Westinghouse Technology Systems Manual

Section 1.2

<u>Introduction to Pressurized Water Reactor Generating Systems</u>

TABLE OF CONTENTS

LIST OF FIGURES

1.2 INTRODUCTION TO PRESSURIZED WATER REACTOR GENERATING SYSTEMS

Learning Objectives:

1. Define the following terms:

    a. Primary cycle,
    b. Secondary cycle, and
    c. Reactor coolant system (RCS) average temperature (T_{avg}).

2. Explain why T_{avg} is programmed to increase with an increasing plant load.

1.2.1 General Description

The pressurized water reactor (PWR) generating system described in this manual is a dual cycle unit. The two cycles are called the primary and the secondary. As shown in Figure 1.2-1, the unit consists of a primary cycle that includes the reactor vessel, the pressurizer, and four closed reactor coolant loops connected in parallel (of which only one loop is shown). The secondary cycle includes the steam system, the high and low pressure turbines, and the condensate and feedwater system. The secondary systems together are sometimes referred to as the power conversion system. The sole function of the power conversion system is to generate electricity.

As shown, the primary cycle is located entirely inside the containment building. This building is designed to act as a shield to minimize the exposure of plant personnel to radiation. In addition, the design of the containment structure prevents or minimizes the release of radioactive material to the environment during normal operation or under an accident condition. The use of a dual cycle design reduces the amount of radioactive material transferred to the power conversion system components. By minimizing the amount of radioactivity transferred to the secondary, the potential exposure of plant personnel is reduced, and any potential releases of radioactive material to the atmosphere are minimized.

1.2.2 Primary Cycle (Reactor Coolant System)

Each of the four reactor coolant loops contains a reactor coolant pump, a steam generator, piping, and associated instrumentation. Attached to one of the four loops is an electrically heated pressurizer. The pressurizer maintains the pressure of the reactor coolant at a high value, which prevents the high temperature (>500°F) coolant from boiling. Reactor coolant (pure water with boric acid in solution) is pumped through the reactor core to remove the heat generated by nuclear fission. The heated water exits the reactor vessel, passes through loop piping and enters the steam generator.

Inside the steam generator, reactor coolant flows through U-tubes and transfers heat to the feedwater inside the steam generator (secondary system). The U-tubes act as a barrier between the primary and secondary cycles. Reactor coolant, now cooler, exits the steam generator and is directed to the suction of the reactor coolant pump. The reactor coolant pump returns the reactor coolant to the reactor vessel, completing the primary cycle.

1.2.3 Secondary Cycle (Power Conversion System)

The power conversion system begins in the shell sides of the four steam generators. At these locations the feedwater contacts the U-tubes and picks up heat from the hot reactor coolant. Since the pressure in the secondary side is less than that of the primary, the heated feedwater boils and becomes saturated steam. Saturated steam is steam that is at the same temperature as boiling water for a given pressure. The saturated steam produced in the shell sides of the steam generators exits via the main steam lines. The steam flows through the main steam line isolation valves (MSIVs) to the high pressure turbine. After flowing through the high pressure turbine, the low-energy, moisture-laden steam is routed to the moisture separator reheaters (MSRs). Each MSR, as its name implies, removes moisture from this low pressure steam and reheats it. The moisture-free steam is superheated by extraction steam from the high pressure turbine and by steam from the main steam lines. Superheated steam is steam that is at a temperature which is greater than the saturation temperature for a given pressure. The dry, superheated steam is directed to the low pressure turbines. This steam passes through the low pressure turbine blades and exits to the main condenser. The high and low pressure turbines are mounted on a common shaft that drives the main generator.

The main generator produces electrical power, which is supplied to the utility's distribution network or "grid."

Inside the condenser, the exhausted steam is condensed (cooled and depressurized) by passing over tubes containing water from the condenser circulating water system. The condensed steam (now called condensate) is collected in the condenser's hotwell. The condensate is pumped from the condenser hotwell by condensate pumps. The condensate pumps discharge the condensate through condensate demineralizers, which remove impurities. The condensate then passes through several stages of low pressure feedwater heaters, in which the temperature of the condensate is increased by heat transfer from steam extracted from the low pressure turbines. The condensate exits the low pressure feedwater heaters and enters the suctions of the high pressure main feedwater pumps.

The main feedwater pumps (normally driven by steam turbines) increase the pressure of the condensate (now called feedwater) so that it can enter the steam generators. From the discharge of the main feedwater pumps, the feedwater is heated in the high pressure feedwater heaters by extraction steam from the high pressure turbine. After this final heating, the feedwater passes through the feedwater regulating valves (FRVs), enters the containment, and finally enters the steam generators, thereby completing the secondary cycle.

1.2.4 Support and Emergency Systems

Attached to each reactor coolant loop cold leg is an accumulator, pressurized with nitrogen. The purpose of the accumulator is to inject borated water into the RCS if the reactor coolant system pressure boundary ruptures; i.e., a loss of coolant accident (LOCA) occurs.

When the pressure in the RCS drops below the pressure in the accumulators, the nitrogen forces the borated water out of the accumulators into the RCS, providing both water to cover and cool the reactor core and boron (a neutron absorber) to keep the reactor shut down.

The residual heat removal (RHR) system is designed to provide both safety-related and nonsafety-related functions. Its safety-related function is to provide borated water at a low pressure and a high flow rate to the RCS following a loss of coolant accident. The RHR system pumps water from the refueling water storage tank (RWST) to the RCS for the short term and recycles water from the containment building sump back into the reactor coolant system for long term cooling. Its nonsafety-related function is to remove decay heat from the core after a shutdown. Decay heat removal is accomplished by pumping hot water from an RCS hot leg through heat exchangers and then back into the RCS via the cold legs.

The safety injection (SI) system is another emergency core cooling system located in the auxiliary building. Its function is to inject borated water from the RWST into the RCS after a LOCA. Although the SI system discharge capacity is much less than that of the RHR system, its discharge pressure is greater.

The chemical and volume control system (CVCS) maintains the purity of the reactor coolant by means of demineralizer beds that continuously purify a small letdown stream from the RCS. This purified water is charged back into the RCS at a controlled rate to maintain the proper volume of water in the RCS. The CVCS charging pumps also serve as the high pressure safety injection pumps. Their function is to supply borated water to the RCS in emergency situations.

In the event of a LOCA, the hot reactor coolant spills from the RCS into the containment and flashes to steam. This action causes a pressure increase inside the containment building.

The containment spray system is designed to transfer water from the refueling water storage tank to spray rings located high inside containment. The cool water sprayed into the containment quenches the steam and maintains the pressure inside of the containment within design limits. This action prevents the rupture of the containment building and the subsequent uncontrolled release of radioactive materials to the environment.

The component cooling water (CCW) system provides a cooling medium to various components, such as the CVCS letdown heat exchanger and the RHR heat exchangers. This system is a closed loop system and is cooled by the service water system (SWS), which receives its water from a nearby river, lake, or ocean. Both of these systems (CCW system and SWS) are safety systems and are required to function in order to mitigate the consequences of analyzed accidents.

1.2.5 Plant Layout (Figure 1.2-2)

The entire RCS, including the steam generators, is located within the containment building. This structure isolates the radioactive reactor coolant from the environment in the event of a leak or a loss of coolant accident. The containment building is designed to withstand the pressure resulting from the complete rupture of a reactor coolant pipe or main steam line. The containment must be able to perform this safety function during and following a “design basis earthquake.” The containment building is therefore designated as a Seismic Category I structure.

A "design basis earthquake" (also called a "safe shutdown earthquake”) is defined in 10 CFR Part 100, Appendix A, of the Code of Federal Regulations as the maximum ground motion potential considering local and regional geology and seismology. It is that earthquake which produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. "Safety-related" systems, structures, and components designed to remain functional during a design basis earthquake are designated "Seismic Category I."

Safety-related and potentially radioactive auxiliary systems are located inside the Seismic Category I auxiliary building. This building is normally located between the turbine building and the containment building. Ventilation from the auxiliary building is passed through high efficiency particulate filters and/or charcoal filters to minimize the release of radioactive material to the environment. A fuel storage building (sometimes a part of the auxiliary building) is provided for the handling and storage of new and spent reactor fuel. The fuel storage building is also designated as a Seismic Category I building. The control building (also sometimes part of the auxiliary building) is a Seismic Category I structure, which houses the main control room, the cable spreading room, the auxiliary instrument room, the plant computer, and the battery rooms.

The turbine building is not safety related and contains most of the secondary cycle equipment and secondary support systems. The main turbine, moisture separator reheaters, main condenser, condensate and feedwater pumps, and feedwater heaters are all located inside the turbine building.

1.2.6 Plant Control

The power output of the reactor and the outlet temperature of the coolant from the reactor core are controlled by manipulating several factors which affect the core's reactivity (a measure of neutron population change). The position of neutron absorbing control rods, the concentration of boric acid in the RCS, and the steam flow rate can be changed to affect reactor power and the coolant outlet temperature.

The automatic control systems are designed to provide power change (load change) capability between 15% and 100% of rated power at 5% per minute (ramp) or with a 10% instantaneous (step) change in power without causing an automatic reactor shutdown (trip). Additionally, the plant's steam dump system is designed to direct steam at a high rate of flow to the main condenser, allowing the unit to accept a large power reduction (load rejection) without tripping the reactor.

The power level of the reactor is normally changed by selecting a desired electrical load and load rate via the turbine control system and allowing the reactor to follow the turbine load change. Various methods are possible for controlling the reactor's power, as the turbine load is changed, and are discussed below.

1.2.7 Reactor Control

The basic formula defining heat (or power) transferred across a heat exchanger (in this case, the steam generators) is:

$$\dot{Q} = UA\Delta T$$

where:

$\dot{Q}$ = the rate of heat transfer,
U = the heat transfer coefficient,
A = the area of heat transfer, and
ΔT = the differential temperature across the heat exchanger (in this case, the difference between the average temperature of the reactor coolant [T_{avg}] and the temperature of the steam [T_{stm}]).

For all practical purposes, both the heat transfer coefficient (U) and the heat transfer area (A) are constant, since the heat transfer coefficient is a function of the materials used in the construction of the steam generator and the U-tubes are completely covered with water. The equation may be reduced to:

$$\dot{Q} \propto \Delta T \text{ , or}$$

$$\dot{Q} \propto (T_{avg} - T_{stm})$$

There are three basic modes of controlling a pressurized water reactor with U-tube steam generators. Each of these modes of control could be used to adjust reactor power in response to changes in one of two measurable parameters. These parameters are as follows:

1. T_{avg} - the average reactor coolant system temperature:

$$T_{avg} = \frac{T_h + T_c}{2}$$

where:

T_h = hot leg temperature, and
T_c = cold leg temperature.

2. Steam pressure - the secondary steam pressure either at the outlet of the steam generator or the inlet to the main turbine.

1.2.7.1 Constant T_{avg} Control Mode

With a constant T_{avg} control scheme (Figure 1.2-3), the reactivity of the core is adjusted to maintain a constant reactor coolant average temperature as turbine load is varied. For example, increasing the output of the turbine causes a decrease in T_{avg}, because the turbine uses more energy than that produced by the reactor. The rod control system senses this temperature decrease and withdraws the control rods, adding positive reactivity to the core and returning T_{avg} to the programmed value. An anticipatory signal comparing turbine load and reactor power might also be utilized to optimize the transient response of this control scheme.

The constant T_{avg} control mode has the advantage of an unchanging RCS temperature and density, regardless of power level. Since the coolant volume does not change, the pressurizer level is constant for all load conditions.

A major disadvantage of the constant T_{avg} control is that it produces an unacceptable secondary system pressure when the turbine is fully loaded. As shown above, the rate of heat transfer from the reactor coolant across the steam generator tubes to produce steam is proportional to the differential temperature between the reactor coolant and the secondary water. If the reactor coolant T_{avg} remains constant, then the saturation temperature (T_{stm}) in the shell side (secondary) of each steam generator must drop when the steam demand (load) increases. This effect produces a significant decrease in steam pressure (P_{stm}) as secondary power is increased from hot zero power to full load. The low steam pressure produces unacceptable steam conditions at the main turbine inlet.

Large PWR generating stations with U-tube steam generators do not use constant T_{avg} control. The advantage of a constant pressurizer level is greatly offset by the disadvantage of low steam pressure.

1.2.7.2 Constant Steam Pressure Control Mode

With a constant steam pressure control scheme (Figure 1.2-4), the reactivity of the reactor core is adjusted to maintain a constant pressure in the steam system as turbine load is changed. As described in the previous section, increasing turbine load causes steam pressure (P_{stm}) to decrease. The rod control system would sense this decrease in P_{stm} and withdraw control rods to increase the reactor coolant temperature.

With this type of reactor control, the ΔT between primary and secondary is increased by raising T_{avg} and allowing T_{stm} (and therefore P_{stm}) to remain constant. This produces ideal steam conditions at the main turbine inlet for all loads from hot zero power to 100% load.

The disadvantage of this type of control scheme is that it results in a high reactor outlet (hot leg) temperature (T_h), which approaches saturation values. The constant steam pressure control may be used in other PWR vendor designs but is impractical for plants with U-tube steam generators.

1.2.7.3 Sliding T_{avg} Control Mode

A sliding T_{avg} control scheme (Figure 1.2-5) is a compromise between a constant T_{avg} and a constant steam pressure control scheme. This control scheme incorporates the advantages of both but also retains some of their disadvantages.

With a sliding (programmed) T_{avg} control scheme, the reactivity in the reactor core is adjusted to maintain a programmed T_{avg} as the turbine load is varied. As with the previously described control schemes, varying the load on the turbine causes T_{avg} and steam pressure to change. To compensate, the rod control system repositions the control rods to add positive or negative reactivity to the reactor core and maintain T_{avg} equal to its programmed value for a given load.

The heat transfer rate from the primary cycle to the secondary cycle is directly proportional to the value of the temperature difference (ΔT) between the primary and the secondary. This ΔT is also a direct indication of power. As shown in Figure 1.2-5, as secondary power increases, the difference between T_{avg} and T_{stm} increases. The ΔT increases with an increasing load on the turbine generator as (1) the turbine control valves open (causing the steam pressure to decrease), and (2) control rods withdraw or operators dilute the reactor coolant (adding positive reactivity and increasing reactor coolant temperature).

This mode of control produces acceptable steam conditions at the main turbine inlet at 100% power, while requiring a lower T_h than does the constant steam pressure control mode. Most Westinghouse designed nuclear units use some form of a sliding T_{avg} control program. One difference in the various control programs is the value of the programmed range of temperature control. The programmed temperature range varies from 12°F to the most often used value of 30°F. Another difference is in the value of the no-load setpoint (the programmed T_{avg} at 0% power). The most commonly used values of no-load T_{avg} are 547°F and 557°F.

1.2.8 Summary

Pressurized water reactor units use a dual cycle concept, in which the closed primary cycle is separate from the secondary cycle. The point of heat transfer between the two cycles is the steam generator(s). The RCS is the primary cycle and is located inside the containment building. The secondary cycle includes the steam system, the turbine generator (where steam energy is used to generate electric power), and the condensate and feedwater systems. Secondary cycle systems, subsystems, and components are principally located in the turbine building.

Support and emergency systems serve many purposes affecting both the primary and the secondary. Systems, components, structures, and buildings which have safety functions or are required to maintain the integrity of the RCS or the reactor core, under accident conditions, must be built to Seismic Category I standards. Safety systems, components, and structures built to Seismic Category I specifications will provide their intended safety functions during the maximum credible seismic event.

The constant T_{avg} and constant steam pressure reactor control modes are viable control modes but are not generally used at Westinghouse designed plants. The control scheme most often selected to control the reactor is the sliding T_{avg} control mode. This mode of control programs T_{avg} to increase as secondary load increases. The sliding T_{avg} mode is a compromise between the other two modes of control and contains some of the advantages and disadvantages of each.

1101-329
1.2-1; 1-2

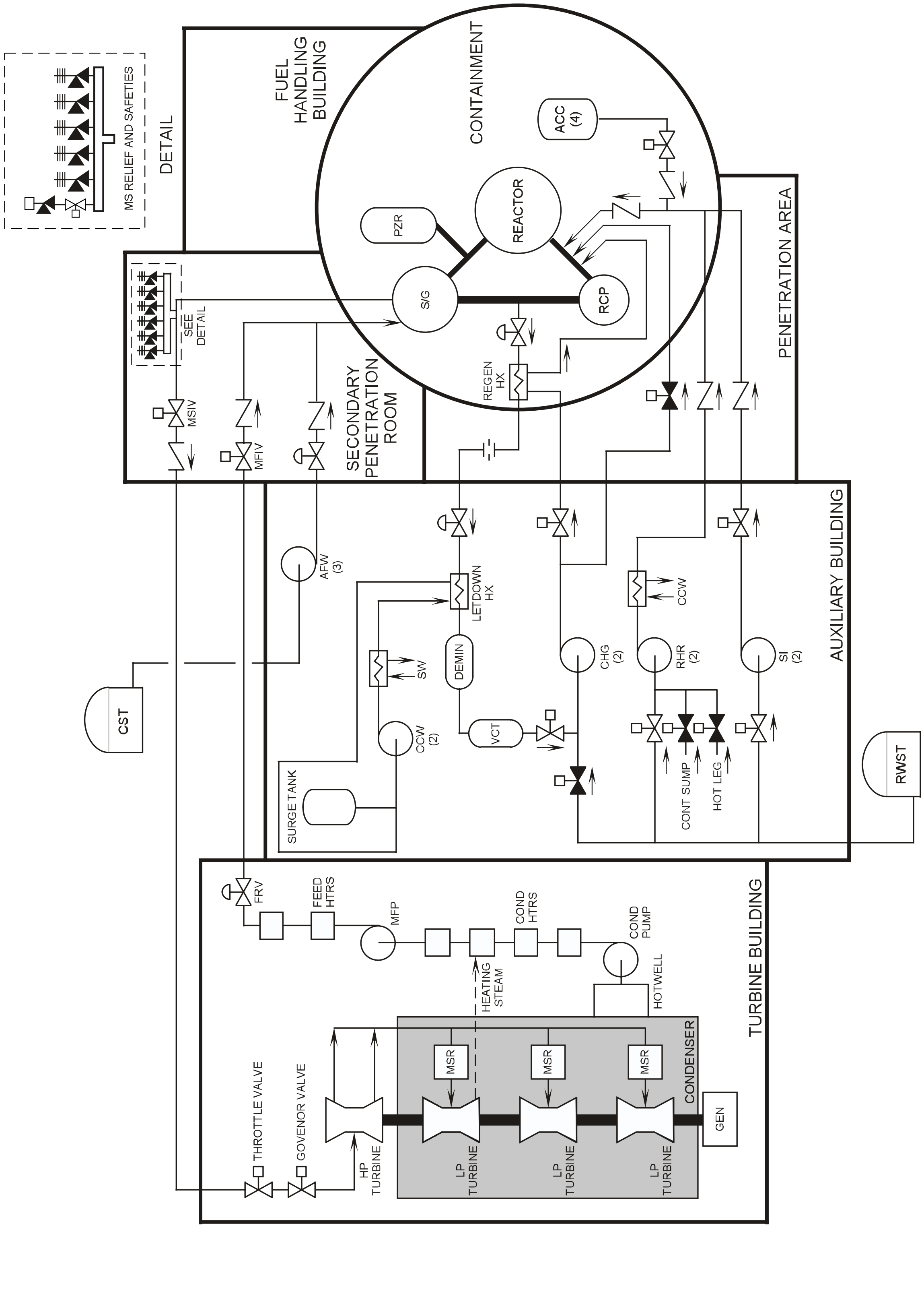


Figure 1.2-1 Plant Systems Composite

0199-3

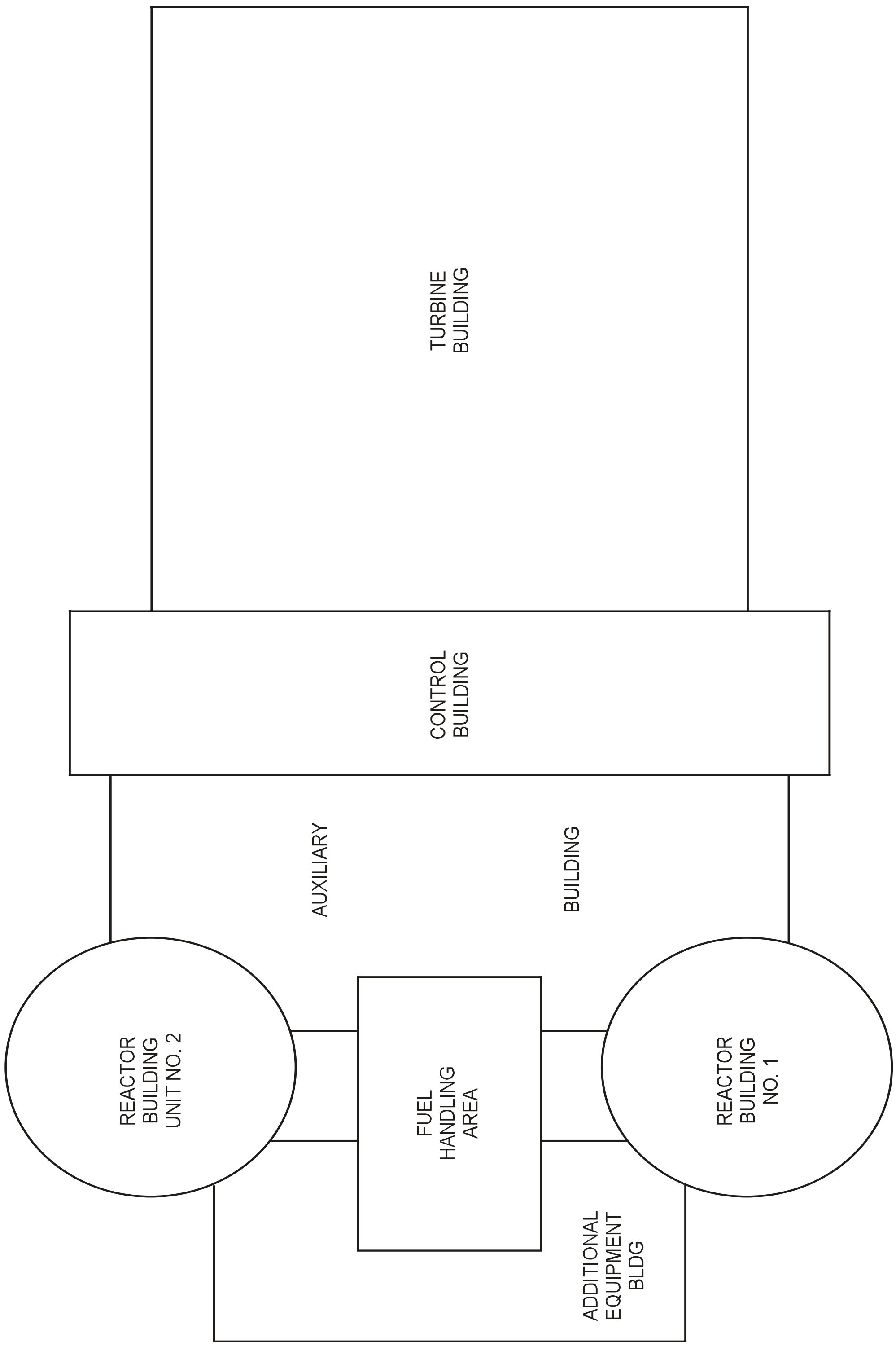


Figure 1.2-2 Plant Layout

0599-3

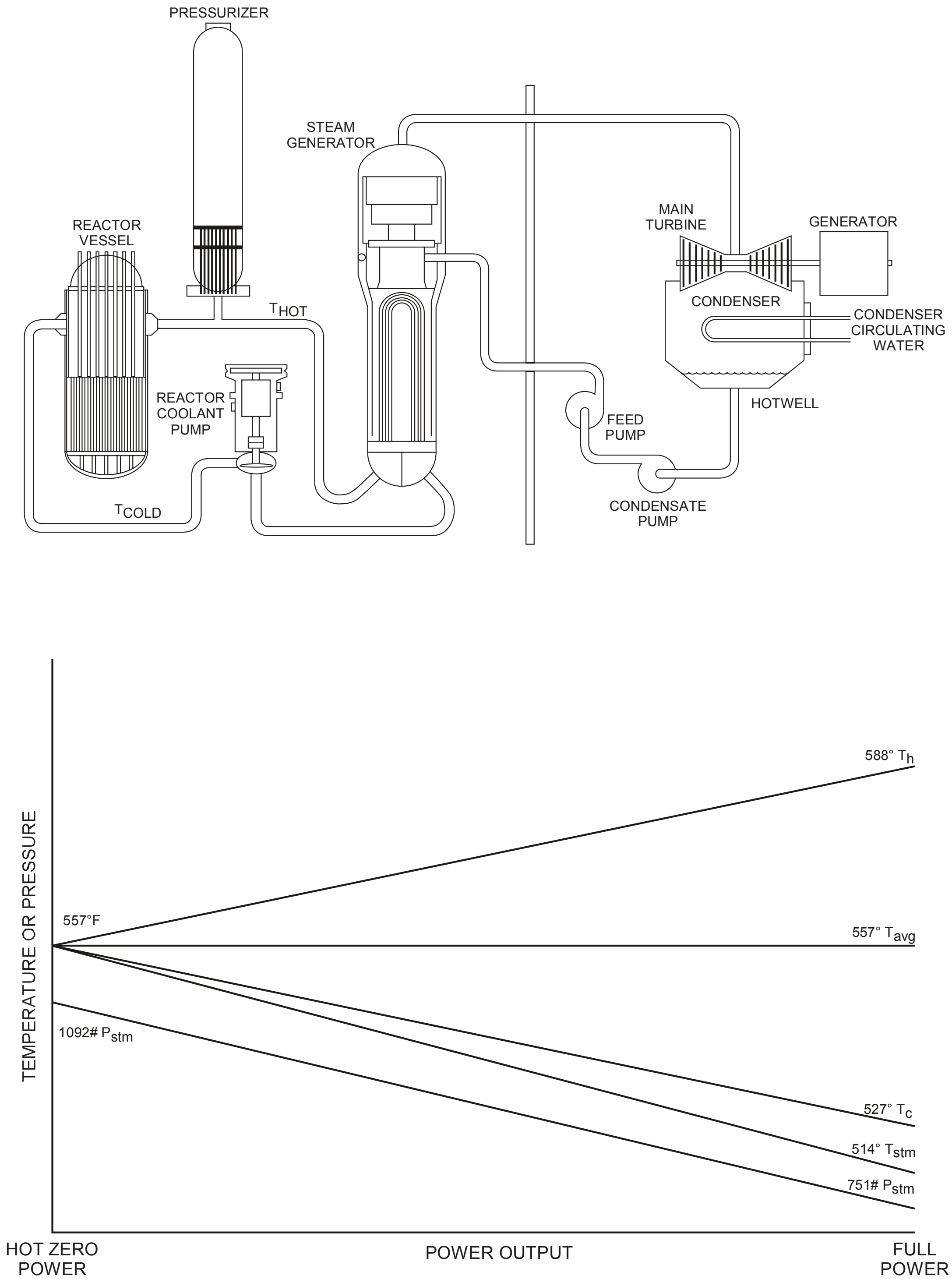
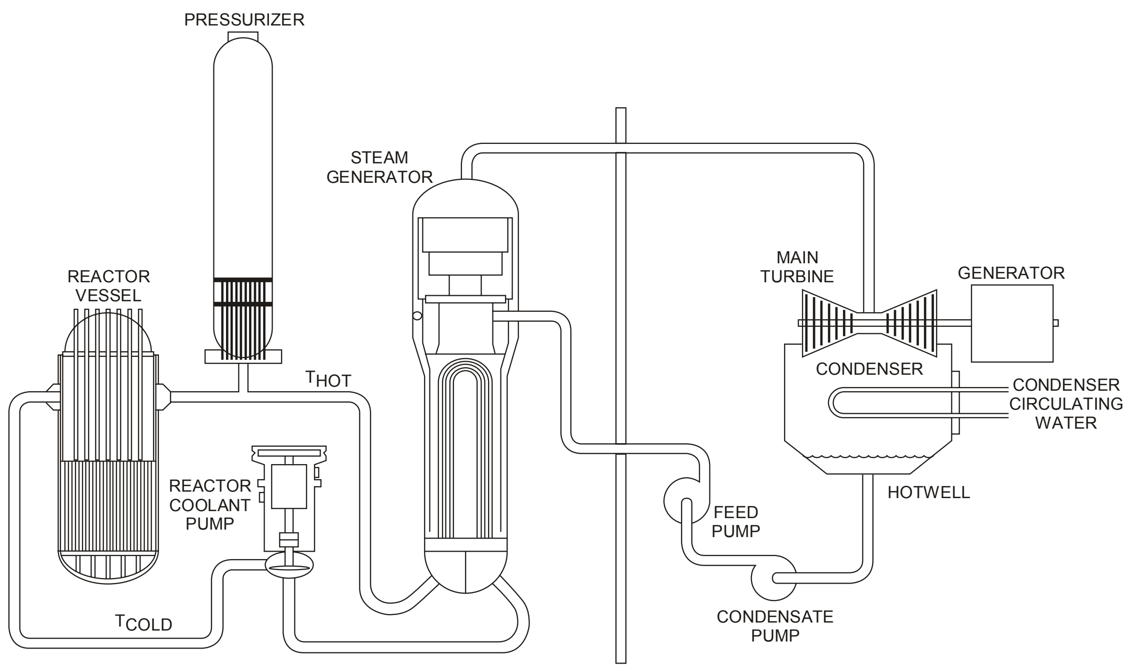


Figure 1.2-3 Characteristics of a Constant Average Temperature Program



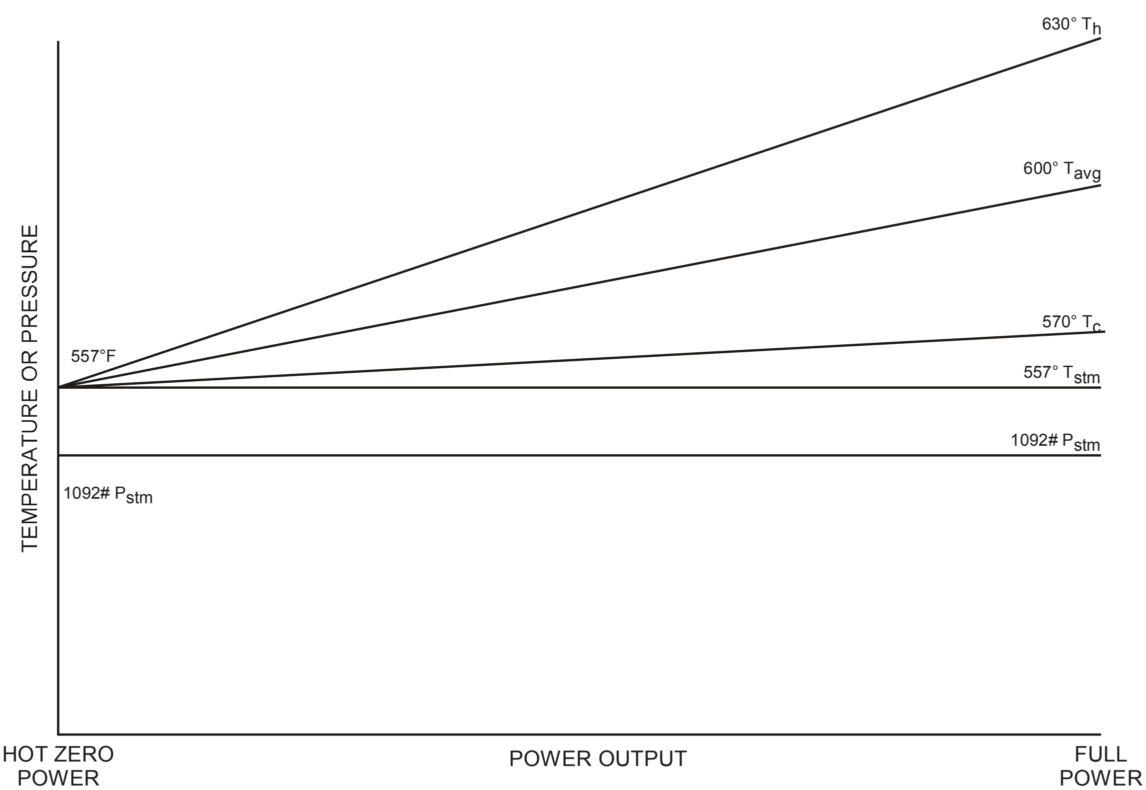
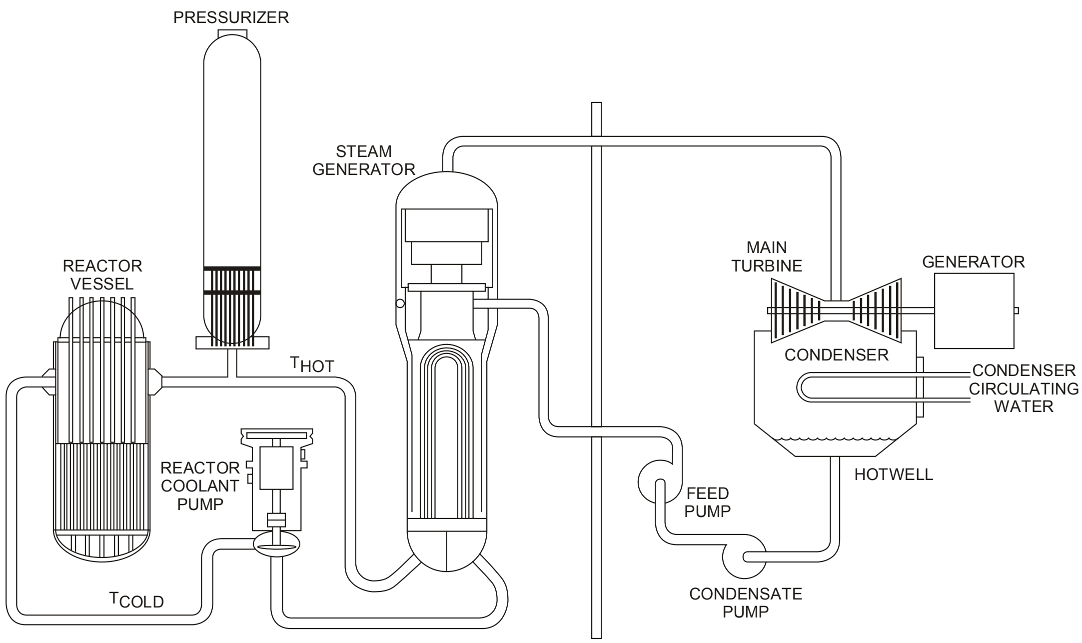


Figure 1.2-4 Characteristics of a Constant Steam Pressure Program

0199-3



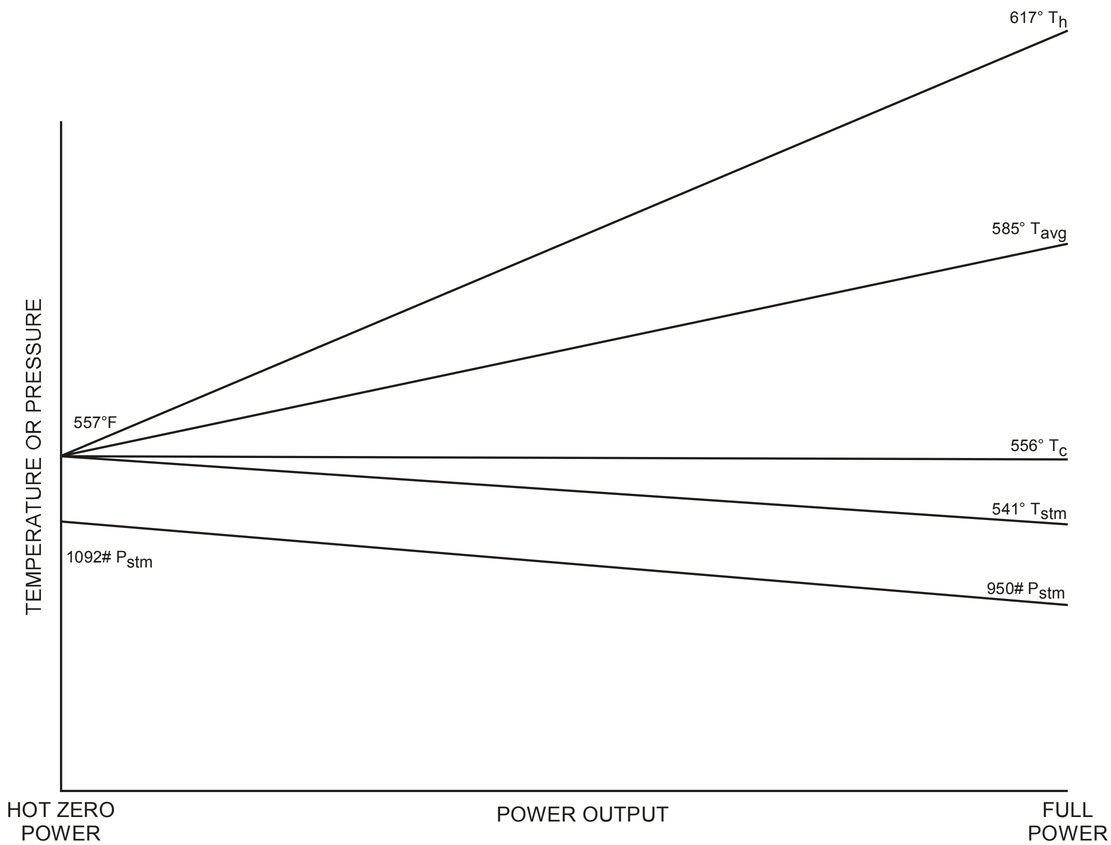


Figure 1.2-5 Characteristics of a Sliding Average Temperature Program