Nuclear Engineering Handbook

Chapter Two: Pressurized Water Reactor

2

Pressurized Water Reactors (PWRs)

Richard Schreiber
Westinghouse Electric Co. (Retired)

CONTENTS

2.1 Introduction 11
2.2 Overview 11
2.3 The Power Plant 11
2.4 Vendors 12
2.5 General Description of PWR Nuclear Power Plants Presently in Use 13
  2.5.1 Fuel 14
  2.5.2 Control 14
  2.5.3 Burnable Poison (BP) 15
  2.5.4 Coolant Pumps 17
  2.5.5 Steam Generation 18
  2.5.6 Pressurizer 18
2.6 Operations 20
2.7 Detailed Description of Present Systems 24
  2.7.1 Primary Loop 24
  2.7.2 Secondary Loop 24
  2.7.3 Tertiary Loop 24
  2.7.4 Confinement of Radioactivity 25
2.8 Component Design 25
  2.8.1 Fuel Assembly 25
  2.8.2 Grid Assemblies 25
  2.8.3 Other Features of Assemblies 26
  2.8.4 Control Rods 26
  2.8.5 Enrichment 27
  2.8.6 Startup 27
  2.8.7 Construction Materials 28
2.9 Auxillary Systems 29
  2.9.1 Auxiliary Flows 29
  2.9.2 Water Sources 30
  2.9.3 BTRS 31
  2.9.4 Residual Heat Removal System (RHRS) 33
  2.9.5 BRS 35
  2.9.6 Steam Generator Blowdown Processing System (SGBPS) 36

2.10 Engineered Safeguards Systems....36
2.10.1 SIS....37
2.10.2 High-Pressure Injection....38
2.10.3 System Safeguards....39
2.10.4 SIS Components....40
2.10.5 Cold Leg Recirculation Mode....41
2.10.6 Emergency Feedwater for Secondary Loop....41
2.10.7 Component Cooling Water System (CCWS)....44
2.11 Containment Systems....47
2.11.1 DBA....48
2.11.2 Thermal Loads....48
2.11.3 Dead Loads....48
2.11.4 Live Loads....49
2.11.5 Earthquake Loads....49
2.11.6 Wind Forces....49
2.11.7 Hydrostatic Loads....49
2.11.8 External Pressure Load....49
2.11.9 Prestressing Loads....49
2.11.10 Containment Design Criteria....49
2.11.11 Design Method....50
2.11.12 Containment Liner Criteria....50
2.11.13 Equipment and Personnel Access Hatches....51
2.11.14 Special Penetrations....51
2.11.15 Containment Isolation System (CIS)....51
2.11.16 Containment Spray System (CSS)....52
2.11.17 Initial Injection Mode....53
2.11.18 RCFC System....53
2.11.19 Hydrogen Control in Containment....54
2.12 Instrumentation....54
2.13 Fuel Handling....55
2.13.1 Spent Fuel Handling....55
2.13.2 New Fuel Handling....56
2.14 Waste Handling....56
2.14.1 Liquid Waste Processing....56
2.14.2 Gaseous Waste Processing....57
2.14.3 Solid Waste Processing....57
2.14.4 Radwaste Volume Reduction....58
2.15 Advanced Passive Reactor....58
2.15.1 New PWR Designs....58
2.15.2 Chemical Control of the Coolant System....64
2.15.3 RCP....65
2.15.4 Steam Generator....67
2.15.5 Reactor Coolant Pressurizer....72
2.15.6 ADS....72
2.15.7 RNS....75
2.16 PXS....76
2.17 Detection and Ignition of Hydrogen....77
2.18 IRWST....77

2.19 Safety Design Rationale for Venting the Reactor Vessel Head 79
2.20 Other Passive Emergency Systems 82
References 82

2.1 Introduction

In the 1960s, the U.S. Government, as well as other countries, promoted the development and application of nuclear energy for the production of electric power. The employment of nuclear navies throughout the world provided a knowledge base for the type of reactor using high-pressure "light" water as coolant and moderator. The fuel selected for domestic power stations was uranium dioxide in pellet form, slightly enriched in the isotope U-235, and protected from the coolant by stainless steel or a modified zirconium–tin alloy that came to be known as "Zircaloy." Zircaloy-4 has been the tubular cladding material of choice today because of its corrosion resistance when preoxidized, and its low absorptive "cross-section" for neutrons. In the present century, PWRs are the most popular design, providing nearly two-thirds of the installed nuclear capacity throughout the world.

2.2 Overview

For general discussion purposes, a nuclear power plant can be considered to be made-up of two major areas: a nuclear "island" and a turbine island composed of a turbine/generator (T-G). Only the former is being described in detail in this chapter. To a large extent, the design of the non-nuclear portion of a Rankine cycle power plant depends only on the steam conditions of temperature, pressure, steam "quality" (how little liquid is present with the vapor), and flow arriving at the turbine, regardless of the heat source. There are safety systems in the non-nuclear part of a nuclear plant that are unique, such as a diesel generator for emergency power. All essential nuclear systems are discussed below.

2.3 The Power Plant

For PWRs, the part of the coolant system (primary loop, Figure 2.1) that contains radioactivity is surrounded by a sturdy containment structure whose main purpose is to protect operating personnel and the public. Various auxiliary and safety systems attached to the primary are also located within the containment. This protected array of equipment we call the nuclear island is also called the "Nuclear Steam Supply System" (NSSS). The NSSS and the balance-of-plant (including the T-G and all other systems) are composed of fluid, electrical, instrumentation, and control systems; electrical and mechanical components; and the buildings or structures housing them. There are also several shared fluid, electrical, instrumentation and control systems, as well as other areas of interconnection or interface. The principal operating data for current Westinghouse NSSS models are listed in Table 2.1.

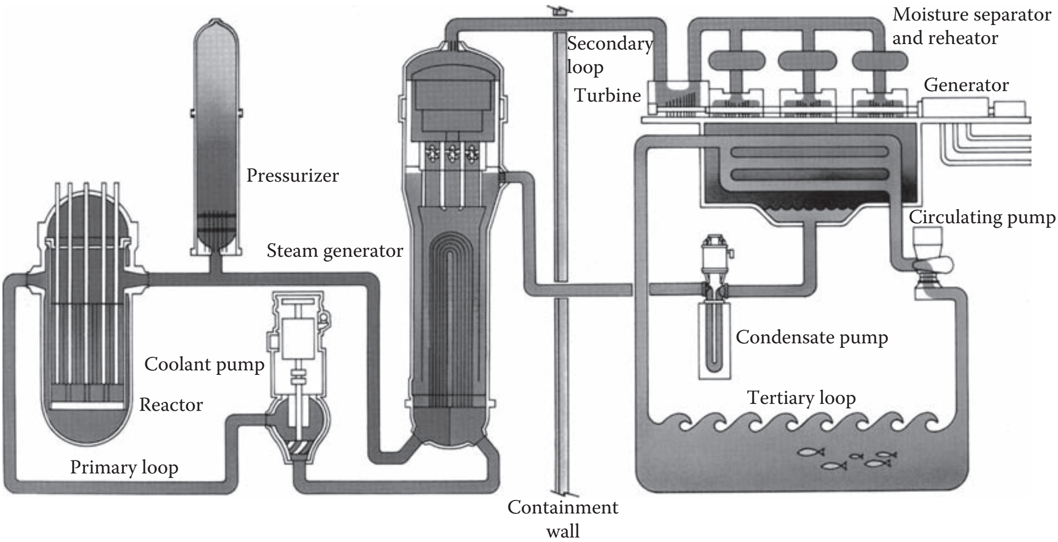


FIGURE 2.1
Nuclear steam supply system (schematic).

TABLE 2.1
Principal Data for Current Westinghouse NSSS Models

Model:	**212**	**312**	**412**	**414**
Number of loops:	**2**	**3**	**4**	**4**
NSSS power, megawatt thermal	1882	2785	3425	3819
Approximate electrical output, Mwe	600	900	1150	1280
Steam pressure, psia (bar)	920 (63)	960 (66)	1000 (69)	1100 (76)
Reactor vessel ID, in. (cm)	132 (335.3)	157 (398.8)	173 (439.4)	173 (439.4)
Steam generator model	F	F	F	H
Reactor coolant pump type	93A1	93A1	93A1	93A1
RCP motor horsepower	7000	7000	7000	9000
Hot leg ID, in. (cm)	29 (73.7)	29 (73.7)	29 (73.7)	29 (73.7)
Cold leg ID, in. (cm)	27.5 (69.9)	27.5 (69.9)	27.5 (69.9)	27.5 (69.9)
Number of fuel assemblies	121	157	193	193
Fuel length, feet (cm)	12 (365.8)	12 (365.8)	12 (365.8)	14 (426.7)
Fuel assembly array	16×16	17×17	17×17	17×17

2.4 Vendors

In the United States, the principal suppliers of the present generation of NSSS were units of Babcock & Wilcox (B & W), Combustion Engineering (C-E), General Electric (boiling water reactors (BWRs)) and Westinghouse. These and several other organizations supply the fuel assemblies. Other consortiums have been formed throughout the world. In Europe, a group named AREVA has been organized. Since March 1, 2006, all first-tier subsidiaries of the AREVA group have new names. The trade name of COGEMA is now AREVA NC, Framatome ANP is now AREVA NP, and Technicatome is AREVA TA. This initiative also applies to second-tier subsidiaries and sites in France or abroad where "COGEMA" or "Framatome ANP" is part of the name. Japanese suppliers include Mitsubishi Heavy Industries (MHI) for PWRs, as well as local and international manufacturers for reactor equipment and fuel. In South Korea, PWR vessel and equipment suppliers include Doosan Heavy Industries/Construction and Korea Power Engineering. Fuel suppliers are Korea Nuclear Fuel and international suppliers. In Germany, Siemens is the major player, but they also have absorbed Exxon Nuclear in the United States by way of Kraftwerk Union (Germany). Siemens has also turned over their nuclear assets to a joint venture with Framatome ANP of France. The new company is to be called AREVA NP. Many other companies and consortia worldwide supply the nuclear power industry. MHI has aligned with AREVA to form a joint venture ATMEA to build nuclear plants, but MHI has also joined with Westinghouse on some bids and as a sole bidder in others. AREVA has absorbed the former B & W nuclear unit in Lynchburg, VA.

In the 1960s, C-E began selling commercial nuclear power steam supply systems, having cut their teeth on naval systems, just as many other firms had done. C-E was generally credited with a superior design to its competitors, evidenced by the fact that the megawatt yield of its nuclear reactors was typically about 10% higher than that of comparable PWRs. The basis for this increase in efficiency was a computer-based system called the Core Operating Limit Supervisory System (COLSS), which leveraged almost 300 in-core neutron detectors and a patented algorithm to allow higher power densities. In 1990, C-E became a subsidiary of Asea Brown Boveri (ABB), a Swiss–Swedish firm based in Zurich. In late December 1999, the British firm British Nuclear Fuels Limited (BNFL) agreed to purchase ABB's worldwide nuclear businesses, including the nuclear facilities of C-E. In March 1999, BNFL had acquired the nuclear power businesses of Westinghouse Electric Company with the remaining parts of Westinghouse going to Morrison Knudson (MK) Corporation. In late 2006, Toshiba completed its acquisition of those nuclear units from BNFL, bringing C-E and Westinghouse design and manufacturing capabilities together. Westinghouse has also developed the ability to design and build BWRs and fuel. These rearrangements have taken place in the last 20 years while nuclear power dropped from the headlines. Expansion and development of new designs continues in the twenty-first century.

2.5 General Description of PWR Nuclear Power Plants Presently in Use

The central component of the Reactor Coolant System (RCS) is a heavy-walled reactor vessel that houses the nuclear core and its mechanical control rods, as well as necessary support and alignment structures. It is shown schematically in Figure 2.1, in relation to other parts of the system in Figure 2.2, and as a cut-away showing the internal details in Figure 2.3. The vessel is cylindrical in shape with a hemispherical bottom head and a flanged and gasketed upper head for access. It is fabricated of carbon steel, but all wetted surfaces are clad with stainless steel to limit corrosion. The internal core support and alignment structures are removable to facilitate inspection and maintenance, as is the alignment structure for the top-mounted control rod drive mechanisms. Vessel inlet and outlet nozzles for the primary loops are located at a level well above the top of the fuel core.

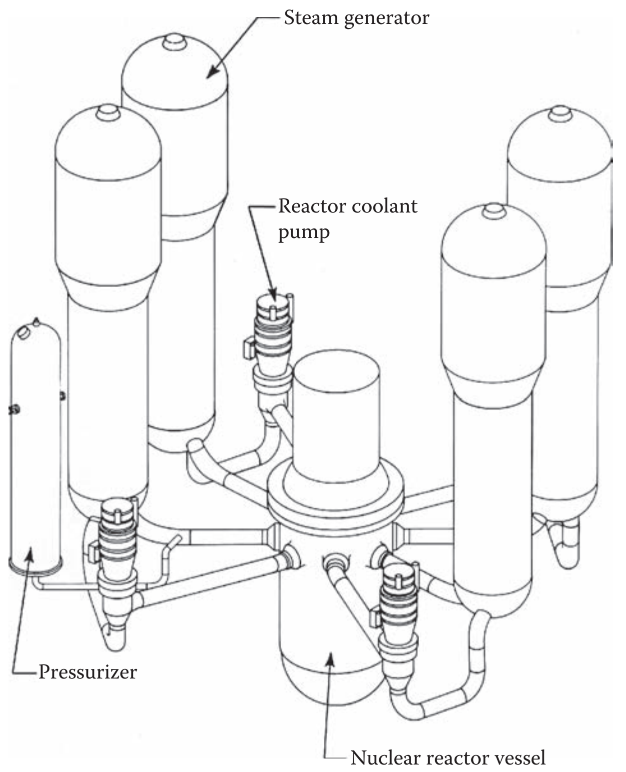


FIGURE 2.2
Layout of nuclear island.

2.5.1 Fuel

The nuclear core comprises several fuel assemblies arranged in three regions to optimize fuel performance. All fuel assemblies are mechanically identical, but enrichment of the uranium dioxide fuel differs from assembly to assembly. In a typical initial core loading, three fuel enrichments are used. Fuel assemblies with the highest enrichments are placed in the core periphery, or outer region, and the groups of lower enrichment fuel assemblies are arranged in a selected pattern in the central region. In subsequent refuelings, one-third of the fuel (the highest "burnup") is discharged and fresh fuel is loaded into the outer region of the core. The remaining fuel is rearranged in the central two-thirds of the core as to achieve optimal power distribution and fuel utilization. Figure 2.4 shows the details of the PWR fuel assembly. Figure 2.5 shows how they are distributed by enrichment within the core. Table 2.2 gives fuel rod design details. Further details regarding nuclear fuel are given elsewhere in this handbook.

2.5.2 Control

Rod cluster control (RCC) assemblies used for reactor control consist of absorber rods attached to a spider connector which, in turn, is connected to a drive shaft. The absorber (control) rods are loaded with a material that has a high affinity "cross section" for neutrons. Above the core, control rods move within guide tubes that maintain alignment of

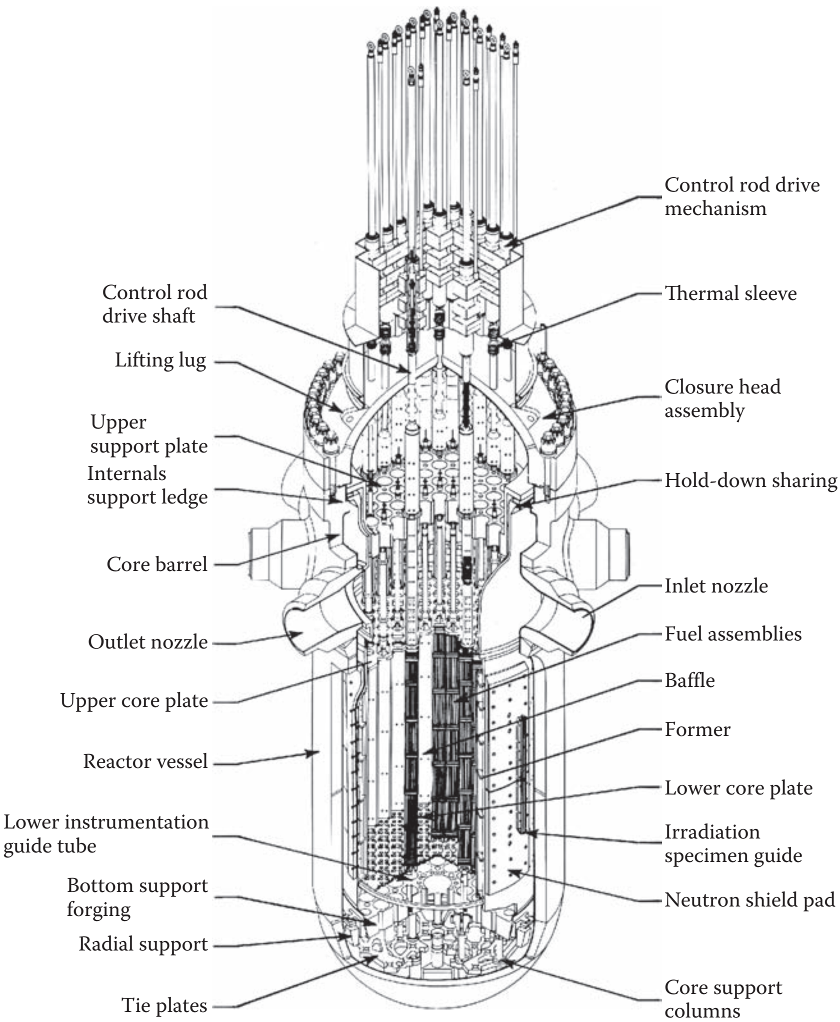


FIGURE 2.3
Cut-away of reactor vessel.

the control rods with empty thimbles of certain fuel assemblies at particular locations in the core. RCC assemblies are raised and lowered by a drive mechanism on the reactor vessel head. The drive mechanism allows the RCC assemblies to be released instantly, "trip," when necessary for rapid reactor shutdown. Insertion of the assemblies during a trip is by gravity. Figure 2.6 shows the relationship of the fuel assembly and the RCC arrangement within the core. The intent is to equalize ("flatten") the power distribution across the core as much as possible.

2.5.3 Burnable Poison (BP)

In addition to control rods, there is a distribution of absorber (BP) rods that are mounted on RCC-like fixtures, but are not connected to drive mechanisms. The BP rods remain in the core during operation, but may be moved to new locations during shutdown.

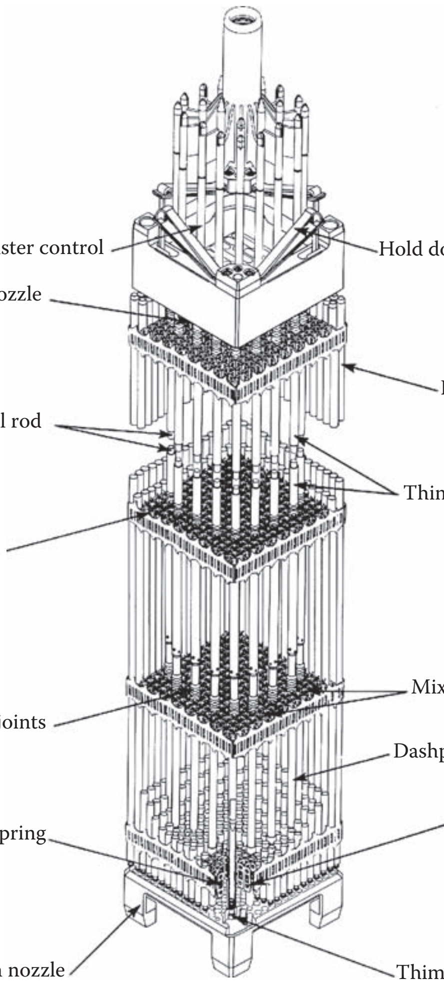


FIGURE 2.4
Typical fuel assembly for the present generation of reactors.

Figure 2.7 shows their distribution in a typical large core. Their intent is to suppress the large excess of nuclear reactivity during the early part of the cycle, using up the absorber during operation. They also allow a lower concentration of soluble boron poison during operation. There is a small burnup penalty (Figure 2.8). The configuration of each BP assembly is similar in appearance to an RCC assembly with the exception of the handling fitting. Positions in the cluster not occupied by BP rods contain loose-fitting plugs that balance the coolant flow across the host fuel assembly. The plugs are also connected to the fixture. The fuel assemblies that contain neither control rods (including safety rods) nor BPs, nor neutron startup sources, contain "pluggers." Pluggers are all flow-balancing plugs mounted on a fixture for support and handling. Special handling tools are needed for each of these inserts into a fuel assembly because they all become "hot" in use, but must be switched between assemblies. The long dangling rods are kept from splaying by the use of "combs" that keep them properly oriented for reinsertion. All of these manipulations are done deep underwater.

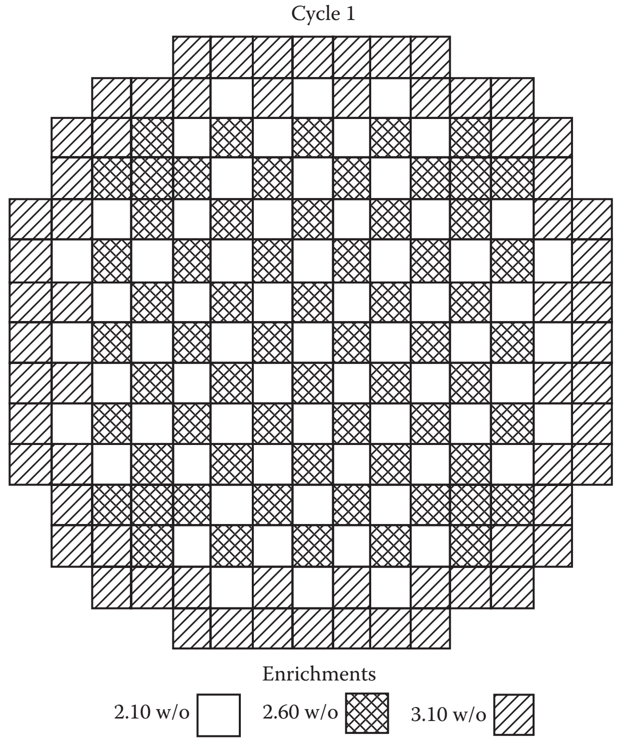


FIGURE 2.5
Pattern of initial fuel load, three regions.

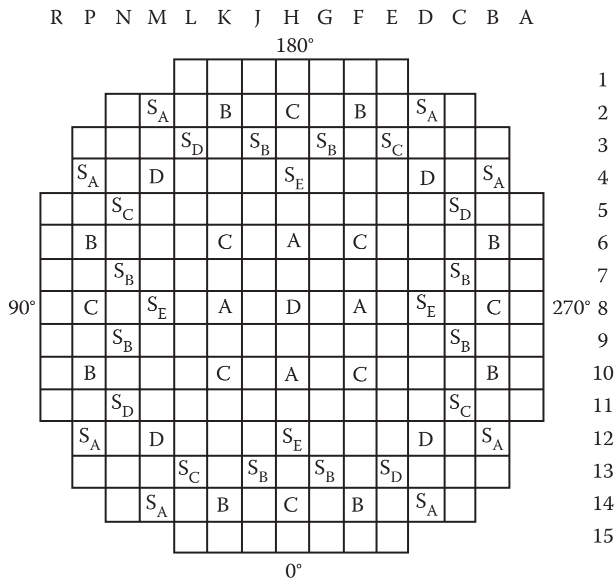
TABLE 2.2

Fuel Rod Parameters (Four-Loop Plant)

Fuel rod length	12 ft (365.8 cm)
Outside diameter	0.360 in. (0.914 cm)
Cladding thickness	0.0225 in. (0.0572 cm)
Cladding material	Zircaloy-4
Diametral gap	0.0062 in. (0.0157 cm)
Pellet diameter	0.3088 in. (0.7844 cm)
Lattice pitch	0.496 in. (1.260 cm)
Rods array in assembly	17×17
Rods in assembly	264
Total number of fuel rods in core	50,952

2.5.4 Coolant Pumps

Reactor coolant pumps (Figure 2.9) are vertical, single-stage, mixed flow pumps of the shaft-seal type. A heavy flywheel on the pump motor shaft provides long coastdown times to preclude rapid decreases in core cooling flow during pump trips. Interlocks and automatic reactor trips ensure that forced cooling water flow is present whenever the reactor is at power. Additionally, two separate power supplies are available to the pump motor when the plant is at power.



Control bank	Number of rods
A	4
B	8
C	8
D	5
Total	25

Shutdown bank	Number of rods
S_A	8
S_B	8
S_C	4
S_D	4
S_E	4
Total	28

FIGURE 2.6
Arrangement of control rod banks in the reactor core.

2.5.5 Steam Generation

Steam generators are of a vertical U-tube design with an expanded upper section that houses integral moisture separation equipment to produce steam with a quality of at least 99.75% (Figure 2.10). Table 2.3 lists many design parameters. Preheated feedwater enters the top of the unit, mixes with effluent from the moisture separators and then flows downward on the outside of the tube bundle. The feed is distributed across the bundle and then flows upward along side the heated tubes. An alternate design used by another vendor (B & W) has a bundle of straight tubes. Water in the secondary loop is boiled in the lower section of the steam generator, dried to all steam in the middle section and superheated in the upper section, obviating the need for moisture separators before passing the dry steam to the turbines. Reactor coolant piping, the reactor internals, and all of the pressure-containing and heat transfer surfaces in contact with reactor water are stainless steel or stainless steel clad, except the steam generator tubes and fuel tubes, which are Inconel and Zircaloy, respectively.

2.5.6 Pressurizer

An electrically heated pressurizer connected to one of the reactor coolant hot legs maintains RCS pressure during normal operation, limits pressure variations during plant load transients, and keeps system pressure within design limits during abnormal conditions.

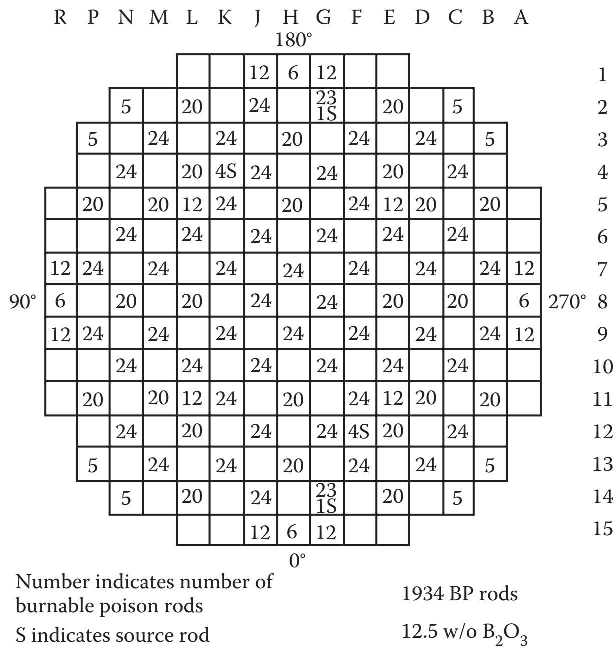


FIGURE 2.7
Arrangement of burnable poison rods, initial core loading.

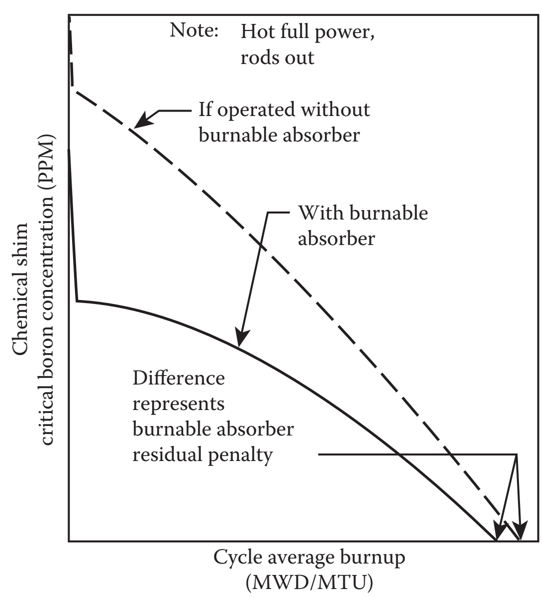


FIGURE 2.8
Effect of burnable poison rods on soluble poison requirements.

A typical design is given in cut-away pictorial in Figure 2.11. For example, a transient that could decrease system pressure is counteracted by flashing water within the pressurizer which is kept at saturation temperature by the automatic heaters. An increasing pressure transient is limited by spraying cooler water from the primary loop into the pressurizer

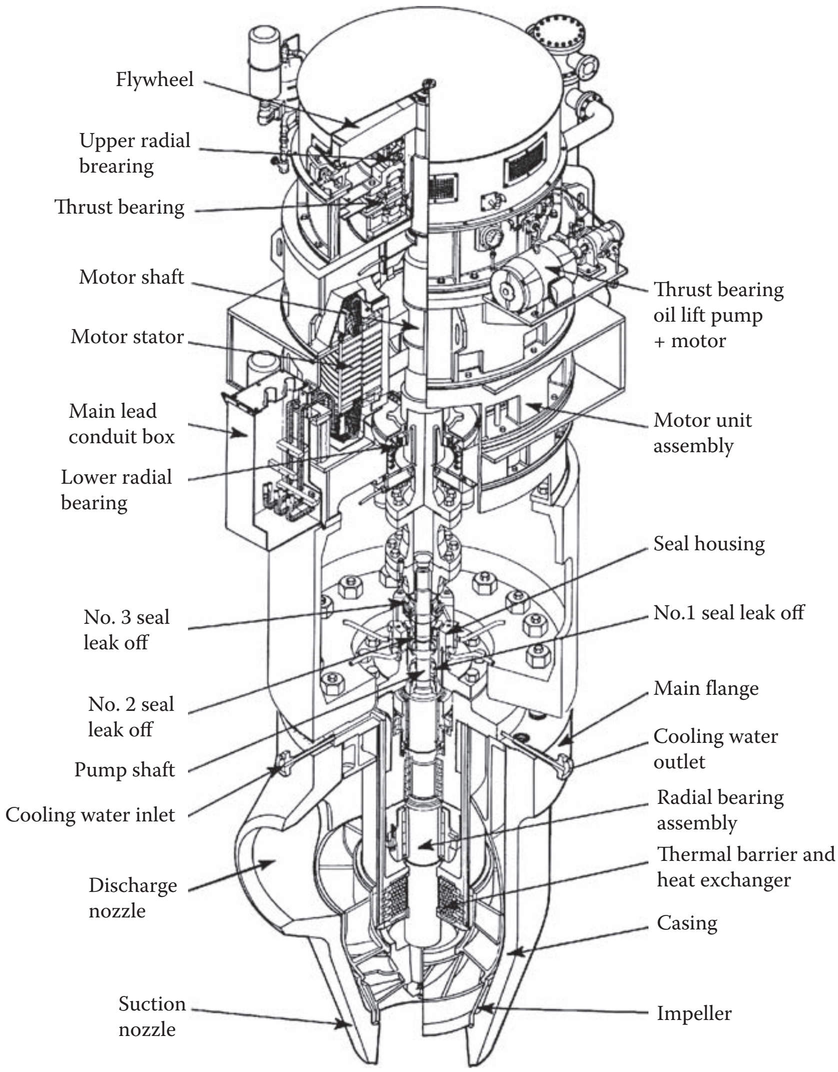


FIGURE 2.9
Cut-away of reactor coolant pump.

steam space to partially collapse the steam bubble, or by automatic operation of relief and safety valves.

2.6 Operations

Normal and emergency operation of the RCS requires several support functions to: maintain water inventory; purify and treat primary coolant; remove residual heat after a plant shutdown; provide cooling water to pumps and motors; supply ventilation air;

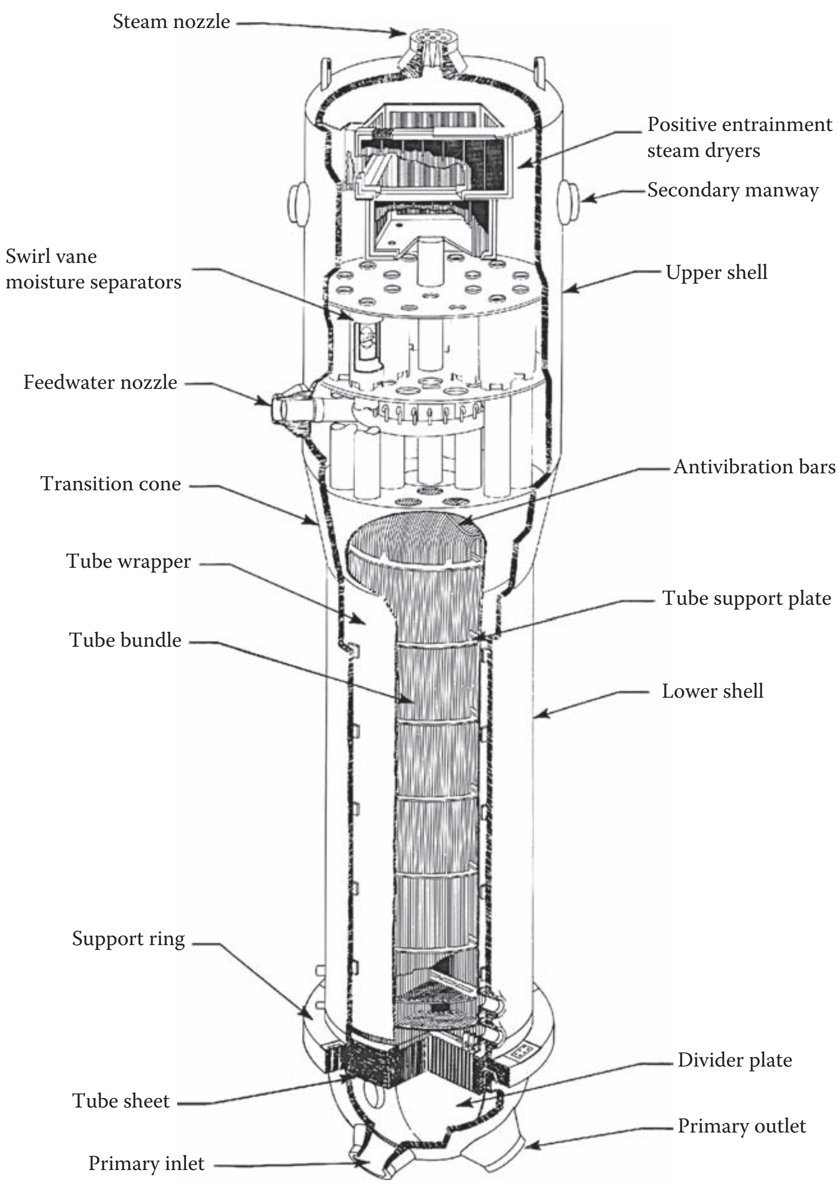


FIGURE 2.10
Cut-away of steam generator.

and provide emergency supplies of core cooling water. These functions are provided by auxiliary systems described later in this section.

Transient power behavior of a nuclear core is determined by a condition known as "reactivity." For a core operating at a steady power level, the various factors that affect reactivity are balanced so that the net reactivity is zero. If the net reactivity is positive, power level will increase and, conversely, decrease if reactivity is negative. Power control of a PWR is based on balancing reactivity through the use of mechanical and

TABLE 2.3

Steam Generator Principal Design Data

Number and Type	One Vertical, U-tube Steam Generator with Integral Steam-Drum Per Loop
Height overall	67 ft, 8 in. (20.6 m)
Upper shell OD	14 ft, 7-3/4 in. (4.5 m)
Lower shell OD	11 ft, 3 in. (3/4 m)
Operating pressure, tube side	2250 psia (155 bar)
Design pressure, tube side	2500 psia (172 bar)
Design temperature, tube side	650 °F (343 °C)
Full load pressure, shell side	
Two-Loop Plant	920 psia (63 bar)
Three-Loop Plant	964 psia (66 bar)
Four-Loop Plant	1000 psia (69 bar)
Steam flow per steam generator	3,813,000 lb/hr (480 kg/sec)
Maximum moisture at outlet (full load)	0.25%
Design pressure, shell side	1200 psia (82.7 bar)
Reactor coolant flow rate	35,075,000 lb/hr (4419 kg/sec)
Reactor coolant inlet temperature	621°F (327°C)
Reactor coolant outlet temperature	558°F (292°C)
Shell material	Mn–Mo steel
Channel head material	Carbon steel clad internally with stainless steel
Tube sheet material	Ni–Mo–Cr–V clad Inconel on primary face
Tube material	Thermally treated Inconel
Steam generator weights	
Dry weight, in place	346 tons (314,000 kg)
Normal operating weight, in place	422 tons (384,000 kg)
Flooded weight (cold)	560 tons (508,000 kg)

chemical neutron absorbers and appropriate allowance for physical phenomena that influence reactivity.

The principal natural phenomena that influence transient operation are the temperature coefficients of the moderator and fuel and the buildup or depletion of certain fission products. Reactivity balancing may occur through the effects of natural phenomena or the operation of the reactor control system using the RCCs or chemical “shim.” A change in the temperature of moderator or fuel (e.g., due to an increase or decrease in steam demand) will add or remove reactivity (respectively) and cause the power level to change (increase or decrease, respectively) until the reactivity change is balanced out. RCC assemblies are used to follow fairly large load transients, such as load-follow operation, and for startup and shutdown.

The chemical shim system uses the soluble neutron absorber boron (in the form of boric acid), which is inserted in the reactor coolant during cold shutdown, partially removed at startup, and adjusted in concentration during core lifetime to compensate for such effects

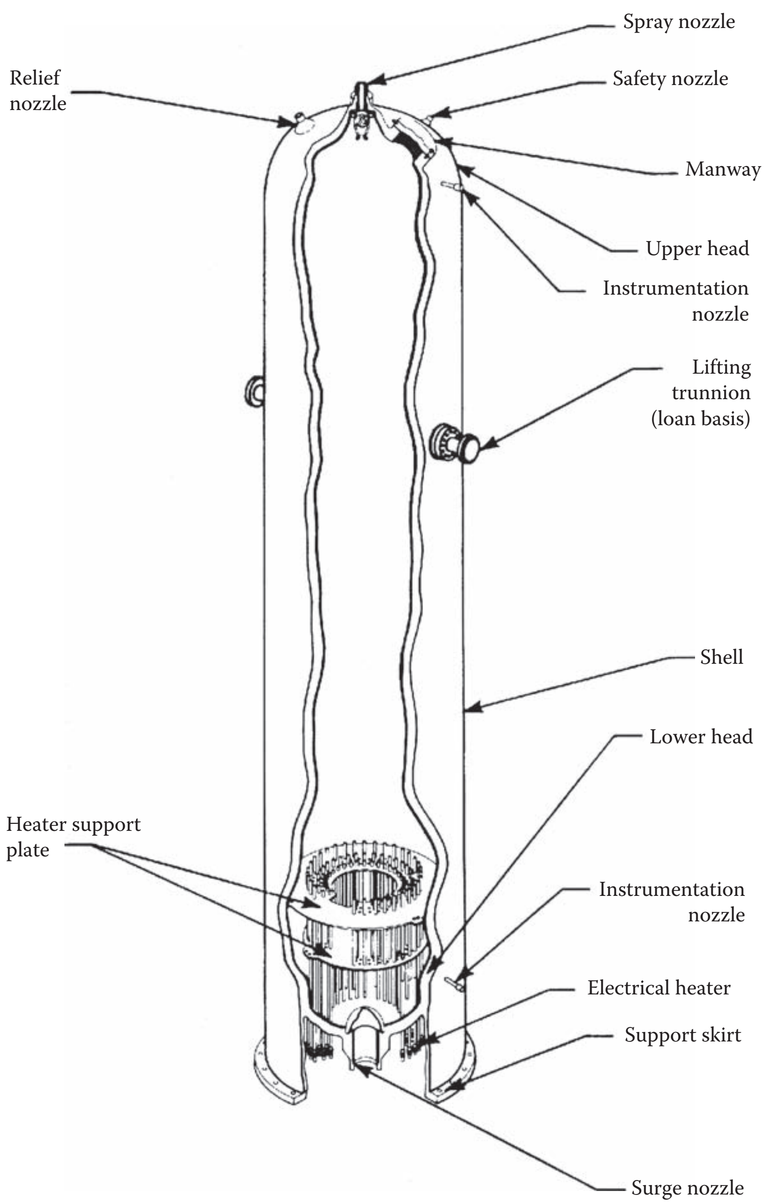


FIGURE 2.11
Primary system pressurizer.

as fuel consumption and accumulation of fission products which tend to slow the nuclear chain reaction. The control system allows the plant to accept step load increases of 10% and ramp load increases of 5% per minute over the load range of 15–100% of full power, subject to "xenon" limitations near the end of core life. Equal step and ramp load reductions are possible over the range of 15–100% of full power. Losses of reactor load up to 100% of rated power without a reactor trip can be accommodated by steam dump to the condenser in conjunction with the control system. Complete supervision of the nuclear and turbine-generator islands is accomplished from a single plant control room.

2.7 Detailed Description of Present Systems

This section describes the basic design and operating characteristics of a Westinghouse PWR plant. Differences from other manufacturers are mentioned if appropriate. Keep in mind that B & W and C-E made many of the domestic reactor vessels used by Westinghouse. The design is available in several ratings from approximately 600 megawatts electrical (MWe) to 1200 MWe. Exact ratings depend on several specific constraints (e.g., heat sink characteristics, customer system needs). The different ratings are attained through use of two, three, or four reactor coolant piping loops, each loop comprising a steam generator, reactor coolant pump, and interconnecting piping. The loops are each connected to a reactor vessel sized to contain nuclear cores composed of fuel elements of 12- or 14-foot length from 121–193 assemblies. The construction of the groups of fuel-bearing tubes ("rods") into assemblies will be described later. The objective of NSSS design is to satisfy the full range of utility requirements while maximizing the use of standard components. The principal parameters for the various power ratings are given in Table 2.1. The description given in this section is based on a four-loop plant with a 12-foot core having an electrical capacity of 1100 MWe. Refer to Figure 2.1 for the NSSS segment descriptions described below.

2.7.1 Primary Loop

The primary loop contains the heat source consisting of a nuclear fuel core positioned within a reactor vessel where the energy resulting from the controlled fission reaction is transformed into sensible heat in the coolant/moderator. The coolant is pumped to the steam generator where the heat is transferred to a secondary loop through several U-type tubes. The reactor coolant then returns back to the reactor vessel to continue the process. An electrically heated pressurizer connected to the loop maintains a pressure above the saturation pressure so that bulk boiling does not occur. See Figure 2.2 for the layout of the RCS. Figure 2.3 is a cut-away of the reactor vessel. Figure 2.10 gives details of the steam generator. Figure 2.9 shows the reactor coolant pump (RCP), and Figure 2.11 is the pressurizer. In the RCS design of plants by C-E, two of the exit loops from the reactor vessel join to feed one steam generator. In a nominal "four-loop" plant, there are two such very large steam generators instead of four.

2.7.2 Secondary Loop

The secondary loop is the heat utilization circuit where dry steam produced in the steam generator flows to a turbine-generator where it is expanded to convert thermal energy into mechanical energy and hence electrical energy. The expanded steam exhausts to a condenser where the latent heat of vaporization is transferred to the cooling system and is condensed. The condensate is pumped back to the steam generator to continue the cycle.

2.7.3 Tertiary Loop

The tertiary loop is the heat rejection loop where the latent heat of vaporization is rejected to the environment through the condenser cooling water. Depending on the specific site, this heat is released to a river, lake, ocean, or cooling tower system. The latter is becoming the more common within the United States in part because of increasingly stringent environmental rules to reduce the thermal impact on natural water bodies.

2.7.4 Confinement of Radioactivity

Use of a steam generator to separate the primary loop from the secondary loop largely confines the radioactive materials to a single building during normal power operation and eliminates the extensive turbine maintenance problems that would result from radioactively contaminated steam. Radioactivity sources are the activation products from the small amount of corrosion that is present in the primary loop over the 12–18-month reactor cycle, as well as from the occasional (<1 in 10,000) fuel rod that develops a crack and releases a small portion of its volatile fission products. Uranium dioxide fuel is very resistant to erosion by the coolant, so the rod does not dump its entire fission product inventory into the RCS.

2.8 Component Design

Table 2.2 gives the essential parameters of the fuel rods and assemblies.

2.8.1 Fuel Assembly

A square array of fuel rods structurally bound together constitutes a fuel assembly. Control rod guide thimbles replace fuel rods at selected spaces in the array and are fastened to the top and bottom nozzles of the assembly. Spring clip grid assemblies are fastened to the guide thimbles along the height of the fuel assembly to provide support for the fuel rods. The fuel rods are contained and supported, and the rod-to-rod centerline spacing is maintained within this skeletal framework. Figure 2.4 is a cut-away of the assembly showing the structural details. The bottom nozzle of the fuel assembly controls the coolant flow distribution and also serves as the bottom structural element. The top nozzle functions as the fuel assembly upper structural element and forms a plenum space where the heated reactor coolant is mixed and directed toward the flow holes in the upper core plate. The spring clip grids provide support for the fuel rods in two perpendicular directions. Each rod is supported at six points in each cell of the grid. Four support points are fixed: two on one side of the grid strap, and two similarly located on the adjacent side. Two more support points are provided by spring straps located opposite the fixed points. Each spring strap exerts a force on the fuel rod such that lateral fuel rod vibration is restrained, but small thermal expansions are allowed. Because the fuel rods are not physically bound to the support points, they are free to expand axially to accommodate thermal expansion and radiation-induced growth of the highly textured crystal structure of the Zircaloy cladding.

2.8.2 Grid Assemblies

Two types of grid assemblies are employed. One type features mixing vanes that project from the edges of the straps into the coolant stream to promote mixing of the coolant in the high heat region of the fuel assemblies. The other, a nonmixing type of grid (to minimize flow pressure losses), is located at the bottom and top ends of the assembly where mixing for heat transfer purposes is not needed. The outside straps on all grids contain vanes which aid in guiding the grids and fuel assemblies past projecting surfaces during fuel handling or while loading and unloading the core. All fuel assemblies employ the same basic mechanical design.

2.8.3 Other Features of Assemblies

All assemblies can accept control rod clusters (the term "control rod cluster" is also referred to as "RCC" in the literature) but these are not used at every core location. Selected fuel assemblies have neutron sources or burnable absorber rods installed in the control rod guide thimbles. Fuel assemblies not containing RCCs, source assemblies, or burnable absorber rods, are fitted with plugs in the upper nozzle to restrict the flow through the vacant control rod guide thimbles. This plug includes an end-flow mixing device to assure that these fuel assemblies have approximately the same coolant flow as those containing RCCs.

The fuel assembly design provides optimum core performance by minimizing neutron absorption in structural materials and maximizing heat-transfer capabilities. Mixing vane grids increase the heat-transfer capability of the fuel rods. High fuel utilization is achieved by minimizing the parasitic absorption of neutrons in the core. In the assembly design, the only structural materials in the fuel region are the spring clip grids, Zircaloy control rod guide thimbles, and Zircaloy fuel cladding. Zircaloy is used because it absorbs relatively few neutrons and has good mechanical and heat-transfer properties.

Because all fuel vendors bid on reloads for each others' reactor core designs, as well as their own, all design features of each core must be public knowledge. Each reload constitutes one-third of the core, so it must be compatible neutronically and mechanically to fuel already in the core to avoid power generation distortions within the core and equipment mismatches. This applies to all types of reactors, not just PWRs.

2.8.4 Control Rods

RCC assemblies are used for reactor startup or shutdown, to follow load changes, and to control small transient changes in reactivity. The control elements of a RCC assembly consist of cylindrical neutron absorber rods (control rods), having approximately the same dimensions as a fuel rod and connected at the top by a spider-like bracket to form rod clusters. Details of the RCC system are shown in Figures 2.6 and 2.7. The control rods, which are stainless steel tubes encapsulating a hafnium absorber material, extend the full length of the core when fully inserted. Full-length RCCs provide operational reactivity control and can shut the reactor down at all times, even with the most reactive RCC stuck out of the core. Each RCC is coupled to its drive shaft, which is actuated by a separate magnetically actuated drive mechanism mounted on the reactor vessel head. RCCs are arranged into groups and electrically interconnected so that the entire group moves together. Reactivity of the core is changed by raising or lowering a group in the core. Each control rod of a RCC moves vertically in its own tubular guide thimble (an empty tube with a reduced diameter near the base, a plug at the bottom and holes in the smaller diameter section to allow some coolant in or out). Located symmetrically within fuel assemblies, these thimbles replace fuel rods within the fuel assembly lattice. All fuel assemblies are built the same because each could be in a control rod position, or any other position in the core. When an assembly is not in a RCC position, thimbles are plugged or contain BP rods (absorber that is used up during the reactor cycle). The thimbles: (1) act as guides for the control rods; and (2) serve as "dashpots" (shock absorbers) for slowing control rod motion during reactor trip. In their fully withdrawn position, control rods do not leave the upper end of the guide thimbles. This assures that the rods are always properly aligned, and reduces reactor coolant bypass through the thimbles. RCC design contributes to core performance improvement by providing a relatively homogeneous means of control. When control rods are withdrawn from the core, the resulting small water gaps do not cause significant power peaking.

2.8.5 Enrichment

Of the 193 fuel assemblies originally (no new reactors of this design have been built in many years) contained in a four-loop reactor first core, approximately 65 were of a low enrichment (2.10 weight percent uranium-235), 64 were an intermediate enrichment (2.60 w/o U-235), and 64 were a high enrichment (3.10 w/o U-235). Low and intermediate enrichment assemblies were arranged in a checkerboard pattern in the central portion of the core, whereas high enrichment assemblies were arranged about the periphery of the core (Figure 2.5). The first fuel cycle usually contained more excess reactivity than subsequent cycles due to the loading of all fresh (unburned) fuel. If soluble boron were the sole means of control, the concentration would be of the order of 1700 ppm, and the moderator temperature coefficient would be on the order of +7 pcm/°F (pcm stands for percent mille, i.e., 10^{-5}) [a 15°F (9.5°C) *increase* in moderator temperature creates 0.001 reactivity *increase* under these high boron conditions]. Because a large *positive* coefficient is undesirable, a reduction of the amount of control to be provided by chemical shim is accomplished by placing aluminum oxide–boron carbide burnable absorber material in the core. This material is depletable in the same fashion as uranium-235. Figure 2.7 shows the approximate BP locations in a four-loop core; Figure 2.8 shows the reduction in chemical shim brought about by this burnable absorber. As the fuel and BP deplete, the power shifts toward the center of the core, and this shift must be accounted for in design calculations. At the end of life, the power distribution is again quite uniform. Today, these older systems have achieved an optimal balance of BP and reload enrichment to achieve longer core life and much higher burnups than were initially believed possible.

2.8.6 Startup

Reactor startup neutron sources must be used to raise the neutron multiplication rate to levels detectable by the flux detectors outside the reactor, but still less than criticality. The fresh fuel configuration and the initially low core reactivity (from random fissions) by themselves would not permit a safely controlled startup. The source range as seen by the reactor instrumentation is shown in Figure 2.12. Neutron sources are of two types: (1) a primary source (which is active for initial reactor startup and startup early in the life of the first core) and (2) a secondary source (used for later startup of the reactor and which is activated during the operation of the reactor). The primary source is usually a spontaneously fissionable californium isotope. Secondary sources contain a mixture of antimony and beryllium (Sb–Be). The Sb becomes radioactive, emitting high-energy gamma particles that spall neutrons from the Be. The primary and secondary sources are similar to a control rod in mechanical construction. Both types of source rods are clad in stainless steel. The secondary source rods contain Sb–Be pellets which are not initially active. The primary source rods contain sealed capsules of source material at a specified axial position. Cladding encapsulation is completed by seal-welding

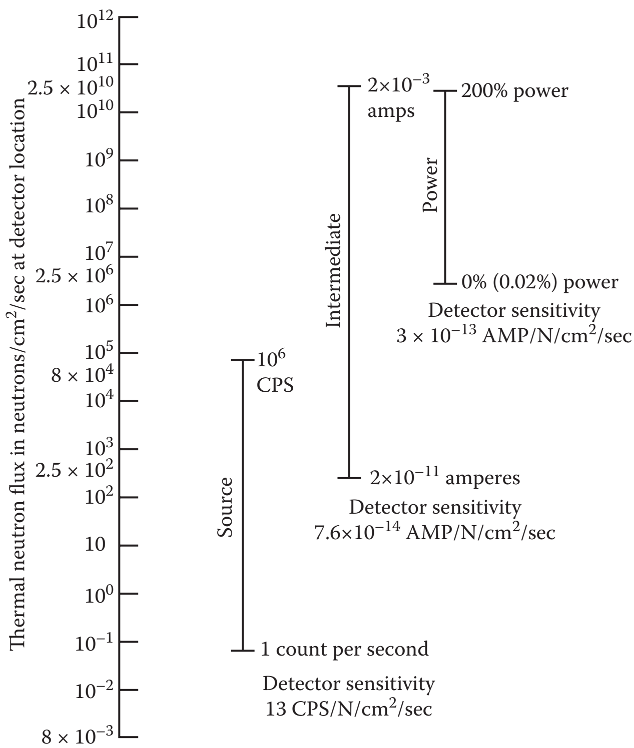


FIGURE 2.12
Nuclear instrumentation system neutron detectors and ranges of operation.

the end plugs. The specific core location of the sources is determined during final design of the core to assure adequate neutron flux at the source range detectors at all times.

Reactor coolant piping and fittings are made of stainless steel or are carbon steel-clad with stainless steel. Carbon steel is used for the pressurizer relief line which connects the pressurizer safety and relief valves to the flanged nozzle on the pressurizer relief tank, and for the nitrogen supply, vent, and drain lines for the pressurizer relief tank. The pressurizer surge and spray lines, loop drains, and connections to other systems are of austenitic stainless steel. Except for the flanged pressurizer safety valve outlet nozzles, all joints and connections are welded. Thermal sleeves are installed where high thermal stresses could develop because of rapid changes in fluid temperature during transients. Valves, piping, and equipment which operate at elevated temperatures are normally covered with thermal insulation to reduce heat losses. Insulation covering the piping and components of the RCS are designed to facilitate its removal for periodic in-service inspections. Insulation used for the RCS is strictly specified to limit chlorides and other halogens. Reactor vessels are frequently insulated with reflective metal insulation systems.

2.8.7 Construction Materials

All valves in contact with reactor coolant are constructed primarily of austenitic stainless steel and employ special materials such as corrosion-resistant hard surfacing and packing. Back seats and stem leak-offs reduce leakage to the containment to essentially zero. The pressurizer safety valves are spring-loaded and self-actuating, with back pressure compensation. The pressurizer also has power-operated relief valves which operate automatically to prevent overpressure. Remotely operated stop valves are provided to isolate these relief valves should excessive leakage occur. Automatic spray valves regulate the pressurizer spray to provide overpressure control. Locally adjustable throttling valves in parallel with the spray control valves deliver a small continuous flow through each spray line and the pressurizer. Local vents permit filling of the system. The piping is arranged so that any liquid discharged through a vent is collected in a container or drain without spilling. Reactor coolant loop isolation valves may be included in a plant design to facilitate some maintenance operations or operation with a loop out of service.

2.9 Auxillary Systems

The Chemical and Volume Control System (CVCS; Figure 2.13) is designed to carry out the functions listed below in support of the RCS:

- Maintain required water inventory in the RCS by maintaining programmed water level in the pressurizer.
- Reduce the concentration of corrosion and fission products in the reactor coolant.
- Provides a means for filling, draining, and pressure testing the RCS.
- In conjunction with the Boron Thermal Regeneration System (BTRS), adjust the boric acid concentration of the reactor coolant for chemical shim control solution.
- Provide a means for control of RCS chemistry.
- Provide high-pressure seal water for the reactor coolant pump (RCP) seals.

2.9.1 Auxiliary Flows

During power operation, a continuous feed-and-bleed stream is maintained to and from the RCS. The feed rate to the RCS is automatically controlled by the pressurizer water level, while the bleed rate can be set by selecting the proper combination of letdown orifices to meet plant operational requirements. Letdown water leaves the RCS and flows through the shell side of the regenerative heat exchanger, where it gives up its heat to makeup water being returned to the RCS. The letdown water then flows through letdown orifices where its pressure is reduced, then through a nonregenerative heat exchanger, followed by a second pressure reduction by a low-pressure letdown valve. After passing through a mixed-bed demineralizer where ionic impurities are removed, the water flows through the reactor coolant filter and into the volume control tank (VCT) via a spray nozzle. An alternate path downstream of the mixed-bed demineralizers can be used to direct the letdown flow to the Boron Recycle System (BRS). The vapor space in the VCT contains hydrogen, which dissolves in the coolant and determines the hydrogen concentration in the RCS. Fission or other noncondensable (He, H_2) gases present are removed from the system by venting of the VCT continuously, intermittently, or prior to plant shutdown. Continuous purging of the VCT considerably reduces the activity level of the reactor coolant. The charging pumps take the coolant from the VCT or the

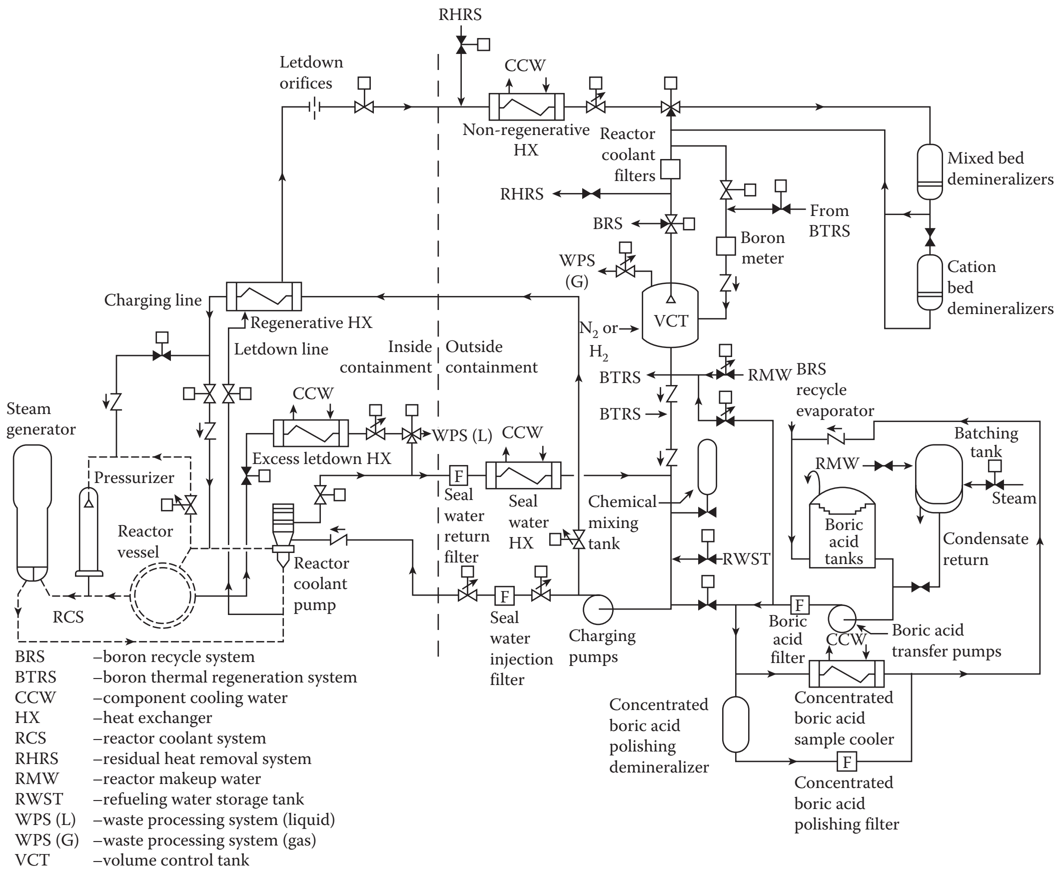


FIGURE 2.13
Chemical and volume control system flow diagram.

BTRS and send it along two parallel paths: back to the RCS through the tube side of the regenerative heat exchanger and to the seals of the reactor coolant pumps. The RCP seal injection flow enters the pump between the labyrinth seals and the number 1 seal. Here the flow divides with some water flowing into the RCS and the remainder leaving the pumps as controlled seal leakage. From the pumps, the controlled leakage water goes to the seal water heat exchanger and then returns to the charging pump suction for another circuit. If the normal letdown and charging path through the regenerative heat exchanger is not operable, water injected into the RCS through the RCP seals is returned to the charging pump suction through the excess letdown heat exchanger and seal water heat exchanger. Surges in the RCS volume are normally accommodated by the pressurizer. However, the VCT is designed to allow for pressurizer level mismatches that may occur. If the VCT is full, a high-water level signal diverts letdown flow to the BRS or waste processing system.

2.9.2 Water Sources

Low-level signals from the VCT initiate reactor makeup control or flow from the Refueling Water Storage Tank (RWST) as a backup. Makeup to the RCS can come from the sources listed below:

- Demineralized and deaerated water supply used to reduce the concentration of dissolved boric acid in chemical shim to accommodate core burnup
- Chemical mixing tank to add small quantities of hydrazine for oxygen scavenging or lithium hydroxide for pH control
- RWST for emergency makeup of borated water
- Boric acid from the boric acid tanks
- A blend of demineralized and deaerated water and concentrated boric acid to match the reactor coolant boron concentration for normal plant makeup

A continuing source of clean water and concentrated solution of boric acid must be provided to replenish these sources. An alternate flow path downstream of the VCT directs the charging pump suction flow to the BTRS to effect boron concentration changes during load follow operations and other plant operations. A boron concentration meter provides accurate measurement and monitoring of boron concentration in the reactor coolant. The readout provides control information which the plant operator can utilize in routine operations with makeup, letdown, chemical treatment of reactor coolant, and the regulation of boron concentration. These data are also used in a boron follow program to assist the operator during various boron concentration changes, for example, load follow, fuel burnup, hot standby, and cold shutdown. Sample points for the measurement unit are located upstream and downstream of the boron thermal regeneration demineralizers. Provisions are also included for boric acid purification. Concentrated boric acid is passed through a demineralizer to remove water-borne impurities such as aluminum, calcium, and magnesium. A filter is also provided for removal of suspended solids.

2.9.3 BTRS

The primary function of the BTRS is to vary the RCS boron concentration during reactor power changes which include daily load follow operations. Boron concentration changes are accomplished automatically by the BTRS through an ion exchange (passive) method. The BTRS makes use of a temperature-dependent ion exchange process to store boron from and release boron to the RCS. Operation of the BRS and evaporators is not required during normal load follow operation. As a result, there is a significant reduction in the water to be processed during normal plant operations and a reduction in BRS requirements. The BRS provides a diverse and redundant method of making boron concentration changes during load follow operations. The BTRS can also assist in making RCS boron concentration changes associated with core burnup, shutdowns, and refuelings. BTRS schematics are shown in Figure 2.14.

The BTRS makes use of a temperature-dependent ion exchange process to store boron from and release boron to the RCS without discharging water to the BRS for later evaporation. The BTRS, which operates in conjunction with the CVCS, consists mainly of several demineralizers, a chiller unit, a chiller pump, heat exchangers, a BTRS recirculation pump, a BTR filter, valves, and associated piping. This equipment controls the temperature and flow rate of the fluid entering the BTRS demineralizers. An alternate charging pump suction path in the CVCS is provided which allows the flow from the VCT to pass through the BTRS when boron concentration changes are required. The fluid temperature is reduced to 50°F (10°C) to store boron on the ion exchange resin and thus dilute the reactor coolant. The fluid temperature is raised to 140°F (60°C) to release boron from the ion exchange resin and thus borate the reactor coolant. The rate of boration or dilution

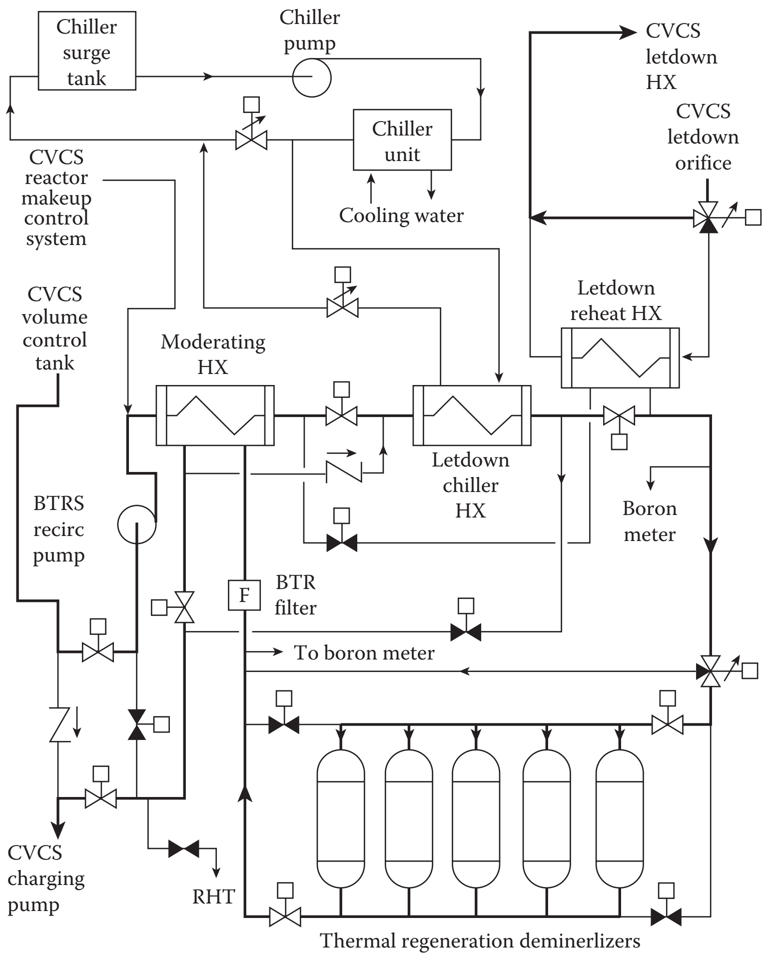


FIGURE 2.14
Boron thermal regeneration system in dilution mode.

is controlled by varying the flow rate through the BTRS demineralizers. The flow path through the BTRS is different for dilution (storage) than for boration (release). During dilution, the letdown stream from the VCT enters the moderating heat exchanger, and from there it passes through the letdown chiller heat exchanger. These two heat exchangers cool the BTRS stream to 50°F (10°C) before it enters the demineralizers. The source of cold water for the letdown chiller heat exchanger is a closed-loop circuit consisting of the chiller unit, chiller pump, and chiller surge tank. The letdown reheat heat exchanger is valved out on the tube side during dilution. The temperature of the BTRS stream at the point of entry to the demineralizers is controlled automatically by the temperature control valve which controls the shell-side flow to the letdown chiller heat exchanger. The letdown stream passes through the demineralizers and enters the moderating heat exchanger shell side where it is heated by the incoming letdown stream. The flow is then returned to the VCT outlet line. Therefore, for dilution, a decrease in the boric acid concentration in the reactor coolant is accomplished by sending the flow from the VCT, at relatively low temperatures (50°F) (10°C), to the thermal regeneration demineralizers. The resin, which was depleted of boron at high temperature during a prior boration operation, is now capable of storing boric acid from the low temperature flow from the VCT. Reactor coolant with a lower boron concentration leaves the demineralizers and is directed to the CVCS. During boration, the charging pump suction stream enters the moderating heat exchanger tube side, bypasses the letdown chiller heat exchanger, and passes through the shell side of the letdown reheat heat exchanger. The moderating and letdown reheat heat exchangers heat the BTRS stream to 140°F (60°C) before the stream enters into the demineralizers. The temperature of the BTRS stream at the point of entry to the demineralizers is controlled automatically by the temperature control valve that controls the flow rate on the tube side of the letdown reheat heat exchangers. The BTRS stream then passes through the demineralizers, enters the shell side of the moderating heat exchanger, passes through the tube side of the letdown chiller heat exchanger, and enters the charging pump suction. The temperature of the BTRS stream entering the charging pump is controlled automatically by the temperature control valve that controls the shell-side flow rate on the letdown chiller heat exchangers. Thus, for boration, an increase in the boric acid concentration of the reactor coolant is accomplished by sending the flow from the VCT, at relatively high temperatures (140°F) (60°C), to the thermal regeneration demineralizers. The water flowing through the demineralizers releases boron that was stored on the resin at low temperature during a previous dilution operation. The boron-enriched reactor coolant is returned to the RCS via the CVCS. The BTRS recirculation pump provides the motive force to route flow from the VCT through the demineralizers during boration and dilution modes. When the BTRS is not being used for boration/dilution, it is isolated from CVCS and the recirculation pump can operate to change the temperature of the demineralizer beds for the next anticipated mode of operation. For example, after completion of a dilution mode, with demineralizers at 50°F (10°C), the BTRS recirculation pump is used to increase the temperature of the demineralizers to 140°F (60°C) for the subsequent boration mode.

2.9.4 Residual Heat Removal System (RHRS)

The primary function of the RHRS is to transfer heat energy from the core and RCS during plant cooldown and refueling operations. The system is designed to reduce the RCS temperature to 140°F (60°C) within 20 hours after reactor shutdown. Provisions are made for continued flow of the reactor coolant to the CVCS during shutdown. The RHRS may also be used to transfer refueling water between the refueling cavity and the RWST at the beginning and end of refueling operations. The residual heat removal pumps and heat exchangers are also utilized as part of the Safety Injection System for emergency core cooling in the event of a Loss-Of-Coolant Accident (LOCA). The accident mode of operation is more fully described in subsection 2.10, Safety Injection System (SIS; part of the Engineered Safeguards System). A flow diagram of the RHRS is shown in Figure 2.15, and represents the configuration for a four-loop NSSS. The configuration for two-loop and three-loop plants differs only in the size of the components and number of branch injection lines to the RCS.

The RHRS consists of two independent, redundant mechanical subsystems, each of which receives electrical power from one of two separate and redundant electrical power trains. Each subsystem consists of one RHR pump, one RHR heat exchanger, and the required piping, valves, and instrumentation. The RHR pumps and heat exchangers are located in the auxiliary building as close as practical to the containment. The piping configurations of the two subsystems are identical, with no major piping cross connects between subsystems in the auxiliary building. Each subsystem has a suction line from an RCS hot leg with

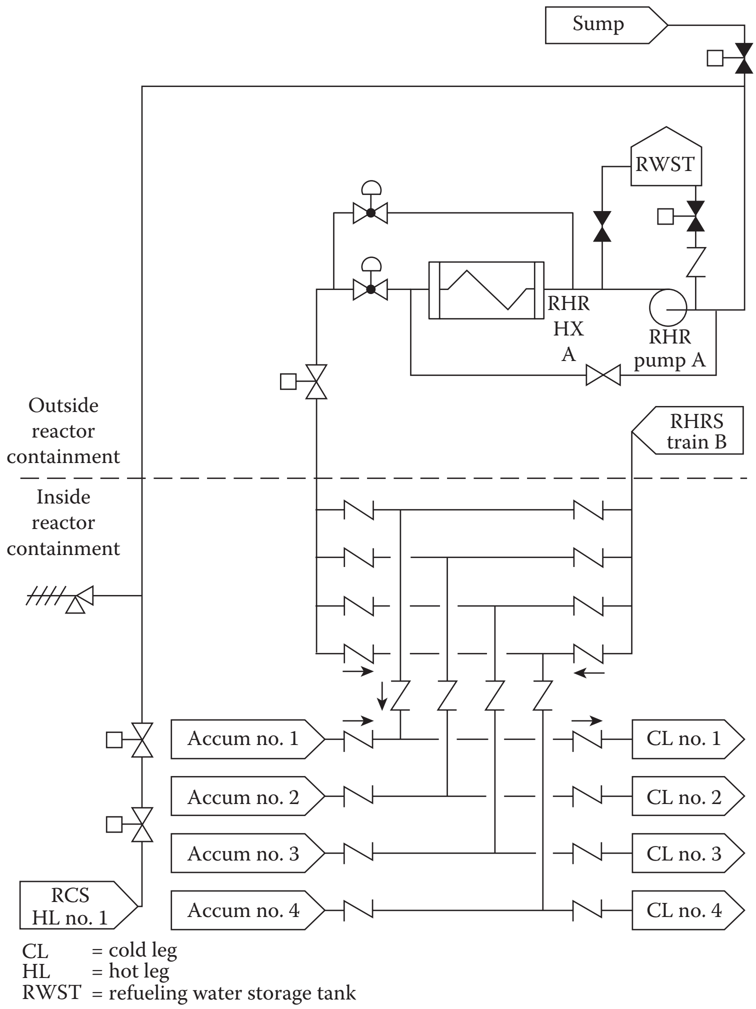


FIGURE 2.15
Safety injection/residual heat removal system in RHR mode.

two normally closed motor-operated series isolation valves providing the isolation necessary to meet the RCS pressure boundary and containment isolation requirements. Each subsystem also has suctions from the containment sump and RWST. These suction sources are required by the RHRS for its operation during mitigation of a LOCA as part of the SIS. The discharge line of each pump is routed to its respective RHR heat exchanger and thence into the containment. A bypass loop is provided around the RHR heat exchanger to permit control of RCS cooldown while maintaining a constant total pump flow. A minimum flow for the RHR pump is provided by a recirculation line from the heat exchanger outlet to the pump suction, which remains open in all system operating modes. Inside the containment, each pump discharge is divided into four branch lines. Each branch line delivers to its respective cold leg through the appropriate accumulator discharge line. No cross connects are present between the two subsystems prior to the discharge headers injecting into the accumulator discharge lines. The branch lines inside the containment are isolated from the RCS cold legs by check valves. During system operation, each RHR pump takes suction from one of the RCS hot legs by its separate suction line. The pumps then discharge flow through the RHR heat exchangers which transfer heat from the hot reactor coolant fluid to the component cooling water circulating through the RHR heat exchanger shell side. The cooled RCS flow is then returned to the RCS cold legs by the shared SIS piping.

2.9.5 BRS

The BRS (Figure 2.16) collects and processes deaerated reactor coolant effluents for reuse as makeup to the RCS as boric acid and reactor makeup water. The BRS provides a means of recycling borated reactor coolant so as to minimize activity releases. The BRS collects water from the RCS through the CVCS letdown line. The letdown is diverted to the BRS as a result of changes made to the RCS boron concentration by the CVCS reactor makeup control. In addition, the BRS collects the overflow of the RCS during heatup operations. These occur during plant shutdown and startup, refueling, and dilution resulting from the slow burnup of the core. Normally, all boron changes required for load changes are made by the BTRS with no discharges to the BRS. However, as a backup to the BTRS, boron concentration changes for load follow can be made by using CVCS makeup and

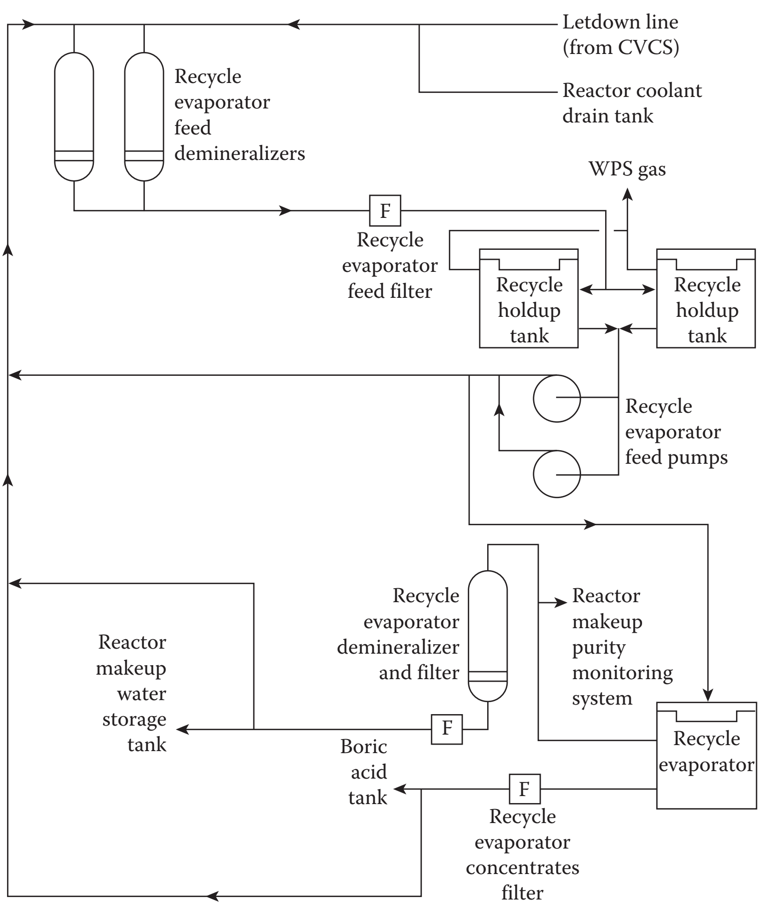


FIGURE 2.16
Boron recycle system.

by discharging reactor coolant effluent to the BRS for processing. The BRS, by means of demineralization, gas stripping, and evaporation, reclaims the boric acid and the primary water. Letdown reactor coolant from the CVCS and deaerated liquid drains is passed through the recycle evaporator feed demineralizers where lithium and fission products are removed. The fluid then flows through a recycle evaporator feed filter and into the recycle holdup tanks. The borated liquid is then pumped to the recycle evaporator package by one of the recycle evaporator feed pumps. Here hydrogen and residual fission gases are removed in the stripping column before the liquid enters the evaporator shell. The evaporative process produces a batch of four-weight-percent boric acid solution. Distillate from the evaporator is pumped to the evaporator condensate demineralizer and then to the reactor makeup water storage tank. Also located in this flow path is a recycle evaporator condensate filter. To provide that the reactor makeup water storage tank will not become contaminated, a scintillation counter is located in the line leading into the tank. A high radiation level signal at the monitor directs distillate back to the recycle evaporator feed demineralizer. Concentrates at four-weight-percent boric acid are normally pumped to the boric acid tanks through the recycle evaporator concentrates filter. If for some reason the boric acid cannot be discharged to the boric acid tanks, it can be diverted back to the recycle holdup tanks. A small sidestream is routed to the reactor makeup purity monitoring system. This system is provided to monitor the condensate from the recycle evaporator to ensure that it conforms with chemical specifications before flowing to the reactor makeup water storage tank.

2.9.6 Steam Generator Blowdown Processing System (SGBPS)

The SGBPS (Figures 2.17 and 2.18) processes blowdown from steam generators to meet chemical specifications that are suitable to allow recycle into the main condenser and to meet radiochemical specifications that are suitable for discharge to the environment when discharge is required. During normal operation, blowdown from each steam generator enters a heat exchanger where the temperature is reduced by condensate water. The flows are then measured before being manifolded. The pressure is reduced, and the blowdown is directed through the prefilter and mixed-bed demineralizers in series. The fluid flows through a radiation monitor and is normally recycled to the main condenser, but may be discharged to the environment through the discharge canal when required. Instrumentation is provided at strategic points to monitor the functional integrity and operating efficiency of the system. An increase in conductivity, indicated at the outlet of each demineralizer, signals resin bed exhaustion. When the upstream bed is exhausted, flow is directed around that bed. With the aid of the spent resin sluice pump, spent resin is transferred to the spent resin storage tank, and new resin is charged to the demineralizer. This fresh demineralizer is then valved back online as the new downstream bed. Before resin disposal, the spent resin storage tank is fluidized with nitrogen gas or sluice water to loosen the resin. The tank is pressurized with nitrogen to discharge the resin to the solid waste processing system.

2.10 Engineered Safeguards Systems

Engineered Safeguards Systems are integrated with the other main and auxiliary systems to protect the plant, its personnel and the general public. After 2001, security systems were beefed-up in the United States and elsewhere to meet the terrorist threat. These latter

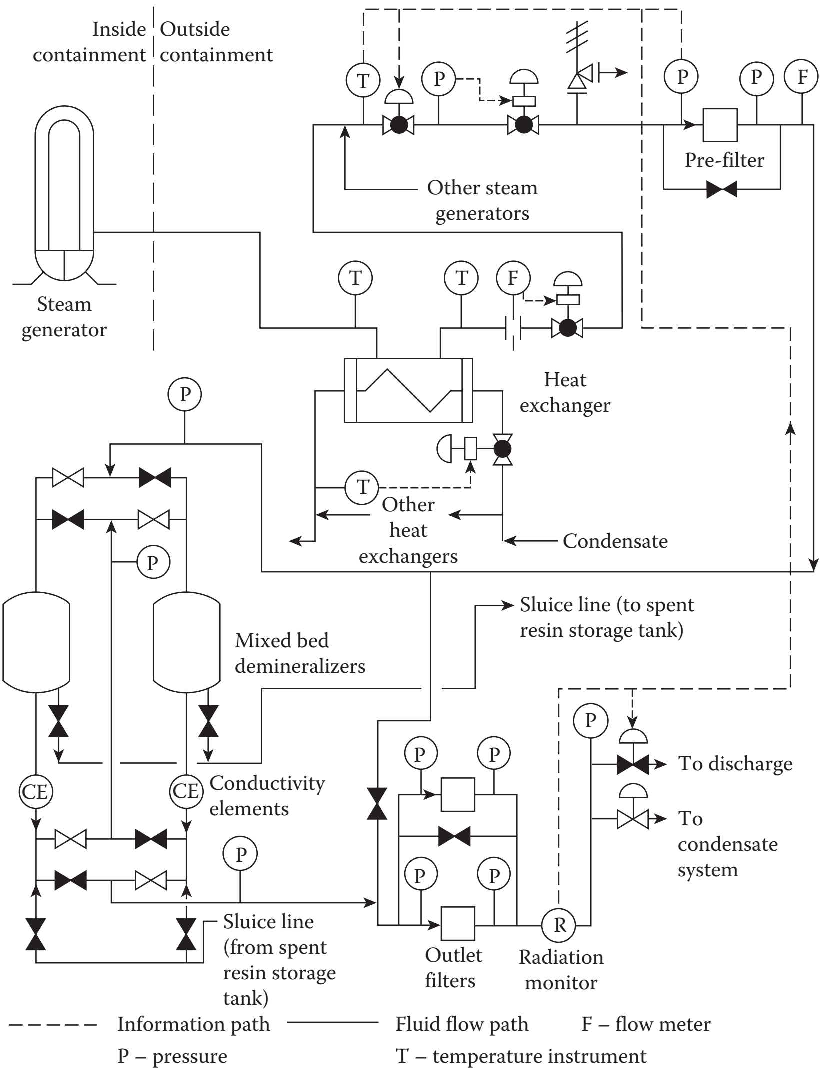


FIGURE 2.17
Steam generator blowdown processing system.

systems, of course, are not discussed in the literature. Along with the plant Safeguards Systems are computers, alarms, interlocks and sophisticated software to see that all systems work as they are supposed to in a wide variety of hypothesized upset and accident conditions, those that could be part of the everyday operation of the facility, as well as those that are extremely rare but are postulated to be potentially catastrophic if not contained or prevented from developing. What follows is a breakdown by component subsystems of the function and interaction of the various parts of the Safeguards System.

2.10.1 SIS

The SIS has multiple functions. Its primary function is to provide emergency core cooling in the event of a LOCA resulting from a break in the RCS. The SIS also provides a safety grade method for addition of negative reactivity via injection of borated water to meet

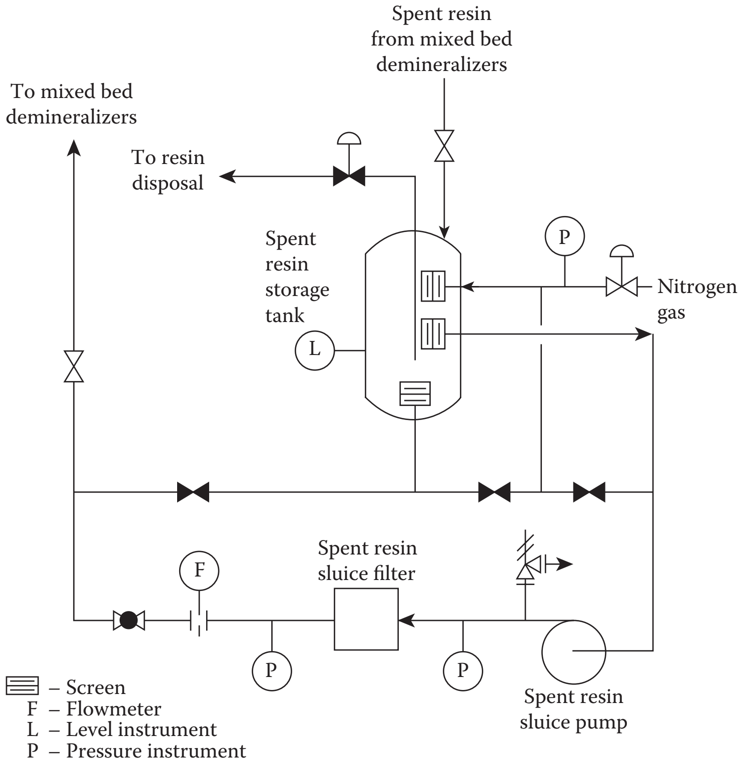


FIGURE 2.18
SGBPS spent resin storage tank.

shutdown requirements, or to compensate for the reactivity increase caused by cooldown transients such as a steam line break. The SIS positive displacement hydrotest pumps also provide a backup source of reactor coolant pump seal injection water, diverse to the CVCS. In the unlikely event of a LOCA, the SIS is designed to limit increases in fuel clad temperatures, core geometry distortion, and metal–water reaction for all breaks up to and including the double-ended severance of a reactor coolant loop. For the more probable break sizes inside diameter, (≤5 inches (12.7 cm), the SIS is designed to minimize core damage by providing flow to the core that is sufficient to prevent the mass depletion-related uncovering of the core based on current analytical models. The system is designed to provide not only emergency core cooling, but also continued cooling during the long-term phase after the accident.

2.10.2 High-Pressure Injection

High-pressure safety injection water is provided by separate high-head pumps while lower pressure injection water is supplied by the RHRS pumps. Passive accumulator tanks are inside the containment to provide for fast injection of water after a LOCA. Figure 2.19 shows the injection phase of the SIS, and is the configuration for a four-loop plant. The configurations for two- and three-loop plants differ only in the size of the components and the number of branch injection lines to the RCS.

The SIS consists of two independent and redundant mechanical subsystems, each of which receives electrical power from one of two separate and redundant class 1E electrical

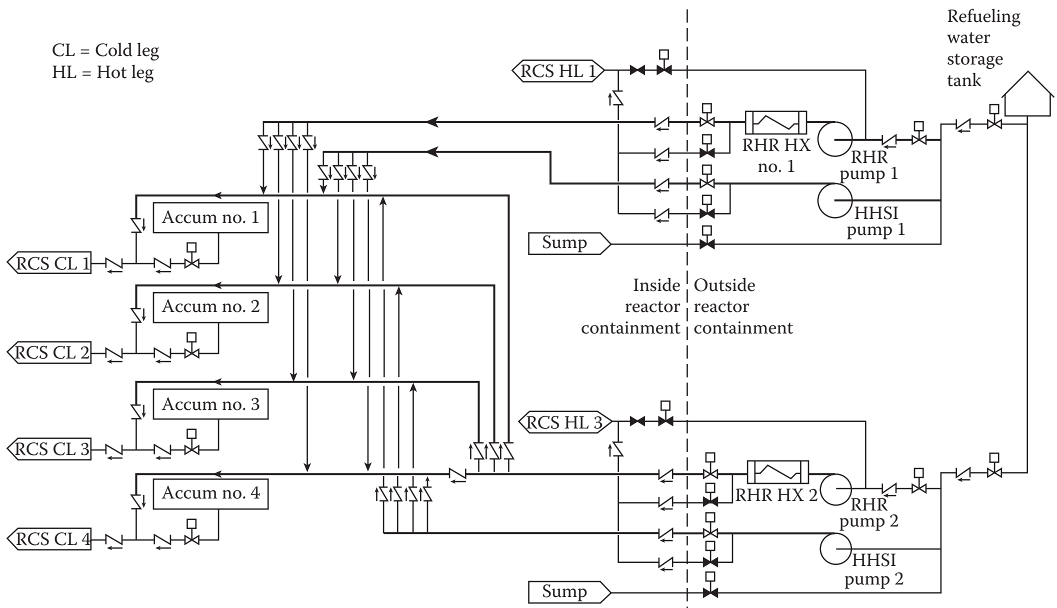


FIGURE 2.19
Safety injection/residual heat removal system, injection mode.

power trains, and each of which receives an actuation signal from one of two separate and redundant actuation trains. The major components are two high-head safety injection (HHSI) pumps, two RHR pumps, and four accumulators. There is also a single positive displacement pump which provides emergency injection flow if the centrifugal charging pumps are not available. It is not directly associated with subsystem A or B, and can be powered from either electrical train. The piping configurations and valving associated with each of the two subsystems are identical. There are no major piping interconnections between the two subsystems in the auxiliary building with the exception of the containment, the individual loop injection branch lines from each of the four main supply headers from the RWST. Inside, the pumps are directed into a common pipe header and injected into the respective accumulator discharge lines to the RCS cold legs. Each pump delivers to all of the cold legs. Additionally, each subsystem delivers to the hot leg of one loop.

Each of the two mechanical subsystems contains one HHSI pump and one RHR pump. With the exception of drawing from a common suction header from the RWST, these pumps share no piping and are not connected outside the containment. Lowhead safety injection is provided by the RHRS. The HHSI pump is a dedicated component of the SIS and does not provide any function required for normal plant operation. This permits the HHSI pump and associated piping to be retained in a ready configuration which requires only the pump to start to initiate safety injection. Each HHSI pump is also provided with a full flow test loop which permits periodic verification of hydraulic performance. Design of the HHSI pump is such that testing or inadvertent start will not result in injection to the cold leg while the RCS is at normal operating temperature and pressure.

2.10.3 System Safeguards

An inadvertent safety injection signal will not result in lifting the pressurizer power-operated relief or safety valves. A continuous minimum flow path which does not require isolation during the injection phase is provided so that an inadvertent pump start will not result in the deadheading of the HHSI pump. Each HHSI pump discharge header feeds each RCS cold leg through individual branch lines. Redundancy of the two independent subsystems is incorporated to provide for reliability of operation and continued core cooling, even in the event of a failure of a single component in the fluid system or related control and power supply systems to respond actively in accordance with its design function. The HHSI and the RHR pumps can take direct suction from the containment sump, thereby increasing the system reliability for long-term post-accident recirculation. The HHSI and RHR pumps and all motor-operated valves, except the normally open accumulator isolation valves, are located outside the containment to provide for ease of access for maintenance. The accumulator tanks are passive components normally requiring no maintenance, and are located inside the containment to provide rapid injection of water following a LOCA.

Operation of the SIS following a LOCA is described in the following paragraphs in three distinct phases:

- Injection phase
- Cold-leg recirculation phase
- Cold-leg and hot-leg recirculation phase

In addition, this system can be operated with a nonfaulted RCS to provide an alternate source of borated makeup water injection. This function is described separately in the following paragraphs.

2.10.4 SIS Components

The principal mechanical components of the SIS that function immediately following a LOCA are the accumulators, the HHSI pumps, the RHR pumps, the RWST, and the associated piping and valves. Because the SIS components have no active function during normal power operation, they are maintained in a configuration aligned for safety injection immediately upon pump start; only check valves are required to change position. For large pipe ruptures, the RCS would be rapidly depressurized and voided of coolant. A high flow rate of emergency coolant would therefore be required to quickly recover the exposed fuel rods to limit possible core damage. This high flow is provided by the accumulators, followed by the HHSI and RHR pumps, all discharging into the cold legs of the RCS. The HHSI and RHR pumps are aligned to take suction directly from the RWST. For smaller breaks, depressurization of the RCS will occur at a slower rate. The HHSI and RHR pumps will be started upon receipt of the actuation signal and will run on miniflow until the RCS pressure falls below the respective shutoff heads of the pumps. Similarly, the accumulator discharge will begin automatically as the RCS pressure decreases below the accumulator tank pressure. The active safety injection phase is actuated by any one of the following:

- Low pressurizer pressure reported by two out of four signals.
- Excessive cooldown protection signals.
    - Low compensated steam line pressure as reported by two out of four signals in any steam line. This protects against secondary breaks during power generation.

    - Steam pressure high negative rate as reported by two out of four signals in any one steam line plus low pressurizer pressure in two out of four channels. This protects against secondary breaks during planned cooldown from power generation.
- High containment pressure as reported by two out of four signals.
- Manual actuation.

The receipt of the safety injection signal initiates the following emergency core cooling actions:

- The onsite emergency diesel generators are started
- HHSI pumps are started
- RHR pumps are started

2.10.5 Cold Leg Recirculation Mode

The changeover from the injection mode to the cold-leg recirculation mode is initiated automatically when the RWST level channels indicate an RWST level less than a low-level setpoint in conjunction with the initiation of the engineered safeguards actuation signal. Protection logic is provided to automatically open the recirculation valve in each of the SIS subsystems when the above signals are generated. To prevent backflow to the RWST of contaminated water from the containment sump, the HHSI pump mini-flow isolation valves are automatically closed at this time. This automatic action would align each of the SIS subsystems for cold-leg recirculation. In this mode, the HHSI and RHR pumps for each subsystem take suction directly from one of the two containment sump recirculation lines. All pumps will continue to operate and will deliver water to the RCS cold legs without interruption throughout the automatic recirculation switchover operation. As in the injection phase, no operator action is required to initiate realignment of either of HHSI or the RHR pump. Furthermore, each pump draws directly from the containment sump and operates independently of the operation of any other pump.

Approximately 24 hours after the switchover to cold-leg recirculation, simultaneous flow to the RCS cold legs and hot legs is established for the purpose of long-term core cooling. The alignment of the HHSI pumps and the RHR pumps remains essentially the same as during the cold-leg recirculation phase, except that a substantial portion of the flow is directed to the RCS hot legs. The purpose of the change in flow alignment is to provide for termination of boiling and to prevent buildup of boron from interfering with core cooling following a large cold-leg break. Switchover to this phase is performed manually from the control board.

The HHSI pumps also serve as safety grade backup to the CVCS charging pumps for RCS boration. Manual starting of the HHSI pumps will result in the injection of borated water as RCS pressure is decreased below the shutoff head of the pump. A manual throttling valve, operable from the main control board, is provided to permit control of the injection flow rate as the RCS pressure is reduced.

2.10.6 Emergency Feedwater for Secondary Loop

The primary function of the Emergency Feedwater System (EFWS; Figure 2.20) is to supply feedwater to the steam generators following accident or transient conditions when the

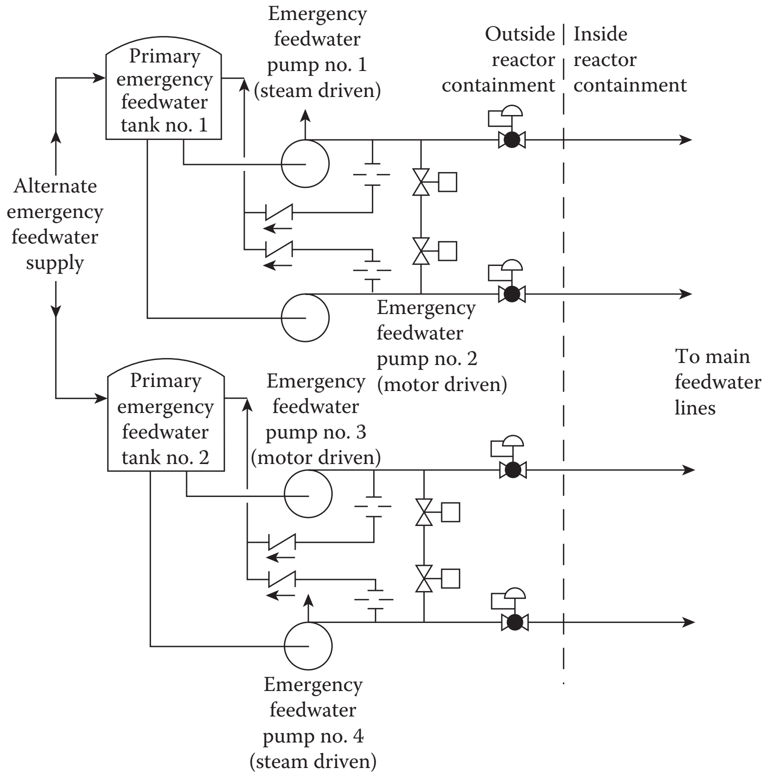


FIGURE 2.20
Emergency feedwater system, flow diagram.

main feedwater system is not available. The EFWS thereby maintains the capability of the steam generators to remove plant-stored heat and core decay heat by converting the emergency feedwater to steam which is then discharged to the condenser or to the atmosphere. Although the EFWS is also capable of supplying feedwater to the steam generator during normal plant operations of startup, shutdown, and hot standby, it is normally not used for this service. Instead, a small startup pump(s) is located in the main feedwater system and used to supply the reduced amount of feedwater required during these normal plant operations.

The EFWS consists of two identical subsystems, each of which receives electrical power from one of two separate Safety Class 1E electrical power trains. Each subsystem consists of a primary emergency feedwater tank, one motor-driven emergency feedwater pump, one turbine-driven emergency feedwater pump, and the required piping, valves, instruments, and controls necessary for system operation. The motor-driven and turbine-driven pumps are located in the emergency feedwater pump building. The use of motor-driven and turbine-driven pumps satisfies the requirement that the pumps be powered by diverse power sources. In operation, the emergency feedwater pumps take suction from the primary emergency feedwater tanks and discharge the water into the main feedwater piping between the steam generator feed nozzle and the last check valve in the main feedwater line. Each pump is provided with an orificed recirculation line leading back to the primary emergency feedwater tank. This line provides recirculation flow any time the pumps are operating. The steam supply line for each turbine-driven pump is connected to the main steam line from a steam generator. This line is fitted with a steam admission valve which is a pneumatically operated valve arranged to fail-open on loss of air or electrical power. A primary emergency feedwater supply tank, to which the suction of the emergency feedwater pumps are normally aligned, is provided in each subsystem. The tanks are safety grade and seismically qualified. Each tank contains a quantity of condensate quality water sufficient to allow the plant to be maintained in hot standby for 13 hours then allow a five-hour cooldown of the plant to 350°F (176.7°C). The maximum permissible water temperature in the primary emergency feedwater supply tank is 120°F (48.9°C).

An alternate emergency feedwater source (condensate storage tanks) should also be provided. The alternate water source should contain sufficient water to allow the plant to be maintained in hot standby conditions, with one RCP operating, for 2 days (48 hours) beginning 18 hours after reactor trip. The maximum water temperature in the alternate water supply should be 120°F (48.9°C). Normally open, fail-open, air-operated, flow-modulating valves are located in each pump discharge line. These valves will normally be full open when the system is activated. The valves are provided to allow operator control of the emergency feedwater flow rates to the steam generators so that, in the long term, steam generator water levels can be restored and maintained in the narrow control range. For conservatism, it is assumed that no operator action can be taken for 30 minutes, and for this period it is assumed that these valves will be full open.

The EFWS can be used to supply feedwater to the steam generators during a plant startup when only small amounts of feedwater are required; however, the system is not normally used for this purpose. Other equipment in the main feedwater system design is provided to supply this reduced amount of feedwater during the plant startup and heatup.

The EFWS is not operated during normal plant operations, but remains in a state of readiness to provide emergency feedwater to the steam generators in the event of transient or accident conditions. In the event of such occurrences, the emergency feedwater pumps are automatically started as follows:

- Low-low level in 2/4 level channels in any one steam generator: motor-driven pumps
- Low-low level in 2/4 level channels in any two steam generators: turbine-driven pumps
- Safety injection: motor-driven pumps.

All valves in the system discharge path are open, so the automatic startup of the pumps will result in the immediate delivery of emergency feedwater into the steam generators. The system is designed to supply at least the minimum required flow, within one minute of the actuation signal, to at least two effective steam generators (or one effective steam generator in the case of a two-loop plant), and to continue this delivery for an indefinite period without operator action. When operator action can be taken (after an assumed 30-minute delay), the emergency feedwater flow is adjusted by positioning the flow-modulating valves to restore and maintain the steam generator water levels within the narrow control range.

With the reactor tripped, and with the EFWS supplying water to the steam generators at a rate equivalent to the rate at which steam is being removed to dissipate core decay heat and the heat input of one RCP (assumed to be operating), the plant is in a stable hot standby condition. The plant can be maintained in this condition for a period limited only by the amount of water in the primary and alternate emergency feedwater supplies. If the initiating event can be resolved, plant power operations can be resumed. Normal feedwater flow to the steam generators by the main feedwater system is resumed and the emergency feedwater pumps are manually stopped. If the initiating event cannot be resolved, a plant cool down must be performed.

In this case, the EFWS continues to supply feedwater to the steam generators throughout the cool down until the primary system hot leg temperature is reduced to 350°F (176.7°C). At this temperature, the RHRS is activated and the EFWS is secured. The RHR continues the cool-down to cold shutdown conditions.

The EFWS can be used to supply feedwater to the steam generators for normal plant cool downs. To accomplish this, the pumps are manually started and the flow rates manually controlled by positioning the flow modulating valves. When the reactor coolant hot leg temperature is reduced to 350°F (176.7°C) and the RHRS has been activated, the EFWS is secured. The EFWS is not normally used for plant cool down during shutdown because other design provisions contained in the main feedwater system should be used to supply the steam generators in this case.

2.10.7 Component Cooling Water System (CCWS)

The CCWS provides a continuous supply of cooling water to plant components which handle potentially radioactive fluids. In doing so it forms an intermediate barrier between these potentially radioactive systems and the Service Water System, thereby reducing the possibility of discharging radioactivity to the environment. Component cooling water is supplied to NSSS as required by the following operations:

- Removal of heat from various components during normal power generation
- Removal of residual and sensible heat from the RCS through the RHRS during plant cool down
- Cooling of safeguards equipment following an accident

The CCWS is designed to provide cooling for the following components (The acronyms in the following list are for the systems of which these components are a part, e.g., CSS is containment spray system.):

- RCPs (RCS)
- Residual heat exchangers (RHRS)
- Residual heat removal pumps (RHRS)
- Letdown heat exchanger (CVCS)
- Excess letdown heat exchanger (CVCS)
- Seal water heat exchanger (CVCS)
- Centrifugal charging pumps (CVCS)
- Control rod drive mechanism coil cooler (RCS)
- Refrigerated dryer
- Recycle evaporator package (BRS)
- Chiller unit (BTRS)
- Waste evaporator package(s) (WPS)
- Waste gas compressor package (WPS)

- Hydrogen monitor
- Reactor coolant drain tank heat exchanger (WPS)
- Spent fuel pit heat exchangers (SFPCS)
- Sample heat exchangers (SS)
- Gross failed fuel detector
- Containment spray pump heat exchangers (CSS)
- Containment fan coolers
- Service air compressor
- Instrument air compressor
- Positive displacement pump (CVCS)

The CCWS consists of a Safeguards subsystem and an Auxiliary subsystem (Figure 2.21). The CCWS Safeguards subsystem is considered an engineered safeguards system because it is required to remove decay heat during post-accident and to provide cooling water to safeguard equipment. The subsystem is designed to retain total physical separation consisting

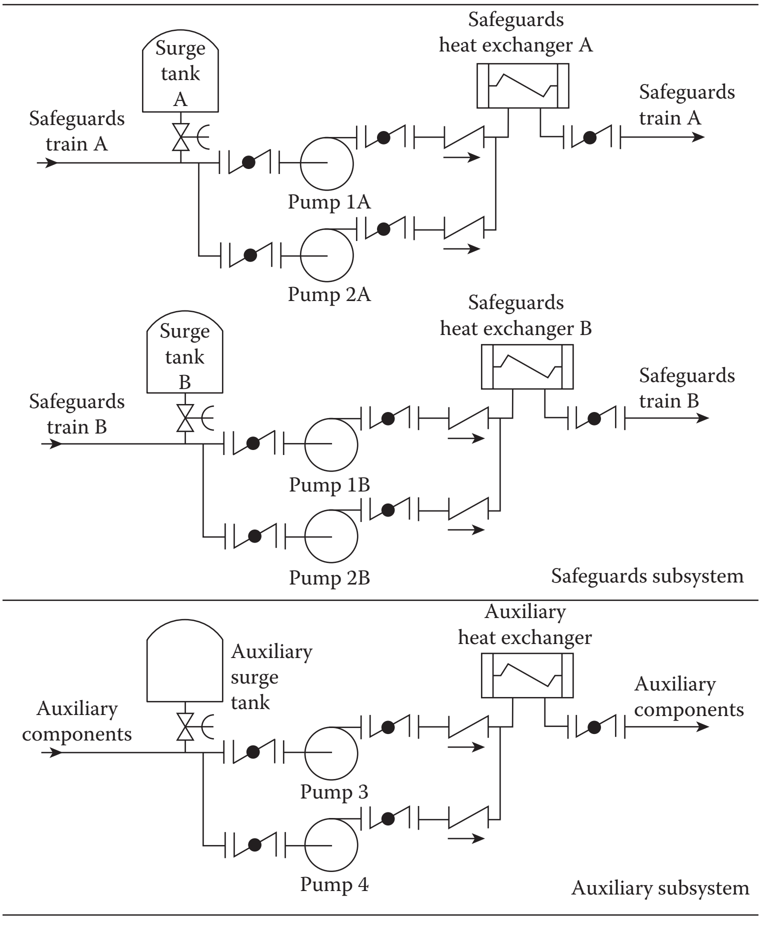


FIGURE 2.21
Component cooling water system, flow diagram.

of two separate, independent safeguards trains with no cross connection between the two trains. Components of each safeguards train (pumps and heat exchanger) are housed in separate safeguards equipment cubicles. The plant separation facilitates compliance to standards in the area of fire protection, flooding prevention, and plant sabotage.

The Safeguards subsystem is divided into two separate and redundant safeguards trains and provides cooling water to safeguards equipment such as the residual heat removal heat exchangers and pump, the RCP thermal barriers, the containment fan coolers, the containment spray pump heat exchangers, the spent fuel pit heat exchangers, and the positive displacement pump. Each safeguards train is sized to supply 100% post-accident component cooling flow requirements. Each train consists of two 100%-capacity component cooling pumps, one 100% component cooling heat exchanger, one component cooling surge tank, a chemical addition tank, cooling lines to the various safeguards components, and associated piping, valves, and instrumentation. This arrangement permits single failure criteria to be met with up to one pump per train removed from service for maintenance. The component cooling water flows from the pump, through the shell side of the component cooling heat exchangers, through the components being cooled, and back to the pump. The surge tank is connected to the suction side of the component cooling pumps.

The component cooling Auxiliary subsystem provides cooling water to nonsafeguards components in the CVCS, BRS, Waste Processing systems, and other auxiliary subsystems. The Auxiliary subsystem provides 100% component cooling for normal operation. It consists of two 100% component cooling pumps, one 100% component cooling heat exchanger, one component cooling surge tank, a chemical addition tank, and associated piping, valves and instrumentations. The subsystem's flow path is similar to that of the Safeguards subsystem. The surge tanks of Safeguards and Auxiliary subsystems provide like functions. They provide a surge volume to accommodate thermal expansion and contraction of system volume during transients, and collects water that may leak into the system from components being cooled. The tanks also contain sufficient water volume to provide component cooling water until a design basis passive failure can be isolated.

Water chemistry control of the CCWS is accomplished by chemical additions to the chemical addition tank. A safety grade makeup source is provided by the demineralized water system or reactor makeup system (emergency makeup only) and delivered to the surge tank.

During normal full-power operation, one component cooling safeguards train and the component cooling auxiliary subsystem (each operating with one pump and one heat exchanger) are required to accommodate the heat removal loads. The safeguards train services the containment fan coolers, the RCP thermal barriers, and the spent fuel pit heat exchangers; the auxiliary train supplies component coolant to other components required during normal operation.

During plant normal cool down, the standby train of the Safeguards subsystem is placed in operation at approximately four hours after reactor shutdown. The Safeguards subsystem provides component cooling water flow to the operating residual heat exchangers and pumps. The component cooling water inlet temperature to various components during normal cooldown is permitted to increase to 120°F (49°C), but it must return to 105°F (40°C) after four hours. Pump failure will not affect the time required for cooldown because a standby pump in each train of the Safeguards subsystem is provided. A spare pump is also available in the Auxiliary subsystem. Failure of a safeguards train will not affect the plant's capability to remove decay heat; but cooldown time will be extended. The CCWS pumps, heat exchangers of the Safeguards subsystem and its associated piping, valves and instrumentations are seismically qualified as required for an engineered safeguards system.

2.11 Containment Systems

Multiple barriers designed into the power plant provide containment of radioactive products at three fundamental levels:

- Zircaloy tubes of the fuel rods
- RCS pressure boundaries
- Reactor containment vessel (building)

Diversity of concept, defense in depth, and redundancy of systems as it relates to protection and containment are discussed in other sections of this handbook. This subsection discusses the design of the Reactor Containment Building (hereafter referred as "the containment") that will contain and control any release of radioactivity to the environment under normal, upset, or emergency conditions. Also included are containment systems that will protect the integrity of the containment by reducing steam pressure and controlling hydrogen to avoid an explosive mixture. Ventilation systems provided with filters will reduce radioactivity in the containment atmosphere to permit safe access into the containment. The structure provides biological shielding for normal and accident situations.

Several types of containment structures have been designed and proposed to utilities. Those designs in prevalent use incorporate steel vaults or concrete vessels lined with steel plate. Steel vessels can be cylindrical or spherical in shape. Reinforced concrete vessels, which may in some cases be post-tensioned, are cylindrical with hemispherical domes. The type of structure chosen by a utility is dependent on plant layout, site characteristics, and relative costs of alternatives for a particular project. Typical containments are shown in Figure 2.22. A description is given of the prestressed concrete containment having a cylindrical shell, a hemispherical dome, and a flat base slab (with a pit). The inside face of the concrete shell, dome, and floor is steel plate lined to ensure a high degree of leak tightness. Other designs while varying in detail will meet the same functional requirements. The cylindrical shell is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a two-way post-tensioning system. There are three buttresses equally spaced around the containment. Hoop tendons are anchored at buttresses 240° apart, bypassing the intermediate buttress. Each successive hoop is progressively offset 120° from the one beneath it. (Another possible tendon arrangement includes U-tendons and hoops; a third system includes helicoidal tendons in opposite patterns.)

The foundation base slab is a concrete structure conventionally reinforced with high-strength reinforcing steel. A continuous access gallery is provided beneath the base slab for the installation and inspection of vertical tendons. The base liner, installed on top of the structural slab, is covered with concrete for post-tension. The containment completely encloses the entire reactor and RCS and ensures that an acceptable upper limit for leakage of radioactive materials to the environment would not be exceeded even if gross failure of the RCS were to occur. The approximate dimensions of the containment are: 124 feet (37.8 m) inside diameter, 205 feet (62.5 m) inside height, 3-1/2 feet (1.07 m) wall thickness, and 2-1/2 feet (0.76 m) dome thickness. The internal net free volume approximates 2,000,000 cubic feet (56,600 cubic meters).

The containment is designed for all credible conditions of loading, including normal loads, loads during LOCA, test loads, and loads due to adverse environmental conditions.

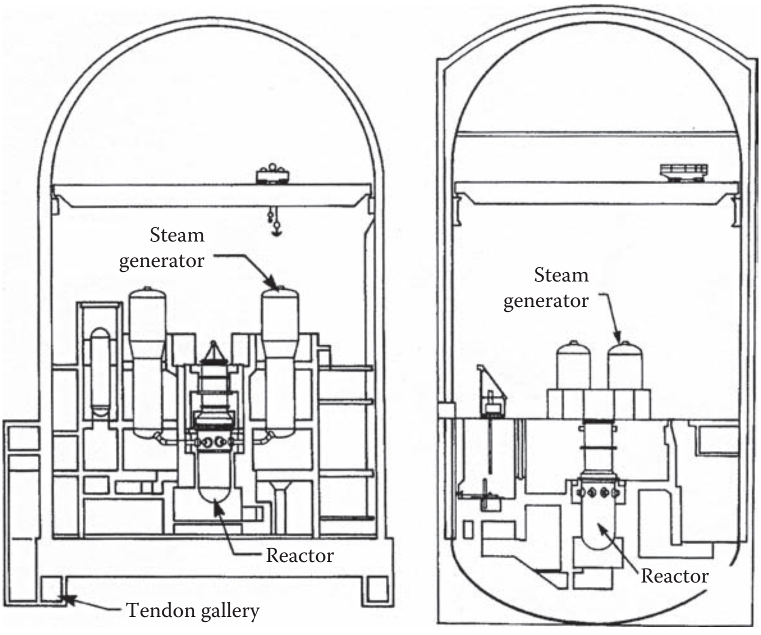


FIGURE 2.22
Different types of containment, prestressed concrete on left and cylindrical steel on right.

The two critical loading conditions are those caused by the design basis accident (DBA) resulting from failure of the RCS, and those caused by an earthquake.

Loading considerations are applied as detailed below:

2.11.1 DBA

The minimum design pressure and temperature of the containment are equal to the peak pressure and temperature occurring as the result of any rupture of the RCS up to and including the double-ended severance of a reactor coolant pipe. The supports for the RCS are designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping so that its coincidental rupture with that of the steam or the feedwater systems is not considered credible. Transients resulting from the DBA serve as the basis for a containment design pressure of 60 psig (4.46 bar). Transients resulting from other accidents could be controlling for specific compartments of the containment.

2.11.2 Thermal Loads

The variation of temperature with time and the expansion of the liner plate are considered in designing for the thermal stresses associated with the DBA.

2.11.3 Dead Loads

Dead loads consist of the weight of the concrete wall, dome, base slab, internal concrete and permanent equipment, machinery, components, and the like.

2.11.4 Live Loads

Live loads consist of all loads except dead, accident, seismic, flood, and wind, and include snow loads on the domed roof of the containment. Live loads are assumed for the design of internal slabs consistent with the intended use of the slabs.

2.11.5 Earthquake Loads

Earthquake loading is predicated upon a design earthquake with a ground acceleration equal to the Operating Basis Earthquake (OBE) for the site selected. In addition, a maximum hypothetical earthquake having a ground acceleration equal to the Design Safe Shutdown Earthquake (SSE) for the site selected is used to check the design and ensure no loss of function. A vertical component having a magnitude equal to the horizontal component is applied simultaneously. A three-dimensional dynamic analysis is used to arrive at equivalent static loads for design. Soil–structure interactions are included in the dynamic model.

2.11.6 Wind Forces

Wind loading results from site conditions, including consideration of hurricane winds. This wind loading is considered for the design of all structures. However, wind loads are not applied simultaneously with seismic loads.

2.11.7 Hydrostatic Loads

Uplift forces created by hydrostatic pressure are included in the design of all structures.

2.11.8 External Pressure Load

External pressure loading with a differential of 2-1/2 psi (0.17 bar) from outside to inside are considered. The external design pressure is also adequate to permit the reactor building to be cooled from an initial maximum operating condition of 120°F (49°C) to an internal temperature during shutdown of 50°F (10°C) (winter).

2.11.9 Prestressing Loads

Prestressing loads allow for slip at anchorage, elastic shortening of concrete, creep of concrete, shrinkage of concrete, relaxation of steel stress, and frictional losses in the tendon ducts. Load factors are in accordance with the ACI/ASME Code.

2.11.10 Containment Design Criteria

The safety of the structure under extraordinary circumstances and the performance of the containment at various loading stages are the primary considerations in establishing the structural design criteria. The two basic criteria are

- Integrity of the containment liner is guaranteed under all credible loading conditions.
- Structure has a low-strain elastic response such that its behavior is predictable under all design loadings.

The strength of the containment at working stress and overall yielding is compared with the allowable values under the various loading combinations to ensure safety. To ensure proper performance, the analysis and design of the containment is carried out with consideration for strength, the nature and the amount of cracking, the magnitude of deformation, and the extent of corrosion. The structure is designed to meet performance and strength requirements under the following conditions:

- Before prestressing
- At transfer of prestress
- Under sustained prestress
- At design loads
- At factored loads

The base slab acts primarily in bending rather than membrane stress and therefore, it is not prestressed.

2.11.11 Design Method

The containment shell is analyzed for individual and various combinations of loading cases of dead load, live load, prestress, temperature, and pressure. The design output includes direct stresses, shear stresses, principal stresses, and displacements of each nodal point. Stress plots which show total stresses resulting from appropriate combinations of loading cases are made and areas of high stress identified. If necessary, the modulus of elasticity is corrected to account for the nonlinear stress–strain relationship at high stresses. Stresses are then recomputed if a sufficient number of areas requiring attention exist.

2.11.12 Containment Liner Criteria

To meet the specified leak rate under accident conditions, the containment liner satisfies the following criteria:

- Containment liner is protected against damage by missiles.
- Containment liner strains are limited to allowable values that have been shown to result in leak tight vessels or pressure piping.
- Containment liner is prevented from developing significant distortion.
- All discontinuities and openings are anchored to accommodate the forces exerted by the restrained containment liner, and careful attention is paid to details at corners and connections to minimize the effects of discontinuities. The following fatigue loads are considered in the design of the containment liner:
  - Thermal cycling due to annual outdoor temperature variations. Daily temperature variations do not penetrate a significant distance into the concrete shell to appreciably change the average temperature of the shell relative to the containment liner. The number of cycles for this loading will be 40 cycles for the plant life of 40 years.
  - Thermal cycling due to containment interior temperature varying during the startup and shutdown of the reactor system. The number of cycles for this loading is assumed to be 500.

- Thermal cycling due to DBA is assumed to be one cycle. Thermal load cycles in the piping systems are somewhat isolated from the containment liner penetrations by the concentric sleeves between the pipe and the liner. All penetrations are verified for a conservative number of cycles to be expected during the plant life. The containment liner as well as any carbon steel surface exposed to the containment atmospheres is coated with a coating system qualified to resist accident conditions without peeling, scaling, or blistering.

2.11.13 Equipment and Personnel Access Hatches

An equipment hatch 21 feet (6.4 m) in diameter is provided for access to the containment. It is fabricated from welded steel and furnished with a double gasketed flange and bolted dished door. Equipment up to and including the size of the steam generators can be transferred into and out of the containment through this hatch. Two personnel locks are provided. One of these is for emergency egress only. Each personnel lock is a double door, welded steel assembly. Quick-acting valves connect the personnel lock with the interior and exterior of the containment for the purpose of equalizing pressure in the two systems when entering or leaving the containment. The two doors in each personnel lock are interlocked to ensure that one door is completely closed before the opposite door can be opened.

Remote indicating lights and annunciators situated in the control room indicate the door operational status. Provision is made to permit bypassing the door interlocking system to allow doors to be left open during plant cold shutdown. Each door hinge is designed to be capable of independent, three-dimensional adjustment to assist proper seating. An emergency lighting and communication system operating from an external emergency supply is provided in the lock interior.

2.11.14 Special Penetrations

Fuel Transfer Penetration: A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment building and the spent fuel pool in the fuel handling building. The penetration consists of a 20-inch (51 cm) stainless steel pipe installed inside a casing pipe. The inner pipe acts as the transfer tube and is fitted with a double gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pit. The casing pipe is provided with expansion joints and is connected to the containment liner. This arrangement prevents leakage in the event of an accident.

Containment Supply and Exhaust Purge Ducts: The ventilation system purge duct is equipped with two tight seating valves to be used for isolation purposes. The valves are remotely operated for containment purging.

2.11.15 Containment Isolation System (CIS)

The CIS provides the means of isolating the various fluid systems passing through the containment walls as required to prevent the release of radioactivity to the outside environment. Design bases of the CIS consider several factors. Subsequent to an accident which may release radionuclides within the containment, there must be a barrier in all pipes or ducts that penetrate the containment. Leakage from the containment through these pipes or ducts which penetrate the containment is minimized by a double barrier. This double barrier ensures that failure of a single active component along a leakage path will not result in loss of ability to isolate the containment. A barrier may be a valve, a blind flange (two barriers by use of a double gasket), or a closed piping system or vessel. The barriers must be missile protected, Seismic Class 1 and designed for a pressure equal to or greater than the design pressure of the containment. The isolation barriers are located as close to the containment penetration as practicable.

2.11.16 Containment Spray System (CSS)

The CSS (Figure 2.23) is an engineered safeguard designed to limit the peak pressure in the reactor containment building to a pressure less than the containment design pressure, in the event of a LOCA or a steam break accident inside the containment. The system also acts to remove airborne fission products (principally iodine) from the containment atmosphere should they be present due to a fuel cladding break.

The CSS achieves the above objectives by spraying a sodium hydroxide solution of borated water throughout a large volume portion of the containment atmosphere. The system consists of a containment spray storage tank (CSST), a spray additive tank (SAT), two pumps, two heat exchangers, and a set of spray ring headers located in the upper

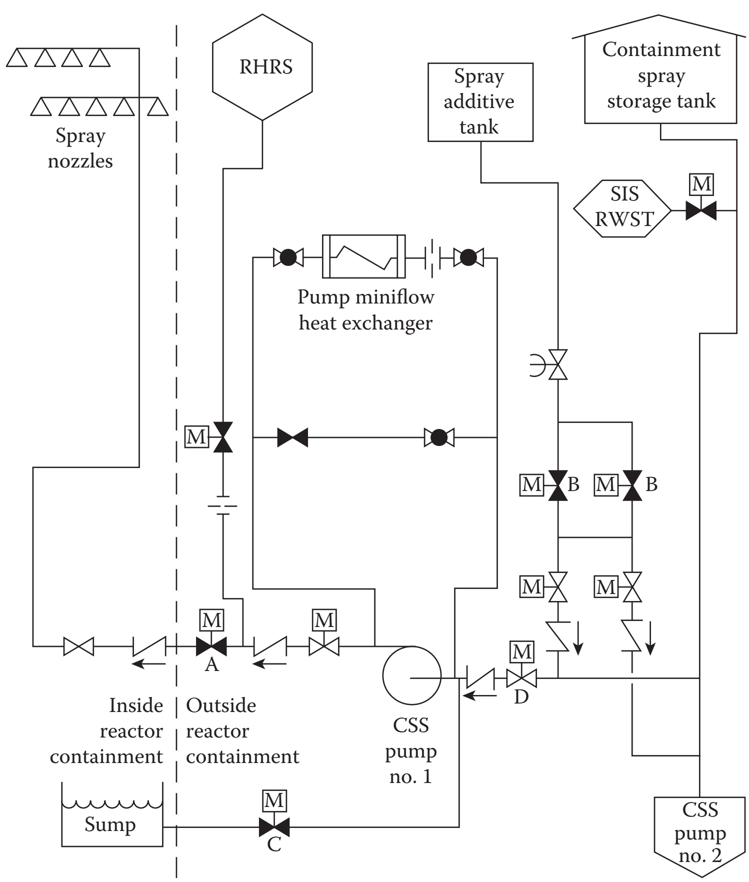


FIGURE 2.23
Containment spray system (one of two trains) flow diagram.

dome of the containment and having nozzles designed to provide adequate containment spray coverage. A pump mini-flow line, containing a heat exchanger and flow limiting orifice, and a test loop with a throttling valve are connected in parallel across each pump. The test line is open only when testing the pump at various flows. The pumps are normally aligned to take suction from the CSST and the SAT, which drain down together to provide the correct spray pH. Each one of these subsystems is also independently capable of delivering the necessary flow to limit containment pressure in conjunction with the Reactor Containment Fan Cooler (RCFC) System.

2.11.17 Initial Injection Mode

Containment spray is initiated automatically in response to a containment high pressure signal. This signal starts the spray pumps, opens the spray header isolation valves "A," and opens the SAT isolation valves "B." Suction is provided to the spray pump from the CSST, with the SAT providing sodium hydroxide solution. The SAT drains into each spray pump's suction header from the CSST; this mixing results in a solution pH that is conducive to the entrainment of iodine (airborne fission product). The spray pumps deliver this solution to the ring headers, and the solution is dispersed throughout the containment atmosphere by the spray nozzles. As this water comes in contact with the steam in the containment atmosphere, the steam is condensed and falls to the containment floor as drops of a borated sodium hydroxide solution. Also, as the spray is introduced to the containment atmosphere, the solution entrains airborne iodine particles that it contacts; these are then carried to the floor with the spray droplets. The storage tanks are sized to provide approximately 30 minutes of spray with spray additive, and when the CSST is drained to the "Lo-Lo" setpoint, the spray pumps are manually switched to the recirculation mode of spray operation. An additional function which is performed by the CSS during its normal operation is the pH adjustment of the containment sump. The sodium hydroxide (spray additive) delivered through the injection phase is sufficient to raise the long-term sump pH to a level adequate for inhibiting chloride stress corrosion/cracking of stainless steel components, and for keeping the iodine entrained in the water.

The recirculation mode of the CSS will be initiated manually by the operator. To accomplish this, the sump isolation valves "C" are opened, and the spray pump suction isolation valves "D" are closed. The pumps will then deliver spray from the sump to the spray headers. Spray additive is not injected during this mode.

2.11.18 RCFC System

The RCFC system is designed to remove heat from the containment building during normal operation and in the event of a LOCA. The RCFC is an engineered safeguard system. In a containment, the system includes four fan units that operate in parallel. A minimum of two units must function to satisfy design requirements for normal operation and post-accident operation. During normal operation, air is drawn from the upper part of the containment, through the return air ductwork, and continues through the normal flow inlet damper into the roughing filter plenum. This bypasses the HEPA filters and other components upstream of the roughing filter, which is mainly for the protection of the fan from foreign objects and minimizing the dust in the containment. The air continues its flow through the cooling coils and discharged by the fan into the ventilation system distribution ductwork.

During the post-LOCA operating mode, the air flow is routed from the return air ductwork, through the emergency flow inlet damper, and into the filtration package plenum. Here the air flows through moisture separators, HEPA filters, and discharges through the emergency flow outlet dampers into the roughing filter plenum and then into the cooling coils. When a LOCA is sensed, the fan motors are automatically switched to low speed to provide proper flow of the steam–air mixture. A gravity-actuated backdraft damper is installed in the ventilation system discharge ductwork of each fan. These dampers serve to isolate units from the ventilation system when the fan is not in use and to protect each unit from damage due to reverse flow during a LOCA pressure transient. The cooling coils remove heat from the air with the fan providing the required air flow rates. Cooling water is supplied by the essential service water system. Drain troughs and piping are provided to remove condensate humidity from the cooling coils. The drain piping is routed to the containment sump.

2.11.19 Hydrogen Control in Containment

Under accident conditions, coolant radioactivity and the Zirconium water reaction can result in the release of hydrogen gas to the containment atmosphere. Eventually the gas could accumulate to an explosive hydrogen/oxygen concentration. To control this potential risk the reactor containment is provided with:

- A hydrogen detection system that can detect and report in a continuous manner the hydrogen content in the containment atmosphere. Alarms at proper setpoints are provided.
- A containment ventilation system that circulates the containment atmosphere to remove pockets of high hydrogen concentration after an accident.
- Hydrogen recombiners that burn the free hydrogen. The hydrogen/oxygen combination results in water.

2.12 Instrumentation

This is an extremely detailed subject involving not only detection systems for radiation (or its surrogates: electronic currents, radioactive isotopes, etc.), but also control theory is involved, computer representations of data and their interpretation, and automatic systems that take over when events happen too fast for human intervention.

The Instrumentation and Control Systems can be viewed as the "central nervous system" of the plant and consist of the following major systems:

- *Nuclear Instrumentation System*—Provides continuous indications of the reactor core power level from shutdown to full power.
- *In-core Instrumentation System*—Senses the distribution of the nuclear flux within the core.
- *Digital Rod Position Indication System*—Detects the position of the control rods in the reactor core.

- *Process Instrumentation System*—Senses the state of the plant, when used together with the Nuclear Instrumentation, In-Core Instrumentation, and Digital Rod Position Indication System.
- *Nuclear Steam Supply Control Systems*—Implements the operator's control decisions and automatically changes the plant to and maintains it at selected operating states.
- *Reactor Protection System*—Protects the reactor core and the NSSS by monitoring operating parameters and initiating safeguards actions on the detection of abnormal conditions.
- *The Control Room*—Provides information to the operator to enable him/her to comprehend the plant's state and to make and implement operating decisions. Increasingly, ergonomic considerations have come to dominate the design of the layout: immediately in front of the operator are the controls and read-outs for normal operation. Up a little higher on the panel are instruments that show the effect of control manipulations, together with setpoint limits on key variables. Above that display are the alarms and signals of upset and accident conditions. All key instrumentation for each panel is within sight of the operator. Panels are arranged from the reactor and its auxiliaries, to the secondary systems, to the tertiary plant systems, and finally the electrical systems and their adjuncts. Arrangements are logical for normal power operation, shutdown and maintenance configurations, and emergency operations.
- *Plant Computer System*—Provides computational, data processing, and data presentation services for the plant. Flow maps and instrumentation diagrams may be called-up and data-logged to allow sequence analysis after events. Computers are especially valuable in the training center to model the reactor and all other systems, and to simulate upset and accident conditions that could not be conducted on the actual plant. This tool is also important in the development of emergency procedures.

2.13 Fuel Handling

Systems for fuel handling can vary widely. In reactor sites on which multiple plants are located, a single storage pool for spent fuel may be most convenient. Where long term storage is necessary, a large pool, and dry storage pads for shielded containers may be used for spent fuel. After the radioactivity of the fuel has died down for several years, it may be stored dry in vertical concrete vaults that allow air circulation while preventing people and small animals from being exposed.

2.13.1 Spent Fuel Handling

Spent fuel is handled underwater from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site, or placed in dry storage. The water provides an effective, economical, and transparent radiation shield as well as a reliable medium for decay heat removal. Boric acid is added to the water to ensure subcritical conditions during refueling. The fuel handling facilities are generally divided into two areas:

(1) The refueling cavity and fuel transfer canal, which are flooded only during the refueling shutdown.
(2) The fuel storage pit, which is kept full of water and is always accessible to operating personnel.

These two areas are connected by the fuel transfer tube through which an underwater conveyor or fuel transfer system carries the new fuel into the reactor containment and spent fuel into the fuel handling building.

In the refueling cavity, fuel is removed from the reactor vessel by a refueling machine, transferred through the water, and placed in the fuel transfer system. In the fuel handling building, the fuel is removed from the transfer carriage and placed in storage racks in the fuel storage pit by using a manually operated spent fuel assembly handling tool suspended from an overhead bridge crane. After a sufficient decay period, the spent fuel can be removed from the fuel racks and loaded into a shipping cask for removal from pool into dry storage, or shipped from the site. The shipping cask is inspected for internal or external contamination, dirt, etc before being lowered onto a special pad at the end of the pool near the truck door of the fuel handling building. The cask lid is removed and the spent fuel assemblies are placed therein, the lid replaced, the cask raised and drained, the lid fastened down, and the cask is loaded onto the truck.

2.13.2 New Fuel Handling

New fuel assemblies normally arrive at the site shortly before refueling is to commence. Following depressurization of the container and site receipt inspection to check for any possible shipping distortion or damage, new assemblies are attached to the "new" (short-handled) fuel handling tool and are lowered into the fuel storage pit by means of the new fuel elevator and are placed underwater in the storage racks. During the refueling operation, the new assemblies are transported one at a time from their storage locations in the pit to the fuel transfer system by means of the fuel handling machine and the "spent" (long-handled) fuel handling tool.

2.14 Waste Handling

Waste Processing Systems, which can vary greatly from plant to plant, process liquid, gaseous, and solid plant effluents during power operation and plant shutdowns. The systems consist of the Liquid Waste Processing System, the Waste Gas Processing System, and the Solid Waste Processing Systems. The Liquid Waste Processing System is designed to collect, process, monitor, and recycle for reuse the liquid waste effluents generated during various plant operations. The Waste Gas Processing System stores waste gases for fission product decay and eventual release. Wastes which cannot be recycled and must be disposed of safely are volume reduced and packaged for disposal by the Solid Waste Systems.

2.14.1 Liquid Waste Processing

The Liquid Waste Processing System is provided for use in the processing and handling of radioactive wastes generated during various modes of plant operation. The system is designed to receive, segregate, process, monitor, and recycle for reuse all primary system waste effluents. The system is designed so that tritium-containing water can be segregated from nontritium-containing water and includes a separate laundry waste treatment system, thereby allowing for easy tritium control. In addition, provisions are made to handle spent regenerant chemicals from condensate polishers and waste from decontamination of spent fuel shipping casks. The system consists of several waste holdup and collection tanks, corresponding pumps, waste evaporators, demineralizers, filters, monitors, and associated piping. Liquid wastes from floor drains, laundry, hot showers, laboratory rinses, equipment drains, and the like are collected in their respective holdup tanks. After collection, some wastes are processed for disposal; most are processed through the waste evaporators or the reverse-osmosis package (separation of pure water from dissolved salts). Sample analysis is performed and the wastes are either recycled for reuse or disposed of safely.

The system design incorporates features specifically aimed at minimizing the environmental impact of plant operation. In addition, the design provides adequate processing capacity to accommodate unforeseen occurrences of high liquid waste leakage. Sufficient capacity, redundancy, and flexibility provide a wide range of operability.

2.14.2 Gaseous Waste Processing

The Gaseous Waste Processing System receives noble fission gases which have been stripped from the reactor coolant through the use of hydrogen gas as a carrier. The stripping of the fission gases by hydrogen reduces the fission gas concentration in the reactor coolant to a low residual level. This minimizes the release of radioactive gases during maintenance operations on the RCS or through unavoidable equipment leaks in the RCS. The system consists of a waste gas dryer, charcoal adsorption beds, guard beds, a charcoal fines filter, a surge tank, and recycle line compressors. The hydrogen carrier gas from the VCT (part of the CVCS) first enters the refrigerated waste gas dryer which cools the hydrogen purge stream and condenses and removes the water vapor. The dried gas then flows to the guard beds which protect the charcoal adsorption beds from water contamination. The flow is then routed through the charcoal adsorption tanks where the noble fission gases contained in the waste stream are absorbed. Xenon-133 is delayed for 60 days, after which time the concentration of xenon in the exiting hydrogen stream is negligible. The system is designed to also delay krypton-85 for three days. The hydrogen carrier gas experiences no delay; it passes through the charcoal beds to the plant vent. The recycle line compressors provide the system with the capability to process gases from other areas of the plant by routing the gases to the VCT for processing. In addition, the compressors may be used for hydrogen gas recycle where its use is economically justified.

2.14.3 Solid Waste Processing

The Solid Waste Processing Systems include the Radwaste Volume Reduction/Solidification System and the Radwaste Incinerator. The systems reduce in volume and solidify low-level radioactive plant wastes to prepare them for safe storage and/or disposal. The Radwaste Volume Reduction/Solidification System employs a vacuum-cooled crystallization process to effect volume reduction, coupled with high speed, higher shear mixing of the waste with cement to achieve solidification. For combustible plant wastes, the Radwaste incinerator utilizes a controlled air incineration process.

2.14.4 Radwaste Volume Reduction

Volume reduction of concentrated evaporator bottoms, which may include boric acid wastes, laundry wastes, chemical wastes, and other floor drain wastes, is accomplished in the Radwaste Volume Reduction System. The major components of the system are the crystallizer chamber and recirculation system, condenser, and vacuum pump system. The crystallizer chamber consists of a conical tank and an inner circular baffle to separate solid crystals from a clear recycle stream. Combustible wastes such as clothing, filter cartridges, and wood, are volume-reduced in the Radwaste Incinerator. Solidification of volume-reduced wastes and other low-level radioactive wastes, such as spent resins and contaminated tools, is performed in the Cement Solidification System. The major components of the Cement Solidification System include the high shear Radwaste mixer, waste dispensing system, flushwater recycle steam, cement storage and feed system, and the container handling system.

2.15 Advanced Passive Reactor

The year 2006 was a seminal one for the design of a new generation of PWRs. The Westinghouse design of the AP-1000 was one of the first certified. It was initially approved by the Nuclear Regulatory Commission (NRC) for a lower power model, the AP-600, when it seemed that revitalization of the nuclear industry would center around smaller units that were expected to come online sooner than the larger-capacity plants. They would also appeal to smaller utilities who would not want too-large a share of their total capacity dominated by any one plant. In the same year that new designs were being announced, many mergers were taking place that changed the vendor array available to the world's utilities. These changes were discussed earlier in this section and in Section 1.1. Discussion of these other new designs by other organizations is contained in the handbook sections after this one. At the same time as new vendor organizations were developing, several utilities formed consortia to operate the existing and future nuclear plants. These consortia are not under the control of state public utility commissions (PUCs). Consequently, individual utilities under PUC control do not risk having their investment in nuclear power arbitrarily declared "imprudent investment," thus disallowing the costs to be recovered in the rate base. The consortia are under the control of the NRC, as well as the Federal Energy Regulatory Commission (FERC) because electric power is often sold across state lines. Increasing numbers of interties between systems and "wheeling" of power across the country have greatly increased the reliability of the nation's electrical distribution system. Other innovations on the horizon include cryogenic (low resistance losses) and direct current (DC) long-distance transmission lines (possibly underground) that produce a negligible radiation field.

2.15.1 New PWR Designs

For the convenience of the reader, the AP-600 and AP-1000 are compared with the largest conventional reactor design that preceded them. The comparison of key parameters is given in Table 2.4. Following that is a detailed discussion of changes in the design from that described above in earlier subsections. Much has been done to improve neutron economy, system and component reliability. Also, equipment design and layout have been reworked to reduce cost and simplify plant operation. Finally, dramatic changes have been made in Safeguards Systems to provide for passive measures that would operate after a severe accident, assuming no operator action is possible in the containment vessel for a considerable period of time.

Figure 2.24 is a schematic of the RCS. Table 2.5 provides the principal pressures, temperatures, and flow rates of the system at the locations noted in Figure 2.24 under

TABLE 2.4

Reactor Design Comparison Table

Thermal and Hydraulic Design Parameters	AP1000	AP600	Typical XL Plant
Reactor core heat output (MWt)	3400	1933	3800
Reactor core heat output (10^6 Btu/hr)	11,601	6596	12,969
Heat generated in fuel (%)	97.4	97.4	97.4
System pressure nominal (psia)	2250	2250	2250
System pressure, minimum steady-state (psia)	2190	2200	2204
Minimum departure from nuclear boiling (DNBR)			
Typical flow channel	>1.25[d], >1.22[d]	>1.23	>1.26
Thimble (cold wall) flow channel	>1.25[d], >1.21[d]	>1.22	>1.24
Departure from nucleate boiling (DNB) correlation[b]	WRB-2M[b]	WRB-2	WRB-1[a]
Coolant Flow[c]			
Total vessel thermal design flow rate (10^6 lbm/hr)	113.5	72.9	145.0
Effective flow rate for heat transfer (10^6 lbm/hr)	106.8	66.3	132.7
Effective flow area for heat transfer (ft^2)	41.5	38.5	51.1
Average velocity along fuel rods (ft/s)	15.9	10.6	16.6
Average mass velocity (10^6 lbm/hr-ft^2)	2.41	1.72	2.60
Coolant Temperature			
Nominal inlet (°F)	535.0	532.8	561.2
Average rise in vessel (°F)	77.2	69.6	63.6
Average rise in core (°F)	81.4	75.8	68.7
Average in core (°F)	578.1	572.6	597.8
Average in vessel (°F)	573.6	567.6	593.0
Heat Transfer			
Active heat transfer surface area (ft^2)	56,700	44,884	69,700
Average heat flux (BTU/hr-ft^2)	199,300	143,000	181,200
Maximum heat flux for normal operation (BTU/hr-ft^2)[f]	518,200	372,226	489,200
Average linear power (kW/ft)[g]	5.72	4.11	5.20
Peak linear power for normal operation (kW/ft)[f, g]	14.9	10.7	14.0
Peak linear power (kW/ft)[f, h] (resulting from overpower transients/operator errors, assuming a maximum overpower of 118%)	≤22.45	22.5	≤22.45
Heat flux hot channel factor (F_Q)	2.60	2.60	2.70
Peak fuel centerline temperature (°F) (for prevention of centerline melt)	4700	4700	4700

(*Continued*)

TABLE 2.4 (Continued)

Thermal and Hydraulic Design Parameters	AP1000	AP600	Typical XL Plant
Fuel assembly design	17 × 17 XL Robust Fuel	17 × 17	17 × 17 XL Robust Fuel/ No IFM
Number of fuel assemblies	157	145	193
Uranium dioxide rods per assembly	264	264	264
Rod Pitch (in.)	0.496	0.496	0.496
Overall dimensions (in.)	8.426 × 8.426	8.426 × 8.426	8.426 × 8.426
Fuel weight as uranium dioxide (lb)	211,588	167,360	261,000
Clad weight (lb)	43,105	35,555	63,200
Number of grids per assembly			
Top and bottom (Ni–Cr–Fe Alloy 718)	2[i]	2[i]	2
Intermediate	8 ZIRLO™	7 Zircaloy-4 or 7 ZIRLO™	8 ZIRLO™
Intermediate flow mixing	4 ZIRLO™	4 Zircaloy-4 or 5 ZIRLO™	0
Loading technique, first cycle	3 region nonuniform	3 region nonuniform	3 region nonuniform
Fuel Rods			
Number	41,488	38,280	50,952
Outside diameter (in.)	0.374	0.374	0.374
Diametral gap (non-IFBA) (in.)	0.0065	0.0065	0.0065
Clad thickness (in.)	0.0225	0.0225	0.0225
Clad material	ZIRLO™	Zircaloy-4 or ZIRLO™	Zircaloy-4/ ZIRLO™
Fuel Pellets			
Material	UO_2 sintered	UO_2 sintered	UO_2 sintered
Density (% of theoretical)	95.5	95	95
Diameter (in.)	0.3225	0.3225	0.3225
Length (in.)	0.387	0.387	0.387
Neutron Absorber			
RCCA	24 Ag–In–Cd rodlets	24 Ag–In–Cd rodlets	24 Hafnium or Ag–In–Cd
GRCA	20 304 SS rodlets 4 Ag–In–Cd rodlets	20 304 SS rodlets 4 Ag–In–Cd rodlets	
Cladding material	Type 304 SS, cold-worked	Type 304 SS, cold-worked	Type 304 SS, cold-worked
Clad thickness (Ag–In–Cd)	0.0185	0.0185	0.0185
Number of clusters	53 RCCAs 16 GRCAs	45 RCCAs 16 GRCAs	57 RCCAs 0 GRCAs
Core Structure			
Core barrel, ID/OD (in.)	133.75/137.75	133.75/137.75	148.0/152.5
Thermal shield	None	None	Neutron Panel
Baffle thickness (in.)	Core Shroud	Radial reflector	0.875

(*Continued*)

TABLE 2.4 (Continued)

Thermal and Hydraulic Design Parameters	AP1000	AP600	Typical XL Plant
Structure Characteristics			
Core diameter, equivalent (in.)	119.7	115.0	132.7
Core height, cold, active fuel (in.)	168.0	144.0	168.0
Fuel Enrichment First Cycle (Weight Percent)			
Region 1	2.35	1.90	Typical
Region 2	3.40	2.80	3.8 to 4.4
Region 3	4.45	3.70	(5.0 max)

[a] WRB-2M will be used in future reloads.
[b] See subsection 4.4.2.2.1 of reference AP-1000 Design Control Document for the use of the W-3, WRB-2 and WRB-2M correlations.
[c] Flow rates and temperatures are based on 10% steam generator tube plugging for the AP-600 and AP-1000 designs.
[d] 1.25 applies to core and axial offset limits; 1.22 and 1.21 apply to all other RTDP transients.
[e] Coolant temperatures based on thermal design flow (for AP-600 and AP-1000).
[f] Based on F_Q of 2.60 for AP-600 and AP-1000.
[g] Based on densified active fuel length.
[h] See subsection 4.3.2.2.6 of reference AP-1000 Design Control Document.
[i] The top grid may be fabricated of either nickel-chromium-iron Alloy 718 or ZIRLO™.

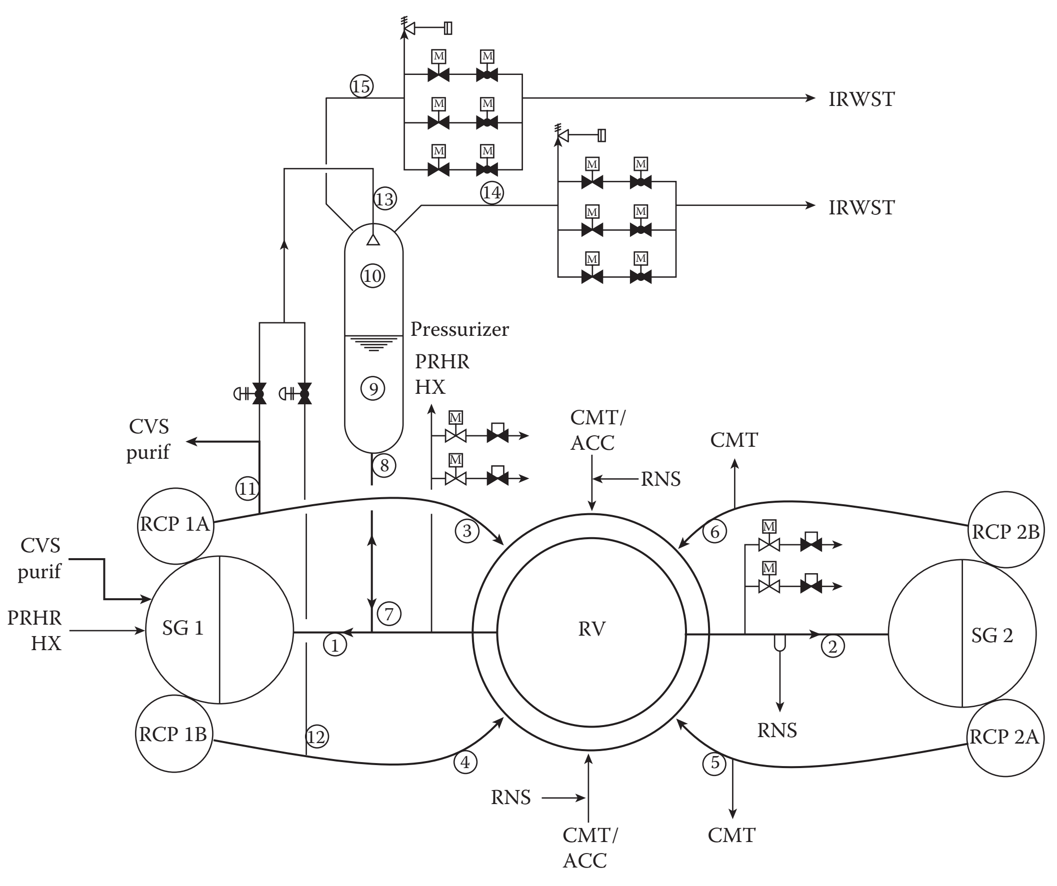


FIGURE 2.24
AP-1000 reactor coolant system schematic flow diagram.

TABLE 2.5

Principal System Pressures, Temperatures and Flow Rates (Nominal Steady-State, Full Power Operating Conditions) Location Numbers are Identified on Figure 2.24

Location	Description	Field	Pressure (psig)	Nominal Temp (°F)	Flow[a] (gpm)
1	Hot Leg 1	Reactor Coolant	2248	610	177,645
2	Hot Leg 2	Reactor Coolant	2248	610	177,645
3	Cold Leg 1A	Reactor Coolant	2310	537.2	78,750
4	Cold Leg 1B	Reactor Coolant	2310	537.2	78,750
5	Cold Leg 2A	Reactor Coolant	2310	537.2	78,750
6	Cold Leg 2B	Reactor Coolant	2310	537.2	78,750
7	Surge Line Inlet	Reactor Coolant	2248	610	–
8	Pressurizer Inlet	Reactor Coolant	2241	653.0	–
9	Pressurizer Liquid	Reactor Coolant	2235	653.0	–
10	Pressurizer Steam	Steam	2335	653.0	–
11	Pressurizer Spray 1A	Reactor Coolant	2310	537.2	1–2
12	Pressurizer Spray 1B	Reactor Coolant	2310	537.2	1–2
13	Common Spray Line	Reactor Coolant	2310	537.2	2–4
14	ADS Valve Inlet	Steam	2235	653.0	–
15	ADS Valve Inlet	Steam	2235	653.0	–

[a] At the conditions specified.

normal steady-state, full-power operating conditions. These parameters are based on the best-estimate flow at the pump discharge. Table 2.6 contains a summary of nominal system design and operating parameters under normal steady-state, full-power operating conditions. These parameters are based on the best-estimate conditions at nominal full power. The RCS volume under these conditions is also provided. The RCS consists of two heat-transfer circuits, each with a steam generator, two RCPs, and a single hot leg and two cold legs for circulating reactor coolant. In addition, the system includes the pressurizer, interconnecting piping, valves, and instrumentation for operational control and safeguards actuation. All RCS equipment is located in the reactor containment. During operation, the RCPs circulate pressurized water through the reactor vessel then the steam generators. Water serves as coolant, moderator, and solvent for boric acid (chemical shim control) and is heated as it passes through the core. It is transported to the steam generators where the heat is transferred to the steam system. Then it is returned to the reactor vessel by the pumps to repeat the process like all previous designs. Spring-loaded safety valves are installed above and connected to the pressurizer to provide overpressure protection for the RCPs. These valves discharge into the containment atmosphere. Three stages of RCS automatic depressurization valves are also connected to the pressurizer. These valves discharge steam and water through spargers to the in-containment refueling water storage tank (IRWST) of the passive core cooling system (PXS). This is an entirely new system. On older designs, the RWST was located outside of containment. Most (initially all) of the steam and water discharged to the spargers is condensed and cooled by mixing with the water in the tank. The fourth-stage automatic depressurization valves are connected by two redundant paths to each reactor coolant loop hot leg and discharge directly to the containment atmosphere. The RCS is also served by several auxiliary systems, including the CVCS, PXS, the normal residual heat removal system (RNS), the steam generator system (SGS), the primary sampling system (PSS), the liquid Radwaste system (WLS), and the

TABLE 2.6

Nominal System Design and Operating Parameters for the AP1000

General	
Plant design objective, years	60
NSSS power, MWt	3415
Reactor coolant pressure, psia	2250
Reactor coolant liquid volume at power conditions (including 1000 ft^3 pressurizer liquid), ft^3	9600
Loops	
Number of cold legs	4
Number of hot legs	2
Hot leg ID, in.	31
Cold leg ID, in.	22
Reactor Coolant Pumps	
Type of reactor coolant pumps	Canned-motor
Number of reactor coolant pumps	4
Nameplate motor rating, hp	7000
Effective pump power to coolant, MWt	15
Pressurizer	
Number of units	1
Total volume, ft^3	2100
Water volume, ft^3	1000
Spray capacity, gpm	500
Inside diameter, in.	90
Height, in.	607
Steam Generator	
Steam generator power, MWt/unit	1707.5
Type	Vertical U-tube Feedring-type
Number of units	2
Surface area, ft^2/unit	123,540
Shell design pressure, psia	1200
Zero load temperature, °F	557
Feedwater temperature, °F	440
Exit steam pressure, psia	836
Steam flow, lb/hr per steam generator	7.49×10^6
Total steam flow, lb/hr	14.97×10^6

CCWS. With the exception of the PXS, these connections are similar to previous designs. The safeguards system in previous designs was called the SIS.

The design of the AP1000 reactor vessel closely matches the existing vessel designs of Westinghouse three-loop plants. New features have been incorporated without departing from the proven features of existing vessel designs. The vessel has inlet and outlet nozzles positioned in two horizontal planes between the upper head flange and the top of the core. The nozzles are located in this configuration to provide an acceptable cross-flow velocity in the vessel outlet region and to facilitate optimum layout of the RCS equipment. The inlet and outlet nozzles are offset, with the inlet positioned above the outlet, to allow mid-loop operation for removal of a main coolant pump without discharge of the core.

RCS and steam system overpressure protection during power operation are provided by the pressurizer safety valves and the steam generator safety valves, in conjunction with the action of the reactor protection system. Combinations of these systems provide compliance with the overpressure protection requirements of the NRC for pressurized water reactor (PWR) systems. Low temperature overpressure protection is provided by a relief valve in the suction line of the RNS system. The sizing and use of the relief valve for low temperature overpressure protection is also consistent with government guidelines.

The new design by AREVA-NP is called the Evolutionary Power Reactor (EPR). It is going through the approval process. General features include:

- Airplane crash resistance
- Core melt catcher
- Quadruple redundancy with independent trains for each safeguard system
- Lowered probability of core damage
- Double containment shell
- Accident consequences limited by an optimal combination of passive and active safety systems
- Maximum power output per site
- Lower natural uranium consumption per MWh
- Reduced long-lived high-level waste generation during operation
- Capability to use mixed uranium and plutonium oxide fuel from recycled used fuel
- Minimal quantities of high-activity metal structure
- Less water usage than present operating reactors thanks to improved thermal efficiency
- Thermal power will be 4250 MW or 4500 MW
- Electrical power will be approximately 1600 MW or 36% efficiency
- There will be four primary loops like most current generation PWRs
- There will be 241 fuel assemblies operating to >60 GWd/tonne of oxide fuel
- Steam will be produced at a pressure of 78 Bar
- The design seismic limit is 0.25 g
- The service life of the plant is expected to be 60 years

2.15.2 Chemical Control of the Coolant System

Returning to the Westinghouse AP reactor design, the primary coolant system water chemistry is selected to minimize corrosion. Routinely scheduled analyses of the coolant chemical composition are performed to verify that the reactor coolant chemistry meets the specifications. Other additions, such as those to reduce activity transport and deposition, may be added to the system. The CVCS provides a means for adding chemicals to the RCS. The chemicals perform the following functions:

- Control the pH of the coolant during prestartup testing and subsequent operation
- Scavenge oxygen from the coolant during heatup

- Control radiolysis reactions involving hydrogen, oxygen, and nitrogen during power operations following startup

The pH control chemical is lithium hydroxide, enriched in the lithium-7 isotope to 99.9%. This chemical is chosen for its compatibility with the materials and water chemistry of borated water/stainless steel/zirconium/nickel–chromium–iron systems. In addition, lithium-7 is produced in solution from the neutron irradiation of the dissolved boron in the coolant. The lithium-7 hydroxide is introduced into the RCS via the charging flow. The concentration of lithium-7 hydroxide in the RCS is maintained in the range specified for pH control. The other major isotope of lithium, Li-6, is a strong neutron absorber. The concentration of isotope Li-6 in naturally occuring lithium is much higher than 0.1% and consequently would interfere with the chemical shim (Boron-10) system. During reactor startup from the cold condition, hydrazine is used as an oxygen-scavenging agent. The hydrazine solution is introduced into the RCS in the same manner as described for the pH control agent. The reactor coolant is treated with dissolved hydrogen to control the net decomposition of water by radiolysis in the core region. The hydrogen reacts with oxygen introduced into the RCS by the radiolysis effect on water molecules. Hydrogen makeup is supplied to the RCS by direct injection of high-pressure gaseous hydrogen, which can be adjusted to provide the correct equilibrium hydrogen concentration. Boron in the form of boric acid is added to the RCS for long-term reactivity control of the core. Suspended solid (corrosion product particulates) and other impurity concentrations are maintained below specified limits by controlling the chemical quality of makeup water and chemical additives and by purification of the reactor coolant through the CVCS.

2.15.3 RCP

The RCP is a single stage, hermetically sealed, high-inertia, centrifugal canned-motor pump. It pumps large volumes of reactor coolant at high pressures and temperature. Figure 2.25 shows the RCP. Table 2.7 gives the RCP design parameters. A RCP is directly connected to each of two outlet nozzles on the steam generator channel head. The two pumps on a steam generator turn in the same direction. A canned motor pump contains the motor and all rotating components inside a pressure vessel. The pressure vessel consists of the pump casing, thermal barrier, stator shell, and stator cap, which are designed for full RCS pressure. The stator and rotor are encased in corrosion-resistant cans that prevent contact of the rotor bars and stator windings by the reactor coolant. Because the shaft for the impeller and rotor is contained within the pressure boundary, seals are not required to restrict leakage out of the pump into containment. A gasket and canopy seal type connection between the pump casing, the stator flange, and the thermal barrier is provided. This design provides definitive leak protection for the pump closure. To access the internals of the pump and motor, the canopy seal weld is severed. When the pump is reassembled a canopy seal is re-welded. Canned-motor RCPs have a long history of safe, reliable performance in military and commercial nuclear plant service. The RCP driving motor is a vertical, water-cooled, squirrel-cage induction motor with a canned rotor and a canned stator. It is designed for removal from the casing for inspection, maintenance and replacement (if required). The stator can protects the stator (windings and insulation) from the controlled portion of the reactor coolant circulating inside the motor and bearing cavity. The can on the rotor isolates the copper rotor bars from the system and minimizes the potential for the copper to plate out in other areas. The motor is cooled by component cooling water circulating through a cooling jacket on the outside of the motor housing and

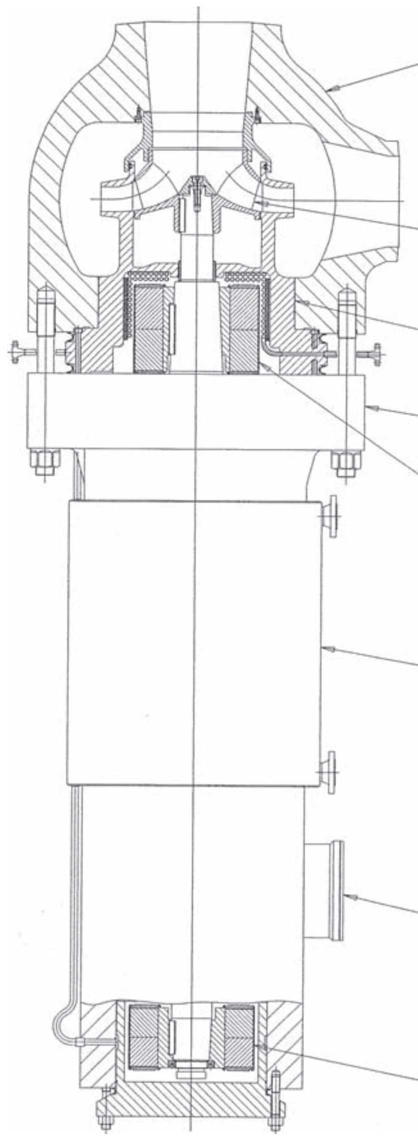


FIGURE 2.25
AP-1000 reactor coolant pump.

through a thermal barrier between the pump casing and the rest of the motor internals. Inside the cooling jacket are coils filled with circulating rotor cavity coolant. This rotor cavity coolant is a controlled volume of reactor coolant that circulates inside the rotor cavity. After the rotor cavity coolant is cooled in the cooling jacket, it enters the lower end of the rotor and passes axially between the rotor and stator cans to remove heat from the rotor and stator. Each pump motor is driven by a variable speed drive, which is used for pump startup and operation when the reactor trip breakers are open. When the reactor trip breakers are closed, the variable frequency drives are bypassed and the pumps run at constant speed. A flywheel, consisting of two separate assemblies, provides rotating inertia

TABLE 2.7

Reactor Coolant Pump Design Parameters

Unit design pressure (psia)	2500
Unit design temperature (°F)	650
Unit overall length, (ft-in)	21–11.5
Component cooling water flow (gpm)	600
Maximum continuous component cooling water inlet temperature (°F)	95
Total weight motor and casing, dry (lb) nominal	184,500
Pump	
Design flow (gpm)	78,750
Developed head (feet)	365
Pump discharge nozzle, inside diameter (inches)	22
Pump suction nozzle, inside diameter (inches)	26
Speed (synchronous)(rpm)	1800
Motor	
Type	Squirrel Cage Induction
Voltage (V)	6900
Phase	3
Frequency (Hz)	60
Insulation class	Class H or N
Current (amp)	
Starting	Variable
Nominal input, cold reactor coolant	Variable
Motor/pump rotor minimum required moment of inertia (lb-ft^2)	16,500

that increases the coastdown time for the pump. Each flywheel assembly is a composite of a uranium alloy flywheel casting or forging contained within a welded nickel–chromium–iron alloy enclosure. The upper flywheel assembly is located between the motor and pump impeller. The lower assembly is located within the canned motor below the thrust bearing. Surrounding the flywheel assemblies are the heavy walls of the motor end closure, casing, thermal barrier flange, stator shell, or main flange.

2.15.4 Steam Generator

The steam generator channel head, tubesheet, and tubes are a portion of the RCP boundary. The tubes transfer heat to the steam system while retaining radioactive contaminants in the primary system. The steam generator removes heat from the RCS during power operation and anticipated transients and under natural circulation conditions. The steam generator heat transfer function and associated secondary water and steam systems are not required to provide a safety-related safe shutdown of the plant. The steam generator secondary shell functions as containment boundary during operation and during shutdown when access opening closures are in place. The AP1000 steam generator is a vertical-shell U-tube evaporator with integral moisture separating equipment. Figures 2.26 and 2.27 shows the steam generator, indicating several of its design features. The design of the Model Delta-125 steam generator, except for the configuration of the channel head, is similar to an upgraded Model Delta-75 steam generator. The Delta-75 steam generator has been placed in operation as a replacement steam generator in older design

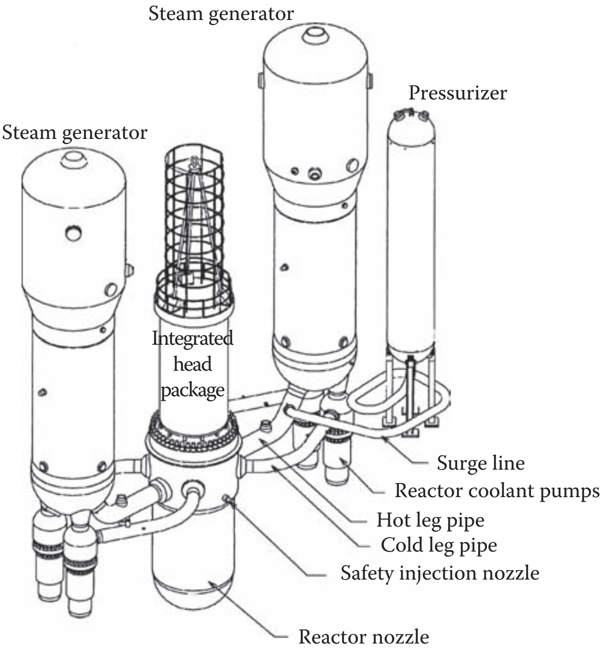


FIGURE 2.26
AP-1000 primary system showing two rather than four steam generators.

plants. Steam generator design features are described in the following paragraphs. The steam generator design requirements are listed in Table 2.8 and the design parameter are listed in Table 2.9.

On the primary side, the reactor coolant flow enters the primary chamber via the hot leg nozzle. The lower portion of the primary chamber is elliptical and merges into a cylindrical portion, which mates to the tubesheet. This arrangement provides enhanced access to all tubes, including those at the periphery of the bundle, with robotics equipment. This feature enhances the ability to inspect, replace and repair portions of the AP1000 unit compared with the more spherical primary chamber of earlier designs. The head is divided into inlet and outlet chambers by a vertical divider plate extending from the apex of the head to the tubesheet. The reactor coolant flow enters the inverted U-tubes, transferring heat to the secondary side during its traverse, and returns to the cold leg side of the primary chamber. The flow exits the steam generator via two cold leg nozzles to which the canned-motor reactor coolant pumps are directly attached. A high-integrity, nickel–chromium–iron (Alloy 690) weld is made to the nickel–chromium–iron alloy- buttered ends of these nozzles.

A passive residual heat removal (PRHR) nozzle attaches to the bottom of the channel head of the loop-1 steam generator on the cold leg portion of the head. This nozzle provides recirculated flow from the PRHR heat exchanger to cool the primary side under emergency conditions. A separate nozzle on one of the steam generator channel heads is connected to a line from the CVCS. The nozzle provides for purification flow and makeup flow from the CVCS to the RCS.

The AP1000 steam generator channel head has provisions to drain the head. To minimize deposits of radioactive corrosion products on the channel head surfaces and to enhance the decontamination of these surfaces, the channel head cladding is machined

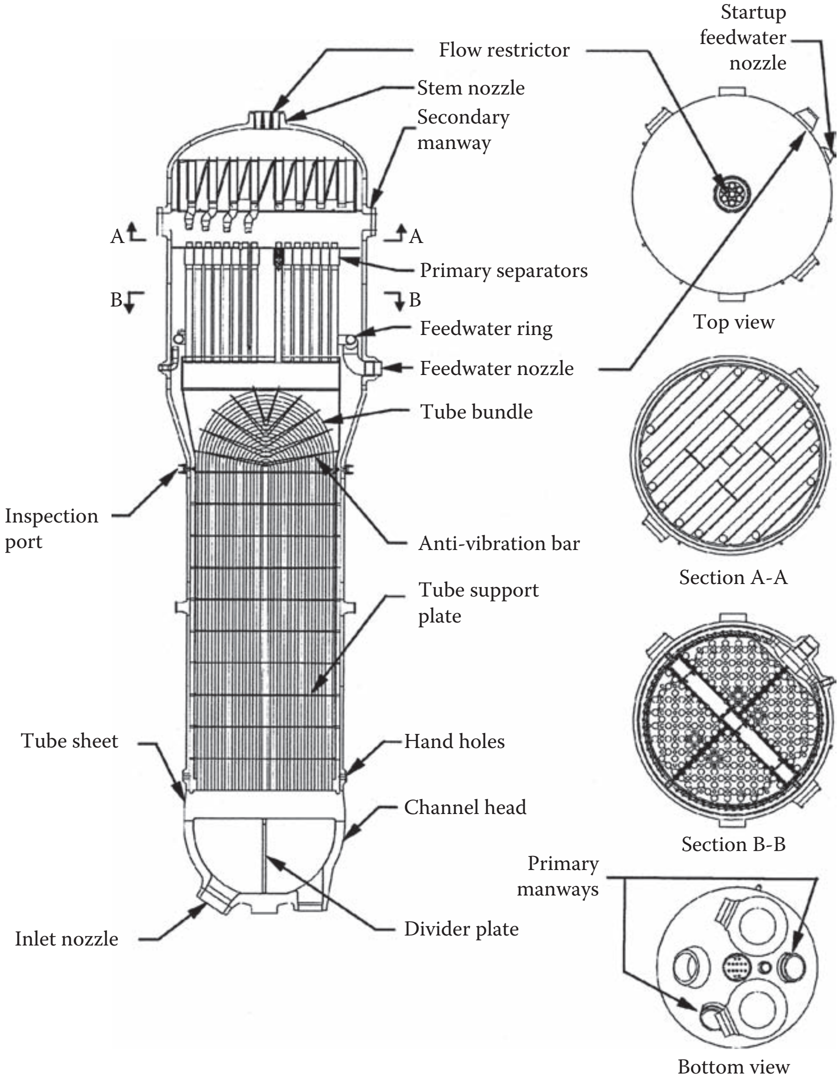


FIGURE 2.27
AP-1000 steam generator details.

or electropolished for a smooth surface. The primary manways provide enhanced primary chamber access compared with previous model steam generators. Should steam generator replacement using a channel head cut be required, the arrangement of the AP1000 steam generator channel head facilitates steam generator replacement in two ways. It is completely unobstructed around its circumference for mounting cutting equipment, and is long enough to permit post-weld heat treatment with minimal effect of tubesheet acting as a heat sink. The tubes are fabricated of nickel–chromium–iron Alloy 690. The tubes undergo thermal treatment following tube-forming operations. The tubes are tack-expanded, welded, and hydraulically expanded over the full depth of the tubesheet. Westinghouse has used this practice in F-type steam generators. It was selected because of its capability to minimize secondary water access to the

TABLE 2.8
Steam Generator Design Requirements

Type	Vertical U-Tube Feedring-Type
Design pressure, reactor side (psia)	2500
Design Pressure, steam side (psia)	1200
Design pressure, primary to secondary (psi)	1600
Design temperature, reactor coolant side (°F)	650
Design temperature, steam side (°F)	600
S/G Power, MWt/unit	1707.5
Total heat transfer surface area (ft^2)	123,538
Steam nozzle outlet pressure (psia)	836
Steam flow, lb/hr per S/G	7.49×10^6
Total steam flow, lb/hr	14.97×10^6
Maximum moisture carryover (weight percent) maximum	0.25
No load temperature, °F	557
Feedwater temperature, °F	440
Number of tubes per unit	10,025
Tube outer diameter, inch	0.688
Total wall thickness, inch	0.040
Tube pitch, inches	0.980 (triangular)

TABLE 2.9
Steam Generator Nominal Design Parameters

Tube pitch (in.)	0.980 (triangular)
Overall length (in.)	884.26[a]
Upper shell I.D. (in.)	210
Lower shell I.D. (in.)	165
Tubesheet thickness (in.)	31.13[b]
Primary water volume (ft^3)	2077
Water volume in tubes (ft^3)	1489
Water volume in plenums (ft^3)	588
Secondary water volume (ft^3)	3646
Secondary steam volume (ft^3)	5222
Secondary water mass (lbm)	175,758
Design fouling factor (hr-°F-ft^2/BTU)	1.1×10^{-4}

[a] Measured from steam nozzle to the flat, exterior portion of the channel head.
[b] Base metal thickness.

tube-to-tubesheet crevice. Residual stresses smaller than from other expansion methods result from this process and are minimized by tight control of the pre-expansion clearance between the tube and tubesheet hole. Support of the tubes is provided by ferritic stainless steel tube support plates. The holes in the tube support plates are broached with a hole geometry to promote flow along the tube and to provide an appropriate interface between the tube support plate and the tube. Anti-vibration bars installed in the U-bend portion of the tube bundle minimize the potential for excessive vibration. Steam is generated on the shell side, flows upward, and exits through the outlet nozzle at the top of the vessel. Feedwater enters the steam generator at an elevation above the top of the U-tubes through a feedwater nozzle. The feedwater enters a feedring via a welded thermal sleeve connection and leaves it through nozzles attached to the top of the feedring. The nozzles are fabricated of an alloy that is very resistant to erosion and corrosion with the expected secondary water chemistry and flow rate through the nozzles. After exiting the nozzles, the feedwater flow mixes with "saturated" (at the boiling point) water removed by the moisture separators. The flow then enters the downcomer annulus between the wrapper and the shell.

Fluid instabilities and water hammer phenomena are important considerations in the design of steam generators. Water-level instabilities can occur from density wave instabilities which could affect steam generator performance. Density wave instability is avoided in the AP1000 steam generator by including appropriate pressure losses in the downcomer and the risers that lead to negative damping factors. Steam generator bubble collapse water hammer has occurred in certain early PWR steam generator designs having feedrings equipped with bottom discharge holes. Prevention and mitigation of feedline-related water hammer has been accomplished through an improved design and operation of the feedwater delivery system. The AP1000 steam generator and feedwater system incorporate features designed to eliminate the conditions linked to the occurrence of steam generator water hammer. The steam generator features include introducing feedwater into the steam generator at an elevation above the top of the tube bundle and below the normal water level by a top discharge feedring. The top discharge of the feedring helps to reduce the potential for vapor formation in the feedring. This minimizes the potential for conditions that can result in water hammer in the feedwater piping. The feedwater system features designed to prevent and mitigate water hammer include a short, horizontal or downward sloping feedwater pipe at steam generator inlet. These features minimize the potential for trapping pockets of steam which could lead to water hammer events. Stratification and striping are reduced by an upturning elbow inside the steam generator which raises the feedring relative to the feedwater nozzle. The elevated feedring reduces the potential for stratified flow by allowing the cooler, denser feedwater to fill the nozzle/elbow arrangement before rising into the feedring. The potential for water hammer, stratification, and striping is additionally reduced by the use of a separate startup feedwater nozzle. The startup feedwater nozzle is located at an elevation that is just below the main feedwater nozzle and is rotated circumferentially away from the main feedwater nozzle. A startup feedwater spray system independent of the main feedwater feedring is used to introduce startup feedwater into the steam generator. The layout of the startup feedwater piping includes the same features as the main feedwater line to minimize the potential for water hammer.

At the bottom of the wrapper, the water is directed toward the center of the tube bundle by the lowest tube support plate. This recirculation arrangement serves to minimize the low-velocity zones having the potential for sludge deposition. As the water passes the tube bundle, it is converted to a steam–water mixture. Subsequently, the steam–water mixture from the tube bundle rises into the steam drum section, where centrifugal moisture separators remove most of the entrained water from the steam. The steam continues to the secondary separators, or dryers, for further moisture removal, increasing its quality to a designed minimum of 99.75% (0.25% by weight maximum liquid). Water separated from the steam combines with entering feedwater and recirculates through the steam generator. A sludge collector located amidst the inner primary separator risers provides a preferred region for sludge to settle away from the tubesheet and tube support plates.

The dry, saturated steam exits the steam generator through the outlet nozzle, which has a steam-flow restrictor.

2.15.5 Reactor Coolant Pressurizer

The pressurizer maintains the system pressure during operation and limits pressure transients. During the reduction or increase of plant load, the pressurizer accommodates volume changes in the reactor coolant. It is directly connected to the hot leg of the RCS. The pressurizer is a vertical, cylindrical vessel having hemispherical top and bottom heads constructed of low alloy steel. Internal surfaces exposed to the reactor coolant are clad austenitic stainless steel. The general configuration of the pressurizer is shown in Figure 2.28. The design data for the pressurizer are given in Tables 2.10 and 2.11. Nickel–chromium–iron alloys are not used for heater wells or instrument nozzles for materials compatibility reasons. The spray line nozzles and the automatic depressurization and safety valve connections are located in the top head of the pressurizer vessel. Spray flow is modulated by automatically controlled air-operated valves. The spray valves can also be operated manually from the control room. In the bottom head at the connection of the surge line to the surge nozzle a thermal sleeve protects the nozzle from thermal transients. A retaining screen above the surge nozzle prevents passage of foreign matter from the pressurizer to the RCS. Baffles in the lower section of the pressurizer prevent an in-surge of cold water from flowing directly to the steam/water interface. The baffles also assist in mixing the incoming water with the water in the pressurizer. The retaining screen and baffles also act as a diffuser. The baffles also support the heaters to limit vibration. Electric direct-immersion heaters are installed in vertically oriented heater wells located in the pressurizer bottom head. The heater wells are welded to the bottom head and form part of the pressure boundary. Heaters can be removed for maintenance or replacement. Heaters are grouped into a control group and backup groups. Heaters in the control group are proportional heaters which are supplied with continuously variable power to match heating needs. Heaters in the backup group are off or at full power. The power supply to the heaters is a 480-V 60 Hz. three-phase circuit. Each heater is connected to one leg of a delta-connected circuit and is rated at 480 V with one-phase current. A manway in the upper head provides access to the internal space of the pressurizer in order to inspect or maintain the spray nozzle. The manway closure is a gasketed cover held in place with threaded fasteners. Brackets on the upper shell attach the structure (a ring girder) of the pressurizer safety and relief valve (PSARV) module. The PSARV module includes the safety valves and the first three stages of ADS valves. The support brackets on the pressurizer represent the primary vertical load path to the building structure. Sway struts between the ring girder and pressurizer compartment walls also provide lateral support to the upper portion of the pressurizer. Four steel columns attach to pads on the lower head to provide vertical support for the vessel. The columns are based at elevation 107′-2″. Lateral support for the lower portion of the vessel is provided by sway struts between the columns and compartment walls.

2.15.6 ADS

ADS valves are part of the RCS and interface with the PXS. Twenty valves are divided into four depressurization stages. These stages connect to the RCS at three locations. The ADS first, second, and third stage valves are included as part of the PSARV module and are

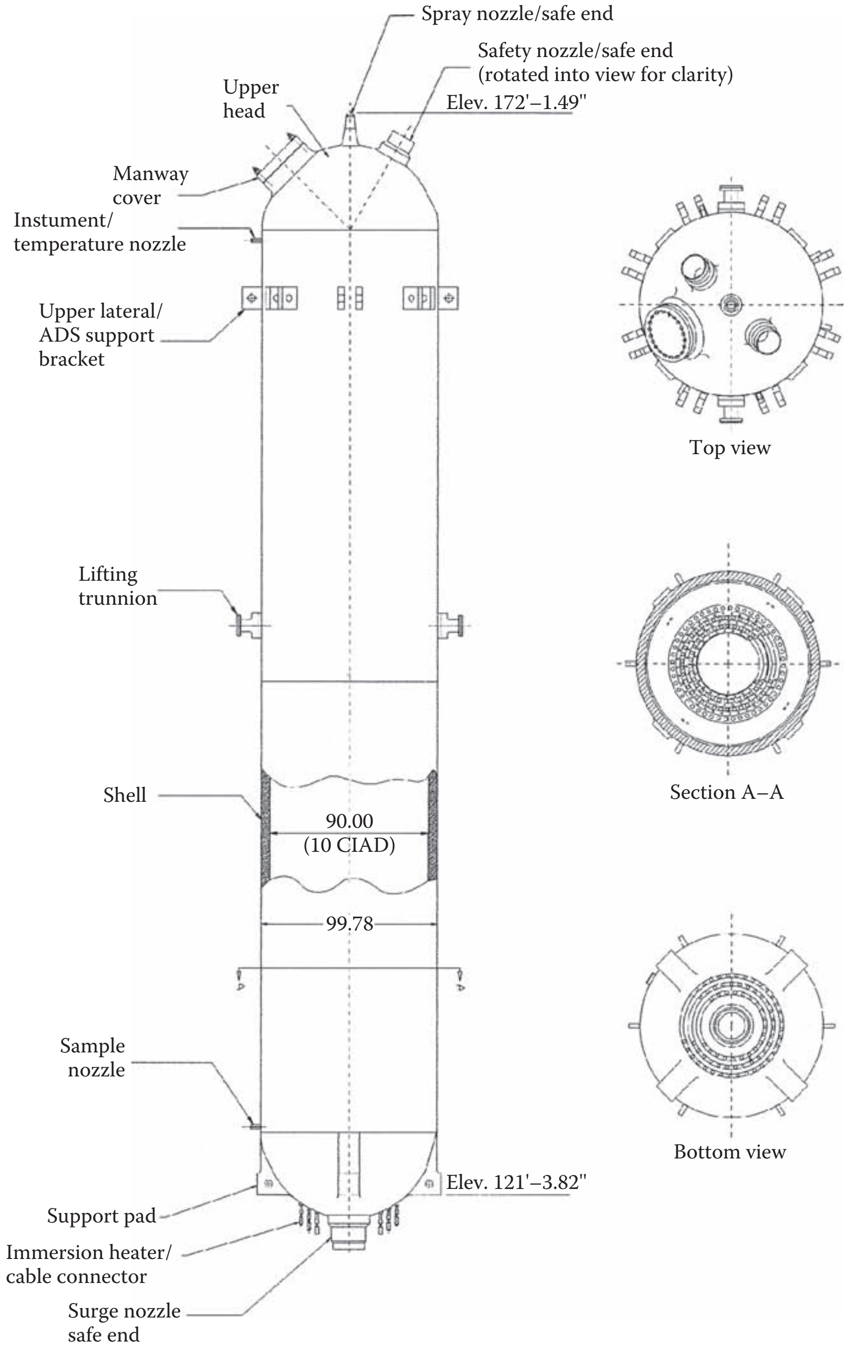


FIGURE 2.28
AP-1000 primary system pressurizer.

connected to nozzles on top of the pressurizer. Fourth-stage valves connect to the hot leg of each reactor coolant loop. Opening of the ADS valves is required for the PXS to function as required to provide emergency core cooling following postulated accident conditions. First-stage valves may also be used as required following an accident to remove

TABLE 2.10

Pressurizer Design Data Including Heater Group Parameters

Pressurizer Design Data	
Design pressure (psig)	2485
Design temperature (°F)	680
Surge line nozzle nominal diameter (in.)	18
Spray line nozzle nominal diameter (in.)	4
Safety valve nozzle nominal diameter (in.)	14
Internal volume (ft^3)	2100
Pressurizer Heater Group Parameters	
Voltage (Vac)	480
Frequency (Hz)	60
Power Capacity (kW)	
Control Group	370
Backup Group A	245
Backup Group B	245
Backup Group C	370
Backup Group D	370

TABLE 2.11

Pressurizer Safety Valves–Design Parameters

Number	2
Minimum required relieving capacity per valve (lbm/hr)	750,000 at 3% accumulation
Set pressure (psig)	2485 ±25 psi
Design temperature (°F)	680
Fluid	Saturated steam
Backpressure	
Nominal (psig)	3–5
Expected maximum during discharge (psig)	500
Environmental conditions	
Ambient temperature (°F)	50–120
Relative humidity (percent)	0–100
Residual Heat Removal Relief Valve–Design Parameters	
Number	1
Nominal relieving capacity per valve, ASME flow rate (gpm)	850
Nominal set pressure (psig)	500
Full-open pressure, with accumulation (psig)	550
Design temperature (°F)	400
Fluid	Reactor coolant
Backpressure	
Nominal (psig)	3–5
Expected maximum during discharge (psig)	200
Environmental conditions	
Ambient temperature (°F)	50–120
Relative humidity (percent)	0–100

noncondensable gases from the steam space of the pressurizer. The flashing coolant is directed to the IRWST by means of spargers. The tank itself is vented to containment with the buildup of very low internal pressure.

2.15.7 RNS

The RNS performs the following major functions:

- **RCS Shutdown Heat Removal—**Remove heat from the core and the RCS during shutdown operations.
- **Shutdown Purification—**Provide RCS and refueling cavity purification flow to the CVCS during refueling operations.
- **IRWST Cooling—**Provide cooling for the IRWST.
- **RCS Makeup—**Provide low-pressure makeup to the RCS.
- **Post-Accident Recovery—**Remove heat from the core and the RCS following successful mitigation of an accident by the PXS.
- **Low-Temperature Overpressure Protection—**Provide low-temperature overpressure protection for the RCS during refueling, startup, and shutdown operations.
- **Long-Term, Post-Accident Containment Inventory Makeup Flowpath—**Provide long-term, post-accident makeup flowpath to the containment inventory.
- **Spent Fuel Pool Cooling—**Provide backup for cooling the spent fuel pool.

The layout of the RNS is shown in Figure 2.29 and the component data are listed in Table 2.12.

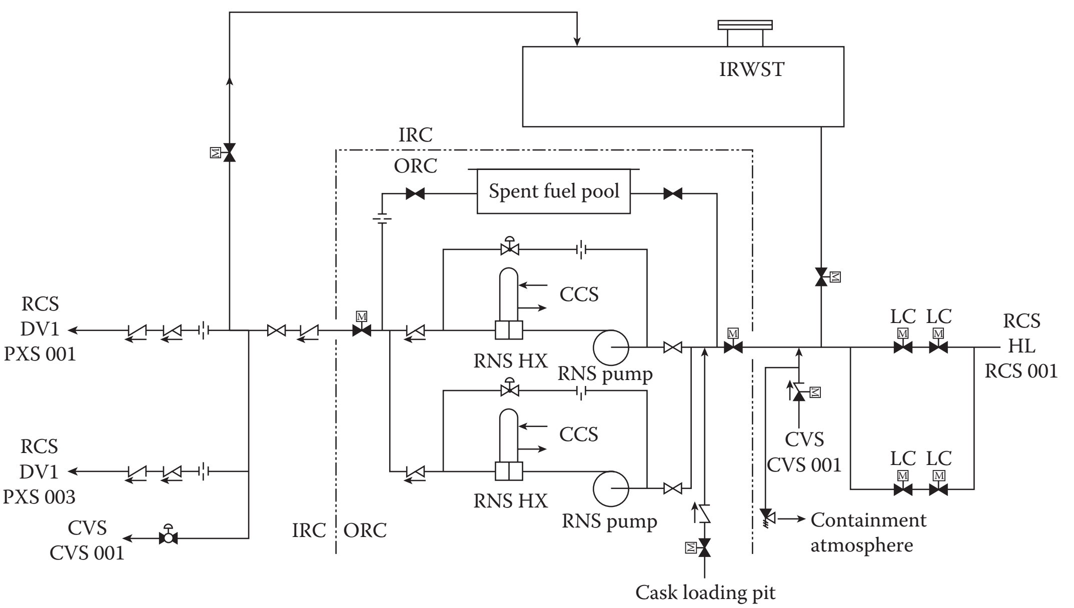


FIGURE 2.29
AP-1000 normal residual heat removal system.

TABLE 2.12

Heat Removal System Component Data

Nominal RHR Pumps (per pump)		
Minimum flow required for shutdown cooling (gpm)	1425	
Minimum flow required for low pressure makeup (gpm)	1100	
Design flow (gpm)	1500	
Design head (ft)	360	
Normal RHR Heat Exchangers		
Minimum UA required for shutdown cooling (BTU/hr-°F)	2.2×10^6	
Design heat removal capacity (BTU/hr)[a]	23×10^6	
	Tube Side	**Shell Side**
Design flow (lb/hr)	750,000	1,405,000
Inlet temperature (°F)	125	87.5
Outlet temperature (°F)	94	104
Fluid	Reactor Coolant	CCS
Design Bases for Normal Risidual Heat Removal System Operation		
RNS initiation (hours after reactor shutdown)	4	
RCS initial pressure (psig)	450	
RCS initial temperature (°F)	350	
CCS design temperature (°F)	95	
Cooldown time (hours after shutdown)	96	
RCS temperature at end of cooldown (°F)	125	

[a] Design heat removal capacity is based on decay heat at 96 hours after reactor shutdown.

2.16 PXS

The primary function of the PXS is to provide emergency core cooling following postulated design basis events. To accomplish this primary function, the PXS is designed to perform the functions detailed below.

- *Emergency core decay heat removal*—Provide core decay heat removal during transients, accidents, or whenever the normal heat removal paths are lost. This heat removal function is available at RCS conditions including shutdowns. During refueling operations, when the IRWST is drained into the refueling cavity, other passive means of core decay heat removal are utilized.
- *RCS emergency makeup and boration*—Provide RCS makeup and boration during transients or accidents when the normal RCS makeup supply from the CVCS is unavailable or is insufficient.
- *Safety injection*—Provide safety injection to the RCS to provide adequate core cooling for the complete range of LOCA, up to and including the double-ended rupture of the largest primary loop RCS piping.
- *Containment pH control*—Provide for chemical addition to the containment during post-accident conditions to establish floodup chemistry conditions that support radionuclide retention with high radioactivity in containment, and to prevent corrosion of containment equipment during long-term floodup conditions.

The PXS is designed to operate without the use of active equipment such as pumps and AC power sources. The PXS depends on reliable passive components and processes such as gravity injection and expansion of compressed gases. The PXS requires a one-time alignment of valves upon actuation of the specific components.

2.17 Detection and Ignition of Hydrogen

Another important component of the safety-related systems is the Hydrogen Detection and Ignition system. Reaction of steam with the Zircaloy-4 or ZIRLO™ fuel cladding can release hydrogen into the containment if there is a pipe rupture or other venting of RCS coolant outside the pressure boundary. An explosive mixture can result if the hydrogen concentration with dry air reaches about 20%. The limit is higher if a large amount of steam is also present. The objective of the hydrogen control system is to detect it at levels <20% and neutralize it by reacting it with air in catalytic heaters located at 64 places around the inside of containment such that natural circulation will assure adequate mixing and prevent the formation of an explosive mixture anywhere.

Zircaloy-4 is an alloy of zirconium with about 1.5% tin and lesser amounts of iron, nickel, chromium and about 200 ppm of carbon. It is an excellent window for neutrons, forms a tough, adherent oxide coating to prevent corrosion, and its mechanical properties provide adequate strength and ductility. It does not interact with the uranium dioxide fuel and barely reacts with fission products. Large ingots of the alloy are cut down into “trexes” a few inches’ thick and several inches in diameter. A hole, an inch or so in diameter, is carefully centered in the trex. From there it is worked by a three-role process that scuffs the material into a high tensile hydrostatic stress at the center. A mandrel pushes from behind and smoothes the inside surface as the material gets longer, thinner and smaller in diameter through many steps until it becomes tubing. Another process is called the Pilger process; a pair of mandrels squeeze the material in one spot, then the trex/tube is turned slightly and the process is repeated. A spiral path is followed until the tube reaches the desired dimensions for that step. The zirconium cold works readily and must be annealed between steps to maintain ductility for the next step in the process. Westinghouse has patented what they believe is an improvement, called ZIRLO™. The newer alloy includes 0.5–2.0 weight percent niobium as well as the other ingredients of Zircaloy-4. The alloy is also preferably subjected to intermediate recrystallization anneals at about 1200–1300°F, and to a beta quench two steps prior to final size.

2.18 IRWST

Another vital piece of equipment inside the containment, but not inside the RCS pressure boundary, is the IRWST (Figure 2.30). In older plant designs, the tank was located outside the shield building and was considered vulnerable. Locating the IRWST inside the containment allows it to participate in additional important functions. The IRWST is a large, stainless steel-lined tank located underneath the operating deck inside the containment. The tank is designed to meet seismic requirements and is constructed as an integral part of the containment internal structures, but is isolated from the steel containment vessel. The bottom of the IRWST is above the RCS loop elevation so that the borated refueling water can drain by gravity into the RCS, after it is sufficiently depressurized. The IRWST is connected to the RCS through both direct vessel injection lines. The IRWST and contents are at the existing temperature and pressure in the containment. Vents are installed in the roof of the tank. These vents are normally closed to contain water vapor and radioactive gases within the tank during normal operation and to prevent debris from entering the tank from the containment operating deck. The vents open with a slight pressurization of the IRWST. These vents provide a path to vent steam released by the spargers or generated by the PRHR heat exchanger into the containment atmosphere. Other vents also open on small-pressure differentials, such as during a LOCA, to prevent damage to the tank. Overflows are provided from the tank to the refueling cavity to accommodate volume and mass increases during PRHR heat exchanger or ADS operation, while minimizing the floodup of the containment. The IRWST does not contain material in the tank or the recirculation path that could plug the outlet screens. The tank contains one PRHR heat exchanger and two depressurization spargers. The top of the PRHR heat exchanger tubes are located underwater and extend down into the IRWST. The spargers are also submerged in the tank, with the spargers midarms located below the normal water level. The tank is sized to provide the flooding of the refueling cavity for normal refueling, the post-LOCA flooding of the containment for RCS long-term cooling mode, and to support the PRHR heat exchanger operation. Flow out of the IRWST during the injection mode includes conservative allowances for spill flow during a direct vessel injection line break. The IRWST can provide sufficient injection until the containment sump floods up high enough to initiate recirculation flow. The injection duration varies greatly depending upon the specific event. A direct vessel injection line break more rapidly drains the tank and speeds containment floodup. The containment floodup volume for a LOCA is <73,500 cubic feet (excluding the volume of the IRWST itself) below a containment elevation of 108 feet. Connections to the tank provide for transfer to and from the RCS/refueling cavity via the RNS, purification and sampling via the spent fuel pit cooling system, and remotely adjusting boron concentration to the CVCS. Also, the RNS can provide cooling of the IRWST. The water level and temperature in the tank are monitored by indicators and alarms. The operator can take action, as required, to meet the technical specification requirements for system operability.

The PRHR exchanger consists of inlet and outlet channel heads connected together by vertical C-shaped tubes (Figure 2.30). The tubes are supported inside the IRWST. The top of the tubes is several feet below the tank water surface. The PRHR heat exchanger is designed to meet seismic requirements. The heat exchanger inlet piping connects to an inlet channel head located near the outside top of the tank. The inlet channel head and tubesheet are attached to the tank wall via an extension flange. The heat exchanger is supported by a frame which is attached to the IRWST floor and ceiling. The extended flange is designed to accommodate thermal expansion. The heat exchanger outlet piping is connected to the outlet channel head, which is vertically below the inlet channel head, near the tank bottom. The outlet channel head has an identical structural configuration to the inlet channel head. Both channel head tubesheets are similar to the steam generator tubesheets and they have manways for inspection and maintenance access. The PRHR heat exchanger is designed to remove sufficient heat so that its operation, in conjunction with available inventory in the steam generators, provides RCS cooling and prevents water relief through the pressurizer safety valves during loss of main feedwater or main feedline break events. PRHR heat exchanger flow and inlet and outlet line temperatures are monitored by indicators and alarms. The operator can take action, as required, to meet the technical specification requirements or follow emergency operating procedures for control of the PRHR

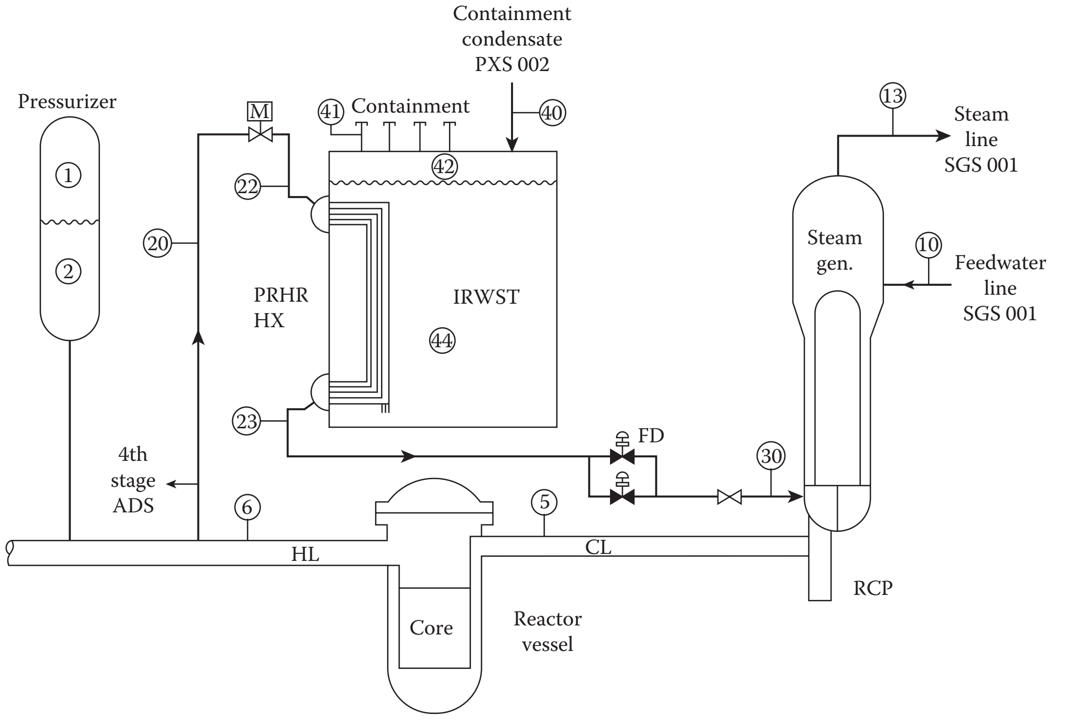


FIGURE 2.30
AP-1000 passive decay heat removal system.

heat exchanger operation. Two reactor coolant depressurization spargers are provided as shown in Figure 2.31. Each one is connected to an ADS discharge header (shared by three ADS stages). They are also submerged in the IRWST. Each sparger has four branch arms inclined downwards. The connection of the sparger branch arms to the sparger hub are submerged below tank overflow level by ≤11.5 feet. The spargers are designed to meet seismic requirements. The spargers perform a nonsafety-related function: minimizing plant cleanup and recovery actions following automatic depressurization. They are designed to distribute steam into the IRWST, thereby promoting more effective steam condensation. The first three stages of ADS valves discharge through the spargers and are designed to pass sufficient depressurization venting flow, with an acceptable pressure drop, to support the depressurization system performance requirements. The installation of the spargers prevents undesirable and/or excessive dynamic loads (water hammer) on the IRWST and other structures. Each sparger is sized to discharge at a flow rate that supports ADS performance, which in turn, allows adequate PXS injection.

2.19 Safety Design Rationale for Venting the Reactor Vessel Head

The Three Mile Island accident in 1979 was exacerbated by uncertainty over the existence of a steam bubble in the reactor vessel head as well as in the pressurizer. The crew were reluctant to pressurize the coolant, believing that it may result in system rupture. The time

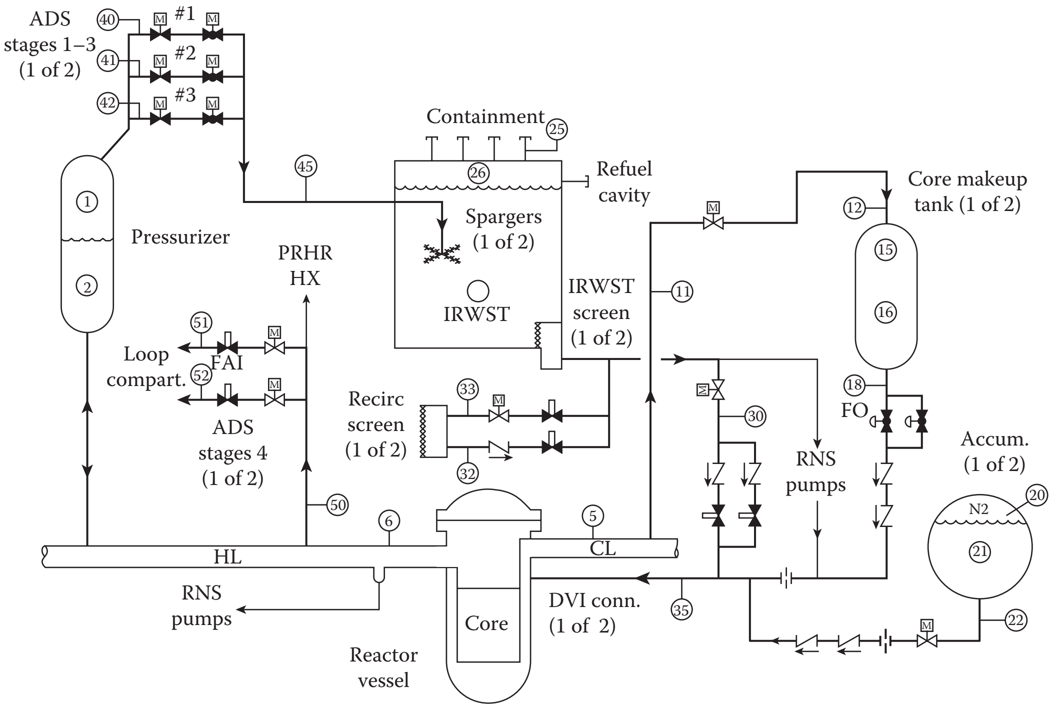


FIGURE 2.31
AP-1000 passive safety injection.

delay resulted in partial core melting because of inadequate emergency cooling. Since that event, it has been a requirement to vent the reactor vessel head remotely to be certain there is no bubble there during an accident (or at any time). The requirements for high point vents are provided by the reactor vessel head vent valves and the ADS valves. The ADS system has been described. The primary function of the reactor vessel head vent is for use during plant startup to properly fill the RCS and vessel head. Both reactor vessel head vent valves and the ADS valves may be activated and controlled from the main control room. The AP1000 does not require use of a reactor vessel head vent to provide safety-related core cooling following a postulated accident. The reactor vessel head vent valves (Figure 2.32 and Table 2.13) can remove noncondensable gases or steam from the reactor vessel head to mitigate a possible condition of inadequate core cooling or impaired natural circulation through the steam generators resulting from the accumulation of noncondensable gases in the RCS. The design of the reactor vessel head vent system is in accordance with NRC requirements. The reactor vessel head vent valves can be operated from the main control room to provide an emergency letdown path which is used to prevent pressurizer overfill following long-term loss of heat sink events. An orifice is provided downstream of each set of head vent valves to limit the emergency letdown flow rate. The first stage valves of the ADS are attached to the pressurizer and provide the capability of removing noncondensable gases from the pressurizer steam space following an accident. Venting of noncondensable gases from the pressurizer steam space is not required to provide safety-related core cooling following a postulated accident. Gas accumulations are removed by remote manual operation of the first-stage ADS valves. The discharge of the ADS valves is directed to the IRWST. The PRHR heat exchanger piping and the core makeup tank inlet piping in the PXS include high point vents that provide the capability of removing noncondensable

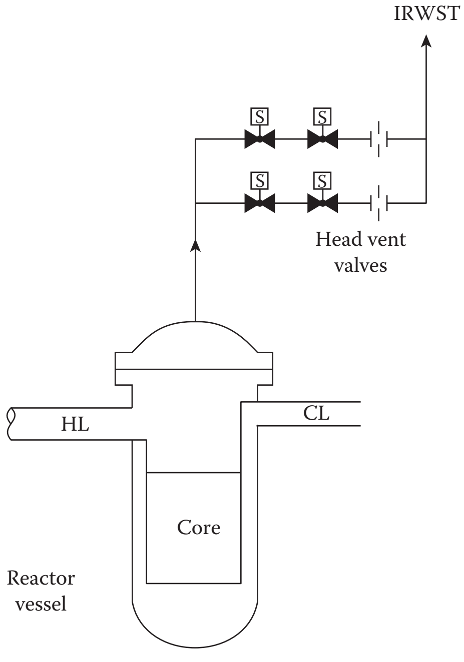


FIGURE 2.32
Reactor vessel head vent system.

TABLE 2.13
Reactor Vessel Head Vent System Design Parameters

System design pressure (psig)	2485
System design temperature (°F)	650
Number of remotely-operated valves	4
Vent line, nominal diameter (in.)	1
Head vent capacity (lbm/sec) (assuming a single failure, RCS pressure at 1250 psia)	8.2

gases that could interfere with heat exchanger or core makeup tank operation. These gases are normally expected to accumulate when the RCS is refilled and pressurized following refueling shutdown. Any noncondensable gases that collect in these high points can be manually vented. The discharge of the PRHR heat exchanger high point vent is directed to the IRWST. The discharge of the core makeup tank high point vent is directed to the reactor coolant drain tank. The reactor vessel head vent arrangement is designed to remove noncondensable gases or steam from the RCS via remote manual operations from the main control room through a pair of valves. The system discharges to the IRWST. The reactor vessel head vent system is designed to provide an emergency letdown path that can be used to prevent long-term pressurizer overfill following loss of heat sink events. The reactor vessel head vent is designed to limit the emergency letdown flow rate to within the capabilities of the normal makeup system. The reactor vessel head vent system can also vent noncondensable gases from the reactor head in case of a severe accident. The system vents the reactor vessel head by using only safety-related equipment. The reactor vessel head vent system satisfies applicable requirements.

2.20 Other Passive Emergency Systems

In addition to passive systems inside the containment, there are passive systems for cooling the containment itself if it is filled with steam and gives high-pressure signals. All passive systems are intended for keeping the radioactivity contained and the fuel and other equipment protected from further degradation for the long-term following the DBA. It is assumed that the reactor cannot be accessed by personnel during this time (about 1.5 months). The equipment has other uses during normal or upset conditions, but the safety function is not compromised by other uses, such as fire protection or refueling water make up. The Passive Containment Cooling Water Storage Tank (PCCWST) is incorporated into the shield-building structure above the containment vessel. The inside wetted walls of the tank are lined with stainless steel plate. It is filled with demineralized water and has the minimum required useable volume for the passive containment cooling function, over 756,000 gallons. The passive containment cooling system functions as the safety-related ultimate heat sink. The passive containment cooling water storage tank is seismically designed and missile protected. The surrounding reinforced concrete supporting structure is part of the protection for the passive cooling system. The welded seams of the plates forming part of the leak tight boundary are examined by liquid penetrant after fabrication to confirm that the boundary does not leak. The tank also has redundant level measurement channels and alarms for monitoring the tank water level and redundant temperature measurement channels to monitor and alarm for potential freezing. To maintain system operability, a recirculation loop that provides chemistry and temperature control is connected to the tank. The tank is constructed to provide sufficient thermal inertia and insulation such that draindown can be accomplished without heater operation. In addition to its containment heat removal function, the passive containment cooling water storage tank also serves as a source of makeup water to the spent fuel pool and a seismic Category I water storage reservoir for fire protection following a SSE. The PCCWST suction pipe for the fire protection system is configured so that actuation of the fire protection system will not infringe on the usable capacity allocated to the passive containment cooling function. System actuation consists of opening the passive containment cooling water storage tank isolation valves. This allows the passive containment cooling water storage tank water to be delivered to the top, external surface of the steel containment shell. The flow of water, provided entirely by the force of gravity, forms a water film over the dome and side walls of the containment structure. The cooling action reduces the pressure inside containment.

References

AP-1000 Design Control Document. Submittal to the Nuclear Regulatory Commission as document number APP-GW-GL-700, September, 2004.

Patent 5112573 gives the composition of ZIRLO™ and its heat treatment. Filed August 28, 1989. Issued May 12, 1992.

The Westinghouse Pressurized Water Reactor Nuclear Power Plant. Copyright 1984.