Babcock & Wilcox Pressurized Water Reactors

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Course Description

This course provides an overview of the reactor and major reactor support systems found in a Babcock & Wilcox (B&W) Pressurized Water Reactor (PWR) Power Plant. Major systems associated with the reactor are discussed, such as, the reactor, steam generators, pressurizer, reactor coolant pumps, control rod drives, high pressure injection, residual heat removal system, reactor protection system, borated water system, and the letdown and make-up systems. In addition, the primary balance-of-plant (BOP) systems are discussed in terms of their system interface with the reactor. A short quiz follows the end of this course.

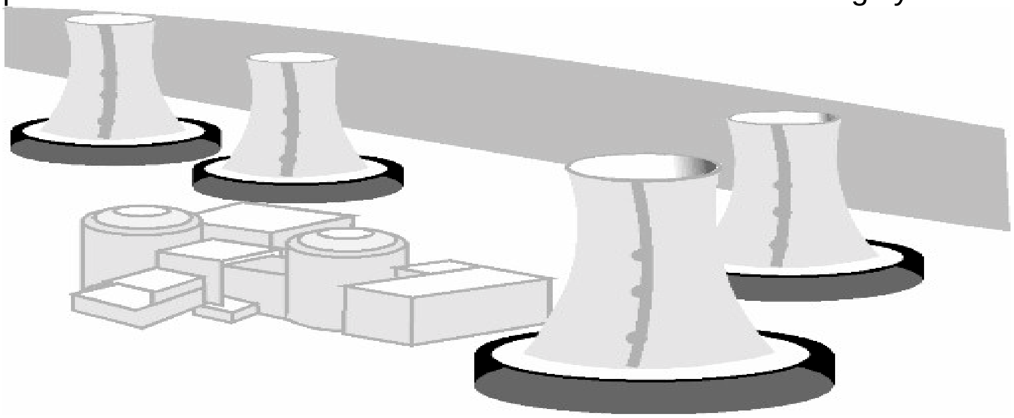
Learning Objectives

At the completion of this course the student should have an understanding of the following PWR systems in a Babcock &Wilcox plant:

- Reactor;
- Steam Generator;
- Pressurizer;
- Reactor Coolant Pumps;
- Control Rod Drive Mechanisms;
- Emergency Core Cooling Systems;
    - High Pressure Injection
    - Low Pressure Injection
    - Core Flood
    - Borated Water Storage
- Reactor Protection System.

Course Introduction

Utility owned and operated nuclear power plants have become a significant portion of the generation mix in the United States over the last thirty years. Although relatively expensive to build, compared to coal and oil power plants of the same vintage, nuclear plants have been very profitable to the utilities due to the low cost of fuel and the highly reliable



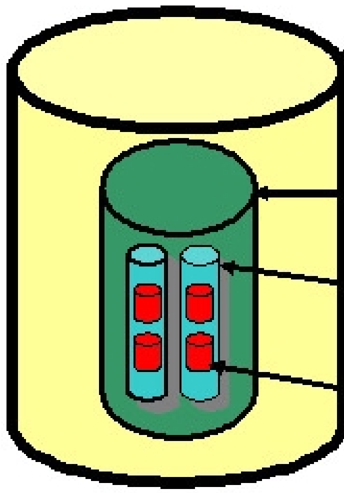
operating record. There were primarily two different types of reactor designs used in the United States: the Boiling Water Reactor (BWR), and the Pressurized Water Reactor (PWR) with the PWR being the predominant type. This course addresses PWR designs only. There were three major suppliers of PWR reactors, known in the industry as NSSS (Nuclear Steam Supply Systems) manufacturers. These suppliers were Combustion Engineering, Westinghouse, and Babcock & Wilcox (B&W). Westinghouse produced the most plants in the United States. This course is based upon the B&W designed reactors.

Disclosure: Each and every reactor in the United States is uniquely different. Even “identical” units have differences in their designs and construction. The material presented here is generic in nature; that is, what is commonly found in most plants, or in plants using a particular reactor manufacturer. The purpose of this course is to impart to a non-nuclear engineer a technical overview of the basic reactor and reactor support systems encountered in the United States.

Course Content

OVERVIEW

The nuclear power plants in the United States were built and operated by public utility companies to produce and sell electricity. The Federal Government issued licenses to operate these plants after an extensive review of volumes of safety analysis reports. During the early days of nuclear power, this governance was performed by the Atomic Energy Commission (AEC) which was created under the Atomic Energy Act of 1954. In the early 1970s, the AEC was split into the Energy Research and Development Agency (ERDA) and the Nuclear Regulatory Commission (NRC). The NRC was assigned the role to oversee commercial nuclear power.



Defense in Depth via Four Barriers

The principle of design for nuclear power plants was and still is defense in depth. Layers and layers of defense were put into place to assure the safety of the public. Some of the elements of the defense in depth philosophy are:

- Redundant Safety Systems – The systems necessary to support the safe shutdown of the nuclear reactor were designed with redundant and diverse backup systems. Only the highest quality materials went into the building of these systems.

- Automatic Reactor Protection Systems – These systems monitor critical parameters of the reactor system and automatically initiate shutdown of the reactor when the parameter limits are exceeded.

- Radiation Containment Barriers – Four physical barriers are designed to prevent radiation from escaping and reaching the public.

  - Fuel Design – the nuclear fuel is composed of ceramic pellets which contain most of the radioactive material within the fuel pellet.
  - Fuel Rods – the nuclear fuel pellets are placed in metal tubes that are welded shut to prevent the release of any material.
  - Reactor piping system – the reactor and the piping associated with the reactor system are composed of thick steel alloys and form a sealed system.
  - Containment Building – the reactor is housed in a steel and concrete building several feet thick. These buildings can

withstand the force of hurricanes and the impact of airplanes.

Figures 1, 2 and 3 in the following discussion will gradually explain a simplified diagram of a Babcock & Wilcox pressurized water reactor nuclear power plant. In order to simplify the drawing, only a single set of components is shown where the actual plant will have two, three or even four of the same components. Again, to simplify the drawing, valves have not been shown; however, different colors for fluid systems have been used and where different colors meet in a system there would be a valve, or pair of valves, in the closed position. Where piping is shown passing through the walls of the containment structure, generally there would be isolation valves.

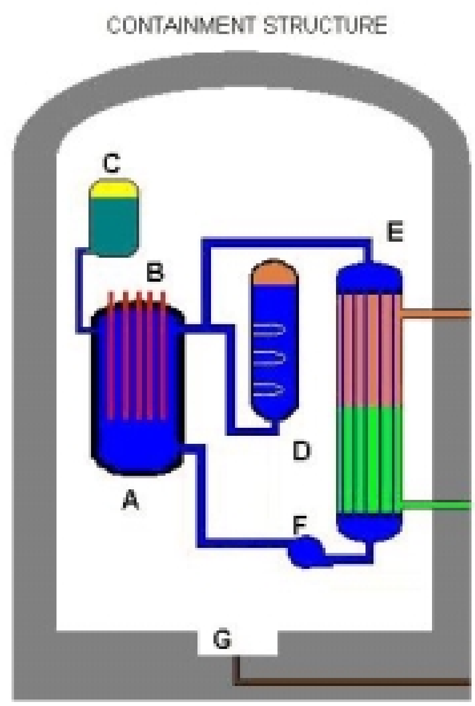


Figure 1 PWR Primary System

DRAWING LEGEND FIGURE 1

A Reactor Vessel
B Control Rods
C Core Flood Tank
D Pressurizer
E Steam Generator
F Reactor Coolant Pump
G Containment Sump

The best way to understand this simplified model of a PWR is to follow a drop of water through the different paths and equipment. In Figure 1, starting with the primary system (the Reactor Coolant System or RCS), a droplet of water begins the journey at the bottom of the Reactor Vessel A (this path is shown as dark blue). It passes through the reactor core and by Core Flood Tank C inlet and the Control Rods B, and then out of the reactor, bypassing the Pressurizer D outlet, and over to the Steam Generator E. The path from the Reactor to the Steam Generator inlet is known as the Hot Leg because this is the high temperature water from the Reactor. Once in the Steam Generator, the drop of water passes through one of the thousands of tubes in the Steam Generator and out through the bottom of the Steam Generator, where it enters the suction side of the Reactor Coolant Pump F. From here the drop of water is pumped through the pump and returns to the reactor to start the journey all over again. This path from the Steam Generator outlet to the reactor vessel is known as the Cold Leg because the water has dropped in temperature after transferring heat to the feedwater.

We are now ready to add another set of systems as shown in Figure 2 below.

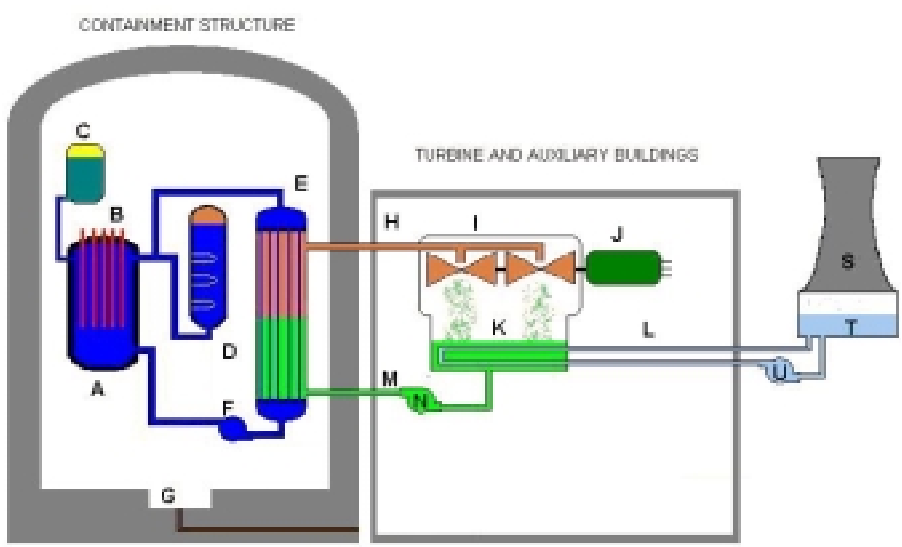


Figure 2 PWR Secondary Systems

DRAWING LEGEND FIGURE 2

A Reactor Vessel	**H Main Steam**	**S Cooling Tower**
B Control Rods	**I Turbine**	**T Tower Basin**
C Core Flood Tank	**J Generator**	**U Circ Water Pump**
D Pressurizer	**K Condenser**	
E Steam Generator	**L Circulating Water**	
F Reactor Coolant Pump	**M Feedwater**	
G Containment Sump	**N Feedwater Pump**	

On the secondary side of the system, a different droplet of water (shown as light green) begins a journey starting at the Feedwater Piping M. The water drop enters the secondary side, or" shell side", of the Steam Generator E, coming in close proximity to many of the thousands of tubes containing the reactor coolant. Here the drop of water is heated sufficiently to transform into steam where it exits the steam generator in the Main Steam Piping H and then travels to the Turbine I. The steam created from the drop of water expends its energy against the blades of the turbine, causing the turbine to spin and, in turn, the Generator J, producing electricity for the plant and for sale to the public. The steam from the water droplet, now greatly reduced in energy, returns to the water phase in the Condenser K and is referred to as condensate. From the condenser, the water droplet moves through a series of feedwater heaters (not shown) where it is preheated prior to entering the suction of the Feedwater Pump N. From the Feedwater Pump, the water drop enters the Feedwater Piping M to begin the journey again.

The Condenser K requires a large amount of water to condense the steam exiting the turbine. The Circulating Water L, which is pumped through the Condenser K by the Circulating Water Pump U in this diagram, passes to a cooling tower where it is sprayed into the tower and cooled by a natural circulation, or draft of air, through the cooling tower. The cooled water is stored in a basin beneath the tower. Other plants use large reservoirs of water, the ocean, lakes, or rivers to provide the circulating water without the need for cooling towers.

There are two more systems to add to complete this simple diagram of a PWR power plant. They are shown in Figure 3 below.

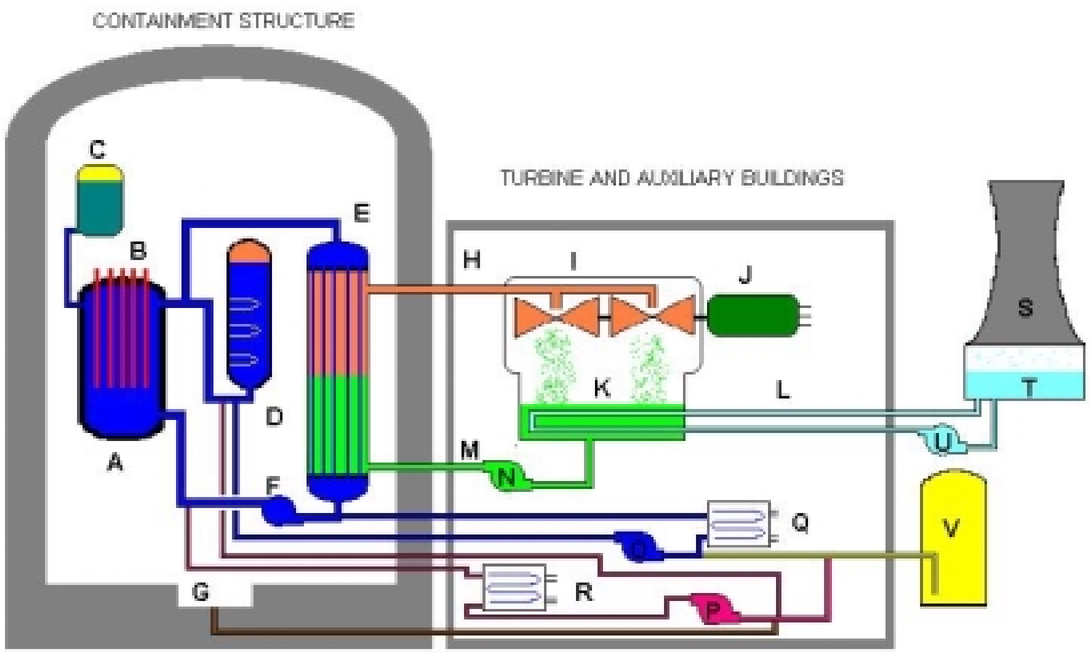


Figure 3 PWR Plant Diagram

DRAWING LEGEND FIGURE 3

A Reactor Vessel	**I Turbine**	**Q Letdown Cooler**
B Control Rods	**J Generator**	**R Decay Heat Cooler**
C Core Flood Tank	**K Condenser**	**S Cooling Tower**
D Pressurizer	**L Circ Water**	**T Tower Basin**
E Steam Generator	**M Feedwater**	**U Circ Water Pump**
F Reactor Coolant Pump	**N Feedwater Pump**	**V Borated Water Storage Tank**
G Containment Sump	**O HPI Pump**	
H Main Steam Line	**P LPI Pump**	

The next system shown is the High Pressure Injection (HPI) system. The HPI Pump O is a part of a system that can deliver fluid to the reactor at normal operating pressures. Water is withdrawn from the Reactor Coolant System (RCS), shown here being drawn from the Steam Generator E outlet, where it passes through the Letdown Heat Exchanger Q, where it is cooled. After the water temperature is reduced, the water is passed through purification systems (not shown) that remove impurities from the reactor water. The water is then reheated in another heat exchanger (not shown) and enters the HPI Pump O, where it is returned to the reactor. The HPI Pump O also can pump water from the Borated Water Storage Tank V to add boron to the RCS for reactivity control.

The final system shown in the diagram is the Low Pressure Injection (LPI) System. This system is used for cooling the reactor core after shutdown when the operating pressure of the RCS is much lower. The LPI Pump P can pump water to the Reactor Vessel A through the LPI Heat Exchanger R. The LPI Heat Exchanger gets cooling water from the design basis heat sink, or Ultimate Heat Sink (UHS) for the plant, which is a separate source of cooling water such as a pond or reservoir. The LPI pump can provide water to the reactor from the Borated Water Storage Tank. The LPI Pump P can also draw water from the reactor building Sump G in the event of a large pipe break in the RCS. Water which was pumped to the reactor to remove residual heat would settle in the basement sump and this water source can be pumped by the LPI Pumps through the LPI heat exchanger and then back to the reactor for post accident cooling.

THE RCS SYSTEMS

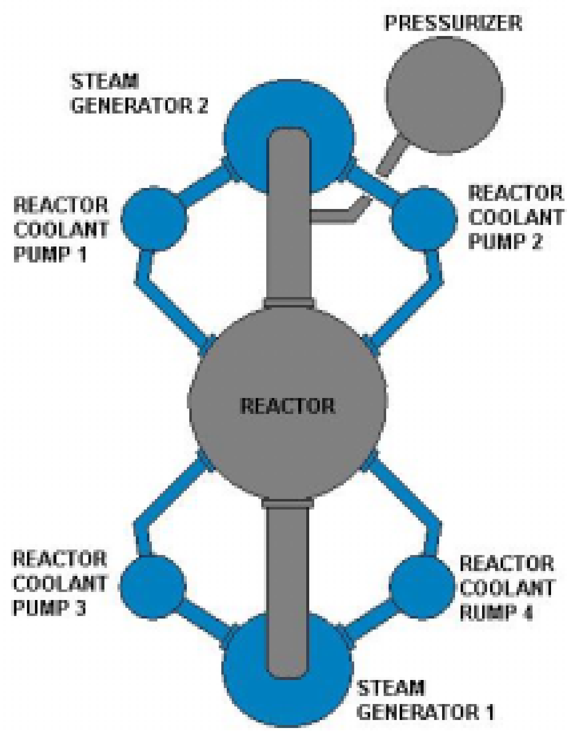
1) Reactor

The primary difference between the Boiling Water Reactor (BWR) and the Pressurized Water Reactor (PWR) is in how the steam that spins the main turbine generator is created. The BWR generates high pressure steam that leaves the reactor and goes directly to the turbine through the main steam piping. This steam is highly radioactive; therefore, the turbine and associated equipment in the steam cycle are highly contaminated. This is not true for the PWR design. In the PWR, the high temperature/pressure water circulates through the reactor in a continuous loop known as the primary system. Within this system, the energy is transferred from the primary water to the secondary water in the steam generator, which is a large vertical heat exchanger.

Pressure is controlled in a PWR by the Pressurizer. The pressurizer serves as a large volume and pressure control tank for the primary system. It has heaters for raising the water temperature inside the pressurizer and water sprays for condensing the steam volume and lowering pressure.

The reactor vessel and reactor coolant systems are typically constructed of carbon steel, with the interior surface weld-clad with stainless steel. The stainless steel cladding is necessary to control corrosion, thus, minimizing corrosion products in the primary system. Corrosion products are highly radioactive, creating radioactive dose for the station employees when they escape from the RCS through steam leaks and valve packing. The stainless steel liner is also necessary because the PWR uses a solution of boric acid and water for the reactor coolant to provide an additional means of controlling the nuclear reaction. The boron atom's nucleus has a large cross sectional area in relation to the size of a neutron. This allows for the capture of neutrons from the nuclear reaction and, as such, acts to slow down (poison) the reaction. The concentration of boron is rigorously controlled to specific limits, as too much will cause the reactor to shutdown, and too little will have the reactor create more power than desired.

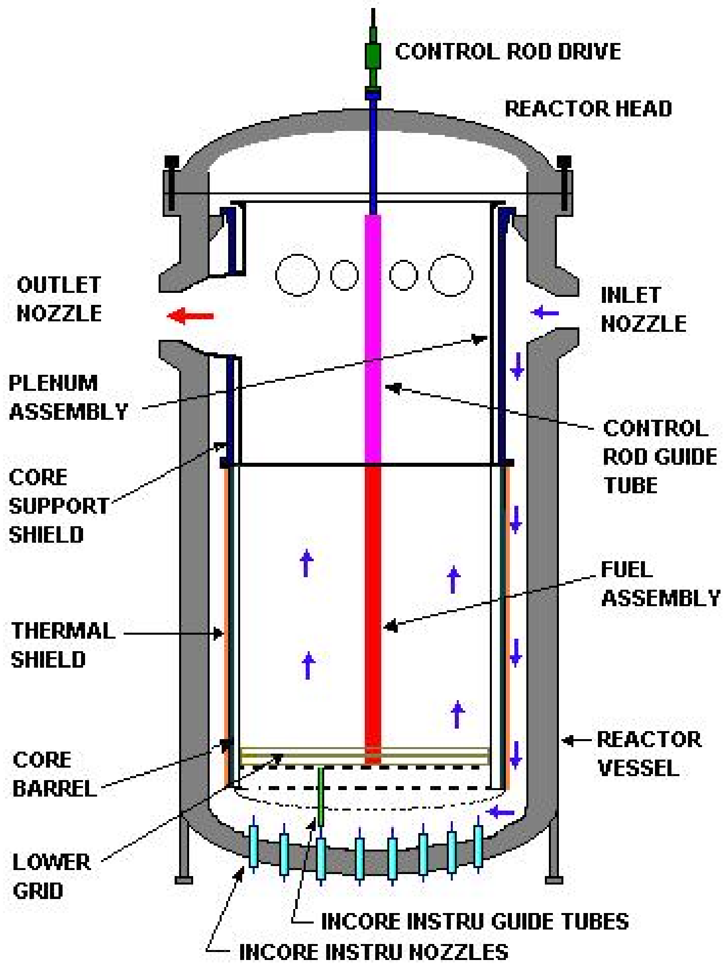
The following arrangement drawing (looking down from the top of the reactor building) shows the layout of a B&W reactor coolant system.



PWR 2 LOOP RCS ARRANGEMENT

Water heated by the reactor core leaves the reactor through the grey pipes and goes to the top of the steam generators. The B&W design uses two "once through" steam generators, which means the reactor coolant passes through the steam generator one time before being returned to the reactor to be re-heated. The four return lines (two for each generator and shown in blue), or "cold-legs", are approximately two and one-half feet in diameter, and the two return lines (one per steam generator), or "hot legs", are approximately three feet in diameter. The single Pressurizer connects to one of the hot legs.

The reactor itself is simply a pressure vessel with several internal assemblies. The primary functions of the reactor internals are to maintain fuel assembly alignment, support the core, and direct the flow of reactor coolant. The following figure will help illustrate some of the major sub-assemblies of the reactor.



REACTOR INTERNALS

The following is a brief description of each component shown in the figure.

1. Core Barrel – The core barrel is a cylinder with a flange on each end. The upper flange bolts to the lower flange of the core support shield assembly**;** the lower flange bolts to the lower grid assembly. The grid

assemblies provide a grid which holds and supports the fuel assemblies. The core barrel surrounds the fuel in the reactor core. The core barrel also supports the incore instrument guide tubes.

2. Thermal Shield – A stainless steel cylinder installed in the annulus, between the inside reactor vessel wall and the core barrel. The thermal shield provides protection to the reactor vessel walls by reducing the thermal stresses caused by gamma radiation from the core.

3. Core Support Shield – The Core Support Shield is a cylinder with a flange on each end. The upper flange mates with a recessed circumferential ledge in the top of the reactor on the reactor vessel closure flange (not shown). The lower flange is bolted to the Core Barrel.

4. Plenum Assembly – The plenum assembly is a group of reactor internals, bolted together, which sits above the reactor core. The entire Plenum Assembly is removed as a single unit when the reactor is refueled. It is composed of a flanged cylinder with openings for the outlet flow of reactor coolant, the upper grid, the Control Rod Assembly (CRA) guide tube assemblies, and the plenum cover.

5. Outlet nozzle – Nozzle for the RCS flow out of the reactor and into the "hot leg" of the RCS piping.

6. Control Rod Drive – The Control Rod Drive Mechanism (CRDM) is the electro-mechanical device which moves the control rods up and down in the reactor core. See the detailed section of this course for more information.

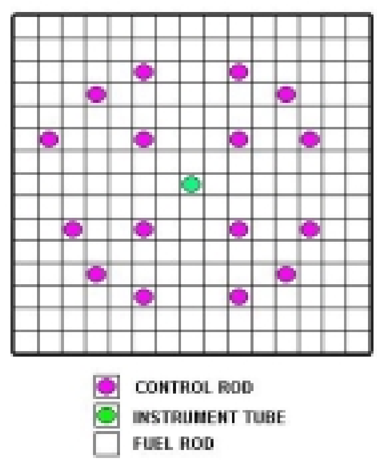
7. Reactor Head – The top portion of the reactor vessel which can be unbolted and removed for refueling the reactor. The reactor head has the Control Rod Drive Mechanism's nozzles and nozzle flanges. The CRDM bolts to these flanges.

8. Inlet Nozzle – The nozzle for the RCS flow into the reactor from the steam generator. As shown in the figure, the inlet nozzle is the return from the steam generator/reactor coolant pump. Here, reactor coolant enters the upper part of the reactor and flows down the annulus, between the reactor vessel wall and the reactor internals (which are surrounded by the core barrel and thermal shield).

9. Control Rod Drive Tube – A housing which provides alignment between fuel assemblies and their respective control rod drives. When

the control rods are withdrawn from the core, they are retracted up into the drive tubes.

10. Fuel Assembly – A fuel assembly is a long square tubular component made up of fuel rods and their supports, individual control rod guide tubes, and instrument tube assembly. The two hundred or so fuel rods are placed in lattice support structures. The control rods, sixteen for the type demonstrated in this course, are held together in a fixed pattern by a spider assembly. This assembly slides into the same pattern of guide tubes built into the lattice. Top and bottom sections complete the assembly. The guide tubes, spacer grids, and end fittings form a structural cage to arrange the rods and tubes in a 15 x 15 array with the center position in the assembly reserved for incore instrumentation.



FUEL ASSEMBLY CROSS SECTION

11. Reactor Vessel – The steel pressure vessel which surrounds the reactor core, control rod assemblies, and other reactor internals.

12. Incore Instrument Guide Tube - The incore instrument guide tube is an assembly which provides a path for the incore instrumentation assemblies from the vessel nozzles up to the instrument tube in the fuel assembly.

13. Incore Instrument Nozzles – Penetrations in the bottom of the reactor vessel for the incore instrumentation assemblies to penetrate the vessel.

14. Lower Grid – A cross hatch (lattice) arrangement of plates, oriented in a vertical position to form a grid of square channels to support the fuel assemblies.

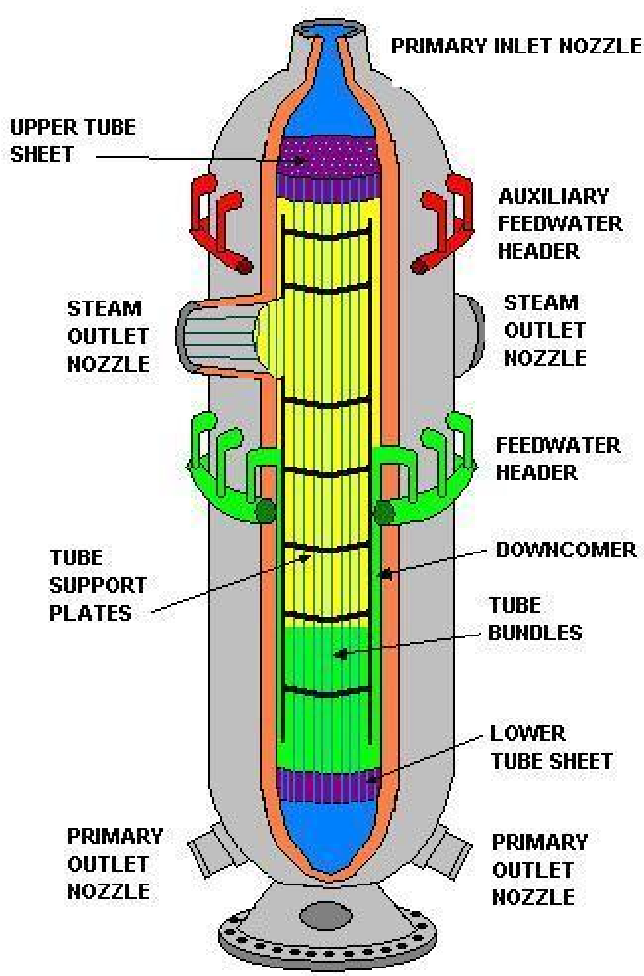
Again, as this is important, the primary functions of the reactor internals are to maintain fuel assembly alignment, support the core, and direct the flow of reactor coolant.

Not shown in the above Figure is another major component of the reactor which is the Reactor Service Structure. The Reactor Service Structure is a large flanged cylinder which is mounted on top of the reactor head. The top of the cylinder supports a platform. This platform provides support for the system of cables that connects to the top of the CRDMs which penetrate the platform. The circumference of the Reactor Service Structure cylinder also has openings for ventilation and may have cooling fans installed, which provide cooling during operation of the reactor.

2) Steam Generator

The Steam Generator design is the primary difference between the B&W and Westinghouse designed plants. The steam generator is a large vertical, tube and shell, heat exchanger. The B&W once-through steam generator, which is discussed in this course, uses a single pass of the primary fluid through a series of straight tubes. The steam generator provides superheated steam to the main turbine for the production of electricity. The steam generator also creates a boundary between the primary system and secondary system, keeping the radioactive fission products in the primary system.

The integrity of each tube is very important or the secondary systems will become contaminated. This can happen from even a small pinhole leak due to the pressure differential between the primary and secondary side. For this reason, the generator tubes are tested during refueling outages and, if thinning has produced a leak path or a potential leak path, that tube is permanently plugged. The steam generator must be able to remove the energy (heat) from the primary fluid to safely operate the reactor. There is a limit to how many tubes can be plugged**;** once the limit for plugged tubes is reached, the steam generator must be replaced.



Once Through Steam Generator

The above figure shows a cutaway view of a once-through steam generator. Reactor coolant (shown as light blue) enters the top of the steam generator through the inlet nozzle and flows through the upper tube sheet (shown as purple), down the inside of the approximately 15,000 tubes (shown as light blue). At the bottom, the reactor coolant passes through the bottom tube sheet and out the primary outlet nozzles, and on to the Reactor Coolant Pump suction (not shown). The upper and lower tube sheets are thick steel plates through which the tubes pass, and are seal welded to prevent mixture of the primary and secondary fluid. The tube support plates provide support and maintain spacing of the tubes over the full length of the tube bundles.

The exterior of the tubes, the tube sheets, and the generator shell form the steam producing surfaces in the steam generator. The tube bundles are surrounded by a cylinder baffle. The gap between this baffle and the shell form an annulus called the downcomer. The feedwater enters the steam generator into the annulus from a series of inlet nozzles connected to a header which circumferences the generator. Holes in the baffle allow crossover steam to mix with the feedwater, producing pre-heating of the water. The water in the downcomer provides a static head that balances the static head from the nucleate boiling regions.

Emergency (auxiliary) feedwater inlet nozzles are provided in a similar circumferential header arrangement at a higher elevation than the main feedwater header. The high elevation is, by design, to assist natural circulation in the unlikely event all Reactor Coolant Pumps are not functioning. In the event of the loss of feedwater to the steam generator, standby auxiliary pumps automatically start and provide emergency water to the auxiliary header. Mitigating the loss of all feedwater pumps is one of the Reactor Protection System's (RPS) automatic reactor trip functions. The auxiliary feedwater is needed immediately following the loss of feedwater event in order to remove residual heat from the reactor core.

There are four distinct heat transfer regions in the once-through steam generator:

1. Feedwater Heating Region – Although not shown in the simplified plant drawing at the beginning of the course, when the steam exiting the main turbine is condensed back to water, it is then pumped through a series of heat exchangers to raise the feedwater temperature prior to entering the reactor. All power plants do this to increase the efficiency of the steam cycle. The final stage of feedwater heating is done in the steam generator. The feedwater is sprayed through the feedwater nozzles down into the annulus. Steam is drawn by aspiration into the same annulus region through the holes in the baffle.

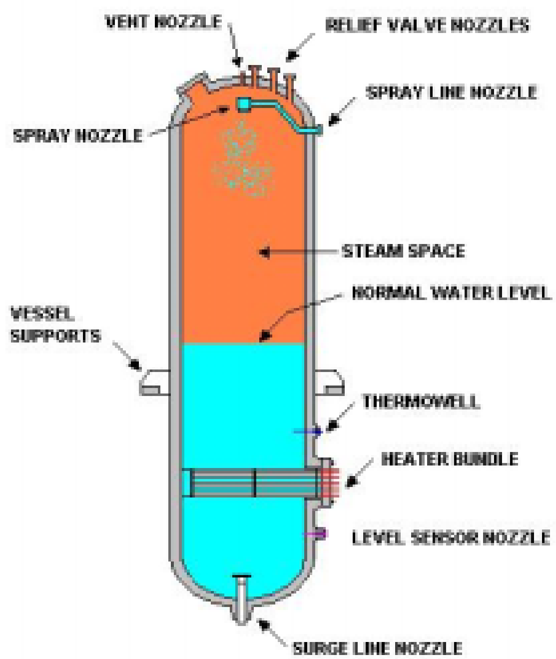
2. Nucleate Boiling Region – Nuclear boiling is the formation of vapor bubbles at a heating surface (such as a steam generator tube). These bubbles form at nucleation sites all over the surface, and the number and location of boiling sites depends upon the material, saturated liquid, and other factors. The saturated feedwater enters the tube bundle just above the lower tube sheet. The water and steam mixture travels up the outer surfaces of the tubes and increases vapor content through nucleate boiling.

3. Film Boiling Region – As the heating surface increases in temperature (the higher the elevation in the steam generator, the higher the reactor coolant inside the tubes), vapor begins to form a film on the tube surface. A Departure from Nucleate Boiling (DNB) occurs and a film boiling takes place.

4. Superheated Steam Region – The saturated steam rises higher in the tube bundle until it is raised to superheated temperatures. The amount of superheat surface available is dependent upon the load (as does the lengths of the other heating regions). The feedwater inventory and steam pressure of the outlet steam are controlled by the feedwater control system.

3) Pressurizer

The pressurizer is a vertical, cylindrical vessel used to maintain the pressure of the reactor coolant system within its design range. The pressurizer also provides a means of compensating for the volumetric changes of reactor coolant due to density changes. Means of pressure control are provided by a spray nozzle which can spray into the steam volume, code safety relief valves, a Power Operated Relief Valve (PORV), and also by electric heaters which can raise the temperature of the water in the water volume, producing steam. The drawing below illustrates the key components of the pressurizer.



PRESSURIZER DRAWING

The Reactor Coolant System (RCS) is a closed loop system with multiple heat sources**:** the reactor core, the pressurizer heaters, and the reactor coolant pumps.

> Sidebar**:** The reactor coolant pumps are large enough (the motors are 9000 Horsepower each) that the four pump impellers can add enough heat to the reactor coolant, such that, with the reactor shutdown, the steam generator can produce sufficient steam to roll and synchronize the turbine generator. This is actually done during the initial startup of the plant prior to the first time the reactor is taken critical.

As there are no isolation valves in the primary system, pressure relief devices are required. The pressurizer has two code safety relief valves and a Power Operated Relief Valve (PORV). Each of the relief valves and the PORV are mounted on individual nozzles on top of the pressurizer. The PORV also has an associated block valve to provide isolation in the event of a failure of the PORV in the open position.

The reactor is protected from over pressurization by the relief valves, by pressurizer spray, and by the Reactor Protection System (RPS). One of the automatic reactor trips in the RPS is initiated by high reactor pressure. The RPS is discussed in more detail later in this course. The spray nozzle's normal source of spray water is from the cold leg inlet to the reactor (reactor coolant pump discharge); thus, as long as that pump is operating, spray control is possible. When the RCP is not available, water from the High Pressure Injection Pumps (HPI) supplies the spray water. The HPI system is used for spray to cool down and finally de-pressurize the reactor coolant system.

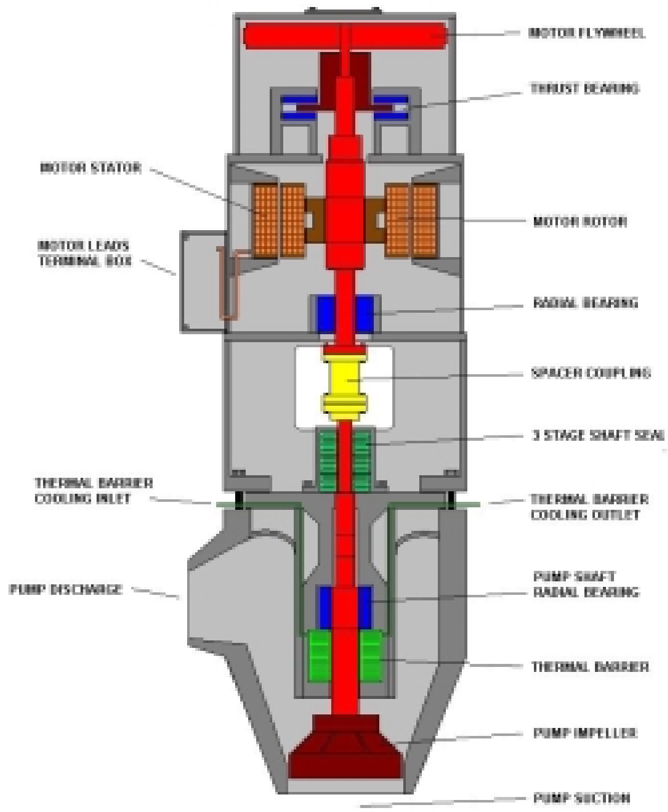
Pressure control is accomplished using the heater bundles. The drawing above shows one set of bundles for simplicity**;** there are three more bundles not shown. These large heater elements are powered by 480 VAC power sources in the plant. Three of the banks are operated in an on/off mode while the other bank is operated at a range of outputs controlled by a heater controller. The heater controller, spray valves, and temperature/level sensors are connected to control systems to maintain design limits. The normal operating pressure for the reactor is around 2200 psig (which is slightly more than twice the pressure at which BWRs operate).

4) Reactor Coolant Pumps

The Reactor Coolant Pumps are the prime movers of the fluid through the reactor coolant system. They must provide the motive force to move a large volume of water through the reactor, reactor coolant piping and steam generators. As previously discussed, B&W reactors were rated for approximately 2500 Mega-watts thermal and 900 Mega-watts electrical (MW) at full power. It takes four 9000 HP Reactor Coolant Pumps totaling 36,000 HP at full power.

Since:	1 HP = 736 Watts (W)	
	1 HP = 736 W x 1 KW/1000 Watts	
	1 HP = .736 KW	
Then:	36,000 HP	= 36,000 HP x .736 KW/HP
		= 26,496 KW
		= 26,496 KW x 1 MW/1000KW
		= 26.5 MW
Thus:	26.5MW / 900 MW = 2.94%	

This simplified calculation ignores motor efficiencies and actual horsepower at full load, but demonstrates that it takes approximately 3 per cent of the output of the power plant just to move the fluid through the reactor.



Reactor Coolant Pump Motor Diagram

The drawing above shows a cutaway view of a typical reactor coolant pump and motor.

The pump is driven by squirrel cage, vertical, induction motors. Each reactor coolant pump motor is equipped with a rotational inertia-increasing flywheel to provide a longer coast down-time following a reactor coolant pump trip. This allows for a more gradual reduction in flow by the pumps as the reactor is being shutdown by the automatic systems (RPS). The flywheel design also includes an anti-rotational device to prevent backflow in the system and to keep the pump from back spinning when not in service. This is done to decrease the starting duty on the large motors by not having to overcome the backward rotational inertia when starting the pump.

The motors and pump assembly is typically supported by a Kingsbury style thrust bearing. The motors have several sensors that provide the operators information about the motor and its support systems. These include:

- Upper bearing thrust plate thermocouples
- Lower bearing thrust plate thermocouples
- Oil reservoir thermocouples
- Oil level alarm
- Vibration alarms for shaft and frame

The purpose of this equipment is to provide the operators in the control room early warning of trouble, such as a bearing failure, so that corrective action can be taken to prevent the failure.

As discussed earlier, due to their large size, the reactor coolant pumps impart considerable energy to the reactor coolant. The reactor coolant pumps are started one at a time. They are used in the initial heat up of the reactor coolant in preparation for bringing the reactor back to service following a shutdown.

The pumps are typically single stage – constant speed, vertical, centrifugal pumps. The reactor coolant pump outer casing is part of the pressure boundary for the reactor coolant system and, as such, is considered equipment important to safety. Likewise, the shaft seals, which prevent the high pressure water from leaking, are pressure boundary safety related equipment. There are different types of shaft seal systems used with reactor coolant pumps**;** however, the details of these systems are beyond the scope of this course.

The remaining pump and motor components are not considered important to safety, as the pump function is not used to safely shutdown the reactor. That function is performed by other equipment. Nonetheless, the reactor coolant pump equipment is essential for reliable operation and is well designed with backup systems, monitoring instrumentation, and well executed maintenance. This equipment often operates for 18 months continuously without shutting down.

5) Control Rods and Control Rod Drive Mechanisms

Control Rods

The reactor power is proportional to the neutron flux within the core. The neutron flux, or neutron density, is controlled by the amount of boron in the RCS and by devices known as control rods. In a PWR there are four different types of rods:

1. Control Rod – Safety Rod

   Safety Rods are control rods that provide Safe Shutdown Margin. Shutdown Margin is the amount of reactivity by which a reactor is sub-critical (shutdown). The Safe Shutdown Margin is a prescribed amount of shutdown margin in the operating license of the plant. The Safe Shutdown Margin is usually the amount of reactivity from the "highest worth" rod when fully withdrawn from the reactor. This is another way of saying you have to be able to have enough control rods to shutdown the reactor if the most "powerful" rod (rod of the highest worth) fails to fall into the core upon demand.

   The Safety Rods are pulled out of the fuel assembly upon startup of the reactor and remain out during operations. They fall into the core upon a reactor trip.

2. Control Rods – Regulating Rod

   Regulating control rods are used to control reactor power during operation as well as shutdown of the reactor. They are located throughout the core, based upon the core design, and fall into the core upon a reactor trip. The regulating rods operate in prescribed groups which overlap each other.

3. Axial Power Shaping Rods

   These special rods are used to help control neutron flux imbalances in the reactor core. These rods do not fall into the core upon a trip due to their special design. They only have neutron absorbing material in the bottom portion of the rod.

4. Burnable Poison Rods (BPRs)

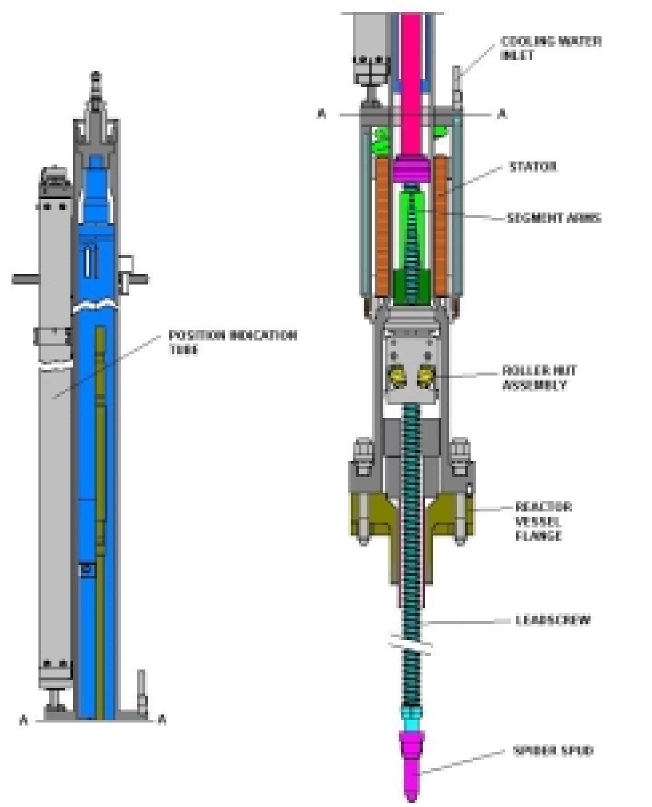
   These rods are designed to compensate for the excess reactivity loaded into the fuel for long fuel cycles. They are placed into the fuel assembly and remain there until the fuel assembly is spent and offloaded to the Spent Fuel Pool.

Control Rod Drive Mechanisms

The Control Rod Drive Mechanisms (CRDMs) are electro-mechanical devices which serve two functions:

1. Raise and lower or maintain the positions of the control rods (both Safety Rods and Regulating Rods) and the Axial Power Shaping Rods.

2. Following a Reactor Trip signal, they rapidly insert large amounts of negative reactivity by dropping the Safety and Regulating Rods into the core.



CONTROL ROD DRIVE MECHANISM

The CRDMs are comprised of the following key subcomponents:

1. Motor Tube

   The Motor Tube is the tubular housing which contains the moving parts of the CRDM. It mounts on top of the reactor head and has a flange which mates with the CRDM nozzle flanges. The motor tube is an extension of the RCS pressure boundary and is filled with reactor coolant.

2. Stator

   The Stator establishes a rotating magnetic field which provides the motive force for the CRDM. The stationary Stator surrounds the motor tube and, as such, is outside the RCS boundary. It is cooled by the Component Cooling system. The magnetic field generated by the stator engages the roller nuts in the motor tube.

3. Segment Arms

   The Segment Arms rotate, following the rotating field of the Stator. The segment arms are made of magnetic material. When the stator is energized, the upper portion of the segment arm is pulled outward, creating a pivot action to the lower portion of the arm, which moves inward. The lower portion of the segment arm contains the roller nut assembly.

4. Roller Nut Assembly

   The Roller Nut Assembly engages the threads of the leadscrew. As the segment arms are rotated, the roller nut translates the rotational movement to the raising or lowering of the leadscrew.

5. Leadscrew

   The Leadscrew is a long threaded shaft which mates with the Roller Nuts. The control element spider hub attaches to the lower end of the Leadscrew. The Leadscrew travels through the Roller Nuts, pulling the control element up or lowering the control element into the core.

Tripping the Reactor

When the reactor trip signal is generated by the RPS, the AC and DC circuit breakers providing power to the CRDMs are tripped, removing power to the Stators. A spring on the segment arms returns the segment arm to its normal position, disengaging the Roller Nut Assembly from the Leadscrew. This allows the control elements to drop into the reactor core **by the force of gravity alone**.

Earlier in the course, during the discussion of the reactor internals, the upper and lower grid assemblies were discussed. The great importance of proper vertical alignment of the fuel assembly and control rod mechanism should now be evident. Gravity is a good and reliable force for having the control rods rapidly insert into the core**;** however, miss-aligned configurations could prevent this rapid insertion. Rod drop tests are performed to assure that the rods do insert in the prescribed amount of time.

6) Emergency Core Cooling Systems (ECCS)

ECCS

There are three systems classified as the Emergency Core Cooling Systems (ECCS). Emergency Core Cooling Systems are equipment important to safety and are designed to safely shutdown the reactor in the event of a design basis event. They are the High Pressure Injection system, the Core Flood System and the Low Pressure Injection System (LPI). The ECCS is designed to provide cooling to the reactor core following any of the following design basis events:

1. A Loss of Coolant Accident (LOCA) is a pipe break or relief valve stuck open that allows coolant to discharge at a greater rate than can be offset by the normal makeup system.
2. Rupture of a control rod drive mechanism causing a rod ejection accident.
3. A Main Steam Line Break (MSLB) or feedwater system break which allows an uncontrolled steam release in the secondary system.
4. A steam generator ruptured tube.

The ECCS primary function is to remove the decay heat from the reactor during and after accident conditions. When the reactor is tripped (shutdown) and all control rods are inserted, the reactor goes sub-critical, which means the chain reaction producing thermal neutrons has stopped**;** however, the fission particles from the previous chain reactions continue to decay and this decay process continues to produce heat. This heat, known as decay heat, must be removed as it is sufficient to damage the fuel cladding if the heat is not removed from the core.

The ECCS also provides shutdown capability for the accident by injecting boron into the reactor coolant system. A large tank of borated water is maintained in a storage tank and can be injected by the HPI at operating pressures and by the LPI at lower pressures.

The ECCS has five modes of operation which are:

1. Injection of borated water from the Borated Water Storage Tank (BWST) by the HPI pumps
2. Injection of borated water by the Core Flood System
3. Injection of borated water from the BWST by the LPI system
4. Long term cooling by recirculation of the water in the containment sump back to the core by the LPI pumps
5. Gravity cooling by pumping the Reactor Building sumps to the reactor outlet piping by the LPI pumps

The high and low pressure systems operate independently and are designed to handle pipe breaks over a range of sizes. The HPI is designed for small pipe breaks, where the system pressure is maintained, and to prevent the core from being uncovered. The low pressure systems are designed for the larger pipe breaks that will cause a depressurization of the RCS and are designed to get the core recovered. The low pressure systems are also designed for allowing long term cooling of the core.

High Pressure Injection

The HPI is designed to operate at the operating pressure of the reactor system while the other systems can only work when the RCS pressure is much lower. The discharge of the HPI pumps connects to a nozzle on the reactor inlet, downstream of the discharge of the reactor coolant pumps.

The HPI system has seven functions it performs:

1. Supplies the reactor coolant system with makeup water to control temporary changes of the water volume due to changes in operating conditions.

2. Provides for the cleanup of the reactor coolant system water by the removal of corrosion and fission products. The HPI pumps allow the letdown water to be pumped back to the RCS following its purification**;** this allows for a continuous cleaning of the RCS water.

3. Provides the means to control the boric acid concentration in the reactor coolant system.

4. Maintains chemical concentrations in the reactor coolant system by providing a means for chemical injection.

5. Provides the high pressure seal injection water necessary for the reactor coolant pump seals.

6. Provides the Emergency Core Cooling System (ECCS) function during accident situations, allowing borated water to be added to the reactor coolant system

7. Provides Pressurizer spray when the reactor coolant pump is not available.

The ECCS function of High Pressure Injection is automatic and is initiated on low reactor pressure or high reactor building pressure, both of which are indicative of a break in the reactor coolant system piping or of a relief valve stuck open.

Low Pressure Injection

The Low Pressure Injection System is designed to maintain core cooling for large break sizes and for controlling the boron concentration when operating in the recirculation mode. LPI is initiated automatically at low reactor pressure (around 500 psig) or high reactor building pressure (3 or 4 psig).

In all, the LPI system has six different functions:

Normal Use

1. Reduces the temperature of the Reactor Coolant System during plant shutdowns from around 250° F to less than 140° F and maintains this temperature for extended periods of time.

Emergency Uses

2. Floods the reactor core with borated water immediately following a significant pipe break Loss of Coolant Accident (LOCA). This is done to prevent a significant amount of cladding failure which would result in the subsequent release of fission products.

3. Removes the residual (decay) heat from the core for extended periods of time following a significant Loss of Coolant Accident.

Sidebar: In B&W plants the LPI pumps are referred to as either the LPI pumps or Decay Heat Pumps. In Westinghouse plants the LPI pumps are referred to as the Residual Heat Removal (RHR) pumps. The same terminology is used in General Electric BWR plants where these low pressure pumps are called RHR.

4. Provides adequate Net Positive Suction Head (NPSH) for the HPI pumps during certain operating conditions when the LPI pump and HPI pump may be operated in series. This is known as piggyback operation.

5. Provides cooling of the water from the Reactor Building Emergency Sump, prior to its use as a suction source for HPI pumps.

Backup Use

6. Fills and drains the Fuel Transfer Canal as a backup to the Spent Fuel Cooling System. The fuel transfer canal is an underwater path that connects the spent fuel pool with the reactor cavity during refueling outages and allows new and used fuel to be moved out of the reactor building underwater.

The LPI systems tend to be one of the more complicated piping systems in most nuclear plants due to the several different modes of operation. The number of valves and valve interlocks can be quite large.

Core Flood Systems

The core flood system is a passive safety system designed to provide continuity of cooling following a large pipe break. The core flood system is comprised of tanks which are connected via piping to their own nozzles near the top of the reactor. The tanks are filled with borated water and have a nitrogen pressurized space at the top third of the tank. The pressurized nitrogen blanket provides the driving force to force the core flood tank borated water into the core once reactor pressure falls to around 600 psig. Reactor coolant is prevented from pressurizing the core flood tanks to RCS pressure levels by two check valves in series that will open at a low pressure (600 psig), allowing the flooding of the reactor core. Stop valves are provided for normal shutdowns to prevent the discharge of the tanks. Pressure relief valves provide over-pressurization protection, and instrumentation provides water level indication.

> Sidebar: On Westinghouse reactors the Core Flood Tanks are called Accumulators. They serve identical functions and have the same general design features.

Borated Water System

The Borated Water System is primarily a storage tank designed to hold a reservoir of purified water, to which a predetermined concentration of boric acid has been added. The boron acts as a posion to the nuclear chain reaction and works in conjunction with the control rods to maintain the reaction at a desired rate. The concentration of boric acid is calculated each fuel cycle based upon the amount and type of fuel loaded in the core.

The Borated Water Storage Tank (BWST) holds greater than 300,000 gallons of borated water that can be injected into the RCS by either HPI or LPI. The BWST can also be connected to the Building Spray System. The Building Spray System is a ring header of spray nozzles located in the top of the reactor building. This system sprays the atmosphere in the reactor building in the event of a LOCA. The spray primarily is used to quench the post-LOCA steam environment, thereby reducing the pressure in the building

The BWST has a normal function as a reservoir for filling the Transfer Fuel Canal during refueling operations.

Boron acid, in the concentrations used in nuclear power plants, will precipitate out of solution when the solution temperature to low. This requires tank heaters for the BWST, and heat tracing for the piping. When boron does precipitate, it will form rock like crystal formations that can plug a pipe completely.

7) Reactor Protection System

Recall from the introduction to this course the defense in depth philosophy. Layers and layers of defense have been put into place to assure the safety of the public. Some of the elements of the defense in depth philosophy are:

- Redundant Safety Systems – The systems necessary to support the safe shutdown of the nuclear reactor were designed with redundant and diverse backup systems.

- Automatic Reactor Protection Systems – These systems monitor critical parameters of the reactor system and automatically initiate shutdown of the reactor when the parameter limits are exceeded.

- Radiation Containment Barriers – Four physical barriers are designed to prevent radiation from escaping and reaching the public.

    - Fuel Design – the nuclear fuel is composed of ceramic pellets which contain most of the radioactive material within the fuel pellet.
    - Fuel Rods – the nuclear fuel pellets are placed in metal tubes that are welded shut to prevent the release of any material.
    - Reactor piping system – the reactor and the piping associated with the reactor system are composed of thick steel alloys and form a sealed system.
    - Containment Building – the reactor is housed in a steel and concrete building several feet thick.

The Reactor Protection System (RPS) is one of the primary defenses in depth in the design of the nuclear power plant. The design functions of the RPS are to monitor parameters associated with the safe operation of the reactor, to shutdown the reactor to prevent damage to the fuel cladding, and to prevent the reactor coolant system from exceeding its design pressure limits. The fuel cladding is the metal tube that holds the fuel pellets. The fuel pellets and fuel rod are two of the radiation containment barriers that are in place to protect the public from exposure to radiation. The RPS is, without question, the most important safety system in the power plant. It is designed to work fast and without any operator action.

The following is a simplified explanation of a more complex system, but serves the purpose of this course. The reactor is tripped by removing power from the Control Rod Drive Mechanisms. The RPS sends trip signals to the CRDM circuit breakers (both AC and DC). The RPS system is comprised of four protection channels. Each of these channels monitors several critical parameters. All of the monitored parameters provide a contact, wired such that the channel terminating relay is normally energized when all systems are within design limits. If the terminating relay de-energizes for any reason, it sends a single trip signal to the final RPS trip logic. Any two channels sending a trip signal will cause the RPS trip logic to send the trip signal to the CRDM breakers undervoltage trip relays. The undervoltage relays cause the CRDM breakers to actually trip.

The protection channels are required to be tested and checked on a regular basis. Bypass switches are located in the control room and allow any one of the four channels to be placed in the bypass mode for testing. When one channel is placed in bypass, the RPS works on a 2 out of 3 logic in which any two of the remaining three logic channels will cause a reactor trip. Bypass switches are typically key operated switches under the administrative control of the control room shift supervisor.

The following is a list of typical RPS protective functions:

1. High Reactor Flux (power) – the nuclear in-core instrumentation monitors the neutron flux in the core. Neutron flux (reactor power) is proportional to the amount of heat in the reactor. Recall that the reactor coolant pumps are constant speed pumps and therefore provide a fixed amount of flow through the reactor. RPS monitors reactor power and trips the reactor upon exceeding a limit based upon the amount of heat the reactor coolant pumps can remove.

2. High Reactor Flux versus Reactor Coolant Flow – RPS looks at two different sets of parameters regarding flux and flow.

    a. First, the flux in the lower half of the core is compared to the flux in the upper half of the core. If power is significantly different between the upper and lower parts of the core, an imbalance exists. This is not good since the total core flux may still be within the limits for the high flux trip. Since flow is the same to the top and bottom of the core, the amount of flux versus the flow must be monitored to assure adequate cooling is available to all parts of the core. RPS looks at the amount of imbalance and the amount of core flow and determines whether the reactor should be tripped.
    b. RPS monitors the number of reactor coolant pumps running and determines if there are sufficient pumps for the power level and trips the reactor if there is not.

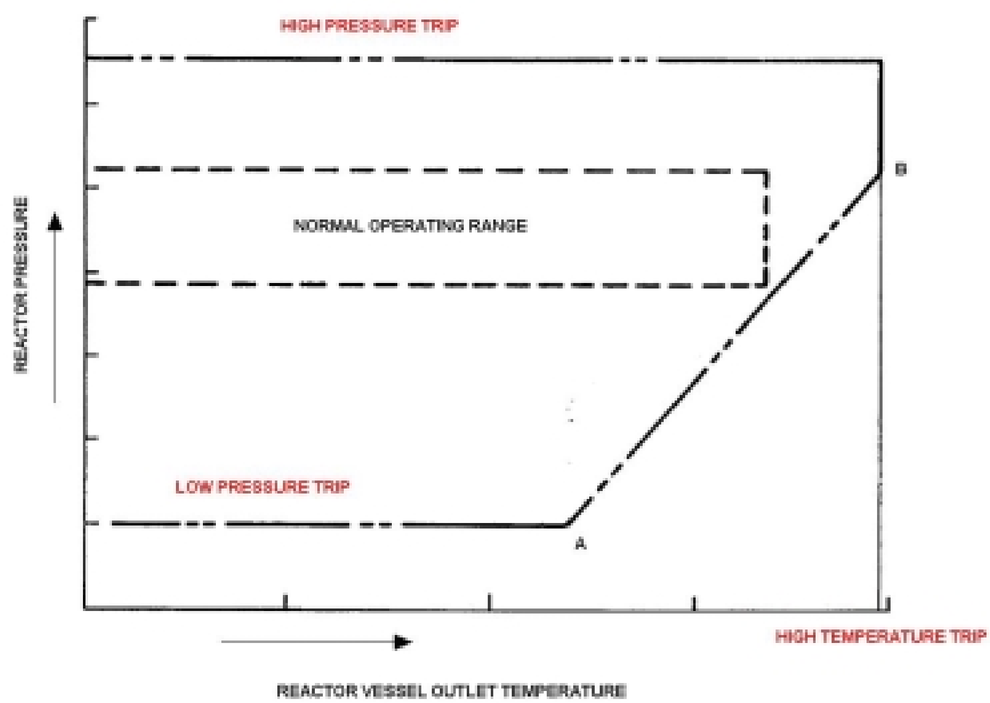
3. High Reactor Flux versus the number of Reactor Coolant Pumps in service – the number of pumps in operation and

the reactor power level are monitored and compared. The reactor is tripped when power exceeds limits for the number of pumps in service (the reactor is tripped upon the loss of two reactor coolant pumps).

4. High reactor coolant outlet temperature – the reactor is tripped when the outlet temperature from the core exceeds a set limit. High reactor coolant outlet temperature is indicative of any number of problems including:

    a. Reactor power is too high
    b. Reactor coolant flow is to low
    c. Insufficient steam is leaving the steam generator
    d. Feedwater to the steam generator is too low

    Rising temperatures could lead to the water boiling in the reactor, which is undesirable due to the possibility of uncovering the core.



PRESSURE AND TEMPERATURE BOUNDARIES

5. High reactor coolant outlet temperature versus pressure – the RPS trips the reactor whenever the temperature and pressure combination goes outside the limits established by the line AB in the above figure.

6. Low reactor coolant pressure – the reactor is tripped whenever the pressure goes below the established trip setting. This trip is necessary when reactor power is high to prevent bulk boiling of the water inside the reactor vessel, which could result in uncovering the core.

7. High reactor pressure – the reactor is tripped whenever the pressure goes above the established trip setting. This trip function is necessary to protect the RCS pressure boundary from over pressure.

8. High reactor building pressure – the reactor is tripped upon detecting higher than normal pressure in the reactor building. Higher than normal pressure may be indicative of a pipe break.

9. All Feedwater Pumps trip – The loss of feedwater to the steam generator causes a reactor trip by the RPS. Without feedwater, the energy from the RCS can not be removed and rising pressure would result if the reactor is not tripped. In anticipation of this transient, the reactor is tripped. This type of reactor trip is called an anticipatory trip.

10. Main turbine trip – the reactor is tripped whenever the main turbine is tripped, as another anticipatory trip. Once the turbine is tripped, there is no place to expend the energy being created by the reactor, so reactor pressure will rapidly rise. The reactor trip is initiated following a turbine trip to avoid the pressure spike. This trip is by-passed at low power (flux) levels during plant startup.

> Sidebar: The loss of all feedwater anticipatory trip and turbine anticipatory trip were added to the B&W plant's RPS following the accident at Three Mile Island (TMI). This was done to reduce challenges to the PROV on the pressurizer during the pressure spike which follows a turbine or feedwater trip. At

TMI, following a loss of feedwater event, the PORV opened, as designed, to relieve pressure from the resulting pressure spike**;** however, when the PORV failed to re-close and was not discovered, a de-pressurization of the RCS occurred, ultimately resulting in boiling and uncovering the core.

References:

Material for this course was summarized from the following references:

General Physics Corporation "Generic Fundamentals: Reactor Theory"
General Physics Corporation "Generic Fundamentals: Components"
Final Safety Analysis Reports for B&W plants
All sketches and diagrams are by the author

Glossary of Terms

- Anticipatory Trip – A protective function performed based upon an expected transient due to some other initiating event
- Burnable Poison Rod – a control rod designed to compensate for the excess reactivity loaded into the fuel for long fuel cycles.
- BWR –Boiling Water Reactor- Reactors designed by General Electric which use a single primary system, and where steam is generated in the reactor versus in a steam generator
- Channel – An instrumentation wiring loop, independent unto itself (i.e. separated and distinct from another channel)
- Cold Leg – The reactor piping that goes from the steam generator outlet to the reactor inlet.
- Control Rod Drive – or Control Rod Drive Mechanism (CRDM) is the electro-mechanical device which moves the control rods up and down in the reactor core
- Control Rod Assembly (CRA) – A reactor internal component composed of 16 control rods, the spider support, and female coupling.
- Core – The reactor fuel assemblies found in the lower half of the reactor vessel.
- Core Barrel – The lower cylinder assembly in the reactor internals that supports the fuel assemblies, lower grid, and incore instrument guide tubes.
- Core Support Shield – A reactor internal cylindrical component with a flange on each end. The upper flange mates with a recessed circumferential ledge in the top of the reactor and the lower flange is bolted to the Core Barrel.
- DBE - Design Basis Event – An accident scenario the plant has been designed to withstand

- Departure from Nuclear Boiling – That point where a vapor film begins to cover the heating surface where nucleate boiling was occurring
- ECCS -Emergency Core Cooling Systems- equipment important to safety, designed to safely shutdown the reactor in the event of a design basis event
- Fuel Transfer Canal - an underwater path that connects the spent fuel pool with the reactor cavity during refueling outages and allows new and used fuel to be moved out of the reactor building underwater
- Hot Leg – The reactor piping from the reactor outlet to the steam generator
- HPI – High Pressure Injection
- Inlet Nozzle - Nozzle for the RCS flow into the reactor from the “cold leg” of the steam generator
- Incore Instrument Assembly – A tubular assembly which slides up into a fuel assembly and contains radiation monitoring instruments
- Incore Instrument Guide Tube – An assembly in the lower reactor internals which provides a path for the incore instrumentation assemblies from the vessel nozzles up to the instrument tube in the fuel assembly.
- Incore Instrument Nozzles – Penetrations in the bottom of the reactor vessel for the incore instrumentation assemblies to penetrate the vessel
- Leadscrew – Sub-component of the CRDM which attaches to the control element spider assembly and is engaged by the roller nut assembly
- Lower Grid – Reactor internal component which supports the fuel assemblies near the bottom of the core
- LPI – Low Pressure Injection – ECCS designed to operate at low pressures following a large pipe break in the RCS
- Nucleate boiling - the formation of vapor bubbles on a heating surface at numerous individual locations
- Outlet nozzle – Nozzle for the RCS flow out of the reactor and into the “hot leg” of the RCS piping
- Plenum Assembly – A group of reactor internals, bolted together, which sits above the reactor core and can be removed as one assembly for refueling
- PORV – Power Operated Relief Valve – One of the pressurizer’s pressure relief valves
- PWR – Pressurized Water Reactor
- NSSS – Nuclear Steam Supply System – name given to the manufacturers of nuclear reactors
- RCS – Reactor Coolant System – the reactor, reactor piping, pressurizer, and steam generators.
- Reactor Service Structure - A steel assembly mounted to the reactor head that, in turn, supports a support platform for attaching the CRDM cables
- Roller Nut Assembly – Sub-component of the CRDM that engages the leadscrew for movement of the rod
- Shutdown Margin - The amount of reactivity by which a reactor is sub-critical (shutdown)

- o Spider Support – A stainless steel reactor internal component that holds 16 control rods in a fixed pattern
- o Stator – Sub-component of the CRDM which produces the electro-magnetic field to rotate the CRDM Segment Arms
- o Trip – The automatic or manual shutdown of a device
- o UHS –Ultimate Heat Sink – A guaranteed water source for decay heat removal following a design basis accident

Conclusions

This course has provided an overview of the reactor and major reactor support systems for a Babcock & Wilcox Pressurized Water Reactor. Primary differences between the B&W design and other PWR designs have been highlighted. Lessons learned from the accident at Three Mile Island have been incorporated into the designs of B&W reactors and some of these were identified in this course.

Nuclear power continues to provide a critical portion of the electricity generated in the United States and will into the future as most utilities seek and obtain twenty year extensions to their operating licenses. As the uncertainty of future oil supplies plagues this country, nuclear power will continue to be the safe, clean, and reliable energy source of energy that it is today.