Westinghouse Technology Systems Manual

Chapter 4

CHEMICAL AND VOLUME CONTROL

Section

Westinghouse Technology Systems Manual

Section 4.1

Chemical and Volume Control System

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4.1 CHEMICAL AND VOLUME CONTROL SYSTEM

Learning Objectives:

1. List the purposes of the Chemical and Volume Control System (CVCS).

2. List in flow path order and state the purpose of the following major components of the CVCS:

   a. Regenerative heat exchanger
   b. Letdown orifice
   c. Letdown heat exchanger
   d. Demineralizers (ion exchangers)
   e. Letdown filter
   f. Volume control tank (VCT)
   g. Charging pump

3. Identify the components in the CVCS that are used to purify the reactor coolant and the types of contaminants each is designed to remove.

4. Describe how the makeup system is used to borate, dilute, and makeup a blended flow of boric acid to the reactor coolant system (RCS).

5. Explain why and for what plant conditions the following chemicals are added to the RCS:

   a. Lithium hydroxide
   b. Hydrogen
   c. Hydrazine

6. Describe the emergency boration flow path, and identify the plant conditions which would require its use.

7. State the purpose of the connection between the Residual Heat Removal System (RHR) and CVCS letdown.

8. List the plant operations that result in large amounts of influent into the Boron Recovery System (BRS).

9. Identify the changes in the CVCS that occur upon the receipt of an Engineered Safety Features Actuation Signal (ESFAS).

10. Explain how the CVCS is designed to prevent the following:

    a. Flashing and pressure transients in the regenerative and non-regenerative heat exchangers.
    b. Loss of suction to the charging pumps.
    c. High temperature in the letdown ion exchangers (demineralizers).
    d. Over and under pressurization of the volume control tank.

11. State when and why excess letdown would be used.

12. List the automatic actions initiated by VCT level instrumentation.

4.1.1 Introduction

The chemical and volume control system is a Seismic Category I system. Its purposes are to:

1. Adjust the RCS boric acid concentration.

2. Maintain the proper water inventory in the RCS in conjunction with the pressurizer level control system.

3. Provide seal water flow to the reactor coolant pump shaft seals.

4. Add corrosion inhibiting chemicals to the RCS.

5. Purify the reactor coolant in order to maintain

it within its design activity limits.

6. Provide borated water for emergency core cooling.

7. Process reactor coolant for the reuse of boric acid and reactor makeup water, through the boron recovery system.

8. Degas the RCS.

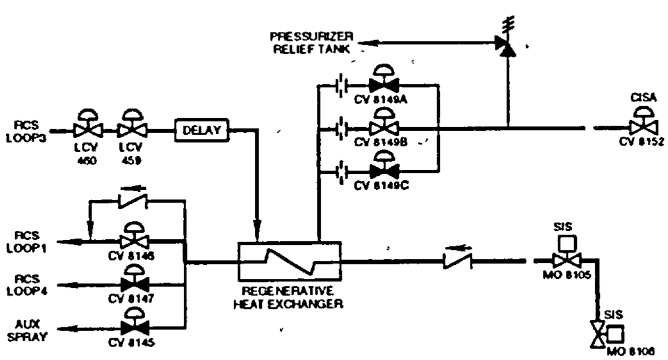
9. Provide a means of emergency boration of the RCS.

A discussion of each of these purposes, a system description, and an individual component description is contained in subsequent paragraphs of this section.

4.1.2 System Description

4.1.2.1 Letdown

The CVCS letdown path, as shown in Figure 4.1-1, taps off of the intermediate section of loop 3 cold leg piping through two series letdown isolation valves (LCV 459, 460). The reactor coolant now referred to as the letdown fluid then flows through a delay pipe to the regenerative heat exchanger. The regenerative heat exchanger provides the initial cooling of the letdown fluid by preheating the returning charging flow. From the regenerative heat exchanger, the letdown fluid passes through one or more letdown orifices. The letdown orifice(s) controls the amount of reactor coolant that is letdown (removed) from the RCS and provides the initial pressure reduction of this high pressure fluid. The letdown, after passing through the letdown orifices and isolation valves, exits the containment and flows through a containment isolation valve (CV-8152) to the letdown heat exchanger where the final cooling of this liquid occurs.



The letdown heat exchanger is cooled by the component cooling water system (CCW). This heat exchanger reduces the letdown fluid temperature to a value that is compatible with the ion exchanger (mixed bed demineralizer) resin. From the letdown heat exchanger the fluid is delivered to the letdown pressure control valve (PCV-131). This valve is sometimes referred to as the back pressure regulator. The letdown pressure control valve automatically maintains a constant pressure of 340 psig in the section of letdown piping upstream of the letdown heat exchanger to prevent the letdown liquid from flashing to steam. The cooled and depressurized letdown is then directed to the mixed bed demineralizers.

The mixed bed demineralizers (ion exchangers) are designed to remove ionic impurities from the reactor coolant. The ion exchangers are called mixed bed demineralizers because they contain a mixture of anion and cation resins. In addition to the mixed bed demineralizers, a special-purpose ion exchanger may be used, the cation ion exchange. It is used when a reduction in the RCS lithium or cesium concentration is desired. All of the ion exchanger resins are temperature sensitive; therefore, a temperature divert valve (TCV-129) bypasses letdown flow around the ion exchangers if the letdown heat exchanger outlet temperature rises to 137°F. The next component in the letdown stream is the letdown filter. The letdown filter removes any resin fines (broken resin beads) that escape from the ion exchangers. The purified, filtered letdown then flows to the volume control tank through a three-way valve. This three-way valve (LCV-112A) diverts the letdown to the boron recovery system BRS on a high level in the volume control tank. The VCT completes the portion of the CVCS known as letdown.

4.1.2.2 Volume Control Tank

The volume control tank collects the RCS letdown and provides a suction reservoir and head for the charging pumps. The tank is pressured with hydrogen gas. The hydrogen will dissolve into the charging fluid for the scavenging of oxygen in the reactor coolant system.

4.1.2.3 Reactor Makeup and Chemical Addition

The reactor makeup system (RMS) provides a method of supplying concentrated boric acid, demineralized reactor makeup water, or a mixture of both, to the volume control tank or the charging pump suction header. Chemicals such as lithium hydroxide and hydrazine may be added to the charging pump suction header via the chemical addition mixing tank. (Section 4.1.3.2)

4.1.2.4 Charging System

The charging portion of the system contains three pumps: two redundant centrifugal charging pumps and a positive displacement pump. Any of the pumps may be used to supply charging; however, a centrifugal charging pump is normally used. In addition to the charging function, the centrifugal charging pumps also serve as the high head safety injection pumps during emergency core cooling system operations. During normal operation, the charging pumps supply both reactor coolant pump seal injection and normal charging requirements.

The seal injection portion of the system consists of seal injection filters and individual flow control valves to supply the required amount of seal injection flow to each reactor coolant pump.

The normal charging header consists of a flow control valve (HCV-182) that is used to divide flow between the seal injection header and the charging header. The charging header contains isolation valves to isolate the charging header during accident conditions. Downstream of the charging header isolation valves, MO-8105 and MO-8106 is the tube side of the regenerative heat exchanger, and three charging paths. The three charging paths allow the preheated charging flow to be directed to loop 1 (normal charging), loop 4 (alternate charging), or to the pressurizer spray line (auxiliary spray).

During normal steady state operations, letdown is in service through one letdown orifice, a mixed bed demineralizer, and then to the VCT. This flowpath allows continuous purification of the RCS. From the VCT, reactor coolant is returned to the RCS via the seal injection piping and the normal charging path by one of the charging pumps. Reactor coolant pump seal return flow is routed to the charging pump suction via the seal water heat exchanger.

A flow balance diagram of the CVCS is provided in Figure 4.1-2. Normally the charging pump discharge flow rate is 87 gpm, of which, 55 gpm is returned to the RCS via the normal charging line and 32 gpm goes into the reactor coolant pump (RCP) seals. This division of flow, is determined by the position of the HCV-182 valve. Five gpm per RCP is returned to the RCS via the thermal barrier heat exchanger located in the reactor coolant pump casing, for a total of 20 gpm. This flow, plus the 55 gpm normal charging results in a total of 75 gpm being pumped into the RCS. In order to maintain a flow balance in the RCS, 75 gpm is removed via the letdown line.

A flow balance is maintained on the VCT by 75 gpm letdown and 12 gpm seal return into the VCT, and 87 gpm output to the charging pumps. In reality since 3 gph per RCP of the seal injection goes to the liquid waste system, slightly less than the 3 gpm per pump is returned to the VCT. Therefore, the VCT has a normal flow imbalance, that is, less entering the tank than leaving. As a result of this flow imbalance over a time period of several operating shifts, the automatic makeup system to the VCT will activate to makeup for this inventory loss.

4.1.2.5 Excess Letdown

Certain plant evolutions such as RCS heatup or the inoperability of the normal letdown path may require the use of the excess letdown. At low RCS pressures, when the letdown orifices do not pass their design flows, the excess letdown may be placed in service. Placing excess letdown in service assists the normal letdown system in removing the expansion volume due to the RCS heatup. The amount of water removed is minimal since it is only a 1 inch diameter line as it is designed to equal the nominal RCP seal injection flow (20 gpm). If plant conditions dictate the removal of the normal letdown flow path from service, then excess letdown can be placed in service to balance RCP injection flow..

The excess letdown consists of a penetration to loop 3 cold leg piping, an excess letdown heat exchanger with associated inlet and outlet valves, and an excess letdown divert valve. To place excess letdown in service, CCW is supplied to the heat exchanger, and excess letdown flow is established by opening the heat exchanger inlet and outlet valves.

Normally the excess letdown is directed to the CVCS through the reactor coolant pump seal return line. However, the excess letdown divert valve may be positioned to divert flow to the reactor coolant drain tank (RCDT).

4.1.2.6 Boron Recovery (Recycle) System

The BRS (sometimes called the boron recycle system), as shown in Figure 4.1-3, collects excess borated water resulting from certain plant operations. In each of these operations, the excess reactor coolant is diverted from the CVCS letdown line to the recycle holdup tanks as a result of a high level in the volume control tank. The following operations result in borated water being diverted to the BRS:

1. Dilution of reactor coolant to compensate for core burnup
2. Load follow operations
3. Heatup of the RCS from cold shutdown to hot standby
4. Refueling operations

Excess liquid effluents containing boric acid flow from the RCS through the letdown line and are collected in the holdup tanks. As this liquid enters the holdup tanks and the level rises, nitrogen cover gas is displaced to the waste gas decay tanks in the gaseous waste disposal System. The concentration of boric acid in the holdup tanks varies throughout core life from the refueling concentration of 2000 ppm to essentially zero ppm at the end of the core cycle. A holdup tank recirculation pump is provided to transfer liquid between the holdup tanks. After a certain amount of fluid has been collected, the liquid effluent in the holdup tanks will be processed as a batch operation. First, the liquid is pumped by a boric acid evaporator feed pump through a pair of evaporator feed ion exchangers. The liquid then flows through a filter, and into the boric acid evaporator.

The fluid enters the stripper column of the evaporator (Figure 4.1-4) via a preheater, where dissolved gasses are removed from the liquid.

These gases are vented through the vent condenser to the waste gas disposal system. The liquid effluent from the stripper column enters the evaporator section, where the evaporator separates the fluid into water vapor and a concentrated boric acid solution. The water vapor rises through the absorption tower where any boric acid or gas carry-over is absorbed by the recirculated condensate (reflux) flow and returned to the evaporator section. The remaining water vapor is condensed in the evaporator condenser and pumped, through an evaporator condensate demineralizer (Figure 4.1-3) and filter, to one of two monitor tanks. The condensate is accumulated in the monitor tanks and sampled before it is moved. Discharge from the monitor tanks may be pumped to any one of the following places.

1. Primary water storage tank
2. Lake discharge tank
3. Holdup tanks
4. Evaporator condensate demineralizers
5. Liquid waste system

The concentrated boric acid solution originally in the evaporator section remains as the bottoms of the distillation process and is concentrated to approximately four percent boric acid. Boric acid evaporator bottoms are sampled and, if analysis indicates that the solution meets specifications for use as boric acid makeup, the solution is sent to the boric acid tanks via the concentrates filter and concentrates holding tank. Otherwise, the concentrates are returned to the holdup tanks for reprocessing or are pumped to the waste disposal system.

4.1.3 Component Description

4.1.3.1 Letdown Components

Letdown Isolation Valves

The letdown isolation valves (LCV-459 and LCV-460) are used to isolate the letdown flow if pressurizer level decreases to the low level 17% setpoint. An interlock prevents opening or closing these valves unless all three letdown orifice isolation valves are closed. This prevents depressurization of the line segment containing the regenerative heat exchanger.

Letdown Delay Pipe

A long section of larger diameter pipe is used to increase the transport time of the coolant to allow most of the highly radioactive nitrogen-16 to decay before the letdown flow exits the containment.

Regenerative Heat Exchanger

The regenerative heat exchanger is a stainless steel, tube and shell heat exchanger with letdown flow on the shell side and charging flow through the tubes. The first stage of letdown cooling is performed by the charging flow. The removal of heat from the letdown drops its temperature from 550°F to 290°F and preheats the charging stream from 130°F to 500°F. This helps conserve energy and minimizes thermal stresses to the charging nozzles.

Letdown Orifices and Letdown Orifice Isolation Valves

Three orifices are provided to control letdown flow. Two of these orifices will pass 75 gpm each at an RCS pressure of 2235 psig. The third orifice is rated at 45 gpm. Normally one of the two 75 gpm orifices is in service. If extra purification flow is desired, or additional letdown flow for boron concentration changes is required, the 45 gpm orifice may be placed in service. Ion exchanger flow limitations (127 gpm maximum), which are administratively maintained, precludes placing both 75 gpm orifices in service at normal system pressure.

Remotely-operated letdown orifice isolation valves are located downstream of the letdown orifices. These valves are interlocked as follows:

1. The letdown orifice isolation valves close on low pressurizer level. This is a redundant isolation to LCV-459 and LCV-460.

2. At least one charging pump must be running in order to open the letdown orifice isolation valves. If the running charging pump(s) is lost, then the letdown orifice isolation valves will close. This interlock ensures that cooling water (charging) is available to the regenerative heat exchanger prior to the establishment of letdown flow.

3. The letdown isolation valves (LCV-459 and LCV-460) must be open prior to opening of the letdown orifice isolation valves. This interlock prevents depressurization of the letdown line.

The interlocks associated with the letdown isolation valves and the letdown orifice isolation valves mandate a specific order of operation when placing the CVCS in service. First, a charging pump is started. Next, the letdown isolation valves are opened. Finally, a letdown orifice isolation valve is opened.

Containment Isolation Valve

The containment isolation valve (CV-8152) is a remotely operated isolation valve that provides redundant isolation of letdown flow, along with the orifice isolation valves, on receipt of a Containment Phase A isolation signal.

Letdown Heat Exchanger

The final cooling of the letdown is accomplished by the letdown heat exchanger. This heat exchanger is of the tube and shell design with letdown flowing through its stainless steel tubes, and component cooling water flowing through its carbon steel shell.

The component cooling water outlet valve is a modulating valve whose position is controlled by the letdown heat exchangers outlet temperature. The normal controller setpoint is 120°F. If the controller is in automatic and the letdown heat exchangers outlet temperature increases, then the letdown heat exchanger component cooling water outlet valve is opened to increase cooling flow. Conversely, if temperature decreases, component cooling water flow through the letdown heat exchanger is reduced.

Letdown Pressure Control Valve (Back Pressure Regulating Valve)

In order to prevent the letdown fluid from flashing to steam upstream of the letdown heat exchanger, a high pressure must be maintained until the letdown temperature can be reduced. The required pressure is maintained by the letdown pressure control valve (PCV-131). A PID controller receives an input from a pressure transmitter downstream of the letdown heat exchanger and compares this pressure with an adjustable setpoint (normally 340 psig). The controller modulates the pressure control valve to maintain letdown system pressure at setpoint.

Temperature Divert Valve

The temperature divert valve (TCV-129) is a three-way valve. In the normal position, the temperature divert valve directs letdown flow to the mixed bed demineralizers and in the bypass position, letdown flow is diverted around the demineralizers. Demineralizer efficiency is reduced and resin bed lifetime is shortened by high temperatures. The position of the temperature divert valve is determined by the letdown heat exchanger's outlet temperature. The temperature divert valve directs flow to the demineralizers when temperature is less than 137°F. When the letdown heat exchanger outlet temperature reaches 137°F, the temperature divert valve automatically bypasses letdown flow around the demineralizers. The flow to the demineralizers through this valve can be operated from the main control board. The operator can select either the normal or the divert position. Once the divert valve has been automatically diverted to the VCT, it takes operator action to return the valve to its normal position.

Letdown Line Over pressure Protection

Two relief valves are installed to prevent over pressurization of the letdown piping. The first relief valve is located downstream of the letdown orifices, and prevents over pressurization of the piping between the orifices and the letdown pressure control valve. The valve has a setpoint of 600 psig, and can relieve the flow from all three letdown orifices (195 gpm). The valve discharges to the pressurizer relief tank (PRT). The second relief valve is located downstream of the letdown pressure control valve and protects the ion exchangers, letdown filter, and low pressure letdown piping from over pressurization. This valve is set at 285 psig, and also has a capacity equal to the flow through all three letdown orifices. This valve discharges to the VCT.

Mixed Bed Demineralizers

Two mixed bed demineralizers are installed to remove ionic impurities from the coolant letdown from the RCS. The demineralizers are called mixed bed because both anion (removes negatively charged ions) and cation (removes positively charged ions) resins are contained in the same vessel.

Each of the stainless steel demineralizer vessels has a capacity of 30 cubic feet of resin. During normal operations, one mixed bed demineralizer will be in service with the other in standby. The minimum design decontamination factor (input activity/output activity) for the resin beds is ten. Since each cubic foot of resin contains millions of beads, the ion exchangers are also very effective mechanical filters. Either Li-OH or H-OH resin may be used in the mixed bed ion exchangers.

Cation Demineralizer

During power operations, lithium is formed from the boron-neutron reaction. This extra lithium could result in excessively high pH values in the RCS. One of the purposes of the cation demineralizer is to remove excess lithium. This demineralizer is placed in service, as required, to maintain lithium concentration within limits. In the event of fuel clad failure, the cation demineralizer can also remove fission product ions especially cesium from the letdown stream. As its name implies, the cation ion exchanger contains only cation type resin.

VCT Level Divert Valve

The level divert valve (LCV-112A) is a three-way valve controlled by volume control tank level. An operator adjusted VCT level setpoint is compared with actual VCT level. If actual tank level exceeds the setpoint, then LCV-112A will begin to divert a portion of the letdown flow to the BRS holdup tanks. As the level error increases, an increased amount of letdown is diverted. With decreased flow into the VCT (due to partial diversion of letdown flow) and normal charging outflow, the VCT level will drop. As VCT level drops, the amount of letdown diverted to the holdup tanks will decrease. This action will continue until the level setpoint is again reached. At or below level setpoint, LCV-112A will be positioned to the VCT.

Backup level control from separate level transmitters is provided in the case of normal controller failure. In the backup control, if VCT level reaches the Hi alarm setpoint, LCV-112A will divert all letdown flow to the BRS. The level divert valve can also be operated from the main control board. The operator can select either the normal or divert position. If the divert position is selected the valve will go to full divert, with no flow being supplied to the VCT.

Volume Control Tank

The VCT assists the pressurizer with RCS volume changes, provides an interface with the RMS (Section 4.1.3.2), provides a means of hydrogen addition to the RCS, and allows for RCS de-gasification.

If the level in the pressurizer is above its setpoint, then CVCS charging flow will be reduced. Reduced charging flow and a constant letdown flow will restore pressurizer level to normal. The mass removed from the RCS to lower pressurizer level results in an increased VCT level. If the level in the pressurizer is below its setpoint, then the charging flow will be increased. Increasing the charging flow and maintaining a constant letdown will raise pressurizer level to its normal setpoint. The water required to raise pressurizer level comes from the VCT. Pressurizer level control is discussed in Chapter 10.3.

A hydrogen over pressure is maintained in the volume control tank. The letdown flow enters the VCT through a spray nozzle and absorbs the hydrogen gas. When this fluid reaches the reactor core, radiation causes the combination of the excess hydrogen with any free oxygen to form water.

The following list contains additional penetrations into the VCT :

1. Nitrogen supply - this tap shares a VCT penetration with the plant hydrogen system. Nitrogen may be used to purge hydrogen and fission gasses from the VCT during periods of shutdown or maintenance. Nitrogen should not be added during power operations because of possible nitric acid formation in the RCS.

2. VCT vent - this connection is used to vent fission gasses and hydrogen from the VCT gas space to the waste gas system. The vent line contains a remotely operated solenoid valve and pressure regulating valve. The pressure regulating valve senses VCT pressure and closes if pressure drops to 15 psig to ensure an adequate net positive suction head (NPSH) for the charging pumps.

3. VCT relief valve - The VCT relief valve line shares a penetration with the VCT vent, and provides over pressure protection for the tank. The valve relieves at 75 psig and discharges to the recycle holdup tanks.

4. RCP seal return line - Allows the seal return from the reactor coolant pumps to be directed to the VCT. However, the seal return is normally aligned directly to the charging pump suction.

As the letdown is depressurized through the spray nozzle in the VCT, dissolved fission product gases come out of solution and collect in the VCT. De-gasification of the RCS is accomplished by opening the VCT vent to the waste gas system and dropping VCT pressure. When pressure reaches 15 psig, the vent is closed. VCT level is increased to compress the gasses, and again, the VCT vent is opened until pressure drops to 15 psig. Letdown flow is manually diverted to the holdup tanks until the automatic makeup point is reached. This decrease in VCT level allows a volume for the accumulation of more hydrogen and fission gasses from the RCS. When the automatic makeup setpoint is reached, letdown flow is manually returned to the VCT. The level oscillations and venting operations are continued until the desired gas concentration is reached.

4.1.3.2 Reactor Makeup System Components

As stated in Section 4.1.2.3, the RMS provides a method of supplying concentrated boric acid, demineralized reactor makeup water, or a mixture of both to the VCT or the charging pump suction. The reactor makeup system can also supply borated water to the refueling water storage tank (RWST), the spent fuel pool, and the waste holdup tanks. The system consists of storage tanks, transfer pumps, and associated pipes and valves. The makeup system is illustrated in Figure 4.1-5.

Primary Water Storage Tank

The primary water storage tank provides a source of demineralized water for the RMS. This 203,000 gallon capacity tank may be filled from the plant secondary makeup system or the distillate from the boric acid evaporators. Demineralized water is transferred from the primary water storage tank by one of the two primary water transfer pumps. During normal operations one pump is in service and the other is in standby. If the running pump is stopped, the standby pump will automatically start.

Boric Acid Tanks

Two boric acid tanks are located in the auxiliary building. Each boric acid tank is constructed of stainless steel and has a capacity of 24,228 gallons. The concentration of the stored boric acid is approximately 4 weight percent (7000 ppm). The boric acid is maintained in solution by maintaining the room temperature above 65°F. A major advantage of this system over the 12 weight percent systems found at some plants is that special heat tracing is not required for the 4 weight percent solution.

The Technical Specifications for the plant require a minimum volume of 14,418 gallons of 7,000 ppm boron from the boric acid tanks or 74,752 gallons of 2000 ppm boron from the refueling water storage tank to insure that a 1.6%ΔK/K shutdown margin can be achieved. This assumes that the plant is at cold shutdown, at the end of its fuel cycle, and that the most reactive rod is fully withdrawn from the core. The boric acid tank may be filled from either the boric acid batch tank or the boric acid evaporator concentrates.

Boric Acid Batch Tank

The batch tank is used for mixing the makeup supply of boric acid solution for the boric acid tanks. The mixing evolution consists of heating the required volume of demineralized water to a temperature above saturation temperature for the boric acid concentration to be mixed, adding the correct amount of boric acid crystals, and agitating the mixture with an electric-driven agitator. If desired, the concentrated boric acid mixture may be transferred directly to the suction of the boric acid transfer pumps.

Boric Acid Transfer Pumps

Two electric driven pumps transfer boric acid from the boric acid tank to the reactor makeup system. Operations requiring the addition of boric acid to the RCS will automatically start both pumps. If emergency boration is required, the operator must open MO1804 and start a pump.

Boric Acid Filter

The boric acid filter collects particulates from the boric acid solution being pumped to the

CVCS. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously.

Reactor Makeup Control

The RMS consists of a boric acid flow transmitter, boric acid batch integrator, boric acid flow control valve, total flow transmitter, total flow batch integrator, primary water flow control valve, a blend tee, blender supply valves to the VCT, and charging pump suction, a mode selector switch, and a makeup permissive switch. These components are used to add boric acid or demineralized water to the RCS through five modes of operation controlled by the mode selector switch. These modes are borate, dilute, alternate dilute, auto, and manual. Changing the position of the MODE SELECTOR SWITCH automatically turns off the makeup system. Therefore, when selecting a new mode, the system must be re-energized by placing the Makeup Permissive Switch to RUN.

1. BORATE - The mode selector switch is selected to the "borate" position. The desired amount of boric acid to be added is set into the boric acid batch integrator. A desired boric acid flow rate is chosen, and the makeup permissive switch is selected to "run". The boric acid transfer pumps are automatically started, and boric acid is added to the charging pump suction via the boric acid flow control valve (FCV-110A), the blender, and the blender supply to the charging pumps (FCV-110B).

   The output of the boric acid flow transmitter controls the boric acid flow control valve and supplies a signal to the boric acid batch integrator. When the batch integrator senses that the desired amount of boric acid has been added, valves FCV-110A and FCV-110B are closed, and the boric acid pumps are stopped.

   After this operation is over, the operator must either select a new mode and energize the system with the MAKEUP PERMISSIVE SWITCH or repeat the operation in the present mode. This sequence of actions must also be performed by the operator when operating the makeup system in the dilute or alternate dilute modes.

2. DILUTE - The "dilute" mode is used to reduce the boron concentration in the RCS. The mode selector is placed in the "dilute" position. The desired quantity and flow rate are set into the total flow batch integrator and flow controller. The makeup permissive switch is selected to "run". The primary flow control valve (FCV-111A) opens, and demineralized water is added to the VCT via the blender and the blender supply valve to the VCT (FCV-111B). As water is added to the VCT, the flow is sensed by the total flow transmitter. The output of the total flow transmitter controls the primary flow control valve, and supplies a signal to the total flow batch integrator. When the total flow batch integrator senses that the desired amount of primary water has been added to the VCT, the valves FCV-111A and FCV-111B are closed.

3. ALTERNATE DILUTE - Operation in the alternate dilute mode is the same as the dilute mode except that the flow path of primary water is different. In alternate dilute a portion of the dilution flows directly to the charging pump suction via FCV-110B and a portion flows through the normal dilute path through FCV-111B. The alternate dilute mode is used to reduce the dilution delay time of the VCT. Since some of the primary water bypasses the VCT, it does not absorb hydrogen. Excessive use of the alternate dilute mode could result in high RCS oxygen concentrations.

4. AUTOMATIC - The "automatic makeup" mode of operation, the reactor makeup control system provides boric acid solution preset to match the boron concentration in the RCS. The automatic makeup compensates for normal leakage of reactor coolant without causing changes in the reactor coolant boron concentration.

   Under normal plant operating conditions, the mode selector switch and makeup valves are set in the "automatic" position and the permissive switch set to "auto". The operator sets the boric acid controller to a desired flow rate that results in the final desired boron concentration. A preset low level signal from the volume control tank level controller actuates automatic makeup that starts both boric acid transfer pumps, opens the blender supply valve (FCV-110B) to the charging pump suction. The primary water flow control valve (FCV-111A) opens to a preset position to give a fixed flow rate.

   Makeup addition to the charging pump suction header causes the water level in the volume control tank to rise. At a preset level in the VCT, the primary makeup water flow control valve closes, the boric acid transfer pumps stop, the concentrated boric acid flow control valve closes, and the blender supply valve to charging pump suction closes.

5. MANUAL - The "manual" mode of operation permits the addition of a pre-selected quantity and blend of boric acid solution to the CVCS, the refueling water storage tank, the holdup tanks in the boron recovery system, or some other location via a temporary connection. While in the manual mode of operation, automatic makeup to the RCS is prevented. The discharge flow path must be aligned manually.

   The operator sets the mode selector switch to "manual", the boric acid and total flow controllers to the desired flow rates, the boric acid and total flow batch integrators to the desired quantities and positions the makeup permissive switch to "run". When the preset quantities of boric acid and primary makeup water have been added, the boric acid transfer pumps stop and the boric acid and primary makeup water flow control valves close. This operation may be stopped manually by placing the makeup permissive switch to the stop position, and manually closing any valves the operator had previously opened.

   If either batch integrator is satisfied before the other has recorded its required total, the pump and valve associated with the satisfied integrator will terminate that portion of flow. The flow controlled by the other integrator will continue until that integrator is also satisfied.

Boric Acid Blender

The boric acid blender, as shown in Figure 4.1-6, insures thorough mixing of boric acid and primary water for the RMS. The blender consists of a conventional pipe tee fitted with a perforated tube insert. The boric acid flow is supplied through the perforated tube, and primary water enters at the bottom. A blended concentration exits the tee, and is supplied to the CVCS. The blender and boric acid flow control valve limit boric acid flow to a maximum of 10 gpm in plants with a 12 weight percent boric acid concentration in the Boric Acid Tanks.

Emergency Boration Path

Certain operational events require rapid addition of large quantities of boric acid into the RCS. This is accomplished through the emergency boration flow path. Events that would require emergency boration are shutdown margin related, and include, Anticipated Transients Without a Scram (ATWS), reactor trips with more than one rod stuck out of the core, or inadequate boron concentrations during shutdown or refueling conditions. Boric acid addition is an alternate, diverse method of negative reactivity addition to assure the reactor can be shut down.

The emergency boration flow path as shown in Figure 4.1-5, consists of a motor operated emergency boration valve (MO-8104) and a flow transmitter. Emergency (immediate) boration is accomplished by throttling the emergency boration valve to achieve the desired flow rate. In the event that both the normal (blender) and emergency boration flow paths are unavailable, an alternate emergency boration path is provided. To use this path, FCV-110A and a locked close local manual valve 8439 must be opened. Regardless of the emergency boration path chosen, emergency boration flow is determined by charging pump flow.

Electrical Heat Tracing (For 7 and 12 weight percent boric acid plants only)

Electrical heat tracing is installed under the insulation on all piping, valves, line-mounted instrumentation, and components normally containing concentrated boric acid solution. The heat tracing is designed to prevent boric acid precipitation due to cooling. Redundant heat tracing on sections of the CVCS normally containing concentrated boric acid solution provides redundancy if the operating heat tracing system malfunctions. Power for the electric heat tracing is supplied from the 480Vac vital power. Heat tracing is addressed by the plants Technical Specifications.

Chemical Addition

The chemical addition portion of the CVCS, as shown in Figure 4.1-5, consists of a chemical mixing tank and the piping which connects the tank to the charging pump suction. This allows the addition of lithium hydroxide for pH control of the RCS, and the addition of hydrazine for oxygen control when the plant is in cold shutdown. Chemicals are added through a funnel and flushed to the charging pump suction with primary water.

4.1.3.3 Charging System Components

VCT Outlet Valves

Two series, motor-operated valves (LCV-112B and LCV-112C) are located in the VCT outlet line. These valves provide redundant isolation of charging pump suction path from the VCT on low VCT level or an ESFAS.

RWST Suction Supply

Two parallel, motor-operated valves (LCV-112D and LCV-112E) supply the charging pump suction header with borated water from the refueling water storage tank (RWST). The valves are normally closed, and open to supply a suction to the charging pumps on a low VCT level or an engineered safety features actuation signal.

RHR Suction Supply

A suction supply to the centrifugal charging pumps is supplied by the residual heat removal (RHR) pump discharge. This line is used to supply the high head injection pumps during the recirculation phase of a loss of coolant accident.

Charging Pumps

Three charging pumps are installed in the system to provide charging flow to the RCS. Two of the pumps are single speed, horizontal centrifugal pumps powered from vital (Class 1E) ac power and the third is a positive displacement (reciprocating) pump equipped with variable speed drive. The positive displacement pump is powered from a non-vital ac source. All parts in contact with the reactor coolant are fabricated of austenitic stainless steel or other materials of adequate corrosion resistance. The centrifugal pump seals and the reciprocating pump stuffing box are provided with leak offs to collect reactor coolant leakage before it can escape to the atmosphere. There is a minimum flow recirculation line for the centrifugal charging pumps to protect them from a low flow condition. The reciprocating pump has a recirculation line and a relief valve to the VCT for over pressure protection.

The charging flow rate is determined by the pressurizer level control system. Control of the flow rate from the reciprocating pump is accomplished by varying the speed of the pump. When operating a centrifugal charging pump, the flow paths remain the same but charging flow rate controlled by varying the position of a modulating valve (FCV-121) on the discharge of the centrifugal pumps.

The centrifugal charging pumps also serve as the high head safety injection pumps in the emergency core cooling system.

RCP Seal Flow Control Valve

The reactor coolant pump seal flow control valve (HCV-182), located in the charging header, determines the division of flow between the RCP seal injection path and the charging flow path. This valve is remotely adjusted by the control room operator. RCP seal flow is increased by throttling closed HCV-182. If seal flow is increased, then the charging flow is decreased. Conversely, if HCV-182 is throttled open, the seal flow decreases and the charging flow increases.

Charging Isolation Valves

Two series, motor-operated isolation valves, (MO-8105 and MO-8106) isolate the charging header on an engineered safety features actuation signal.

Two charging paths to the RCS are provided. The normal path is into loop 1 through an air operated isolation valve, (CV-8146). A spring loaded check valve set to open at 200 psid is in parallel with the isolation valve. The purpose of this spring loaded check valve is to relieve the volumetric expansion of coolant if charging were to be isolated and letdown continued. The alternate charging connection (CV-8147) taps into loop 4, and can be used if the normal charging line is inoperable.

Pressurizer Auxiliary Spray

The charging header can supply spray to the pressurizer if the reactor coolant pumps are not running. Depressurization while on residual heat removal is a normal evolution requiring the use of the auxiliary spray valve.

The auxiliary spray line is routed from the outlet of the regenerative heat exchanger to the pressurizer spray line downstream of the normal pressurizer spray valves. Auxiliary spray flow is controlled by a remotely controlled air operated valve (CV-8145).

4.1.3.4 Seal Injection and Seal Return

Seal Injection Header

The reactor coolant pump seal injection header connects to the CVCS at the discharge of the charging pumps, and directs flow to the seal injection filter(s). The seal injection filters are 5 micron filters installed to collect particulate matter that could damage the reactor coolant pump seal faces. The filtered seal injection water is directed to each reactor coolant pump through individual injection lines.

A total seal injection flow of 32 gpm, controlled by the seal injection flow control valve (HCV-182), is divided equally among the four reactor coolant pumps. Each seal injection line contains a manual throttle valve, locked in a throttled position, used to balance pump seal injection flow rates, and a manually operated isolation valve.

Seal Return Header

A seal return flow of 3 gpm/RCP is returned to the CVCS. The individual pump seal return lines join together and exit the containment through one penetration. Motor operated isolation valves (MO-8112 and MO-8100), located on opposite sides of the containment building penetration, provide redundant isolation of the seal return line should a containment phase "A" isolation occur. A relief valve, located in the containment, routes seal return flow to the pressurizer relief tank (PRT) if the seal return motor operated isolation valves are closed. The relief valve lifts at 150 psig.

A filter is installed in the seal return piping to collect particulates from the seal return and excess letdown systems. The filtered seal return water passes through the seal water heat exchanger to the charging pump suction header.

The seal water heat exchanger is also used to cool the centrifugal charging pump recirculation flow. This heat exchanger is cooled by the component cooling water system.

4.1.3.5 Excess Letdown

Excess letdown Figure 4.1-1 is supplied from the loop 3 cold leg through excess letdown heat exchanger and isolation valves to the CVCS.

Excess Letdown Control Valves

Four remotely-operated valves are associated with the Excess Letdown System. Two valves (FCV- 8153, and 8154) are located on the excess letdown heat exchanger inlet. A valve (HCV-123) on the excess letdown heat exchanger outlet is positioned by a hand controller in the control room, and is used to throttle excess letdown flow. A three way valve (CV-8143) is used to direct excess letdown flow to the CVCS or to the reactor coolant drain tank (RCDT). Excess letdown flow is normally routed to the CVCS.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown flow and has a capacity, at full system operating pressure equal to that portion of the seal injection flow which enters into the RCS through the reactor coolant pump labyrinth seals (normally, 20 gpm).

The excess letdown heat exchanger can be employed either when normal letdown is temporarily out of service to maintain the reactor in operation or it can be used to supplement maximum letdown during the final stages of heatup. The letdown fluid flows through the tube side of the excess letdown heat exchanger and component cooling water is circulated through the shell.

4.1.4 System Features and Interrelations

4.1.4.1 Deborating Ion Exchangers

Some Westinghouse plants are equipped with deborating ion exchangers. Late in core life, the necessary boron concentration of the RCS approaches zero. At low boron concentrations, the volume of dilution required to further decrease the boron concentration is extremely large. The large dilution volume required must be processed as liquid radioactive waste because boron recovery becomes impractical at low boron concentrations.

The deborating ion exchangers provide a means of reducing RCS boron concentration without generating a large volume of liquid radioactive waste. The ion exchangers contain anion type resin and remove boric acid from the RCS by ion exchange.

4.1.4.2 Boron Thermal Regeneration System

To allow maximum load following while minimizing the amount of processing in the boron recovery system, some plants are equipped with a boron thermal regeneration system (BTRS). The BTRS takes the letdown flow at the outlet of the ion exchangers, adds or removes boron, and returns the flow to the inlet of the letdown filter. The BTRS is discussed in detail in Section 4.2.

4.1.4.3 Boron Concentration Monitoring System

All plants can determine boron concentration by obtaining a sample and taking it to the chemistry lab for analysis. Some plants have an on-line monitoring system that continuously measures the boron concentration with a remote readout in the control room.

The boron concentration monitoring system (BCMS) consists of a neutron radiation source and a sample chamber in a tank full of water. The water moderates and shields against neutron leakage. A BF3 proportional neutron detector is located in the center of the sample chamber to detect neutrons that get through the boron solution. The neutrons come from an Am-Be source located just outside the chamber. The time required to detect a given number of neutrons is proportional to the boron concentration.

Sample flow to the BCMS is from downstream of the ion exchangers (or upstream of the letdown filter). The return flow is via the relief line to the VCT.

4.1.4.4 Pressurizer Level Control

The CVCS maintains the proper water inventory in the RCS by adjusting the changing flow rate as pressurizer level deviates from its program. The error signal from the pressurizer level control system (Chapter 10.3) increases or decreases the charging system flow. The charging flow may be controlled by varying the speed of the positive displacement (PD) charging pump, or by varying the position of the charging flow control valve (FCV-121) if a centrifugal charging pump is in service. If the level in the pressurizer is below its setpoint, the PD pump speed is increased, or FCV-121 is opened to increase charging flow. If the level in the pressurizer is above its setpoint, then the PD pump speed is decreased, or FCV-121 is closed to decrease charging flow.

4.1.4.5 Purification During Residual Heat Removal Operation

A connection from the RHR located between the letdown line containment isolation valve (CV-8152) and the letdown heat exchanger, allows purification of the RCS while in cold shutdown. Purification is accomplished by directing a portion of the RHR flow to the CVCS. Flow from the RHR system to the CVCS letdown line is controlled by HCV-128. The mixed bed ion exchanger and letdown filter purify the coolant prior to its entry into the VCT. The purified coolant is then returned to the RCS by the normal charging flow path.

4.1.4.6 CVCS Alignment During ECCS Operation

The following valves close during an ESFAS:

1. Containment Isolation Valve (CV-8152) - closed by a Containment Phase "A" signal.
2. Seal return isolation valves (MO-8112 and MO-8100) - closed by a Containment Phase "A" signal.
3. VCT outlet valves (LCV-112B and LCV-112C) - closed by an ESFAS.
4. Charging flow control valves (MO-8105 and MO-8106) - closed by an ESFAS.
5. Centrifugal charging pump recirculation line isolation valves (MO-8110 and MO-8111) - closed by an ESFAS.

The following valves open during an ESFAS:

1. Charging Pump Supply from the RWST (LCV-112D and LCV-112E)
2. Boron injection tank inlet and outlet valves

Closing the above valves isolates letdown, normal charging, the VCT, and the seal return. The valves that open supply borated water to the RCS from the RWST via the BIT by the centrifugal charging pumps.

4.1.4.7 Volume Control Tank Level Functions

The level in the VCT controls the position of the level divert valve (LCV-112A), the VCT outlet valves (LCV-112B and LCV-112C), the RWST suction supply valves (LCV-112D and LCV-112E), and the reactor makeup system if the mode selector switch is in automatic. Figure 4.1-7 summarizes the VCT level control functions.

4.1.5 Summary

The CVCS is a major process system that controls RCS inventory, maintains RCS chemistry, provides the reactor coolant pumps with seal water, and controls the soluble poison concentration of the RCS. In addition, the CVCS supplies borated water to the RCS in the event of an accident.

The CVCS interfaces with the following system interfaces:

1. Pressurizer level control system
2. Reactor makeup system
3. Emergency core cooling system
4. Boron recovery system

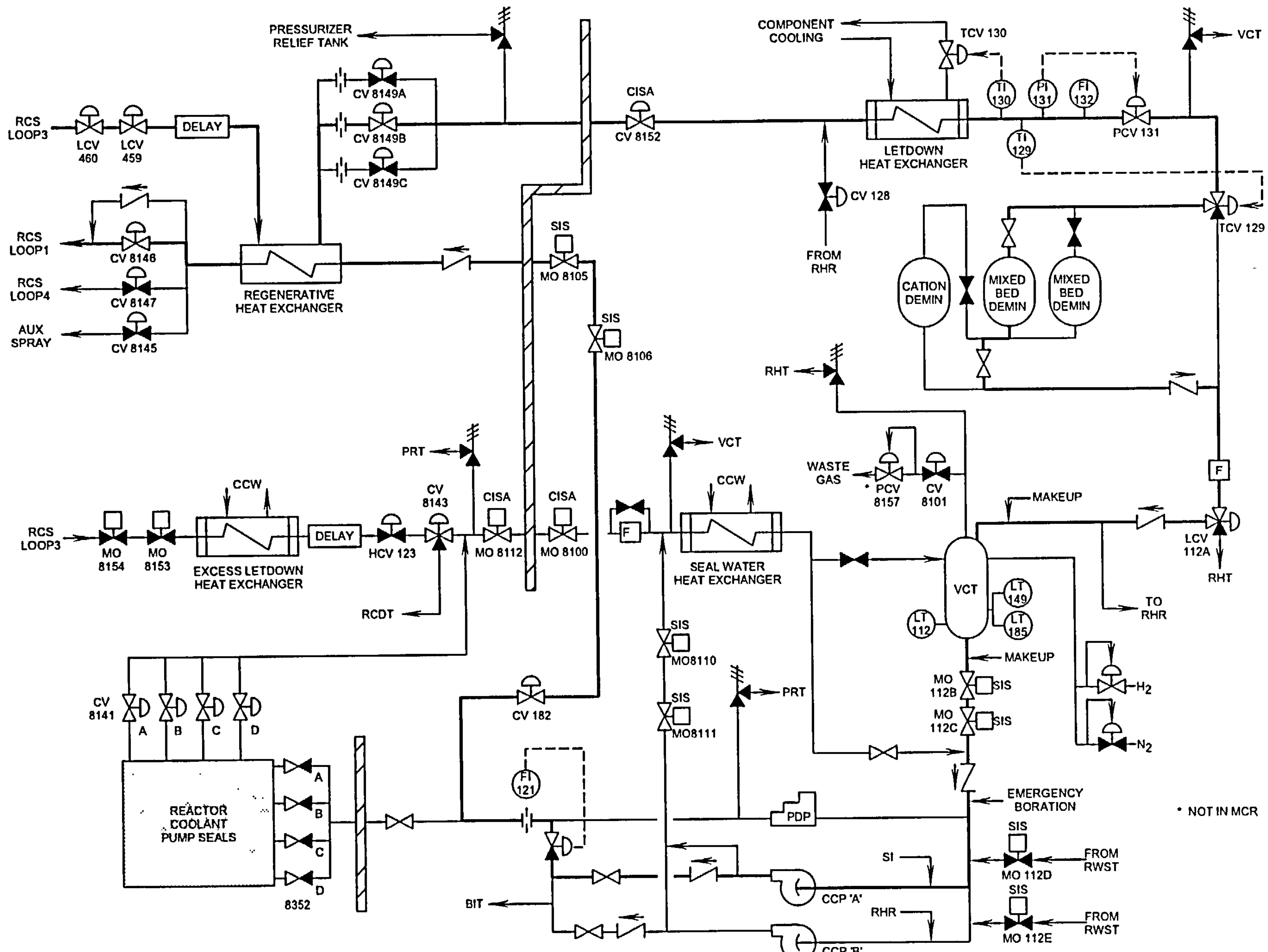


Figure 4.1-1 Chemical and Volume Control System

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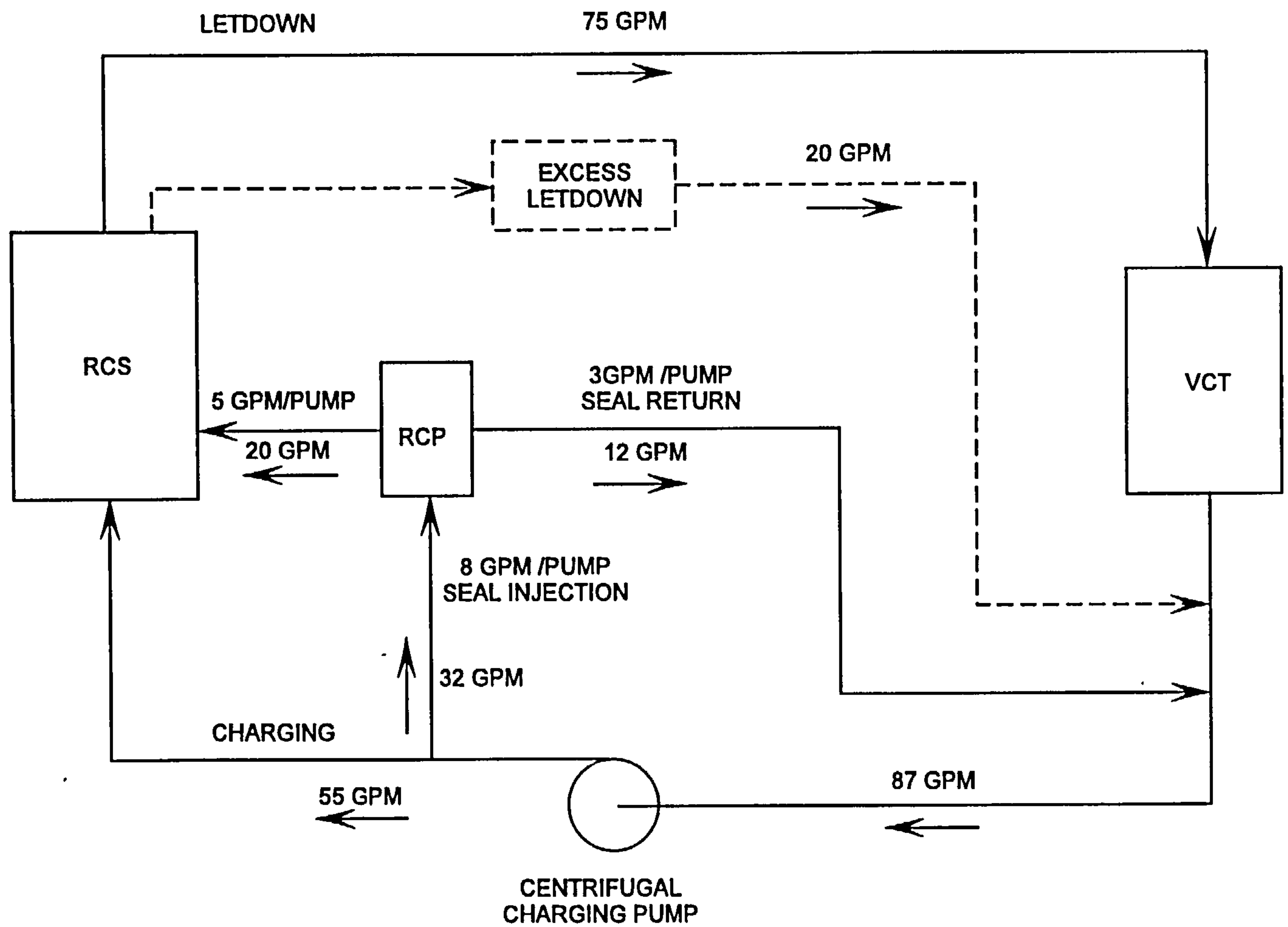

Figure 4.1-2 CVCS Flow Balance

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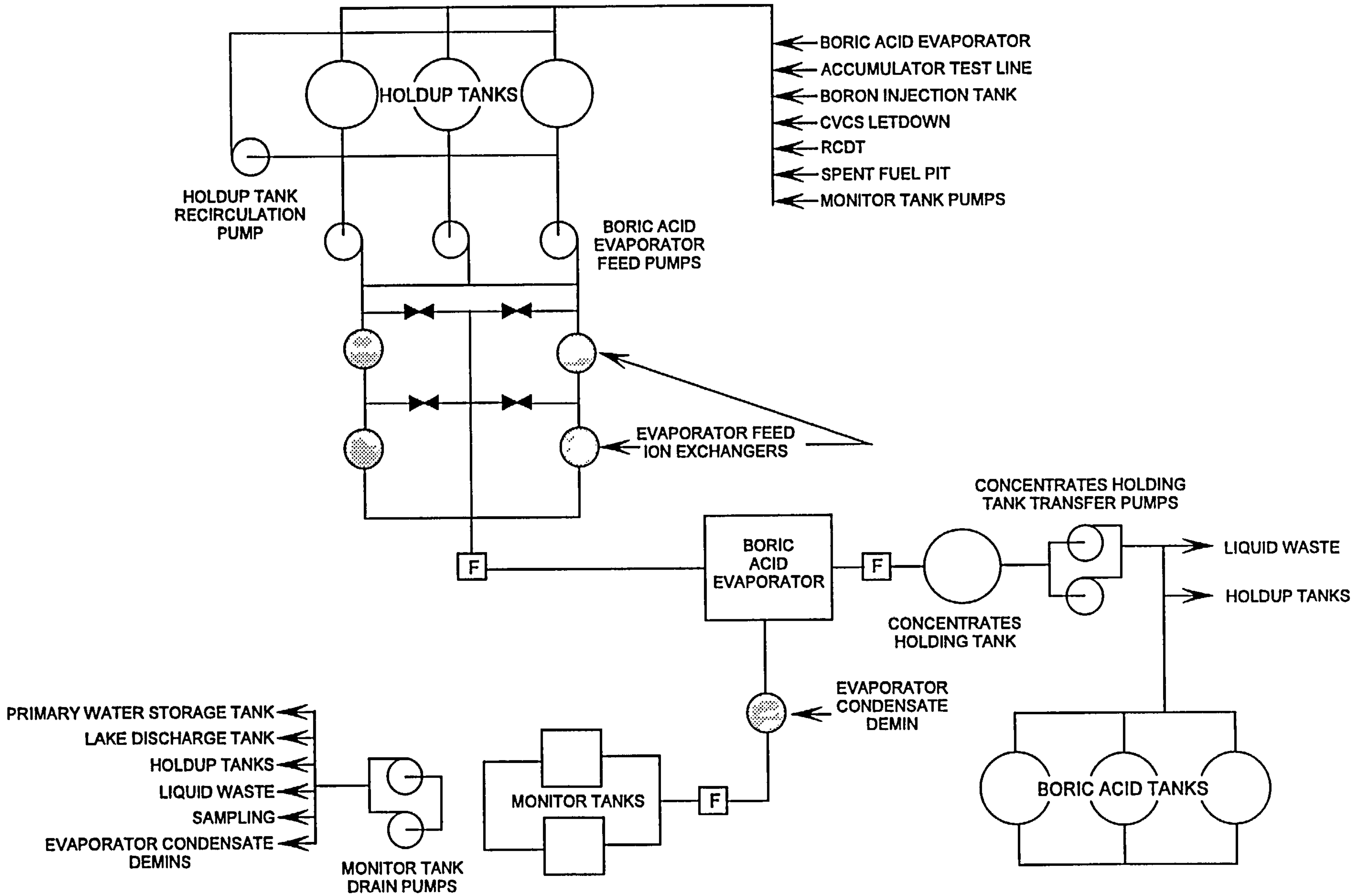

Figure 4.1-3 Boron Recycle System

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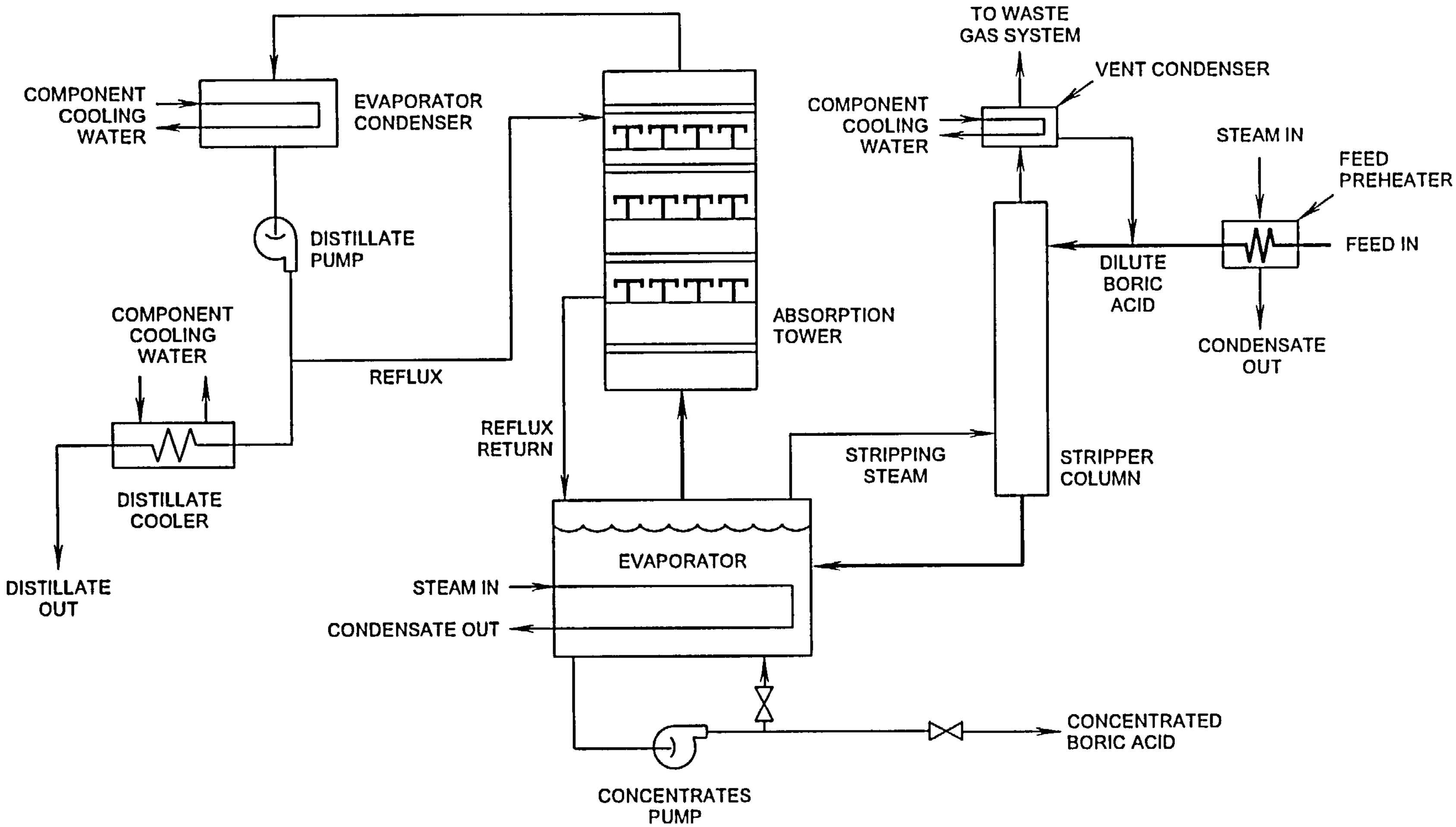

Figure 4.1-4 Boric Acid Evaporator

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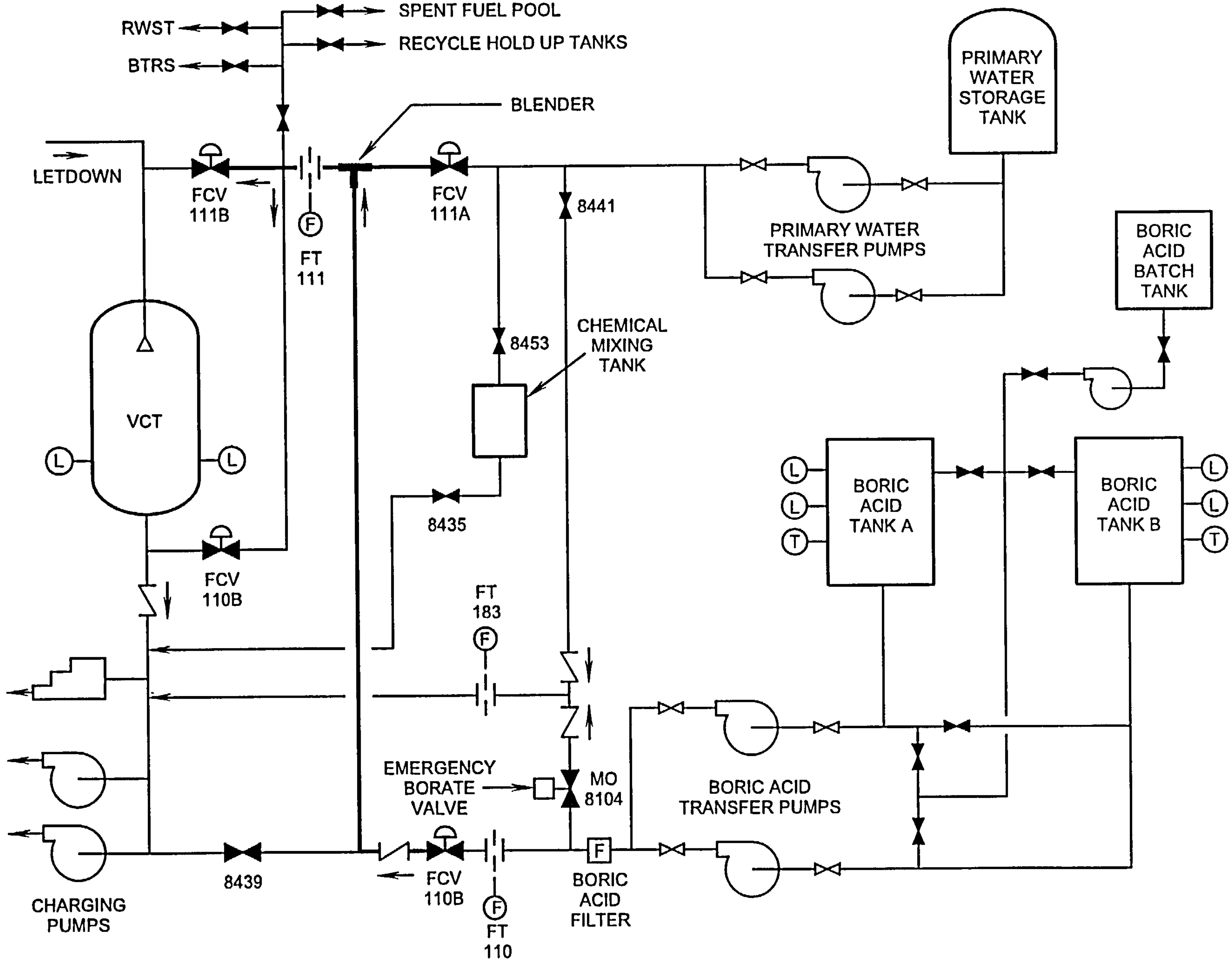

Figure 4.1-5 Reactor Makeup System

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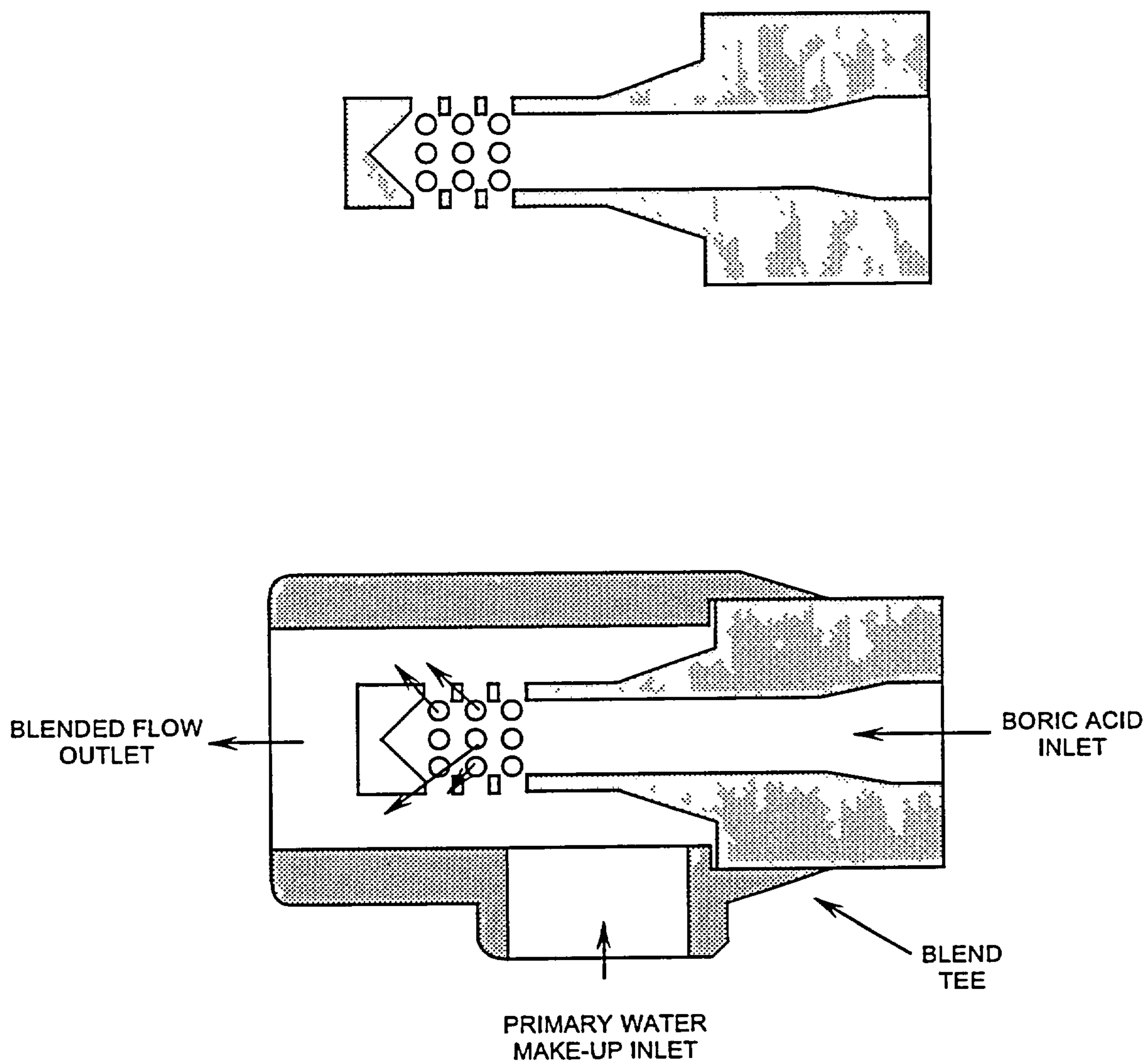

Figure 4.1-6 Boric Acid Blender

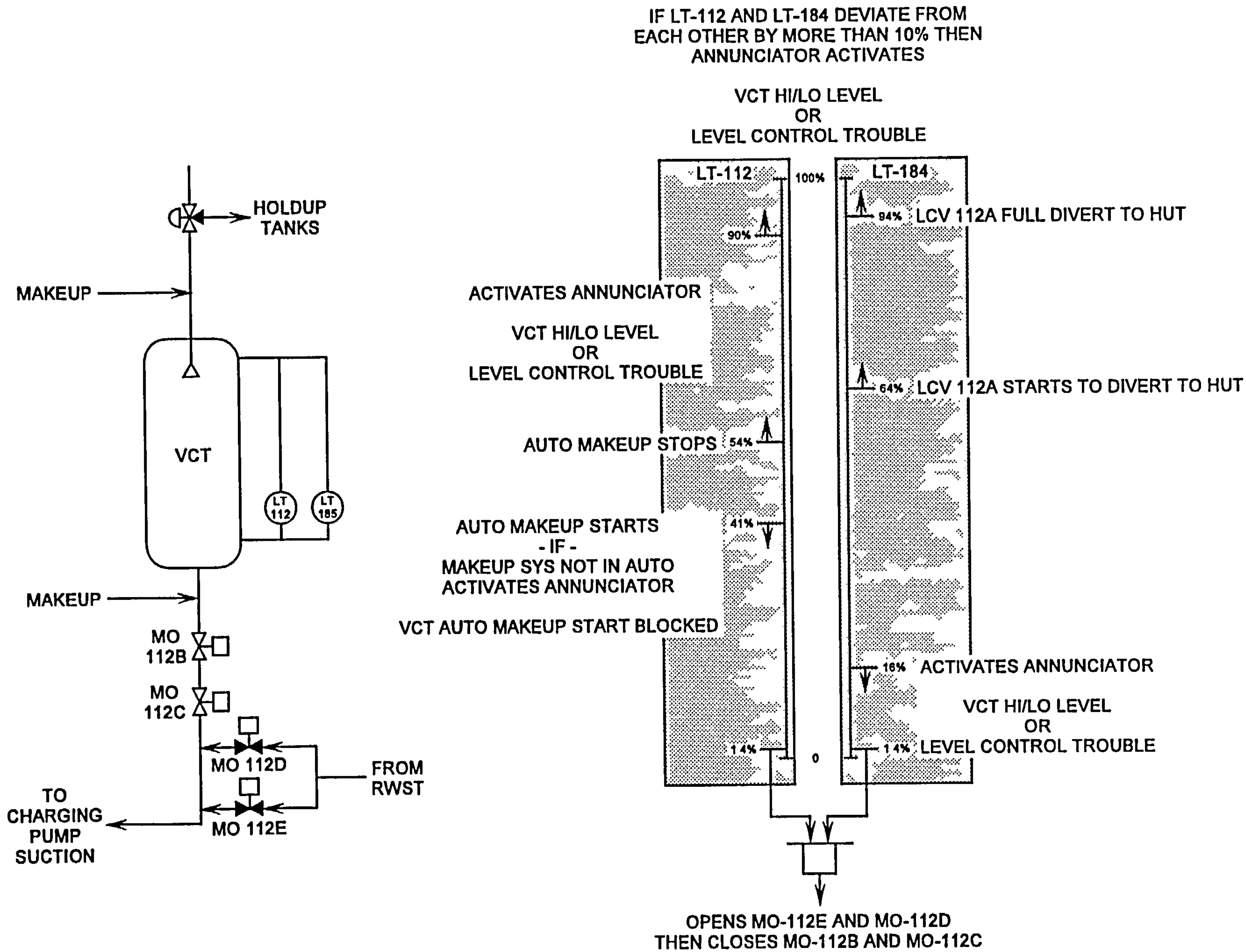

Figure 4.1-7 VCT Level Functions

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Section 4.2

Boron Thermal Regeneration System

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4.2 BORON THERMAL REGENERATION SYSTEM

Introduction

The Boron Thermal Regeneration System (BTRS) varies the reactor coolant system (RCS) boron concentration to compensate for xenon transients and other reactivity changes which occur when power is changed. The BTRS is installed to allow for maximum load following while minimizing the liquid radioactive effluents from boron concentration changes. The design load follow capability for the BTRS is 12 hours at 100% power, 3 hour ramp to 50% power, 6 hours at 50% power, and a 3 hour ramp to 100% power. The operation of the BTRS is described below.

4.2.1 System Operation

Downstream of the mixed bed demineralizers in the Chemical and Volume Control System (CVCS), the letdown flow can be diverted to the BTRS. The letdown flow, all or part, may be treated when boron concentration changes are desired for load follow operations. After processing, this fluid is returned to a point upstream of the reactor coolant filter in the CVCS letdown line. Storage and release of boron during load follow operations are determined by the temperature of the fluid entering the thermal regeneration demineralizers. A chiller which is cooled by using a separate refrigeration system, and a group of heat exchangers are employed to provide the desired fluid temperatures to the demineralizers for either storage or release operation of the system.

4.2.1.1 Dilution Mode

The flow path through the BTRS is different for the boron storage and release operations. During boron storage (dilution) (Figure 4.2-1), the letdown stream enters the moderating heat exchanger tube side and from there it passes through the letdown chiller heat exchanger tube side.

These two heat exchangers cool the letdown stream prior to entering the BTRS demineralizers. The letdown reheat heat exchanger is valved out on the tube side and performs no function during boron storage operations. The temperature of the letdown stream at the point of entry to the demineralizers is maintained automatically by the temperature control valve which regulates the shell side flow to the letdown chiller heat exchanger. After passing through the BTRS demineralizers, the letdown enters the moderating heat exchanger shell side, where it is heated by the incoming letdown stream.

The letdown flow then enters the normal letdown flowpath upstream of the RCS filter before going to the volume control tank. Therefore, for boron storage, a decrease in the boric acid concentration in the reactor coolant is accomplished by sending the letdown flow at relatively low temperatures to the thermal regeneration demineralizers. The resin, which was depleted of boron at high temperature during a prior boron release operation (borate), is capable of storing boron from the low temperature letdown stream. Letdown flow with a decreased concentration of boric acid leaves the demineralizers and is directed to the RCS via the CVCS normal charging flowpath.

4.2.1.2 Boration Mode

During the boron release operation (borate) (Figure 4.2-2), the letdown 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 letdown stream prior to its entering the resin beds. The temperature of the letdown at the point of entry to the demineralizers is maintained automatically by the temperature control valve which regulates the flow rate on the tube side of the letdown reheat heat exchanger. After passing through the demineralizers, the letdown stream enters the shell side of the moderating heat exchanger, passes through the tube side of the letdown chiller heat exchanger.

An increase in the boric acid concentration in the reactor coolant is accomplished by sending the letdown flow at relatively high temperatures through the thermal regeneration demineralizers. The higher temperature water flowing through the demineralizers releases boron which was stored by the resin at low temperature during a previous boron storage (dilution) operation. The boron enriched letdown is returned to the RCS via the CVCS normal charging flowpath.

Although the BTRS is primarily designed to compensate for xenon transients occurring during load follow operations, it can also be used to handle boron swings far in excess of the design capacity of the demineralizers.

For dilution during a reactor start-up and power escalation for example, the resin beds are first saturated with boron, then washed off to the boron recovery system, then again saturated and washed off. This operation continues until the desired dilution in the RCS is obtained.

As an additional function, a thermal regeneration demineralizer can be used as a deborating demineralizer, which would be used to dilute the RCS during very low boron concentrations at the end of the core cycle. To make such an operation effective, the effluent concentration from the resin bed must be kept very low, close to zero ppm boron. This low effluent concentration can be achieved by using fresh resin. The use of fresh resin can be coupled with the normal replacement cycle of the resin; one resin bed being replaced during each core cycle.

4.2.2 Summary

The BTRS provides a means of varying the boron concentration of the reactor coolant. This system reduces the operational demands on the boron recovery system thereby generating less liquid radioactive waste. The BTRS makes use of a temperature dependent ion exchange process in order to both remove boron from (dilute) and release boron to (borate) the CVCS letdown stream.

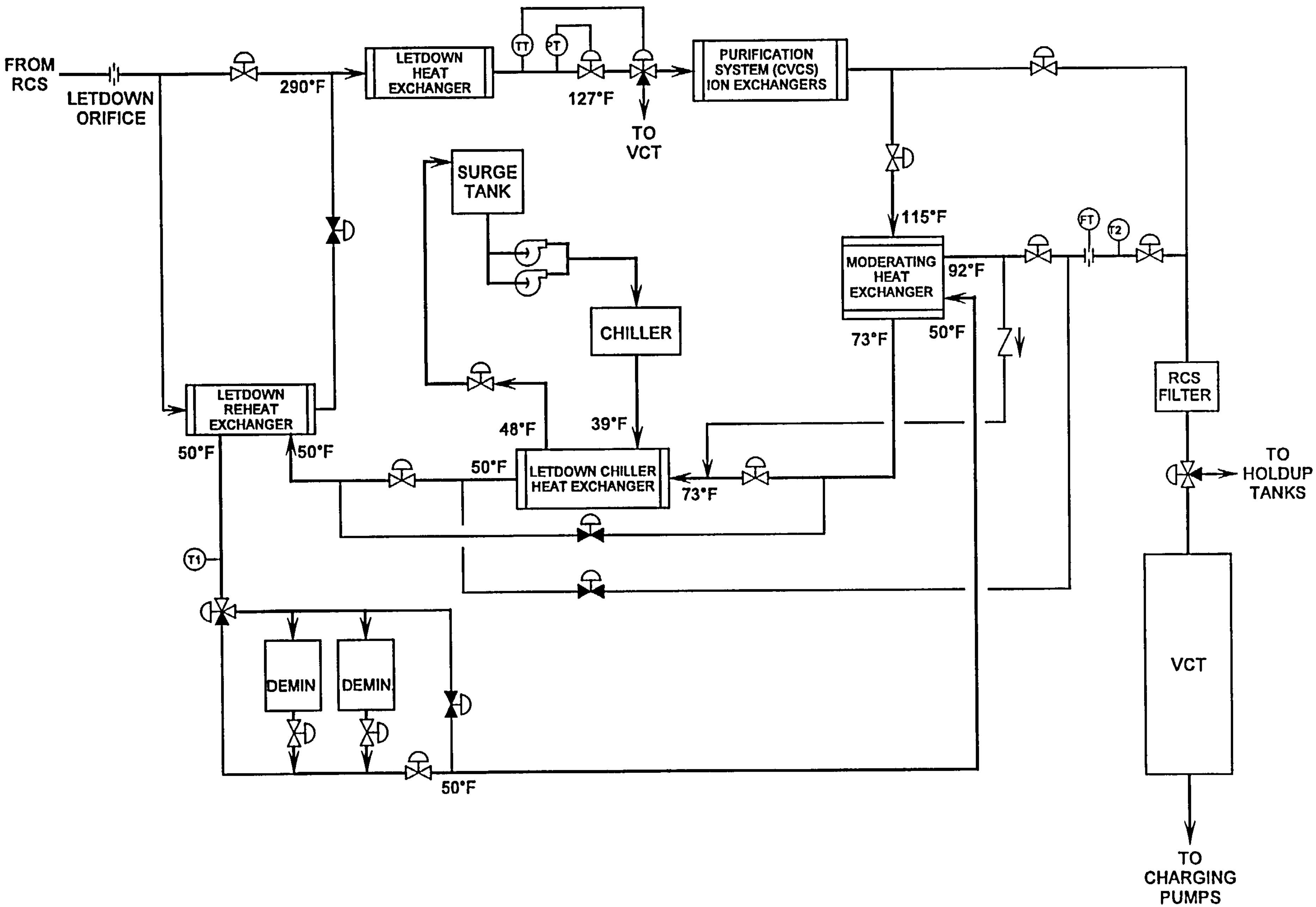

Figure 4.2-1 Boron Thermal Regeneration System (Dilution)

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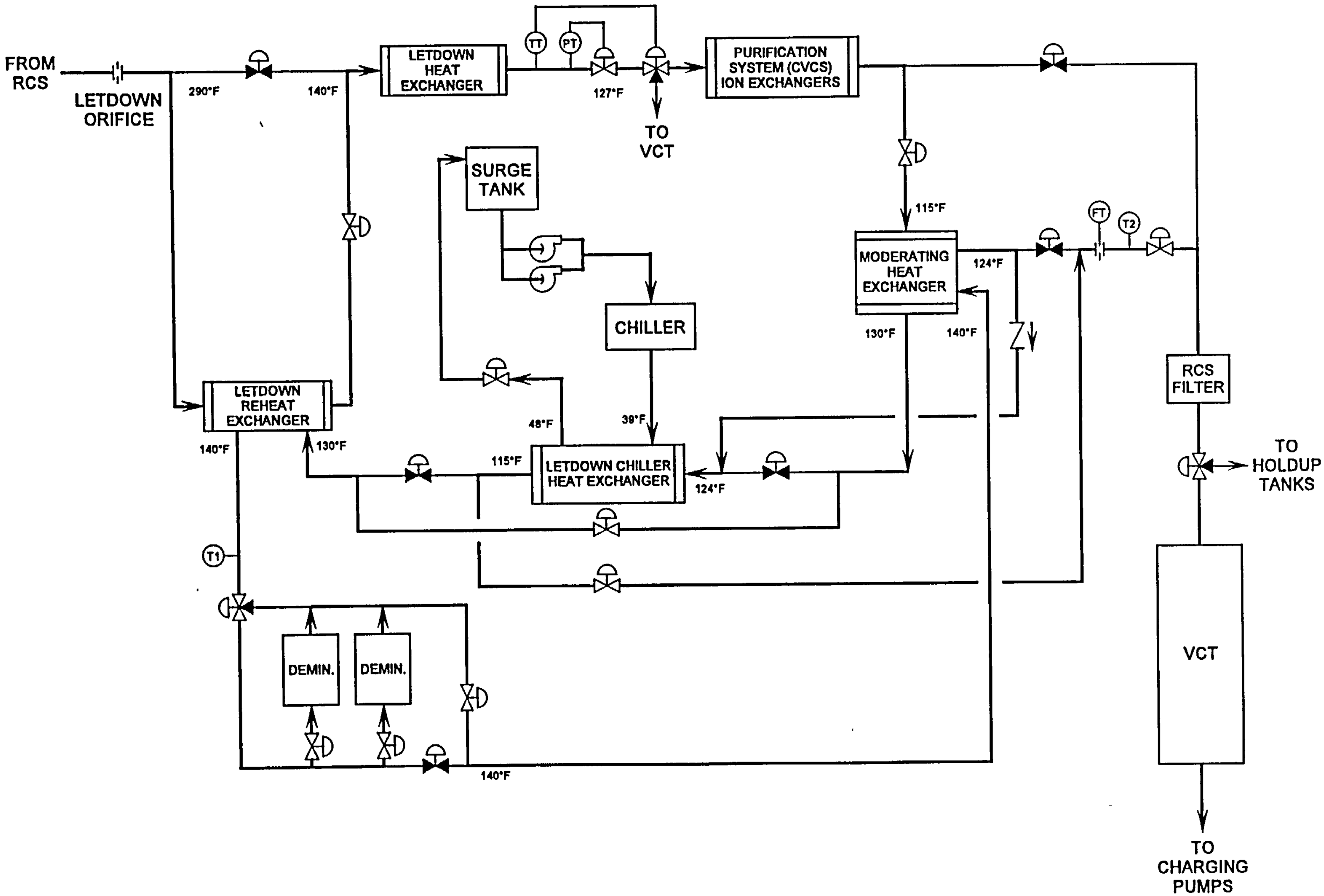

Figure 4.2-2 Boron Thermal Regeneration System (Boration)

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Section 5.0

Introduction to Engineered Safety Features

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5.0 INTRODUCTION TO ENGINEERED SAFETY FEATURES

Learning Objectives:

1. State the purposes of the Engineered Safety Feature (ESF) Systems.

2. Identify the three barriers designed to limit the escape of fission products to the environment.

3. Explain the following terms:
   a. Redundancy
   b. Physical and electrical separation
   c. Regulatory terms associated with safety

      - Important to Safety
      - Safety Related
      - Safety Grade

   d. Seismic Category I
   e. Diversity
   f. Single Failure
   g. Active Failure
   h. Passive Failure
   i. ESF Train

4. List the four design categories or conditions of operation, and give an example of each.

5. List the five acceptance criteria for the Emergency Core Cooling Systems (ECCS).

5.0.1 Introduction

In a large nuclear generating station with a core output rated at over 3,000 megawatts thermal, about 6 pounds of fission products are produced each day the unit is operated at full power. To protect the public from these fission products, a multi-barrier concept is incorporated into the design of the plant.

The first barrier consists of an enclosed cylindrical cladding that surrounds the fuel pellets and is designed to contain the fission products.

The second barrier is the Reactor Coolant System (RCS) pressure boundary. This barrier is designed to withstand high pressures and temperatures. The thickness of this barrier varies from a few inches for the reactor coolant loop piping, about 8 inches for the reactor vessel walls, and about one-tenth of an inch for the steam generator U-tubes. Since the RCS pressure boundary surrounds the first barrier, it will contain any fission products that escape from the cladding.

The final barrier of protection is the containment (reactor building). Many approved designs are used, but all contain the reactor coolant system and provide a third barrier to the release of radioactivity to the public.

The three barriers and the protection against the loss of each barrier is required by the Code of Federal Regulations. The reactor trips based upon Departure from Nucleate Boiling (DNB) and local power densities (kw/ft.) are installed to protect the first barrier (fuel cladding) from damage. The high pressure reactor trip and the pressurizer code safety valves provide protection for the second barrier (RCS) by limiting the pressure in the RCS to less than its design pressure. Finally, the design of the containment, its support systems, and allowable leakage specifications help to insure the integrity of the containment.

5.0.1.1 LOCA Description

Consider the effect on the fuel cladding and the containment if a gross failure of the reactor coolant system occurs without the benefit of the engineered safety features. When a Loss of Coolant Accident (LOCA) occurs, hot pressurized reactor coolant is forced out of the reactor coolant system, and pressure in the RCS rapidly decreases. An automatic reactor shutdown occurs because of this decrease in pressure and the control rods drop into the core. The fission process in the core is all but stopped and heat production drops to the decay heat value. As the coolant rapidly escapes via the break, containment temperature and pressure begin to increase. In a short time the reactor coolant system has flashed to steam and the pressure has equalized with the pressure inside the containment. At this time the blowdown phase of the loss of coolant accident has ended.

The fuel cladding begins to heat up following the blowdown phase of the accident. When the cladding temperatures exceed 2200°F, a zirconium-water reaction starts to occur. The hydrogen produced from this chemical reaction escapes into the containment, and the exothermic reaction causes an extra heat input into the cladding. The extreme pellet temperatures cause increased fuel rod pressures and the weakened cladding begins to bow. As the zirconium-water reaction continues, the zircalloy cladding begins to undergo metallurgical changes and becomes brittle, adding to its destruction.

As a result of the cladding degradation, fission products are released from the reactor coolant system to the containment through the RCS break. Two of the three barriers have now failed, and the third barrier is threatened. The containment pressure has increased due to the blowdown phase and will cause the escape of fission products to the environment through minute leakage paths. For some plant designs, the potential for a gross failure of the containment due to a hydrogen explosion is of great concern. For others the continuous heat input into the containment can cause the pressure and temperature to increase to the point where the containment could rupture.

However, the engineered safety feature systems are installed in nuclear units to mitigate the consequences of the loss of the reactor coolant system pressure boundary. Among these systems are the emergency core cooling systems which are actuated automatically during accident situations.

When an engineered safety features actuation occurs, borated water is pumped from the refueling water storage tank to the reactor coolant system, and the reactor vessel is refilled. This reflooding of the core from the refueling water storage tank is called the injection phase of the LOCA. With a flow of water into the core, decay heat is removed and the fission product release due to cladding failure is minimized. However, a few major points concerning ECCS should be made at this time.

First of all, if only one pump is installed and it failed, the consequences of a LOCA are the same as those previously discussed. But if a redundant pump that is also capable of supplying 100% of the required core cooling is installed, then the public can be protected. General Design Criterion (GDC) 35 of 10 CFR 50, Appendix A requires redundant emergency core cooling systems.

Next, if only one sensor is used to actuate the ECCS and it fails in a non-conservative direction, the core will not be reflooded, regardless of the number of ECCS pumps installed. If the sensor fails in the conservative direction, then an unnecessary actuation of ECCS equipment would occur. Therefore, IEEE-279 requires that redundant sensors, as well as redundant instrument strings, logic devices, and actuating devices be installed.

After providing protection for the first two contingencies as stated above, a loss of power to the emergency core cooling system pumps must be considered. The normal power supply to the electrical busses, which power the pump motors, comes from the electrical grid by the way of transformers, breakers, and bus work. Because this distribution system is vulnerable to thunderstorms, tornadoes, hurricanes, icing, and other acts of nature; a standby (emergency) power system is provided to insure a power supply to the ECCS pump motors.

Redundant diesel generators are normally used as the standby power supply. GDC 17 of 10 CFR 50, Appendix A outlines the requirements for electrical power distribution.

Finally, the emergency equipment must be designed to remain operational during a postulated seismic event. Paraphrasing 10 CFR 100, nuclear plant components designated to prevent or mitigate the consequences of accidents must be designed and built to remain functional during the design basis earthquake. Components satisfying this requirement are called Seismic Category I.

In summary, the emergency core cooling systems must be redundant, actuated by redundant sensors, powered from redundant electrical power sources, and designed to be operational during seismic events.

5.0.1.2 Emergency Core Cooling Systems

Certain systems are installed to provide emergency cooling to the core in the event of a LOCA. These systems include the high, intermediate, and low pressure injection systems, and the cold leg accumulators .

The high pressure injection system, consisting of two redundant trains, provides protection for Small Break Loss of Coolant Accidents (SBLOCA). This system pumps water from the Refueling Water Storage Tank (RWST) into the RCS cold legs during the injection phase. The high pressure system can also be used during the recirculation phase by connecting its suction to the residual heat removal system. In addition, this system injects boric acid from the RWST into the RCS during steam break accidents to offset the positive reactivity added by the rapid cooldown of the RCS.

The intermediate pressure injection system (safety injection system), provides protection for small to intermediate sized loss of coolant accidents. It has a greater capacity than the high pressure injection system but less than the low pressure injection system. Both safety injection system trains take a suction from the RWST during the injection phase of the accident. Like the high pressure injection system, it can also be used during the recirculation phase by connecting its suction to the residual heat removal system.

The low pressure injection system (residual heat removal system) provides protection for a large loss of coolant accident. This system consists of two trains which can pump large quantities of water at a low pressure from the RWST to the reactor coolant system. These pumps are also supplied with a suction from the containment emergency core cooling recirculation sump. This suction source will be used during the recirculation phase of the accident In addition to supplying cooling water to the reactor vessel, these pumps can also supply the suctions of the intermediate and high pressure injection systems.

The cold leg accumulators consists of four tanks filled with borated water and pressurized with nitrogen. When the RCS is pressurized and the reactor is at power, check valves, held closed by the higher reactor coolant system pressure, prevent the entry of water from the accumulators into the reactor coolant system. However, if a loss of coolant accident results in a reactor coolant system pressure lower than the pressure in the accumulators, then borated water will flow from the accumulators to the reactor vessel, via the cold legs. This system, unlike the high, intermediate, and low pressure injection systems, requires no actuation signal.

In addition to emergency core cooling systems, the engineered safety features design includes provisions to protect the containment barrier and to remove the core decay heat (auxiliary feedwater system).

5.0.1.3 Containment Barrier Protection

The containment spray system is installed to reduce the pressure inside the containment following a loss of coolant accident or a steam line break within the confines of the containment. The containment spray system satisfies the same design requirements as the emergency core cooling system. In addition to the containment spray system, other methods of controlling the containment pressure include containment fan coolers and hydrogen recombiners.

5.0.1.4 Decay Heat Removal

When the reactor trips, the rapid insertion of the control rods stops the heat production due to the fission process. However, the heat production from fission product decay continues. This decay heat is normally removed by converting feedwater to steam in the steam generators. The main feedwater system does not meet safety system design criteria and if it is lost, the steam generators will boil dry. When the secondary inventory is lost, the decay heat will not be removed from the core. As the fission product decay continues, the resultant heat production causes an increase in reactor coolant temperature and a corresponding increase in RCS pressure. If left unchecked, the temperature and pressure increases will result in a challenge to the reactor coolant system pressure boundary.

The auxiliary feedwater system is designed to provide feedwater to the steam generators to remove decay heat in the event of a loss of main feedwater. The auxiliary feedwater system meets the same design criteria as the emergency core cooling system.

The emergency systems installed to mitigate the consequences of accidents are among the most important systems in the nuclear plant. Normally these systems are aligned for the injection phase. However, if the safety system is in use performing one of its non-safety functions, it will be automatically re-aligned to perform its safety function upon receipt of an engineered safety features actuation signal.

5.0.2 Terms

5.0.2.1 Redundancy

If a component such as a pump is installed to provide an engineered safety feature, a second 100% backup pump must also be installed. This is known as redundancy. Redundant instrument sensors, instrument strings, and logic devices are required to ensure that no single failure will prevent at least one of these components from performing their intended function.

5.0.2.2 Physical and Electrical Separation

All engineered safety feature systems must be physically separated so that a catastrophic failure of one system will not prevent another engineered safety feature system from performing its intended function. Electrical power to the engineered safety features comes from the transmission grid via transformers, breakers and bus work. Because this distribution systems is vulnerable to thunderstorms, hurricanes, icing, and other acts of nature, a standby power system is provided to insure a reliable source of power. Redundant diesel generators are normally used for this standby power supply. These standby power supplies are normally referred to as trains: such as Train A and Train B.

5.0.2.3 Regulatory Terms Associated With Safety

- Important to Safety - Those structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. This encompasses the broad class of plant features that contribute in an important way to the safe operation and protection of the public in all phases and aspects of facility operations. This includes Safety Grade or Safety-Related as a subset.

- Safety-Related - Those structures, systems, or components designed to remain functional during a Safe Shutdown Earthquake (SSE) and are necessary to assure:

  1. The integrity of the reactor coolant system pressure boundary;
  2. The capability to shut down the reactor and maintain it in a safe shutdown condition; or
  3. The capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guidelines in 10CFR 100 Appendix A.

- Safety-Grade - This term is not explicitly used in regulations. It is equivalent to Safety-Related, and is a subset of Important to Safety.

5.0.2.4 Seismic Category I

Those structures, systems, and components, including their foundations and supports, designed to remain functional if the safe shutdown earthquake occurs have been designated as Seismic Category I.

The following list is typical of the systems and components designed to Seismic Category I.

1. The reactor coolant pressure boundary.
2. The reactor core and vessel internals, and
3. Systems or portions of systems that are required for emergency core cooling, post-accident containment heat removal, and post accident containment atmosphere cleanup.

Seismic Category I design requirements extend to the first seismic restraint beyond the identified boundaries.

5.0.2.5 Diversity

Diversity refers to different methods of providing the same safety protection or function. An example of this is the containment fan cooler system and the containment spray system. Both of these systems are designed to lower the pressure inside the containment following a steam break or a loss of coolant accident inside the containment.

5.0.2.6 Single Failure (Active/Passive)

A Single Failure is an occurrence which results in the loss of capability of a component to perform its intended safety function when called upon. Multiple failures resulting from a single occurrence are to be considered as a single failure. Fluid process systems are considered to be designed against an assumed single failure if neither a single active nor a single passive failure results in a loss of the safety function to the nuclear unit.

An Active Failure is a malfunction, excluding passive failures, of a component which relies on mechanical movement to complete its intended function on demand. Examples of active failures include the failure of a powered valve or a check valve to move to its correct position, or the failure of a pump, fan, or diesel generator to start.

Spurious action of a powered component

originating within its actuation system shall be regarded as an active failure unless specific design features or operating restrictions preclude such spurious actions.

A Passive Failure is a breech of a fluid pressure boundary or blockage of a process flow path.

The design flow limit for a passive failure (i.e. flow from a pressure boundary breech or flow through the blockage of a process flow path) is analyzed by the utility. This is done by performing an analysis of passive failures in the system, considering conditions of operation and possible failure or leakage modes as appropriate. For example, a review of a system that contains piping, heat exchangers, valves, flanged joints, and system interface barriers might result in the definition of a design leak rate for passive failure in that system based on the maximum flow through a failed packing or mechanical seal rather than complete severance of the piping.

In lieu of this analysis, the designer must consider complete severance of the piping and an independently complete blockage of the process path as a passive failure.

5.0.3 General Design Requirements

The engineered safety features systems are provided in nuclear power plants to mitigate the consequences of reactor plant accidents. Sections of the General Design Criteria (refer to Chapter 1.1) require that specific systems be provided to serve as ESF systems. Containment systems, Residual Heat Removal (RHR) system, emergency core cooling systems, containment heat removal systems, containment atmospheric cleanup systems and certain cooling water systems are typical of the systems required to be provided as ESF systems. Each of the ESF systems is designed to withstand a single failure without loss of its protective function capabilities during or following an accident condition.

5.0.3.1 Failure Criteria

The ESF systems are designed to withstand any single failure without loss of core and containment protection. However, this single failure is limited to either an active failure during the injection phase (a short period of time until the RWST empties) following an accident, or an active or a passive failure during the recirculation phase (a long period of time when water is taken from the containment recirculation sump, cooled by the RHR heat exchangers and pumped back into the RCS).

Most accidents are analyzed assuming a loss of off-site power conditions. This loss of off-site power is considered in addition the "single active failure".

5.0.3.2 ESF Train

The engineered safety features which contain active components are designed with two independent trains. One example of this is the emergency core cooling system (ECCS) of which either train can satisfy all the requirements to safely shut down the plant or meet the final acceptance criteria following an accident.

As an example of this, the Technical Specifications state that one ECCS train consists of:

1. One centrifugal charging pump.
2. One safety injection pump.
3. One RHR pump.
4. An operable flow path from the RWST to the RCS and from the containment sump back to the RCS.
5. Power supplies and instrumentation for the

above items.

To guarantee that at least one train of the ECCS is available if an accident occurs, both trains must meet the operability requirements in the Technical Specifications whenever the temperature of the RCS is above 350°F. This will allow for the single active failure of the emergency power source (diesel generator), and still retain one train of ECCS for accident mitigation.

5.0.3.3 Typical Engineered Safety Features Systems

The typical ESF systems that may be found at a Westinghouse nuclear power plant are listed below.

1. Containment - The Seismic Category I building that provides a virtually leak-tight barrier to prevent the release of fission products to the environment.
2. Containment heat removal systems - These systems, containment spray and containment fan coolers, reduce the pressure and temperature inside the containment and remove fission products from the containment atmosphere.
3. Containment isolation system - Provides isolation capability for the various lines penetrating the containment.
4. Containment combustible gas control system - The hydrogen recombiners control the concentration of hydrogen gas which may be released to the containment following an accident to assure containment integrity.
5. Emergency core cooling systems - Deliver the borated water from the RWST to the core following various postulated accidents.
6. Habitability systems - Provide the control room with adequate shielding, air purification and climatic control.
7. Auxiliary feedwater system - Provides make up to the steam generators whenever the main feedwater system is not available. It also maintains heat removal capability of the steam generators by heat transfer from the RCS.
8. Class 1E electrical system - The vital electrical distribution system, including the emergency diesel generators, which provides a reliable source of power to the active components of the ESF.
9. Essential support systems - Any system or subsystem (component cooling water, essential raw cooling water, etc.) which would be required to operate in the support of the above systems.

5.0.3.4 Accident Analysis

Each facility shall, as a condition of licensing, include in its Final Safety Analysis Report (FSAR) a section titled Accident Analysis. The requirements for the information contained in this section are outlined in 10 CFR 50.34 and are detailed in Regulatory Guide 1.70 Standard Format and Content of Safety Analysis Reports.

This Regulatory Guide sets up a standard approach to ensure the license applicant has analyzed both expected and unexpected transients and accidents, that conservative values are applied in the analysis, and that the plant design has incorporated an adequate margin of safety.

The accident analysis section of the FSAR shall include the analysis of several transients and accidents. These analyzed incidents are placed in four (4) categories or conditions of operation.

5.0.3.5 Design Categories or Conditions of Operation

Condition I - Normal Operations

Condition I events are those occurrences which are expected to happen frequently or regularly in the course of power operation, refueling, maintenance, or changing power in the plant. Therefore, Condition I occurrences are accommodated with a margin between any plant parameter and the value of that parameter which would require either automatic or manual protective actions. Since Condition I events occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions which can occur during Condition I operation. Below is a typical list of Condition I events:

1. Steady state and shutdown operations.
   a. Power operation (15-100% of full power)
   b. Start up (or standby) (critical, 0-15% of full power)
   c. Hot shutdown (subcritical, residual heat removal system in operation)
   d. Refueling

2. Operation with permissible deviations- Various deviations which may occur during continued operation as permitted by the plant Technical Specifications must be considered in conjunction with other operational modes. These include:
   a. Operation with components or systems out of service (such as power operation with a reactor coolant pump out of service)
   b. Leakage from fuel with clad defects
   c. Radioactivity in the reactor coolant
      - Fission Products
      - Corrosion Products
      - Tritium
   d. Operation with steam generator leaks up to the maximum allowed by the Technical Specifications
   e. Testing as allowed by the Technical Specifications

3. Operational testing
   a. Plant heatup and cooldown (up to 100°F/hour for the RCS; 200°F/hour for the pressurizer cooldown only)
   b. Step load changes (up to 10%/minute)
   c. Ramp load changes (up to 5%/minute)
   d. Load rejection up to and including a full load rejection transient

Condition II - Faults of Moderate Frequency

These faults, which are expected to happen on a once per year basis and at worst, result in a reactor trip with the plant being capable of returning to operations. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., a Condition III or IV event. In addition, Condition II events are not expected to result in fuel rod failures or the over pressurization of the RCS. The operational events that meet the above assumptions have been grouped below:

1. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition.
2. Uncontrolled rod cluster control assembly bank withdrawal at power.
3. Rod cluster control assembly misalignment.
4. Uncontrolled boron dilution.
5. Partial loss of forced reactor coolant flow.
6. Start-up of an inactive reactor coolant loop.
7. Loss of external electrical load and/or turbine trip.
8. Loss of normal feedwater.
9. Loss of offsite power to the station auxiliaries.
10. Excessive heat removal due to feedwater system malfunctions.
11. Excessive load increase incident.
12. Accidental depressurization of the RCS.
13. Accidental depressurization of the main steam system.
14. Inadvertent operation of emergency core cooling system during power operation.

Condition III - Infrequent Faults

Condition III occurrences are faults which are expected to happen once in the forty year life of the plant. This event will result in the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of power operations for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system or containment barriers. The following faults have been grouped into this category:

1. Loss of reactor coolant from small ruptured pipes or from cracks in large pipes which actuates the emergency core cooling system.
2. Minor secondary pipe breaks.
3. Inadvertent loading of a fuel assembly into an improper position.
4. Complete loss of forced reactor coolant flow.
5. Single rod cluster control assembly withdrawal at full power.

Condition IV - Limiting Faults

Condition IV occurrences are faults which are never expected to take place but are postulated because the events are so severe that the consequences would include the potential for the release of significant amounts of radioactive material. These accidents are the most drastic and therefore represent the limiting design cases. Condition IV faults shall not cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of the guideline values of 10 CFR Part 100. A single Condition IV fault shall not cause a consequential loss of required functions of systems needed to cope with the fault, including those of the emergency core cooling system and the containment. The following faults have been classified in this category:

1. Major rupture of pipes containing reactor coolant up to and including the double ended rupture of the largest pipe in the reactor coolant system (Loss of Coolant Accident).
2. Major secondary system pipe ruptures.
3. Steam generator tube rupture.
4. Single reactor coolant pump locked rotor.
5. Fuel handling accident.
6. Rupture of a control rod drive mechanism housing (rod cluster control assembly ejection).

5.0.3.6 Acceptance Criteria for the Emergency Core Cooling Systems (10 CFR 50.46)

In 1974, the final acceptance criteria for emergency core cooling systems for light water reactors were established. These criteria required the ECCS to be designed and built so that the calculated cooling performance following a full spectrum of postulated LOCA's would meet the following limits during or following the accident.

1. Peak Cladding Temperature (PCT) < 2200°F.
2. Maximum cladding oxidation shall not exceed 17% of the total cladding thickness.
3. Maximum hydrogen generated by the reaction of water or steam with the cladding shall not exceed 1% of the amount that would be generated if all the cladding reacted chemically with water or steam.
4. Coolable geometry - Any changes in core geometry shall be such that the core remains in a coolable configuration.
5. Long-term cooling - Core temperature, after the injection phase, shall be maintained at an acceptably low value. Decay heat shall be removed for the extended period of time required.

5.0.3.7 Engineered Safety Features Actuation Signals

The engineered safety features of a facility will be actuated by the signals listed below. It should be noted that all ESF actuation signals achieve identical results. That is, regardless of the protection signal being generated, each signal causes the same equipment to be operated.

The engineered safety features actuation signals, often referred to as the Safety Injection (SI) signal, are listed below:

1. High steam line flow - coincident with low steam line pressure or low-low T_{avg}.
2. High steam line differential pressure - One steam line lower than two of the remaining three by 100 psid/or more.
3. Low pressurizer pressure - Two of three or two of four pressurizer pressure instruments less than setpoint.
4. High containment pressure - Two of three or two of four containment pressure instruments greater than setpoint.
5. Manual - one of two control board switches.

An additional signal is the containment spray actuation signal which actuates the containment spray system when containment pressure reaches approximately half of design pressure.

5.0.4 Typical Analysis Limits and Assumptions

For conservatism, each accident that is analyzed assumes the most conservative conditions, setpoints, equipment operability, and other factors which could conceivably affect the severity of the event. Listed below are some of the assumptions used in the accident analysis.

1. Maximum time delays for reactor trip, safety injection actuation, steam line isolation valve closure, etc., are assumed.
2. Starting values for the various plant parameters will be assumed to be at their worst case conditions.
3. Plant history, reactivity coefficients and other variables affecting the accident will be chosen to produce a more severe transient.

5.0.4.1 Steam Line Break Accident Analysis

The steam break accident is outlined here as an example of the goals and concerns that are assumed in performing accident analysis.

The analysis is performed to demonstrate that:

1. There is no consequential damage to the primary system and the core remains intact.
2. Energy release for the worst case break does not cause failure of the containment.
3. There is no return to criticality after the reactor trip for a break equivalent to a stuck open steam bypass, relief, or safety valve.

5.0.4.2 Assumptions for Steam Line Break Analysis

The assumptions used for the steam line break accident are listed as follows:

1. The design end of life shut down margin at no-load, equilibrium Xenon conditions, and with the most reactive rod stuck in its fully withdrawn position.
2. The negative moderator temperature coefficient corresponding to the end of life rodded core with the most reactive rod in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included.
3. Minimum capability for injection of concentrated boric acid solution corresponding to the most restrictive single active failure in the safety injection system. This corresponds to

the flow delivered by one centrifugal charging pump delivering its full contents to the cold leg header.

4. Design value of the steam generator heat transfer coefficient including allowance for tube fouling.
5. Hot channel factors corresponding to the worst stuck rod at end of core life.
6. Several combinations of break sizes and initial plant conditions have been considered in determining the core power transient which can result from large area pipe breaks.
   a. Complete severance of a pipe downstream of the steam flow restrictor with the plant initially at no load conditions and all reactor coolant pumps running. (Figure 5.0-1).
   b. Complete severance of a pipe inside the containment at the outlet of the steam generator with the same plant conditions as above (Figure 5.0-2).
   c. Case (a) above with loss of outside power simultaneous with the generation of the safety injection signal (loss of ac power results in coolant pump coast down).
   d. Case (b) above with the loss of offsite power simultaneous with the safety injection signal. A fifth case was analyzed consistent with the criterion stated earlier that there should be no return to criticality after reactor trip in the event of the spurious opening of a steam bypass or relief valve.
   e. A break equivalent to a steam flow of 247 lbs. per second at 1100 psi from one steam generator with offsite power available.
7. Initial hot shutdown conditions were considered for all of the above cases since this represents the most pessimistic initial condition for the accident.
8. The containment pressure response was evaluated for case (b) above, since this results in the most severe steam break containment transient. In addition to the full contents of the faulty steam generating unit being delivered to the containment, it was also assumed that break flow was delivered from the other three steam generators through a failed non-return valve in the broken steam line. The latter flow continues until the main steam stop valves close in the intact steam lines.
9. Perfect moisture separation in the steam generator is assumed. The assumption leads to conservative results since, in fact, considerable water would be discharged. Water carryover would reduce the magnitude of the temperature decrease in the core and the pressure increase in the containment.

5.0.4.3 Systems Which Provide Protection Against Steam Line Break Accidents

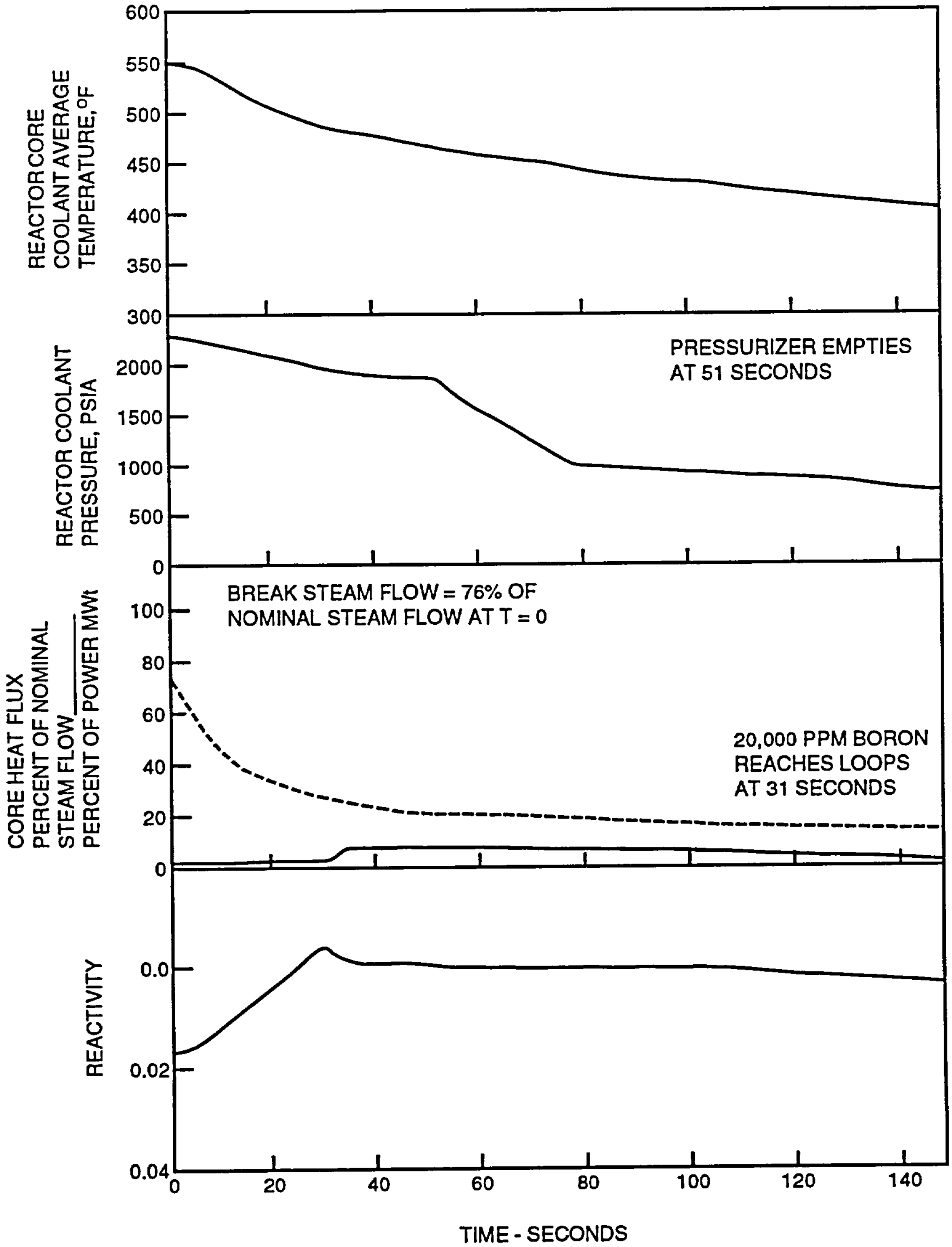
The following signals will be generated to actuate: a reactor trip, a safety injection, or components to mitigate the consequences of a main steam line rupture.

1. Safety Injection (ESF) Actuation
   a. High steam line flow
   b. High steam line differential pressure
   c. Low pressurizer pressure
   d. High containment pressure
2. Reactor protection system actuation
   a. High neutron flux
   b. Overpower delta T
   c. Safety injection reactor trip
3. Main feedwater isolation
   a. Close feedwater isolation valves
   b. Close feedwater control valves
   c. Close feedwater control bypass valves
   d. Trip main feedwater pumps
4. Trip of main steam stop valves, main steam check valve will prevent blowdown of more than one steam generator for any break location.
5. Steam line flow venturi will limit break flow.

5.0.5 Summary

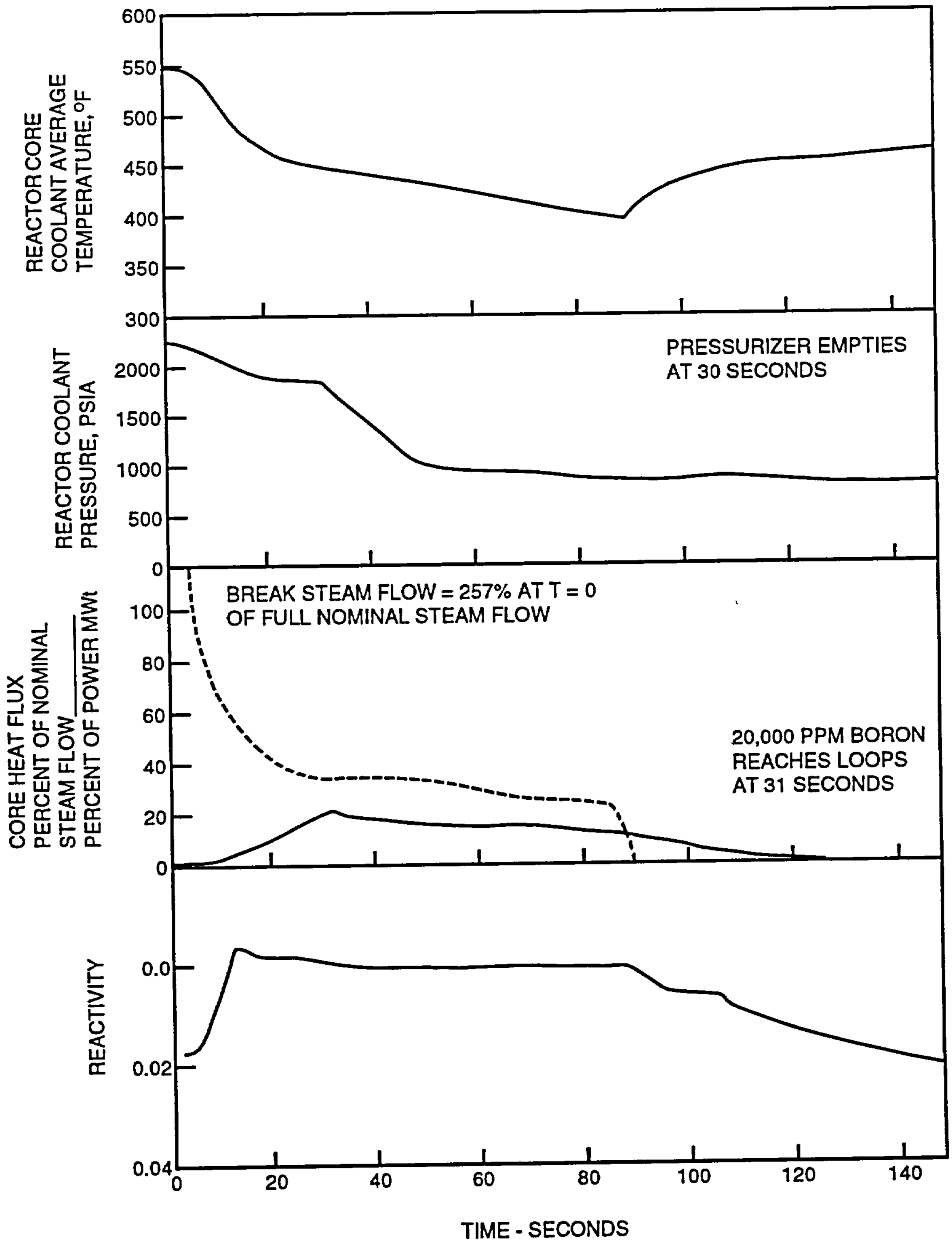
To protect the public from fission products, a multi-barrier of protection is incorporated in the design of a commercial pressurized water reactor.

If the reactor enters unsafe regions of operation the reactor protection system will generate a reactor trip to protect the cladding and reactor coolant system pressure boundary (first two barriers of protection). If an accident has occurred, such as a loss of coolant accident (LOCA), the reactor protection system actuates the engineered safety feature systems to mitigate the consequences of the accident and protect the containment (third and final barrier of protection). The engineered safety feature systems must be redundant, actuated by redundant sensors and instrument strings, physically separated, powered from redundant electrical power supplies, and be designed to remain operational during seismic events.



Safety Injection with Offsite Power

Figure 5.0-1 Steam Line Break Downstream of Flow Measuring Nozzle



Safety Injection with Offsite Power

Figure 5.0-2 Steam Line Break at Exit of Steam Generator

Westinghouse Technology Systems Manual

Section 5.1

<u>Residual Heat Removal System</u>

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5.1 RESIDUAL HEAT REMOVAL SYSTEM

Learning Objectives:

1. State the purposes of the Residual Heat Removal (RHR) System.
2. Describe the RHR system flow path including suction supplies, discharge points and major components during decay heat removal.
3. Describe the normal, at-power line-up of the RHR system.
4. Explain why Reactor Coolant System (RCS) pressure and temperature limits are placed on the initiation of RHR cooldown.
5. Explain how the RHR system is protected against over pressurization.
6. Explain how an intersystem LOCA is initiated in the residual heat removal system and what affect it can have on long-term core cooling.

5.1.1 Introduction

The purposes of the residual heat removal system are as follows:

1. Removes decay heat from the core and reduces the temperature of the RCS during the second phase of plant cooldown.
2. Serves as the low pressure injection portion of the Emergency Core Cooling System (ECCS), following a loss of coolant accident.
3. Transfers refueling water between the refueling water storage tank and the refueling cavity before and after refueling.

The RHR system transfers heat from the reactor coolant system to the component cooling water system. During shut down plant operations, the RHR system is used to remove the decay heat from the core and reduce the temperature of the reactor coolant to the cold shutdown temperature (<200°F). The cooldown performed by the RHR (from 350°F to <200°F), is referred to as the second phase of cooldown. The first phase of cooldown is accomplished by the Auxiliary Feedwater (AFW) system (Chapter 5.8), Steam Dump Control system (Chapter 11.2), and the steam generators.

Once the plant is in cold shutdown, the RHR system will maintain RCS temperature until the plant is started up again. The residual heat removal system also serves as part of the emergency core cooling system during the injection and recirculation phases of a loss of coolant accident. The residual heat removal system is used to transfer refueling water between the refueling water storage tank and the refueling cavity before and after the refueling operations.

5.1.2 System Description

The residual heat removal system as shown in Figure 5.1-1 consists of two heat exchangers, two residual heat removal pumps, and the associated piping, valves, and instrumentation necessary for operational control. The inlet lines to the residual heat removal system for the second phase of cooldown are connected to the hot leg of reactor coolant loop 4, and the return lines are connected to each cold leg of the reactor coolant system. These return lines also function as the emergency core cooling system low pressure injection lines.

The RHR pump suction line from the reactor coolant system is normally isolated by two series motor-operated valves (8701&8702). The suction line has a relief valve located downstream of the isolation valves, all of which are located inside the containment. Each RHR pump discharge line is isolated from the reactor coolant system by two check valves located inside the containment and by two normally open motor-operated valves (8809A and 8809B) located outside the containment. These motor operated valves are part of the emergency core cooling system and receive confirmatory open signals from the engineered safety features actuation system. During the second phase of cooldown, reactor coolant flows from the RCS to the residual heat removal pumps, through the tube side of the RHR heat exchangers, and back to the RCS. The heat from the reactor coolant is transferred to the component cooling water which is circulating through the shell side of the RHR heat exchangers.

If one of the two pumps or one of the two heat exchangers is not operable, the ability to safely cooldown the plant is not compromised; however, the time required for the cooldown is extended. When the residual heat removal system is in operation, the water chemistry requirements are the same as that of the reactor coolant system. Provisions are made for extracting samples from the flow of reactor coolant downstream of the RHR heat exchangers for analysis. A local sampling point is also provided on each residual heat removal train between the pump and its associated heat exchanger.

To insure the reliability of the RHR system, the two residual heat removal pumps are powered from separate vital electrical power supplies. If a loss of offsite power occurs, each vital bus is automatically transferred to a separate emergency diesel power supply. A prolonged loss of offsite power would not adversely affect the operation of the residual heat removal system.

The residual heat removal system is normally aligned to perform its safety function. Therefore, no valves are required to change position. For the RHR system to perform its safety function, the RHR pumps must start when the engineered safety features actuation signal is received, ar the pressure in the reactor coolant system must drop below the discharge pressure of the RHR pumps.

The materials used to fabricate the RHR system components are in accordance with the applicable ASME code requirements. All parts or components in contact with borated water are fabricated of, or clad with austenitic stainless steel or an equivalent corrosion resistant material.

5.1.3 Component Description

5.1.3.1 Residual Heat Removal Pumps

Two pumps are installed in the residual heat removal system. The pumps are vertical, centrifugal units with mechanical seals on the shafts. These seals can be cooled by either component cooling water or service wate depending on the plant design. All pump surfac in contact with reactor coolant are manufactured from austenitic stainless steel or an equivalent corrosion resistant material. The pumps are sized to deliver reactor coolant flow through the residual heat exchangers to meet the plant cooldown requirements.

The residual heat removal pumps are protected from overheating and loss of suction flow by minimum flow bypass lines that assure flow to the pump suction for pump cooling. A control valve located in each minimum flow line (MO-610, 611) is regulated by a signal from the flow transmitters located in each pump discharge header. These control valves open when the RHR pump discharge flow is less than 500 gpm and the pump is running, and close when the flow exceeds 1000 gpm or the pump is not running. A pressure sensor in each pump header provides a signal for an indicator on the main control board. A hig pressure annunciator alarm is also actuated by t pressure sensor.

5.1.3.2 Residual Heat Removal Heat Exchangers

Two heat exchangers are installed in the system. The heat exchanger design is based on the heat load and temperature differences between reactor coolant and component cooling water twenty hours after the reactor has been shutdown. The temperature difference between these two systems at that time is at its minimum, thus creating the minimum heat transfer capability.

The heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes while component cooling water circulates through the shell. The tubes are welded to the tube sheet to prevent leakage of reactor coolant.

5.1.3.3 Residual Heat Removal System Valves

The valves that perform a modulating function are equipped with two stem packing glands and an intermediate leak-off connection that discharges to the drain header.

Manual and motor operated valves have backseats to facilitate repacking and to limit stem leakage when the valves are open. Leakage connections are provided where required by valve size and fluid conditions. The suction line from the reactor coolant system is equipped with a pressure relief valve sized to relieve the combined flow of all the charging pumps at the relief valve set pressure. This relief valve is installed to provide over pressure protection for the reactor coolant system under solid plant operations.

Each discharge line to the reactor coolant system is equipped with a pressure relief valve to relieve the maximum possible back-leakage through the check valves which separate the residual heat removal system from the reactor coolant system. The design of the residual heat removal system includes two isolation valves (8701 and 8702) in series on the inlet line between the high pressure reactor coolant system and the lower pressure RHR system.

Each isolation valve is interlocked with one of two independent reactor coolant system pressure transmitters. These interlocks prevent the valves from being opened unless the reactor coolant system pressure is less than 425 psig to ensure that the RHR system is not over pressurized. After the valves are open, another set of interlocks will cause the valves to automatically close when the reactor coolant system pressure increases to approximately 585 psig.

5.1.4 System Features and Interrelationships

5.1.4.1 Plant Cooldown

The initial phase of reactor cooldown is accomplished by transferring heat from the RCS to the steam and power conversion system through the use of the steam generators. The second phase of cooldown starts with the RHR system being placed in operation. The RHR system is placed in operation approximately four hours after reactor shutdown when the temperature and pressure of the RCS are approximately 350°F and 425 psig, respectively.

Assuming that two heat exchangers and two RHR pumps are in service, and that each heat exchanger is being supplied with component cooling water at its design flow rate and temperature, the RHR system is designed to reduce the temperature of the reactor coolant from 350°F to 140°F within 16 hours. The heat load handled by the residual heat removal system during the cooldown includes residual and decay heat from the core, and reactor coolant pump heat. The design heat load is based on the decay heat fraction that exists at 20 hours following reactor shutdown from extended operations at full power. Coincident with operation of the residual heat removal system, a portion of the reactor coolant flow may be diverted from downstream of the residual heat removal heat exchangers to the Chemical and Volume Control System (CVCS) low pressure letdown line for cleanup and/or pressure control.

Start-up of the residual heat removal system includes a warm up period during which time reactor coolant flow through the heat exchangers is limited to minimize thermal shock to the heat exchangers. The rate of heat removal from the reactor coolant is manually controlled by regulating the coolant flow through the RHR heat exchangers.

Component cooling water is supplied at a constant flow rate to the RHR heat exchangers. The temperature of the return flow can be controlled by manually adjusting the control valves (606, 607) downstream of the RHR heat exchangers. In coincident with the manual adjustment of the heat exchanger outlet valves, a heat exchanger bypass valve (HCV-618) is manually adjusted to maintain a constant flow through each train of the RHR system.

The reactor coolant system cooldown rate is limited by equipment cooldown rates based on allowable stress limits. The available cooldown rate can be affected by the operating temperature limits of the component cooling water system. As the reactor coolant temperature decreases, the reactor coolant flow through the RHR heat exchangers is gradually increased by adjusting the control valve in each heat exchanger outlet line. The normal plant cooldown function of the residual heat removal system is independent of any engineered safety features function.

The normal cooldown return lines are arranged in parallel redundant flow paths and are also utilized as the low pressure emergency core cooling injection lines to the reactor coolant system. Utilization of the same return lines f emergency core cooling as well as for norma cooldown lends assurance to the proper functioning of these lines for engineered safety features purposes.

5.1.4.2 Solid Plant Operations

The residual heat removal system is used in conjunction with the chemical and volume control system (Figure 5.1-2) during cold shutdown operations (<200°F) to maintain reactor coolant chemistry and pressure control. Solid plant operations (i.e. no bubble in the pressurizer) is one method of operating the plant during the cold shutdown period. This mode of operation is generally limited to system refill and venting operations. Solid plant operations receives its name from the fact that the reactor coolant system is completely filled to the top of the pressurizer with coolant.

The RHR system is used to circulate reactor coolant from loop 4 hot leg to the cold leg connections on each loop. The RHR system is essentially operating as an extension of the reactor coolant system and is completely filled with reactor coolant. Pressure in the system can be changed by either changing the temperature of the reactor coolant or by varying the mass of the reactor coolant within the system. Using the temperature of the reactor coolant is not an effective method of RCS pressure control due to the time required to heat the coolant and the large pressure changes that are experienced for a small temperature change. Volume control of the reactor coolant is preferred because it is fast responding and any desired pressure change can be obtained within controllable limits. Since it is preferred to control the mass in the RCS for pressure control, a portion of the RHR flow is diverted to the chemical and volume contr system through valve HCV-128.

The volume of water diverted to the CVCS is controlled by the position of the back-pressure control valve PCV-131, which is located downstream of the letdown heat exchanger. During solid plant operations the volume of water returned to the reactor coolant system is determined by the charging rate, which is controlled by manually positioning the charging flow control valve HCV-182. The chemical and volume control system is also a water solid system with the exception of the volume control tank, which acts as a buffer or surge volume for the purpose of pressure control. Pressure is controlled by maintaining a constant charging rate and varying the flow rate of the water into the chemical and volume control system (PCV-131). To maintain a constant pressure in the RCS, both flow rates (charging and letdown), must be equal. If the charging rate exceeds the letdown rate then the pressure in the RCS will increase. Conversely pressure in the RCS will decrease if the letdown flow rate exceeds the charging flow rate.

Normally the back pressure regulating valve, PCV-131, is maintained in the automatic mode of operation and set to control the reactor coolant pressure at a desired set point. The volume control tank will absorb any mismatches in flow rates between charging and letdown. Pressure regulation is necessary to maintain the pressure in the RCS to a pre-selected range dictated by the fracture prevention criteria requirements of the reactor vessel.

5.1.4.3 Refueling

Both residual heat removal pumps are utilized during refueling to pump borated water from the refueling water storage tank to the refueling cavity. During this operation, the isolation valves in the inlet line from the reactor coolant system (8701, 8702) are closed, and the isolation valve from the refueling water storage tank (8812) is opened. The reactor vessel head is lifted slightly, refueling water is pumped into the reactor vessel through the normal RHR system return lines, into the refueling cavity through the open reactor vessel. The reactor vessel head is gradually raised as the water level in the refueling cavity increases. After the water level reaches the normal refueling level, the reactor coolant system inlet isolation valves are opened and the refueling water storage tank supply valves are closed.

During refueling, the residual heat removal system is maintained in service with the number of pumps and heat exchangers in operation as required by the heat load and Technical Specification minimum flow requirements.

After completing the refueling, the RHR system is used to return the water from the refueling cavity to the refueling water storage tank via a manual isolation valve. The water level is brought down to the flange of the reactor vessel. The remainder of the water in the refueling cavity is removed through drains located in the bottom of the refueling canal.

5.1.4.4 Emergency Core Cooling

The residual heat removal system functions in conjunction with the high pressure portion of the emergency core cooling system to inject borated water from the refueling water storage tank into the RCS cold legs during the injection phase following a loss of coolant accident. The residual heat removal system is aligned as shown in Figure 5.1-1. After the injection phase, the RHR system provides long term recirculation capability for core cooling. This function is accomplished by aligning the residual heat removal system to take water from the containment sump (opening valves 8811A&B and closing valves 8700A&B and 8812), cool this fluid by circulating it through the residual heat removal heat exchangers, and pumping it back to the core through the cold leg penetrations. If pressure in the RCS is greater than the discharge pressure of the RHR pumps, water may be returned to the core by the centrifugal charging pumps. (8804A) and the safety injection pumps (8804B).

In the event of a loss of coolant accident, fission products may be recirculated through part of the residual heat removal system exterior to the containment. If the residual heat removal pump seal should fail, the water would spill out on the floor in a shielded compartment. Each pump is located in a separate, shielded room. If one of the rooms flood, it would have no effect on the other train since there are no interconnections between trains during the recirculation phase.

Provisions are made for draining spillage into a sump which is provided with dual pumps and suitable level instrumentation so that this spillage can be pumped to the waste disposal system.

5.1.5 PRA Insights

A type of LOCA which has become a safety concern is the intersystem loss of coolant accident, sometimes called Event V. An intersystem LOCA is of concern because the coolant is lost outside the containment, and therefore would not be available for recirculation from the containment sump.

The intersystem LOCA is a small contributor to core damage frequency (4% at Surry, 0.1% at Zion, and 0.4% at Sequoyah). The failure that leads to an intersystem LOCA is the failure of the check valves in the low pressure injection system (RHR). This would lead to a loss of coolant outside the containment building. The water would not be available for recirculation from the containment sump.

Probable causes of an intersystem LOC include:

1. Transfer open of one check valve followed by the rupture of the second interface valve.
2. Failure of one valve to close on re-pressurization followed by the rupture of the second valve.
3. Rupture of the interface valve.
4. Operator failure to isolate the interfacing valve.

NUREG-1150 studies on importance measures have shown that Event V is a contributor to risk achievement, but a very minor contributor to risk reduction. Specifically, a large increase in the probability of the check valve rupture event involved in the intersystem LOCA would increase the core damage frequency (a factor of 30 at Sequoyah and a factor of 270 at Surry). Reducing the probability of the Event V initiator did not have a significant effect on the ri. reduction factors.

5.1.6 Summary

The residual heat removal system performs both normal plant functions and accident functions. The normal plant function is the transfer of heat from the reactor coolant system to the component cooling water system during shutdown operations. This is referred to as the second phase of plant cooldown, which starts when RCS T_{avg} is at 350°F. The RHR System is designed to remove decay heat associated with the shutdown reactor until the plant is restarted. During the shutdown, if solid plant operations are desired, the RHR system is used in conjunction with the chemical and volume control system for solid plant pressure control.

The RHR system is normally aligned to perform its accident function. During the injection phase following a loss of coolant accident, water is supplied from the refueling water storage tank to the reactor coolant system cold legs. For long term cooling and recirculation, the RHR system utilizes the containment sump as a source of water, and the RHR heat exchangers to cool the water prior to returning the water to the reactor coolant system.

The RHR system is also used during refueling to remove decay heat and to transfer water between the refueling water storage tank and the refueling cavity.

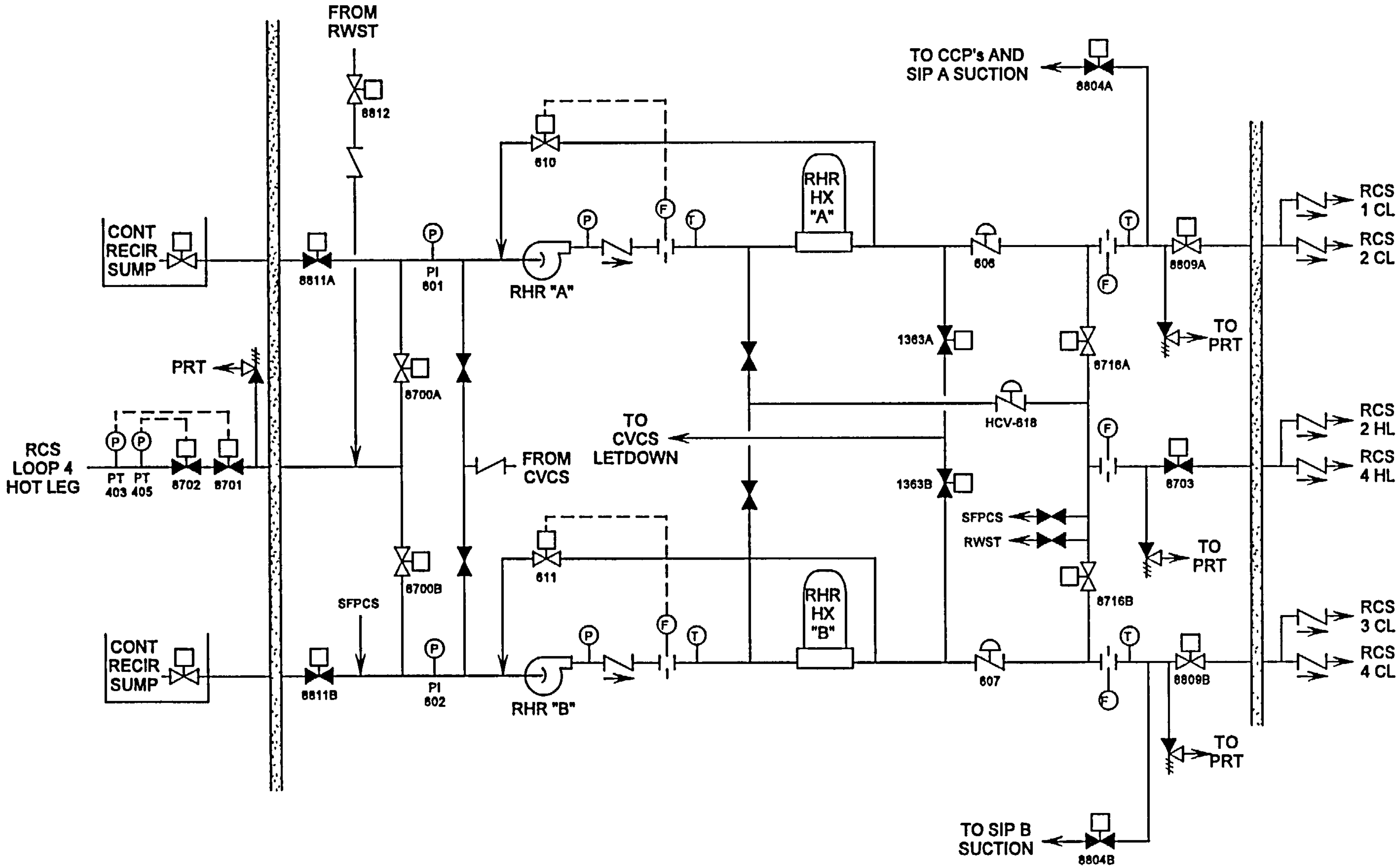

Figure 5.1-1 Residual Heat Removal System

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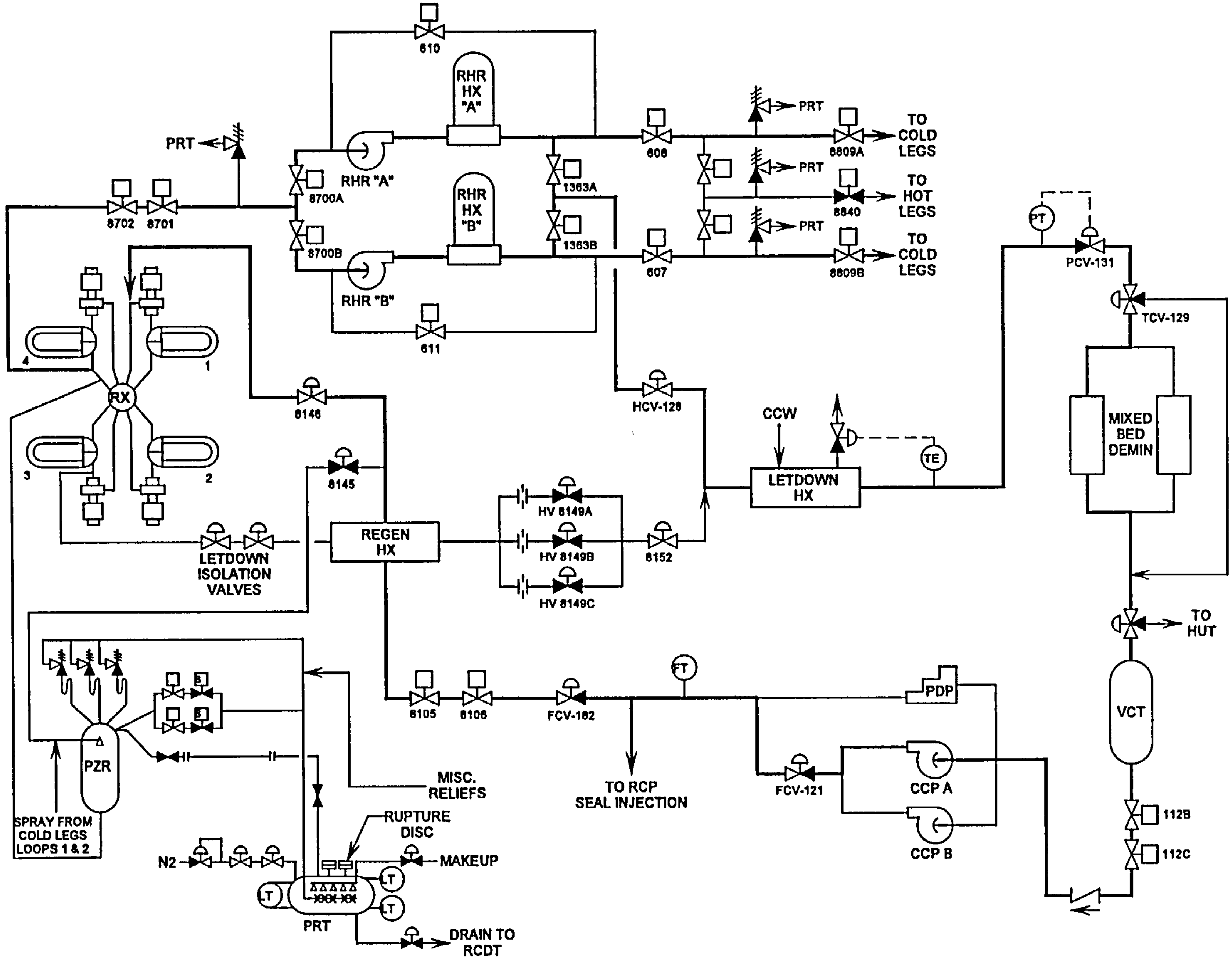

Figure 5.1-2 Solid Plant Control

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Westinghouse Technology Systems Manual

Section 5.2

Emergency Core Cooling System

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5.2 EMERGENCY CORE COOLING SYSTEMS

Learning Objectives:

1. Explain why Emergency Core Cooling Systems (ECCS) are incorporated into plant design.

2. State the purpose of the following systems:
   a. Accumulator injection system,
   b. Safety Injection (SI) pump system,
   c. High head injection system, and
   d. Residual Heat Removal (RHR) system

3. State the purpose of the following major components:
   a. Refueling Water Storage Tank (RWST), and
   b. Containment recirculation sump

4. List the order of ECCS injection during the following abnormal conditions:
   a. Small (slow depressurization) loss of coolant accident, and
   b. Large loss of coolant accident.

5.2.1 Introduction

The purposes of the emergency core cooling systems are as follows:

1. Emergency core cooling systems
   - Provide core cooling to minimize fuel damage following a loss of coolant accident (LOCA).
   - Provide additional shutdown margin following a steam line break accident.

2. Accumulators (passive system) (Section 5.2.3.1).
   - Rapidly reflood the core following a LOCA.

3. High head safety injection system (active system (Section 5.2.3.2).
   - Provide high pressure, low volume safety injection for small to intermediate size LOCAs.
   - Maintain shutdown margin by injecting highly concentrated boric acid through the Boron Injection Tank (BIT) following a steam line break.
   - Provide charging flow for the Chemical Volume and Control System (CVCS) during normal operations.

4. Safety injection pump system (active system) (Section 5.2.3.5).
   - Provide intermediate pressure, low volume safety injection for small to intermediate size LOCAs.

5. Residual heat removal system (active system) (Section 5.2.3.6).
   - Provide low pressure, high volume safety injection to complete the reflooding of the core following a LOCA.
   - Provide a flowpath and heat sink for long term core cooling following a LOCA.
   - Provide for decay heat removal during a plant cooldown below 350°F.

As listed above, the emergency core cooling system is divided into several subsystems consisting of both passive systems and active systems.

The passive system (accumulators) consist of large volume tanks of borated water pressurized with nitrogen. Pressure in the passive systems is less than that of the Reactor Coolant System (RCS). Following an accident, when reactor coolant system pressure decreases below tank pressure, the borated water will be injected.

The active systems (high, intermediate, and low pressure injection systems) consist of several pumping systems of varying discharge pressures and flow rates. These systems do not start until they receive an accident initiation signal (ESF). Once started, these systems will inject borated water into the reactor coolant system as reactor coolant system pressure decreases below the discharge pressure of the system pumps.

The ECCS is designed to cool the reactor core and provide additional shutdown capability following initiation of the following accident conditions:

1. Loss of coolant from the reactor coolant system in excess of normal make up.
2. Steam generator tube rupture.
3. A pipe break in the main steam system.

The emergency core cooling system provides shutdown capability for the accidents listed above by means of chemical poison (boron) injection.

5.2.2 System Description

5.2.2.1 General Design Criteria

The emergency core cooling systems are designed in accordance with 10 CFR 50 Appendix A, General Design Criteria 35, 36, and 37. Criteria 35 is given below; Criteria 36 and 37 are associated with system testing.

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

5.2.2.2 ECCS Acceptance Criteria

The emergency core cooling systems must also meet the requirements of 10 CFR 50.46, Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors. 10CFR50.46, in part, reads:

Each light water reactor fueled with uranium oxide pellets within cylindrical zircaloy cladding shall be provided with an emergency core cooling system. Of which these systems shall be designed so that their calculated cooling performance following postulated loss of coolant accidents conforms to the following criteria.

1. Calculated Peak Cladding Temperature remains less than 2200 °F.
2. Maximum cladding oxidation shall not exceed 0.17 times the total cladding thickness.
3. Maximum hydrogen generation shall not exceed .01 times the hypothetical amount generated from the chemical reaction of the cladding with water.
4. Changes in core geometry will allow for core cooling flow.
5. Long term cooling can be maintained.

5.2.2.3 General Description

The principal components of the emergency core cooling system which provide core cooling immediately following a loss of coolant accident are the accumulators, the safety injection pumps, the centrifugal charging pumps, the residual heat removal pumps, refueling water storage tank, and the associated valves, and piping. Table 5.2-1 lists normal status of these components. (See Figure 5.2-1). The order of component injection into the reactor coolant system is dependent on the size of the break.

For large pipe ruptures, the reactor coolant system would be depressurized and voided of coolant rapidly, and a high flow rate of emergency coolant is required to quickly cover the exposed fuel rods and limit possible core damage. This high flow is provided by passive cold leg accumulators; and active charging pumps, safety injection pumps, and the residual heat removal pumps discharging into the cold legs of the RCS.

Emergency cooling is provided for small ruptures primarily by the high head injection pumps. Small ruptures are those with an equivalent diameter of 6 inches or less, and which do not immediately depressurize the reactor coolant system below the accumulator discharge pressure. The centrifugal charging pumps (high head injection) deliver borated water at the prevailing RCS pressure to the cold legs of the RCS. During the injection, the charging pumps take suction from the refueling water storage tank. The discharge from the pumps initially sweeps the concentrated boric acid in the boron injection tank into the RCS.

5.2.3 Component Descriptions

5.2.3.1 Cold Leg Injection Accumulators (Figure 5.2-2)

The accumulators are pressure vessels filled with borated water and pressurized with nitrogen gas. During normal operation each accumulator is isolated from the RCS by two check valves in series. Should the RCS pressure fall below the accumulator pressure, the check valves open and borated water is forced into the RCS. One accumulator is attached to each of the cold legs of the reactor coolant system. Mechanical operation of the swing disc check valves is the only action required to open the injection path from the accumulators to the core via the cold leg.

Accumulator injection occurs immediately when the RCS is depressurized below accumulator operating pressure. This setpoint will be reached only in event of a large RCS rupture.

Accumulator pressure is exerted by nitrogen gas and can be adjusted as required during normal plant operation. The accumulators are normally isolated from the source of nitrogen supply. Gas relief valves on the accumulators protect them from pressures in excess of design pressure. The accumulators are located within the containment but outside of the secondary shield wall which protects them from missiles. Since the accumulators are located within the containment, a release of the nitrogen gas in the accumulators would cause an increase in normal containment pressure. Containment pressure increase following release of the gas from all accumulators has been calculated and is well below the containment pressure setpoint for emergency core cooling system actuation.

Release of accumulator gas is detected by the accumulator pressure indicators and alarms. Thus the operator would take action promptly as required to maintain plant operation within the requirements of the Technical Specification covering accumulator operability.

Connections are provided for remotely adjusting the level and boron concentration of the water in each accumulator during normal plant operation as required. Accumulator water level may be adjusted either by draining to the reactor coolant drain tank or by pumping borated water from the refueling water storage tank to the accumulator.

Minimum accumulator water level is based on the requirement of the accumulators to refill and reflood the reactor vessel to the core mid-plane during the design basis LOCA. This assumes one of the four accumulators injects directly to the break and is not available for core cooling.

Relief Valves

The accumulator relief valves are sized to pass nitrogen gas at a rate in excess of the accumulator gas fill line delivery rate. The relief valves will also pass water in excess of the expected accumulator in-leakage rate, but this is considered precautionary, because the time required to fill the gas space gives the operator ample opportunity to correct the situation.

Other relief valves are installed in various sections of the emergency core cooling system to protect lines which have a lower design pressure than the RCS. Relief valves discharge to the pressurizer relief tank. The valve stem and spring adjustment assembly are isolated from the system fluids by a bellows seal between the valve disc and spindle. The closed bonnet provides an additional barrier for enclosure of the relief valves.

Accumulator Check Valves

The low pressure accumulator check valve is designed with a low pressure drop configuration with all operating parts contained within the body. The disc is permitted to rotate, providing a new seating surface after each valve opening. Design considerations and analysis which assure that leakage across all the check valves located in each accumulator injection line will not impair accumulator availability are as follows:

1. During normal operation the check valves are in the closed position with a nominal differential pressure across the disc. The differential pressure is approximately 1650 psid for the check valves in the cold leg lines. Since the valves remain in this position except for testing or when called upon to function, and are not, therefore, subject to the abuses of flow operation or impact loads caused by sudden flow reversal and seating, they do not experience significant wear of the moving parts, and hence are expected to function with minimal leakage.

2. When the reactor coolant system is being pressurized during the normal plant heatup operation, the check valves are tested for leakage as soon as there is a stable differential pressure of about 100 psi or more across the valve. This test confirms the seating of the disc and whether or not there has been an increase in the leakage since the last test. When this test is completed, the discharge line motor operated isolation valves are opened and the RCS pressure increase is continued. There should be no increase in leakage from this point on since increasing reactor coolant pressure increases the seating force and decreases the probability of leakage.

3. The accumulators can accept some in-leakage from the RCS without affecting availability. In-leakage would require, however, that the accumulator water volume be adjusted according to Technical Specification requirements. An alarm is also provided as an added safeguard against excessive accumulator in-leakages. Cold leg accumulator design parameters are listed in Table 5.2-2.

5.2.3.2 High Head Injection System

The high head injection system (Figure 5.2-4) utilizes the centrifugal charging pumps, which are used during normal plant operations with the chemical and volume control system (Chapter 4.1) for normal charging supply. Under accident conditions these pumps deliver water from the refueling water storage tank (Section 5.2.3.4), to the reactor coolant system at the prevailing RCS pressure. Each centrifugal charging pump is a multistage, diffuser design, barrel type casing with vertical suction and discharge nozzles. The unit has a self contained lubrication system, and mechanical seal cooling system. Component cooling water is the normal heat exchange medium.

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the volume control tank after cooling in the seal water heat exchanger to protect the pumps at the shutoff head. The charging pumps may be tested during normal operation through the use of the minimum flow bypass line.

Receipt of an engineered safety features actuation signal will initiate the following actions in the high head injection system.

1. The centrifugal charging pumps will receive a start signal.
2. The suction valves from the volume control tank (112B, 112C) will close and the suction valves from the refueling water storage tank (112D, 112E) open. This provides a source of borated water to the suction of the pumps.
3. In order to limit the bypass flow around the boron injection tank the minimum flow recirculating valves (8110, 8111) close.
4. To establish a flowpath to the RCS the boron injection tank inlet (8803A&B) and outlet (8801A&B) valves open.

The high head injection system will continue to operate in this configuration until the lineup is changed by the reactor operator. The discharge pressure of the centrifugal charging pumps is sufficient to provide flow to the RCS for any postulated size break. Pump capacity and pressure capability allow the pumps to deliver adequate flow under all pressure conditions up to and including pressurizer safety valve lift pressure. This is the last method of cooling the core (known as feed-and-bleed), if both forced circulation and natural circulation were unavailable. In this hypothetical case, the only flow through the core would be delivered by the charging pumps and displaced out the power operated relief valves on the pressurizer.

5.2.3.3 Boron Injection Tank (BIT)

The boron injection tank is being phased out of use. At most Westinghouse units the function of the BIT has been eliminated, however the tank still remains in the high head injection flow path. However, since some units may incorporate the BIT into their design this section is provided to explain the design and function of this tank.

The boron injection tank (Figure 5.2-3) contains a boric acid solution (2000 to 20000 ppm) and is connected to the discharge of the centrifugal charging pumps. Upon actuation of the safety injection signal, the charging pumps provide the pressure to inject the boric acid solution into the reactor coolant system when the isolation valves open. To prevent cold spots and stratification within the tank during normal operation, the contents of the boron injection tank are continuously recirculated with boron injection recirculation pumps. These pumps are rated at 2 horsepower and provide a flow of 20 gpm. The recirculation flow path includes a 75 gallon stainless steel surge tank, which accommodates volumetric changes in the system.

The boron injection tank incorporates a sparger type inlet which distributes the incoming boric acid in 360 degrees as it enters the tank. Redundant tank heaters and line heat tracing are provided to ensure that the solution will be stored at a temperature greater than the solubility limit of boric acid (normally greater than 135°F).

The design basis of the boron injection tank is to provide the negative reactivity needed to ensure the reactor remains subcritical even during the worst-case cooldown accident. This is the cooldown (and the associated positive reactivity addition) from a steam line break. Boron injection tank design parameters are listed in Table 5.2-4.

5.2.3.4 Refueling Water Storage Tank (Figure 5.2-1)

The refueling water storage tank is designed to hold enough borated water (borated to 2000 ppm), to fill the refueling cavity for refueling operations and to provide water for ECCS operation. The volume of the RWST is 438,000 gallons with a Technical Specification minimum volume of 428,000 gallons. The RWST is always aligned for safety injection operation to provide water for the centrifugal charging, safety injection, residual heat removal, and containment spray pumps.

The refueling water storage tank is protected from back flow of reactor coolant from the reactor coolant system. All connections to the refueling water storage tank are provided with check valves to prevent back flow. When the RCS is hot and pressurized there is no direct connection between the RWST and the RCS. When the reactor coolant system is being cooled down and the residual heat removal system is placed into operation, the RHR system is isolated from the RWST by a motor operated valve (8812) in addition to a check valve. RWST design parameters are listed in Table 5.2-5.

5.2.3.5 Intermediate Head (Safety Injection System)

The intermediate head injection system (Figure 5.2-4), also referred to as the safety injection system, utilizes two safety injection pumps. Each intermediate head safety injection pump is a multistage centrifugal pump. The pump is driven directly by an induction motor. The unit has a self contained lubrication system and mechanical seal cooling system. Component cooling water is the normal pump heat exchange medium.

A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event the pumps are started with the normal flow paths blocked. This line also permits pump testing during normal operation. Two motor operated valves (8814& 8813) are provided in this line. These valves are closed by operator action during the recirculation mode.

The safety injection pump system is designed to provide water from the RWST to the four reactor coolant system cold or hot legs in the case of a relatively small break where the system pressure continues to remain high for a relatively long period.

The safety injection pump system is aligned during normal operations to perform its accident function. Upon receipt of an engineered safety features actuation signal, the pumps will start and recirculate to the RWST until RCS pressure decreases below pump discharge pressure. Initial injection is into the cold legs of the RCS. Injection into the RCS hot leg valves (8802A&8802B) is a manual realignment completed by the operator. Safety injection component design parameters are listed in Table 5.2-6.

5.2.3.6 Residual Heat Removal System

The residual heat removal system (Figure 5.2-5) utilizes the residual heat removal (low head injection) pumps to deliver water from the refueling water storage tank or the containment recirculation sump to the reactor coolant system. Each residual heat removal pump is a single stage, vertical, centrifugal pump. It has an integral motor-pump shaft driven by an induction motor. The unit has a self contained mechanical seal cooling system. Component cooling water is the pump heat exchange medium. The residual heat removal pumps are discussed in Chapter 5.1.

The residual heat removal (low head injection) system is designed to provide water from the RWST or containment recirculation sump to the four RCS cold or hot legs in the case of a large break up to and including the Design Basis Accident (DBA) where the system pressure decreases to containment pressure in a relatively short period of time.

The RHR system is aligned during normal plant operations to perform its accident function. All valves in the flowpath from the RWST to the RCS cold legs are open. Upon receipt of an engineered safety features actuation signal, the RHR pumps will start and recirculate around the heat exchangers (valves 610, 611) back to the pump suctions. When reactor coolant system pressure is less than RHR pump discharge pressure, the cold leg injection line check valves will open and the RHR system will provide flow to the core through the cold legs. Switch over to the containment sump as a source of water to the core is manually aligned when RWST level decreases to 48%. Injection into the hot legs of the Reactor Coolant System is also a manual operation initiated by the plant operators.

5.2.3.7 Containment Recirculation Sump

The containment recirculation sump (Figure 5.2-6) is designed to prevent trash and debris from entering the recirculation flow path which could affect the operation of the residual heat removal and containment spray systems. In addition, the sump provides for adequate NPSH for the containment spray pumps and residual heat removal pumps to operate in the cold leg recirculation mode.

Trash is prevented from entering the sump by providing a trash curb around the sump. Sufficient flow area around the sump maintains a low fluid velocity at the entrance. Therefore, the debris that is more dense than water will settle to the containment floor rather than enter the sump. Baffles are provided to protect the sump screens from damage and to prevent floating debris from entering the sump. Two screens of graduated mesh size to less than 1/4" prevents entrance of suspended matter that could jeopardize the containment spray or residual heat removal systems operation. The top of the sump above the screen sides is covered with 14 gauge sheet steel on 3" grating. The sump design, the materials used in fabrication of the sump, the suction piping, the guard pipes, and the isolation valves are designed to provide the assurance that the sump will remain functional for long-term circulation during existence of a post accident environment.

Each recirculation line from the sump is routed outside the containment to a sump isolation valve. This valve is surrounded with a leak tight steel enclosure and the section of piping joining it to the sump is run within a guard pipe welded to the enclosure. Any leakage from the sump piping or valve body will be contained and cannot leak into the atmosphere or cause a loss of recirculation fluid. Any excessive leakage or passive failure downstream of the sump valves can be controlled and isolated by closure of the sump valve in the affected train.

The emergency core cooling system recirculation loop piping and components external to containment is surrounded by shielding. This shielding is designed to permit access for maintenance to a component such as a pump while the redundant component is recirculating sump fluid.

5.2.4 System Features and Interrelationships

5.2.4.1 Emergency Core Cooling System Materials

Materials employed for components of the ECCS are selected to meet the applicable material requirements of the codes and the following additional requirements:

1. All parts of all components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or equivalent corrosion resistant material with the exception of pump seals and valve packing.
2. Valve seating surfaces are hard-faced with Stellite No. 6 or equivalent to prevent galling and reduce wear.
3. Valve stem materials are selected for their corrosion resistance, high tensile properties, and resistance to surface scoring by the packing.
4. All parts of components in contact with containment recirculation sump solution during recirculation are fabricated of austenitic stainless steel or equivalent corrosion resistant material.

The elevated temperature of the sump solution is within the design temperature of all the emergency core cooling system components. In addition, consideration has been given to the potential for corrosion of various types of metals exposed to the fluid conditions prevalent immediately after the accident or during the long term recirculation operations.

The emergency core cooling system equipment inside the containment, which is required to operate following a loss-of-coolant accident, is environmentally tested. The chemistry used in the test program was obtained by using a spray solution of 1.5 weight percent boric acid solution and adjusting the pH to a value of approximately 9.5 with sodium hydroxide. This solution is similar to that of a post accident environment resulting from the release of sodium into the containment recirculation sump. The results of the test program indicate that the emergency core cooling system components will operate satisfactorily during and following exposure to the combined containment post-accident environments of temperature, pressure, chemistry, and radiation.

Pertinent design and operating parameters for the components of the emergency core cooling system are given in Table 5.2-1. The component design and operating conditions are specified as the most severe conditions to which each respective component is exposed during either normal plant operation, or during operation of the emergency core cooling system. For each component, these conditions are considered in relation to the code to which it is designed. By designing the components in accordance with applicable codes, and with due consideration for the design and operating conditions, the fundamental assurance of structural integrity of the emergency core cooling system components is maintained. In addition, components of the emergency core cooling system are designed to withstand the appropriate seismic loadings in accordance with their safety class.

5.2.4.2 Emergency Core Cooling System Piping

All piping joints are welded except for the pump and butterfly valve flanged connections. Weld connections for pipes sized 2" and larger are butt welded. Reducing tees are used where the branch size exceeds one-half of the header size. Branch connections of sizes that are equal to or less than one-half of the header size conform to the ANSI code. Branch connections ½ inch through 2 inches are attached to the header by means of full penetration welds, using pre-engineered integrally reinforced branch connections.

Minimum piping and fitting wall thicknesses are increased to account for the manufacturer's permissible tolerance to minus 12 percent on the nominal wall and an appropriate allowance for wall thinning on the external radius during any pipe bending operations in the shop fabrication of the subassemblies.

Heat tracing is installed on all piping, valves, flanges, and instrumentation lines carrying the nominal 12 weight percent concentrated boric acid solution. The heat tracing system is designed in accordance with the following criteria:

1. One hundred percent redundant and separate heat tracing systems are provided.
2. Each heat tracing system is designed to maintain the fluid temperature between 160°F and 170°F with an ambient air temperature of 60°F.
3. Each redundant heat tracing system is supplied from a separate electrical bus capable of being connected to the redundant emergency diesel generators.

5.2.4.3 Emergency Core Cooling System Valves

The design features employed to minimize valve leakage include:

1. Where possible, packless valves are used.
2. Globe valves are installed with recirculation fluid pressure under the seat to prevent stem leakage of recirculated (radioactive) water when the valves are closed.
3. Relief valves are enclosed, i.e., they are provided with a closed bonnet.
4. All control and motor operated valves (2 inches and above) exposed to recirculation flow have double packed stuffing boxes and stem leak-off connections to the waste processing system.

Motor Operated Valves

The seating design of all motor operated valves is of the parallel disc design or the flexible wedge design. These designs release the mechanical holding force during the first increment of travel so that the motor operator works only against the frictional component of the hydraulic imbalance on the disc and the packing box friction. The discs are guided throughout the full disc travel to prevent chattering and to provide ease of gate movement. The seating surfaces are hard faced to prevent galling and to reduce wear.

Where a gasket is employed for the body to bonnet joint, it is either a fully trapped, controlled compression, spiral wound asbestos gasket with provisions for seal welding, or it is of the pressure seal design with provision for seal welding. The valve stuffing boxes are designed with a lantern ring leak-off connection with a minimum of one-half of a set of packing above the lantern ring. A full set of packing is defined as a depth of packing equal to 1 times the stem diameter.

The motor operator incorporates a "hammer blow" feature that allows the motor to impact the discs away from the main seat or back seat upon opening or closing. This "hammer blow" feature not only impacts the disc but allows the motor to attain its operational speed prior to impact. Valves which can function against system pressure are designed such that they function with a pressure drop equal to full system pressure across the valve disc.

Manual Globe, Gate and Check Valves

Gate valves are either wedge design or parallel disc and are straight through. The wedge is either split or solid. All gate valves have back seat and outside screw and yoke. Globe valves, "T" and "Y" style are full ported with outside screw and yoke construction.

Check valves are spring loaded lift piston types for sizes 2 inches and smaller, swing type for size 3 inches to 4 inches and tilting disc type for size 4 inches and larger. Stainless steel check valves have no penetration welds other than the inlet, outlet and bonnet. The disc hinge is serviced through the bonnet.

The stem packing and gasket of the stainless steel manual globe and gate valves are similar to those described above for motor operated valves. Carbon steel manual valves are employed to pass non-radioactive fluids only and therefore do not contain the double packing and seal weld provisions.

Diaphragm Valves

The diaphragm valves are of the Saunders patent type which uses the diaphragm member for shut off with even weir bodies. These valves are used in systems not exceeding 200°F design temperature.

5.2.4.4 Emergency Core Cooling System Reduced Availability

Certain modifications (i.e., reduced component availability) to the normal operating status, as given in Table 5.2-1, of the emergency core cooling system are permissible without impairing the ability of the emergency core cooling system to provide adequate core cooling capability. Accordingly, Technical Specifications have been established to cover these modifications.

As an example, Technical Specifications permit one cold leg accumulator to be isolated for check valve leakage testing. They also permit various pumps of the emergency core cooling system to be inoperable during power operation and for an additional time period while the reactor is at hot shutdown, provided that the duplicate pump has been demonstrated to be operable prior to the repair. Technical Specification times are based on the following:

1. Repairs will be completed within the allowable time period.
2. If it is determined that this time period is not adequate to restore the component to the operable condition, putting the reactor in the hot shutdown condition significantly reduces the cooling requirements following a postulated loss of coolant accident.
3. Failure to complete repairs within the designated time interval after going to hot shutdown is considered indicative of a requirement for major maintenance and therefore the reactor is put in the cold shutdown condition where the emergency core

cooling system capability is not required.

The minimum number of active components will be capable of delivering full rated flow within a specified time interval after process parameters reach the set points for the engineered safety features actuation signal. Response of the system is automatic with appropriate allowances for delays in actuation of circuitry and active components. The active portions of the system are actuated by the safety injection (ESF) signal. In analysis of system performance, actuation of components are established on the basis that only emergency onsite power is available.

Since redundant flow paths are provided during recirculation, a leaking component in one of the flow paths may be isolated. This action curtails any further leakage and renders the component available for corrective maintenance. In the loss of coolant accident analysis, no credit is assumed for partial flow prior to the establishment of full flow, and no credit is assumed for the availability of normal offsite power sources.

For smaller loss of coolant accidents, there is some additional delay before the process variables reach their respective programmed trip set points since this is a function of the severity of the transient imposed by the accident. This is allowed for in the analysis of the range of loss of coolant accidents.

5.2.4.5 Emergency Core Cooling Systems Integrated Operations

The operation of the emergency core cooling system, Figure 5.2-1, following a loss of coolant accident, can be divided into three distinct modes of operation summarized as follows:

1. The injection mode:
   - Any reactivity increase following the postulated accident is terminated, initial cooling of the core is accomplished, and coolant lost from the primary system is replenished.
2. The cold leg recirculation mode:
   - Long term core cooling is provided during the accident recovery period.
3. The hot leg recirculation mode:
   - Flow through the core is reversed.

Injection Mode After Loss of Primary Coolant

The injection mode of emergency core cooling is initiated by the emergency safety features actuation signal (SI Signal). This signal is actuated by any of the following:

- Low pressurizer pressure.
- High containment pressure.
- High steam flow, coincidence with either low low T_{avg} or low steam pressure.
- Steam line differential pressure.
- Manual actuation.

These signals are discussed in detail in Chapter 12.

Operation of the emergency core cooling system during the injection mode is completely automatic. The engineered safety features actuation signal in addition to starting the ECCS pumps automatically initiates the following actions:

1. Starts the diesel generators and, if all other sources of power are lost, aligns them to the 4.16-kV safety related boards.
2. Aligns the charging pumps for injection. (Section 5.2.3.2)
3. Starts auxiliary cooling water pumps to

provide cooling for safety related components.

Remotely operated valves for the injection mode which are under manual control (i.e., valves which normally are in their ready position and do not require a safety injection signal) have position indication on a common portion of the control board. If a component is out of proper position, a monitor light will indicate this misalignment on the status panel. At any time during operation when one of these valves is not in the ready position for injection, this condition is shown visually on the board, and an audible alarm is sounded in the control room.

During large pipe ruptures, the reactor coolant system would be depressurized and voided of coolant rapidly. A high flow rate of emergency coolant is required to quickly cover the exposed fuel rods and limit possible core damage. This high flow is provided by the passive cold leg accumulators, the charging pumps, safety injection pumps, and the residual heat removal pumps discharging into the cold legs of the RCS. The residual heat removal and safety injection pumps deliver into the accumulator injection lines during the injection mode. The charging pumps deliver through the boron injection tank directly into the cold legs during the injection mode.

For small pipe ruptures, emergency core cooling is provided primarily by the high head injection pumps. Small ruptures are those with an equivalent diameter of 6 inches or less which do not immediately depressurize the reactor coolant system below the accumulator discharge pressure. The centrifugal charging pumps deliver borated water at the prevailing RCS pressure to the cold legs. The charging pumps take suction from the refueling water storage tank during the injection mode. The discharge from the pumps initially sweeps the concentrated boric acid in the boron injection tank into the reactor coolant system.

The safety injection pumps also take suction from the refueling water storage tank and deliver borated water to the cold legs of the RCS. The safety injection pumps begin to deliver water to the RCS after the pressure has fallen below the pump shutoff head. The residual heat removal pumps take suction from the refueling water tank and deliver borated water to the RCS. These pumps begin to deliver water to the reactor coolant system only after the pressure has fallen below the pump shutoff head.

The injection mode continues until the low level is reached in the refueling water tank at which time the operator must change system alignment to the cold leg recirculation mode.

Cold Leg Recirculation Mode

Water level indication in the containment recirculation sumps and the refueling water storage tank, and low level alarms on the refueling water storage tank provide ample warning to terminate the injection mode while the operating pumps still have adequate net positive suction head. Since the injection mode of operation following a loss of coolant accident is terminated before the refueling water storage tank is completely emptied, all pipes are kept filled with water.

The change-over from the injection mode to the recirculation mode of operation may be accomplished automatically or manually dependent upon plant design. The sequence of operations is followed regardless of which power supply is available (offsite or emergency onsite).

After the injection phase of operation, water collected in the containment sump is cooled and returned to the reactor coolant system by the residual heat removal system. In the event of a small rupture, where the depressurization proceeds more slowly, the reactor coolant system pressure may still be in excess of the shutoff head of the residual heat removal pumps at the onset of recirculation. In this case, the flow from the RHR heat exchanger outlet can be aligned to the suction piping of the charging pumps and the safety injection pumps and returned to the RCS.

Hot Leg Recirculation Mode

Approximately 24 hours after the switch over to cold leg recirculation, hot leg recirculation will be initiated. Hot leg recirculation will cause the coolant flow through the core to be reversed. This accomplishes the following:

1. Since the core is essentially operating as an evaporator/concentrator, hot leg recirculation will sweep the concentrated boric acid solution out of the core to mix with the less concentrated solution in the containment sump.
2. Any boiling or steam formation in the core or upper head region will be terminated as the reverse flow sweeps the steam out of the core.

Switch over to hot leg recirculation is a manual operation accomplished by the operator. Hot leg and cold leg recirculation will be used alternately following the loss of coolant accident for as long as required.

5.2.5 PRA Insights

During loss of coolant accidents, the emergency core cooling systems provide core cooling to minimize core damage. The systems are designed to provide this cooling over the full range of break sizes, small (<2" break), intermediate (2" to 6" break), and large (>6"break). For the purposes of the PRA, the high pressure system is made up of both the centrifugal charging pumps and the safety injection pumps. The low pressure system is the residual heat removal pumps.

The loss of coolant accident is a major contributor to the core damage frequency (59% at Sequoyah, 28% at Surry, and 18% at Zion). The major failures in the LOCA sequences that lead to core damage is the failure of either the high pressure emergency core cooling systems in the recirculation mode or the failure of the low pressure emergency core cooling system in the recirculation mode. The loss of the recirculation capability, especially during small and intermediate size breaks, will allow the core to continue to heat up and boil off, resulting in core damage.

Probable causes of failure of the ECCS include:

1. Failure of the high pressure recirculation system to shift to the recirculation mode.
2. Failure of the high pressure injection discharge valves to open.
3. Failure of the room cooling for the high pressure injection pumps.
4. Failure to shift from cold leg to hot leg recirculation.
5. Failure to shift from the low pressure injection to the low pressure recirculation.
6. Failure of the low pressure recirculation sump suction valves to open.
7. Failure of the RWST to the LPI suction valves to close.
8. Failure of the operators to realign the system after testing.
9. Failure of the low pressure injection pumps to start.

NUREG-1150 studies on importance measures have shown that the ECCS can be a major contributor to both the risk reduction and the risk achievement. Specifically the core damage frequency is most sensitive to increases in the probability of component faults and operator errors associated with the containment sump recirculation. The greatest reduction in core damage frequency could be achieved if the probabilities of the initiating event, operator errors, and the failure of the valves could be reduced.

5.2.6 Summary

The emergency core cooling system is designed to provide protection to the core and to reduce the consequences of the major accidents.

The emergency core cooling systems are designed in accordance with the General Design Criteria of 10CFR50 and they must meet the acceptance requirements of 10 CFR 50.46. The emergency core cooling systems are divided into both passive and active systems.

The passive system is the cold leg injection accumulators. The active systems include the high head injection (centrifugal charging pumps), intermediate head injection (safety injection pumps), and the low head injection (residual heat removal pumps) systems.

All active components are actuated by an engineered safety features actuation signal (also known as safety injection signal). These signals originate in the reactor protection system (Chapter 12.3).

The order of injection into the reactor coolant system following the loss of coolant accident will be dependent on the size of the rupture. In general, a large rupture will initially require the passive accumulators to inject first followed by the active pumping systems. A small rupture will be characterized by a slow rate of pressure reduction in which case the ECCS components will inject in order from the highest to lowest pressure systems.

All active systems use the RWST as a source of water following the loss of coolant accident. A low level in the RWST signals the end of the injection mode of operation. Cold leg recirculation mode of operation will be initiated automatically or by the operator. This mode will provide long term cooling for the core. Hot leg recirculation mode of operation will be used alternately to provide reverse flow to the core.

The emergency core cooling systems are also designed to meet single failure criteria by providing 100% redundancy in components and system flow paths.

Table 5.2-1
Normal Operating Status of Emergency Core Cooling System Components

Number of charging pumps operable	2
Number of safety injection pumps operable	2
Number of residual heat removal pumps operable	2
Number of residual heat removal heat exchangers operable	2
Minimum refueling water storage tank volume, gal.	428,000
Boron concentration in refueling water storage tank, ppm	2,000 - 2,500
Number of cold leg accumulators	4
Cold leg accumulator water volume, ft^3	870 - 930
Cold leg accumulator pressure, psig	600 - 650
Minimum boron concentration in cold leg accumulators, ppm	1,900

Table 5.2-2
Accumulator Design Parameters

Number	4
Design pressure, psig	700
Design temperature, °F	300
Operating temperature, °F	100-150
Normal operating pressure, psig	650
Minimum operating pressure, psig	600
Total volume, gal	10,100 ea.
Minimum water volume, gal	6,500 ea.
Boric acid concentration, minimum ppm	1,900

Table 5.2-3
Centrifugal Charging Pump Design Parameters

Number	2
Design pressure, psig	2,800
Design temperature, °F	300
Design flow rate, gpm	150
Developed head at maximum flow rate, psig	1,400
Shutoff head, psig	2,670

Table 5.2-4
Boron Injection Tank Design Parameters

Number	1
Total volume, gal	900
Boric acid concentration, ppm	1,900-2,000
Design pressure, psig	2,735
Design temperature, °F	300
Operating temperature, °F	150-180

Table 5.2-5
Refueling Water Storage Tank Design Parameters

Number	1
Total volume, gal	438,000
Minimum volume, gal	428,000
Boric acid concentration, ppm	2,000 - 2,500
Normal pressure	Atmospheric
Operating temperature, °F	37 - 90
Design temperature (Tank),°F	200

Table 5.2-6
Safety Injection Pump Design Parameters

Number	2
Design pressure, psig	1,700
Design temperature, °F	300
Design flow rate, gpm	425
Developed head at maximum flow rate, psig	650
Shutoff head, psig	1,520

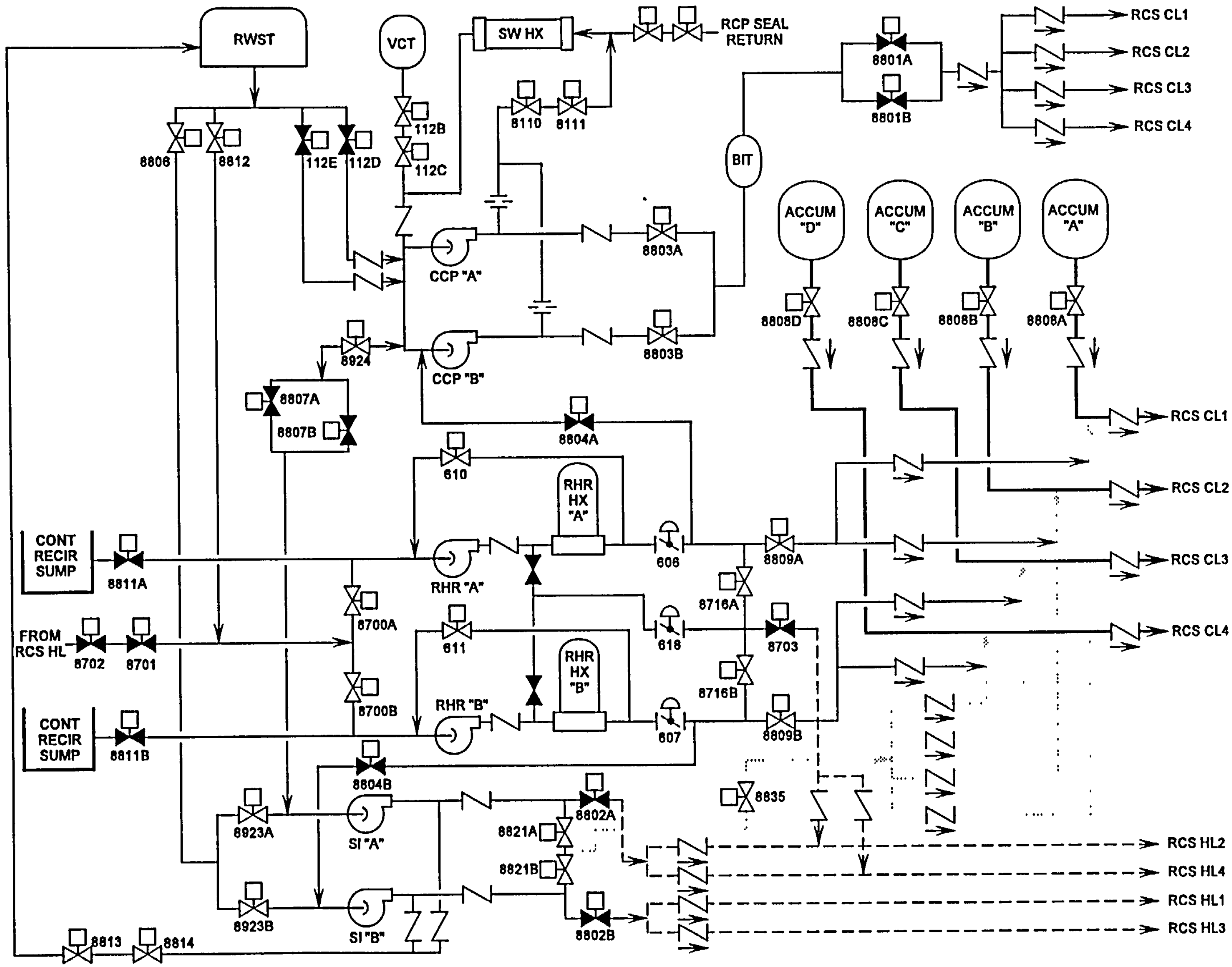

Figure 5.2-1 ECCS Composite

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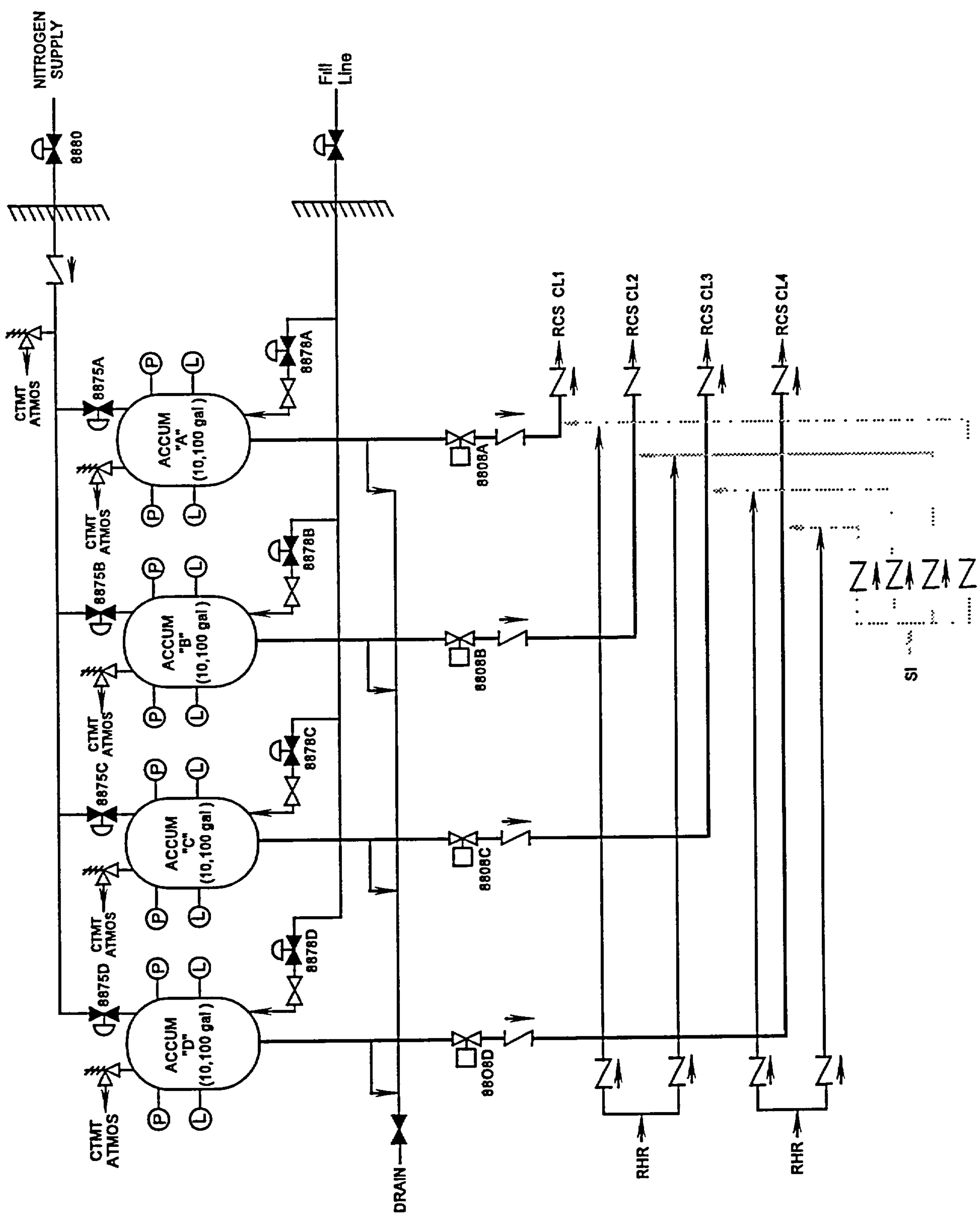

Figure 5.2-2 Cold Leg Accumulator System

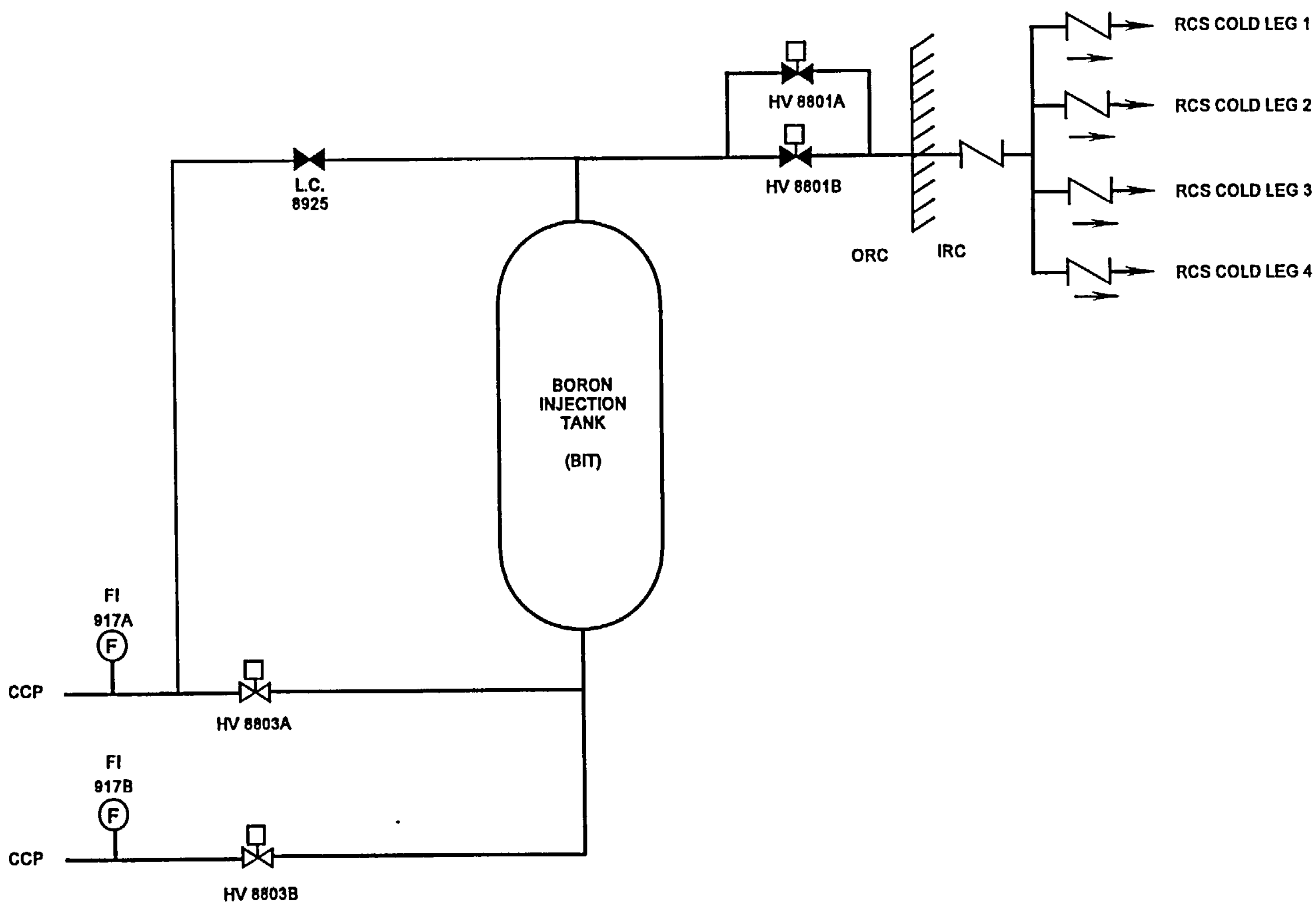

Figure 5.2-3 High Head Injection System

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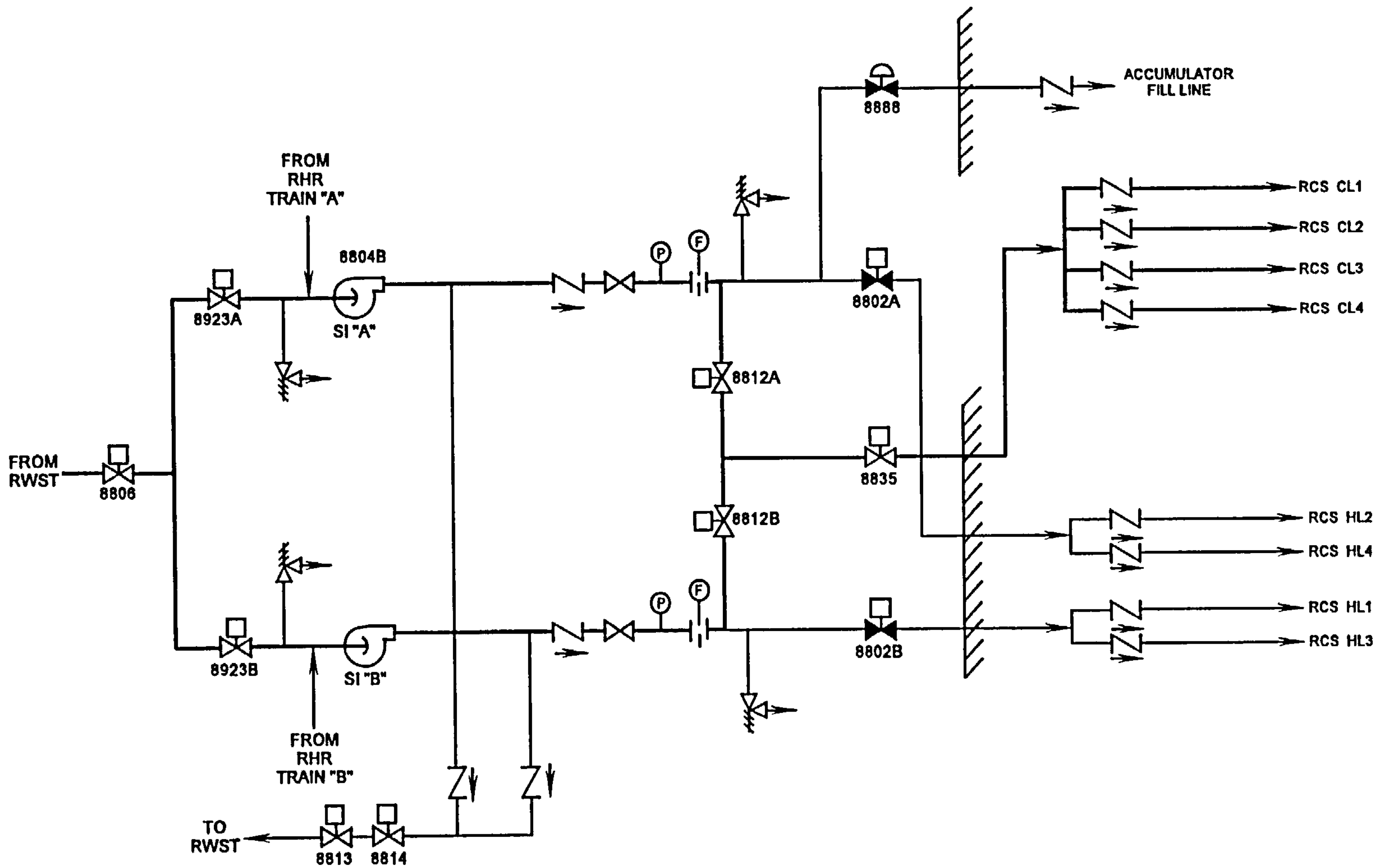

Figure 5.2-4 Safety Injection System

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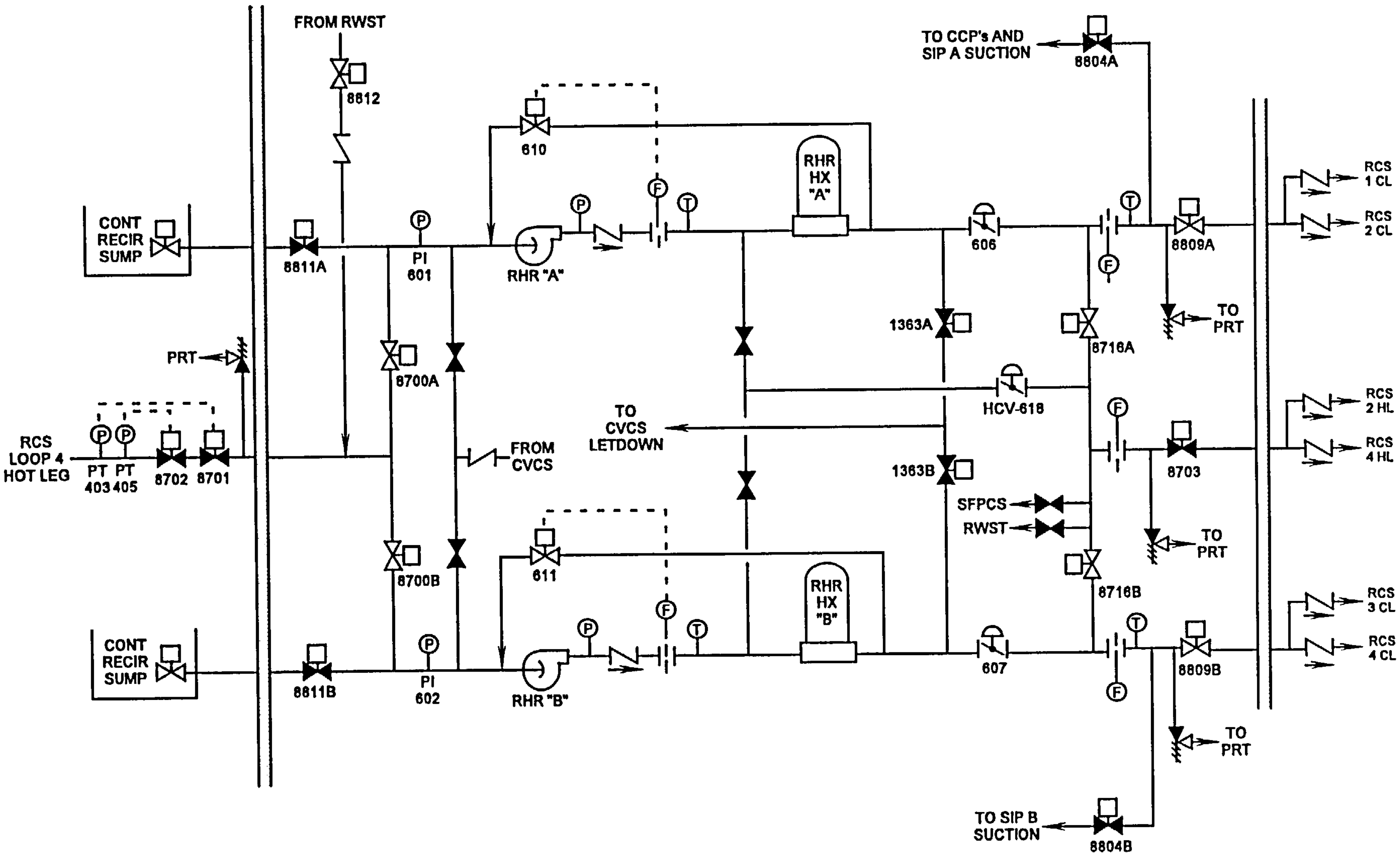

Figure 5.2-5 Residual Heat Removal System

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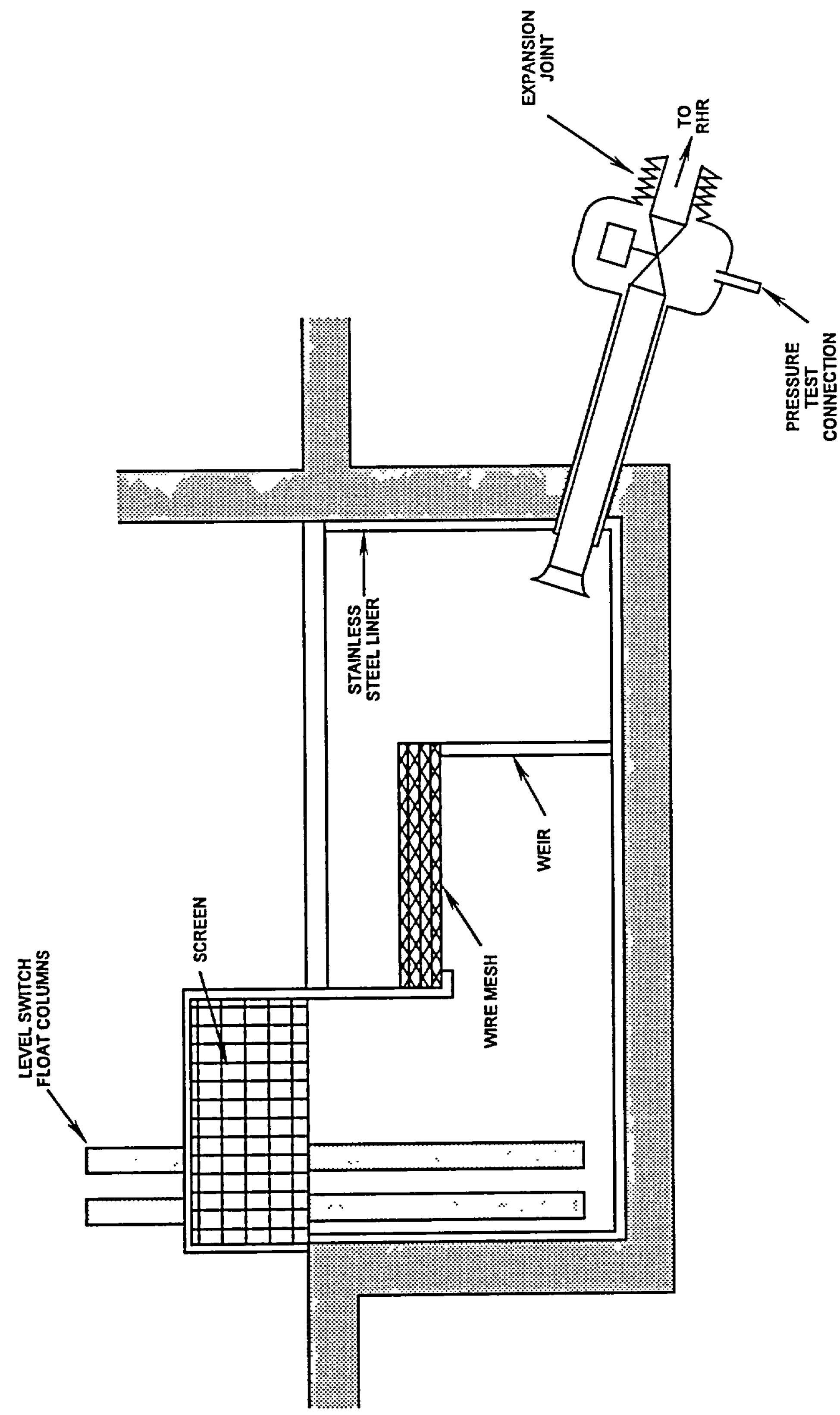

Figure 5.2-6 Containment Recirculation Sump

Westinghouse Technology Systems Manual

Section 5.3

Containment

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5.3 CONTAINMENT

Learning Objectives:

1. State the purposes of the containment.
2. Briefly describe the function of the following:
   a. Containment liner
   b. Primary shield wall
   c. Secondary shield wall
   d. Refueling canal
   e. Containment sumps
   f. Containment recirculation sump
   g. Containment hydrogen analyzer
3. Briefly describe the methods of monitoring the containment environmental conditions.

5.3.1 Introduction

The purposes of the containment are as follows:

1. Provide a barrier to prevent the escape of radioactivity during normal and accident conditions.
2. Provide protection against internally and/or externally generated missiles.
3. Provide biological shielding during normal and accident conditions.
4. Provide Seismic Category I supports for the Reactor Coolant System (RCS) and its associated support systems.

The containment completely encloses the reactor and the RCS, and serves to prevent the inadvertent release of radioactive fission products to the atmosphere. The containment also provides biological shielding during normal operations and during the unlikely event of a Loss of Coolant Accident (LOCA).

Several different types of containments have been developed for PWR applications, and almost all are premised on the use of the containment structure to contain the large volume of high pressure, high temperature steam-water mixture that would result from a LOCA or a Steam Line Break (SLB) inside the containment. After a LOCA or a SLB, the pressure and temperature inside the containment will increase to a peak level, and then decrease as the containment support systems are activated.

The containment structure must be shown to be functionally available for the life of the plant. From the viewpoint of design, the containment design must consider the following loadings:

1. Pressure and temperature transients that occur as the result of a Design Basis Accident (DBA).
2. Thermal loads such as the temperature gradient through the containment wall during normal and transient conditions.
3. Dead loads consisting of the weight of the concrete wall, dome, base slab, internal concrete, machinery and other permanent load contributing stresses.
4. Live loads which consist of snow loading, movable equipment loads, and other loads which vary with intensity and occurrence.
5. Earthquake loads such as the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE).
6. Wind and tornado loads with consideration given to missile impingement.
7. Hydrostatic loads based on the worst case flood conditions with a water level significantly above mean sea level.
8. External pressure loads based on a maximum differential pressure, inside to outside, on the containment.
9. Prestressing loads are considered in all loading combinations.

10. Pressure test loads which is 1.15 times the design pressure.

The loading conditions caused by the DBA, resulting from gross failure of the RCS, and those caused by earthquake are considered to be the critical loading conditions.

5.3.2 System Description

The large dry primary containment utilizing a three dimensional (3-D) post-tensioned, prestressed, reinforced concrete cylinder with a steel liner was first used in 1966 and is deployed in the majority of the PWRs in the United States.

Because of the inherent weakness of concrete in tension and its strength in compression, prestressing systems were developed to superimpose compressive loads onto concrete structures so that when these structures are loaded by exterior force systems (such as a LOCA) the net stresses in the concrete are still generally compressive. Therefore, prestressing keeps the concrete in compression under all postulated loading conditions, and the concrete will be available to carry seismic shear forces without any need for additional reinforcement. The prestressed containment's single greatest disadvantage is the active nature of the structural system which requires monitoring during the life of the plant. This monitoring is accomplished in the form of periodic lift-off tests of a sample of tendons, to insure no prestress loss, and added in-service inspection requirements of the tendon strands.

5.3.3 Component Description

5.3.3.1 Containment Shell

The containment shell (Figure 5.3-1) is a Seismic Category I prestressed post-tensioned concrete cylinder with a hemispherical dome and a flat foundation slab. The reactor cavity and instrumentation tunnel are located below the foundation slab. A continuous access gallery for the installation, tensioning and inspection of the vertical U-tendons is also provided below the foundation slab. A 19 foot diameter equipment hatch and two personnel airlocks of approximately 10 feet in diameter are provided in the shell (Figure 5.3-2). Table 5.3-1 shows the design parameters of the containment shell.

The foundation slab is conventionally reinforced with high-strength reinforcing steel. Three (3) buttresses equally spaced at 120° intervals around the outside of the containment are provided as anchor points for the horizontal tendons.

A transfer tube penetration (Figure 5.3-4) is provided for fuel movement between the refueling canal in the containment building and the fuel

Table 5.3-1
Containment Shell Design Parameters

Inside Diameter, ft.	124
Inside Height, ft.	203
Wall Thickness, ft.	3 1/2
Dome Thickness, ft.	2 1/2
Internal Free Volume, ft.^{3}	2 x 10^6
Design Pressure, psig	60 max (DBA)3 1/2 min.
Design Temperature, °F	120 normal 288 DBA
OBE, g	0.15
SSE, g	0.25
DBA, ft^2 (double - ended pump suction shear)	10.48
Buttresses, #	3
Material	Prestressed reinforced concrete

handling building. Numerous smaller penetrations for electrical conduits, piping and other systems are also provided.

5.3.3.2 Internal Structures

The major components of the internal structure (Figure 5.3-3) are :

1. Primary shield wall.
2. Secondary shield wall.
3. Operating floor.
4. Refueling canal.
5. Intermediate floor, platforms, and hatches.
6. Removable shield slabs.
7. Pipe supports and restraints.
8. Nuclear Steam Supply System (NSSS) equipment supports and restraints.

The primary shield wall is reinforced concrete and completely surrounds the reactor vessel and provides biological shielding. The shield wall is located in the center of the containment and extends from the foundation slab to the operating floor. The primary shield wall also gives support to the reactor.

The secondary shield wall is a reinforced concrete structure which provides radiation shielding and protection to the RCS. The RCS is completely enclosed by the secondary shield wall which extends from the foundation slab to above the operating floor. The wall also provides lateral support to the steam generators, reactor coolant pumps, pressurizer and associated piping.

The operating floor is constructed of reinforced concrete and structural steel framing. The floor is supported by the refueling canal walls, secondary shield walls and the containment shell. Separation is provided in the supports and between the floor and shell to allow for differential horizontal movement. Floor hatches are provided for equipment and tank removal.

A refueling canal, of reinforced concrete with a stainless steel liner, is provided for transportation of new and used fuel between the reactor vessel and the fuel transfer penetration. During periods of maintenance or refueling the canal is used for temporary storage of vessel internals or fuel.

The intermediate floor is constructed of reinforced concrete with steel grating platforms and walkways supported by structural steel framing. Hatches are provided for equipment removal. The floor is also used as a laydown area during maintenance or refueling.

Removable shield slabs, of precast concrete, are supported above the reactor vessel by the secondary shield walls. These shield slabs protect the steel liner from damage by missile impingement.

5.3.3.3 Steel Liner

To ensure a high degree of leak tightness, the inside face of the containment shell is lined with one-quarter (1/4) inch of carbon steel plate which is thickened in the regions adjacent to all penetrations. The carbon steel has a minimum yield strength of 30,000 psi and an elongation, in an eight (8) inch section, of 21 percent.

The liner is designed to function as a leak tight seal only, Any tensional stresses due to a DBA are transferred from the liner to the concrete wall. Table 5.3-2 shows the liner design parameters.

5.3.4 System Features and Interrelationships

5.3.4.1 Prestressing

Prestressing is a method by which internal compressive stresses are induced in concrete so that when a load is applied the tensile stresses in the concrete are minimized.

In prestressed containment shells, compressive stresses are produced in both directions in the cylinder and the dome. The level of prestressing is adjusted such that when tensile forces are acting on the shell, most load combinations do not produce tensile stresses in the concrete.

The concrete compression is achieved by installing high strength steel tendons in ducts in the concrete and tensioning them. To minimize the adverse affects of corrosion, the ducts are filled with a corrosion inhibiting grease. The number and placement of the tendons is determined by the containment design such that the minimum force required is provided to all sections of the containment shell in both the horizontal and vertical directions. Consideration is also given for stress losses in the tendons which result from friction forces during tensioning, elastic shortening of concrete, concrete creep under long term loads and tendon steel relaxation.

Table 5.3-2
Containment Liner Design Parameters

Material	ASTM A-442, Grade 55
welded	
	steel plate
Thickness, in.	1/4
Yield Strength, psi	>30,000
Elongation, %	
(8 inch specimen)	21
Tendons	
Number of Vertical	70
Number of Horizontal	132
Wires per Tendon	170
Size of Wire, in.	1/4
Ultimate Strength, tons	>1000
Type of Wire	High strength relieved steel IAW ASTM A-421

5.3.4.2 Tendons

Each tendon is composed of 170 stress relieved, high strength one-quarter (1/4) inch diameter wires. Each tendon has an ultimate yield strength in excess of 1000 tons. Button heads are employed at the ends of the tendons to transfer tensile forces to the anchor plates (Figures 5.3-5, 5.3-6 and 5.3-8). There are 70 vertical tendons arranged in two (2) families that are perpendicular to each other in the upper region of the dome (Figure 5.3-7).

The vertical tendons are continuous and stretch from one anchor point in the tendon gallery, up through the wall, through the roof, and back down the opposite wall to another anchor point in the tendon gallery.

There are three (3) buttresses equally spaced around the outside of containment. The buttresses serve as anchor points for the hoop tendons, which stretch from one buttress, past the next and to the third buttress. The 132 horizontal tendons continue in this manner up the wall and onto the roof, with each successive tendon anchored to a different buttress.

Sufficient prestressing is provided in the cylindrical and dome portions of the containment to eliminate any tensile stress across the interior wall thickness under design loads. There is a loss of approximately 12 percent of prestress, due to elastic and creep losses, which reduces the prestress to design level.

Each tendon is pretested at the time of initial tensioning. The stress in the tendons during accident loading is approximately 80 percent of the stress induced at tensioning. This ensures that the possibility of tendon failure under DBA loading is remote. The coincident failure of two (2) or three (3) side by side tendons during DBA conditions will have no significant affect on containment integrity. This is ensured by designing the walls sufficiently thick to transmit the force to adjoining tendons without resulting in any serious local stress. The tendons are inspected periodically and lift-off readings made to ensure adequate performance of the prestressing system during the life of the plant.

5.3.4.4 Penetrations

There are four types of penetrations into the containment, all are welded assemblies, except the equipment hatch which is manufactured in two (2) bolted halves and seal welded after installation. All penetrations are pressure resistant and leaktight. The steel liner is thickened in the region of the penetrations.

The four types of penetrations are:

1. Electrical.
2. Piping.
3. Equipment hatch and personnel airlocks.
4. Special purpose.

The containment penetrations are presented in more detail in Section 5.6 of this manual.

5.3.4.5 Cranes

The containment is equipped with two (2) cranes and a hoist for installation of equipment and for plant maintenance. The reactor area polar bridge crane is supported by brackets embedded in the containment shell concrete. The polar crane is equipped with two (2) 15 horsepower bridge motors, a 10 horsepower trolley motor, an auxiliary hoist, and a main hoist. The auxiliary hoist has a capacity of 25 tons and is driven by a 60 horsepower motor. The main hoist has a capacity of 125 tons and is driven by a 60 horsepower motor.

The containment jib crane is a one (1) ton capacity 14 foot crane installed on the steam generator shield wall near the refueling cavity. The crane can pivot over the refueling cavity and against the wall for storage.

The stairwell hoist crane provides lifting capability for the stairwell.

5.3.4.6 Containment Recirculation Sump

The containment recirculation sump serves a vital safety related function as a large collection reservoir designed to provide an adequate water supply to both the Containment Spray (CS) and Residual Heat Removal (RHR) systems. A detailed presentation of the recirculation sump is provided in Section 5.2 of this manual.

5.3.4.7 Containment Building Sumps

Two containment building sumps provide low points for leakage collection within the containment building. These sumps are referred to as the North and the South sump. Each sump has a dedicated sump pump which is utilized to pump water to the dirty waste drain tank when necessary (Figure 5.3-9).

Technical Specifications require that the containment building sumps and their associated sump pumps be operable for a RCS leakage indicator. The frequency with which the sump pumps operate is an indication of leakage.

5.3.4.8 System Isolation and Integrity

The containment isolation systems provide a means of isolating fluid, air or gas systems that penetrate containment. This confines any radioactivity that may be released during and after a design basis LOCA to the containment structure. Containment isolation is achieved by applying common design criteria to penetrations in the many different fluid systems and by use of a Containment Isolation Signal (CIS) to actuate appropriate valves. A detailed presentation of containment isolation is provided in Section 5.6 of this manual.

5.3.5 System Instrumentation

Containment instrumentation is provided for the detection of radioactive and nonradioactive leakage in the containment. Continuous monitoring of the environmental conditions within the containment also provides a background level of overall normal leakage from primary systems and components. Detection of deviations from the normal containment environmental conditions provides indication in the control room of increases in leakage rates.

The instrumentation provided for containment monitoring consists of pressure transmitters, Resistance Temperature Detectors (RTDs), sump level transmitters, area and process radiation monitors, a humidity monitor and a Containment Hydrogen Analysis System (CHAS).

5.3.5.1 Containment Pressure

Four pressure transmitters (Figure 5.3-10) and seven pressure switches provide indication in the main control room and actuation signals to Engineered Safety Features (ESF) systems. Employing a two-out-of-three logic, pressure switches provide a "Containment Pressure High" alarm in the main control room at 3.5 psig. Additionally, Safety Injection (SI) and CIS signals are generated. Using a two-out-of-four logic, pressure switches provide a "Containment Pressure High-High" alarm in the main control room at 30 psig. Containment spray and steam line isolation signals are also generated. To prevent an inadvertent spray down of the containment, the Hi-Hi pressure switches energize to cause actuation preventing containment spray down on loss of power to the switches.

5.3.5.2 Containment Temperature

Containment ambient temperature is monitored by eight (8) RTDs located above the 205 foot elevation. In addition, local temperatures are available in the steam generator cubicles, the pressurizer cubicle, the reactor vessel cavity, alongside the containment wall and bio-shield wall and in the incore instrumentation switching room. Main control room indication for the containment temperatures is provided on scanning temperature recorder.

5.3.5.3 Containment Sump Level

Level indications and alarms are provide for the containment recirculation sump (Figure 5.3-11) and for the containment building sumps (Figure 5.3-9).

The containment recirculation sump is designed to operate in the post-DBA environment to provide water for the CS pumps and to provide water for long term core cooling by RHR. The recirculation sump is provided with dual level alarm and indication circuitry. Each channel has three (3) indicating lights for levels of 24 inches, 30 inches and 34 inches above the sump floor. There are also high level alarms at 30 inches and 34 inches. All indications are in the main control room.

Containment sump level indication and alarm is provide on the radwaste control panel. The low level alarm setpoint is set at three (3) inches above the bottom of the sump and the high level alarm setpoint is 42 inches above the bottom of the sump. Level indication for each sump is provided by a series of six (6) lights which indicate at eight-inch (1/8) intervals from three (3) inches to 42 inches. Additional narrow range and wide range level indication is provided in the main control room for the containment sumps.

5.3.5.4 Containment Radiation

The containment Process Effluent Radiation Monitoring System (PERM-1) detectors continuously monitor gaseous, iodine, and air particulate activity levels in the containment atmosphere during normal operation. PERM-1 detectors also monitor the gaseous, iodine and air particulate activity levels of the containment purge exhaust flow during containment purge operations. Increasing airborne radiation levels are one indication of potential RCS leakage. By comparing the activity in the coolant to the activity in the containment atmosphere a magnitude of leakage can be determined.

There are five (5) Area Radiation Monitors (ARMS) in containment. Two (2) of them are high range post-accident detectors used to determine the effectiveness of accident mitigation efforts. The other three (3) detectors monitor containment radiation levels in areas likely to experience high levels. A detailed presentation of the radiation monitoring system is provided in Chapter 16 of this manual.

5.3.5.5 Containment Humidity

The containment Humidity Detection System (HDS) provides a means of measuring the overall leakage within the containment. The psychometric detector measures wet and dry bulb temperature and displays percent humidity in the main control room on both a meter and a chart recorder. By comparing these values for specific humidity within the containment over a period of time, a means of measuring the overall leakage within the containment is achieved.

5.3.5.6 Containment Hydrogen Analysis System (CHAS)

The CHAS (Figures 5.3-12, 5.3-13) is a standby system is only used when directed by procedure. The system is used to determine the need for operation of hydrogen recombiners and hydrogen mixing fans in containment. In addition, since the amount of hydrogen in containment is proportional to the amount of core damage, a rough approximation of damage can be made using the CHAS.

In the event of a LOCA the CHAS is designed to operate up to 30 days without servicing and provide accurate hydrogen measurement at containment pressures up to 50 psig and containment temperatures up to 445 °F. Remote readouts of hydrogen concentration are available in the main control room.

The CHAS consists of two (2) large analysis panels and two (2) smaller control panels with appropriate indication and control capability. The control units are located in an area that would be accessible during an accident and will normally be operated by chemistry personnel to obtain the sample. The system uses redundant sample paths with CHAS A using the normal PERM 1 sample path and CHAS B using a separate penetration.

The CHAS has a selectable span for the indication. In LOW, the indication will readout from zero (0) to ten (10) percent. In HIGH, the indication will readout from zero (0) to thirty (30) percent. The system is normally selected to the LOW span and the control room must be notified if the span is changed to the HIGH position.

To initiate sampling of the containment atmosphere for hydrogen, containment isolation signals must be cleared, a sample flow path must be established and the CHAS turned on. The CHAS A and CHAS B inlet and outlet valves are controlled from the main control room. The system will take about 20 minutes to warm up and be operational. During the warmup, the alarms from the system will be disabled.

5.3.6 Containment Evacuation Alarm

A containment evacuation alarm circuit is provided to alert personnel that may be in containment of the need to evacuate. The alarm is of particular significance during maintenance and outage activities when there may be many people in containment.

The alarm consists of an audible siren and a portion of the normal containment lighting that is made to flash during actuation. The alarm is actuated by the high flux at shutdown bistables in the excore nuclear instrumentation source range indication circuit and manually from the main control room.

5.3.7 Summary

The containment is a cylindrical fully reinforced concrete Seismic Category I structure with a hemispherical roof and a flat foundation slab. The cylindrical portion is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome is prestressed by a two way post-tensioning system consisting of horizontal and vertical tendons.

The containment structure provides biological shielding for both normal and accident conditions. By completely enclosing the reactor and RCS, the containment system ensures that an acceptable upper limit for leakage of radioactive materials to the environment is not exceeded even with gross failure of the RCS.

The containment is designed for all credible conditions of loading, including normal loads, LOCA loads, test loads and loads due to adverse environmental conditions. The loading conditions caused by a DBA, resulting from gross failure of the RCS, and those caused by earthquake are considered to be the critical loading conditions.

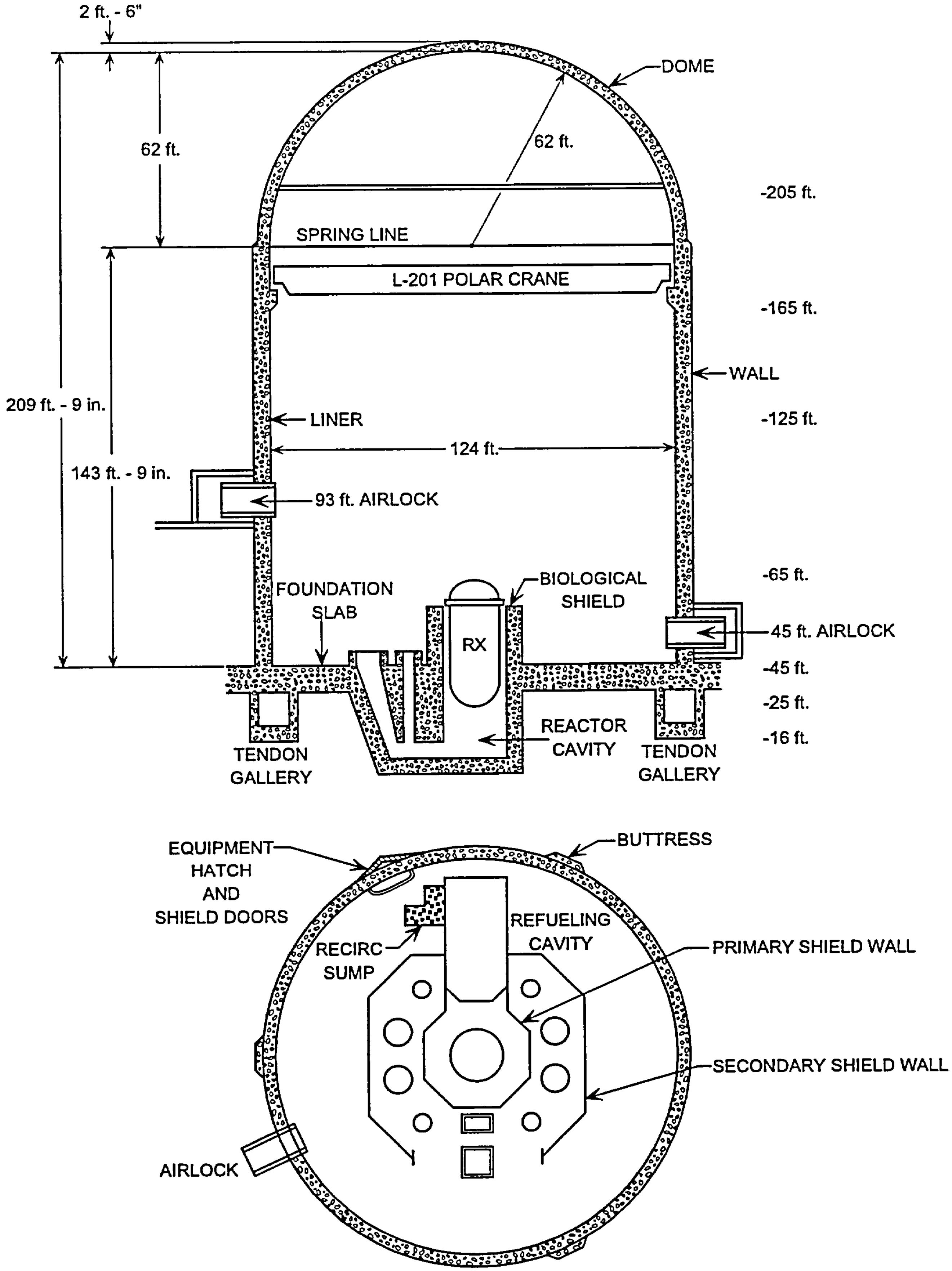

Figure 5.3 -1 Containment and Internals

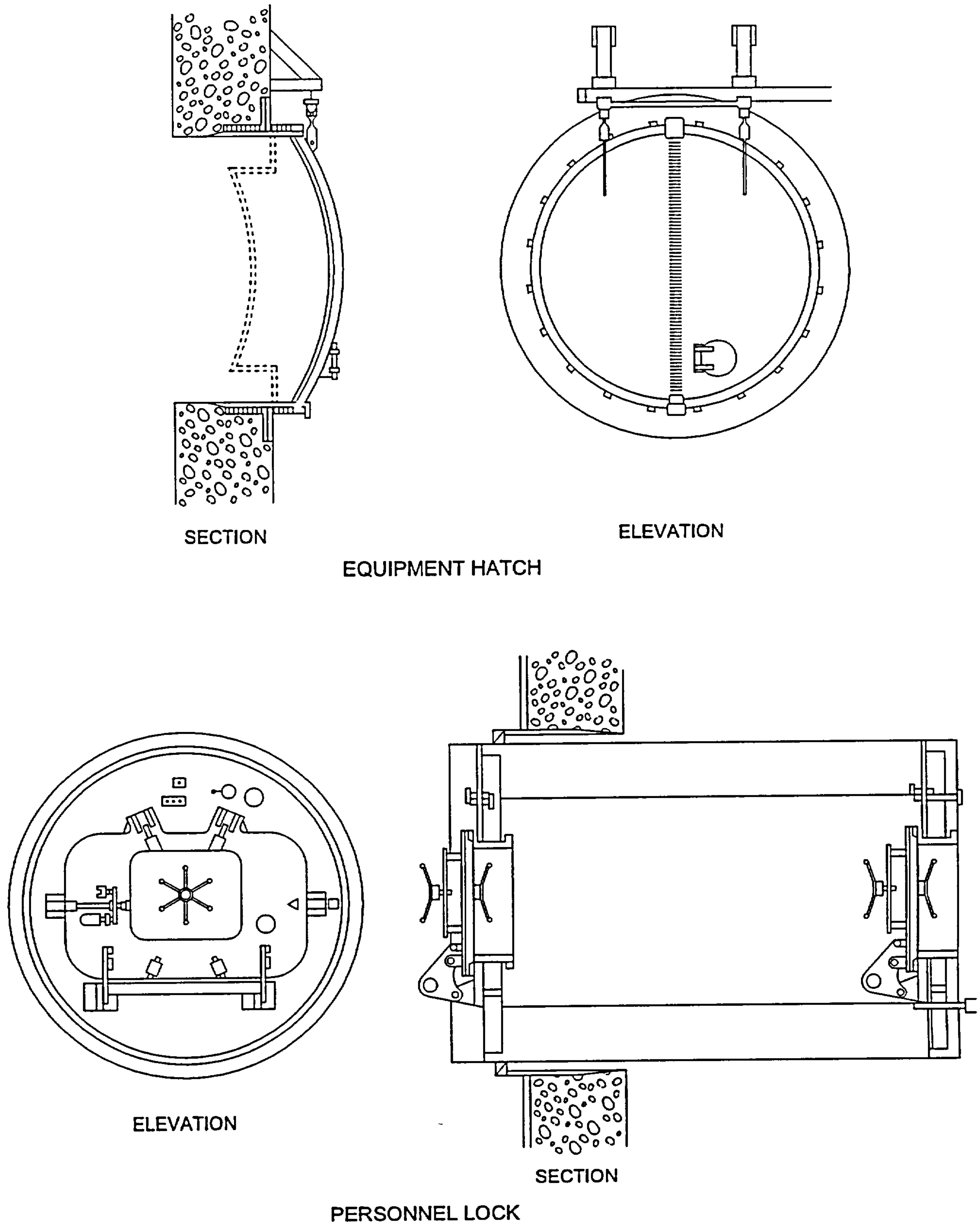

Figure 5.3-2 Equipment Hatch and Personnel Lock

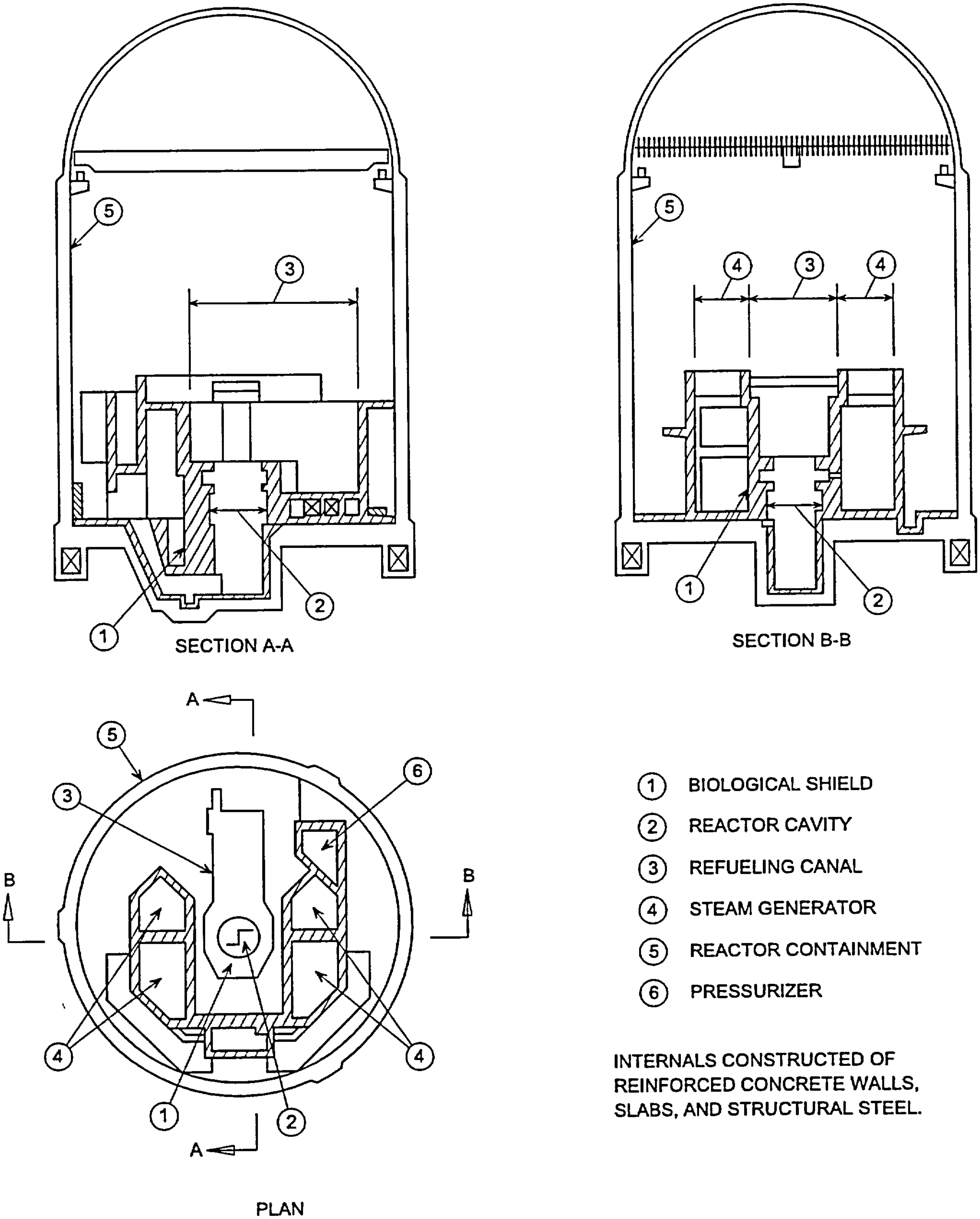

Figure 5.3-3 Containment Internals (Simplified)

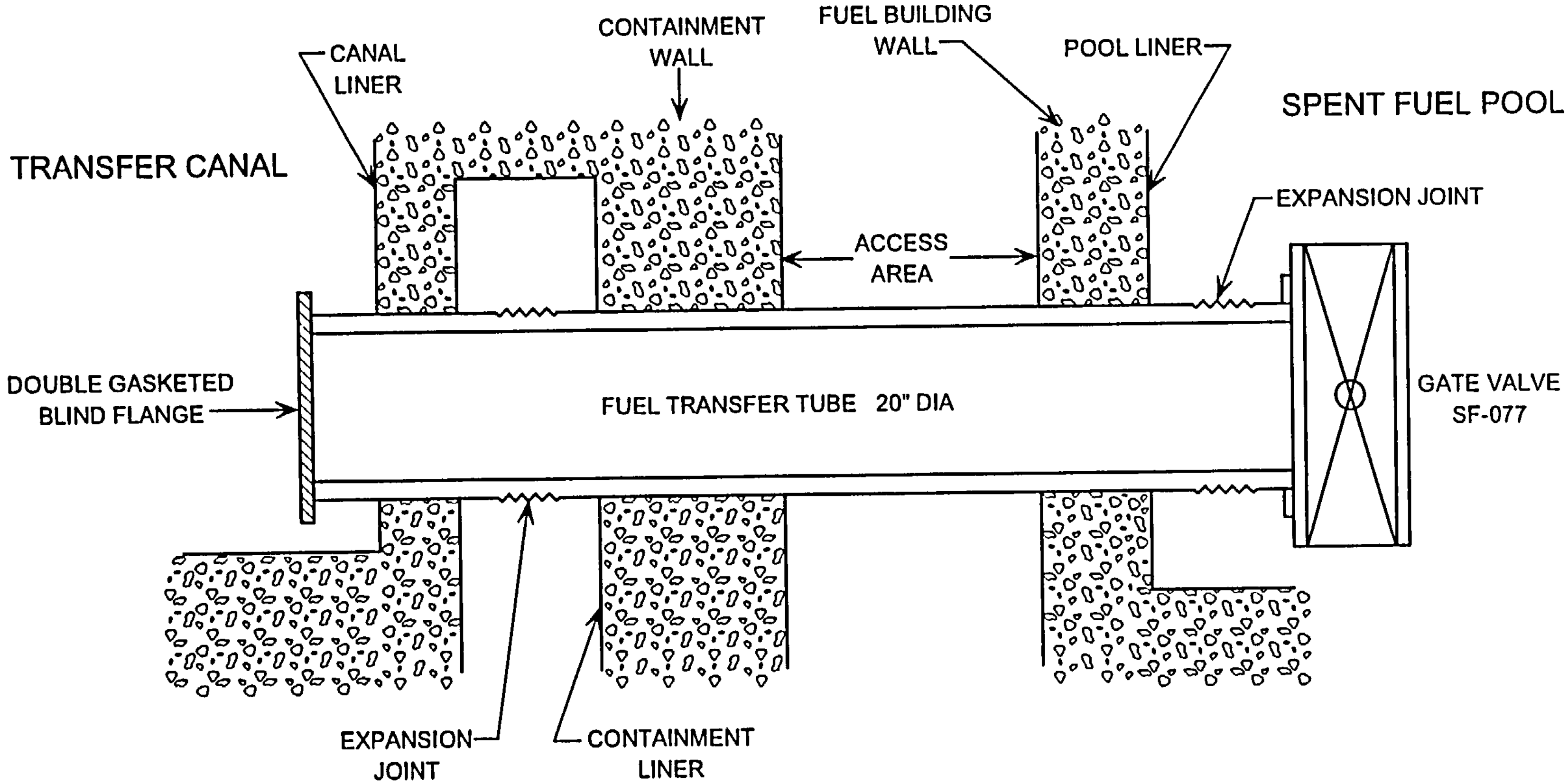

Figure 5.3-4 Fuel Transfer Tube Penetration

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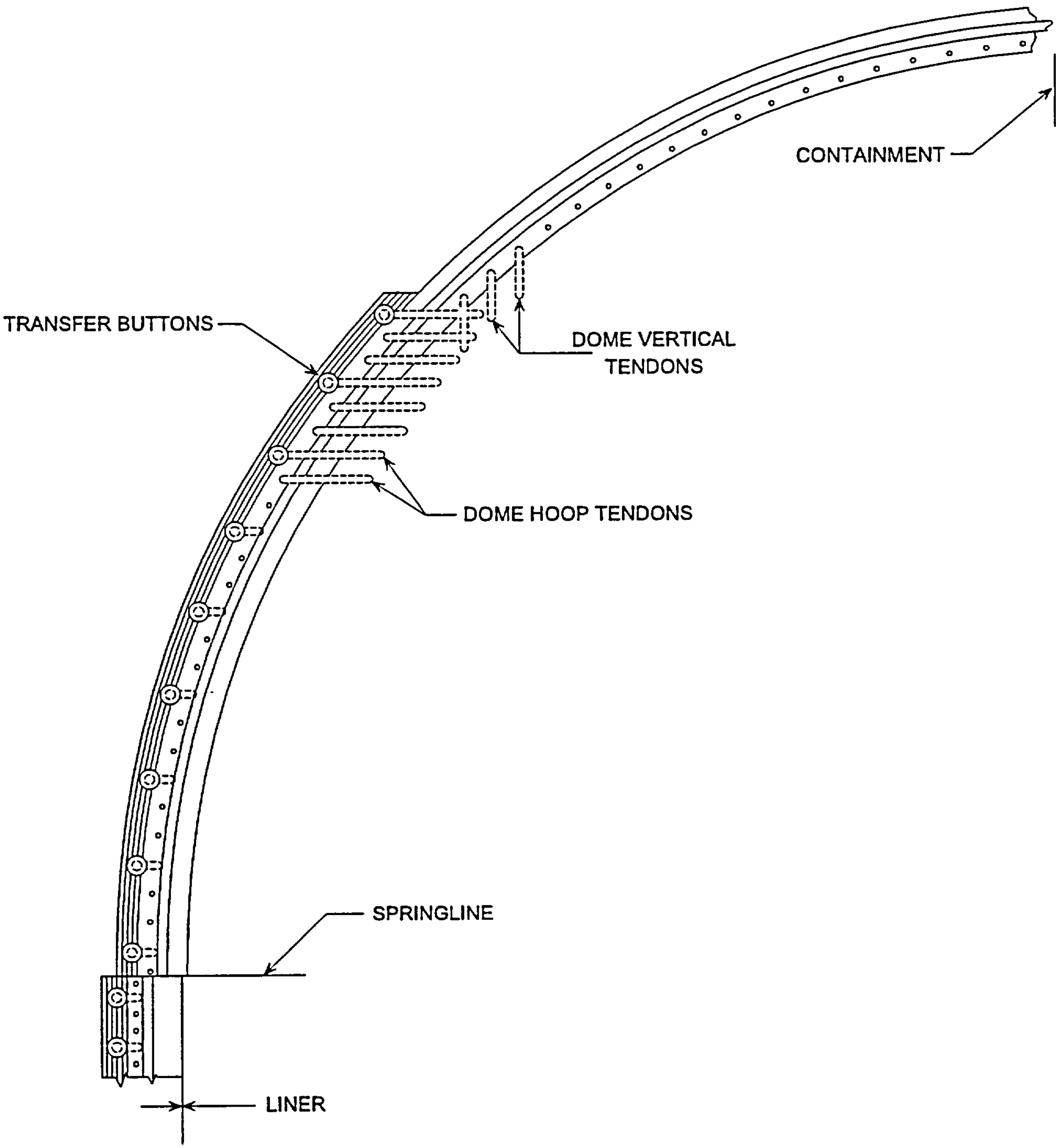


Figure 5.3-5 Transfer Buttons

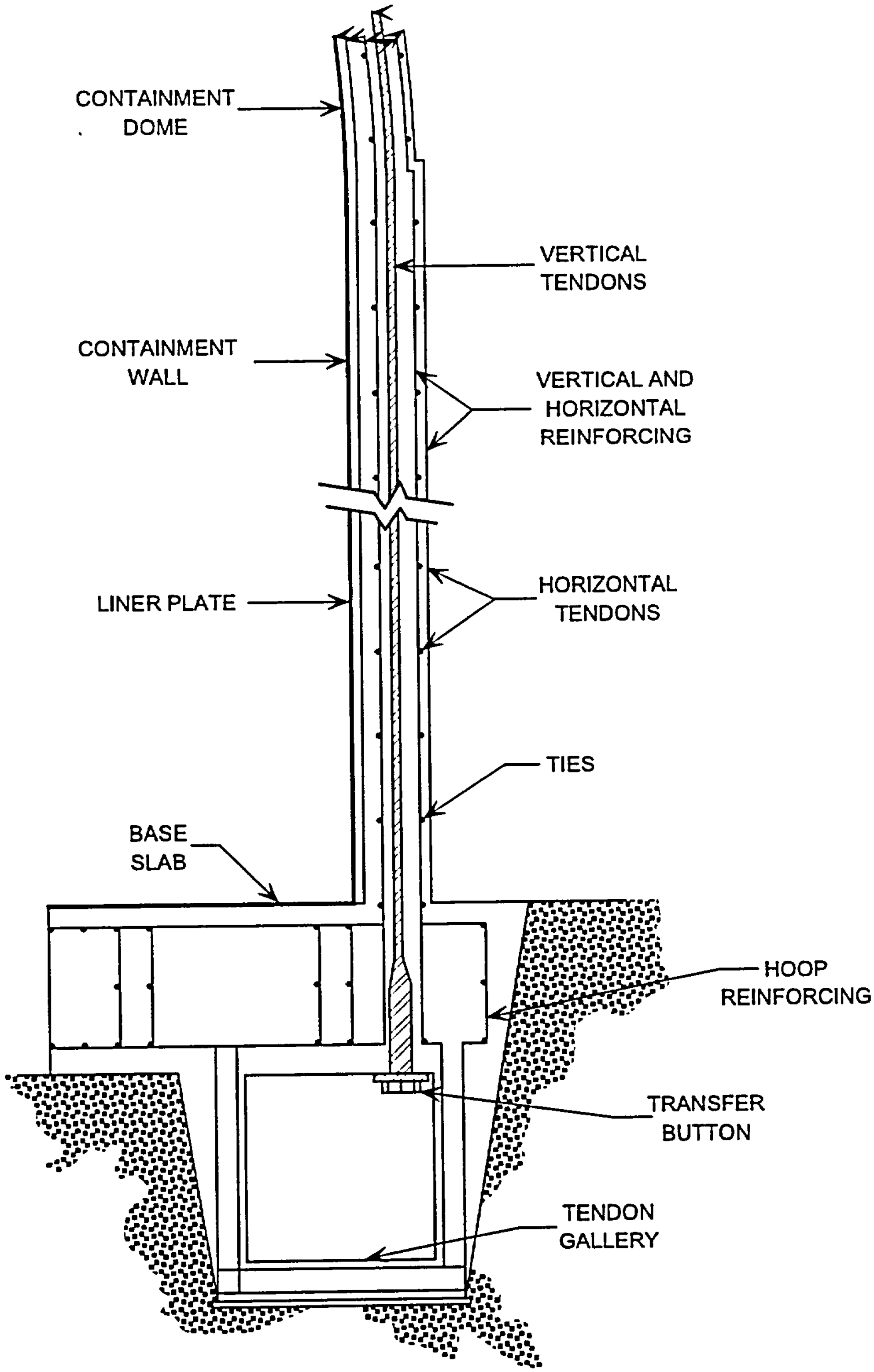


Figure 5.3-6 Section View of Wall and Vertical Tendons

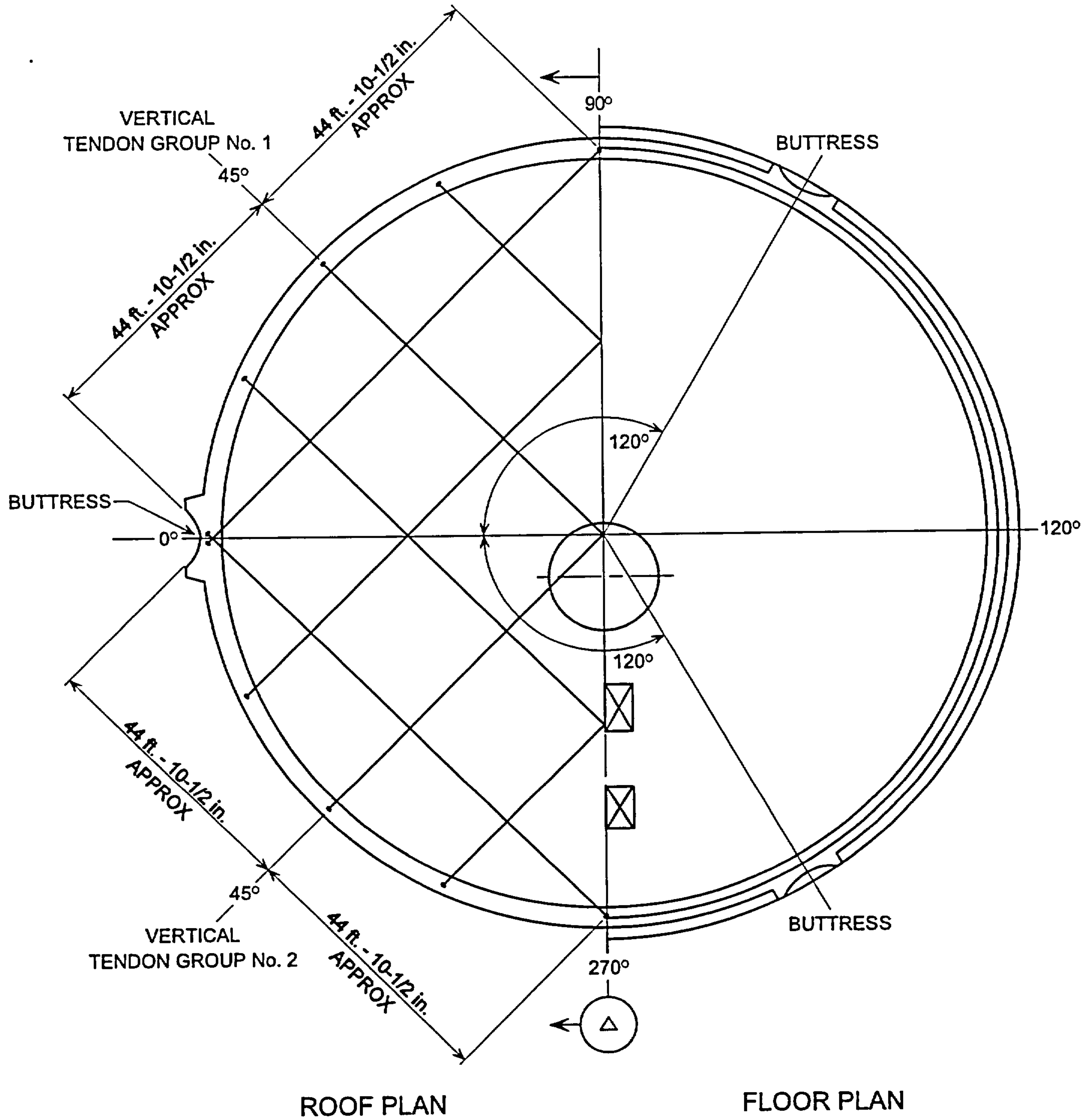
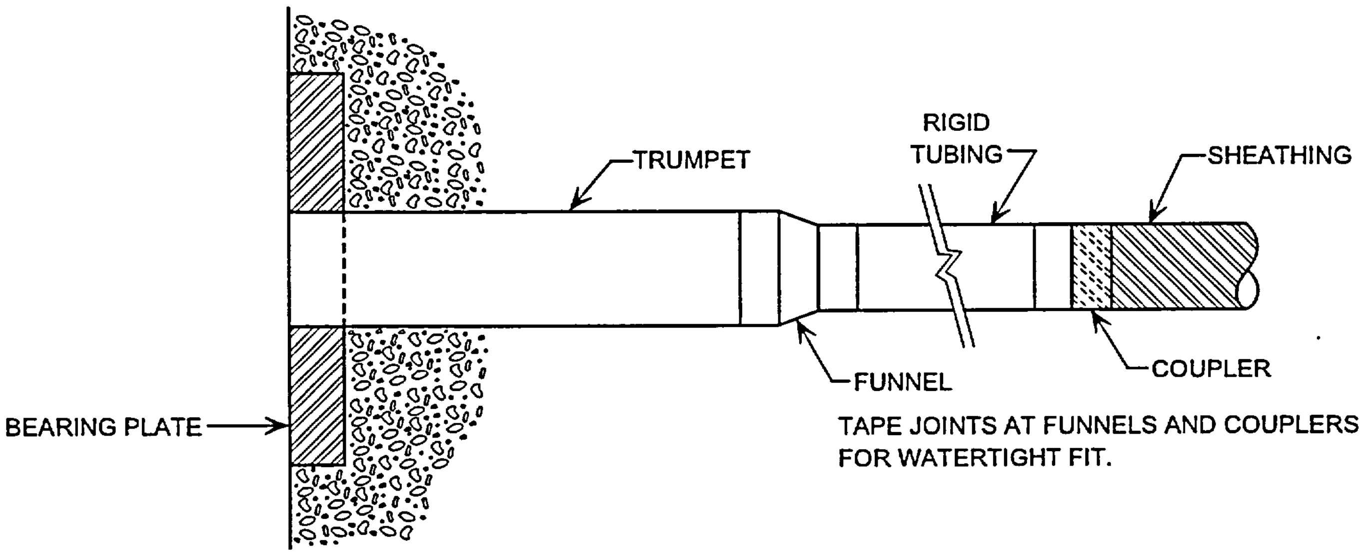
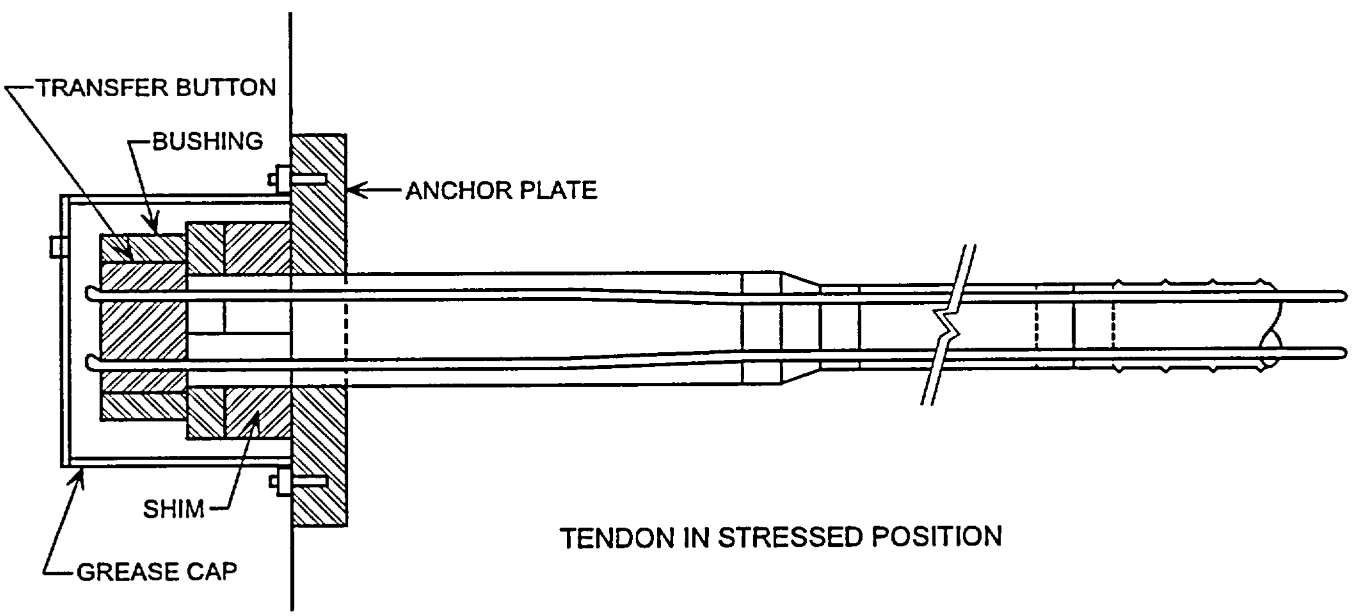


Figure 5.3-7 Dome Tendon Families



TRUMPLATE AND SHEATHING AS PLACED



TENDON IN STRESSED POSITION

Figure 5.3-8 Tendon Anchor Assembly

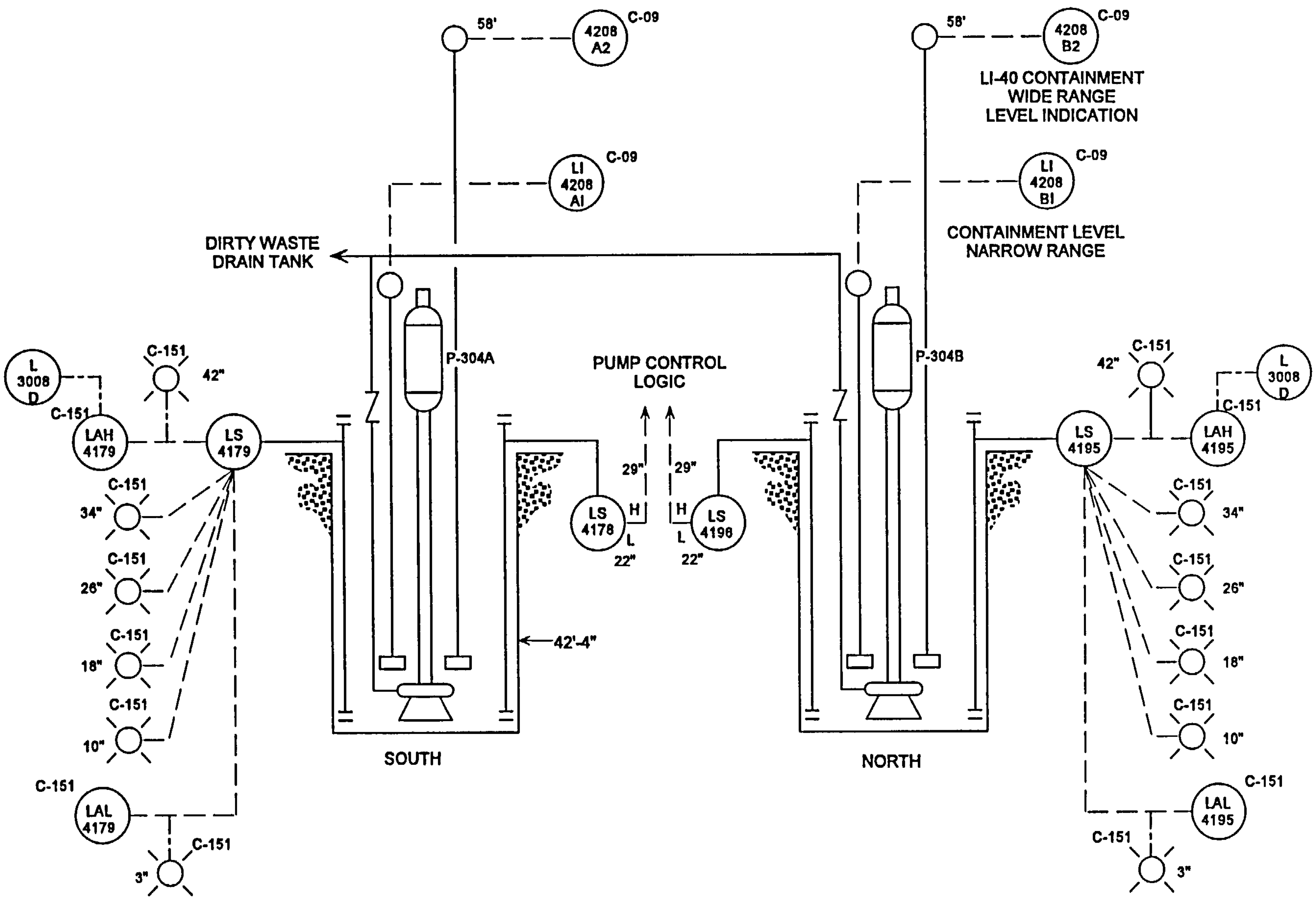


Figure 5.3-9 Containment Building Sump Level Indication

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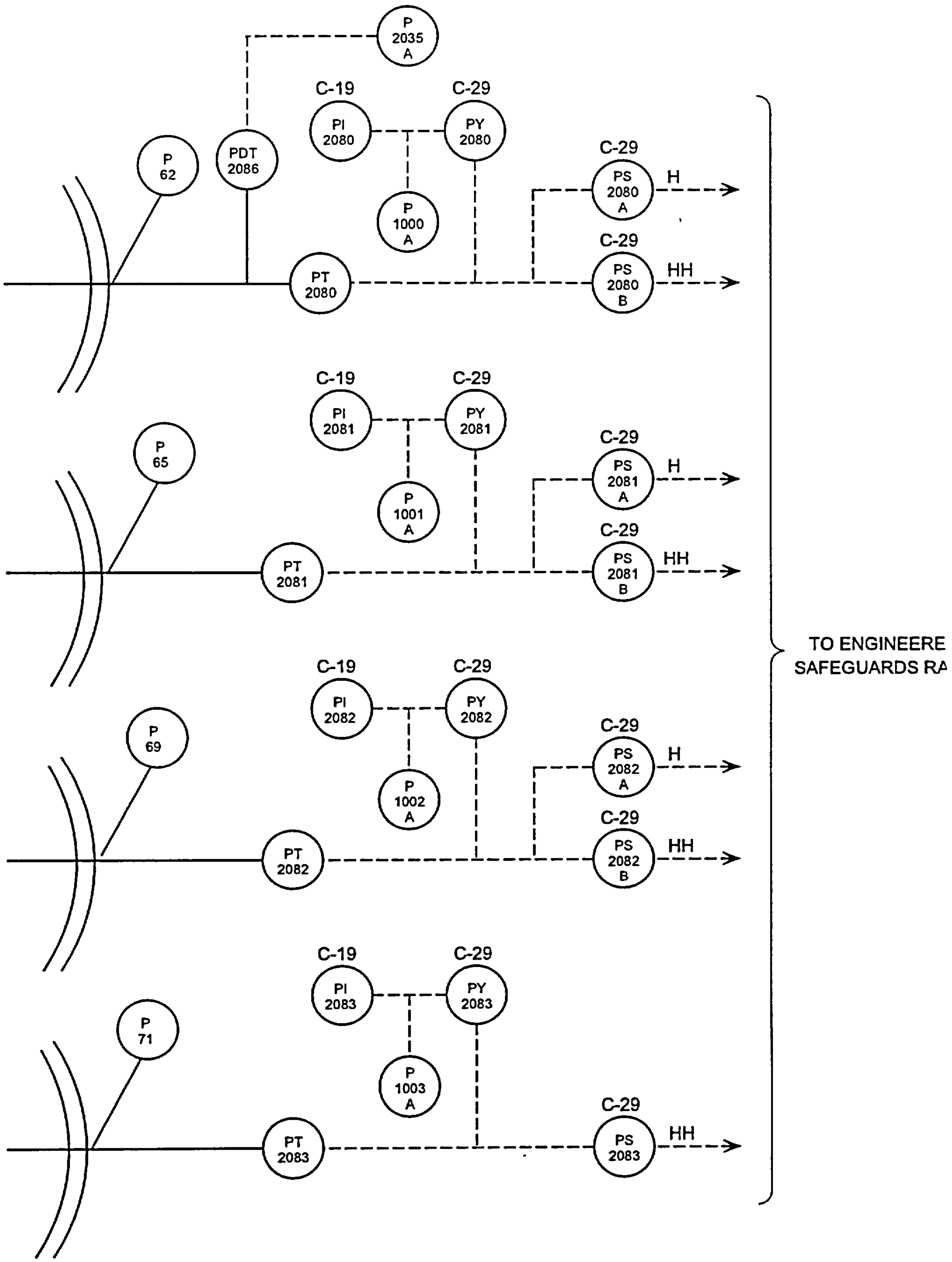

Figure 5.3-10 Containment Atmosphere Pressure

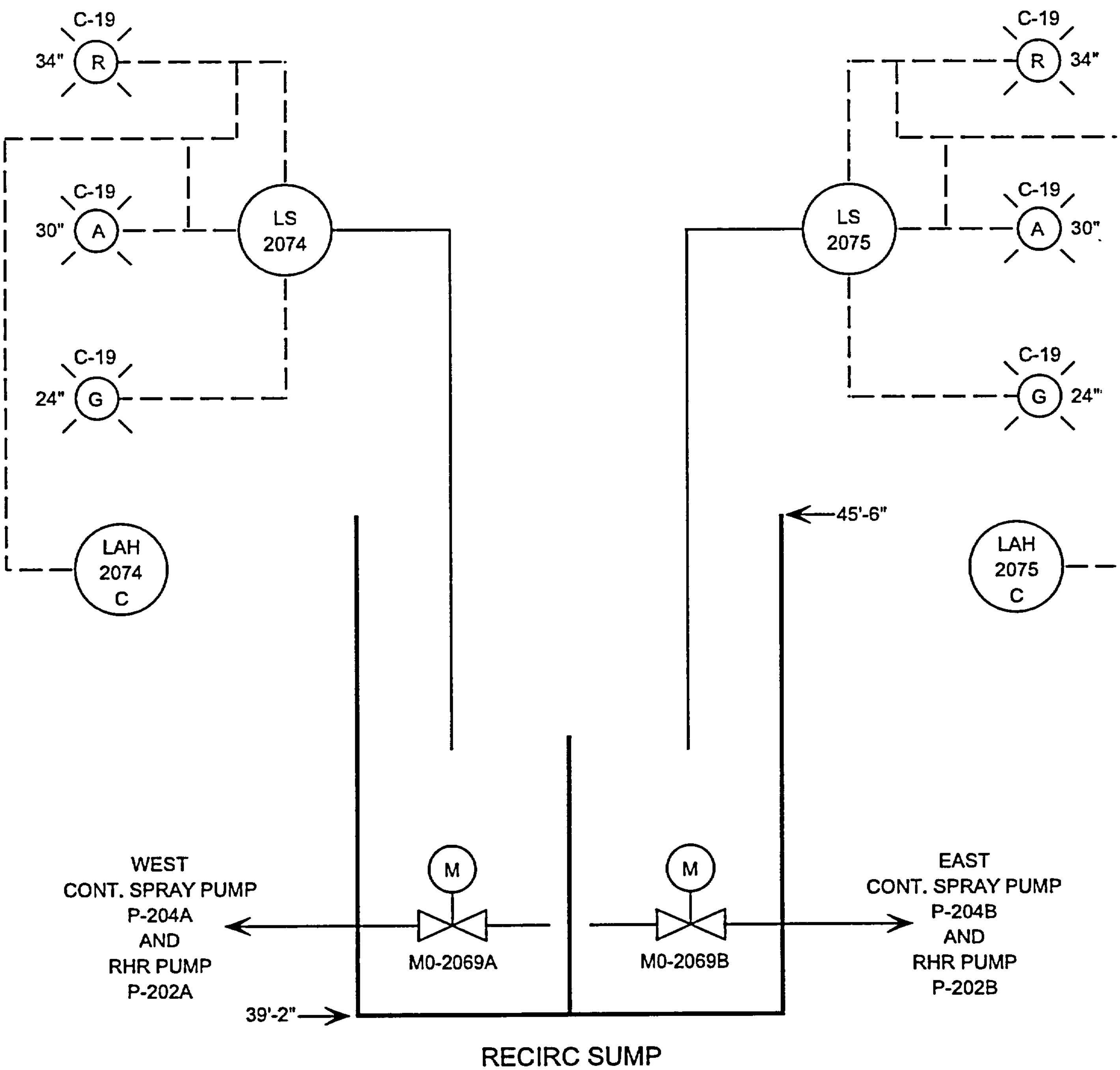

Figure 5.3-11 Recirculation Sump Level

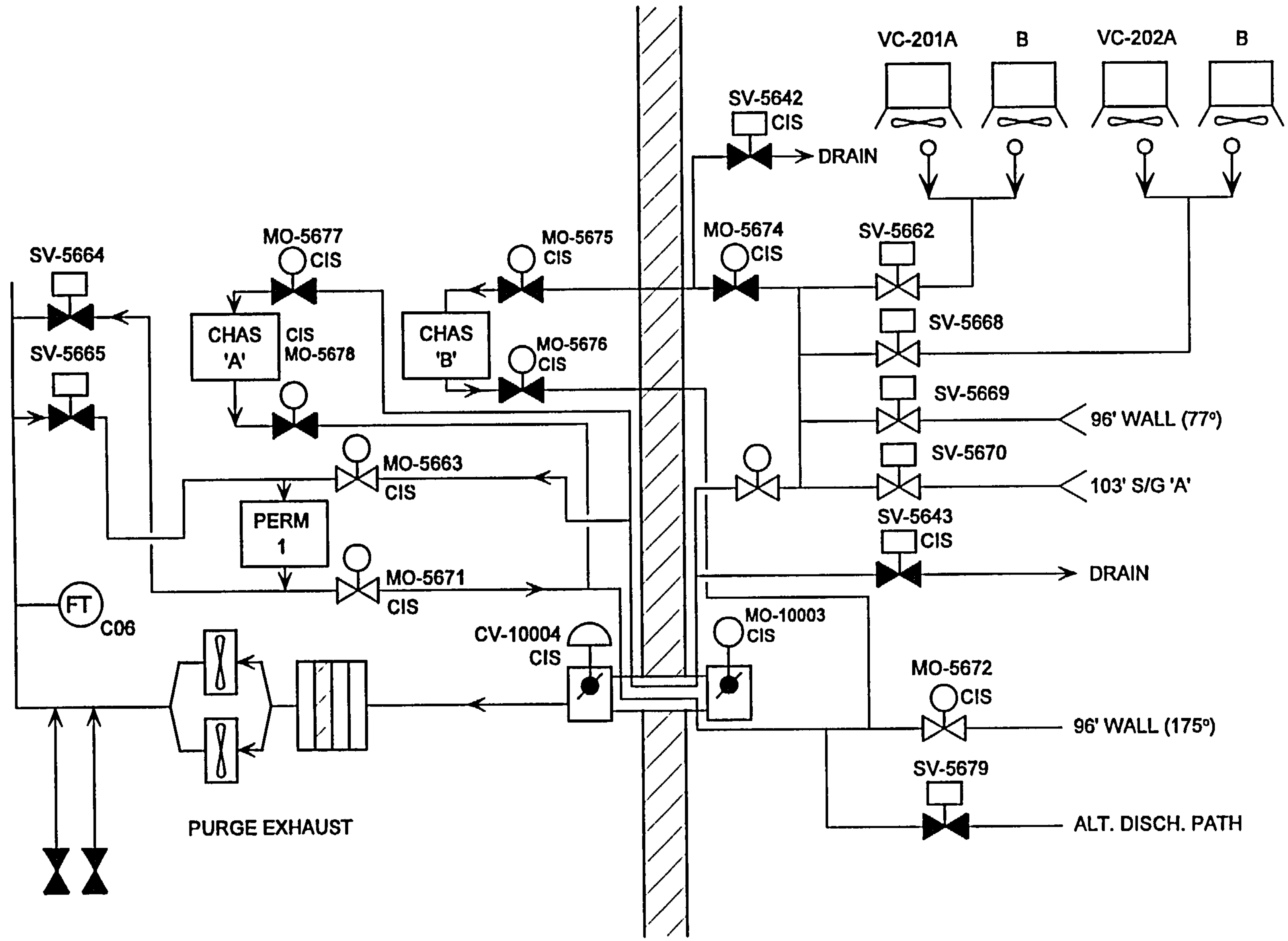

Figure 5.3-12 Containment Sampling Valves & Paths

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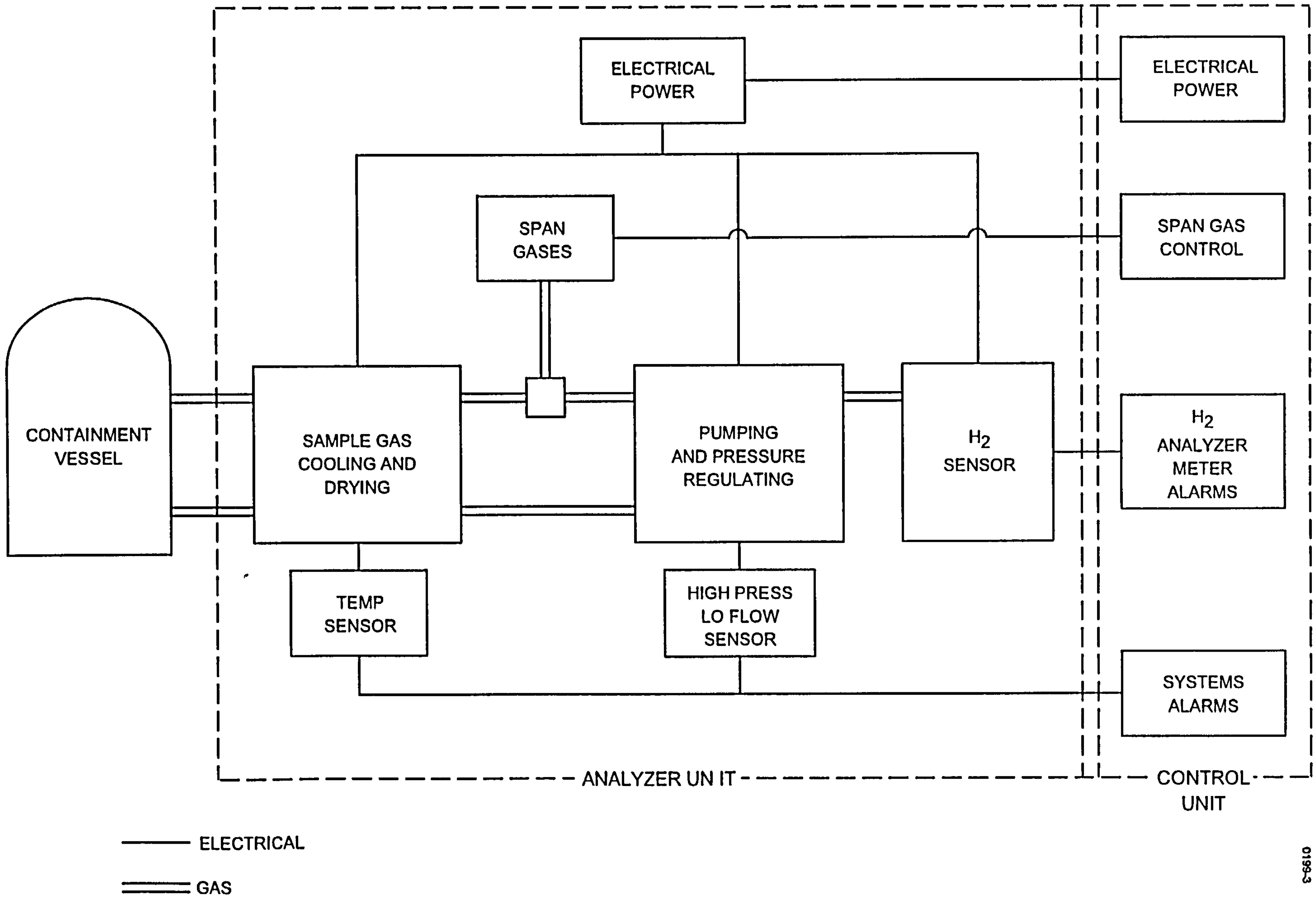

Figure 5.3-13 Hydrogen Sampling System

Westinghouse Technology Systems Manual

Section 5.4

Containment Temperature Pressure and Combustible Gas Control

TABLE OF CONTENTS

LIST OF FIGURES

5.4 CONTAINMENT TEMPERATURE PRESSURE AND COMBUSTIBLE GAS CONTROL

Learning Objectives:

1. State the purposes of Containment Ventilation Systems.
2. List the signals that automatically initiate purge and exhaust system isolation.
3. State the purposes of the Containment Spray System.
4. List the signals that automatically initiate the Containment Spray System.
5. State the purpose of the containment Hydrogen Recombiners.

5.4.1 Introduction

The purposes of the containment temperature, pressure and combustible gas control systems are as follows:

1. Control the containments temperature and pressure during normal operations.
2. Protect the containment barrier and minimize the leakage of radioactivity to the environment following an accident by reducing containment temperature and pressure.
3. Remove hydrogen from containment atmosphere to prevent explosive mixtures.
4. Remove radioactive iodine from containment atmosphere after LOCA.

The central safety objective in reactor plant design and operation is the control of the radioactive fission products. To ensure this objective is met, the containment must be designed and maintained so that the fission products are retained after operational and accidental releases inside the containment.

The containment temperature, pressure and combustible gas control systems are those systems which are necessary for reducing the release of airborne radioactivity and ensuring continued containment integrity. These containment systems function as necessary during normal operation and during the period following a postulated accident.

To minimize the leakage from the containment and any subsequent release of fission products after an accident, it is necessary to reduce the pressure and temperature inside the containment. Also the capability to remove the additional energy produced by reactor decay heat must be provided, so that the containment design pressure is not exceeded. Since it is not permissible to cool the containment by means of once-through ventilation (due to the increased radioactive release to the environment) the containment ventilation systems and the Containment Spray System (CS) provide the required heat removal.

To further limit the release of radioactive iodine, the CS system includes a chemical additive (sodium hydroxide) to scrub any iodine from the containment atmosphere and keep it in solution in the containment sump water.

It is also necessary to control the buildup of hydrogen gas produced by the metal-water reactions in the core, evolution of dissolved hydrogen from the reactor coolant, and other sources to prevent reaching flammable or explosive levels. This is necessary to protect containment integrity and its support equipment. Post accident combustible gas control systems are capable of sampling and analyzing containment samples and reducing the hydrogen levels by use of thermal recombiners.

The design requirements of the containment also limit the amount of materials such as aluminum and zinc which will chemically react with hydroxyl ions in the reactor coolant to produce hydrogen in the containment in a post-accident environment.

5.4.2 Containment Ventilation Systems

The purposes of the containment ventilation systems are as follows:

1. Control the containments temperature and pressure during normal operations to maintain operability of the containment and associated equipment.
2. Provide localized area ventilation for equipment inside containment.
3. Provide cleanup of the containment atmosphere for limited personnel access while at power, and for continuous access while shutdown.

To amplify the purposes as previously stated, the containment ventilation system is designed to accomplish the following:

a. Limit the average containment temperature between 50 - 120 °F during normal operation.
b. Provide 70,000 CFM of air to the control rod drive mechanisms with a maximum inlet air temperature of 120 °F.
c. Supply air to maintain the Reactor Coolant Pump (RCP) air temperature below 120 °F.
d. Provide cooling air for the primary concrete shield and the enclosed nuclear instrumentation thimbles.
e. Provide air to cool the reactor vessel supports.
f. Control containment airborne fission product gases, halogens and particulates in order to allow containment access without exceeding the occupational exposure dose limits of 10 CFR Part 20.
g. Control the releases of fission product halogens, particulates, and noble gases during containment purge such that the requirements of 10 CFR Part 50, Appendix A, may be met for overall plant radioactivity releases.

Nine (9) systems, operating together, meet the diverse design requirements for containment ventilation. These systems are as follows:

1. Purge Supply System.
2. Purge Exhaust and Refueling Cavity Supply and Exhaust System.
3. Control Rod Drive Mechanism (CRDM) Cooling System.
4. Pressurizer Compartment and Incore Instrumentation Switching Room Cooling System.
5. RCP Cooling System.
6. Reactor Cavity Cooling System.
7. Containment Air Cooler System.
8. Unit Heater System.
9. Cleanup Recirculation Units.

5.4.2.1 Purge Supply System

The purge supply system (Figure 5.4-1) is designed to insure safe, continuous access to the containment within after a planned or unplanned reactor shutdown by reducing the airborne particulates of the containment atmosphere. The system performs no function with respect to reactor safety. Prior to activating the purge and exhaust systems, the particulate and radioactive gas activity levels inside the containment must be monitored and will be used as a guide for routing the release from the containment. A sample is taken and analyzed before purging begins. The containment is normally isolated from the environment during power operations, with purging at power limited by the plants Technical Specifications

The purge supply system consists of an outside air intake, an automatic roll filter, a bank of High Efficiency Particulate Air (HEPA) filters, two 100 percent supply fans arranged in parallel, backdraft dampers, an outboard air operated isolation valve, a containment penetration, an inboard motor operated isolation valve and associated duct work.

The duct work out to and including the containment outboard isolation valve is designed to Seismic Category I specifications. The isolation valves are quick closing and are capable of closing within five (5) seconds for the motor operated valves and within three (3) seconds for the air operated valves upon receipt of a Containment Ventilation Isolation Signal (CVIS).

The purge supply fans are 480 Vac, 3 phase, 125 horsepower vane axial fans, each is capable of providing 100 percent of the required 50,000 CFM of air. The fans are controlled from the main control room and are interlocked such that only one fan runs at a time. To start a fan, both isolation valves must be open and the other fan must be in off. A pressure switch that senses pressure in the common discharge ductwork will start the standby fan on low pressure if the running fan is lost. When starting a fan, the control switch for the redundant fan must be in Pull-To-Lock until the running fan has increased the duct pressure above the pressure switch setpoint. The supply fans also have undervoltage and overcurrent protection.

The automatic roll filter is an 80 percent efficient filter designed to reduce the particulate and dust in the supply air to serve as a prefilter for the HEPA filter. The filter drive motor is interlocked with the supply fans such that it will run when either fan is on and is deenergized by a filter runout interlock.

The inboard and outboard dampers are controlled from the main control room. With the control switch in the normal position (not in lockout) they will automatically close on either a CVIS, a containment high radiation signal (PERM), or a manual Containment Spray Actuation Signal (CSAS). Damper open and closed indication is sensed by limit switches and is provided at the control switch and also on both the Containment Isolation System (CIS) and CVIS status panels in the main control room.

5.4.2.2 Purge Exhaust and Refueling Cavity Supply and Exhaust System

The purge exhaust and refueling cavity supply and exhaust system (Figure 5.4-2) work in conjunction with the containment purge supply system to provide one and one-half (1 1/2) containment air changes per hour. The purge exhaust is operated in mode 5 and 6 when continuous containment occupancy is desired.

The purge exhaust system consists of an inboard motor operated isolation damper, a containment penetration, an outboard air operated isolation damper, an automatic roll filter, a bank of HEPA filters, two 100 percent exhaust fans arranged in parallel, backdraft dampers, and associated ductwork. These components meet the same requirements as noted for the purge supply system.

The purge exhaust system exhausts to the containment purge vent at the top of the containment building. Exhaust air is monitored for gaseous, particulate and iodine activity.

The refueling cavity supply consists of two (2) fans drawing air from the containment atmosphere and discharging horizontally across the refueling cavity. The refueling cavity exhaust consists of two (2) fans drawing air from the inlets at the surface of the refueling cavity and discharging to the purge exhaust system. The refueling cavity supply and exhaust system's design objective is to rapidly remove and exhaust water vapor and fission products escaping from the fuel pool surface, to reduce the burden in the containment atmosphere during refueling.

5.4.2.3 CRDM Cooling System

The CRDM cooling system is designed to remove heat from the CRDMs and release it to the containment atmosphere. Air quantity and static pressure drop requirements are based on maintaining the CRDM temperatures at ≤300 °F with normal operation of two (2) of the four (4) fans and a containment air temperature of 120 °F. The system shall be in operation anytime the RCS temperature is ≥ 350 °F. The system draws a minimum of 70,000 CFM of air through the shroud enclosing the CRDMs an discharges it upward where it rises by convection to the containment air coolers. The four (4) fans are mounted on the CRDM missile shield. The system is Seismic Category II and is not connected to an emergency power supply.

5.4.2.4 Chill Water System

The chill water system (Figure 5.4-4) is not a safety related system, however, it is an important system that removes heat generated in the digital rod position indication and control cabinets, the pressurizer compartment, the incore instrumentation switch room, the reactor cavity, and the process sample system. The majority of the equipment is located in the containment where it is exposed to ambient temperatures that range from 90 °F to 110 °F. Any equipment cooled by chill water may overheat if flow is interrupted it is, therefore, necessary to run the chill water system as much as possible.

5.4.2.5 Pressurizer Compartment and Incore Instrumentation Switching Room Cooling System

The pressurizer compartment and incore instrumentation switching room cooling system (Figure 5.4-5) is designed to circulate containment air to these areas during normal operation and to supply 80 °F air when personnel must access these areas. The system consists of two independent systems, one for the pressurizer compartment and the other for the incore instrumentation switching room.

Each system consists of a dust filter, a chill water cooling coil, a supply fan, and distribution ductwork. The systems can cool their areas sufficiently for personnel entry by use of the fans and cooling coils, and are capable of maintaining suitable compartment temperature for normal equipment operation by use of the fans only. The cooling coils can be served by either of two chillers, each of which can provide 100 percent of the required design capacity. The system is designed as a Seismic Category II system.

5.4.2.6 Reactor Coolant Pump Cooling System

The RCP cooling system (Figure 5.4-6) is designed to distribute cooling air to each of the RCP motors. The system consists of four (4) independent subsystems, one for each RCP. Each subsystem consists of two (2) 100 percent capacity fans arranged in parallel, backdraft dampers, and associated ductwork The system is arranged so that the subsystem serving one RCP is entirely independent of the subsystems serving the other RCPs. The RCP cooling system is a Seismic Category II system.

5.4.2.7 Reactor Cavity Cooling System

The reactor cavity cooling system is designed to circulate chilled air through the incore instrumentation tunnel and up through the cavity around the reactor vessel and its supports. The system is designed to handle the portion of the cooling load which occurs below the cavity seal. An upper exit air temperature limit of 110 °F is imposed on the system based upon the fact that this air subsequently enters the CRDM cooling system. The system is a Seismic Category II system.

5.4.2.8 Containment Air Cooler System

The containment air cooler system (Figure 5.4-8), when operating under normal conditions, and in conjunction with other normal containment ventilation systems, removes the heat losses from normally operating equipment and from radiation and convective transfer from the primary and secondary coolant systems.

The system provides the heat removal capacity to maintain containment temperature below 120 °F during normal plant operation. The system also has the capacity to significantly reduce the containment heat energy following a LOCA. The containment air cooler system is an Engineered Safety Features (ESF) system and is Seismic Category I.

The system consists of eight (8) individual air cooler units of equal capacity mounted above the containment spray header. Each unit contains a fan enclosed by an airtight roof and floor and surrounded on four sides by 12 cooling coils. The fans draw air horizontally across the cooling coils and discharge the cooled air downward into the containment. The cooling coils are supplied with cooling water from the Component Cooling Water System (CCW). The eight (8) air coolers are divided into two (2) groups of four (4) units. Each group of cooling units is designated as train A, and train B. Each of the four (4) train A cooling coils is supplied with cooling water from the train A CCW loop. Each of the four (4) train A cooler fans is supplied power from the train A 480 Vac ESF electrical bus. Air cooler fan and cooling coils for train B are similarly supplied. Physical separation and barriers are provided between all train A and train B components.

The containment air cooler system is designed to maintain containment air temperature below 120 °F during normal operation with six (6) of the eight (8) cooler units in operation when CCW is at design flow and temperature. The containment air cooler system will also remove heat energy from the containment atmosphere in the event of a LOCA in order to suppress any resultant increase in containment pressure and temperature. Additionally, the system is designed such that a single failure of any active component during the injection phase, or any active or passive failure during the recirculation phase, will not degrade the system's ability to meet the design objectives.

The air cooler fans are 480 Vac, three phase, 125 horsepower, downblast discharge, vane axial fans. Each fan has a design capacity of 100,000 CFM and is powered from the ESF electrical buses. The fan motors are designed to ensure required air and steam mixture flow is achieved under design basis event conditions. Under these conditions the fan motor output horsepower is increased because fan brake horsepower requirements vary with the density of the containment atmosphere.

The containment air cooler fans are normally controlled from the main control room. The control switches have STOP, NORMAL, and START positions with PULL-TO-LOCK and spring return to NORMAL features. In NORMAL, the air coolers can be automatically started by the DBA load sequencer.

During normal operation the CCW system supplies both containment air cooler trains. The CCW supply line has two (2) motor operated valves which are normally closed with an orificed bypass line which supplies 1950 gpm of CCW to each train. The air cooler fans are started as necessary to maintain containment temperature between 50 and 120 °F. Not more than three (3) air coolers in each train are run simultaneously except for periodic testing. Normal air cooler operation has six (6) of the eight (8) coolers in operation. In modes 1 - 4, at least three (3) of the coolers in each train shall be running.

Humidity in the air will condense on the coolers. The condensation from the coolers is collected in a drip pan below each cooler. The drip pans are connected to a common collection header and directed to a condensate collection pot. When the level in the pot reaches a setpoint, a motor operated valve opens and allows the pot to drain to the containment sump. The number of times that the motor operated valve opens is recorded in the control room. Condensate pot drain cycles are proportional to the condensation rate. The condensation rate is proportional to humidity and CCW temperature. Humidity is proportional to the leak rate of liquid systems inside the containment.

A Safety Injection Signal (SI) causes startup of any standby CCW train and the separation of the two CCW trains. The motor operated valves in the containment air cooler supplies open and provide a minimum of 6000 gpm flow to each cooler train. The DBA load sequencer automatically starts any air cooler fans that are in NORMAL.

5.3.2.9 Unit Heater System

The unit heater system consists of ten unit heaters located throughout the containment building. The unit heaters are designed to provide heating to allow for personnel comfort during periods of personnel access. The electrical heating coils and blower units are locally controlled and have a variable temperature control.

5.3.2.10 Cleanup Recirculating Units

During normal operation two (2) Seismic Category II recirculating filter units are available for intermittent or continuous operation to control the buildup of airborne halogens and particulates which result from small RCS leaks within the containment. The cleanup filters (in conjunction with the containment purge system) have two (2) objectives. The first is to maintain airborne fission product levels below the 10 CFR Part 20 limits for occupational exposures to allow safe access to the containment. The second is to reduce fission product releases to the environment to levels as low as reasonably achievable when containment purging is required. The cleanup recirculation system performs no function with respect to reactor safety.

The cleanup recirculation system (Figure 5.4-9) consists of two (2) separate units. Each unit draws 4000 CFM if containment air through a prefilter, a HEPA filter, and a carbon absorber. With both units in operation, approximately one (1) containment air volume will be recirculated every four (4) hours.

5.4.3 Containment Hydrogen Control

Three (3) systems are designed to control hydrogen in the containment building. The systems are the hydrogen control system, the hydrogen mixing system and the hydrogen vent system.

Following a LOCA, hydrogen gas may accumulate within the containment as a result of:

1. Metal-water reaction involving the zirconium fuel cladding and the reactor coolant.
2. Radiolytic decomposition of the post accident emergency cooling solutions.
3. Corrosion of metals caused by the solutions used for emergency cooling or containment spray.

If a sufficient amount of hydrogen is generated, it may react with the oxygen present in the containment atmosphere or with the oxygen generated following the accident. In this event, the reaction could take place at rates rapid enough to lead to high temperatures and significant pressures in the containment. This could conceivably result in the loss of integrity and/or damage to systems and components essential to the control of the post LOCA conditions.

5.4.3.1 Hydrogen Control System

The hydrogen control system consists of two (2), in containment, hydrogen recombiner units. The recombination units draw in containment air by natural convection and heat it up to a temperature range of 1150 - 1400 °F. At a temperature of approximately 1135 °F, hydrogen and oxygen recombine and thereby reduce the containment hydrogen concentration. The units are completely enclosed and the internals are protected against impingement by containment spray. The units are designed as Seismic Category I components and are capable of functioning during a LOCA.

The recombination units (Figure 5.4-10) consist of an inlet preheater section, a recombination section, and a mixing chamber. Containment air is drawn into the units by natural convection via inlet louvers and passes through the preheater section. The preheater section consists of a shroud placed around the central heaters to take advantage of heat conduction through the walls. Preheating the air accomplishes the dual function of increasing the system efficiency and evaporating any moisture droplets which may be entrained in the air. The warmed air then passes through a specifically sized flow orifice and flows vertically upward through the recombination section where it's temperature is raised and the hydrogen and oxygen recombine.

The recombination section contains five (5) banks of vertically stacked electric heaters. Each heater bank contains 60 U-type heating elements. After heating and recombination, the air rises to the mixing chamber. In the mixing chamber the hot air is mixed with the cooler containment air that enters the mixing chamber through the lower part of the louvers located on three sides of the mixing chamber and is then discharged to the containment atmosphere.

In the event of a LOCA, the hydrogen concentration in containment will be determined using the CHAS and the recombination units will be started as required (normally with the hydrogen concentration >.5 percent and ≤ 4.0 percent). The units are started from the main control room.

5.4.3.2 Hydrogen Mixing System

The hydrogen mixing system (Figure 5.4-11) consists of two (2) fans which take a suction from the highpoint of containment. The exhaust from the fans is directed downward into the lower containment air space where rapid mixing by turbulence created by the containment air coolers occurs.

The hydrogen mixing system is an ESF system and is designed to meet Seismic Category I requirements. The system must be operable for continued plant operation.

5.4.3.3 Hydrogen Vent System

The hydrogen vent system (Figure 5.4-12) consists of two (2) redundant subsystems to permit controlled purging of the containment. Exhaust flow is provided by one of two fans. Flow is ducted outside of the containment through motor operated isolation dampers which are remotely opened during venting. Each fan has a filter assembly containing a roughing filter, a HEPA filter, a carbon absorber, and a HEPA after filter to reduce the activity of the halogens and particulates in the vent flow. The fans exhaust through motor operated dampers to the purge exhaust system.

Filtered air from the purge supply system is supplied by the hydrogen vent system to replace the air that is removed and maintain containment pressure within normal limits. The intake path is through a normally open manual valve and remotely controlled motor operated dampers.

The eight (8) motor operated containment isolation dampers can be operated from the main control room. A CIS phase A (channel A for inboard dampers and channel B for outboard dampers) or a CVIS will cause the dampers to automatically close. Damper open and closed indication is provided in the main control room.

5.4.4 Containment Spray System

The Containment Spray (CS) system is designed to accomplish three (3) main objectives:

1. Limit the peak containment pressure below its design pressure of 60 psig following a worst case LOCA.
2. Reduce the concentration of fission product iodine in the containment atmosphere following a LOCA.
3. Keep spray entrained iodine in the containment recirculation sump while maintaining the sump water at a basic pH.

The Containment Spray (CS) system is an engineered safety features system which functions to reduce the containment pressure and airborne fission products in the containment atmosphere following a steam break or a loss of coolant accident. The pressure reduction is accomplished by spraying cool, borated water from the refueling water storage tank into the containment atmosphere. Sodium hydroxide is added to the containment spray water to enhance absorption of the airborne fission product iodine and to retain the radioactive iodine in solution.

5.4.4.1 System Description

The containment spray system (Figure 5.4-13) is comprised of two 100 percent independent and identical subsystems, with the exception of a common spray additive tank and a common containment spray pump recirculation test line to the Refueling Water Storage Tank (RWST). The system is composed of two separate trains from the pump suction to and including the containment spray headers. Redundancy of equipment in the containment spray systems satisfies the single failure design criteria.

The containment spray system consists of two (2) pumps, two (2) spray ring headers, a spray additive tank, a number of motor operated isolation valves, and all necessary piping, instrumentation and accessories to make the system operable.

During the injection phase, the containment spray pumps take a suction from the RWST. The fluid being pumped is borated water with 30 weight percent sodium hydroxide (NaOH) from the spray additive tank. The NaOH is added to the spray water to enhance removal of elemental iodine from the containment atmosphere. The spray water pH value is maintained between the range of 9.5 and 11.0 during injection. This pH range helps ensure iodine removal from the containment atmosphere by the spray droplets, and maintains compatibility between the spray fluid and the materials with which it comes in contact.

An inert atmosphere of nitrogen is maintained over the NaOH solution to prevent exposure to air which would result in degradation of the solution. The NaOH is piped to a liquid jet eductor located on a bypass line around each of the spray pumps.

The bypassed spray water from the pump discharge passes through the eductor and draws the NaOH solution into the eductor where it mixes with the spray water and is piped back to the suction of the pump.

The containment spray pump discharge is piped to ring headers located on the inside of the containment dome. Each ring header contains spray nozzles aimed in various predetermined directions to ensure maximum spray coverage of the containment. The ring headers are piped to provide adequate coverage of the containment even if one spray pump fails to start on demand.

There are recirculation lines from each pump to the refueling water storage tank to allow testing of the pumps without initiating containment spray. There is a locked-closed, manually operated globe valve in each line and one in the common line for test flow alignment and throttling.

5.4.4.2 Component Description

Containment Spray Pumps

The Spray pumps are vertical, single stage centrifugal pumps. Each pump is rated for 2800 gpm at approximately 200 psig. The pumps are made of a material compatible with the NaOH additive and the boric acid solution of the RWST. The spray pumps each have a design flow equal to 100 percent of the heat removal capability necessary to maintain the pressure in the containment below its design maximum. The design discharge pressure of the pumps is sufficient to continue at rated flow with the RWST almost empty, against a head equivalent to the sum of design containment pressure, the head of the upper most nozzle, and the line and nozzle pressure losses.

As a component of an ESF system, each pump is supplied electrical power from an independent ESF vital bus. The CS pumps can be individually controlled from two (2) separate locations. A selector switch, mounted on the local switchgear panel, determines which station, LOCAL or REMOTE has control. In LOCAL, START or STOP may be selected on the switchgear panel. In REMOTE, control is shifted to the main control room. With the pump control selected to REMOTE and the main control room switch selected to AUTO, the CS pump will start upon receiving a containment spray system actuation signal (CSAS) coincident with a safety injection actuation signal (SIS).

Spray Additive Tank

The spray additive tank is constructed of carbon steel and lined with a special coating on it's interior surface to protect the tank from the highly corrosive caustic contained within. The tank capacity is 4000 gals. of a 30 weight percent solution of NaOH.

To prevent decomposition and evaporation of the NaOH solution during storage a nitrogen blanket is maintained above the solution. The normal nitrogen pressure is two (2) psig. The capacity of the spray additive tank is selected so that upon mixing with the water from the RCS, the RWST and the safety injection accumulators the final concentration in the containment sump will have a pH that ensures effective iodine removal and retention during sump recirculation.

Spray Additive Eductor

The means of adding NaOH to the spray water is provided by a liquid jet eductor. This device uses the kinetic energy of a pressurized liquid to entrain another liquid, mix the two, and discharge the mixture against a counter pressure. The pressurized liquid is the spray pump discharge which is used to entrain the NaOH solution and discharge the mixture into the suction of the spray pumps. During the initial injection phase of CS operation, the RWST supplies water to each spray pump. Ninety five percent of the pump flow is directed to it's respective spray ring, the remaining five (5) percent bypasses the spray ring to supply motive force for that train's additive eductor. From each eductor, the mixture of NaOH and borated water enters the suction of it's associated CS pump.

Spray Headers and Nozzles

There are two (2) concentric spray rings in the dome of the containment. The rings are redundant with half being supplied from each spray pump through a normally shut motor operated isolation valve. The design of these rings provides a full 360° coverage of the containment by each ring. Protruding from each spray ring are hollow core nozzles. Train A has 176 spray nozzles and Train B has 178 spray nozzles. Each nozzle is capable of delivering approximately 15 gpm of atomized spray water. The spray nozzles are not subject to clogging by particles less than 1/4 inch in maximum dimension.

The spray rings are a critical portion of the CS system. The pumps provide proper flow of spray water and the eductors supply the proper amount of NaOH solution. The spray nozzles deliver this mixture in a fine mist to the containment atmosphere. Atomization increases the effective surface area of the water spray. This allows each drop of water to absorb more internal energy from the atmosphere, increasing the rate at which the containment pressure and temperature decrease. Secondly, atomized spray allows each droplet to be exposed to more iodine and therefore increase removal efficiency.

The positioning of the spray nozzles is such that both spray ring headers develop a blanket of mist covering over 90 percent of the containment building volume. The remaining 10 percent consists of completely flooded lower regions as well as the volume above the spray rings. A portion of the spray pattern intentionally strikes the containment inner walls to create a liquid layer that acts as an additional barrier to fission product leakage.

5.4.4.3 Containment Spray Actuation Signals

Containment spray actuation signals are developed by two completely redundant channels (A and B). Either of the following conditions will cause both a Train A and a Train B CSAS:

1. Containment high-high pressure as sensed by two (2) out of four (4) containment building pressure detectors, or
2. Manual operator initiation from the main control room.

Manual initiation of a CSAS from the main control room is unique from manual initiation of other safeguard functions such as SI or CIS. Two (2) actuation switches are provided, however, both must be simultaneously positioned to the ACTUATE position to initiate a CSAS. The simultaneous operation of these switches is also required to cause a CIS phase B (CCW to the reactor coolant pumps for the respective train will close). A CSAS, either from manual or automatic initiation, is insufficient, by itself, to start the CS pumps. A SI signal must be present to energize the respective DBA load sequencer, allowing it to start it's associated pump. A SI signal is not required to open the spray ring isolation valves or NaOH tank outlet valves, the CSAS is sufficient to cause valve movement. Also the containment pressure bistables for spray actuation are energized to actuate where other bistable are deenergized to actuate. The status of the four individual high-high pressure channels is monitored by trip status lamps on the engineered safety features trip status panel in the main control room. These design features were included to make the possibility of an inadvertent and unnecessary spraydown of the containment building unlikely.

Each CSAS has an associated retentive memory with an actuation block, this means that once an actuation signal is initiated, it will remain present even if the containment building pressure returns to normal. The operator must manually reset the signals when the appropriate plant conditions are met. If the operator resets the CSAS prior to the containment building pressure returning below the high-high actuation setpoint, automatic reinitiation will not occur because of the actuation block feature. When the containment building pressure decreases below the setpoint, the block feature will automatically clear.

5.4.5 Summary

Containment ventilation systems are provided to maintain the containments temperature, pressure, humidity, and activity within appropriate limits for equipment and personnel protection.

A purge supply and exhaust system maintains proper environmental quality for unlimited access during periods of reactor shutdown. This is accomplished by continuously circulating outside air through the containment.

Internal recirculation systems provide cooling to remove heat generated by plant components and heat losses from reactor coolant system piping. Other recirculation systems force this cooled air directly to components requiring additional cooling. Recirculation systems with HEPA and charcoal filters are used to reduce containment activity.

Additional systems distribute air from recirculation or purge systems to minimize exposure of refueling personnel to airborne initiated water vapor.

A containment spray system is included to limit the peak containment pressure following a LOCA and to reduce the concentration of iodine in the containment building. Additionally, the system keeps spray entrained iodine in the containment recirculation sump while maintaining the sump water at a basic pH.

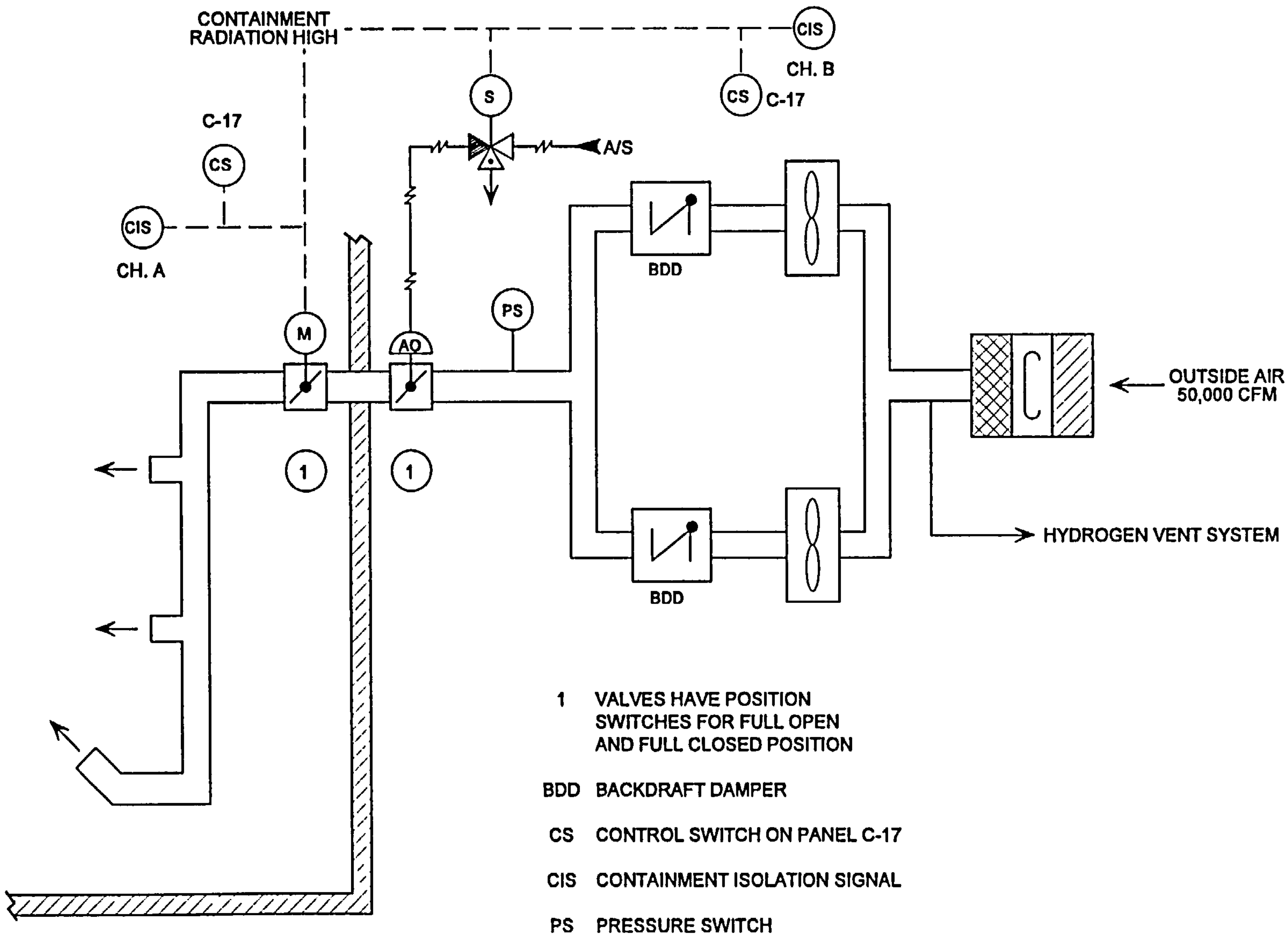

Figure 5.4-1 Purge System

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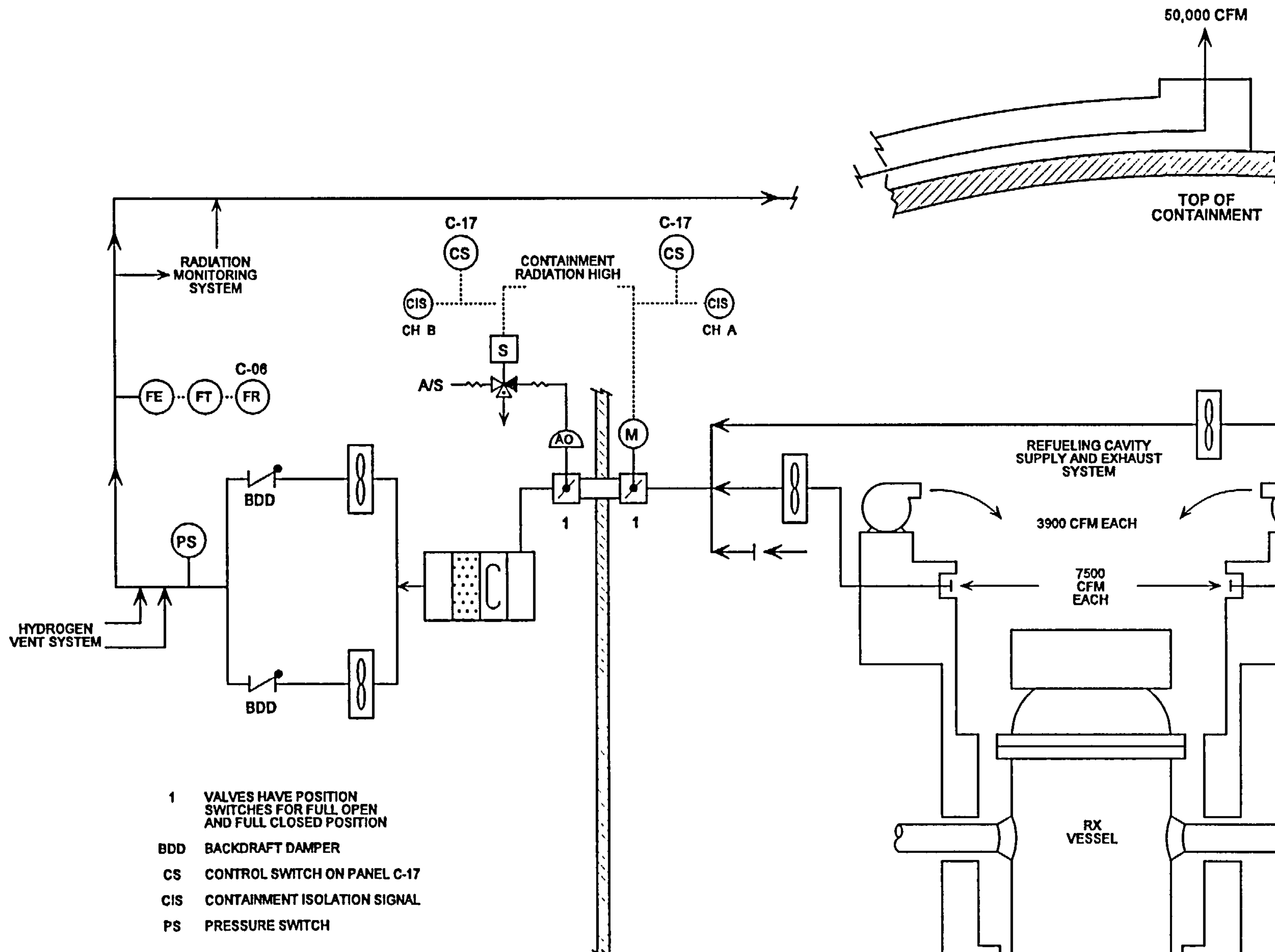

Figure 5.4-2 Purge Exhaust and Refueling Cavity Supply and Exhaust

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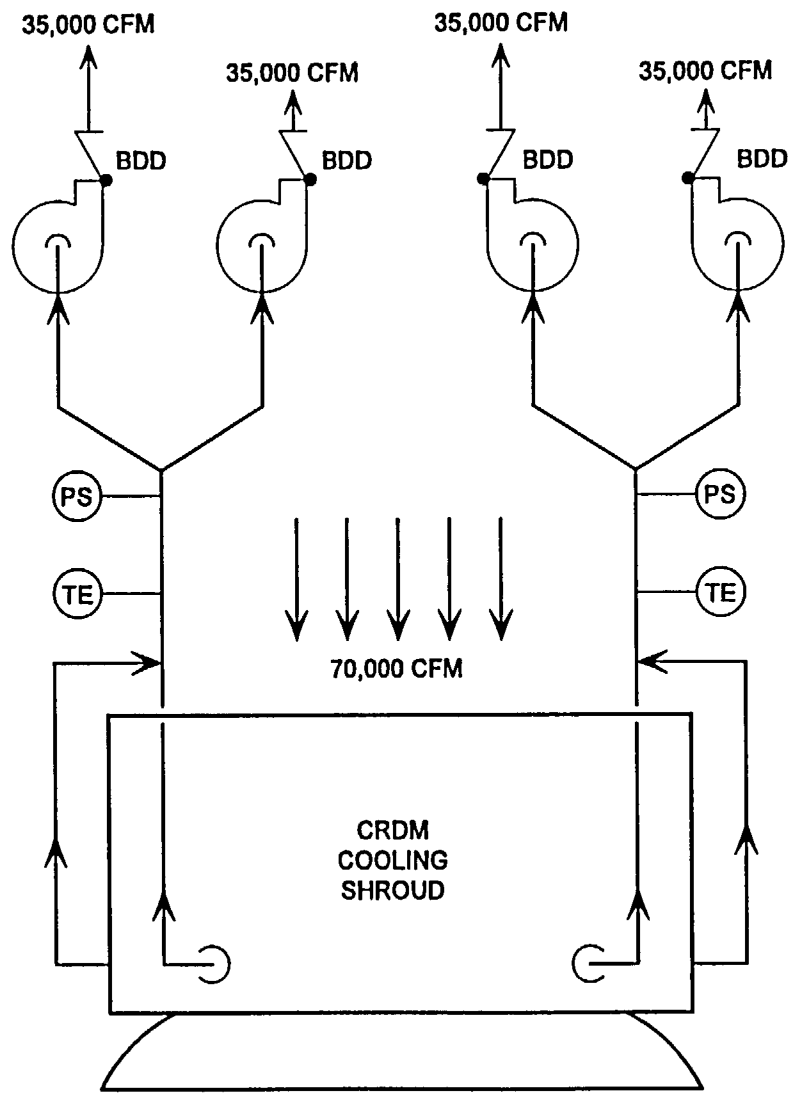

BDD - BACKDRAFT DAMPER
PS - PRESSURE SWITCH
TE - TEMPERATURE ELEMENT

Figure 5.4-3 CRDM Cooling System

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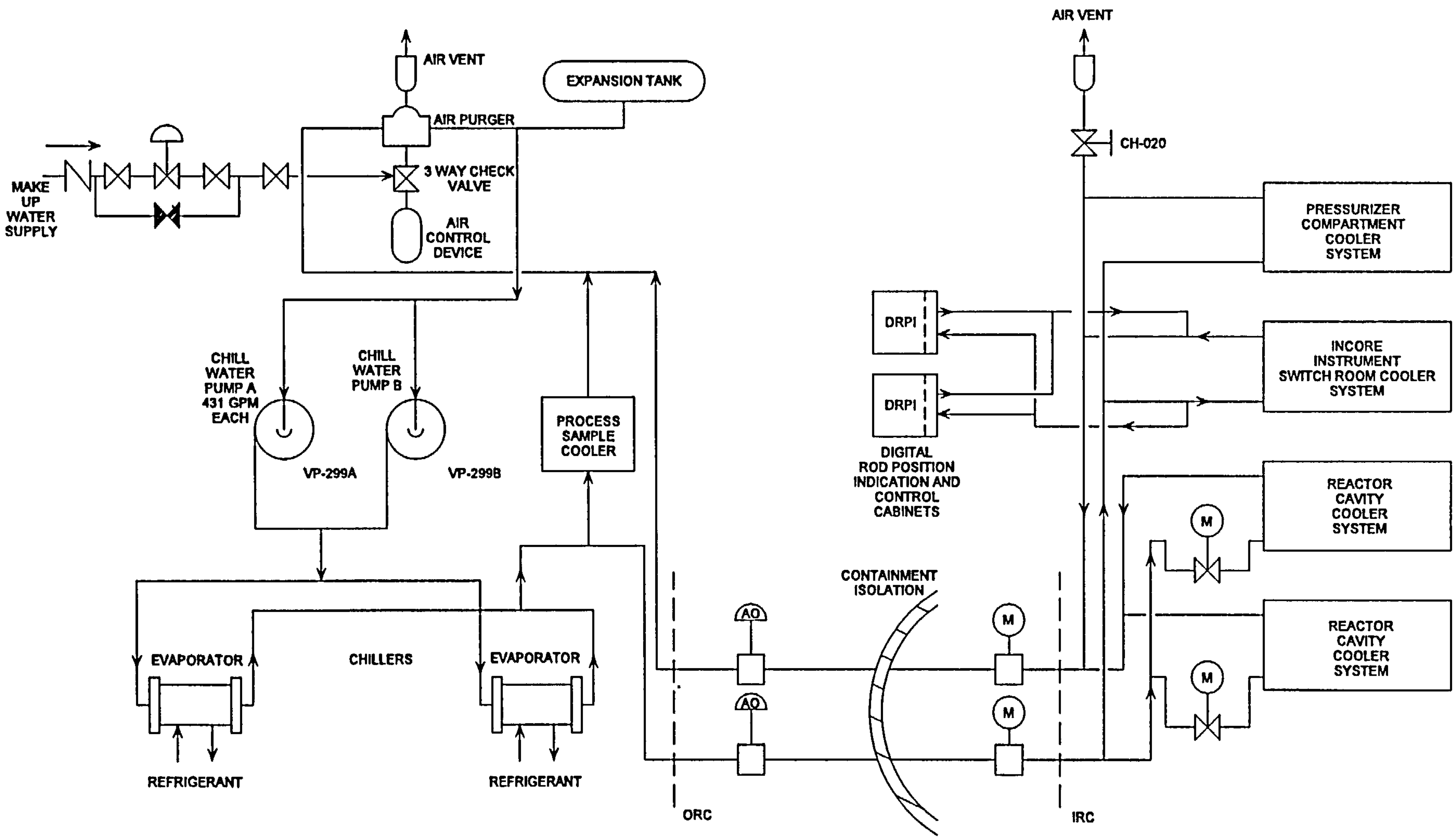
Figure 5.4-4 Chill Water System



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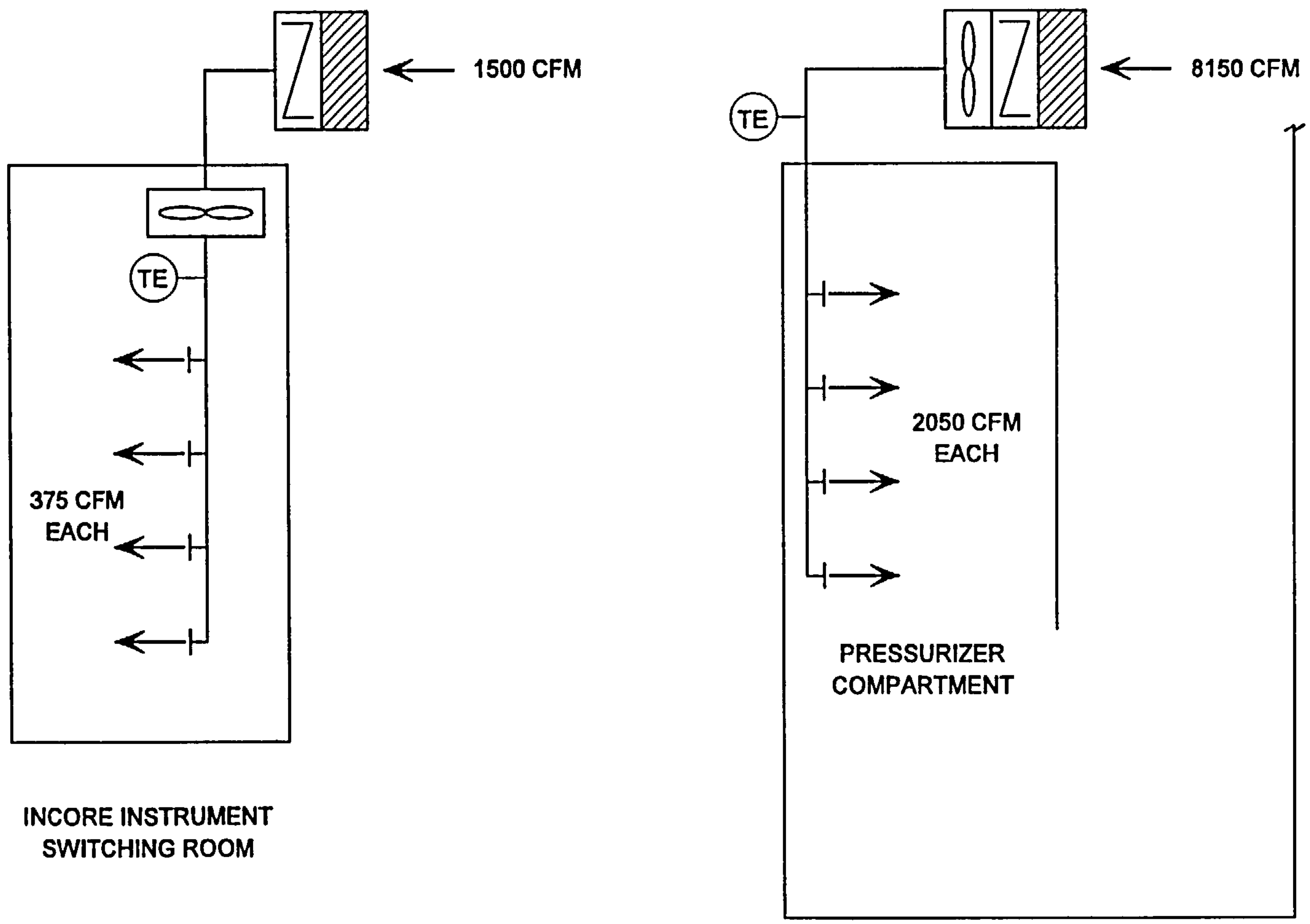

Figure 5.4-5 Pressurizer Compartment & Incore Instrument Switching Room Cooling

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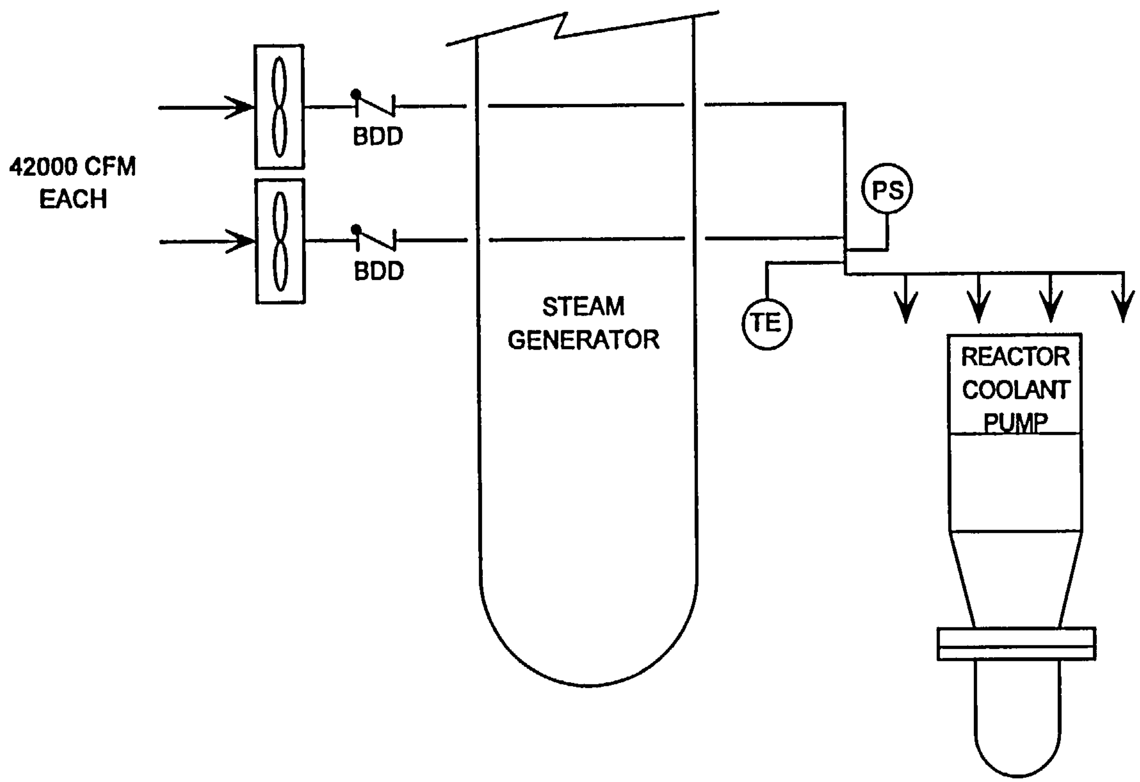

Figure 5.4-6 Reactor Coolant Pump Cooling System

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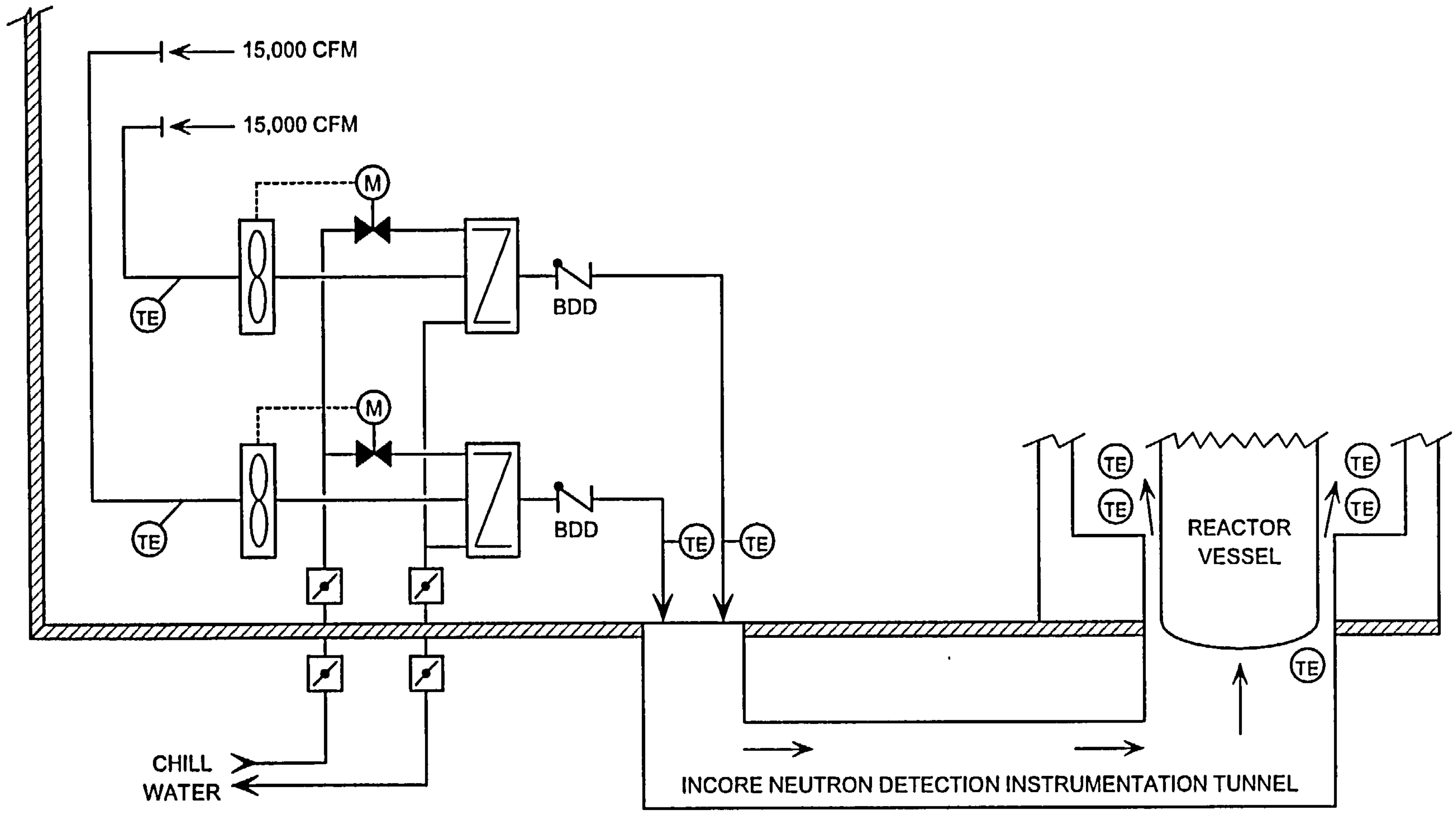

Figure 5.4-7 Reactor Cavity Cooling System

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CONTAINMENT AIR COOLER SYSTEM

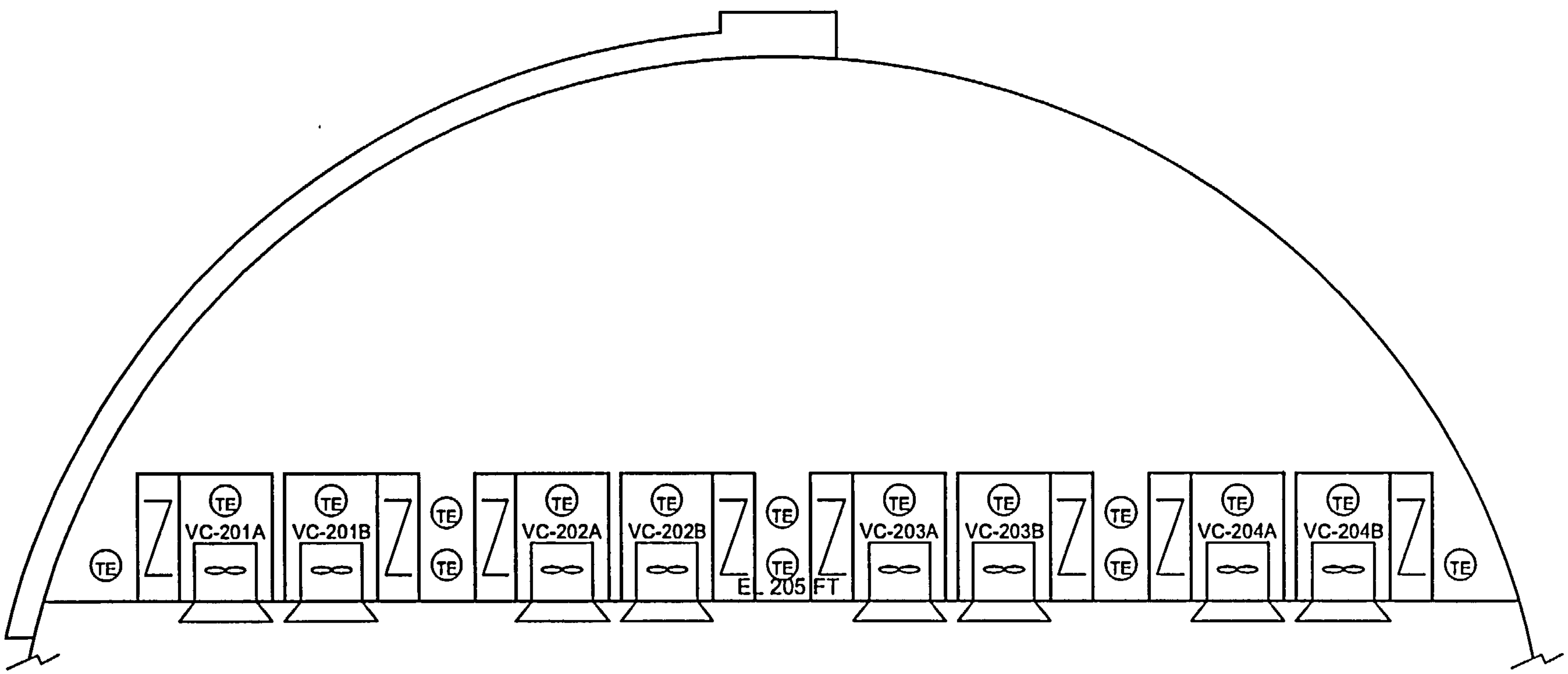

Figure 5.4-8 Containment Air Cooler System

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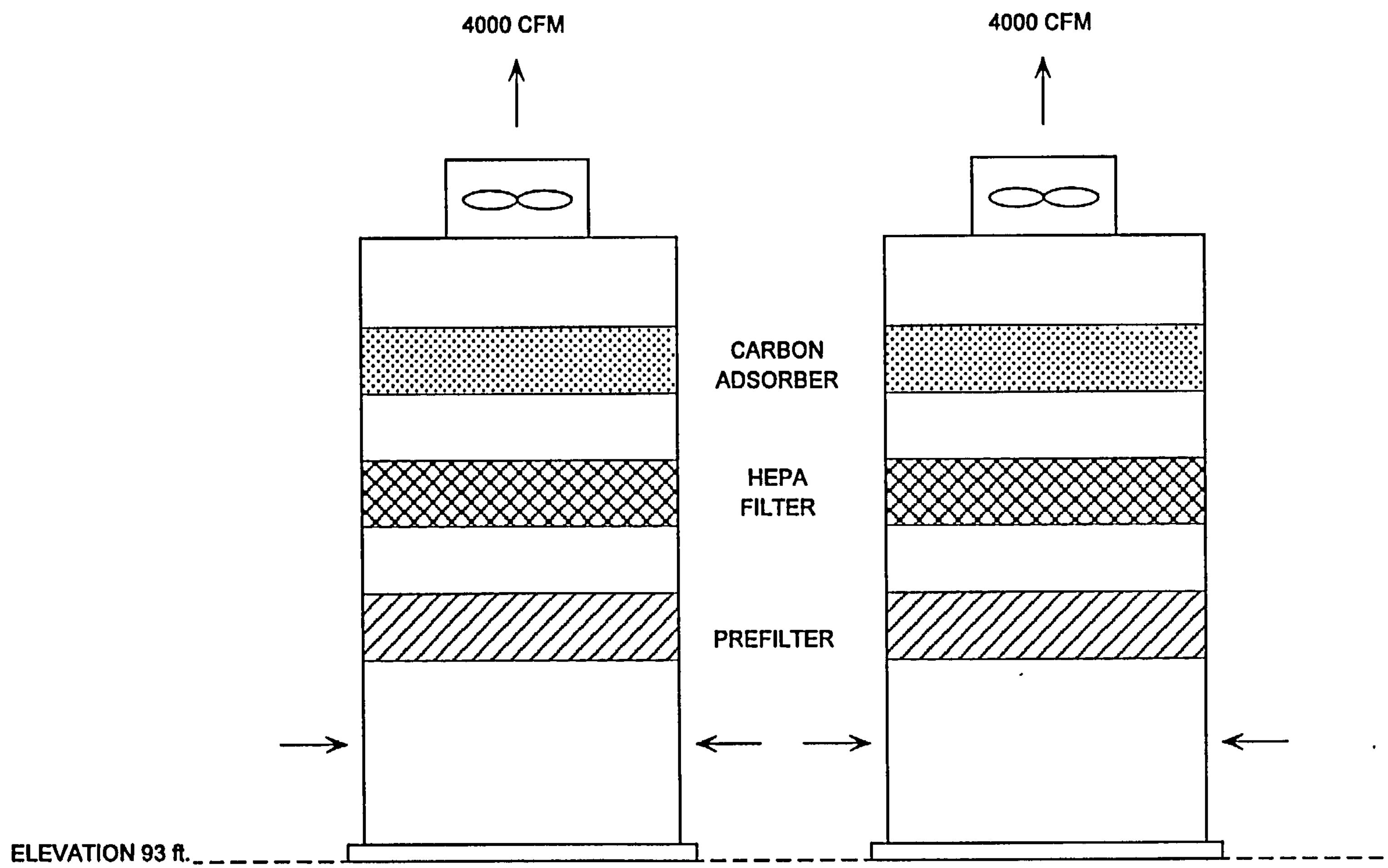

Figure 5.4-9 Clean Up Recirculation Units

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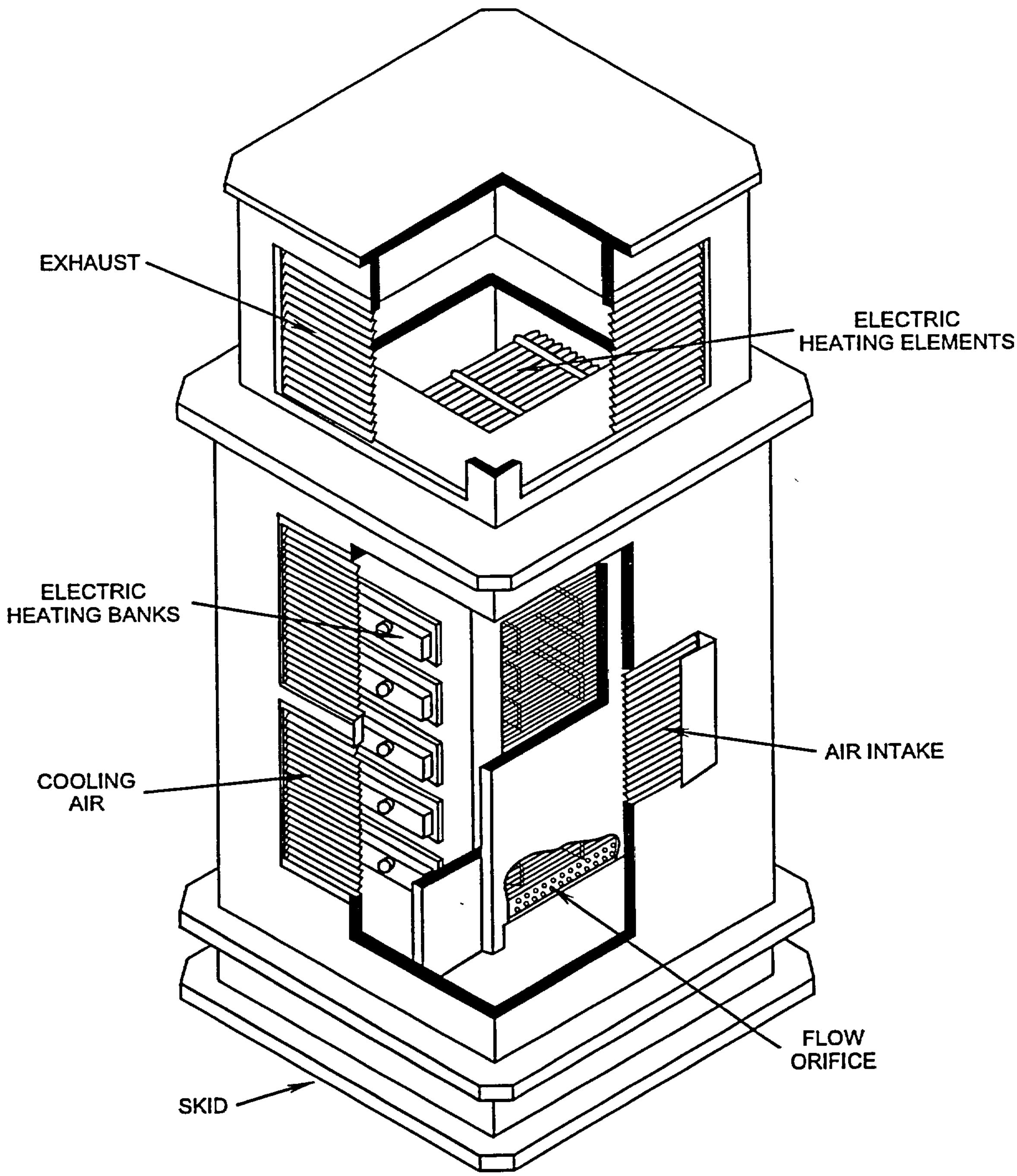

FIGURE 5.4-10 Electric Hydrogen Recombiner

HYDROGEN MIXING SYSTEM

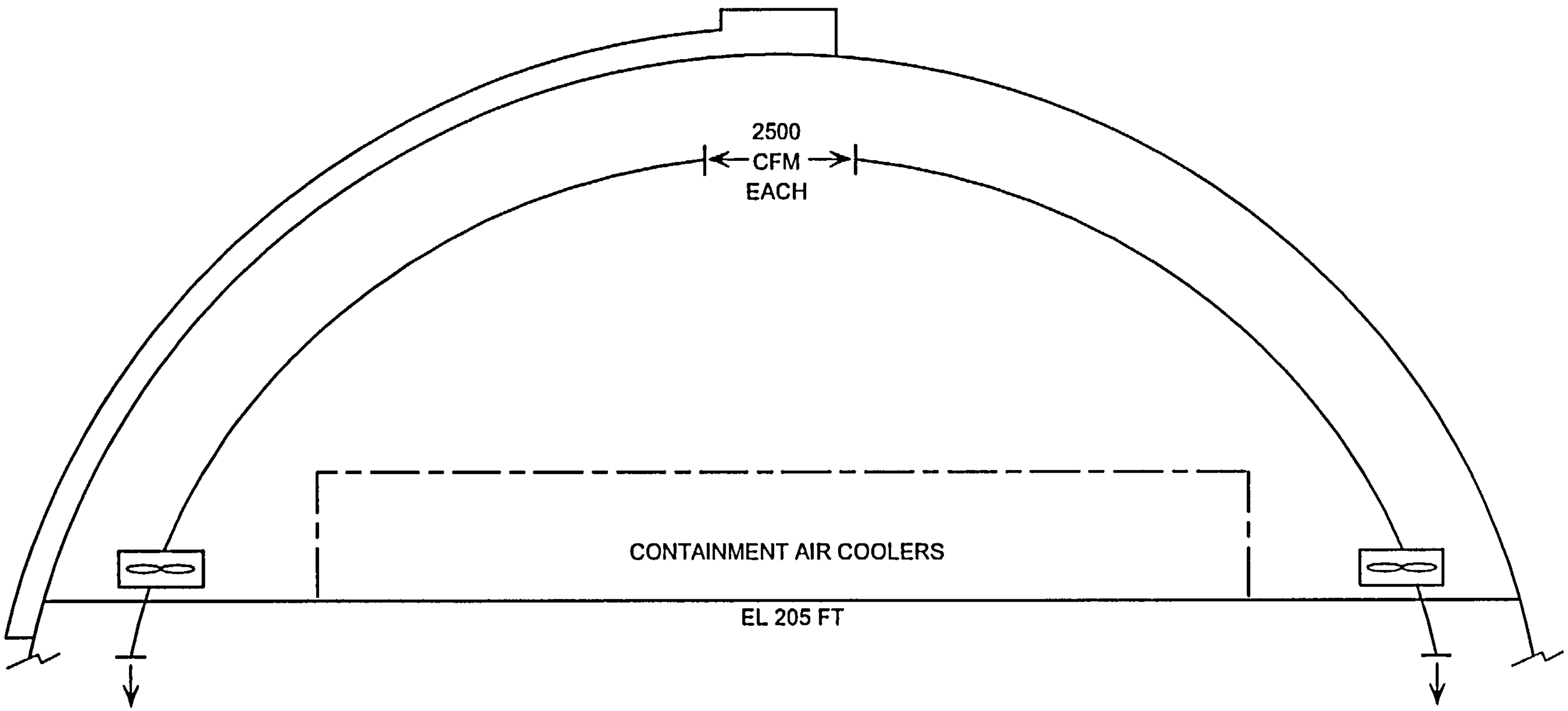

Figure 5.4-11 Hydrogen Mixing System

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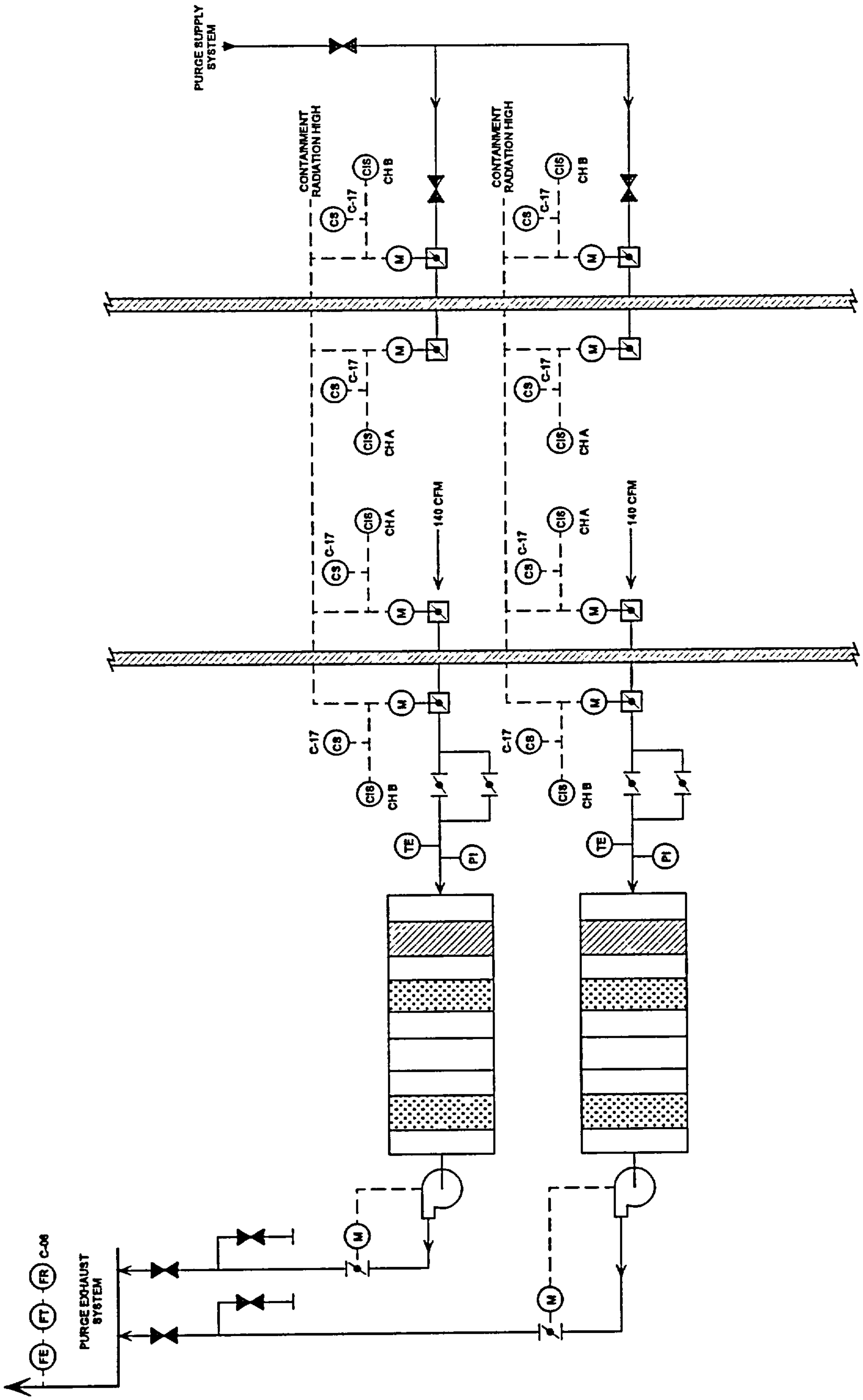

Figure 5.4-12 Hydrogen Vent System

CONTAINMENT SPRAY SYSTEM

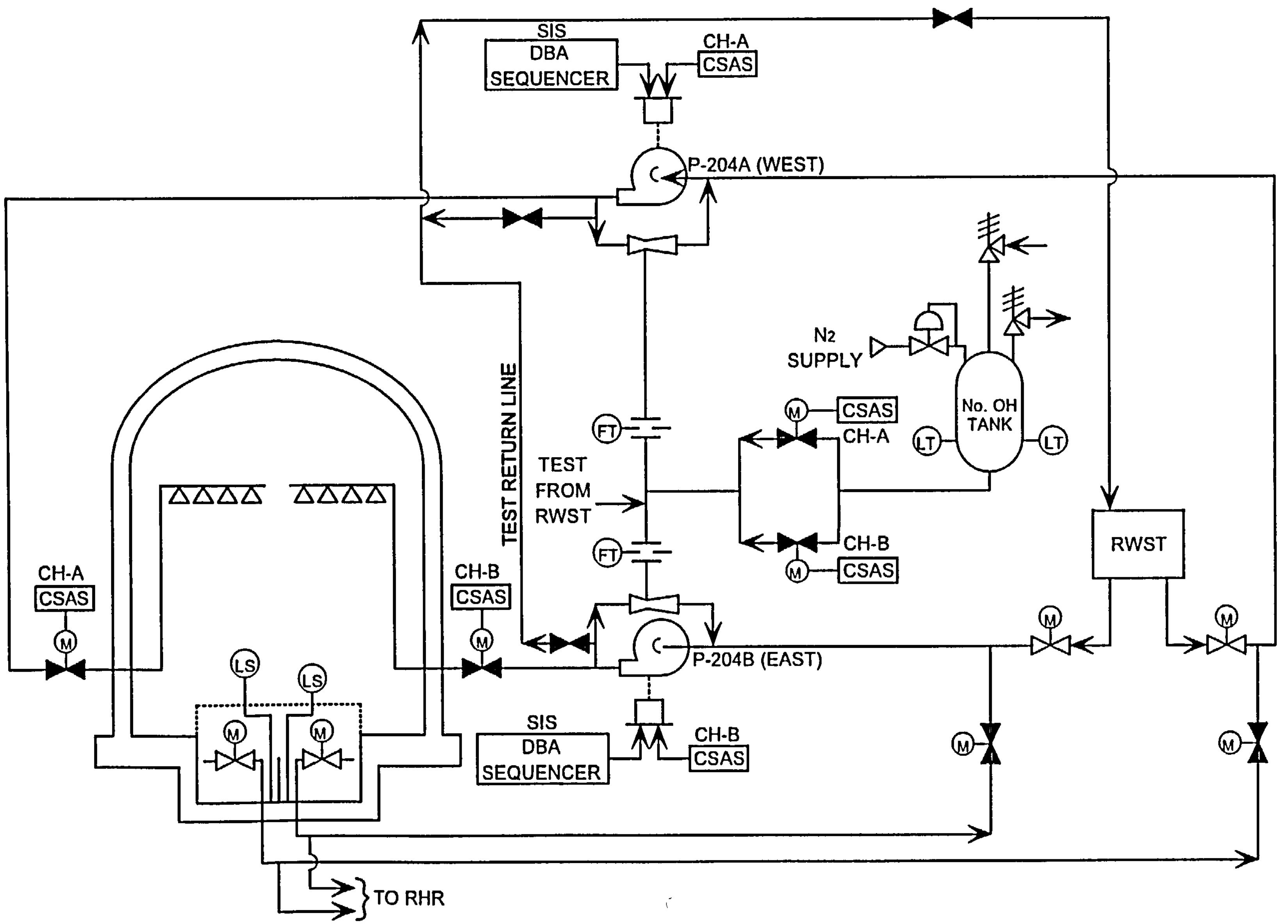

Figure 5.4-13 Containment Spray System

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Section 5.6

Containment Penetration and Isolation Systems

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5.6 CONTAINMENT PENETRATION AND ISOLATION SYSTEMS

Learning Objectives:

1. State the purposes of the system.

2. Define the following:

    a. Containment Integrity
    b. Containment Isolation Phase A
    c. Containment Isolation Phase B

3. List the signals that initiate phase A and phase B isolation.

5.6.1 Introduction

The purposes of the containment penetration and isolation systems are as follows:

1. Isolates non-essential containment penetrations during accident conditions.
2. Provide leak-tight mechanical and electrical containment penetrations.

The primary means of minimizing the release of radioactive gases and particulates after an accident is the isolation of the reactor containment from the environment. Containment penetration and isolation systems are designed to limit any releases to within those allowed by Technical Specifications. Common leakage limits are between one tenth and four tenths of one percent of the containment volume per day.

The Code of Federal Regulations, Title 10, Part 50, Appendix J (Reactor Containment Leakage Testing for Water Cooled Power Reactors) requires that the actual leak rate be verified experimentally prior to operation and periodically thereafter.

All containment penetrations (both electrical and piping) are double barrier assemblies consisting of a closed sleeve, in most cases, or a double gasketed closure for special penetrations, such as the fuel transfer tube. The void space between the double barriers is continuously pressurized, by the penetration pressurization system, to a pressure in excess of the containment design pressure. Leakage from this system is monitored to indicate penetration leakage.

The containment isolation system provides the means of isolating fluid systems that pass through containment penetrations so as to confine any radioactivity that may be released into the containment following a loss of coolant accident. The containment isolation systems are required to function following an accident, to isolate non-safety-related fluid systems penetrating the containment. A particular system does not exist for containment isolation, but isolation design is achieved by applying common criteria to penetrations in many different fluid systems and by using ESF signals to actuate appropriate valves.

The main function of the containment isolation system is to provide containment integrity when needed. Containment integrity is defined to exist when:

1. The non-automatic containment isolation valves and blind flanges are closed as required.
2. The containment equipment hatch is properly closed.
3. At least one door in each containment personnel air lock is properly closed.
4. All automatic containment isolation valves are operable or are deactivated in the closed position or at least one valve in each line having an inoperable valve is closed and,
5. All requirements of the Technical Specification with regard to containment leakage and test frequency are satisfied.

The integrity of the containment is required in operating modes 1, 2, 3 and 4, and in mode 6 when irradiated fuel is being moved inside containment. All containment penetrations and isolation systems are designed to be Seismic Category 1.

5.6.2 Containment Penetration Types

5.6.2.1 Electrical Penetrations

Canister type electrical penetration assemblies are used to extend conductors through the containment penetration nozzles. Figure 5.6-1 shows a typical electrical penetration assembly in place within a containment penetration nozzle. Hermetic (airtight) seals between each conductor and the metallic canister end header plates are obtained by the use of high strength, high temperature glass and ceramic materials.

There are five types of electrical penetration assemblies, classified by service and applications as follows:

1. Type 1-Medium voltage power service
2. Type 2-Low voltage power service
3. Type 3-Control service
4. Type 4-Shielded instrumentation service
5. Type 5-Triaxial instrumentation cable service

The penetrations associated with engineered safety features within the containment are fully rated for both normal and post loss of coolant accident (LOCA) operation.

5.6.2.2 Piping Penetrations

Double barrier piping penetrations are provided for all piping which passes through the containment as shown in Figure 5.6-2. The pipe is contained in a sleeve which is welded to the liner. Closure heads are welded to the sleeve and to the pipe, both inside and outside the containment, to form the double barriers. The annulus between the pipe and sleeve is continuously pressurized. Several pipes may pass through the same sleeve to minimize the number of containment penetrations required. In these cases, each pipe is welded to both closure heads.

The penetrations for main steam, feedwater, and steam generator blowdown are anchored outside the containment in such a way as to ensure containment integrity should any one of these lines rupture. These penetrations are provided with a bellows expansion joint on both the inside and the outside of the containment, which allows for differential movement between the containment wall and the anchor. The inside joint will also take up the differential movement of the hot pipe relative to the liner. The expansion joints have been designed so that, in the event of a pipe rupture within the sleeve, the inside joint will subsequently rupture and the outside joint will remain intact, thereby maintaining containment integrity.

Hot penetrations are normally provided with some type of cooling to minimize drying of the containment concrete around the penetrations. Other hot penetrations, which are not anchored outside the containment for pipe rupture, are where required, provided with a single expansion joint, either inside or outside the containment, to allow for thermal movement of the pipe relative to the anchor.

5.6.2.3 Equipment and Personnel Access Hatches

The equipment hatch (as shown on Figure 5.6-3) is fabricated from welded steel, and furnished with a double-gasketed flange and a bolted dished door. Equipment up to and including the size of the reactor vessel upper head O-ring seal can be transferred into and out of containment via this hatch. The hatch barrel is welded to the containment liner.

Two personnel air locks are provided for the containment, one of which is for normal access and penetrates the dished shaped door of the equipment hatch. The other is an emergency escape hatch located on the side of containment opposite the equipment hatch at grade level. Each personnel air lock is a double door, hydraulically-latched, welded steel assembly as shown in Figure 5.6-3. A quick-acting type, equalizing valve connects the personnel lock with the interior of the containment structure. Its purpose is equalizing pressure in the escape lock and the containment when entering or leaving the containment structure.

The two doors in each personnel lock are interlocked to prevent both doors being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened. Remote indicating lights and annunciators in the control room indicate the status of these doors. An emergency lighting and communication system, operated from an external emergency power supply, is provided in the personnel lock interior.

5.6.2.4 Special Penetrations

A fuel transfer penetration is provided (see Figure 5.6-4) for fuel movements between the refueling cavity in the reactor containment and the spent fuel pit. The penetration consists of a 20" pipe located inside a 24" sleeve. The inner pipe acts as the transfer tube and is fitted with a double gasketed blind flange in the refueling cavity inside containment. A seal plate is welded to the containment liner and also to the inside tube. This seal plate and the blind flange act as the containment boundary.

Bellows expansion joints have been installed in the sleeve. These joints, which are welded to the sleeve and connected to the tube by welding to end plates, provide for normal and seismic differential building movements. The sleeve and expansion joints also serve to cover most of the welds on the tube inside the containment. The annulus between the sleeve and tube is continuously pressurized to demonstrate containment integrity.

Lines originating from the containment recirculation sumps penetrate the containment through the slab below the containment walls. In this case the pipe is contained in a sleeve which is buried in the slab. Both sleeve and pipe extend into the sump, where the sleeve is seal welded to both the pipe and the sump liner.

The sleeve and pipe also extend outside the containment to an isolation valve which is completely enclosed in a small tank. The sleeve terminates at the entrance to the tank where it is welded to the tank. At the outlet of the tank, the pipe is seal welded to the tank through an expansion joint which takes up the thermal expansion of the pipe.

The annulus between the pipe and sleeve, and the volume within the tank are continuously pressurized through a connection on the tank. The sleeve and tank serve as an extension of the containment boundary to maintain containment integrity in the event of a pipe break outside containment.

5.6.3 Containment Isolation Systems

5.6.3.1 Design Basis

The design basis for the containment isolation systems include provisions for the following:

1. A double barrier at the containment penetration in those fluid systems that are not

required to function following a LOCA or a secondary rupture inside the containment.
2. Automatic, fast closure of those valves required to close for containment integrity following an accident to minimize the release of any radioactive material.
3. A means of leak testing barriers in fluid systems that serve as containment isolation.
4. The capability to periodically test the operability of the containment isolation valves.

5.6.3.2 Initiation Signals

Containment isolation is accomplished by two separate signals designated as phase A and phase B. The phase A isolation is initiated by any Engineered Safety Feature (ESF) actuation signal (safety injection signal) or manually from the control room. After the containment phase A is actuated, it isolates non-essential piping penetrations into the containment such as; the containment sump pump discharge, containment ventilation connections to atmosphere, etc. The most important penetrations not isolated on phase A are component cooling water to the reactor coolant pumps.

The phase B isolation is initiated by a hi-hi containment pressure signal or manually from the main control board, and isolates the remainder of the non-safety penetrations including component cooling to the reactor coolant pumps. This same hi-hi pressure signal also closes the main steam line isolation valves and initiates containment spray.

5.6.3.3 Isolation Classes

The criteria defining the number and location of containment isolation valves in each fluid system depends on the function of the system and whether it is open or closed to the containment atmosphere, or reactor coolant system. Four isolation classes of penetrations are defined as follows:

1. Isolation class I - lines which are open to the atmosphere outside the containment and are connected to the reactor coolant system or are open to the containment atmosphere. Each isolation class I system has a minimum of two isolation valves in series. Where system design permits, one valve is located inside and one valve is located outside containment.
2. Isolation class II - lines which are connected to a closed system outside the containment, and are connected to the reactor coolant system or are open to the containment atmosphere. Also included in isolation class II are fluid lines which are open to the atmosphere outside the containment and are separated from the reactor coolant system and the containment atmosphere by a closed system inside the containment. Each isolation class II system has a minimum of one isolation valve.
3. Isolation class III - lines which are connected to a closed system both inside and outside the containment. Isolation class III systems have as a minimum one isolation valve.
4. Isolation class IV - lines which must remain in service subsequent to a LOCA or a secondary pipe break inside the containment, such as the emergency core cooling systems (ECCS). Isolation valves on these lines are not automatically closed by the containment isolation signal. Each isolation class IV system has a minimum of one isolation valve (remote-manual operation).

The criteria for containment penetrations ensure that all fluid lines penetrating the containment have at least one isolation valve near the point of penetration. Most fluid lines that communicate directly with the containment atmosphere have, as a minimum, two isolation valves in series. These lines and isolation valves are designed to Seismic Category I specifications and are missile protected.

5.6.4 Systems Features and Interrelationships

A containment isolation signal initiates closing of automatic isolation valves in those lines which must be isolated immediately following an accident. There is no order of sequence of timing for containment isolation valve closure. However, on loss of offsite ac power, the diesel will have to be started prior to the closure of the motor operated valves. Air operated valves used for containment isolation are designed to fail closed on loss of air.

Check valves are sometimes used as containment isolation valves where the differential pressure, under accident conditions, will close the valves to maintain containment integrity. Lines which, for safety reasons, must remain in service subsequent to an accident are provided with at least one isolation valve.

Each automatic isolation valve required to operate subsequent to an accident is additionally provided with a manual control switch for operation. The position of these automatic isolation valves are indicated by status lights in the main control room.

Containment isolation valves that are located inside the containment are designed to function under the radiation, pressure, and high temperature conditions existing during both normal operation and accident conditions.

Containment isolation valves are designed to Seismic Category I requirements. The valves are capable of operation during and after seismic loadings. Valves with operators or similar features of extended proportions are designed to withstand an inertial Safe Shutdown Earthquake (SSE) load in addition to normal operating loads. Electrical switches or other actuating mechanisms are designed to withstand the inertial load as a result of an SSE without changing position and causing change of position of the valve disc.

5.6.5 Summary

The containment isolation systems provide the means of isolating fluid systems that pass through containment penetrations so as to confine, any radioactivity that may be released into the containment following an accident. The containment isolation systems are required to function following a design basis event to isolate nonsafety related fluid systems that penetrate the containment.

Containment isolation is initiated in two levels: phase A and phase B. Containment isolation phase A always exists if containment isolation phase B exists (unless phase B is manually initiated). A containment isolation signal initiates closing of automatic isolation valves in those lines which are not required to respond to an accident. The position of these automatic isolation valves is indicated by status lights in the main control room.

Containment isolation valves are designed to Seismic Category I requirements. These valves are capable of operation during and after seismic loadings.

Containment penetrations are designed to ensure that the leakage from the containment, at a maximum calculated containment pressure and temperature, does not exceed the limits of the plants Technical Specifications.

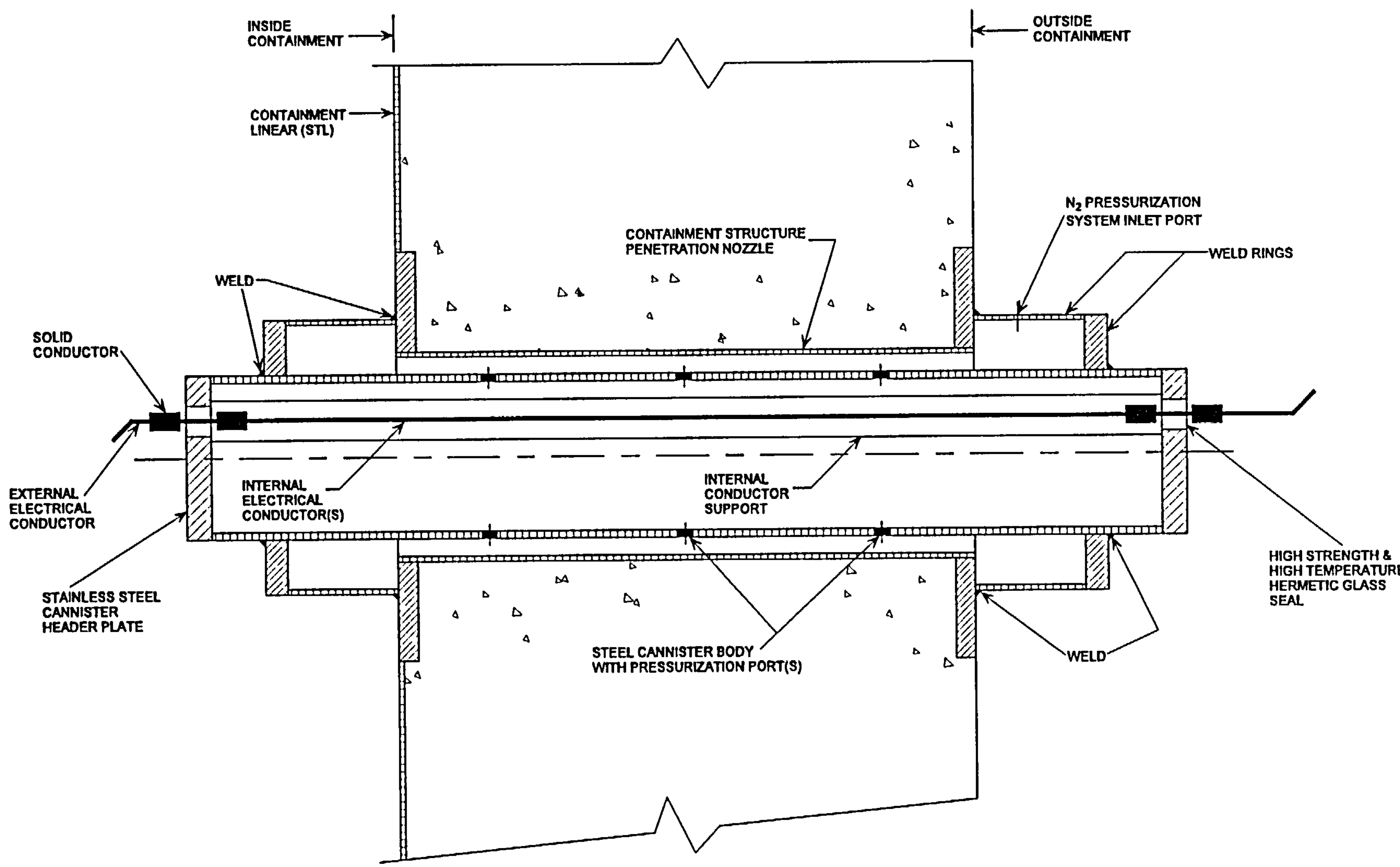

Figure 5.6-1 Electrical Penetration

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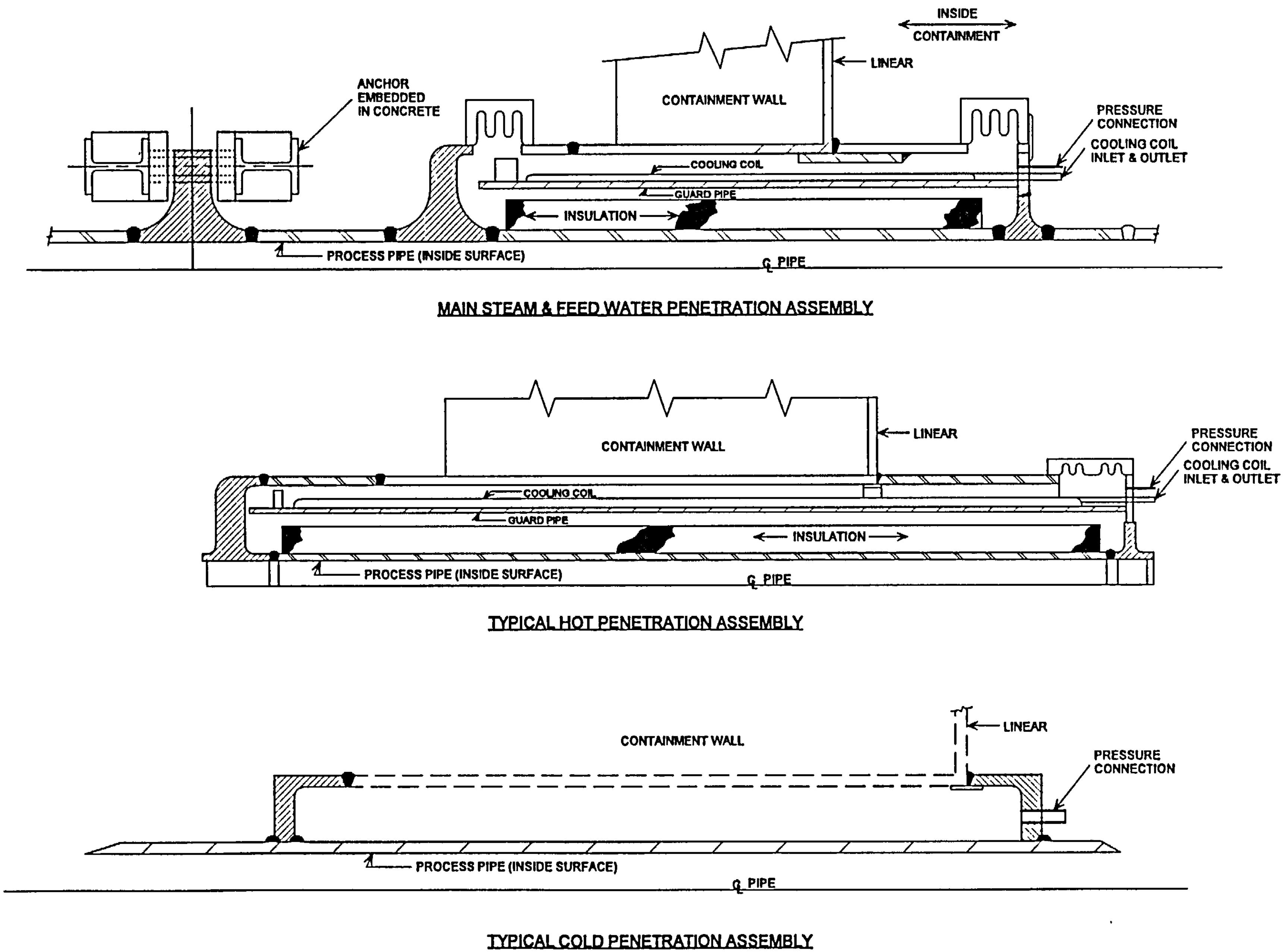
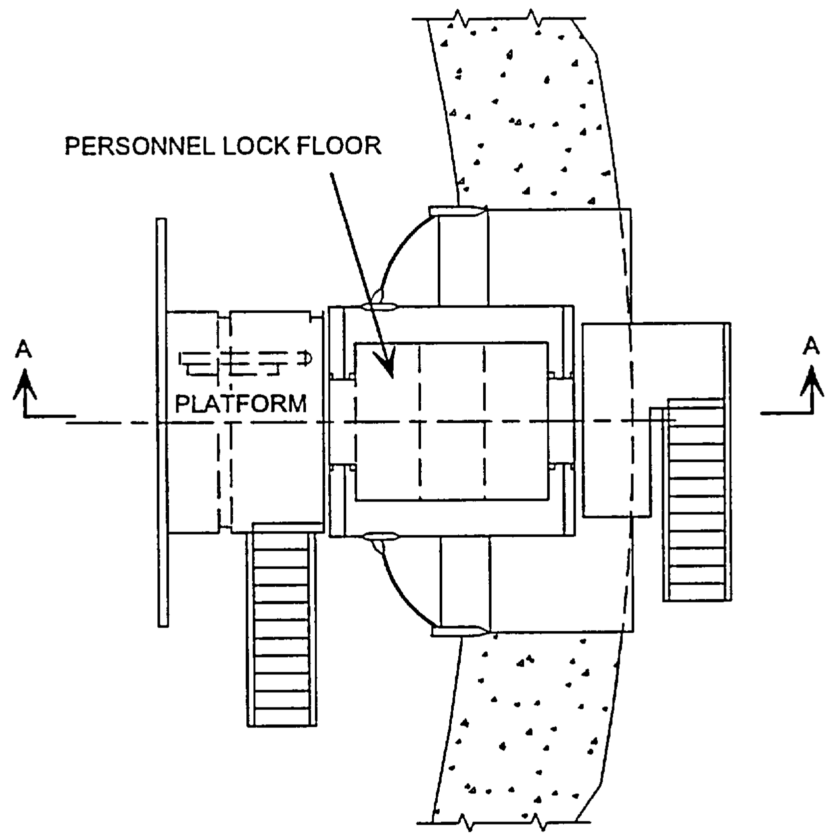


Figure 5.6-2 Piping Penetration

0199-3



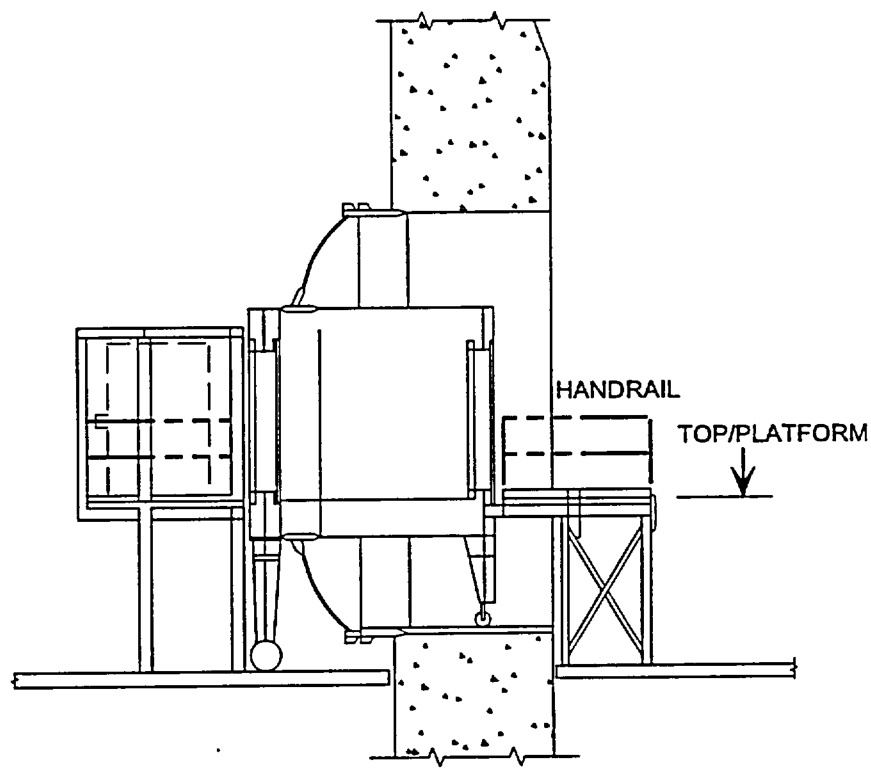


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Figure 5.6-3 Combined Equipment and Personnel Lock

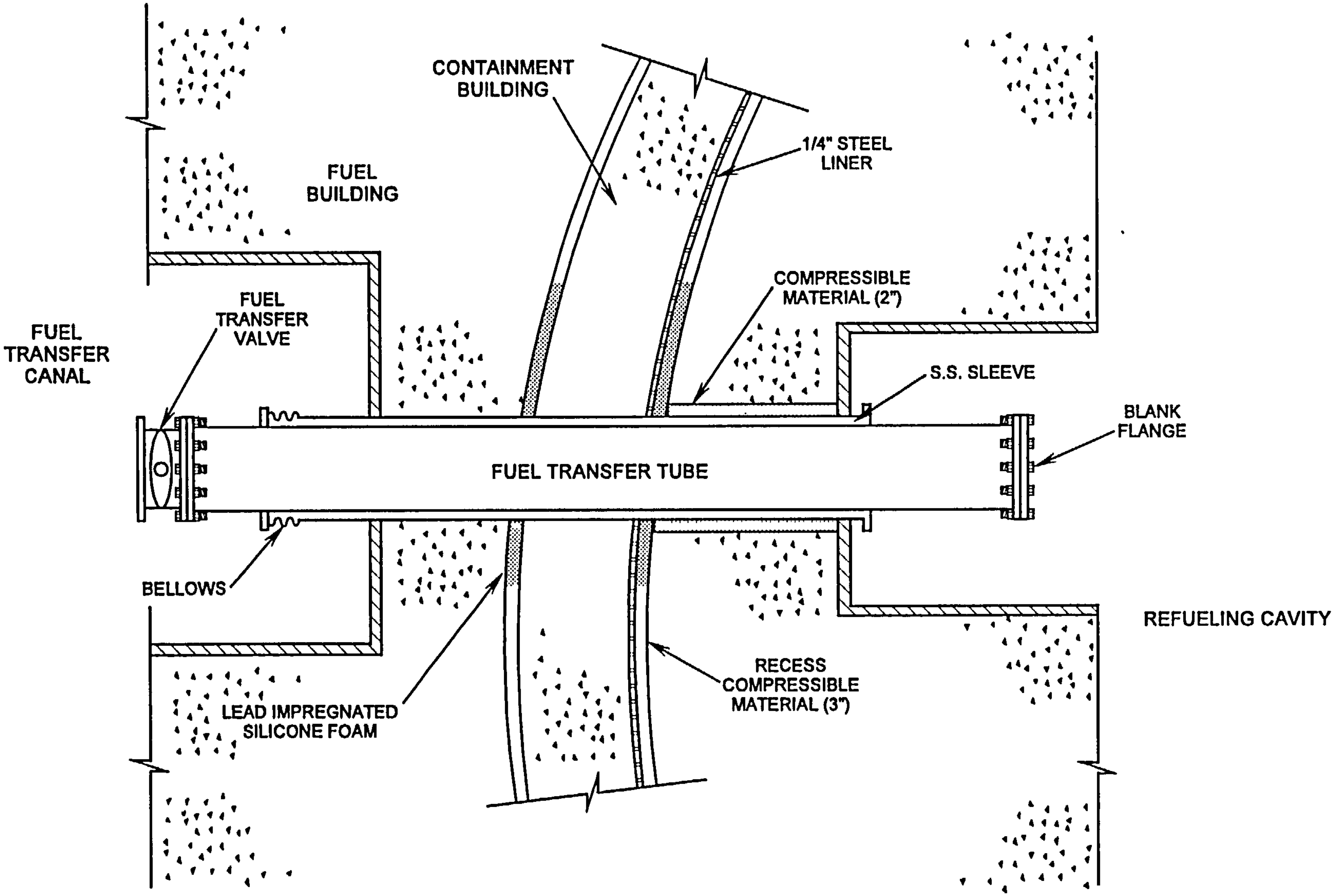

Figure 5.6-4 Fuel Transfer Tube

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Section 5.7

Auxiliary Feedwater System

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5.7 AUXILIARY FEEDWATER SYSTEM

Learning Objectives:

1. State the purposes of the Auxiliary Feedwater (AFW) system.

2. List all suction sources for the AFW pumps and under what conditions each is used.

3. List the five plant conditions that will result in an automatic start of the AFW system.

4. Explain how decay heat is removed following a plant trip and loss of offsite power.

5. Describe the design feature which ensures a minimum volume of water in the condensate storage tank is reserved for the AFW system.

5.7.1 Introduction

The purposes of the AFW system are as follows:

1. Provide feedwater to the steam generators to maintain a heat sink for the following conditions.

   a. Loss of main feedwater (MFW).
   b. Unit trip and loss of offsite power.
   c. Small break loss of coolant accident.

2. Provide a source of feedwater to the steam generators during plant startup and shutdown.

The AFW system supplies, in the event of a loss of the main feedwater, sufficient feedwater to the steam generators to remove primary system stored heat and residual core energy (decay heat). AFW must also be available under accident conditions, such as a small break loss of coolant accident, so the plant can be brought to a safe shutdown condition.

The AFW system is designed to automatically start and supply sufficient feedwater to prevent the relief of primary coolant through the pressurizer safety valves. The AFW system has an adequate suction source and flow capacity to maintain the reactor at hot standby for a period of time and then cool the Reactor Coolant System (RCS) to a temperature at which the Residual Heat Removal System (RHR) may be placed in operation.

5.7.2 System Description

The AFW system as shown in Figure 5.7-1, has two electric-motor-driven pumps and one turbine-driven pump. Each of the electric driven pumps supplies two different steam generators, the turbine driven pump supplies all four steam generators. All three pumps automatically deliver rated flow within 1 minute upon receipt of an automatic start signal.

The preferred source of water for all auxiliary feedwater pumps is the Condensate Storage Tank (CST) which is required by the plants Technical Specifications to contain a minimum amount of water to be used by the AFW system. An additional unlimited backup water supply, Essential Service Water (ESW), is supplied to the AFW system. A separate train of ESW feeds each electric driven pump, (Train A of ESW feeds the A AFW pump) while the turbine driven pump can receive backup water from either train of ESW.

To protect the AFW pumps from a loss of suction, the ESW supply valves are automatically (or remote-manually) opened if the suction pressure is low on two-out-of-three pressure detectors, and the AFW pump is running.

Since the ESW system supplies poor quality water, it is not used except in emergencies when the normal condensate supply is unavailable.

The AFW system is designed to deliver 40 to 120°F water for pressures ranging from the RHR system operating pressure (equivalent to approximately 110 psig in the steam generators) to the highest set point of the steam generator safety valves (1234 psig).

The AFW system piping is designed for pressures up to approximately 1650 psig where necessary. Separate Engineered Safety Features (ESF) quality electrical power subsystems and control air subsystems serve each AFW pump and its associated valves.

In addition to using high quality components and materials, the AFW system provides complete redundancy in pump capacity and water supply for all cases for which the system is required. Under all credible accident conditions each steam generator, not affected by the accident, will be supplied with its required feedwater. Only two steam generators are required to be operable for any credible accident condition.

Redundant electrical power and air supplies assure reliable system initiation and operation. The electric-motor-driven pumps are powered from vital ac distribution sources, while the turbine-driven pump takes steam from either of two main steam lines, up stream of the main steam isolation valves (MSIVs).

5.7.3 Component Descriptions

5.7.3.1 Motor Driven Auxiliary Feedwater Pumps

The motor driven pumps are multi-stage horizontal centrifugal pumps, each of which supplies 440 gpm at a discharge pressure of about 1300 psig. The motor driven pump design data is shown in Table 5.7-1.

Power to the motor driven AFW pumps is supplied from the 4.16 kVac Class IE vital distribution boards. Local switches permit local operation of the pumps. The switches in the control room have three positions: Run, Stop, and Pull to Lock. The Pull to Lock feature prevents the pumps from starting, even if an automatic start signal were present.

Table 5.7-1
Auxiliary Feedwater System
Design Data

Total number of pumps per unit	3
Motor driven	2
Turbine driven	1
Design flow rate, gpm	
Motor driven, each	440
Turbine driven	880
System design pressure, psig	1650
Design feedwater temperature, ° F	40-120
Design discharge head, psig	
Motor driven	1300
Turbine driven	1200

The following five conditions will automatically start both motor driven feed pumps:

1. Low-low steam generator level in any single steam generator.
2. Loss of one main feed pump (MFP) if power is greater than 80 percent.
3. Loss of both MFPs at any power level.
4. ESF actuation signal.
5. Loss of power to the Class IE power distribution system.

5.7.3.2 Turbine Driven Auxiliary Feedwater Pump

The turbine driven pump is a multi-stage, horizontal, centrifugal pump. It is capable of delivering 880 gpm at a discharge pressure of approximately 1200 psig. This pumps design data is shown in Table 5.7-1

The steam driven pump is driven by a horizontal, non-condensing turbine. The turbine is rated at 1100 HP and is designed to operate with a supply steam pressure varying from 1275 psig down to 100 psig. The steam to drive the turbine is supplied from the main steam system, with the piping penetrations located upstream of the MSIVs. The steam supply comes from steam generators number 2 and 3. Each supply line has an air operated main valve and an associated bypass valve. The bypass valves are used for warming the turbine and are operated from the main control board.

During the AFW pump turbine operation, steam is supplied to the unit through a normally shut flow control valve FC-HV-312. The control of the turbine speed is accomplished by adjusting the governor valve (HV-313) from the main control board.

To remove moisture in the main steam supply steam traps have been provided where necessary. Drains have also been provided on the turbine casing, steam chest and exhaust piping. The exhaust from the AFW pump turbine is directed to the atmosphere.

The five automatic start signals for the turbine driven auxiliary feed pump are as follows:

1. Low-low steam generator level in 2 of 4 generators.
2. Loss of one MFP if power is greater than 80 percent.
3. Loss of both MFPs at any power level.
4. ESF actuation signal.
5. Loss of power to either Class IE vital distribution system.

5.7.3.3 Level Control Valves

The AFW system contains eight (8) level control valves. Four (4) valves are located on the auxiliary feed lines feeding the steam generators from the motor-driven AFW pumps. The remaining four (4) level control valves are on the auxiliary feed lines coming from the turbine driven auxiliary feed pump. Each valve has a toggle switch which allows the operator to manually open or close the valve. Normally the valves are fully closed and will automatically control steam generator level to a pre-selected setpoint upon an automatic actuation signal.

At some plants, these level control valves are provided with loop break protection circuitry. This is accomplished by monitoring the pressure between the level control valve and its associated steam generator. If the pressure in this section of piping drops to a pre-selected setpoint (normally 100 psig), a signal is sent to close the level control valve. This protective feature is supplied so that in the event of a steam or feedwater break the auxiliary feedwater system will not continue to feed the faulted steam generator.

5.7.3.4 Water Supplies

The water supplied to the AFW system is redundant with the normal suction source provided by gravity feed from the condensate storage tank which is sized to meet the normal operating and accident needs. Each AFW pump takes its suction on a common header through a check valve and a normally open isolation valve. The suction pressure for the AFW pumps is normally indicated in the control room.

Availability of water from the CST is guaranteed by a stand pipe which is used by the supply lines to the main condenser. Other means may be used at other facilities to ensure a minimum level in the CST. Other methods include a level control valve which closes when the quantity of water in the condensate storage tank drops to a preset value or the supply line to the main condenser is tapped in the side of the condensate storage tank at a height that ensures a minimum level for the AFW system.

In the event that the CST is not seismically qualified, an emergency water supply must be provided as a backup. This backup supply must be seismically rated and must have some method for automatic initiation if needed.

There are 3 pressure switches located on the suction line for each AFW pump which are used to switch the suction from the CST to the safety related emergency service water (ESW) system (which is seismically qualified). To prevent an inadvertent injection of ESW into the steam generators automatic opening of the ESW supply valves is initiated by a 2 out of 3 logic on low pressure coincident with an AFW pump running.

On the discharge of each motor driven pump, there is a blind flange connection for the installation of a spool piece to tie the high pressure fire protection system into the AFW system. The pressure in the steam generator must be below 120 psig for use of the high pressure fire protection system.

5.7.4 System Features And Interrelationships

The safety portions of the AFW system are designed for seismic events and meet the single failure criteria requirements, including the consideration that the rupture of a feedwater line could be the initiating event. This system will provide the required feed flow to two or more steam generators regardless of any single active or passive failure.

The valves associated with the turbine-driven pump are served by both electric and control air subsystems, with appropriate measures precluding any interaction between the two subsystems. The turbine-driven pump receives control power from a third dc electric channel that is distinct from the channels serving the electric pumps.

In the event of a loss of site power, 440 gpm of AFW delivered to two steam generators will prevent relief of reactor coolant via the pressurizer safety valves, and the water levels in the steam generators will remain above the required minimum for tube coverage. If the AFW system did not respond for ten minutes then 880 gpm would be required to meet the above requirements. The AFW system meets the assumptions used in the Final Safety Analysis Report (FSAR), when single failure criterion is applied.

Following a Loss-of-Coolant-Accident (LOCA), the AFW system may be used to supply water to the steam generators to develop a water head within the steam generators and thereby limit potential tube sheet leakage from the primary to the secondary side of the steam generators.

Generally, components are constructed of carbon steel. The condensate storage tanks are lined to prevent corrosion; while other components are protected by chemical additions to the water.

5.7.5 PRA Insights

The AFW system provides feedwater to the steam generators to allow continued heat removal from the primary system when the main feedwater system is not available. In this capacity, the AFW system serves as one means of early core heat removal following a transient or small LOCA.

The AFW system is not a principal contributor to core damage frequency as an initiating event, but does contribute as part of the major accident sequences (1.4% at Zion and 2.6% at Sequoyah). One of the reasons for the relatively small contribution is the ability of the plant to initiate bleed and feed cooling using high pressure injection and the pressurizer PORVs. At a unit such as Surry, where two PORVs are required to open to provide sufficient bleed and feed capability, the AFW system can be a larger contributor (14.8%).

When performing the PRA for the AFW system certain items are plant specific such as human error. If the AFW system is normally configured so that one pump is locked out, failure to start that pump becomes critical. Failure to correctly realign the system after testing or maintenance is another. Common mode failures are also plant specific. One such failure is an undetected flow diversion through a cross-connect line to the second unit on multiple unit sites (Surry Risk Achievement Factor - 400).

A second example of a common mode failure is steam binding of the AFW pumps due to main feedwater leakage through system check valves (Surry Risk Achievement Factor - 400).

PRAs for 13 PWRs were analyzed to identify risk important accident sequences involving the loss of AFW and to identify and risk-prioritize the component failure modes involved. Below is a list of four accident categories explaining how AFW is a contributor to the analysis. Included at the end of the list is the risk important component failure modes.

1. Loss of Power System - A loss of offsite power is followed by the failure of the AFW. Due to the loss of actuating power the power operated relief valves (PORVs) cannot be opened, preventing adequate bleed and feed cooling, resulting in core damage.

   A station blackout fails all ac power except the vital Class IE ac busses from the dc invertors. All decay heat removal systems, except the turbine driven AFW pump, also fail. AFW subsequently fails due to battery depletion or hardware failures resulting in core damage.

   A dc bus fails, causing a trip and failure of the power conversion system. One AFW motor driven pump is failed by the bus loss, and the turbine driven pump fails due to loss of the turbine or valve control power. AFW is subsequently lost completely due to other failures. Bleed and feed cooling fails because PORV control is lost, resulting in core damage.

2. Transient Caused Reactor or Turbine Trip - A transient caused trip is followed by a loss of the power conversion system and the AFW system. Bleed and feed cooling fails either due to a failure of the operator to initiate it, or due to hardware failures, resulting in core damage.

3. Loss of Main Feedwater - A feedwater line break affects the common water source for the steam generators from the main feedwater and the AFW. If the operators fail to provide feedwater from the alternate sources and fail to initiate bleed and feed cooling core damage will result.

   A loss of main feedwater trips the plant, and the AFW system fails due to operator error and hardware failures. If he operators fail to initiate bleed and feed cooling core damage will result.

4. Steam Generator Tube Rupture (SGTR) - A SGTR is followed by a failure of the AFW. Coolant is lost from the primary until the refueling water storage tank is depleted. High pressure injection fails since recirculation cannot be established from the empty containment recirculation sump, and core damage results.

Risk Important Component Failure Modes

The generic component failure modes identified from the PRA analyses as important to AFW system failure are listed below in decreasing order of risk importance:

1. Turbine driven pump failure to start or run.
2. Motor driven pump failure to start or run.
3. Turbine or motor driven pump unavailable due to testing or maintenance.
4. AFW system valve failures such as; steam admission valves, trip and throttle valves, flow control valves, pump discharge valves, pump suction valves, and valves in testing or maintenance.

Risk reduction for core damage frequency through improvements to the AFW system are negligible for the plants studied in NUREG-1150.

5.7.6 Summary

The AFW system supplies high pressure feedwater to the steam generators to maintain a water inventory for removal of heat energy from the RCS by secondary side steam release in the event of the inoperability of the main feedwater system or during startup and shutdown evolutions. The discharge pressure generated by the AFW pumps is sufficient to deliver feedwater into the steam generators at any pressure up to the safety valve setpoint pressure. The capacity of the AFW system is designed so that the four steam generators will not boil dry nor will the primary side relieve fluid through the pressurizer relief valves following a loss of main feedwater flow in conjunction with a unit trip.

The AFW system consists of two sub-systems. One sub-system utilizes a steam turbine-driven pump, with the steam capable of being supplied from No. 2 or No. 3 steam generators upstream of the MSIVs. This system supplies a total of 880 gpm to all four steam generators. The second sub-system utilizes two motor-driven pumps each with a capacity of 440 gpm. The discharge piping is arranged so that each motor driven pump supplies two steam generators.

The automatic start signals for the auxiliary feed pumps are as follows:

1. Motor and Turbine Driven Pumps.
   a. Loss of both main feed pumps.
   b. Loss of one main feed pump if power is >80%.
   c. Low-low level in two or more steam generators.
   d. Engineered safety features actuation signal.
   e. Loss of power to the Class IE vital distribution system.
2. Motor Driven Pumps only.
   a. Low-low level in any single steam generator.

The preferred sources of water for all AFW pumps is the CST. A Technical Specification minimum of 280,000 gallons is required for the AFW system. As an unlimited backup water supply, a separate ESW system connection is provided.

In addition, at some plants, the fire protection system may be connected, with a spool piece, to the auxiliary feed system downstream of each electric driven pump. This feature allows raw water to be supplied directly to the steam generators. It would only be used if there were no other water source, or in the unlikely event that no auxiliary feedwater pumps were available.

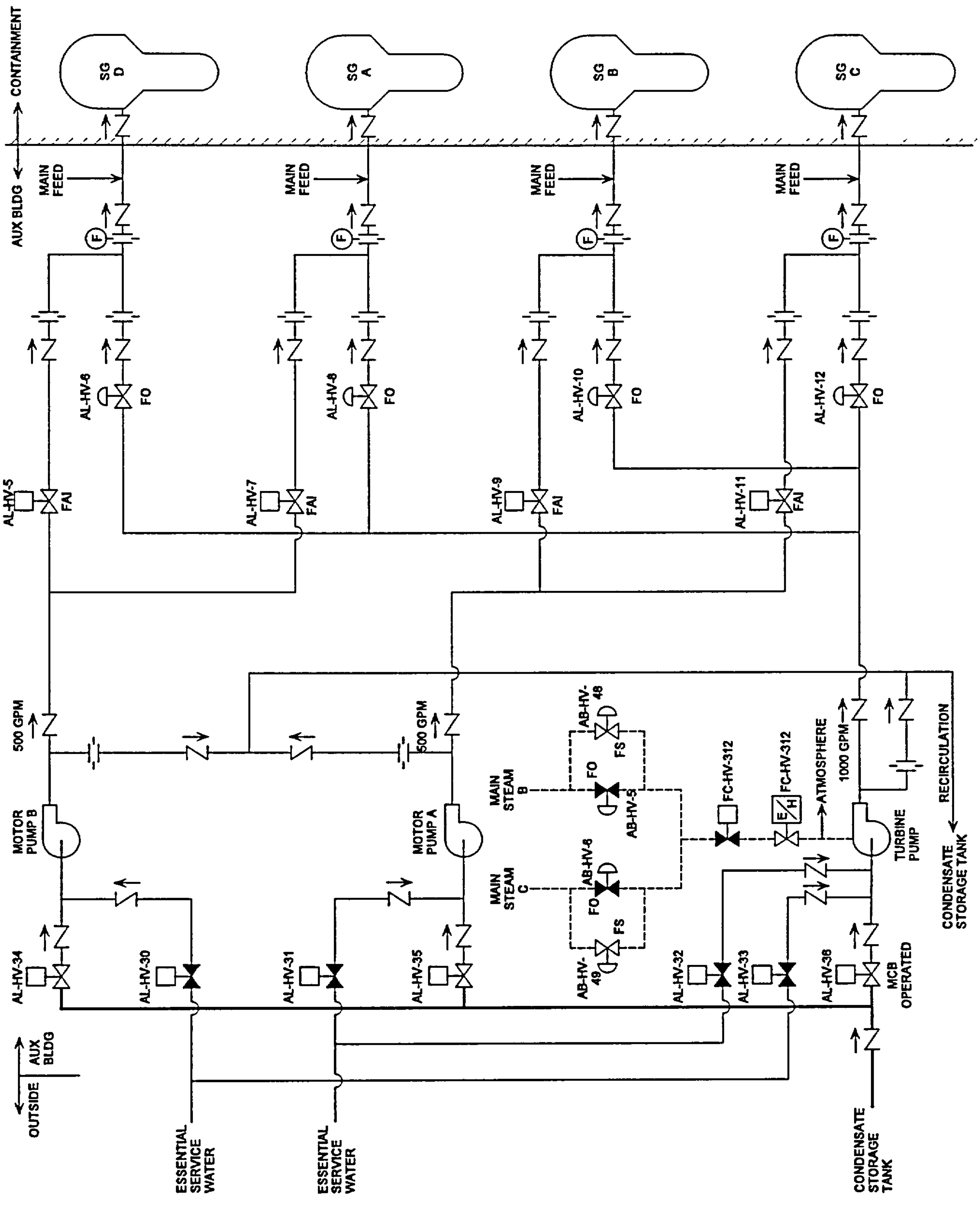

Figure 5.7-1 Auxiliary Feedwater System

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Auxiliary Feedwater System

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Learning Objectives:

1. State the purposes of the Auxiliary Feedwater System (AFW).
2. List all suction sources for the AFW pumps and under what conditions each is used.
3. List the five plant conditions that will result in an automatic start of the AFW system.
4. Explain how decay heat is removed following a plant trip and loss of offsite power.
5. Explain the relationship between the auxiliary feedwater system and the pressurizer Power Operated Relief Valves (PORVs) during a loss of offsite power.

5.8.1 Introduction

The purpose of the Auxiliary Feedwater System (AFW) is to automatically supply feedwater to the steam generators when the main feedwater system is unavailable. The steam generators provide a heat sink for the Reactor Coolant System (RCS), which in turn limits the increase in reactor coolant system pressure. By limiting the pressure increase in the RCS, challenges to the integrity of the RCS pressure boundary via opening of the pressurizer PORV's or safety valves are reduced. The specific transients for which the AFW system is designed, are as follows:

- Loss of all main feedwater
- Turbine trip and loss of offsite power
- Small break loss of coolant accident

In addition an electrically driven auxiliary feedwater pump is used to supply water to the steam generators during startups and shutdowns. This pump is not safety related and its function is to reduce the wear and tear on the safety grade auxiliary feed water pumps and their prime movers.

5.8.1.1 Design Basis

The auxiliary feedwater system is comprised of two safety related trains and is designated as an Engineering Safety Feature (ESF) System. Its design provides two redundant means of supplying feedwater to the steam generators as a means for removing heat from the reactor coolant during accident conditions. This system is designed to operate under the adverse environmental conditions as a result of the design basis accident (Loss of Coolant Accident or LOCA). This accident is the most limiting from a plant safety aspect and the auxiliary feedwater system's capability to operate for this accident assures its proper operation for less severe accidents.

Each of the two ESF AFW pumps is capable of providing 880 gpm to the four steam generators. Each pump is rated at 960 gpm which includes a minimum recirculation flow of 80 gpm. The pumps are rated for a pump head of 3400 feet (approximately 1472 psid) based on the most severe condition of pumping 880 gpm water into the steam generators with steam generator code safeties discharging at their maximum set pressure (1170-1230 psig). Each pump is designed to start and deliver rated flow within 60 seconds of receiving an automatic start signal. This system is designed in accordance with the single failure criteria.

This criteria states that the system must perform adequately with a single active or passive failure. An active failure is one involving the failure of a pump to operate or a valve to open or shut as needed. The use of two independent channels of control/protection signals and two trains of auxiliary feedwater equipment, insures that a single active failure will not prevent proper operation of the system. Therefore, if one AFW pump fails to start (an active failure), the second AFW pump in a separate independent train has sufficient capacity to provide water to the steam generators. A passive failure is one that involves the failure of inactive components such as pipes or pressure vessels. A pipe leak from a weld joint is an example of a passive failure. Design studies assume the maximum credible passive failure results in a 50 gpm leak for 30 minutes.

Specifically, the AFW system is capable of operating to its design values if an active failure occurs during the injection phase of a LOCA (the injection phase consists of water from the refueling water storage tank being injected into the RCS to cover the core). In addition the system must be designed so that no passive failure occurs during the injection phase. When the recirculation phase (water being pumped into the RCS from the containment sump) begins, the AFW system is designed to perform its intended function with either an active or a passive failure.

The design flow that is required to be provided by the AFW system during a complete loss of normal and preferred power (i.e. a station blackout) is 440 gpm to two of four SGs within 60 seconds of the initiation of the total loss of ac power. The actuation time and minimum injection flow prevents an excessive temperature rise in the reactor coolant system leading to overfilling the pressurizer and causing an overpressure event. This overpressure condition will cause the PORV's on the pressurizer to lift. After the PORV's cycle several times there is a probability of their failure which contributes to the core melt frequency of the plants PRA. This is a generic Westinghouse performance requirement which serves to bound all Westinghouse AFW systems. However, for this reference plant, the system design does not have to be as restrictive. The actual criteria as stated by the UFSAR to prevent overpressurizing the RCS for this design is 426 gpm being fed to three out of four SGs within 60 seconds. This works out to an actual minimum flow of 142 gpm being fed from one safety related auxiliary feedwater pump to each of the three SGs.

Taking into account allowable speed variations in the AFW pump prime movers and flow measurement errors (during normal conditions), the indicated minimum is actually set to 165 gpm. Assuming that one steam generator is faulted flow is diverted to three out of the four SGs. With the flow control valves, CV-3004's, throttled to allow 165 gpm per AFW train per steam generator, this results in 330 gpm being fed to each steam generator with both trains operating. This value is below the flow rate thought to be necessary to cause a water hammer (400 gpm).

The Emergency Operating Procedures (EOPs) entered after a reactor trip, when no "accident" is in progress, specifies a minimum adequate AFW flow of 495 gpm (165 gpm X 3 S/G's). For the remaining EOPs, that would be entered only if an accident was in progress, water hammer concerns are secondary to core decay heat removal. In addition, an adverse environmental flow measurement error of 98 gpm per feed line is assumed. This results in an indicated minimum required AFW flow (as stated in the EOPs) of 720 gpm: [142 gpm/SG x 3 S/G's] + [98 gpm/line x 3 lines] = 720 gpm. This minimum required flow (720 gpm) is less than the capacity of one safety related auxiliary feed pump (880 gpm).

As previously stated the AFW system consists of two safety related trains of which one pump is driven by a steam turbine and the redundant pump being driven by a diesel engine. The steam driven turbine is powered from the main steam system. This turbine can operate with steam pressures as low as 110 psig and is ac power independent. The diesel is started by two 28 Vdc batteries, and is dependent upon external power supplies for continued operation (i.e. service water cooling).

The majority of the auxiliary feedwater system is designed to Seismic Category I requirements with the exception of the pump recirculation line, the condensate storage tank, and the electric AFW pump with its associated piping, valves, and support systems. These components are designated Seismic Category II.

5.8.2 System Description

The auxiliary feedwater system (Figure 5.8-1) consists of one steam driven and one diesel engine driven auxiliary feedwater pump (each of which is safety related) with associated feedwater control valves, instruments, pipes, and controls. There is also a non-safety motor driven auxiliary feedwater pump that is used for feed addition during normal startups and shutdowns. Although this component is not safety related, it can fulfill certain action statement requirements in Technical Specifications and does have some operability requirements.

The normal supply of water to the AFW system is the Condensate Storage Tank (CST). An alternate, emergency supply from the Service Water System (SWS) exists should problems occur with the CST. Each safety related AFW pump discharges to its own discharge header. Recirculation lines, with a flow restrictor, return part of each pump's discharge to the CST. This recirculation line is required to allow some flow in case the auxiliary feedwater flow control valves are completely closed. The recirculation flow prevents the pump from being overheated while operating at its shutoff head.

Each pump's discharge header divides into four lines, one for each steam generator. Each steam generator has two auxiliary feedwater control valves; one from the turbine driven pump, and one from the diesel driven pump. The two lines from each steam generator's auxiliary feedwater flow control valves combine and enter the main feedwater line at the containment wall. The motor-operated flow control valves have a downstream check valve along with upstream and downstream isolation valves. In the common line downstream of the auxiliary feedwater flow control valves is a flow element with two flow transmitters which provide indication to meters in the main control room as well as the remote shutdown station. A flow sensing element just upstream of each control valve shuts its respective flow control valve if it detects flow greater than 500 gpm.

A single line from the condensate storage tank supplies the safety related auxiliary feedwater pumps. This line branches into two separate lines just upstream of the pumps. Each pump suction line has a normally open globe valve and a check valve. The suction of each pump has a Seismic Category I service water connection with a normally closed motor operated valve. The valve is normally shut in order to prevent chloride contamination of the steam generators. In the event of a condensate storage tank failure, the service water supply is initiated by manual operation of the isolation valves from the control room.

The electric auxiliary feedwater pump (Figure 5.8-2) can only take a suction on the condensate storage tank. It has a recirculation line that returns a portion of it's flow to the condensate storage tank. The discharge line has an air operated flow control valve in series with two parallel motor operated isolation valves. One of the motor-operated valves taps into the discharge line of the diesel driven auxiliary feedwater pump while the other taps into the discharge line of the steam turbine driven auxiliary feedwater pump.

5.8.2.1 Instrumentation And Control

The AFW system is required to have redundant features. This is accomplished, in part, by having separate trains of pumps, control valves, and pipes. Two independent automatic start signal channels are used. Channel A receives inputs from train A of the Reactor Protection System (RPS). It also receives inputs from non-RPS (i.e. not processed through the RPS) sources such as main feed pump tripped contacts. Similarly, channel B of the auxiliary feedwater automatic start circuitry receives signals from train B of the RPS and from other non-RPS sources. Channel A automatic start circuitry controls the steam turbine driven pump while channel B controls the diesel driven pump.

The control logic for the two pumps is similar. Five signals will cause an auxiliary feedwater pump to automatically start:

1. An undervoltage condition on the ESF 4.16 kV bus A1/A2. (A1 serves the turbine driven pump, A2 serves the diesel driven pump.)
2. Two of three SG level detectors at low-low water level in any one of the four steam generators, $\leq$11.5% (can be blocked when SG are drained for maintenance).
3. Safety injection signal.
4. Both main feed pumps tripped. (Can be blocked individually for either auxiliary feed pump.)
5. ATWS Mitigation System Actuation Circuit (AMSAC)

The fifth automatic start signal for the AFW pumps was added to address the following concerns. Sponsored studies conducted by Westinghouse and the NRC have shown that if a complete loss of normal feedwater or loss of electrical load occurs without a reactor trip, RCS pressure could exceed 3200 psig at thermal power levels above 70 percent. This occurrence assumes a common-mode failure in the reactor protection system which incapacitates auxiliary feed flow initiation and/or a turbine trip in addition to prohibiting a reactor trip. If this hypothetical event were to occur it requires an alternate method of providing AFW flow and turbine trip initiation. The code of federal regulations (10 CFR 50.62) required plants to install an alternate method of starting the AFW system and the ATWS Mitigation Actuation Circuitry (AMSAC) was installed. This circuitry detects a loss of main feedwater and actuates a turbine trip and AFW initiation within a set time if reactor power is greater than 70%. In order to minimize the amount of RCS voiding during an ATWS event a more conservative power level of 40% was chosen for AMSAC actuation.

An example of AMSAC logic is shown in Figure 5.8-3. Note that both trains of AMSAC must actuate to cause a turbine trip and AFW actuation. Four isolated SG level inputs (LT-519, 529, 538, and 547), and two isolated inputs from the turbine first stage pressure inputs (PT 505 and 506) provide actuation signal detection. Time delays of 25 seconds for SG level and 360 seconds for turbine power detection are provided to assist in the prevention of inadvertent actuation. Test actuation circuitry along with an installed maintenance bypass switch allows for system testing and deactivation. This circuitry is also deactivated by the operation of the SG low-low level AFW auto start block switches.

System actuation and status alarms are provided to ensure operators are aware of the AMSAC system status. If any of the above trips occur, a relay powered by 120 Vac preferred instrument bus Y11 is energized. This relay performs the following for the steam turbine driven pump:

1. Opens air-operated steam supply valves CV-1451, 1452, 1453, and 1454; one valve

from each steam generator.

2. Opens the Turbine Trip and Throttle valve, MO-3071.
3. Isolates SG blowdown and SG sampling.
4. Actuates an alarm for AUX FW AUTO START on the Remote Shutdown Station (RSS) panel.

The operation of channel B, which starts the diesel driven auxiliary feedwater pump, is similar to channel A. When channel B's relay is energized (powered by bus Y22), the following occurs:

1. Energizes the diesel start relay.
2. Service water isolation valve, MO-3060B, which supplies cooling water to the diesel lube oil cooler, pump lube oil cooler, diesel engine jacket cooler, and speed increaser lube oil cooler, is opened.
3. Isolates SG blowdown and SG sampling.
4. An annunciator, AFW PUMP A/B AUTO START, is actuated.

One of the recognized problems with insuring proper operation of an essential system like auxiliary feedwater, is the ability to demonstrate that a proper flow path exists when the system is in standby. This means that some way of indicating manual valve position is needed. Micro switches on crucial manual valves in the auxiliary feedwater system sense when the valves are not open and provide alarms in the control room and at the remote shutdown station.

The valves monitored include the individual safety related pump suction and discharge isolation valves, and the two isolation valves for each flow control valve. These switches also provide full open indication at the remote shutdown station and in the control room on the Safety Injection Equipment Status Lamp Panel.

5.8.2.2 Service Water

Recall that the CST is not a Seismic Category I storage tank. Therefore, the service water system provides a safety related backup water supply to the auxiliary feedwater system. In addition, it is also used to cool components in the diesel engine and steam turbine.

The service water supply to the auxiliary feedwater pumps suction is normally isolated by motor-operated gate valves. These valves have no automatic control function and are controlled by switches having incorporated lock out devices.

For the diesel driven auxiliary feedwater pump, service water supplies cooling water to the diesel jacket cooler, engine lube oil cooler and intercooler, pump lube oil cooler, and gear cooler on the speed increaser. Service water is also supplied to the suction of an attached auxiliary water pump which provides additional motive force for cooling flow through the engine lube oil cooler and intercooler (in series). The auxiliary water pump is belt driven off of the diesel engine. For the steam driven auxiliary feedwater pump, service water is a backup cooling supply for the lube oil cooler for the pump and the bearing heat exchangers for the turbine.

5.8.2.3 Auxiliary Feedwater Pump Turbine Steam Supply Valves

The original system design called for motor operated steam supply valves. This design has since been found inadequate because the system becomes dependent on the presence of ac power to operate correctly. For the turbine driven auxiliary feedwater pump to be completely ac power independent, the valves were changed to 3-inch, air-operated globe valves. Each valve has a single, solenoid operated, four-way valve in its air supply. The air supply taps off the existing air supply for the main steam isolation valves. There is also a backup air supply consisting of four 15 gallon accumulators (T-166A, 166B, 166C, and 166D) - one for each stop valve. These accumulators are designed to allow one valve cycle two hours after the normal air supply is lost. If accumulator pressure falls below 80 psig, alarms are initiated to warn of accumulator low pressure. Testing has shown that accumulator pressures as low as 45 psig have been sufficient to open the steam supply valves.

Each steam supply valve has a bypass line. The bypass line has two normally open isolation valves and a flow restricting orifice. The combined flow through all four orifices allows a nominal total of 500 lbm/hr steam flow to warm the steam supply lines and remove condensation. This prevents turbine damage from water hammer and moisture on startup. The bypass steam normally goes to an orificed line to the condenser. It may also vent to the atmosphere via a 3/4-inch line tapping off directly upstream of the turbine trip and throttle valve.

Control and indication power for the steam supply valves is supplied by 120 Vac preferred instrument power bus Y11 for CV-1451, 1452, 1453, and 1454. Control power supplies the solenoids on the four-way valves (SV-1451, 1452, 1453, and 1454). If a solenoid loses power, the valve positions to supply air to open its respective steam supply isolation valve. Thus the steam supply valves will automatically open if ac power is lost. This in itself will not start the turbine as the turbine trip and throttle valve remains closed.

It should be noted that a loss of a 120 Vac preferred instrument bus is relatively unlikely since it can also be powered from a 125 Vdc battery bus through an inverter (Chapter 6). If operating air is lost to the steam supply valves, they fail as is. There is no return spring in the valve actuator against which the operating air works.

The steam supply isolation valves can be manually operated with hand wheels Automatic opening of the steam supply isolation valves is initiated by the same conditions that cause automatic trip and throttle valve actuation.

5.8.2.4 Service Water to Auxiliary Feedwater Pump Cooler Valves

These normally closed valves (MO-3060A, and 3060B) (Figure 5.8-1) can supply service water to cool various steam turbine driven AFW pump and diesel driven AFW pump components. MO-3060A which supplies cooling water to the turbine driven AFW pump is a 2-inch motor-operated globe valve with an upstream isolation valve and a downstream check valve (not shown in this figure). However, because of the self-cooling feature of the turbine driven AFW pump, as discussed earlier, service water is a backup means of cooling. The motor operator for MO-3060A has been electrically disconnected and must be manually opened to provide service water cooling. The service water supply to the diesel driven feedwater pump, MO-3060B, is a 6-inch motor operated butterfly valve, and opens on receipt of a channel B pump start signal.

5.8.3 Component Description

The safety related diesel and turbine driven auxiliary feedwater pumps are identical six stage, horizontal, centrifugal pumps designed for 2000 psig, 70°F water. They operate with a design flow rate of 960 gpm, including 80 gpm recirculation, at a 3400 ft (1472 psid) head. The rated speed of each pump is 4560 RPM.

The turbine auxiliary feedwater pump bearing heat exchangers and lube oil heat exchangers have the capability of being cooled from three sources. There are two sources of self cooling, one from the second stage impeller of the pump via the pump casing and one from the recirculation line, and a third possible supply from the service water system. Using the self cooling supplies for cooling makes the turbine Auxiliary Feedwater Pump (AFP) independent of ac power and able to operate on a loss of all power to the 4160 V buses.

The non-safety electric driven auxiliary feedwater pump is also designed for 2000 psig, 90°F service. It will produce 1020 gpm of flow at 3400 ft head (1472 psid). This includes 140 gpm of recirculation flow back to the condensate storage tank. It is an eight stage centrifugal pump driven by a 1250 hp ac motor. An electric lube oil pump provides bearing lubrication on start up. As the pump accelerates, a shaft driven pump provides the motive force for oil lubrication.

5.8.3.1 Condensate Storage Tank

The condensate storage tank (Figure 5.8-4) has a capacity of 450,000 gallons. By technical specifications, this tank is required to be operable having an indicated volume of 239,000 gallons of water. This insures the availability of 196,000 gallons of usable water. The assumptions are that there are 27,700 unusable gallons at the bottom of the tank (minimum vortexing level) and a possible 14,400 gallons that must be allowed for due to level instrument error. The usable, unusable, and instrument error volumes are added together and rounded off to 239,000 gallons. This volume ensures there is enough water to maintain the plant in hot standby (using the SG safety valves) for two hours, and then cooling down the reactor coolant to 350°F over the next four hours. These times assume that 880 gpm of AFW flow is available. The 350°F cutoff point is where the residual heat removal system is capable of taking over and continuing the cooldown.

The auxiliary feed supply line is an 8-inch Seismic Category I pipe up to the metal bellows expansion joint adjacent to the bottom of CST. There is a debris strainer in the line (near the CST) to prevent foreign material (i.e. from the collapse of the CST floating roof) from entering the suction of the ESF AFW pumps. The recirculation line, which returns to the condensate storage tank, is a 2-inch pipe.

The diesel driven and turbine driven AFW pumps have CST low level trips of 35% and 30% respectively. The turbine driven AFW pump has the lower trip setpoint because it is the ac independent (hence, more reliable) pump. These trips provide an indication to the operator of a diminishing water volume in the CST. Control switches are provided on the MCB that can be used to block the low level trip, allowing subsequent pump restart.

The CST low-low level alarm, at 9 percent (55,000 gallons), alerts the operator to the impending loss of the CST as a source of water to the suction of the AFW pumps. This alarm warns the operator that an alternate source of auxiliary feedwater must be provided to insure a continuing source of water to the steam generators and to prevent damage to the auxiliary feedwater pumps.

CST level indication and protective signals are provided by Seismic Category I instruments. The variable and reference legs to the level transmitters are heat traced to minimize the possibility of freezing conditions causing erroneous level indications. The control power to the instrument channels is Class IE, however the heat trace power supply is non-Class IE.

The recirculation line has an in-line conductivity cell that outputs an alarm. This alarm is indicative of potential service water in-leakage at the auxiliary feedwater pump suction.

5.8.3.2 Diesel Engine

The diesel engine is a 12 cylinder Waukesha Model rated at 1579 bhp (at 1200 rpm) with a speed range of 450-1200 rpm. The diesel operates with a Westinghouse speed increaser, and has integral starting power supplies. The 28 Vdc batteries provides power to electric motor starting units. Diesel fuel oil is supplied to the engine through a day tank via a gravity feed line. The engine has a closed cooling system and a self actuated lubricating oil system with pumps, filters and lube oil coolers. Service water supplies the cooling water for the lube oil coolers, diesel engine cooling and speed increaser.

The diesel is started automatically by a Channel B auxiliary feedwater pump start signal. The diesel may be manually started and controlled remotely from the Main Control Board or from the Remote Shutdown System. It has the following automatic trips:

1. Overspeed - 1350 ±50 rpm,
2. Low lube oil pressure - 20 psig (25 second time delay on startup),
3. High jacket water temperature of 205°F (blocked on auto starts only),
4. Starter over-crank of 112 seconds, and
5. Lo-Lo CST level - 35% (blockable).

The speed of the diesel is automatically controlled (Figure 5.8-6) by the differential pressure between diesel pump discharge (PT-3083B) and auctioneered high steam header pressure (PT-516 or 545). This differential pressure signal is compared with an automatic setpoint in a auto-manual controller. The setpoint is normally at 50% (150 psid).

Diesel Fuel Oil

This system is an ESF system. It supplies diesel fuel for the auxiliary feedwater pump diesel as well as the two site emergency diesel generators. Two independent sources of fuel oil are provided. Each source consists of an underground storage tank, a transfer pump, and transfer piping. Each independent source of fuel supplies the auxiliary feed pump day tank. The day tank has sufficient capacity (500 gallons) to allow diesel operation at design capacity (960 gpm at 3400 ft of head) for 10 hours. The minimum allowed level in the AFW pump diesel fuel oil day tank per technical specifications is 450 gallons (69%) to fulfill the operability requirements for train B of AFW. Each storage tank has sufficient capacity to run both site emergency diesels at full load for four days and the auxiliary feedwater pump diesel for 24 hours.

5.8.3.3 Steam Turbine

A Terry turbine drives one of the auxiliary feedwater pumps. It is a non-condensing, single stage, horizontal shaft turbine, of a split casing design. The turbine is designed to operate with a steam pressure of 110 - 1305 psig. It exhausts to the atmosphere via an 8-inch line, and develops 1045 hp at its rated speed of 4560 rpm.

The turbine receives steam (Figure 5.8-5) from all four steam generators via supply lines upstream of the main steam isolation valves. Each 3-inch steam supply line has an air operated isolation valve. The steam supply lines join to form a combined 4-inch steam header. This steam header has a normally open stop valve and a normally closed trip and throttle valve.

The turbine is started automatically upon receipt of a Channel A start signal as described in section 5.8.2.1. This pump may be operated and controlled remotely from the MCB or from the RSS.

There's an auto-manual controller that develops an output based on an operator adjusted signal and a pressure difference consisting of pump discharge pressure (PT-3083A) minus auctioneered high steam header pressure (PT-514 or PT-536). In automatic control, the operator can vary the internal setpoint to control pump speed. By lowering the setpoint a lower controlling ΔP is called for, causing a lower flow rate. Similarly, a higher setpoint yields a higher flow rate. The controller is normally set to control at 50% (150 psid). In manual control, the controller output depends on the manual potentiometer setting.

The turbine has the following trips:

1. Mechanical overspeed (5500 ±100 rpm - 120% of rated speed),
2. Electrical overspeed (5100±50 rpm - 112% of rated speed), or
3. Low CST level of 30% (blockable).

The first two signals trip shut the turbine trip and throttle valve MO-3071. When MO-3071 is tripped, a spring rapidly forces the valve shut. In this condition, the valve cannot be opened by the motor operator until it is reset. A low CST level condition drives MO-3071 shut by its motor operator. This method allows the turbine to be restarted after CST level is restored (or after the low CST level trip is blocked) without the need to locally reset any mechanical devices.

Turbine Trip and Throttle Valve

MO-3071 is a 4-inch, normally closed, motor operated valve that is downstream of MO-3170 and upstream of the turbine governor valve. The valve is physically mounted on the steam turbine. It serves two functions:

1. On startup it slowly strokes open and helps throttle steam flow to the governor valve until the governor valve can adequately control steam flow and turbine speed;
2. When the turbine undergoes an overspeed condition, the trip valve closes rapidly to cut off steam flow to the turbine.

Auxiliary Feedwater Pump Turbine Stop Valve

MO-3170 is a normally open, 4-inch, motor-operated valve, downstream of the four air operated main steam header supply valves (CV-1451, 1452, 1453, 1454) and upstream of the trip and throttle valve, MO-3071 (Figure 5.8-5). This valve has no automatic opening or shutting functions. Control and indication power is from 480 Vac ESF bus B23.

5.8.3.4 Auxiliary Feedwater Flow Control Valves

Each steam generator has two 3-inch feed flow control valves (Figure 5.8-1) for a total of eight valves. Four of the valves (CV-3004A1, B1, C1, D1) control flow from the turbine driven pump. The other four valves (CV-3004A2, B2, C2, D2) control flow from the diesel driven pump. These valves receive control and indication power from 125 Vdc buses. During standby operation the valves are throttled to provide a pre-determined flow. When a flow control valve is positioned to throttle approximately 165 gpm to a steam generator, a WHITE light is illuminated on the Safety Injection Equipment Status Lamp Panel. The basis for the throttle settings is discussed in section 5.8.1.1. These valves have no automatic control functions. Each valve has remote and local manual controls.

If a high flow condition (>500 gpm) as sensed in a flow element upstream of an auxiliary feedwater control valve, then the respective feedwater control valve will shut as shown in Figure 5.8-7. Once one valve is closed due to a high flow signal, it prevents the other three auxiliary feedwater control valves in that train from closing due to high flow signals in their respective logic circuits. Arranging the logic in this fashion prevents more than one steam generator from losing feed flow due to excessive feed flow. Also, as shown in this logic diagram, if a second steam generator experiences high flow due to a fault in a steam generator its associated valve can be closed from the control room or locally.

The high flow signal serves to isolate feed to a steam generator with excessive flow as it's assumed that a flow rate over 500 gpm would be caused by a steam line or a feed line break. In the steam break case, the steam generator depressurizes, decreasing its flow resistance and causing high feed flow. It is undesirable to feed a faulted steam generator because of excessive cooling of the primary system. During a feed break, the effected feed line has less flow resistance and therefore robs the other steam generators of needed cooling flow.

5.8.4 System Operation

The auxiliary feedwater system is normally aligned to allow automatic initiation of flow to the steam generators as follows. The flow control valves are in a preset throttled position. The auxiliary feedwater pump LOCKOUT switches are in NORMAL (allowing start on loss of both MFPs and Low-Low SG level). The maintenance LOCKOUT switches are in NORMAL. Steam header isolation valves to the turbine driven auxiliary feed water pump are shut, while its associated steam stop valve (MO-3170) is open, and the turbine trip and throttle valve (MO-3071) is shut. Both pumps are in remote control with their respective control switches in AUTO. Their auto-manual controllers are in AUTOMATIC with a setpoint of 50% (150 psid).

When the auxiliary feedwater pumps are in standby, backflow from the main feedwater header is prevented by three check valves in the discharge line of each AFW pump (Figure 5.8-1). Some nuclear plants with similar AFW system designs have had problems with back leakage through the check valves which has rendered the AFW pumps inoperable.

Leakage of hot main feedwater (nominally 440°F) into the AFW pump discharge line will cause voiding in the piping, or if all the discharge check valves leak, could cause voiding in the AFW pumps. If the voiding is limited to the AFW discharge piping, a severe water hammer event could occur when the AFW pumps are started and supply feedwater to the steam generators. If the AFW pumps are voided, they would be unable to supply adequate AFW flow when started, or the high temperature could damage pump components. The AFW pump discharge temperature must be checked when the pump is secured and periodically thereafter to verify there is no back leakage of hot main feedwater into the AFW pump discharge piping.

5.8.4.1 Electric Driven Auxiliary Feed-water pump

The electric auxiliary feedwater pump (Figure 5.8-2) is used to add feedwater to the steam generators during normal startup and shutdown conditions to minimize wear on the safety grade auxiliary feed pumps. This component has no automatic start features and is not safety grade. The electric AFP is powered from 4160 Vac bus A-5.

Even though the electric driven AFW pump has no automatic start features it does have some automatic trip features. Below is a list of the automatic trip signal provided for the this pump:

1. Low suction pressure - 12 psia (can be overridden),
2. Low pump bearing pressure - 3 psig,
3. 4.16 KV bus A5 undervoltage,
4. Phase overcurrent, or
5. Ground overcurrent.

Overriding the low suction pressure trip may be required in an emergency situation if the electric AFW pump is the last means for providing feedwater to the steam generators.

An air operated control valve, CV-2967, is positioned to control pump head. In automatic, the control system senses steam header pressure from PT-535 and compares it with pump discharge pressure, downstream of the control valve, (PT-2967). The ΔP signal is compared with an adjustable setpoint in the auto-manual controller, PDC-2967. The automatic controller setpoint on PDC-2967 is adjustable from 0 - 100%, which represents 0 - 200 psid. The normal setpoint is selected to 50% which is equivalent to 100 psid.

Downstream of the control valve the pump discharge header splits into two lines, each with an isolation valve (MO-2947A, MO-2947B). These valves isolate feedwater to the discharge lines of the turbine driven pump and diesel driven pump. The valves are controlled from the MCB. To meet the requirements for separation of the two auxiliary feedwater trains, these valves must be normally shut. When the electric auxiliary feedwater pump is used, no more than one isolation valve should be opened at a time to prevent cross connecting the two trains of auxiliary feedwater. Check valves provide for Seismic Category I isolation.

5.8.4.2 AFW System Problem Areas - Water Hammer

The degree of feedwater piping displacement indicated that, some time in the past, moderate water hammer events had occurred. These events were generated by small amounts of steam (approximately 5 cubic inches) being rapidly condensed by cold feedwater. During conditions of low feed flow, with SG levels below the feed ring, steam voids were being created in the feed ring.

Two leakage paths out of the feedwater system were identified:

1. Back leakage through the system check valves of the main feedwater system and
2. leakage through a 0.025 inch gap between the SG nozzle and the sleeve of the feed ring.

This gap may have grown to 0.25 inches due to erosion. Steam voids form in the SG feed rings only if the SG level is below the top of the feed ring and the total feed flowrate is less than the leakage out of the system. The maximum leakage rate was calculated to be 30 gpm. AFW flow rates in excess of 400 gpm are thought to be necessary for sudden collapse of the steam voids vice a gradual sweeping action which results in no water hammer. For this reason the auxiliary feed control valves are preset to throttled position to supply approximately 165 gpm per steam generator.

Reactor Trip Scenario

After a reactor trip, the level in the steam generators shrink due to the rapid reduction in steam flow. The main feedwater system is isolated when the Low T_{avg} setpoint (564°F) is reached. This stops feed flow to the S/G's about 2.5 seconds after the reactor trip. There is a 30 - 60 second delay prior to the delivery of AFW flow. With feed flow secured, and SG level below the top of the J-tubes on top of the feed ring (20.6% Narrow Range (NR)), steam voids begin to form in the feed ring. The volume of all the J-tubes is about 1.5 gallons. Void formation increases if SG levels dropped below the bottom of the feed ring (9% NR) due to leakage through the feed ring gap.

Hot Standby/Hot Shutdown Scenario

During hot standby and hot shutdown conditions, it is difficult to maintain continuous feed flow to the S/G's due to the low steaming and blowdown rates. During this operation the S/G's are usually fed intermittently. If the level was allowed to slowly decrease to below 20.6% and AFW flow was less than leakage (would only be affected by check valve back leakage at this point) then steam voids would begin to form. If level continued to decrease to below the bottom of the feed ring (9% NR), then draining of the feed ring would increase due to the additional leak path through the feed ring gap. If subsequent AFW flow rates were allowed to go above 400 gpm, then water hammer may result.

5.8.5 PRA Insights

The AFW system provides feedwater to the steam generators to allow continued heat removal from the primary system when the main feedwater system is not available. In this capacity, the AFW system serves as one of the means to perform the safety function of early core heat removal following a transient or a small break loss of coolant accident (SBLOCA).

The AFW system is not a principal contributor to core damage frequency as an initiating event, but does contribute as a part of the major accident sequences (1.4% at Zion and 2.6% at Sequoyah). One of the reasons for the relatively small contribution is tied to the ability of the plant to initiate feed and bleed cooling using high pressure injection and the pressurizer PORV's. At a unit such as Surry, where two PORV's are required to open to provide sufficient feed and bleed capability, the AFW system can be a larger contributor (14.8%).

When performing the PRA for the AFW system certain items are plant specific such as human error. If the AFW system is normally configured so that one pump is locked out, failure to start that pump becomes critical. Failure to correctly realign the system after testing or maintenance is another. Common mode failures are also plant specific. One such failure is an undetected flow diversion through a cross-connect line to the second unit on multiple unit sites. A second example of a common mode failure is, steam binding of the AFW pumps due to main feedwater leakage through system check valves.

PRAs for 13 PWRs were analyzed to identify risk important accident sequences involving the loss of AFW. These were used to identify and risk-prioritize the component failure modes involved. Below is a list of four accident categories and how AFW is a contributor to the analysis. Also included at the end of this list is the risk important component failure modes.

Loss of Power System

A loss of offsite power is followed by the failure of the AFW. Due to the lack of actuating power, the power operated relief valves (PORVs) cannot be opened preventing adequate feed and bleed cooling, resulting in core damage.

A station blackout fails all ac power except the vital Class 1E ac instrument busses from the dc inverters and all decay heat removal systems except the turbine driven AFW pump. AFW subsequently fails due to battery depletion or hardware failures, resulting in core damage.

A dc bus fails, causing a trip and failure of the power conversion system. One AFW motor driven pump is failed by the bus loss, and the turbine driven pump fails due to loss of the turbine or valve control power. AFW is subsequently lost completely due to other failures. Feed and bleed cooling fails because PORV control is lost, resulting in core damage.

Transient Caused Reactor or Turbine Trip

A transient caused trip is followed by a loss of the power conversion system and the AFW. Feed and bleed cooling fails either due to a failure of the operator to initiate it, or due to hardware failures, resulting in core damage.

Loss of Main Feedwater

A feedwater line break drains the common water source for the main feedwater and the AFW. The operators fail to provide feedwater from the alternate sources, and fail to initiate feed and bleed cooling, resulting in core damage.

A loss of main feedwater trips the plant and the AFW fails due to operator error and hardware failures. The operators fail to initiate feed and bleed cooling, resulting in core damage.

Steam Generator Tube Rupture (SGTR)

A SGTR is followed by a failure of the AFW. Coolant is lost from the RCS until the refueling water storage tank is depleted. High pressure injection fails since recirculation cannot be established from the empty containment recirculation sump and core damage results.

Risk Important Component Failure Modes

The generic component failure modes identified from the PRA analyses as important to AFW system failure are listed below in decreasing order of risk importance.

1. Turbine driven pump failure to start or run.
2. Motor driven pump failure to start or run.
3. Turbine or motor driven pump unavailable due to testing or maintenance.
4. AFW system valve failures such as steam admission valves, trip and throttle valves, flow control valves, pump discharge valves, pump suction valves, and valves in testing or maintenance.

Risk reduction for core damage frequency through improvements to the AFW system are negligible for the plants studied in NUREG-1150.

5.8.6 Summary

The auxiliary feedwater system is an engineering safety feature system designed to Seismic Category I requirements. It is designed to delivering at least 440 gpm to two steam generators within 60 seconds of the loss of all ac power. To meet the single failure requirement, the system has two independent trains of equipment. Each train takes a suction on the condensate storage tank. Service water provides a backup suction supply to the AFW pumps should the condensate storage tank become unavailable.

The auxiliary feed flow to the steam generators is controlled by throttle valves. These valves are preset to provide a desired flow of water to the steam generators. This flow is more than that needed to meet the decay heat requirements of the reactor following a reactor trip. Yet the flow is below the threshold value that could cause water hammers to occur in the feed piping or feed ring inside the steam generators. Pump speed is controlled by comparing the difference in pressure between pump discharge and steam header pressure with a desired setpoint differential pressure. This method of control insures approximately the same feed flow rate for any given steam generator pressure.

The train A pump prime mover is a steam turbine. Each main steam line has a 3-inch steam supply line located upstream of its main steam isolation valve. Each supply line contains an air operated, fail open, isolation valve. The four steam supply lines join to form a common steam header with a motor operated stop valve and a motor operated trip and throttle valve. This pump is ac power independent and will continue to operate in the even of a station blackout.

The train B pump prime mover is a diesel engine. The diesel is cooled by service water. It has two 28 Vdc batteries for starting and various control functions. The diesel AFW pump requires some support from external vital ac power in that service water cooling is necessary during operation.

A non-safety grade electric driven auxiliary feedwater pump is also installed. This pump takes a suction on the condensate storage tank. It delivers flow to the discharge lines of the two safety grade auxiliary feedwater pumps through a flow control valve and two motor operated isolation valves. This pump is used to add feedwater to the steam generators during shutdown periods and startups. It saves wear and tear on the safeguards equipment.

Systems which interface with the auxiliary feedwater system include:

- Condensate - The condensate storage tank provides a source of water for the auxiliary feedwater system. It has high-low and low-low level alarms and blockable low-low level trips of the turbine and diesel AFW pumps.
- Diesel Fuel Oil - A day tank is located above the auxiliary feedwater diesel and supplies fuel oil to the engine by gravity feed.
- Service Water - An alternate supply for auxiliary feedwater should the CST fail or empty. Service water is isolated from the pumps suction by motor operated valves.

There are two independent sets of controls and indications for the auxiliary feedwater pumps. The following signals will automatically start the auxiliary feedwater pumps:

1. Safety Injection Signal.
2. Low-low steam generator water level (2/3 level transmitters on 1/4 S/G's, can be blocked)
3. Loss of 4.16 KVac power to ESF buses A1 and A2
4. Loss of both main feed pumps (this signal can be blocked during a normal shutdown)
5. ATWS Mitigation System Actuation Circuit (AMSAC)

When the steam turbine receives a start signal, the following occur:

1. Steam line isolation valves open.
2. Turbine trip and throttle valve opens.
3. Governor controls steam flow to accelerate turbine to desired speed.
4. Pump room cooling fans energize.
5. Steam generator blowdown and blowdown sample isolation valves shut.
6. Various alarms energize to indicate pump start.

When the diesel gets a start signal, the sequence of events is the same as above except instead of steam supply valves and throttle valves opening the diesel engine is started by the self contained, battery powered, dc electric motors.

The auxiliary feedwater system can be controlled from the control room or the remote shutdown station.

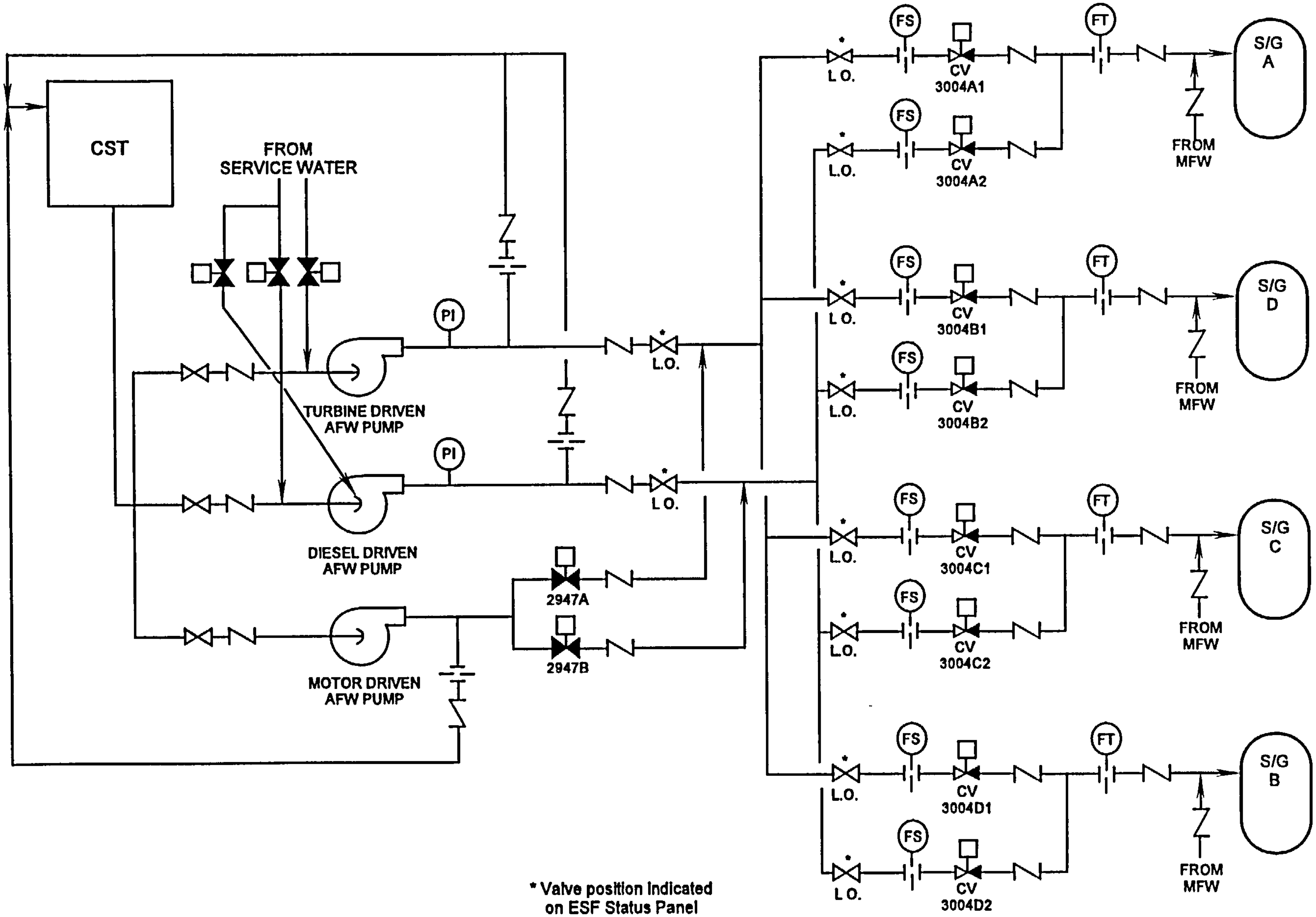

Figure 5.8-1 Auxiliary Feedwater System

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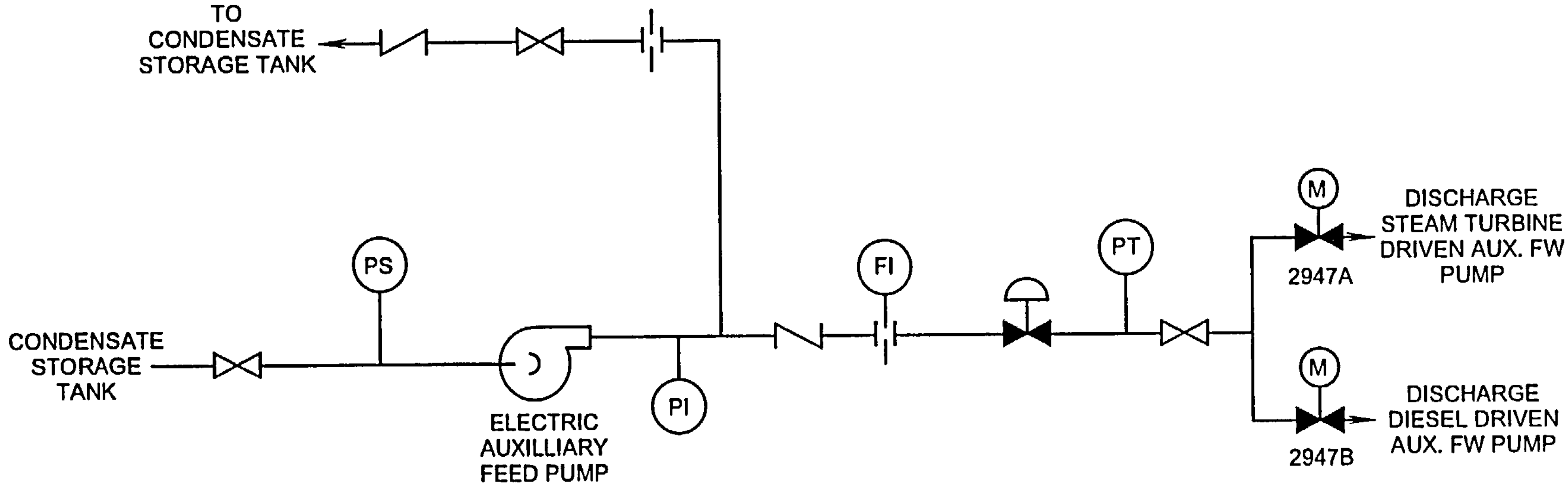

Figure 5.8-2 Electric Auxilliary Feedwater Pump

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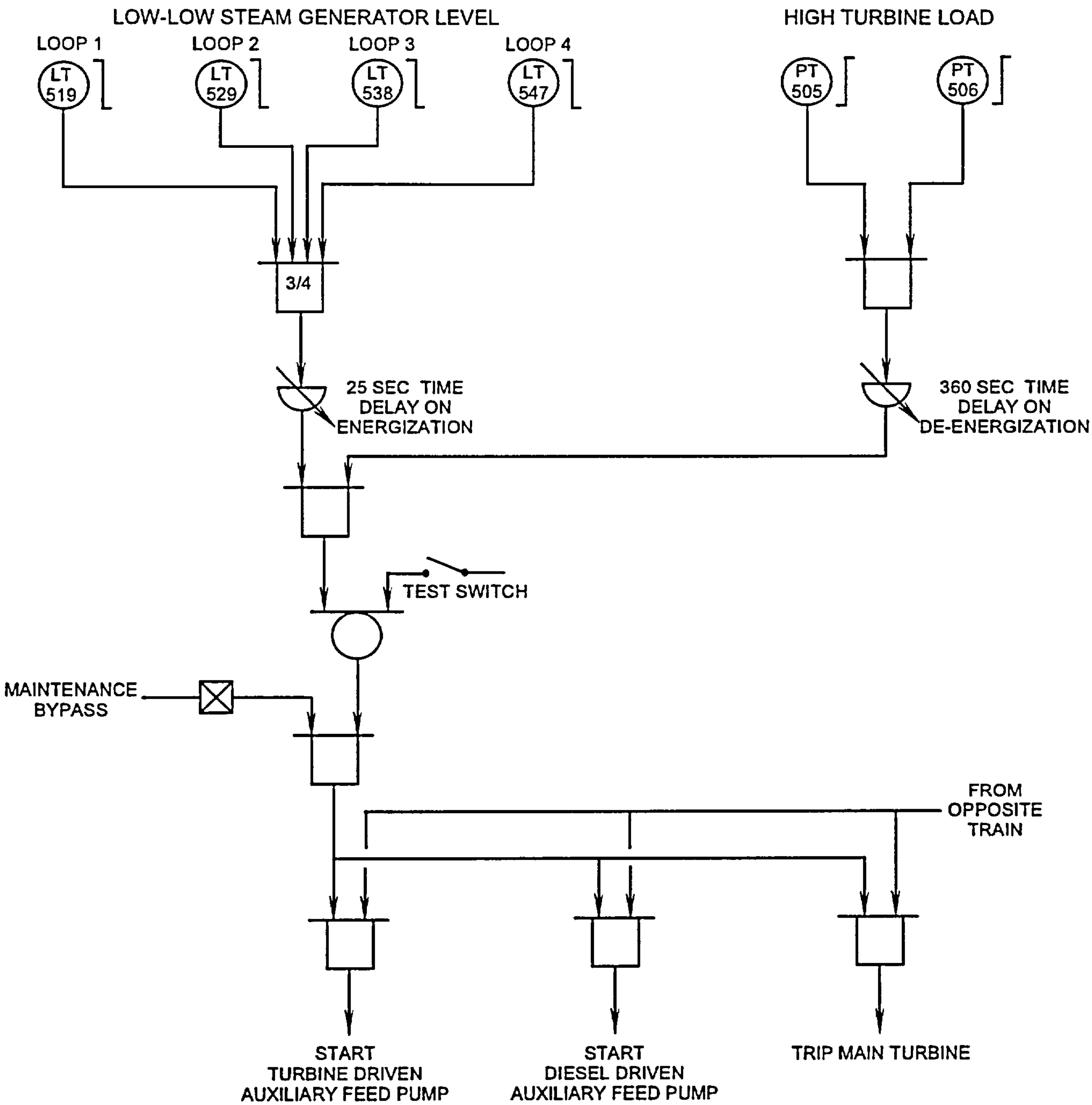

Figure 5.8-3 AMSAC Trip Circuit

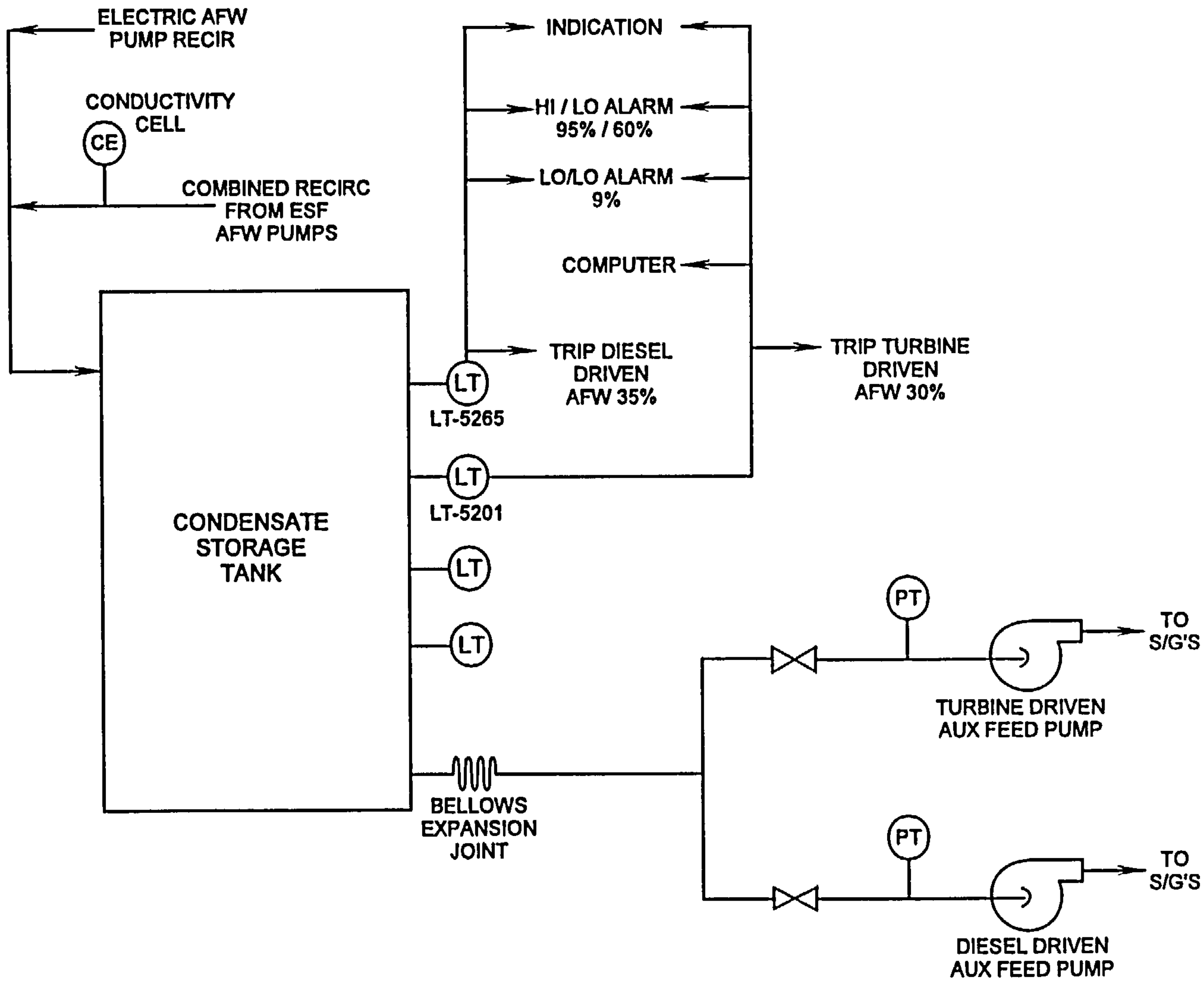

Figure 5.8-4 Condensate Storage Tank

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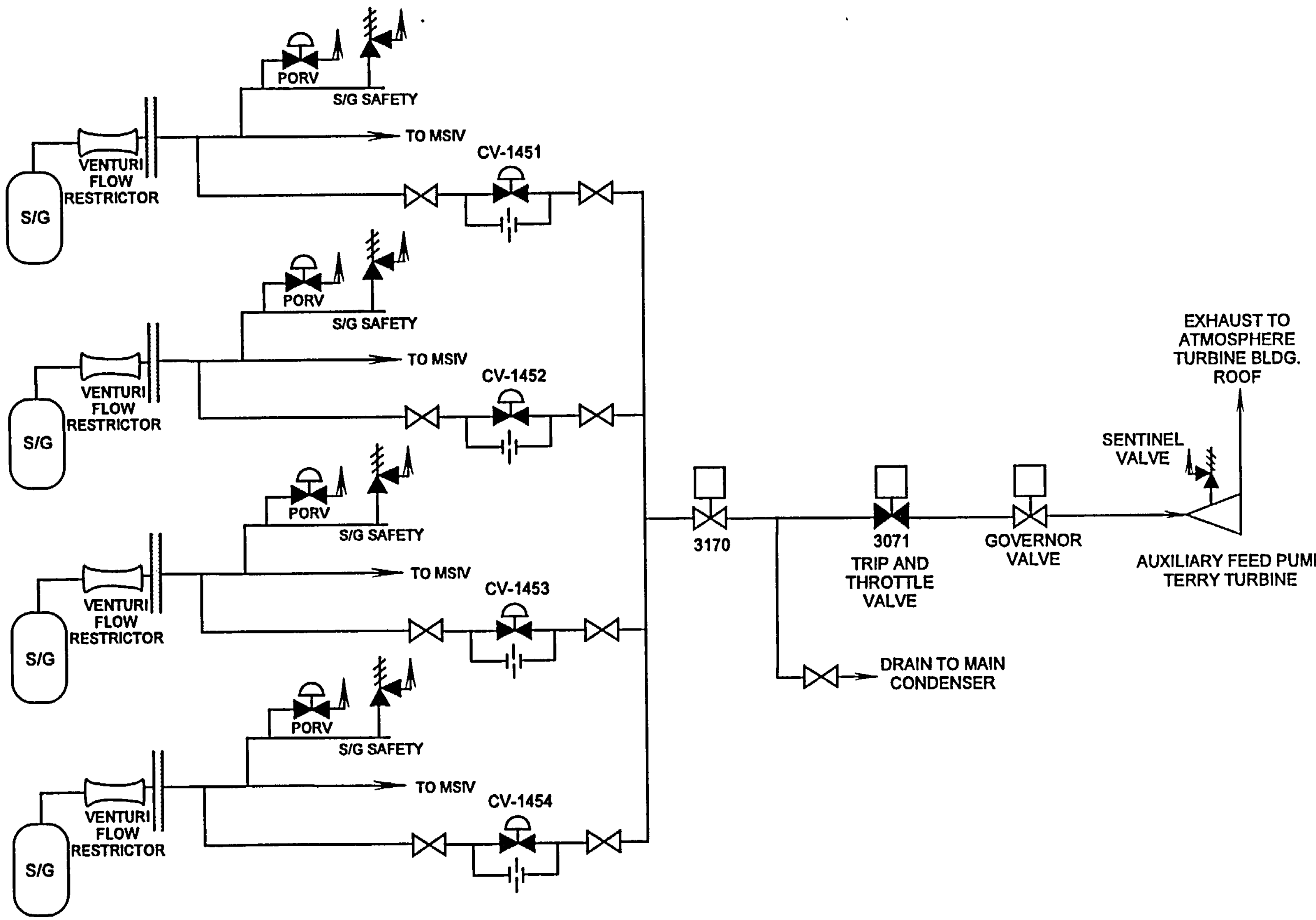

Figure 5.8-5 Steam Supplies To Turbine Driven Auxiliary Feed Pump

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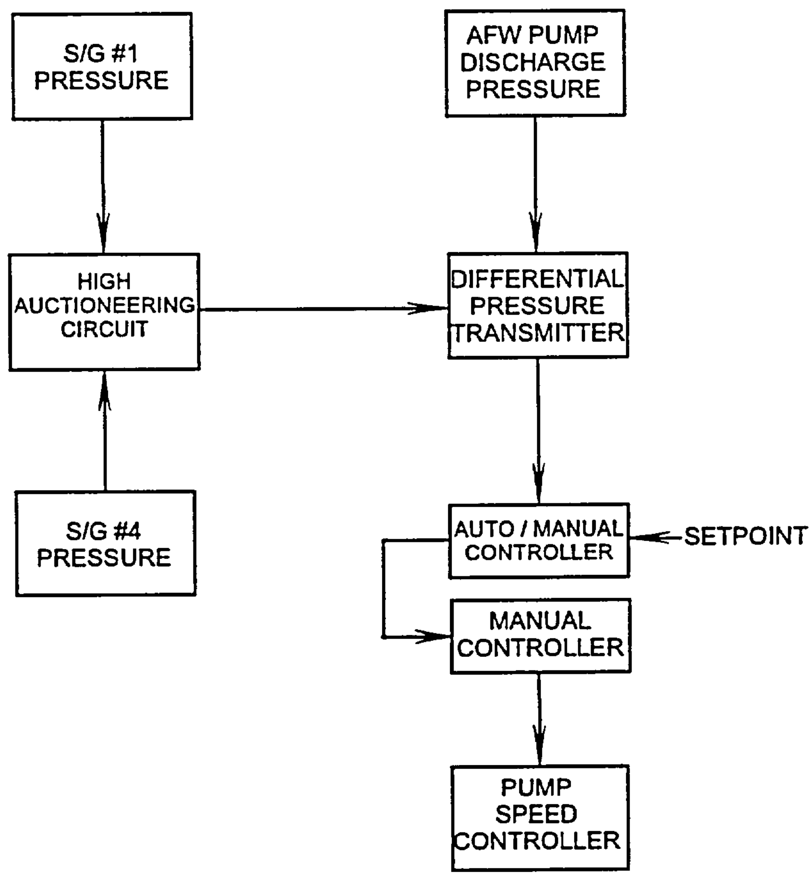

Figure 5.8-6 Train "B" (Diesel) AFW Pump ΔP Control

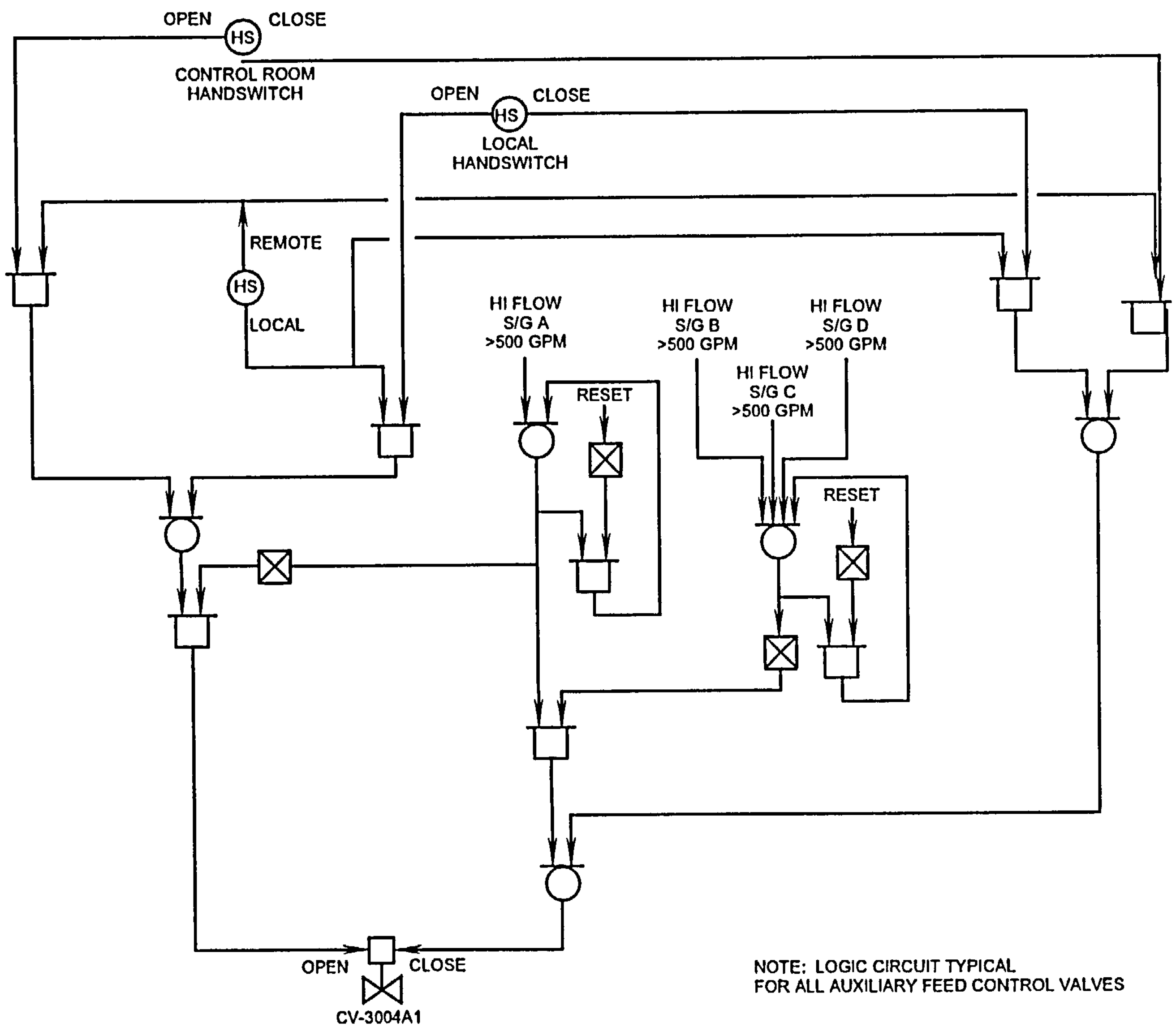

Figure 5.8-7 Auxiliary Feedwater ControValve CV3004A1

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