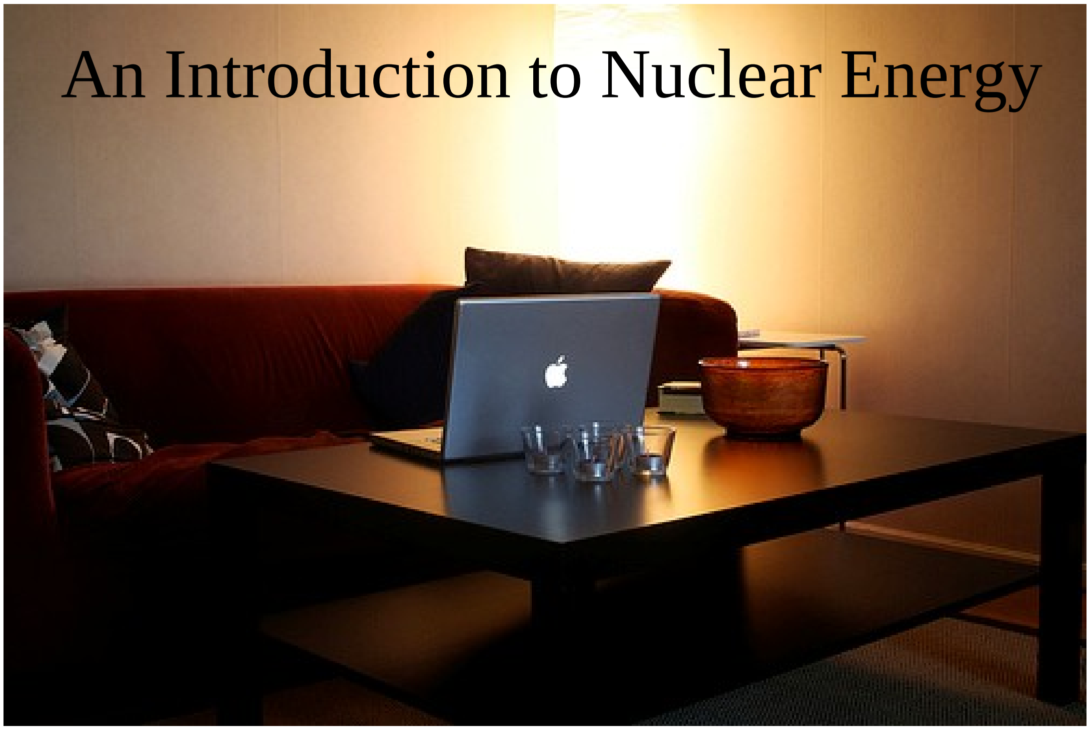

An Introduction to Nuclear Energy

By

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AN INTRODUCTION TO NUCLEAR ENERGY

CONTENTS

NOTE: This course uses generic names and acronyms for nuclear power plant systems, structures, and components (SSCs) and other terminology. These names and acronyms are chosen to best provide an indication of the function of the SSC based on the author's experience. Actual plant SSC names and acronyms vary by reactor type, reactor designer, and the designer of the balance-of-plant SSCs. The reader should not expect exact alignment with these SSC names and acronyms for any particular nuclear power plant. Acronyms are defined at first use in the text. In addition, a list of acronyms is provided at the end of this course material to help the student.

AN INTRODUCTION TO NUCLEAR ENERGY

1. Overview of Nuclear Energy

This chapter begins with a short summary the evolution of nuclear technology from its military beginnings as a weapon used in World War II to its use at commercial nuclear electrical generating stations. It proceeds with summary-level discussions of nuclear power plant designs used in the United States.

1.1 Origins of Commercial Nuclear Technology

Shortly after the Manhattan Project culminated in the use of two nuclear weapons to end World War II, the United States Congress passed the Atomic Energy Act (AEA) to regulate the development and use of nuclear technology for peaceful purposes. While nuclear weapons development continued for decades, two other efforts involving nuclear technology began in the late 1940s: 1) nuclear propulsion for Naval vessels and 2) nuclear energy for commercial electricity generation. The latter involved an expansion of the control of nuclear technology and information from the federal government to include the civilian sector.

The Atomic Energy Commission (AEC) was created to both develop and regulate commercial use of nuclear technology to generate electricity. The federal government's laboratory program was expanded from solely supporting the nuclear weapons program to perform the necessary research and development for the Naval propulsion program and commercial use of nuclear energy. The AEC facilitated a number of demonstration programs around the country in the 1950s and 1960s to design, build, and operate small-scale nuclear power plants to test the technology for generation of electricity.

A key provision of the AEA was that it allowed private companies access to nuclear technology information that has previously been classified as secret. Private companies began investing in this new source of energy that required no transportation infrastructure to continuous deliver fuel to the generating site, as was the case for coal- and oil-fired plants. Private companies worked in concert with the AEC to develop the demonstration plants with an eye on evolving the technology to use for the larger plants that were required to produce a return on the substantial investment required to design, license, construct, and operate the facilities.

The first large-scale nuclear power plants came on line in the late 1960s and grew to over 100 operating reactors at over 70 plant sites around the country by the mid-1980s. As the private nuclear power industry grew, Congress recognized a need to separate its nuclear technology advocacy and research from its licensing and oversight responsibilities. In 1974, Congress dissolved the AEC created two new agencies to assume the AEC's responsibilities. The Nuclear Regulatory Commission (NRC) was created to regulate all civilian use of nuclear materials. The Energy Research and Development Administration (ERDA) was created to coordinate energy programs formerly subdivided among several federal bureaus and serve as focal-point for the major national effort research and develop all forms of energy, including nuclear. In 1977 the ERDA was further consolidated with other energy-related federal agencies to create the Department of Energy (DOE). The NRC currently performs all licensing and oversight of the commercial reactor fleet today and DOE maintains involvement in federal government activities involving nuclear materials and technology.

1.2 Nuclear Energy Versus Nuclear Weapons

Simply put, nuclear power plants cannot explode like a nuclear bomb. While nuclear fission (the splitting of atoms to release energy) was used for both the weapons created by the Manhattan Project and to generate heat in a commercial reactor, there are significant differences in the two technologies. Most importantly, U.S. commercial reactors have always been designed to shut down automatically if certain operating parameters are exceeded; the physics of their design prevents a "runaway" reactor. Furthermore, the nuclear fuel used in commercial reactors is only legally allowed to be enriched in the isotope uranium-235 to relatively low values and not nearly the enrichment levels required to create a detonation. Lastly, while plutonium is created by the fission process in the reactor, it is not in a quantity or form that can detonate. The design of a fission nuclear weapon relies on a specific quantity of highly enriched fissile material (uranium or plutonium) arranged in a precise geometry in order to explode. None of these conditions exist in a nuclear reactor used for generating electricity.

1.3 Nuclear Energy for Electrical Generation

The current fleet of nuclear generating plants is very similar to a fossil-fired steam electric plant. All steam electric plants use a heat source to generate steam and that steam spins a turbine-generator that produces electricity. A nuclear plant simply uses nuclear fission as the heat source rather than the burning of a fossil fuel, like coal, oil, or gas. After the steam exits the turbine and condenses, the water is pumped back to the heat source to repeat the cycle indefinitely, just like other steam electric plants.

The steam, or secondary side of a nuclear plant is essentially identical to that of a fossil-fired steam generating plant. In fact, a common public misperception about nuclear plants is that they all use the tall, hyperbolic natural draft cooling towers. In fact, cooling towers are often used in the media as a representation of a nuclear power plant. However, whether a nuclear power plant (or any power plant) uses a cooling tower is strictly an outcome of the plant design and the characteristics of the associated nearby water source, or “ultimate heat sink” for the plant.

Every steam electric plant requires an ultimate heat sink to discharge waste heat and cool certain plant systems and components. This is why nearly all steam electric plants are built adjacent to a large body of water, such as an ocean, river, or lake. The water is drawn from the ultimate heat sink, pumped through the plant for cooling and either discharged back into the ultimate heat sink at a higher temperature (i.e., a “once-through” system) or cooled with a cooling tower and re-used. The choice of whether to incorporate a cooling tower into the design is based on the plant power output, temperature of the ultimate heat sink water, cost-benefit, and often environmental permitting requirements.

The demonstration commercial nuclear power plants of the 1960s generated around 50 megawatts of electricity. As the size of the plants increased in scale, nuclear plants began generating several hundred megawatts. The largest nuclear plants today each generate on the order of 1,000 to 1,200 megawatts.

Nuclear energy generates about 20 percent of the total electricity in the United States and provides over half of the carbon-free electricity generated, with over 90 operating reactors at 54 sites in 28 states. The last new nuclear reactor to begin operation was Tennessee’s Watt Bar Unit 2 in 2016. Two new reactors at Vogtle plant in Georgia are expected to begin operation in 2022 and 2023, respectively, making Vogtle station the largest nuclear generating station in the United States, and the only site with four operating reactors. Six states have no operating reactors but are the home to former nuclear plant sites that continue to store the spent nuclear fuel produced by reactors that formerly operated in those states.

As mentioned previously, a significant benefit of nuclear power plants is the absence of a need to construct transportation infrastructure, such as a rail spur, to continuous bring fuel to the site to replace the fuel that’s being burned. Nuclear power plants operate for either 18-month or 24-month cycles. At the end of the cycle, the plants are shut down for one to two months to perform maintenance and refuel the reactor. Before such refueling outages, replacement nuclear fuel arrives by truck and the fresh fuel is stored inside the plant until it is loaded into the reactor. Once the reactor re-starts, the fuel lasts for the entire continuous operating cycle. For this reason, reactors today run at very high capacity and have a lower risk of availability disruptions due to transportation system problems or labor actions.

1.4 Types of Nuclear Power Plants in the United States

There are two types of reactors currently used at nuclear power plants in the United States. There is the Pressurized Water Reactor (PWR) and the Boiling Water Reactor (BWR). Worldwide, there are other reactor types beyond PWRs and BWRs that are beyond the scope of this course. About two-thirds of the operating reactors in the United States are PWRs and the remainder are BWRs. Only one site in the country has a mixture of the two types of reactors on the same site – the Salem/Hope Creek Generating Station in New Jersey.

1.4.1 Reactor Design

The nuclear reactor is a bolted-lid pressure vessel designed to safely contain the nuclear fuel assemblies under the high temperature, pressure, and radiation environment created by the nuclear fission process. The reactor is the heart of a system that recirculates water, known as the reactor coolant system (RCS), to remove the heat generated by the nuclear fuel from the fission process, and ultimately convert that heat into electricity.

1.4.1.1 The Reactor Pressure Vessel

The reactor pressure vessels (RPV) is a right-circular cylinder constructed of carbon steel with stainless-steel cladding on the inside surface. The RPV wall is about eight inches thick. The pressure vessel includes a number of nozzles that connect to pipes that allow pumped fluid to flow into and out of the reactor to provide core heat removal. The reactor has a hemispherical top closure head that is bolted to a flange at the top of the vessel body around its circumference. The bolted lid facilitates removing and inserting nuclear fuel assemblies during refueling outages.

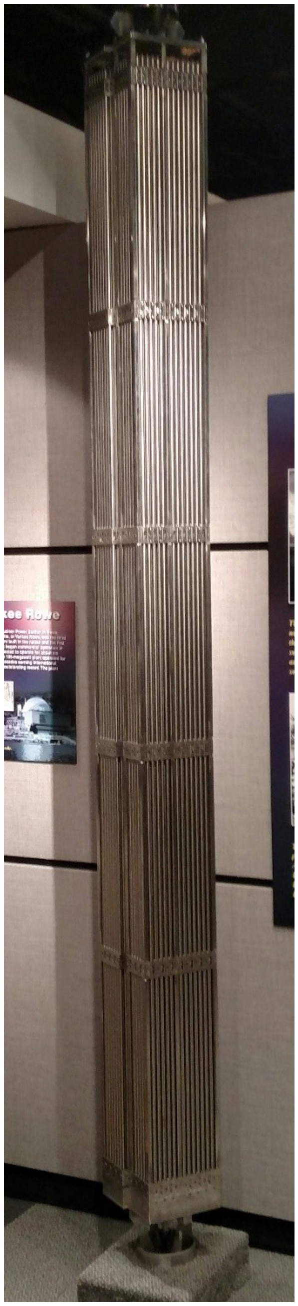
The core of a reactor is composed of a metal lattice and other structural components that support the nuclear fuel assemblies in a stable and predictable configuration under all design conditions. The structural lattice also provides precise geometric spacing of the nuclear fuel assemblies that the reactor core designers rely upon to ensure efficient and safe reactor operation. The differences between PWR and BWR reactor designs are discussed in Subsections 1.4.2 and 1.4.3, respectively, below.

1.4.1.2 Nuclear Fuel Assemblies

Commercial nuclear plant fuel is in the form of small, cylindrical pellets that contain uranium, enriched in the uranium-235 isotope (U-235) to a concentration of five percent or less. The fuel pellets are installed at the fuel fabrication facility inside thin-wall zirconium fuel rods about 12 feet long. The fuel rods are assembled into square arrays that vary based on the type of reactor and fuel design. The fuel assemblies include top and bottom fittings that facilitate lifting and handling and precise placement into each fuel cell location inside the reactor. An example of a typical nuclear fuel assembly and the pellets inside a fuel rod are shown in Figures 1a and 1b.

Figure 1: Example Fuel Assemblies

1a: PWR Fuel Assembly



1b: Fuel Pellets

1.4.2 Overview of PWR Plant Design

This section provides additional details of the PWR reactor and related system component design and a summary of the operation of a PWR plant for generating electricity. In the United States, there are three main vendor designs for PWR reactors – Westinghouse, Babcock and Wilcox (B&W), and Combustion Engineering. All three PWR reactor types follow the same basic principles of design.

1.4.2.1 PWR Reactor

A PWR reactor core is comprised of anywhere from 150 to about 200 fuel assemblies, based on plant design capacity. Each PWR fuel assembly is about 8.5 inches (21.6 cm) square, arranged in a 14x14, 15x15, 16x16, or 17x17 fuel rod array, depending on the reactor vendor and core design. The fuel assemblies are about 14 ft (4.3 m) long, which includes an active fuel region and top and bottom fittings. To start the chain reaction, a reactor core needs neutrons, and a completely fresh core (i.e., at initial operation) produces too few to get the reaction going. So, a few fuel assemblies contain neutron sources to perform this startup function. For operating cycles thereafter, only about a third of the fuel assemblies in the core are replaced. Thus, the refueled core has sufficient startup neutrons. A PWR reactor is outfitted to receive control rods through penetrations in the top closure head and also has incore monitoring instrumentation that is inserted via penetrations in the bottom head of the reactor (see Figure 2).

The nuclear reaction in the PWR core is controlled by managing neutrons produced in the core in two ways: 1) Control rods (or control element assemblies), which are inserted and withdrawn at the top of the reactor and 2) boron in the reactor coolant water. Control rods are made of neutron-absorbing material and move up and down inside some number of the fuel assemblies (usually about a third). Control rods are moved into the core to reduce neutron flux and moved out to increase neutron flux. On a loss of power, the control rods will drop into the core by the force of gravity and shut down the reactor. There is sufficient room inside the RPV above the core to allow the control rods to be removed from the core. The control rod drive mechanisms are situated on top of, and outside the RPV.

Boron concentration in the RCS water is altered by the operators for finer control of neutron flux. Startup of a reactor is a calculation-based process that begins with reducing boron concentration to a level determined by the reactor engineers. Then, control rods are slowing withdrawn to a predicted position where criticality is expected. Plant control room operators monitor neutron flux and are trained to recognize when the reactor has reached a point of self-sustaining fission or "goes critical." This means the reactor core is creating more neutrons resulting from the fission of the uranium-235 atoms in the fuel pellets than it is absorbing to create that fission. The fission process creates heat that is removed by water as discussed further below. Once criticality is achieved, the reactor operators continue to de-borate and pull control rods to increase reactor power to its full design-rated power level. This is a deliberate process that must be coordinated with the rest of the plant as reactor heat increases substantially with the increase in power.

A PWR reactor design maintains the water in the reactor in the liquid phase. That is, the water is kept below the saturation temperature for the operating pressure, also known as "sub-cooled." The difference between the operating temperature and the saturation temperature at a given pressure is known as the subcooling margin. PWR reactors operate at about 2,100 psig (14.5 MPa)[1], for which saturation temperature is about 645^{o}F (340^{o}C) A typical subcooling margin for PWR reactor operation is about 50^{o}F (28^{o}C) at the RPV outlet nozzles, or "hot legs." Maintaining subcooling margin in the RCS is a key goal of PWR reactor operation to avoid any risk of boiling the water in the core.

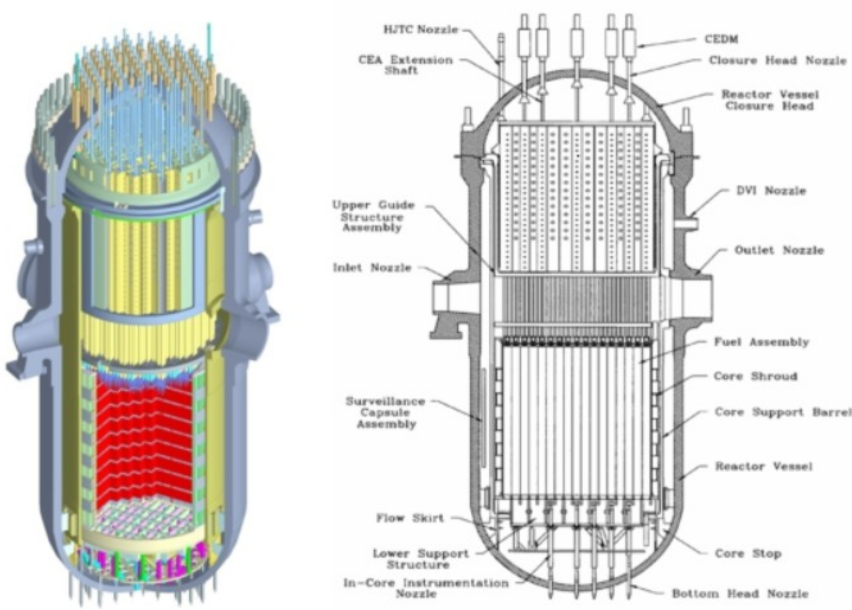
A fundamental safety imperative with any water-cooled nuclear reactor is to keep the nuclear fuel covered with water at all times and under all conditions, including accidents. Failure to do so leaves the nuclear fuel surrounded by low-density steam rather than high density water, resulting in a significant drop in the rate of heat transfer away from the nuclear fuel. The fuel must remain covered with water for adequate heat removal, even if the reactor is shut down

[1] All pressures and temperatures are nominal values used for illustration and should not be considered exact.

(i.e., the control rods are fully inserted and the fission process has stopped). Failure to remove this so-called “decay heat” over a prolonged period of time can eventually result in the fuel cladding melting. This lack of decay heat removal caused the accidents at Three Mile Island Unit 2 in 1979 and at the Fukushima Daiichi plant in 2011. There are several plant emergency systems dedicated solely to ensuring adequate core heat removal as discussed later in this course.

Water enters the RPV inlet (or “cold leg”) nozzles (the number of which vary with reactor vendor design) and flows downward between the inner RPV wall and the outside of the core barrel. At the bottom of the RPV the water moves through the flow skirt and lower support structure, mixes together, and proceeds upward through the core. The water picks up heat from the nuclear fuel assemblies that increases the enthalpy of the sub-cooled liquid. PWR reactor coolant system water enters the reactor at about 550^{o}F (288^{o}C) and exits at about 595^{o}F (313^{o}C). Again, these temperatures vary by reactor design, but will be similar to these values for all PWRs.

Figure 2: PWR Reactor



Three additional components connected to the reactor by piping complete the RCS. Depending on plant design, two, three, or four large reactor coolant pumps recirculate water thought the system. One or more steam generators direct the pumped RCS water through thousands of tubes where the heat picked up by the RCS water in the reactor is transferred to water outside the tubes that becomes steam to generate electricity. Lastly, a tall tank known as a pressurizer is attached to the RCS piping to allow for surge capacity as RCS water temperature rises and falls.

1.4.2.2 PWR Plant Electricity Generation

Figure 3 shows a simplified schematic of the power generation systems at an example PWR nuclear power plant. Like any steam power plant, there are numerous other support systems, such as component cooling water; lubricating oil; plant service air; heating, ventilating, and air-conditioning (HVAC); demineralized water, etc. which are beyond the scope of this course. Suffice to say all components are served by necessary support systems to promote efficient and long-lasting operation.

A PWR plant RCS is a closed-loop system that, simply put, transfers the heat from the nuclear fuel assemblies to a secondary water system in larger heat exchanger known as a steam generator (SG). Depending on plant design and size, there can be one to four reactor coolant pumps to force flow through the system, and two to four steam generators. The pressure tank (or pressurizer) is a tall vessel that maintains a fluid level with an upper steam space. The pressurizer provides system volume control for the expansion and contraction of RCS water as it changes temperature, which ranges from just above ambient temperature during shutdowns to nearly 600°F (316°C) at power. The pressurizer is designed with heaters in the water space and a spray nozzle in the steam space. The operators to control RCS pressure by either heating the water with the heaters or condensing the steam with spray. The pressurizer also prevents the system from “going solid” or allowing steam voids to be created within the RCS, neither of which is desirable.

A small amount of RCS water is continuously removed from the system, cooled, depressurized, directed through resin beds for removal of metal contaminants, and returned to the system as a normal part of power operations. This “letdown and makeup” process reduces the amount of microscopic radioactive metal in the system that could attach to the fuel rods and reactor components. Such “activation products” are created by the microscopic sloughing off of metal in the RCS that is subsequently bombarded by neutrons and becomes radioactive isotopes of iron, cesium, and other elements. The makeup system also replaces RCS inventory that is lost from minor inventory losses in the system that are considered normal.

The SGs are large, vertical shell-and-tube heat exchangers and are the interface point between the RCS and the main steam system (MSS). The RCS water enters the tube side of the SG at an elevated temperature, where it moves through hundreds of small tubes and transfers heat to water on the shell side to produce steam. The RCS water is cooled by this heat transfer process and pumped back to the reactor by the reactor coolant pumps to start the cycle over again.

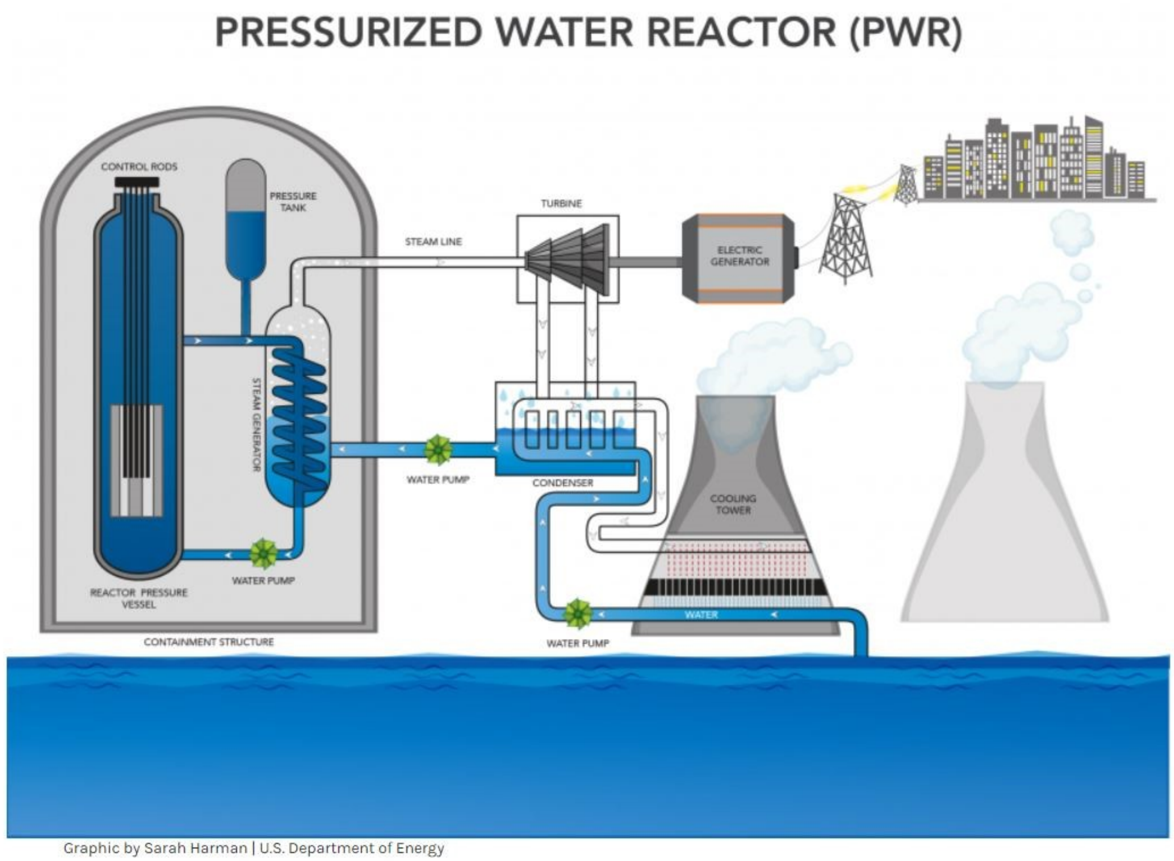
A feedwater system supplies liquid water to the SGs that has been returned from the main condenser via condensate system where it has been demineralized. The feedwater exits the SGs as steam to begin the electric power production process again. The steam travels through a high-pressure turbine, then gets re-heated and fed to a low-pressure turbine, where it spins the turbines and exits as liquid water in the condenser. The high- and low-pressure turbines and the electric generator are connected along a common shaft to produce electricity. That electrical generator output is fed to a switchyard just outside the turbine building and ultimately to the customers via the utility’s transmission and distribution systems.

Steam spins the low-pressure turbine by way of being drawn by vacuum through the turbine blades into a condenser located below the turbine. The condenser is comprised of thousands of tubes containing water from the nearby ultimate heat sink and a collection area at the bottom to collect the condensing steam as it exits the low pressure turbine. The circulating water in the condenser tubes causes the steam condensation, which creates the vacuum needed to force the steam though the turbine blades. The condensate from the steam is pumped to the feedwater system, which directs it back to the SG to start the process again.

Raw circulating water from the nearby ocean, lake, or river that moves through the condenser tubes is heated by the process of steam condensing on the outer surface of the tubes as described above. The raw water can either be discharged directly back into the body of water - a once-through system with no cooling tower – or be directed to a cooling tower. If a cooling tower is used, the raw water is cooled by natural air convection and pumped back to the condenser tubes in a closed loop. The cooling tower air naturally enters at the bottom of the tower and is heated by cooling the raw water. The “chimney effect” causes the air to rise and exit to the environment at the top of the tower. In cooling the circulating water in this manner, the air entrains a substantial amount of the water as small droplets, which creates the well-known water vapor cloud exhausting from the top of an operating cooling tower. This process moderates the environmental impact to the ultimate heat sink by not discharging the hot raw water, but requires a substantial amount of make-up to the circulating water system to maintain the system inventory at design capacity, as shown in Figure 3.

A more detailed discussion of PWR plant systems, structures, and components (SSCs) including emergency core cooling, electrical distribution, instrumentation and control, and structures is provided in Chapter 2.

Figure 3: Example PWR Plant Schematic



1.4.3 Overview of BWR Plant Design

This section provides additional details of the BWR reactor design and a summary of the operation of a BWR plant for generating electricity. In the United States, there is only one main vendor for BWR reactors – General Electric.

1.4.3.1 BWR Reactor

A BWR reactor core is comprised of smaller nuclear fuel assemblies than a PWR reactor and many more of them. BWR fuel assemblies may be found in fuel rod arrays ranging from 6x6 to 10x10 and are about 6 inches (15.2 cm) square. BWR reactor cores contain anywhere from 500 to 760 fuel assemblies. Beyond the fuel size, there are three significant differences between a BWR reactor and a PWR reactor.

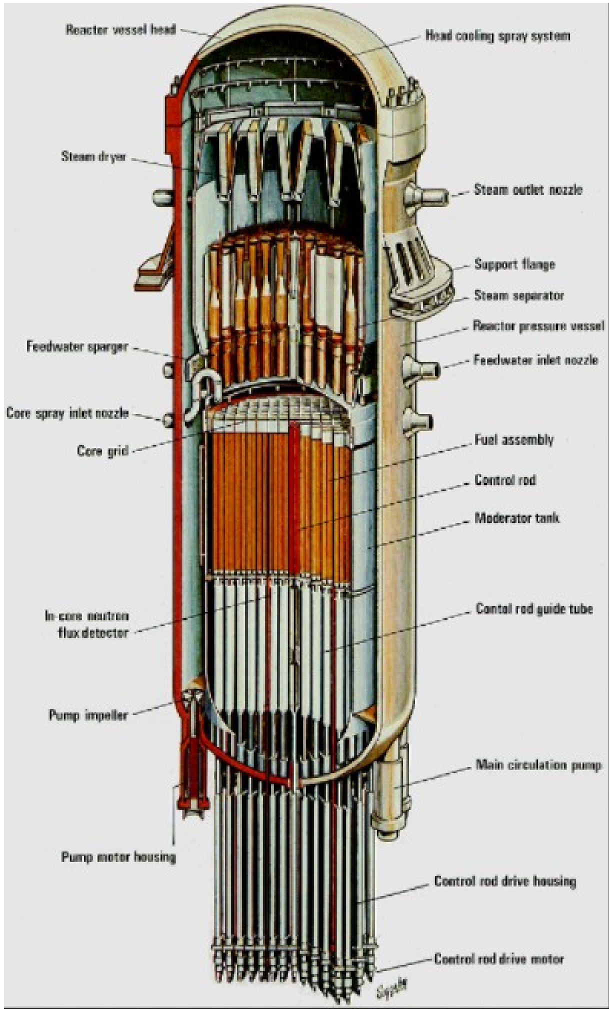
First, as the name suggests, the water in a BWR reactor is allowed to boil. This process is carefully controlled so that the water level in the RPV is kept well above the reactor core such that the nuclear fuel assemblies remain covered. Second, control blades are moved into, and out of the BWR core from the bottom of the RPV with a water-driven hydraulic drive system. A control blade is cruciform-shaped and moves up and down to perform its neutron absorption function within four adjacent nuclear fuel assemblies arranged in a square. Third, there is no boron in the reactor coolant water of a BWR reactor. All core neutron control is performed using the control blades.

A BWR reactor also is outfitted with integral circulation pumps that move the feedwater efficiently into and through the core to ensure distributed heat transfer and steam production. See Figure 4 for a schematic diagram of a typical BWR reactor.

A primary safety objective at the nuclear power plant is core heat removal. Even after the reactor fission process is stopped when the control rods insert, the reactor continues to generate significant residual heat for and extended time due to the radioactive decay of the fuel. Provided adequate water is provided to the core, it will remain stable and eventually cool down to low heat production levels. For a non-emergency shutdown with offsite power available, the initial phase of core residual heat removal is performed by the normal RCS, feedwater, and main steam systems.

As time after shutdown goes on, the reactor will eventually produce too little heat to generate steam and the core heat removal function can be performed by the decay heat removal system (also known as the residual heat removal system). The decay heat removal system takes suction from the RCS and directs that fluid through heat exchangers and back to the RCS. A CCW system rejects that heat to the ultimate heat sink. This is the normal core cooling process during reactor refueling outages.

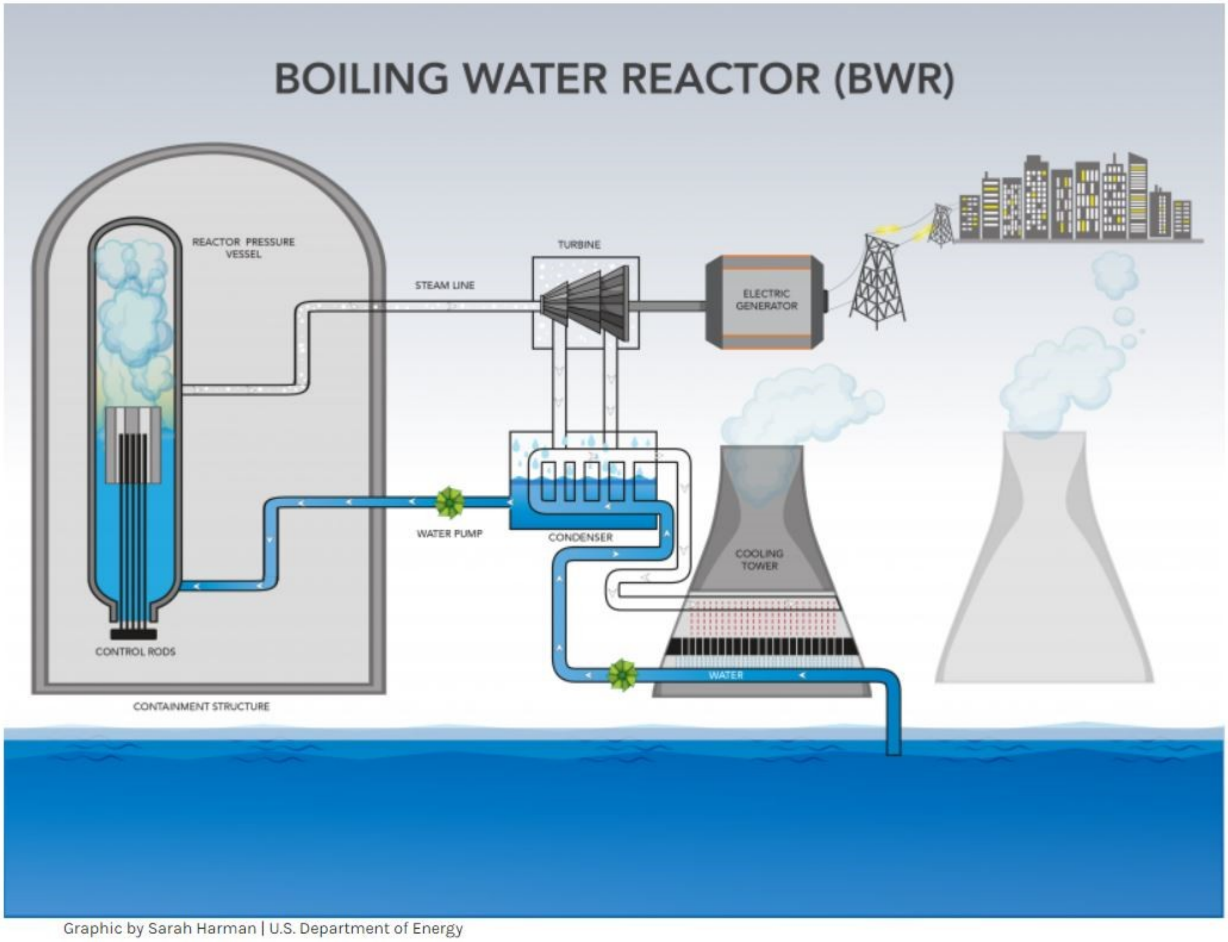
Figure 4: BWR Reactor



1.4.3.2 BWR Plant Electricity Generation

Outside of the reactor containment building, the power generation cycle for a BWR plant is nearly identical to that for a PWR plant. As shown in Figure 5, steam is produced directly in the reactor and piped to the high and low-pressure turbines. The condensate from the steam is demineralized and returned as feedwater to the reactor to repeat the cycle. There are no steam generators in a BWR plant. Whether or not a BWR plant has a cooling tower is predicated on the same reasoning explained previously for PWR plants.

Figure 5: BWR Plant Schematic



2 Nuclear Power Plant Systems

Chapter 2 provides an overview of how and why SSCs are classified as safety-related or non-safety-related at nuclear power plants. Then, the chapter proceeds with generic summaries of the typical mechanical, electrical, and instrumentation and control systems used to operate the facilities. Lastly, a summary of the structural design for a nuclear power plant is provided.

2.1 Classification of Structures, Systems, and Components

Nuclear power plant SSCs are classified as either safety-related or non-safety related. The difference is important because the level of reliability and quality assurance expected of safety-related SSC design, inspection, testing, maintenance, and procurement is substantially more rigorous than that for non-safety-related SSCs.

Safety-related SSCs are those that are relied upon to remain functional during and following design basis events to assure:

1. The integrity of the reactor coolant pressure boundary; or

2. The capability to shut down the reactor and maintain it in a safe shutdown condition; or

3. The capability to prevent or mitigate the consequences of accidents which could result in potentially significant offsite radiation exposures

The above criteria apply not only to the main SSCs directly performing these safety functions, but also the supporting mechanical, electrical, and Instrumentation and Control (I&C) SSCs that provide functions such as protecting, cooling, actuating, and controlling the main SSCs. As such, there are dozens of safety-related SSCs, and hundreds of individual safety-related sub-components at a typical nuclear power plant.

Any SSC not meeting one or more of the above criteria is considered non-safety-related and commercial grade design codes, standards, and quality assurance requirements apply. At a macro level, safety-related SSCs are primarily dedicated to the protection of public health and safety during plant operation and potential accidents. Non-safety-related SSCs are dedicated primarily to supporting electric power generation. There are some exceptions at the interfaces between these two types of SSCs at a system level. For example, the main steam isolation valves (MSIVs) in the main steam system are typically safety-related because they close on demand to provide isolation of containment as part of the response to certain accident events.

2.2 Primary Plant Systems

Primary systems in a nuclear power plant are comprised of those that are directly involved with heat removal from the reactor core during normal power operations. This includes the RCS and connected systems, such as the makeup (or volume control) system. The RPV, along with the RCS piping, SG tubes, and reactor coolant pumps maintain the reactor coolant pressure boundary. RCS inventory management is vital to core cooling and overall plant safety, which is why it is one of the three criteria defining a safety-related SSC.

Given the high operating pressures and temperatures of RCS operation, the water will flash violently to steam if a breach in the system occurs. Anything more than a small RCS leak can result in loss of coolant beyond the capability of the makeup system to make up the inventory, or a Loss of Coolant Accident (LOCA). The system and operator responses to LOCA accident can be quite complex given the wide variety of pipe break sizes and locations that could be postulated. The system and operator responses to LOCAs is discussed in more detail in Chapter 3.

2.3 Secondary Plant Systems

Secondary systems are those that receive the heat removed from the reactor core and convert that energy into electricity via the turbine-generator. The MSS converts water to steam in the steam generators in PWR plants or takes steam directly from the reactor in BWR plants and moves it to the high- and low-pressure turbines. To provide the most efficient thermodynamic power generation cycle, the MSS is also connected to moisture separator/reheaters and provides extraction steam to feedwater heaters in accordance with the overall heat balance for the plant. This is common for the design for any steam electrical generating plant.

In addition to the MSS, secondary plant systems include the feedwater, condensate, circulating water, extraction steam, and steam drain systems, to name a few.

2.4 Component Cooling Water Systems

A nuclear power plant has two major categories of Component Cooling Water (CCW) systems that perform the same basic function – provide heat removal for operating equipment. This is accomplished via providing cool water in a closed loop to the served components (e.g., pumps) and discharging that heat to a tertiary cooling system via heat exchangers. The tertiary system transfers the heat from the heat exchangers to the ultimate heat sink. Tertiary cooling water systems can be open systems (once-through, with the ultimate heat sink) or closed loop, utilizing a cooling tower.

The difference between the two major categories of CCW systems is whether they serve safety-related or non-safety related SSCs. If they serve safety-related SSCs, the CCW systems themselves are also safety-related. Furthermore, safety-related CCW systems require a backup source of electrical power to ensure continued operation if normal power from the grid serving the plant ("offsite power") is lost. These CCW systems are also required to be dual-train, functionally redundant so that a single active failure anywhere in the plant does not cause a complete loss of function. Electrical power and distribution systems are discussed in more detail later in this course.

Other plant CCW systems are non-safety related and require no backup power or redundancy beyond that to protect against an unacceptable commercial risk of losing equipment that affects power plant operation. The two key factors in this regard are the expense to repair or replace the damaged component and the potential for lost electrical generation if that component is unavailable. This is purely a financial decision by the plant owner.

2.5 Emergency Core Cooling and Other Safety Systems

This section summarizes the key safety systems in a nuclear power plant. Again, the discussion focuses on PWR plants, and specifics will vary by reactor type and plant design. Emergency core cooling systems (ECCS) have one function: to provide and maintain water flow to the reactor core when normal cooling is not adequate. Other safety system provide backup feedwater and containment cooling.

2.5.1 Passive Emergency Core Cooling Systems

Because there is an inherent delay in powered systems ramping up to pump water to the core in an emergency (especially with a concurrent loss of offsite power), in many nuclear plants there is a passive system that reacts immediately to deliver water to the core in the case of a significant loss of RCS inventory. Passive core flooding systems are comprised of tanks located at an elevation above the reactor and connected to the primary system via piping and two valves in series – a check valve and an isolation valve. The number and size of the tanks is reactor design-specific.

At operating pressure, the core flooding isolation valves is open, leaving only the check valves between the RCS and the core flooding tanks (also known as cold leg accumulators in some designs). RCS pressure keeps the check valves closed. If depressurization of the RCS occurs, the check valves open and the contents of the core flooding tanks empty into the RCS and the reactor. The rate of discharge into the RCS is dependent on the pressure in the RCS – lower RCS pressure allows a higher rate of flow from the core flooding system. The core flooding system provides both initial emergency core cooling and time for the active ECCS systems to become fully available.

2.5.2 Active Emergency Core Cooling Systems

There are two primary active PWR ECCS systems: high-pressure injection (HPI) and low-pressure injection (LPI). Both systems will actuate in response to a LOCA, with the systems performing hydraulically based on the amount of back pressure in the RCS as a result of the break. The magnitude of RCS backpressure is dependent on the postulated pipe break size. Larger breaks depressurize the RCS quickly, with the LPI system primarily providing cooling. Smaller break sizes may keep RCS pressure relatively high, where the HPI system provides the cooling. The HPI and LPI systems initially draw suction from a large water storage tank known and the Refueling Water Storage Tank (RWST) and inject that water into the RCS. At a pre-determined low level in the RWST, the suction for HPI and LPI is transferred to the containment sump for long-term core cooling.

Chapter 3 provides a more detailed description of the LOCA event and ECCS system operation.

2.5.3 Emergency Feedwater System

As discussed previously, the main feedwater system removes heat from the reactor core either via the SGs (for PWR plants) or directly from the core (for BWR plants). Thus, the primary and secondary systems are hydraulically linked and precisely balanced for a continuous rate of core heat removal during normal power operation. A significant reduction or loss of feedwater event is a serious anomaly at a nuclear power plant because the RCS temperature increases rapidly as the RCS heat removal rate decreases with the feedwater loss. Steam generator (PWR) or reactor (BWR) water level will begin to decrease with the rate of water boil off exceeding the rate of feedwater being supplied.

Without adequate feedwater supply, RCS temperature will rise over time due to reduced primary-to-secondary heat transfer. Over an extended time period with insufficient feedwater, the subcooling margin in a PWR RCS could be lost, resulting in the water in the reactor reaching the boiling point. Left unaddressed, the loss of water level in the core could ultimately result in uncovering the core.[2]

[2] This scenario is exactly what happened at the Three Mile Island Unit 2 accident in 1979 where about half the core was irreparably damaged and the reactor operated again. Unit 1 continued to operate until 2019 when it was permanently retired.

At a BWR, feedwater is provided directly to the reactor. Thus the emergency feedwater system would also feed the reactor to prevent water level from dropping below the top of the core.

Upon detection of a loss of level in the PWR SG or BWR reactor, two things happen. First, the MSIVs close to immediately reduce the rate of heat removal from the RCS. This results in the steam that continues to be generated in the MSS being released to the atmosphere via safety relief valves located upstream of the MSIVs. Second, the emergency feedwater (EFW) system actuates to restore PWR SG or BWR reactor level and provide the needed heat removal from the core. One EFW pump is usually steam-driven, using steam from the MSS delivered from a point upstream of the MSIVs. The redundant EFW pump is motor-driven. The source of EWF system water is a tank of unborated water and, for long-term cooling, other sources of condensate, such as the turbine condenser.

2.5.4 Containment Cooling and Spray Systems

In a LOCA event, a tremendous amount of energy in the form of flashed steam from the high-pressure RCS fluid rapidly depressurizing. To guard against containment overpressurization, one or two systems automatically actuate: containment cooling and containment spray. The first system to actuate is containment cooling, consisting of large fan coolers that pull the steam-laden containment atmosphere over cooling coils to condense the steam and reduce building pressure.

If the accident event causes a containment building pressure increase that is beyond the containment cooling system’s ability to handle, the containment spray system will actuate to inject water into the containment to quench the steam, thus reducing the pressure. The containment spray system draws from the same tank of borated water as the HPI and LPI systems and the spray water collects in the containment sump. Also like the HPI and LPI systems, the suction for the containment spray pumps is eventually switched to the containment sump for long-term containment cooling until such time as the operators deem spray cooling is no longer needed.

2.5.5 Operator Role in ECCS Operation

The ECCS SSCs are designed to actuate and begin operating automatically, as described in further detail in the “Instrumentation and Control Systems” section of this course. In the early stages of an accident event, the plant is designed to allow operators to take no action and simply confirm that all SSCs that were expected to actuate have actuated for the plant conditions observed. If an SSC has not automatically actuated, the operator may manually actuate the SSC in accordance with their emergency operating procedures.

After the initial phase of an accident event, the operators prepare to take certain required manual actions, such as re-aligning ECCS suction from the RWST to the containment sump. As the plant settles into an equilibrium state and relevant parameters allow, the operators can take more discretionary actions based on their training, including securing certain systems. For instance, when RCS pressure is at a sustained low pressure post-LOCA, the HPI system can be turned off.

2.5.6 Single Failure

Safety-related SSCs are required to be able to perform their intended safety function despite the single failure of any active component or sub-component, including cascading effects. The single failure criterion is applied to every active safety component so that the effects of the failure of that component to start, stop, continue running, change position, etc. are understood and accounted for in system design so that the design safety function is still achieved. To ensure no effects are missed, a Failure Modes and Effects Analysis, or FMEA is typically performed to ensure redundant safety functions are always available for any single failure.

The most consequential single failure is usually the loss of a train of emergency electrical power after a loss of offsite power and a coincident accident event. Such a failure would disable every AC electrical-powered component served by that side of the electrical distribution system. Most consequentially this means the pumps, valves, and fans that ensure emergency cooling to the core or containment would be inoperable.

Nuclear power plants in the United States are also designed to withstand a complete loss of all offsite and onsite AC power for a limited time. Such an event, known as a station blackout (SBO), is unlikely but plants are prepared with procedure, training, and equipment to respond to restore power quickly.[3]

2.6 Electrical Systems

Like the mechanical systems in a nuclear plant, the electrical systems are separated into two types: those that power safety-related components (primarily pumps and large motor-operated valves) and those that power non-safety-related components. Also like the safety-related mechanical systems, the safety-related electrical power and distribution systems are completely redundant. That is, the “A” train of the redundant mechanical components that have motors is powered by the “A” train of the electrical power and distribution system. Likewise for the “B” trains of both.

During normal power operations the ECCS systems are in a standby mode and the in-service plant systems used to generate electricity are powered from the grid, or “offsite power.” Offsite power feeds both the emergency safeguards (ES) buses and distribution systems that provide power to safety-related components and the non-ES buses and distribution systems that feed non-safety-related components. Most of the pumps in the safety-related and non-

[3] The events at the Fukushima Daiichi power plant in March 2011 were caused by a prolonged SBO. A tsunami caused by a massive offshore earthquake took out both normal offsite power and submerged the emergency diesel generators in seawater, rendering them inoperable and leaving the plant with no AC power and no way to restore power in a timely manner. Even though the reactors shut down, the necessary core cooling for decay heat removal was unavailable and the fuel in some of the reactors melted. In the United States, the emergency diesel generators at every plant and the related electrical distribution equipment are located above the worst-case flood level.

safety-related systems are powered by 4,160 kV motors. One exception is the non-safety-related reactor coolant pumps, which employ larger motors (e.g., 6,900 kV).

If emergency systems actuate, they are powered from the ES buses via offsite power. As part of such an actuation, signals are sent to shut down the reactor and trip the turbine-generator, ceasing electric power generation. At this point, establishing and maintaining core cooling by removal of decay heat is the sole objective of the power plant systems, powered by all available electrical buses. For example, steam is still being created from reactor heat. That steam, which was once turning the turbines, is instead exhausted to the atmosphere via safety relief valves.

If offsite power is lost, either with or without a coincident ES actuation, again the reactor shuts down, the turbine trips and electrical generation ceases. However, in this scenario, emergency diesel generators (EDGs) automatically start upon detection of undervoltage on and proceed through a loading sequence for the ES buses to power the safety-related components. All non-safety-related components requiring electric power coast down and remain non-functional until offsite power is restored. However, all safety-related components operate on emergency power provided by the EDGs. As discussed previously, the most consequential single failure of a component in a nuclear power is usually the failure of an EDG to start and load in an accident event coincident with a loss of offsite power. This is because the loss of an emergency electrical power source makes an entire train of safety systems unavailable.[4]

EDGs are designed with local “day” fuel tanks near the engines that are re-filled as needed larger onsite tanks that contain about a week’s worth of diesel fuel with an EDG operating at maximum capacity. Usually a week is well in excess of the time required to restore offsite power. However, within that week, the plant can arrange to re-fill the diesel storage tanks, if necessary to continue EDG operation indefinitely.

[4] The one exception is the steam-driven emergency feedwater pump, which will remain operable as long as steam is available to drive the pump turbine.

2.7 Instrumentation and Control Systems

This section summarizes the I&C systems in a typical nuclear power plant. Again, these descriptions are based on the author's experience with one type of PWR plant design. Other PWR designs and BWR plants will have different I&C system types, control parameters, and terminology. But the overall functions are similar. The discussion in this section contemplates a reactor running at full power and is divided into four groups of I&C systems:

1. Those that monitor and control normal power operations
2. Those that automatically shut down the reactor
3. Those that actuate emergency systems and components
4. Those that monitor plant conditions and post-accident parameters

2.7.1 I&C Systems for Normal Power Operations

Like any steam electric station, nuclear power plant operations are fundamentally based on controlling an energy balance. That is, nearly all of the heat generated by nuclear fission in the reactor is 1) converted into electricity via work, 2) used to re-heat main steam and pre-heat feedwater returning to the steam generators, and 3) discharged as waste heat to the environment. The small reminder of reactor heat is assigned to unmonitored losses to the ambient and equipment inefficiencies.

The plant design contemplated in this section is controlled by an Integrated Control System (ICS), which maintains the average RCS temperature (T_{ave}) between RPV inlets (cold legs) and outlets (hot legs) at a constant value. At equilibrium conditions, reactor thermal power and electricity demand are constant, which means feedwater flow to the steam generators and steam production from the steam generators are also constant. But equilibrium only exists in theory due to system losses, changes to electricity demand, and burning the reactor fuel over time. Thus, the ICS is constantly responding to system inputs.

The ICS responds to changes in the RCS, steam systems, and electricity demand at the generator to ensure T_{ave} maintains setpoint. The ICS gradually changes reactor thermal power, feedwater flow, and other system parameters in response to demands to avoid abrupt changes in these parameters if they were manually controlled. Gradual changes help avoid automatically tripping the reactor without a valid reason.

Beyond the ICS, there are numerous I&C systems that monitor plant parameters such as reactor power, RCS temperature and pressure, main steam temperature and pressure, feedwater flow, and a myriad other indicators of plant operation and performance. Some of these parameters feed emergency actuation systems as discussed below and others are used by plant operators and engineers to monitor, adjust, and improve plant performance and efficiency.

2.7.2 I&C Systems that Shut Down the Reactor

The first, and most important response to a plant condition that indicates a significant anomaly in plant operation is to safely shut the plant down, beginning with stopping the fission process in the reactor. Many parameters including, but not limited to reactor power, RCS temperature and pressure, reactor coolant pump status, main feedwater pump status, and turbine operating status are monitored. If a valid trip setpoint is reached by multiple channels of instrumentation that confirm the parameter it is an actual, not spurious signal, a reactor trip signal is initiated. This signal results in the control rods inserting into the reactor core, immediately stopping the fission process.[5] A reactor trip will also send a signal for a turbine trip, which open the output breakers so that the entire electricity generation function of the plant stops.

[5] The reactor can also be tripped manually at any time by a licensed operator pressing the reactor trip button in the control room.

2.7.3 I&C Systems that Actuate Emergency Systems and Components

The Emergency Safeguards (ES) system monitors several system and plant parameters so that upon detection of certain conditions, the reactor will trip, normal power operations will cease, and ES systems will actuate. Upon a valid signal, systems such as the high pressure core injection, low pressure core injection, emergency feedwater, containment cooling, and containment spray systems will automatically start depending on the specific setpoints that were exceeded. Pumps and fans will start and valves will move to their ES positions to ensure cooling is provided where needed and the reactor containment building is sealed. Should the ES event be combined with a loss of offsite power, the emergency diesel generators will also auto-start to power the ES buses, which provide electrical power to the ES equipment.

Parameters monitored by the ES system typically include reactor power, RCS pressure and temperature, feedwater pump status, and main steam pressure and temperature, among others. If any of the ES setpoints are reached, the reactor shuts down and the associated ES equipment actuates.

2.7.4 I&C Systems that Monitor Plant Conditions and Post-Accident Parameters

Like any industrial facility, nuclear power plants have numerous I&C systems that monitor system and plant parameters to provide information to personnel both locally and in the main control room. As discussed above, nuclear power plants are unique in that they require monitoring for control of plant functions and performance in generating electricity as well as actuating and controlling safety systems after an unexpected event such as an accident, fire, or a loss of offsite power. Nuclear plants are also unique because there are areas of the plant that are not accessible by plant personnel during power operations.

Reactor power is measured by instruments located outside the reactor vessel and both along the height of, and around the circumference of the reactor vessel. These instruments detect neutrons escaping the reactor during operation, known as “flux.” Based on the specific fuel type and core design, these detectors are precisely calibrated to correlate this neutron flux measurement to thermal power in the core. Cores are designed for a given rated thermal power, the neutron flux for which is designated as 100% power. Lower neutron flux levels correspond to lower power levels. There is also nuclear instrumentation located inside the reactor core inserted from the bottom of the reactor through a small number of fuel assemblies to measure intra-core parameters.

After an accident, plant areas can potentially become harsh due to heat, humidity, and/or high radiation levels. Furthermore, reactor operators need information about the status of the core and containment, and other information necessary to confirm plant safety systems are performing properly. This was a significant lesson learned from the Three Mile Island accident, resulting is post-accident monitoring being a significantly expanded part of the plant I&C design.

Other monitoring unique to nuclear power plants includes area and effluent radiation monitoring, containment sump level, and containment pressure during normal operation and after an accident. The full list of parameters monitored is too long to list here, and varies by plant design.

2.8 Structural Design

Nuclear power plants are required by federal regulations and the applicable design codes to have safety-related SSCs designed appropriately so that they function under all design basis conditions and events. Safety-related SSCs are defined in Section 2.1. Safety-related mechanical, electrical, and I&C systems have been discussed previously. This section summarizes the structural requirements to protect these systems. The key external structural design criteria to consider in protecting safety-related SSCs include high winds, tornado-driven missiles, earthquakes, and flooding.[6]

[6] Nuclear power plants are also designed to ensure that no fire inside the plant can disable both trains of any redundant safety-related systems.

2.8.1 Containment

As discussed previously, both PWR and BWR nuclear plants operate with the RCS under high temperature and pressure. In the event of a breach in the RCS, the RCS fluid will flash to steam and increase the pressure in the structure housing the RCS. That fluid will also contain radioactive isotopes; the exact type and quantity of radioactive material depends on the severity of the accident. Thus, the reactor and attached system piping out to specific isolation points are housed within a robust building that performs the containment function. Penetrations in the containment building not integral to systems responding to the accident will isolate in response to the appropriate signal. The containment building structure is designed to withstand the maximum pressure from a worst-case accident.

The objective of containment is to minimize any leakage of radioactive material after an accident to an amount that is not a threat to public health and safety. Containments are constructed as steel pressure vessels buttressed by reinforced concrete. PWR plants use a single containment building, usually known as the Reactor Building. BWR plants employ a primary containment housing the reactor and a secondary containment surrounding the primary containment. BWR primary containment is designed as discussed above, while secondary containment is designed to maintain a negative pressure to ensure any leakage from the primary containment is captured and processed.

2.8.2 Other Safety-Related Structures

Many safety-related SSCs at nuclear power plants are not located in the containment because containment is not designed to be readily accessible during power operations, primarily due to the high temperature levels. These SSCs require regular access by Operations, Maintenance, and Engineering personnel to perform maintenance, testing, inspections, and system health walkdowns. These SSCs are housed in buildings adjacent to the containment that are also made of reinforced concrete to withstand all design basis environmental events and protect the SSCs inside. Piping, electrical conduit and cable trays between the buildings traverse through engineered penetrations.

2.8.3 Main Control Room

The main control room at a nuclear power plant is its nerve center, where the main control board displays the status of the reactor and nearly every other component in the plant. The senior-most reactor operators authorize, oversee, and monitor day-to-day plant operations and maintenance to ensure safe and reliable plant operations. They also respond to accidents by executing emergency operating procedures from the control room. Thus, the control room is also a reinforced concrete structure designed withstand environmental events such as earthquakes and also protect the human beings working inside. The main control room has a specialized HVAC system with filtration to ensure the control remains habitable for the operators for the duration of any accident event where hazardous materials are in the vicinity.

2.8.4 Other Plant Structures

In thinking about a nuclear power plant, it is often easy to overlook the reason the plant exists – to generate electricity. While a nuclear power plant does not have a boiler or delivery systems for fossil fuels like coal, oil, or gas, it does have all of the other systems required of a steam generating station. This primarily includes a turbine building (or open-air deck), intake structure for raw cooling water, maintenance shops, and a variety of tanks and piping around the site. These non-safety-related SSCs are designed in accordance with commercial design codes applicable to the system or component function.

2.9 Design Codes and Standards

In the early days of nuclear plant design in the United States, commercial design codes for mechanical, structural, and electrical systems were used. Over time, the owners of these consensus codes and standards developed versions that better reflected the unique characteristics of nuclear power plants, as licensed under the requirements of the U.S. Nuclear Regulatory Commission. The key codes and standards for nuclear power plant design, construction, and operation were developed and are maintained by the American Society of Mechanical Engineers (ASME), the American Concrete Institute (ACI), the Institute and Electrical and Electronics Engineers (IEEE), the American National Standards Institute (ANSI), ASTM International, and American Nuclear Society (ANS). While, with the exception of ANS, each of these organizations' mission is broader than nuclear, they each have a subset of codes and standards unique to nuclear, most notably:

- ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Components"
- ASME Section XI: "Inservice Inspection of Nuclear Power Plant Components"
- ACI-349, "Code Requirements for Nuclear Safety-Related Concrete Structures"
- IEEE Standard 308, "Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations"
- IEEE Standard 497, "Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations"

3. Nuclear Power Plant Design Basis Accident Example

First and foremost, nuclear power plants are designed, constructed, operated, maintained, inspected, and tested regularly to prevent the occurrence of accidents. However, the plants still need to be designed to respond to accidents by shutting the reactor down, keeping the reactor safely shutdown, and containing radioactive material to within analyzed limits. Thus, a series of limiting case, design basis accident are postulated as part of the plant licensing and design process to provide the basis for safety-related SSC design.

For this introductory module only the most significant of these accidents is summarized below to provide insight into how the plant responds. There are numerous other accidents for which a nuclear power plant is designed to withstand that are described in the plant's safety analysis report that supports licensing. However, this module only includes discussion of one as an example due to space constraints.

The primary nuclear power plant design basis accidents are most easily described as loss of inventory events from either the RCS, the main steam system, or the main feedwater system. This is because these systems are hydraulically coupled to remove the heat generated by the nuclear fission process and have the most significant effect on core cooling and requiring emergency system response. The section below describes one such event and the plant/system response. Again, this discussion is based on the author's experience with one type of PWR plant design. Other PWR plant designs and BWR plants respond differently.

Loss of Coolant Accident

A LOCA is defined as a leak from the RCS beyond the capability of the normal makeup system, requiring additional sources of replacement water. Nuclear power plants are built to respond to a full range of RCS pipe break sizes up to a full guillotine break of the largest diameter pipe in the system. The size of the break is an important factor in the response of the ECCS systems. Refer to Figure 5 for an example of a schematic representation of the SSCs discussed below.

Large Break LOCA

The break of the largest diameter RCS pipe is known as a large break LOCA. The large break LOCA depressurizes the RCS quickly and the loss of RCS inventory into the containment occurs at a high rate. Because the RCS operates at high temperature and pressure, the fluid exiting the break immediately flashes to steam and begins to pressurize containment. In this accident, the reactor automatically shuts down and ES systems actuate based on low RCS pressure. The complete loss of RCS pressure causes the core flooding tanks (cold leg accumulators) to empty their contents via gravity feed into the RCS. At the same time, containment is isolated and the HPI and LPI pumps start in response to the ES signal and valves open to provide suction to these pumps from the RWST. Because the RCS is depressurized, both the high pressure and low pressure core injection systems provide core cooling.

Core cooling from the RWST continues until the tank reaches low level. At that time, plant operators swap the suction for the core injection pumps to the containment sump where the fluid from the break has accumulated. This flow path provides a long-term core cooling configuration that prevents fuel damage for an essentially indefinite period of time and lasts until recovery actions are able to be developed.

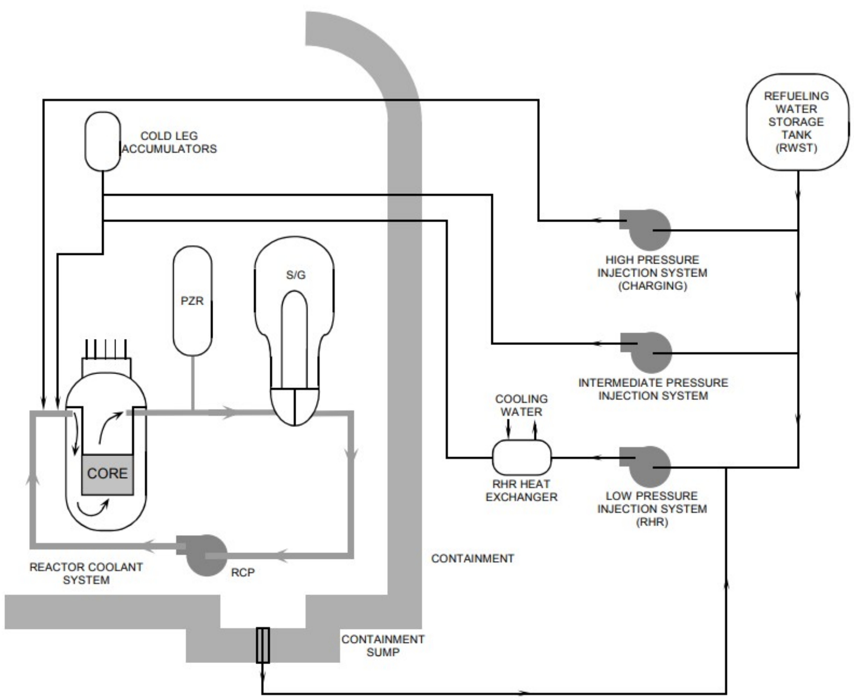
The ES actuation will also auto-start the containment cooling fans. If containment pressure increases substantially due to the RCS break, the containment spray system will also actuate to inject water into the containment atmosphere. These cooling systems are designed to ensure containment temperature and pressure remain below the design values after a LOCA and any leakage of radioactive material remains within analyzed limits.

Small Break LOCA

A small break LOCA is a loss of RCS inventory at a rate high enough that RCS pressure decreases to the ES setpoint, but the system does not depressurize. Because RCS pressure remains elevated, HPI is the primary means of emergency core cooling, RCS inventory is not lost quickly and the response time urgency is not as critical as it is for a large break LOCA. Containment isolation and ECCS system responses are largely the same as for a large break LOCA. However, because RCS pressure remains high, the cold leg accumulators will not dump into the RCS immediately and the low pressure injection system will recirculate to the RWST rather than inject into the RCS.

Again, depending on the size of the pipe break, the RCS pressure will decrease slowly over time. This provides the operators the time to analyze the situation and make decisions on how to proceed. For example, operators can choose to isolate the cold leg accumulators if sufficient core cooling is occurring. High pressure core injection from the RWST continues until the operators swap over to containment sump suction for long-term cooling. Eventually, RCS pressure decreases to a point where the operators will swap from high-pressure core injection to low-pressure core injection from the containment sump to achieve the long-term core cooling configuration.

Figure 5: Example PWR Plant Emergency Core Cooling Systems[7]



[7] Source: C:\WINDOWS\Desktop\Text\04.wpd (nrc.gov). Note that the containment coolers and containment spray systems are not shown and this figure includes an intermediate pressure injection system not discussed here.

LIST OF ACRONYMS

ACI – American Concrete Institute

AEA – Atomic Energy Act

AEC – Atomic Energy Commission

ANS – American Nuclear Society

ANSI – American National Standards Institute

ASME – American Society of Mechanical Engineers

ASTM – American Society for Testing and Materials

BWR – Boiling Water Reactor

CCW – Component Cooling Water

DOE – Department of Energy

ECCS – Emergency Core Cooling System

EDG – Emergency Diesel

EFW – Emergency Feedwater

ERDA – Energy Research and Development Administration

ES – Emergency Safeguards

HPI – High Pressure Injection

I&C – Instrumentation and Control

IEEE - Institute and Electrical and Electronics Engineers

ICS – Integrated Control System

FMEA – Failure Modes and Effects Analysis

LOCA – Loss of Coolant Accident

LPI – Low Pressure Injection

MSIV – Main Steam Isolation Valve

MSS – Main Steam System

NRC – Nuclear Regulatory Commission

PWR – Pressurized Water Reactor

RCS – Reactor Coolant System

RPV – Reactor Pressure Vessel

RWST – Refueling Water Storage Tank

SBO – Station Blackout

SG – Steam Generator

SSCs – Systems, Structures, and Components

An Introduction to Nuclear Energy- Quiz

Updated: 4/22/2022

1. What federal law governs the development and use of nuclear technology for peaceful purposes?

    a. The Nuclear Waste Policy Act
    b. The Nuclear Materials Act
    c. The Atomic Energy Act
    d. The Energy Reorganization Act

2. What United States federal government agency currently performs all licensing and oversight of the commercial reactor fleet?

    a. The Department of Energy
    b. The Nuclear Regulatory Commission
    c. The Atomic Energy Commission
    d. The Energy Research and Development Administration

3. What is the most important reason that a commercial nuclear reactor cannot explode like a nuclear weapon?

    a. Commercial reactors have always been designed to shut down automatically if certain operating parameters are exceeded
    b. Commercial fuel does not contain the any of the isotopes necessary for detonation
    c. Commercial nuclear fuel cannot be reprocessed
    d. Trick question. It can.

4. A nuclear plant uses nuclear fission as the heat source rather than the burning of ________

    a. Coal
    b. Oil
    c. Gas
    d. All of the above

5. Whether a nuclear power plant (or any power plant) uses a cooling tower is strictly an outcome of_________________

   a. Plant design
   b. The characteristics of the associated nearby water source
   c. Both a and b
   d. Neither a nor b

6. The large body of water, such as an ocean, river, or lake, located adjacent to the nuclear power plant is known as:

   a. The big cooling pond
   b. The ultimate heat sink
   c. The cooling tower
   d. The raw water system

7. Nuclear energy generates what percentage of the total electricity in the United States?

   a. 20 percent
   b. 50 percent
   c. 90 percent
   d. 5 percent

8. The reactor pressure vessel has a hemispherical top closure head to facilitate__________

   a. Opening the reactor during operation to perform inspections
   b. Responding to an accident
   c. Tours to visitors
   d. Removing and inserting nuclear fuel assemblies

9. Commercial nuclear plant fuel is enriched in what isotope?

   a. Uranium-238
   b. Plutonium-235
   c. Uranium-235
   d. Plutonium-238

10. Control rods are made of____________

   a. Stainless steel
   b. Neutron absorbing material
   c. Fuel pellets
   d. Zirconium

11. The difference between the operating temperature of the water in the reactor and the saturation temperature at a given pressure is known as_____________

   a. Subcooling margin
   b. Safety margin
   c. Superheat margin
   d. Loss of coolant margin

12. The difference(s) between a PWR plant and a BWR plant is/are________

   a. BWR reactors allow the water in the reactor to boil
   b. BWR control blades are moved into, and out of the BWR core from the bottom of the RPV
   c. There is no boron in the reactor coolant water of a BWR reactor
   d. All of the above

13. Which of the following is <u>not</u> a criterion for a system, structure or component in a nuclear power plant being classified as "safety-related"?

   a. It assures the integrity of the reactor coolant pressure boundary
   b. It assures the capability to shut down the reactor and maintain it in a safe shutdown condition
   c. It assures reliable electricity generation from the power plant
   d. It assures the capability to prevent or mitigate the consequences of accidents which could result in potentially significant offsite radiation exposures

14. A core flooding system is an example of a__________________

   a. Active emergency core cooling system
   b. Passive emergency core cooling system
   c. Static emergency core cooling system
   d. Non-essential core cooling system

15. To guard against containment overpressurization after a LOCA, which of the following systems automatically actuate:

    a. Containment cooling
    b. Containment spray
    c. Both a and b will actuate
    d. a will actuate and b may actuate

16. After the initial phase on an accident event, the operators__________

    a. Prepare to take certain required actions
    b. Leave the control room
    c. Call the local mayor
    d. Do nothing

17. Application of the single-failure criterion is best described as the evaluation of_________

    a. A leak in the reactor vessel bolted lid
    b. A pipe break
    c. The collapse of a safety-related building
    d. The failure of an active component to start, stop, continue running, or change position

18. The Integrated Control System controls the plant based on maintaining what at setpoint?

    a. Electrical demand
    b. Reactor power
    c. Average RCS temperature between reactor inlets and outlets
    d. Turbine pressure

19. The objective of containment is to__________

    a. Minimize any leakage of radioactive material after an accident
    b. Protect the control room operators from an earthquake
    c. Maintain electrical production after an accident
    d. Allow for maintenance of the reactor during operation

20. The key manual operator action after a large break LOCA is___________

a. Actuating emergency feedwater
b. Swapping the suction for the core injection pumps from the RWSTto the containment sump
c. Re-establishing control room ventilation
d. Calculating the radioactive release