General Electric Advanced Technology Manual

Chapter 4.7

Interfacing System LOCA

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4.7 INTERFACING SYSTEM LOCA

Learning Objectives:

1. Define the term "interfacing system LOCA (ISL)"

2. List the major interfacing lines for a BWR.

3. Explain why an interfacing system LOCA is a safety concern.

4.7.1 Introduction

The term "interfacing system LOCA" (ISL) refers to a class of nuclear plant loss of coolant accidents in which the reactor coolant system pressure boundary interfacing with a support system of lower design pressure is breached. This could cause an over pressurization and breach the support system, portions of which are located outside of the primary containment. Thus, a direct and unisolable coolant discharge path would be established between the reactor coolant system and the environment. Depending on the configuration and accident sequence, the emergency core cooling systems as well as other injection paths may fail, resulting in a core melt with primary containment bypassed.

The Reactor Safety Study, WASH-1400, identified an interfacing system LOCA accident in a PWR as a significant contributor to risk from the core melt accidents (event V). The event V arrangements were defined to be two check valves in series or two check valves in series with an open motor operated valve. Such valve arrangements are commonly used in PWRs but rarely in BWRs.

As a result of the WASH-1400 study and the TMI-2 accident, all light water reactors with operating license granted on or before February 23, 1980 were required to periodically test or continuously monitor the event V valves. The periodic test consisted of in-service leak rate testing of each check valve every time the plant is shutdown and/or each time either check valve is moved from the fully closed position.

Since early 1981, the Office of Nuclear Reactor Regulation (NRR) staff commenced back fitting operating reactors by requiring in-service leak rate testing of all pressure isolation valves that connect the reactor coolant system to lower pressure systems. On April 20, 1981, orders were sent to 32 PWRs and 2 BWRs which required leak rate testing of Event V valves.

In February 1985, the NRR staff established new acceptance criteria for leak rate testing. The leak rate of each valve must be no greater than one half gallon per minute for each nominal inch of valve size and no more than 5 gallons per minute for any particular valve.

The current leak rate testing requirements for pressure isolation valves on BWRs are as follows:

- At least once per 18 months (or 24 months based on operating cycle).

- Prior to returning the valve to service following maintenance or replacement work.

- Recent BWR operating experience indicates that pressure isolation valves may not adequately protect against over pressurization of low pressure systems. The over pressurization may result in the rupture of the low pressure piping. This event, if combined with failures in the emergency core cooling systems and other systems that may be used to provide makeup to the reactor coolant system, would result in a core melt accident with an energetic release outside the containment.

4.7.2 Interfacing Lines

The major interfacing lines discussed in the following sections include:

- LPCI injection lines
- shutdown cooling suction line
- shutdown cooling return line
- reactor vessel head spray line
- high pressure core spray suction
- low pressure core spray line

4.7.2.1 LPCI Injection Line

The RHR system consists of two loops, (A & B). Each loop contains two pumps, associated valves; and piping to inject water from the suppression pool to the reactor vessel. Both loops A and B are used for multiple purposes (modes), such as shutdown cooling mode, containment spray mode, and suppression pool cooling mode.

Failure of a LPCI injection testable check valve and/or the normally closed injection valve would over pressurize the RHR system piping and cause failure of that loop. The relief valve located between the inboard and outboard injection valves has a capacity of approximately 185 gpm and a set pressure of 500 psig. The relief valve is capable of handling the flow from the testable check valve bypass valve, but not the amount of flow that would result from a failure of the testable check valve to close.

4.7.2.2 Shutdown Cooling Suction Line

The suction line from recirculation loop B contains an inboard and outboard isolation valve and an individual pump isolation valve. The containment isolation valves automatically close if reactor vessel reaches level 3 or reactor pressure increases to 125 psig. Failure of the containment isolation valves to close would allow the low pressure piping to fail causing an interfacing system LOCA.

4.7.2.3 Reactor Vessel Head Spray

The vessel head spray line is used during the shutdown cooling mode of operation to cool the upper vessel area prior to flood-up of the vessel. If the isolation check valves and the motor operated isolation valves fail, the low pressure RHR system LPCI line will be over pressurized.

The result is identical to paragraph 4.7.2.1 mentioned above. Therefore, the same indications will be available to the operators.

4.7.2.4 Low Pressure Core Spray Injection Line

Failure of the LPCS testable check valve and/or the normally closed injection valve would over pressurize the LPCS piping and possibly cause a rupture. The relief valve lifts automatically at a set pressure of 586 psig and has the same design requirements as the RHR injection line relief valve.

4.7.2.5 High Pressure Core Spray Suction

The HPCS system starts automatically on level 2 or high drywell pressure. Upon actuation, the normally open suction valve from the condensate storage tank is signaled to open, the test return valves are signaled to close, and the normally closed injection valve is signaled to open. Subsequently, the injection valve receives an automatic close signal when vessel level reaches level 8 thus the pump will continue running with flow through the minimum flow line. If the minimum flow valve fails closed and the water leg pump discharge stop check valve fails open, there is a chance of over pressurizing the low pressure suction piping.

4.7.3 Operating Experiences

With two series check valves the probability of at least one of the check valves being seated and not leaking would be extremely high. In addition, if leakage were to occur to the point of causing a LOCA in the low pressure piping, the high differential pressure across the valve should cause the valves to seat, which would terminate the accident. However, actual operating experiences indicate that both check valves have failed to properly close.

The Nuclear Power Experiences Manual reports that between 1974 and 1978 there were nine dilution events in the cold leg accumulators of PWR plants. The following sections discuss other events that pertain to BWRs and interfacing system LOCAs.

4.7.3.1 Cooper Nuclear Station

The HPCI testable check valve failed to remain fully closed due to a broken sample probe wedged under the edge of the valve disc. The origin of the sample probe was traced to the feedwater system. The failure was not recognized until backflow of feedwater to the HPCI pump suction occurred.

4.7.3.2 LaSalle event on October 5, 1982

A testable check valve was tested with the plant at 20% power. The test was accomplished by opening the check valve bypass valve to equalize pressure across the check valve disc and then opening the check valve from the control room. Following the test, both the bypass valve and the testable check valve failed to reclose.

4.7.3.3 Pilgrim event on February 12, 1986 and April 11, 1986

On February 12, both the testable check valve and the normally closed LPCI outboard injection valve leaked, resulting in frequent high pressure alarms. These alarms occurred repeatedly for approximately two weeks prior to this event. Operators simply vented the piping after each alarm. On this date, the outboard injection valve was manually closed and its closing torque switch replaced. The plant continued operation until April 11, at which time, more high pressure alarms occurred. It was discovered that the outboard injection valve started leaking again and subsequently required a plant shutdown to facilitate repairs.

4.7.3.4 Dresden Unit 2 Event

On February 21, 1989, with Dresden Unit 2 operating at power, temperature was greater than normal in the HPCI pump and turbine room. The abnormal heat load was caused by feedwater leaking through uninsulated HPCI piping to the condensate storage tank. During power operation, feedwater temperature is less than 350°F, and feedwater pressure is approximately 1025 psi. Normally, leakage to the condensate storage tank is prevented by the injection check valve, the injection valve, or the discharge valve on the auxiliary cooling water pump.

On October 23, 1989, with the reactor at power, leakage had increased sufficiently to raise the temperature between the injection valve and the HPCI pump discharge valve to 275°F and at the discharge of the HPCI pump to 246°F. Pressure in the HPCI piping was 47 psia. On the basis of the temperature gradient and the pressure in the piping, the licensee concluded that feedwater leaking through the injection valve was flashing and displacing some of the water in the piping with steam. This conclusion was confirmed by closing the pump discharge valve (M034) and monitoring the temperature of the piping. As expected, the pipe temperature decreased to ambient.

The event at Dresden is significant because the potential existed for water hammer or thermal stratification to cause failure of the HPCI piping and for steam binding to cause failure of the HPCI pump. Further, failure of HPCI piping downstream from the injection valves would cause loss of one of two feedwater pipes.

The licensee had not heard the noise that is usually associated with water hammers. Never the less, loosening of pipe supports, damage to concrete surfaces, and the pressure of steam in the piping strongly indicated that water hammers had occurred in the HPCI system, probably during HPCI pump tests or valve manipulations.

4.7.4 PRA Insight

NUREG/CR-5928, ISLOCA Research Program, primary purpose is to assess the ISLOCA risk for BWR and PWR plants. Previous reports (NUREG/CR-5604, 5745, and 5744) have documented the results of ISLOCA evaluations of three PWRs and to complete the picture a BWR plant was examined. One objective of the Research Program is identification of generic insights. Toward this end a BWR plant was chosen that would be representative of a large percentage of BWRs.
The reference BWR plant used as the subject of ISLOCA analysis was a BWR/4 with a Mark-I containment. Power rating for the plant is 3293 MWt. BWRs of similar design include:

- Brown's Ferry 1,2, & 3
- Peach Bottom 2 & 3
- Enrico Fermi
- Hope Creek
- Susquehanna 1 & 2
- Limerick 1 & 2

NUREG/CR-5928 document describes an evaluation performed on the reference BWR from the perspective of estimating or bounding the potential risk associated with ISLOCAs. A value of 1 x 10-8 per year was used as the cutoff for further consideration of ISLOCA sequences.

A survey of all containment penetrations was performed to identify possible situations in which as ISLOCA could occur. The approach taken began with an inventory of these penetrations to compile a list of interfacing systems. Once the list was complete, the design information for each system was reviewed to determine the potential for a rupture given that an over pressure had occurred. The systems included:

- reactor core isolation cooling system

- high pressure coolant injection system

- core spray system

- residual heat removal system

- reactor water cleanup system

- control rod drive system

The results of NUREG/CR-5928 concluded that ISLOCA was not a risk for the BWR plant analyzed. Although portions of the interfacing systems are susceptible to rupture if exposed to full RPV pressure, these are typically pump suction lines that are protected by multiple valves.

4.7.5 Summary

In order to reduce the probability of this type of event even further, license changes have been made to the technical specifications that limit the maximum leak rate through isolation valves.

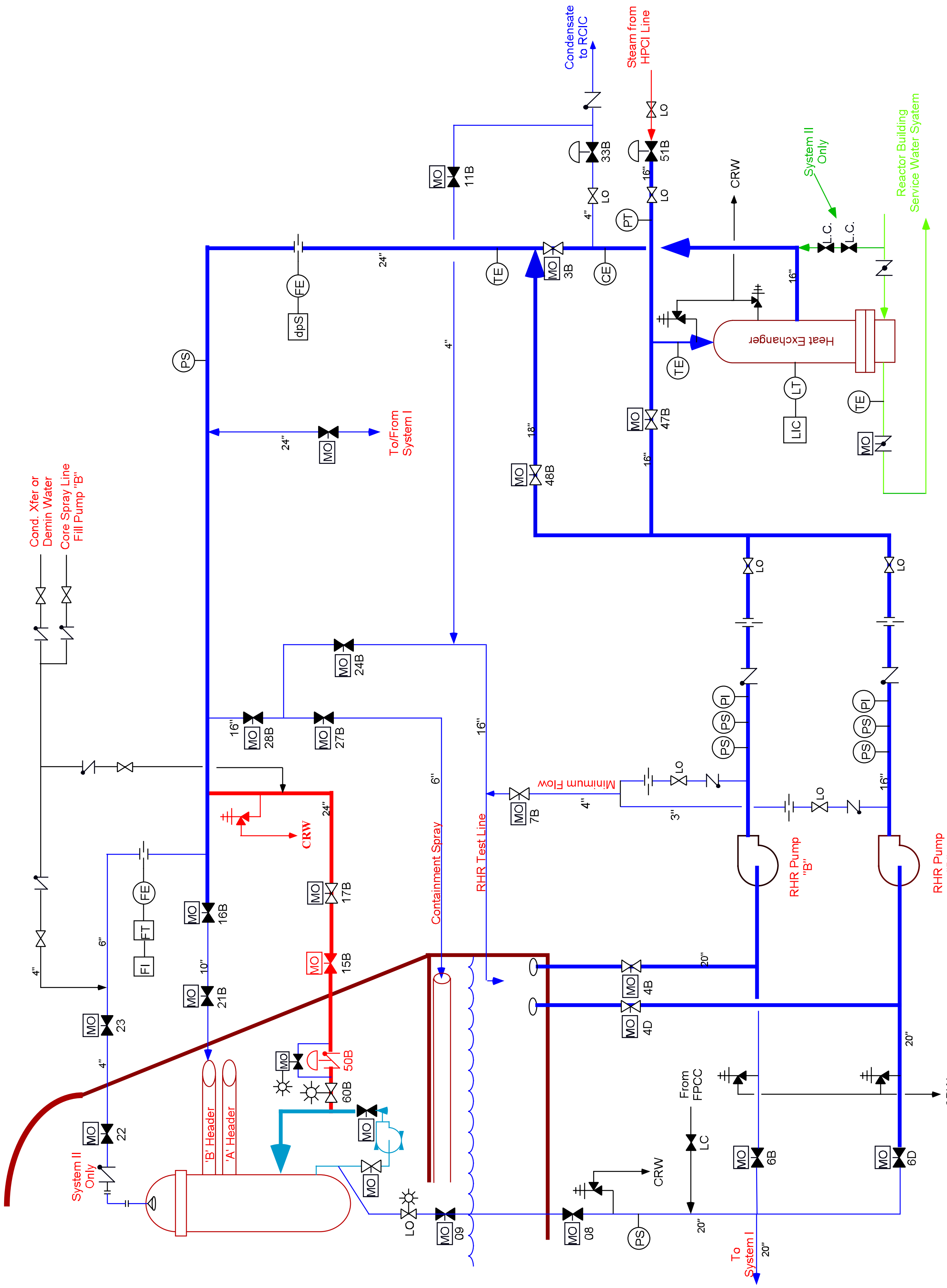

Figure 4.7-1 RHR System LPCI Mode

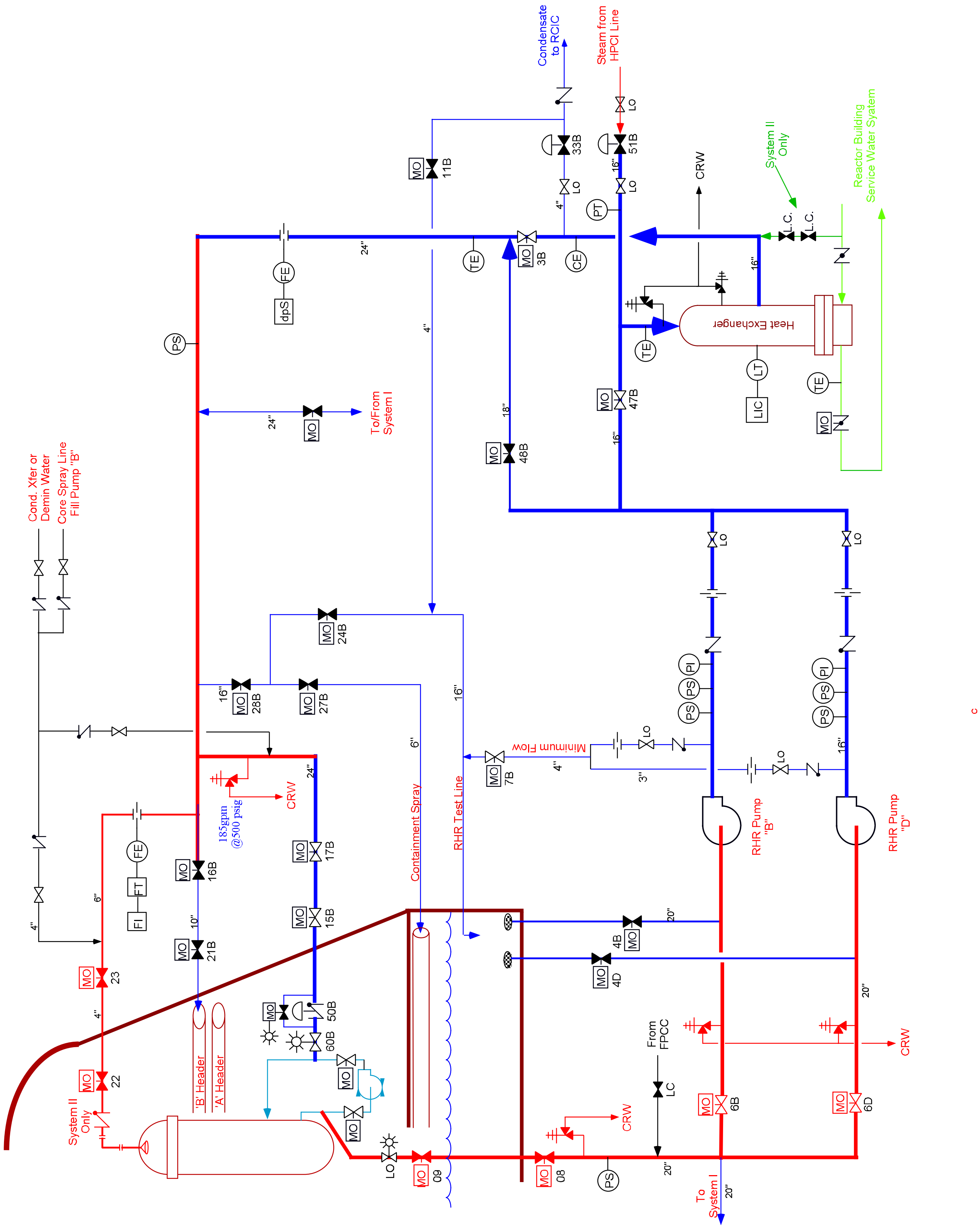

Figure 4.7-2 RHR System Shutdown Cooling Mode

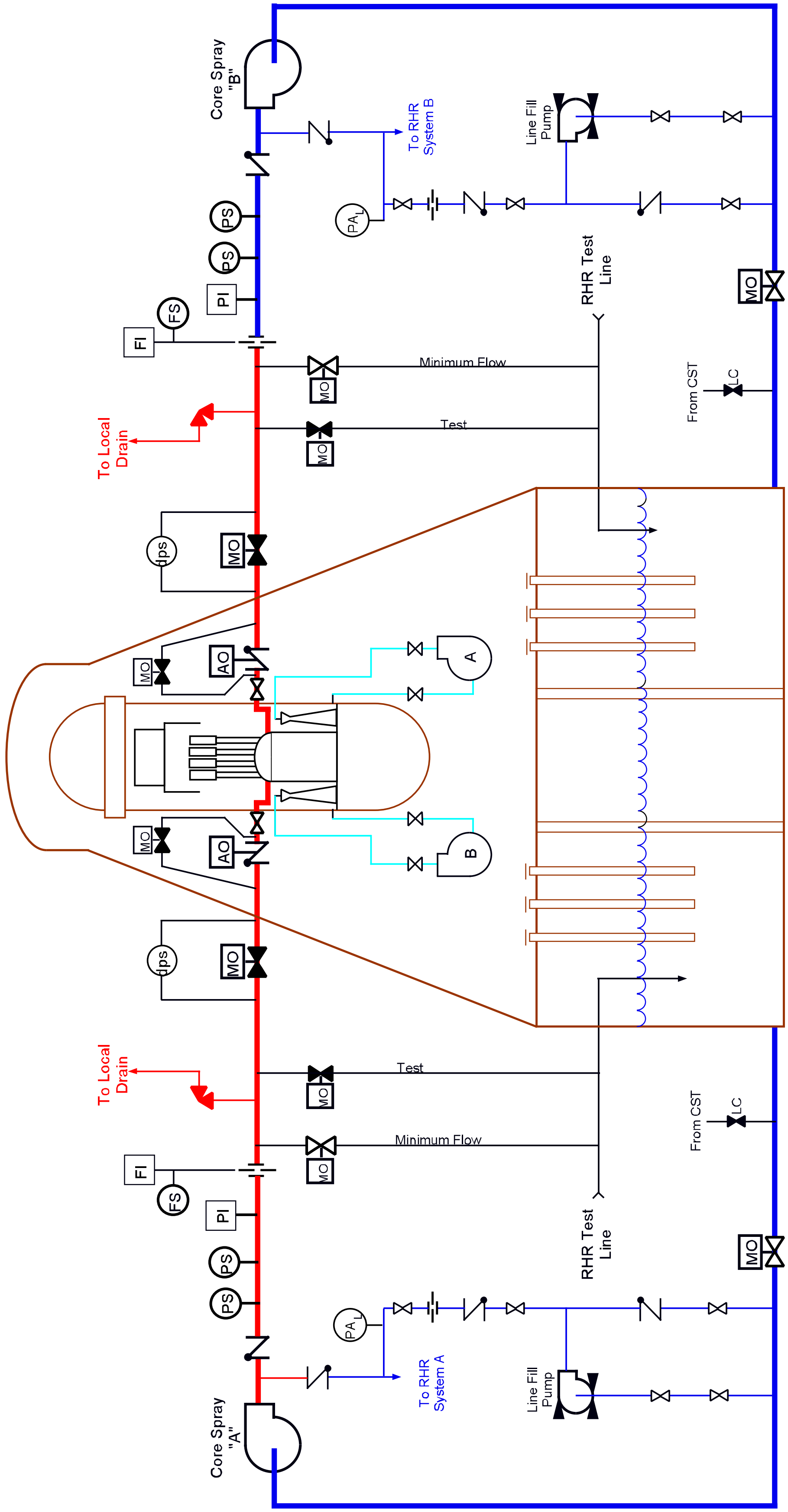

Figure 4.7-3 Core Spray Systems

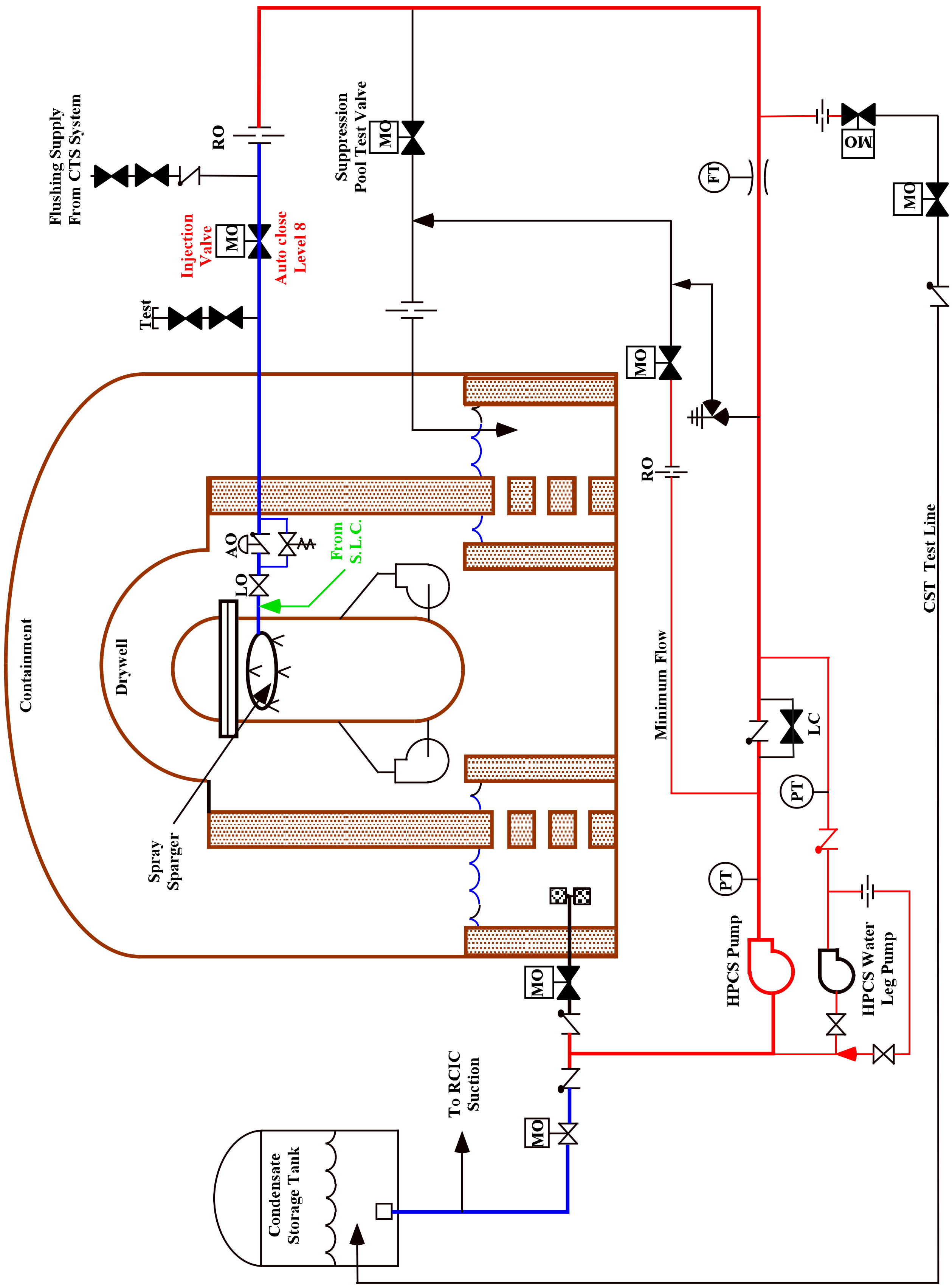

Figure 4.7-4 High Pressure Core Spray

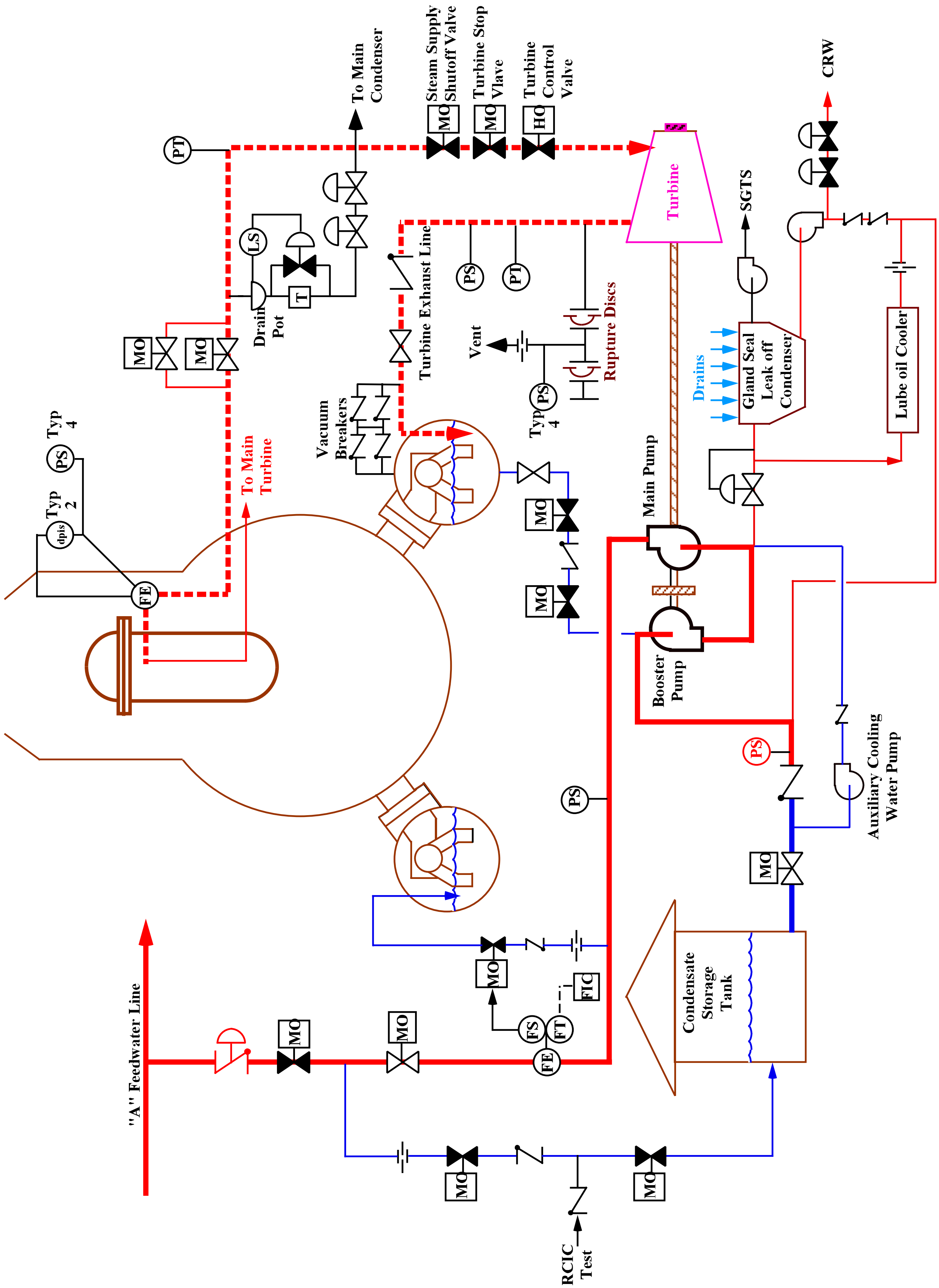

Figure 4.7-5 Dresden High Pressure Coolant Injection System