Pressurized Water Reactor
B&W Technology
Crosstraining Course Manual

Chapter 4.0

Emergency Core Cooling System

And

Nuclear Service Water System

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Learning Objectives:

1. List the purposes of the following Emergency Core Cooling Systems (ECCS):

   a. Decay Heat Removal (DHR) system
   b. High-pressure injection (HPI) system
   c. Core flooding tank system

2. Describe how the DHR system is used to remove decay heat from the core during the later stages of a plant cooldown.

3. State the source of cooling water to the DHR heat exchangers, and explain how the cooldown rate of reactor coolant is controlled.

4. Explain what cavitating venturis are and why they are used in the ECCS.

5. Explain the changes involved in converting the makeup and purification system into the high-pressure injection system.

6. List two accidents or malfunctions for which the high-pressure injection system is designed to provide core cooling.

7. Define the following terms:

   a. LOCA
   b. Blowdown phase
   c. Injection phase
   d. Recirculation phase

8. Explain the integrated operation of the ECCS for the conditions listed in Objective 7.

9. List the purposes of the Nuclear Service Water system (NSW) and the Shutdown Cooling Water system (SCW):

10. Identify the changes that occur in the NSW and SCW systems on the receipt of an engineered safety features actuation signal or loss of offsite power signal.

4.1 Emergency Core Cooling Systems Introduction

In a large, modern nuclear generating station with a rated core output of 3760 MWth, 7.92 lbs. 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 used.

The first barrier consists of 0.048-inch-thick Zircaloy-4 cladding, which surrounds the fuel pellets and is designed to contain the fission products. The second barrier is the reactor coolant system. This barrier is designed to withstand 2500 psig and temperatures up to 670°F. The thickness of this steel barrier ranges from the greater than 2-in. loop piping to the greater than 8-in. reactor vessel walls. Because this barrier surrounds the first barrier, it will contain any fission products that escape from the cladding. The final barrier 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 protection against the loss of each barrier are required by federal law. The mechanisms to effect reactor trips based on departure from nucleate boiling and local power densities (kw/ft) are installed to prevent cladding damage. The high-pressure reactor trip and code safety criteria limit reactor coolant system pressure to less than design pressure. Finally, design of the reactor building and allowable leakage specifications help to ensure building integrity.

Consider the effect on the cladding if a gross failure of the reactor coolant system occurs without the benefit of the emergency core cooling systems (ECCS). When this failure occurs, hot pressurized coolant is forced out of the reactor coolant system, and pressure decreases rapidly. An automatic reactor shutdown occurs because of the decrease in pressure, and the control rods fall into the core. The fission process is all but stopped, and heat production drops to the decay heat value. As the coolant rapidly escapes through the break, reactor building temperature and pressure begin to increase. In less than a minute the reactor coolant system has completely flashed to steam, and its pressure has equalized with the reactor building pressure. The blowdown phase of the loss of coolant has ended.

As the core begins to overheat, several processes come into play. First, with zirconium cladding along with steam in the system, the potential exists for metal-water reactions, which add another significant source term to the core heat up process. Also, hydrogen, a metal-water reaction byproduct, can burn when released to the reactor building and thus affect the building pressure. The core heatup can continue and eventually cause cladding melting followed by fuel meltdown.

When the cladding fails, fission products are released from the reactor coolant system to the reactor building through the break. Two of the three barriers have now failed, and the third barrier is threatened. However, the ECCS is installed in the nuclear units to mitigate the consequences of the loss of the reactor coolant system boundary. When system actuation occurs, water is pumped from a storage tank to the reactor core, and the system is refilled. This refilling of the core from the storage tank is called the injection phase of the loss-of-coolant accident (LOCA). With a flow of water into the core, decay heat is removed, and fission product release resulting from cladding failure is minimized.

However, a few major points concerning the ECCS should be made at this time. First of all, if only one pump is installed and it fails, the consequences of a LOCA are the same as those previously discussed. But if a redundant pump that is capable of supplying 100% of the required core cooling is installed, then the public can be protected. Criterion 35 of 10 CFR Part 50, Appendix A, requires redundant trains of ECCS.

Next, if one sensor is used to actuate the ECCS and it fails in a non-conservative direction, the core will not be refilled regardless of the number of ECCS pumps installed. If the sensor fails conservatively, unnecessary actuation of ECCS equipment occurs. Institute of Electrical and Electronics Engineers Standard IEEE-279 specifies that redundant sensors, as well as redundant instrument strings, logic devices, and actuated devices, be installed.

After protection is provided for the first two contingencies, a loss of power to the ECCS pumps must be considered. The normal power supply to the electrical buses, which powers the pump motors, comes from the utility's grid by 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 power system is provided to ensure a power supply to the ECCS pump motors. Redundant diesel generators are normally used as the standby power supply. Criterion 17 of 10 CFR Part 50, Appendix A, outlines the requirements for electrical power.

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

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

Now that the LOCA and a few of the important design considerations for the equipment installed to mitigate the consequences of the accident have been discussed, the remainder of this section will deal with the Babcock & Wilcox emergency core cooling systems.

In addition to the ECCS, the engineered safety features design also includes provisions to protect the reactor building barrier and remove the reactor's decay heat in the event of a loss of normal feedwater.

Two active systems and one passive system are installed to provide emergency core cooling in the case of an accident. These systems are the high-pressure injection system, the low-pressure injection system, and the core flooding tanks.

1. The high-pressure injection system, consisting of two trains, provides protection for small to intermediate LOCAs. The system pumps water from the borated water storage tank (BWST) during the injection phase, and provisions are installed to allow the use of the system during the recirculation phase.

2. The low-pressure injection system also consists of two trains, and it pumps water from the BWST to the reactor vessel in the event of a large LOCA. The low-pressure injection pumps are supplied with suction from the DHR recirculation sump during the recirculation phase, and in addition to supplying fluid to the reactor vessel, the pumps also supply the high-pressure injection system.

3. The core flooding system consists of two tanks filled with borated water and pressurized with nitrogen. When the unit is at power, check valves prevent the entry of water from the tanks into the reactor coolant system. However, if a LOCA results in a reactor coolant system pressure lower than core flooding tank pressure, then flow from the tanks to the reactor vessel will occur. This system, unlike the high- and low-pressure injection systems, requires no actuation.

When the reactor trips, the rapid insertion of the control rods stops the heat production resulting from 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 once-through steam generators; however, the normal feedwater system does not meet safety system design criteria. If this system is lost, the steam generators can boil dry. When the secondary inventory is lost, the decay heat is not removed. As fission product decay continues, the resultant heat production causes an increase in reactor coolant system temperature and a corresponding increase in system pressure. Left unchecked, the temperature and pressure increases can result in the challenging of the reactor coolant system safety valves. The auxiliary (emergency) feedwater system is designed to provide feedwater to the steam generators to remove decay heat in the event of a loss of normal feedwater. The auxiliary feedwater system meets the design criteria for the emergency core cooling system and is discussed in chapter 5.0.

The emergency systems installed to mitigate the consequences of accidents are among the most important systems in the nuclear station. Normally the systems remain in emergency injection alignment until called upon to perform their accident functions.

4.2 Decay Heat Removal System

4.2.1 System Description

The Decay Heat Removal (DHR) system, as shown in Figure 4-1, consists of two, one-hundred percent redundant trains. Each train alone can meet the minimum requirements for maintaining emergency core cooling during a loss-of-coolant accident. Each train consists of a single-stage centrifugal pump; a heat exchanger cooled by the shutdown cooling water system, and associated connecting piping and valves.

The supply of water to the DHR pumps is dependent on the mode of operation of the system. The supply from the reactor coolant system hot-legs is used during normal plant cooldown. The DHR recirculation sumps and the borated water storage tank (BWST) will be employed during emergency core cooling system operations.

The DHR pumps discharge to the heat exchangers which are cooled by the shutdown cooling water system. For most operations the water will then pass into the reactor building and be injected into the DHR/core flood tank nozzle connections on the reactor vessel. The remaining discharge flowpaths will be discussed in their associated sections.

4.2.2 Component Description

4.2.2.1 Decay Heat Removal/Low Pressure Injection Pumps

The decay heat removal pumps are single-stage centrifugal pumps, which are driven by 700-hp, 4160-Vac electric motors. Because these pumps provide emergency core cooling, redundancy is required. The A decay heat removal pump is powered from one vital ac bus, and the B decay heat removal pump is powered from the opposite vital ac bus. Each pump will provide approximately 5000-gpm flow at a discharge pressure of 167 psig. Figure 4-2 provides a pump head curve for the pumps. During the recirculation phase of a LOCA, the low-pressure injection pumps are required to pump highly radioactive fluid. Mechanical seals installed on the pump shaft minimize the escape of the contaminated sump water into the auxiliary building.

4.2.2.2 Decay Heat Removal Coolers

The decay heat removal (DHR) coolers are of the tube and shell design, with the decay heat removal fluid flowing through the tubes and shutdown cooling water (SCW) flowing on the shell side. The DHR system is designed to reduce reactor coolant system temperature from 305°F to 140°F (refueling temperature) within 14 hours if both coolers are operable. If only one cooler is operating, then a longer period of time is required to reach 140°F. These times are based on a shutdown cooling water temperature of 105°F, which is conservatively high. The actual plant cooldown times will be shorter than those listed above.

4.2.2.3 Cavitating Venturis

A cavitating venturi is a passive flow-restricting device requiring no controls or instrumentation for proper operation. Flow through a low-pressure injection system rupture downstream of the venturi is limited by cavitation. For the design cavitation conditions, the velocity head at the throat is equal to the difference between the total inlet pressure head and the vapor pressure; the fluid will then reach its vapor pressure in the throat and flash, thus limiting flow. Delivery of a sufficient portion of the available low-pressure injection pump flow rate to the reactor vessel is accomplished through the intact line. Manual bypass valves are opened during decay heat removal operations. A cavitating venturi is shown in Figure 4-3.

4.2.3 System Operations

4.2.3.1 Decay Heat Removal Operations

After the RCS temperature has been reduced to 305°F and pressure decreased to approximately 400 psig, the decay heat removal system is placed in service to reduce temperature to the cold shutdown value. The core's decay heat is removed by circulating reactor coolant from the hot leg through the decay heat coolers and injecting the cooled fluid back into the reactor vessel through the low-pressure injection/ core flood tank nozzles.

The DHR system taps off loop A hot-leg piping and is connected to the suction of the decay heat removal pumps. As shown in Figure 4-1, the decay heat removal piping splits into individual pump suction supplies, with each supply containing three motor-operated isolation valves in series.

The motor-operated isolation valves (V-23A, V-25A, and V-31A for the A DHR pump, and V-24B, V-26B, and V-32B for the B DHR pump) serve as isolation valves for the decay heat suction piping. The first two valves (V-23A, V-25A, V-24B and V-26B) in each of the suction lines are interlocked with reactor coolant system pressure. These valves cannot be opened unless pressure is less than 400 psig and will automatically close if pressure exceeds 400 psig. Wide-range pressurizer pressure instrumentation provides interlocking signals for valves V-25A and V-26B. Redundancy in DHR suction isolation is ensured by closing V-23A and V-24B with signals supplied by the engineered safety features actuation system. Because the two series valves in one line have their high-pressure isolation signals generated by separate transmitters, isolation of the decay heat suction lines is ensured in the event of single instrument failure.

After passing through the series motor-operated valves and the various suction line penetrations, water from the reactor coolant system enters the suction of the DHR pumps. Manual cross-connect valves allow either pump to use either suction line.

From the DHR pumps, the fluid travels to the tube side of the decay heat removal coolers where shutdown cooling water, flowing on the shell side of the decay heat removal coolers, removes the energy from the reactor coolant system water. Temperature and cooldown rates are controlled by the amount of reactor coolant flow that is directed through the DHR cooler. This amount of flow, in turn, is determined by the position of the cooler outlet valves (FCV-3A for the A DHR cooler, FCV-3B for the B DHR cooler) and the cooler bypass valves (FCV-14A for the A DHR cooler, and FCV-14B for the B DHR cooler). Hand-indicating controllers located in the control room allow operator control of these valves. From the cooler outlet, the fluid is directed back to the reactor vessel through motor-operated injection valves (V-73A and V-74B).

4.2.3.2 Auxiliary Functions

In addition to the heat removal function, the DHR system also supplies auxiliary pressurizer spray. The auxiliary spray piping consists of a cross connection between the two decay heat removal trains and a connection to the normal pressurizer spray line. An input to the auxiliary spray line from the makeup and purification system allows the makeup pumps to provide auxiliary spray at high reactor coolant system pressure.

Additional connections to the makeup and purification system enable flow through the ion exchangers and filters when purification of the reactor coolant system is desired during cold shutdown. The supply path branches from the auxiliary spray line, and the return path connects to the DHR pump suction lines.

4.2.3.3 Emergency Core Cooling

1. Injection Phase

Several modifications to the decay heat removal flowpath are required to align the decay heat removal system for its low-pressure injection function. First, as shown on Figure 4-4, the series motor-operated suction isolation valves from the A hot-leg are closed. The BWST outlet valves, V-7A and V-8B, are opened to supply the DHR pumps with a source of injection water. Second, the manual isolation valves are repositioned. This consists of opening the manual isolation valves in the suction header from the BWST and closing the manual isolation valves that supply auxiliary spray. Next, the cooler outlet valves are opened fully, and their power supplies are de-energized. The cooler bypass valves are electrically inhibited closed to ensure that all emergency core cooling flow is through the DHR coolers. The motor-operated injection valves are left open; the series check valves downstream of the injection valves prevent backleakage from the reactor coolant system into the DHR system. The realignment of the DHR system is performed after the reactor coolant pumps are started for plant heatup and before Mode 3 is reached.

If an engineered safety features signal is received, the DHR pumps start; however, during the initial phases of an accident, reactor coolant system pressure is usually greater than the decay heat removal pump discharge pressure, and no flow will enter the system. In this situation a recirculation path is provided to prevent pump damage. The path consists of a recirculation line from the outlet of each DHR cooler through an orifice and back to the suction of the pump. The cooled BWST water prevents pump overheating, and the orifice prevents excessive flows that could affect the low-pressure injection delivery rate. When reactor coolant system pressure drops below the discharge pressure of the DHR pumps (~200 psig), low-pressure injection flow enters the system.

The accident flow path is from the discharge of the pumps to the reactor vessel low-pressure injection nozzles via the low-pressure injection valves (V-73A and V-74B). Cross-connections are installed between the low-pressure injection pump discharges to ensure flow through both injection paths even if only one pump starts. Cavitating venturis located in each cross-connecting line (downstream of V-71A and V-72B) and in each pump discharge line (upstream of V-3A and V-3B) are installed to limit flow in the event of a passive low-pressure injection header break. If both trains of low-pressure injection are operable, then each injection path will supply approximately 5000 gpm of flow for emergency core cooling. The system will continue to supply core cooling during the injection phase until a low level occurs in the BWST.

2. Recirculation Phase

When the engineered safety features actuation system (ESFAS) senses a low BWST level, an automatic realignment occurs; the DHR recirculation sump isolation valves (V-1A and V-2B) open to provide a source of suction to the low-pressure injection pumps, and the BWST outlet valves (V-7A and V-8B) may be closed by the operator to isolate the almost emptied tank. The low-pressure injection/high-pressure injection cross-connect valves (V-63A and V-64B) may be opened to supply a sump suction source to the high-pressure injection pumps. The automatic realignment places the low-pressure injection system in the recirculation mode of operation. During the recirculation phase of the LOCA, hot DHR recirculation sump water is supplied to the suction of the low-pressure injection pumps. This fluid is discharged by the pumps to the DHR coolers, where its temperature is reduced by the shutdown cooling water system. The cooled water is then injected into the vessel by means of the low-pressure injection valves (V-73A and V-74B) to remove the heat from the reactor. The flow path from the pump discharge into the reactor vessel during the recirculation phase is identical to the flow path used during the injection phase.

In the event of a LOCA that has not depressurized the reactor coolant system to a pressure less than low-pressure injection pump discharge pressure; the low-pressure injection pumps supply the high-pressure injection system with coolant from the DHR recirculation sump. During this operating mode, the low-pressure injection pumps serve as booster pumps for the high-pressure injection system. When pressure is lowered to below the low pressure injection pump discharge, and the flow rate into the vessel is greater than the high-pressure injection pump design flowrate (~700 gpm), the cross-connect valves may be closed and the high-pressure injection pumps stopped.

3. Long-Term Core Cooling

The low-pressure injection system functions to remove decay heat from the core during the recirculation phase of a LOCA. Steam formation in the core can increase the boron concentration in the remaining liquid-phase coolant to the solubility limit for the existing RCS temperature. Exceeding the solubility limit results in boron precipitation (boron plateout), which can block flow channels and retard heat transfer. To minimize the probability that this phenomenon occurs, the vessel boron concentration must be diluted before the boron concentration reaches the solubility limit.

One of a few plant-dependent flowpaths can be implemented to dilute the vessel boron concentration. First, the discharge of one of the two decay heat removal (DHR) pumps may be aligned to the auxiliary pressurizer (PZR) spray connection. This alignment introduces reversed flow through the core via the PZR surge line, associated hot leg, and hot leg nozzle, thus reducing the boron concentration. The second possible flow path is from the discharge of one of the two DHR pumps back through the suction of an idle pump and into the normal DHR suction line. This hot leg injection path also provides reversed flow through the core, with a much higher flow rate than that of the auxiliary PZR spray flow path. The final flow path is called “dump to the sump” and includes valves downstream of the first decay heat suction isolation valves, which empty to the DHR recirculation sumps. These dump-to-the-sump valves are opened within 24 hours after a LOCA. This alignment prevents boron buildup by ensuring a low-resistance path for the continuous flow of emergency core cooling water through the core region and out of the hot leg.

4.3 High-Pressure Injection System

4.3.1 System Description

The high-pressure injection system, shown in Figure 4-5, consists of two redundant flow paths that supply borated water to the reactor coolant system in the event of an accident. The high- pressure injection system is designed to provide sufficient core cooling for breaks ranging in size from 3/4-inch diameter (0.009 ft^2) to 3-inch diameter (0.05 ft^2) and to delay the uncovering of the core for intermediate-size breaks.

The high-pressure injection system takes a suction from the BWST during accidents. The suction for the A train is supplied through motor-operated valve V-144A, and the suction for the B train is through motor-operated valve V-141B. This suction source is used during the injection phase of the loss-of-coolant accident. Suction for the recirculation phase is supplied by the discharge of the low-pressure injection pumps via motor operated valves V-63A and V-64B.

From either suction source, the emergency core cooling fluid travels to the high-pressure injection (makeup) pumps. The high-pressure injection pumps are motor-driven, 11-stage, centrifugal pumps, which are powered from separate vital ac buses. Pump A is powered from 4160-vac bus EA, pump B is powered from 4160-vac bus EB, and pump C may be selected to either bus. The separated power supplies ensure that at least one pump is available for high-pressure injection even if a loss of offsite power occurs and one emergency diesel fails to start. This condition is consistent with the plant safety analysis, which assumes the availability of only one emergency core cooling system train. Further conservatism is added by assuming that the operable high-pressure injection pump delivers 90% of design flow during accident conditions, with 30% of this flow being lost during a postulated cold-leg break.

Each of the high-pressure injection pumps discharge to two injection paths through motor-operated isolation valves. The A train discharges through V-179A and V-174A, and the B train discharges through V-184B and V-185B.

Cavitating venturis, located in each injection flowpath, are designed to limit high-pressure injection flow in the event of an injection line break and provide a pressure drop that is used for flow indication. The venturis are located upstream of the injection isolation valves. The isolation valves are designed so that the operator may throttle the amount of injection flow being supplied by each train.

High-pressure injection header cross-connections are located downstream of the motor-operated isolation valves. These cross-connections and the cavitating venturis allow for single-failure conditions of the high-pressure injection system as described in the following paragraphs:

1. If a failure of bus EA is assumed, the A pump and injection valves will be inoperable. The B train pump and valves will provide 100% of the assumed flow through all four headers by way of V-184B and V-185B. Conversely, if bus EB is assumed lost, high pressure injection flow will be supplied by the A train pump and valves through the cross-connections to all four injection points.

2. If the loss-of-coolant accident is caused by a break in the high pressure injection piping and one train of high pressure injection is lost, then the remaining high pressure injection pump will deliver 70% of the design flow, with the cavitating venturis limiting break flow to 30%.

The high-pressure injection piping penetrates the reactor coolant system piping at the discharge of each reactor coolant pump. Each penetration is supplied with a thermal sleeve to minimize stresses.

4.3.2 System Operations

4.3.2.1 High-Pressure Injection Sequence

The high-pressure injection system is automatically actuated by ESFAS if reactor coolant system pressure drops to 1600 psig or if reactor building pressure increases to 4 psig. When any one of these signals is sensed, the following actions occur:

1. The BWST suction valves (V-144A and V-141B) open to supply an ECCS suction source.

2. The makeup tank outlet valves (V-136A and V-137B) close.

3. Two high-pressure injection pumps start (one per train).

4. High-pressure injection pumps (HPI) discharge motor-operated isolation valve, V-380A, closes. Valve V-381B remains open to supply seal injection from the B-HPI pump.

5. Pump recirculation valves (V-198A and V-199B) close. This ensures that all flow from the pumps enters the reactor coolant system.

6. Makeup is isolated to prevent high-pressure injection pump runout.

7. Reactor coolant pump seal return is isolated. The closing of V-283B accomplishes this isolation. This prevents potentially highly radioactive reactor coolant system fluid from entering the auxiliary building.

8. The high-pressure injection isolation valves (V-174A, V-179A, V-184B, and V-185B) open to supply emergency core cooling water from the BWST to the reactor coolant system.

9. The letdown isolation valves (V-3A, V-4A, and V-5B) close.

4.3.2.2 Small-Break Spectrum

As previously stated, the high-pressure injection system is designed to provide sufficient core cooling for small breaks ranging in size from 0.009 ft^2 to 0.05 ft^2. Several restrictions must be added to this statement. First, breaks less than 0.009 ft^2 are within makeup system capacity and, therefore, are not defined as loss-of-coolant accidents by 10 CFR 50. Second, sufficient core cooling means that the high-pressure injection system provides enough flow to prevent cladding temperature from exceeding pre-break values (~700°F). Finally, sufficient core cooling does not imply that the low-pressure injection system will not eventually pump fluid into the core. With these restrictions in mind, the spectrum of leaks for which the high-pressure injection system will provide protection includes the following:

1. Pressurizer safety valve failure.
2. Pressurizer power-operated relief valve failure.
3. A double-ended rupture of a single steam generator tube.

4.3.2.3 Steamline Breaks

On a large steamline break, the steam generator depressurization leads to an overcooling of the reactor coolant system. The overcooling causes reactor coolant system contraction and a drop in system pressure. Actuation of the high-pressure injection will occur, and the high-pressure injection system will provide borated water that will aid in reactor coolant system pressure recovery once the steam break has been terminated. Analyses show that system pressure will not decrease below core flood tank pressure or low-pressure injection pump discharge pressure.

4.4 Core Flood Tank System

4.4.1 System Description

The core flooding tanks (CFTs) (Figure 4-6) are filled with borated water (~2000 ppm) and pressurized to 600 psig from the station nitrogen supply. When the unit is at normal operating pressure, the 2195 psig reactor coolant pressure seats check valves V-35A and V-36B and prevents the entry of reactor coolant into the core flood tanks. Check valves V-33A and V-34B provide double isolation of the CFTs from the reactor coolant system. Motor-operated isolation valves, V-31A and V-32B, are open with power removed from the valves during power operations.

If reactor coolant pressure drops below 600 psig, the nitrogen pressure in the core flood tanks forces borated water out of the tanks and into the reactor vessel. Since an actuation signal is not required to cause the system to function, it is called a passive system.

4.4.2 Core Flooding Tanks

The core flooding tanks are vertical cylindrical vessels with elliptical heads. The tanks are constructed of carbon steel, and the internal surface of each tank is clad with stainless steel. A 14-in. outlet nozzle penetrates the lower elliptical head on each tank.

Each core flooding tank has a total volume of 1800 ft^3 and contains 1350 ft^3 of 2270-ppm borated water. The tanks are filled and pressurized through a common line to minimize tank penetrations. The source of makeup water comes from the makeup and purification system; the N_2 supply comes from the station nitrogen banks. The tank relief valve is set at 700 psig, and is sized to relieve the pressure associated with the maximum makeup to the tank.

The tank vent valve is used to depressurize the tank. Electrical power is removed from the vent valve after the valve is closed to prevent core flooding tank inoperability by accidental depressurization.

Redundant tank pressure and level transmitters provide indication and alarm functions for each core flooding tank in the main control room. A sample line allows the periodic sampling of CFT boron concentration.

4.4.3 Motor-Operated Isolation Valves

Motor-operated isolation valves (V-31A for CFT A and V-32B for CFT B) are installed in the outlet lines. These valves are used to isolate the associated core flood tanks when reactor coolant pressure is below tank nitrogen pressure. This situation occurs as the unit is placed in a cold shutdown condition and reactor coolant system pressure is decreased. As reactor coolant system pressure is reduced during a cooldown, an alarm is generated if system pressure is less than 650 psig and the isolation valves are open. Conversely, if reactor coolant system pressure is being increased during a plant heatup and the CFT isolation valves are closed, the valves receive an open signal when reactor coolant system pressure reaches 750 psig. These interlocks originate in the essential controls and instrumentation (ECI) system, with pressure inputs provided by the wide-range pressurizer pressure detectors. In addition, ESFAS sends an open signal to these valves during accident situations. However, these valves are opened with power removed from the motors when the reactor coolant system pressure is above 700 psig. Power is removed because the motor-operated isolation valves will not meet ECCS single-failure criteria.

4.5 Integrated Emergency Core Cooling Systems Operations

Each emergency core cooling system has been discussed in a separate section of this chapter. Section 4.2 described the low-pressure injection system, Section 4.3 discussed high-pressure injection, and Section 4.4 covered the core flood tank system. This section is devoted to the integrated operations of these systems. In the discussion of system operation, it is assumed that a loss of offsite power occurs with the accident.

4.5.1 Loss-of-Coolant Accident

4.5.1.1 Injection Phase

Before a loss-of-coolant accident (LOCA), the emergency core cooling systems are aligned as follows:

1. The high-pressure injection system provides normal reactor coolant system makeup from the A train pump, and the B train pump will be aligned to function as either a standby seal injection supply or as an ECCS pump.

2. The low-pressure injection system suction valves from the BWST are open, and the motor-operated injection isolation valves are open.

3. The core flooding system is pressurized to 600-psig N_2 pressure, with reactor coolant system pressure seating the outlet check valves.

When a break occurs in the reactor coolant system, either a decrease in reactor coolant system pressure or an increase in reactor building pressure will result in actuation of the emergency core cooling systems by the engineered safety features actuation system. Regardless of the specific actuation signal, both the high-pressure injection and low-pressure injection systems will respond.

The high-pressure injection system will start, after a time delay resulting from diesel generator sequencing, and supply injection through the four injection paths to the reactor coolant. As system pressure continues to drop, the nitrogen pressure in the core flooding tanks will force borated water into the reactor coolant system. When reactor coolant system pressure drops below the discharge pressure of the low pressure injection pumps, flow from the low pressure injection system will enter the reactor vessel.

Figure 4-7 shows the ECCS response to a small break loss of coolant accident (SBLOCA) equivalent to a break of 0.44-ft^2 (9-in. dia.) break. This is the same size as a core flood nozzle break. As shown in Figure 4-7, high-pressure injection flow starts at 25 sec. High-pressure injection is the only source of emergency core cooling until pressure drops below the core flooding tank (CFT) nitrogen pressure. Once this happens, CFT flow starts. The flow rate from these tanks is dependent on the differential pressure between tank nitrogen pressure and reactor coolant system pressure.

When reactor coolant system pressure decreases below low-pressure injection pump shutoff head, flow from this system enters the core. Before this the low-pressure injection (LPI) pumps were recirculating from the decay heat cooler outlet back to their suctions. For this particular break size, the 31-sec time delay for LPI pump start is insignificant. In accidents involving larger breaks, adequate core cooling is achieved even though 31 seconds elapse before the low pressure injection pumps start. With the low-pressure injection flow starting at approximately 255 seconds, all three emergency core cooling systems are providing flow. Table 4-1 shows values of flow versus reactor coolant pressure for all three systems.

During the injection phase, both the high- and low-pressure injection systems take a suction on the borated water storage tank (BWST) to provide core cooling. The BWST contains a usable volume of about 680,000 gal of borated water (~2270 ppm) that will be pumped into the reactor coolant system during the accident. At the maximum drawdown rate of about 20,000 gpm (assumes full ECCS and spray), this volume of borated water will supply the engineered safety features pumps for about 30 minutes for this size SBLOCA. At the end of this time period, the BWST level will have dropped to the recirculation switchover point. Figure 4-8 shows a sketch of BWST volumes and setpoints.

4.5.1.2 Reactor Internals Vent Valves

Eight reactor internals vent valves (Figure 2.1-5) are installed in the core support cylinder in Babcock and Wilcox designed reactor internals. In the event of a cold leg pipe rupture, the valves prevent a pressure buildup above the core so that emergency core coolant can be injected. The vent valves are required because the arrangement of the RCS hot leg piping can possibly inhibit the free venting of steam generated in the core after the system is depressurized, if significant quantities of coolant remain in the reactor inlet piping at the end of the blowdown period. Without free venting of the steam, a pressure differential could exist between the core region and the reactor vessel inlet annulus region where emergency coolant is injected. This pressure differential would inhibit flow into the core.

To eliminate the problem, eight vent valves are installed in the core support cylinder to provide a flow path from the region above the core directly to the reactor vessel inlet annulus region (the pipe rupture location). The flowpath provides for pressure equalization and permits emergency coolant water to reflood the core rapidly. Figures 2.1-3 and 2.1-16 show the locations of the internals vent valves. Additional information on the reactor internals vent valves is in Chapter 2.1, Core and Vessel Construction.

4.5.1.3 Recirculation Phase

When the BWST level reaches the switchover setpoint (see Figure 4-8) ESFAS automatically initiates the shifting of low-pressure injection and reactor building spray pump suctions from the BWST to the DHR recirculation sumps and reactor building spray sumps respectively. The DHR recirculation sump valves (V-1A and V-2B) are commanded to open, and the BWST outlet valves (V-7A and V-9B) may be closed by the operator. Check valves installed in the suction header prevent the transfer of BWST fluid to the sump and vice versa. It takes about 100 seconds for the sump suction valves to stroke open. In addition to the suction valve position changes, the low-pressure injection (LPI) discharge may be aligned to provide a suction to the high-pressure injection (HPI) system. This is accomplished by opening the LPI/HPI cross-connection valves (V-63A and V-64B) with a switchover signal.

Once the switchover is completed, core cooling is accomplished by pumping borated water from the DHR recirculation sump through the decay heat removal coolers and back into the system. The temperature of the recirculated fluid is reduced by the shutdown cooling water system.

The recirculation phase of the loss-of-coolant accident involves the circulation of highly radioactive water through components located in the auxiliary building. This represents a potential release path to the public. The HPI and LPI pumps are equipped with mechanical seals to minimize leakage. Activity associated with valve stem leakage is minimized by using valves with back seats and collecting and routing gland leakoff to waste systems.

4.5.1.4 Long-Term Core Cooling

After a large loss-of-coolant accident, the reactor vessel, in effect, becomes a boric acid concentrator. As the emergency core coolant is boiled, the boric acid is concentrated and plates out on the fuel pins. When this occurs, the heat transfer capability from the fuel is decreased. This phenomenon is known as boron precipitation. If a positive flow can be established through the core, boron precipitation can be minimized. Any of the three flow paths listed in section 4.2.3.3 may be used to prevent boron precipitation.

4.5.2 Core Flood Nozzle Break

The ability to provide adequate core cooling in the event of a core flood/LPI nozzle break has been analyzed. In this analysis a break in one nozzle is assumed as the initiating event, and a failure of one of the LPI pumps is the assumed single failure. During this hypothetical break, the LPI cross connects and cavitating venturis effect a flow split as shown in Figure 4-9. The flow split has been shown to provide adequate core cooling. This break is a 9-in. diameter break, and system pressure will respond as shown in Figure 4-7.

4.5.3 PRA Insights

Since the emergency core cooling systems are required to mitigate the consequences of accidents, the failure of these systems during transients and accidents greatly increases the probability of core melt. In particular, the failure of the high pressure injection system in either the injection phase or the recirculation phase of operation is a major contributor to increases in core melt frequency. According to the Arkansas Nuclear One Unit 1(ANO1) PRA, the high pressure injection system failure contribution to core melt frequency is ~80%. The Crystal River Unit 3 PRA shows a similar contribution. Many of the failure sequences are caused by operator error. At ANO1, the low pressure injection system failures contribution to core melt frequency is ~18%. The following sequences provide information concerning system failures and their consequences.

For certain small break sizes (<1.2 inches in diameter), it is postulated that an engineered safety features actuation signal may not be generated until core uncovery begins. In this scenario, a small leak occurs, and the operating makeup pump (i.e. HPI pump) maintains pressurizer level above the heater cutoff setpoint; this will maintain reactor coolant system pressure above the ESF initiation setpoint.

As the leak progresses, the makeup tank is emptied. When the tank is emptied, suction to the operating makeup pump is lost. The loss of suction leads to pump failure. The sequence also assumes that the operator fails to manually initiate HPI. With no makeup or ECCS flow, the core uncovers and overheats.

Even for break sizes that are large enough to actuate the engineered safety features system on low reactor coolant system pressure, the proper operation of all systems is very important. A scenario that increases core melt frequency is outlined below:

1. A LOCA occurs that results in the actuation of the ECCS systems.
2. All ECCS systems function properly in the injection phase of the LOCA.
3. The BWST reaches a low level, and the plant operators switch ECCS suctions to the DHR recirculation sump incorrectly.
4. The incorrect sump suction lineup results in a loss of suction to the ECCS pumps, and no core cooling flow is available.
5. With no cooling flow, the core heats up and melts.

A third sequence that illustrates the importance of ECCS systems to risk is the inter-system or interfacing LOCA (also known as Event V). In this sequence, a failure of both of the series check valves that separate the high pressure reactor coolant system from the low pressure LPI/CFT systems is postulated. With the motor-operated valves in the open position, the low pressure injection discharge piping is exposed to full reactor coolant system pressure. The discharge piping is overpressurized and fails.

The failure results in a loss of coolant accident with some unique features. First, the LOCA is outside of the reactor building; therefore, the containment barrier is lost. Next, the assumed failure has the potential to also fail both trains of low pressure injection. Finally, the leakage from the reactor coolant system is not collected in the DHR recirculation sump. When the BWST empties, no cooling fluid is available for long term core cooling. With no cooling, the core overheats.

Additional causes of failures of ECCS systems are:

1. HPI fails to operate in the injection mode.
2. HPI fails to operate in the recirculation phase.
3. Failure of the BWST suction valve(s) to open.
4. Pipe failures in the HPI and LPI system.

The largest ECCS risk reduction factors are associated with the BWST suction supply valves to the LPI system. The risk reduction factor for the valves is approximately 1.10. Increasing the reliability of the HPI system has a risk reduction factor of 1.01. The largest risk achievement factors are associated with faults in LPI and HPI piping. The risk achievement for LPI piping faults is 401, while the risk achievement factor for HPI piping faults is 5.

4.6 ECCS Summary

The emergency core cooling systems consist of the high-pressure injection system, the core flood tank system, and the low-pressure injection system. The high pressure injection (HPI) system provides core cooling during small- to intermediate-size breaks. The core flood tank system is a passive system that provides cooling when reactor coolant system pressure drops below core flooding tank nitrogen pressure. The low-pressure injection (LPI) system provides core cooling during large-break LOCAs.

Borated water for the high- and low-pressure injection systems is supplied from the borated water storage tank (BWST). Water from this tank is pumped to the core during the injection phase of the accident. An automatic LPI suction shift to the DHR recirculation sump occurs when the BWST reaches a low level. Valves on the LPI pump discharge are also opened to provide suction for the HPI pumps during the recirculation of water from the sump to the core.

4.7 Nuclear Service Water System

4.7.1 Introduction

The nuclear service water (NSW) system transfers heat from safety-related components to the normal heat sink (cooling towers) during normal operations and to the ultimate heat sink (spray pond) during emergencies. The NSW system normally receives its water supply from the service water system (SWS). However, during emergencies, the NSW system is supplied by the emergency shutdown water (ESW) system.

Two completely redundant trains of NSW are installed. Each of the trains removes heat from one-half of the safety-related equipment. A single failure of any component in one NSW train will not result in a loss of cooling capability in the redundant train.

The ESW system provides cooling water from the ultimate heat sink (spray pond) to the NSW system during plant emergencies. The ESW system consists of two 100% redundant trains, each serving one of the NSW trains.

The shutdown cooling water (SCW) system is a subsystem of the NSW system and provides cooling water to the decay heat removal system and reactor building spray system during normal and emergency operations. The SCW system is a closed-loop cooling water system that provides a barrier between the radioactive components that it cools and the non-radioactive NSW system. Heat exchangers in the SCW system are cooled by the NSW system.

4.7.2 Nuclear Service Water System Description

The NSW system is essentially a piping system that supplies cooling water to the following components and systems:

1. Diesel generators
2. Makeup (HPI) pumps
3. Nuclear instrument air compressors
4. Component cooling water (CCW) heat exchangers
5. Shutdown cooling water (SCW) heat exchangers

As shown in Figure 4-10, the NSW system consists of redundant supply and return headers. Approximately one-half of the safety-related equipment is supplied from each header. The C high pressure injection pump may be manually aligned to either train’s supply and return header.

During normal operation, the NSW system is supplied with cooling water from the service water system (SWS). The cool water from the discharge of the SWS pumps is routed to the NSW supply header; through the NSW supplied components; where it removes heat, and then to the NSW return headers. From the NSW return headers, the flow is returned to the cooling towers via the SWS return header. The cooling towers cool the water by evaporation.

In emergencies, with either an ESFAS or a loss of offsite power (LOOP) signal, the NSW system receives its cooling water from the emergency shutdown water (ESW) system. The ESW system consists of two redundant pumps that take a suction from the spray pond and deliver the cool water to each NSW train. The NSW return headers are realigned to the spray pond. The spray pond removes heat from the cooling water by evaporation.

4.7.3 Nuclear Service Water System Component Description

4.7.3.1 Emergency Shutdown Water Pumps

Each of the ESW pumps is a vertical, single-stage, turbine pump that supplies a rated flow of 23,400 gpm. The head of the pumps at rated flow is 144 feet (62 psi). The pumps are driven by 1250-hp, three-phase, 4160-Vac, 60-hertz, induction motors. The ESW pumps are automatically started by an ESFAS or LOOP signal.

Emergency service water pump A supplies the NSW A train components, and ESW pump B supplies the NSW B train components. One NSW train of cooling is sufficient to mitigate the consequences of any accident.

4.7.3.2 Spray Pond

The spray pond has a capacity of 13,000,000 gallons and serves as the ultimate heat sink for the unit in the event of emergencies. The capacity of the pond is designed to provide cooling for 30 days.

Water from the NSW components is returned to the pond by redundant headers. Each of the headers supplies a spray system that is used to cool the returned fluid. Each spray system consists of nine lateral headers. Each of the lateral headers contains nine spray clusters. Each of the nine clusters contains five spray nozzles.

Provisions are installed to control spray pond temperature. The return system contains a spray system bypass valve that is opened if spray pond temperature drops below 85°F. In the bypass mode of operation, spray pond water is recirculated through the NSW system without being cooled. When spray pond temperature increases to above 90°F, the bypass is closed, and the pond is cooled by evaporation.

4.7.3.3 Diesel Generator NSW Supply

Each diesel generator NSW outlet valve is normally closed and receives an automatic open signal when the diesel generator receives a start signal. It should be noted that a diesel generator will not operate correctly if it is not supplied with cooling water.

4.7.3.4 Makeup (HPI) Pump Supply

During normal or emergency operations, the NSW system supplies cooling water to at least two of the three makeup pumps. The NSW system provides cooling for each pump's lubrication system. The A makeup pump receives its cooling water from the A NSW train, while the B pump receives its cooling water from the B NSW train. The C makeup pump may be supplied with cooling water from either train; however, the cooling water supply should correspond to the electrical supply to the pump. For example, if the C makeup pump is aligned to the EB bus (B train), then the cooling water supply to the pump should be aligned to the B NSW train. Cooling water to the pumps is controlled by manual throttle valves in the return lines from the pumps.

4.7.3.5 Nuclear Instrument Air Compressor Supply

During normal plant operations, both of the nuclear instrument air compressors are shutdown and the cooling water to the compressors is isolated. A solenoid valve in the outlet line of each compressor isolates the unit. When a LOOP or an ESFAS signal is received, the compressor starts, and the solenoid valve de-energizes. When the solenoid de-energizes, the outlet valve opens and the air compressor unit is supplied with cooling water.

4.7.3.6 Component Cooling Water Heat Exchanger Supply

During normal operation, two of the three CCW heat exchangers are in service. A temperature control valve on the inlet to each heat exchanger controls the flow of NSW through the heat exchanger tubes. Since CCW is not safety-related, the NSW supply to each heat exchanger is isolated by an ESFAS signal. With NSW isolated to the heat exchangers, the RCPs motors and bearing temperatures will remain within acceptable limits for about 30 minutes. This allows the operators sufficient time to respond to the cause(s) of the ESFAS signal before RCP temperatures become a major concern.

4.7.3.7 Shutdown Cooling Water Heat Exchanger Supply

Cooling water from the NSW system to the SCW heat exchangers is required during two plant operating conditions. The first condition involves normal decay heat removal during shutdown conditions and the second condition involves heat removal during accidents.

During normal operating conditions, the flow of NSW through the tubes of each SCW heat exchanger is governed by a temperature control valve (TCV) on the outlet of the heat exchanger. The TCV throttles the NSW flow to maintain the desired SCW heat exchanger outlet temperature.

In emergencies, the temperature control valve is opened fully by ESFAS channel 5 A/B (BWST to containment sump switchover). The valve is ramped to the full-open position in anticipation of the need to cool the hot water from the sump. After a time delay, the control of the valve is returned to the temperature control mode.

In order to prevent the leakage of radioactive water into the NSW system, the operating pressure of the NSW system is higher than the operating pressure of the SCW system. Therefore, any radioactive leakage will be contained in the SCW system, which is completely located in the auxiliary building.

4.7.4 Emergency Operations

If either an ESFAS or LOOP signals are received, the following sequence of events takes place in the NSW system:

1. The service water system is isolated from the NSW system as follows:

   - The supply header is isolated by check valves upon initiation of flow from the ESW pumps.
   - The motor-operated isolation valves in the return header close.

2. Both ESW pumps start, the ESW supply valves open and the NSW return valves to the spray pond open.

3. Both SCW pumps start.

4. Nuclear service water flow to the SCW heat exchangers is controlled by the NSW TCV.

5. The NSW supplies to the CCW heat exchangers are isolated (ESFAS signal only).

6. The diesel generator cooling water outlet valves open when the diesel generators start.

7. The nuclear instrument air compressor solenoid outlet valves de-energize and open.

4.8 Shutdown Cooling Water System

4.8.1 Shutdown Cooling Water System Description

The shutdown cooling water (SCW) system is a safety-related cooling water system that supplies cooling water to the decay heat removal (DHR) system and the reactor building spray system. In the decay heat removal system, SCW cools the DHR pump seals and the DHR heat exchangers. The reactor building spray pumps and heat exchangers are also supplied with SCW cooling. It should be noted that heat exchangers are not included in the design of the reactor building spray systems for most B&W units. Since the SCW system is a closed loop system, it provides a barrier between the radioactive components that it cools and the non-radioactive NSW system.

The operation of the SCW system is required during plant shutdowns and emergencies. The SCW system may be cross-connected with the component cooling water (CCW) system to supply CCW components if cooling is required during accident situations.

Each of the SCW loops consists of an SCW heat exchanger, a pump, a surge tank and the DHR and reactor building spray heat exchangers. Flow through the system is initiated as a part of the plant shutdown procedure. If an ESFAS signal is received, both SCW loops are automatically placed in service.

Hydrazine and morpholine are added to the SCW system to control the oxygen concentration and pH of the system. The control of these parameters helps to minimize corrosion.

4.8.2 SCW Component Description

4.8.2.1 SCW Pumps

Each of the SCW pumps is a horizontal centrifugal pump rated at 13,500 gpm with a head of 75 ft. (33 psi). The prime mover for the pump is a 4160-Vac squirrel cage induction motor. The pumps are powered from the vital ac distribution system. The pumps are automatically started by a LOOP or an ESFAS signal.

4.8.2.2 SCW Surge Tank

Each of the SCW loops contains a surge tank that provides for thermal expansion and contraction of the SCW system. Each tank is a horizontal cylindrical tank with a capacity of 2000 gallons. The tank is normally vented to a radioactive waste system.

Should radioactive leakage enter the SCW system, it will be sensed by a radiation monitor located on an SCW heat exchanger inlet line. High radioactivity will close the surge tank vent. In the event of continued leakage into the system, a surge tank relief valve will prevent surge tank overpressurization. The discharge of the relief is directed to the radioactive waste system. Surge tank level is monitored in the control room.

4.8.2.3 DHR Heat Exchanger Supply

The SCW flow through the shell side of each DHR heat exchanger is governed by a flow control valve located on the outlet of the heat exchanger. During normal operation, the flow control valve may be positioned by the operator. However, if an ESFAS signal is received, the flow control valve is fully opened. The flow control valve is designed to fail open if either electric or pneumatic power is lost.

4.8.2.4 Decay Heat Removal Pump Seal Coolers

Flow through each DHR pump seal cooler is controlled by a manual throttle valve that is locked in the correct position. Seal cooler outlet temperature may be monitored locally.

4.8.2.5 Reactor Building Spray Heat Exchangers

The SCW flow through the shell side of each reactor building spray heat exchanger is governed by a flow control valve located on the outlet of the heat exchanger. During normal operations, the flow control valve may be controlled by the operator. But, if an ESFAS signal is received, the flow control valve is opened fully. The valve will fail open if electric or pneumatic power is lost.

4.8.2.6 Reactor Building Spray Pump Seal Coolers

Flow through each reactor building spray pump seal cooler is controlled by a manual throttle valve that is locked in the correct position. Seal cooler outlet temperature may be monitored locally.

4.8.3 SCW/CCW Cross-connects

Provisions are made to supply cooling water from the SCW system to the CCW system whenever cooling water is lost to the seismic Category I components of the CCW system. The following CCW-supplied components are seismic Category I.

1. Spent fuel pool heat exchangers
2. Seal return coolers
3. RCP seal heat exchangers
4. Letdown coolers

Manual operation of valves is required to cross-connect the systems. Only one SCW loop may be cross-connected to the CCW system at a time.

4.9 NSW/SCW Summary

The nuclear service water system provides cooling water to safety-related components. The system is supplied by the service water pumps during normal operations and by the ESW pumps during accidents. Water from the NSW system is normally returned to the cooling towers; however, the return headers are aligned to the spray pond in the event of an accident.

The shutdown cooling water system is a closed loop cooling water system that provides a barrier between the radioactive components that it cools and the environment. The SCW system is cooled by the NSW system. The SCW system provides cooling water to the DHR pumps and heat exchangers and the reactor building spray pumps and heat exchangers. The SCW system is a safety-related system.

TABLE 4-1 EMERGENCY CORE COOLING SYSTEM FLOWS (ONE TRAIN)							
	HPI System Flow		LPI System Flow		CFT System Flow	ECCS Total Flow	
RCS pressure	Actual	Analysis	Actual	Analysis		Actual	Analysis
psig	gpm	gpm	gpm	gpm	gpm	gpm	gpm
1600	575	486	0	0	0	575	486
600	730	630	0	0	0	730	630
500	740	630	0	0	9016*	9756*	9646*
200	770	0	3600	2500	**	4370	2500+
100	778	0	5800	4500	**	6578	4500+
0	780	0	6500	4500	**	7280	4500+

(*) Core flooding system flow when D/P (tank – RCS) = 50 psid.
(**) Core flooding system flow depends on break (time and pressure dependent).
\+ Does not include core flooding system flow.

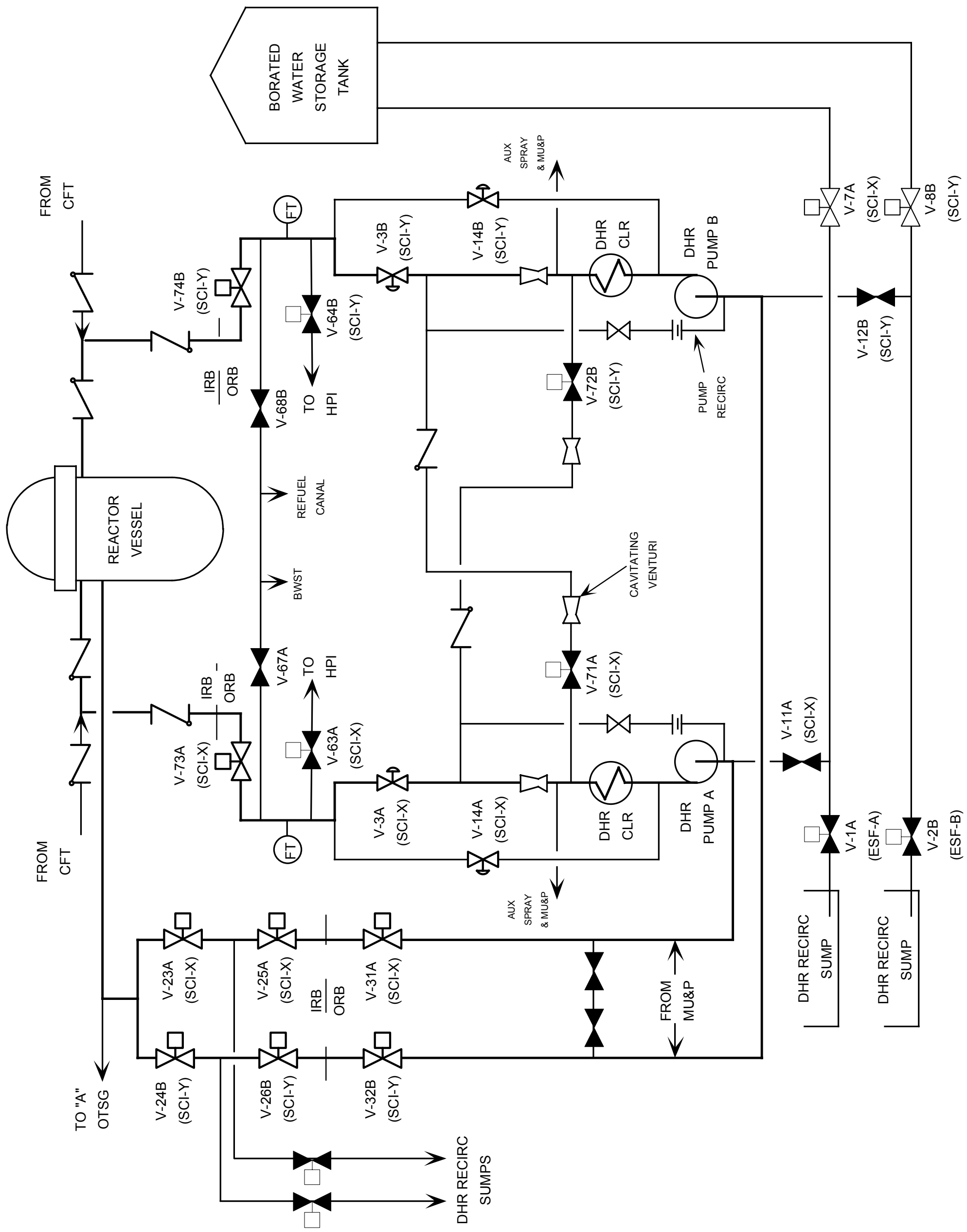

Figure 4-1 Decay Heat Removal System – Plant Cooldown Alignment

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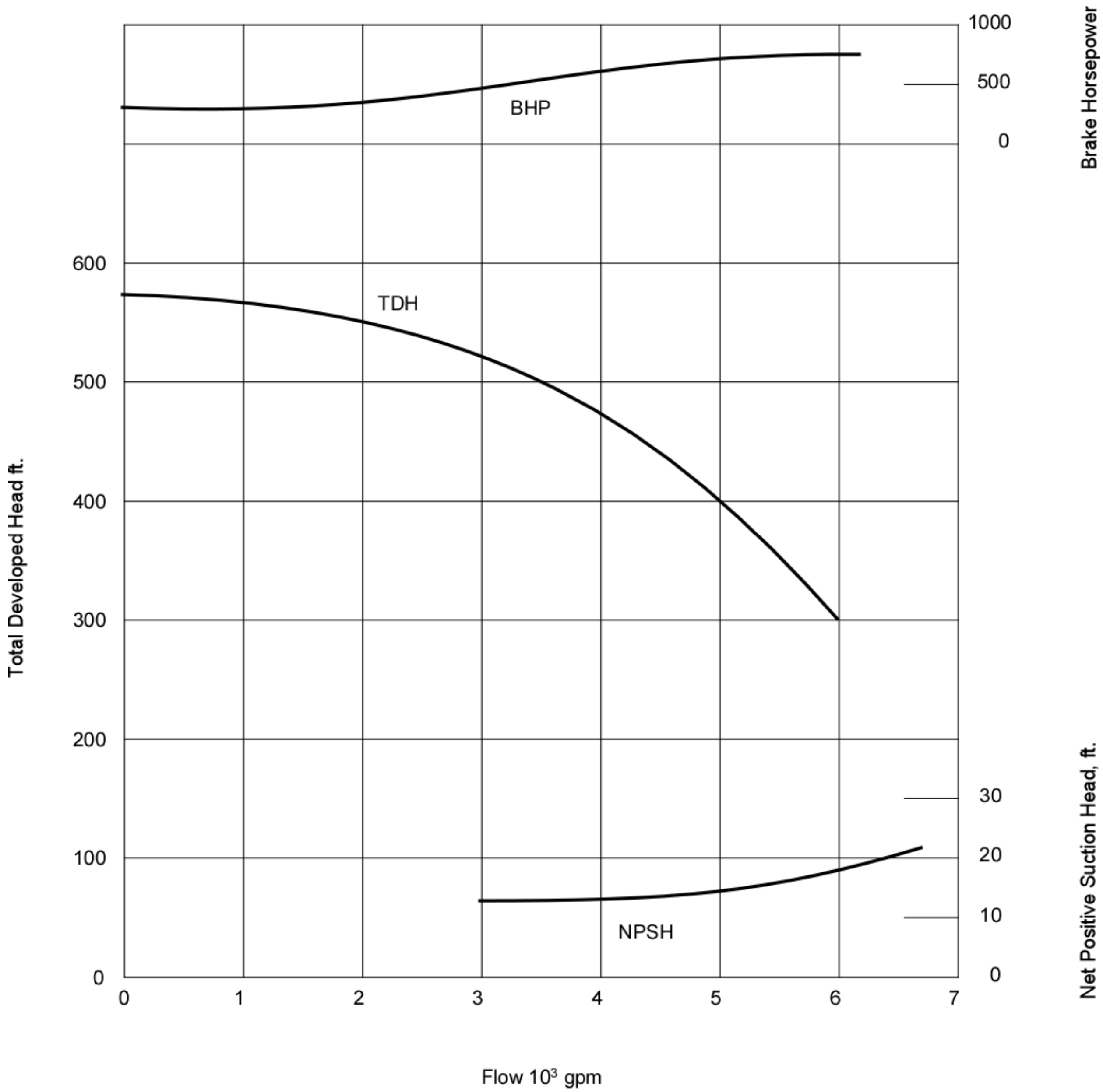

Figure 4-2 Decay Heat Removal/Low Pressure Injection Pump Characteristics

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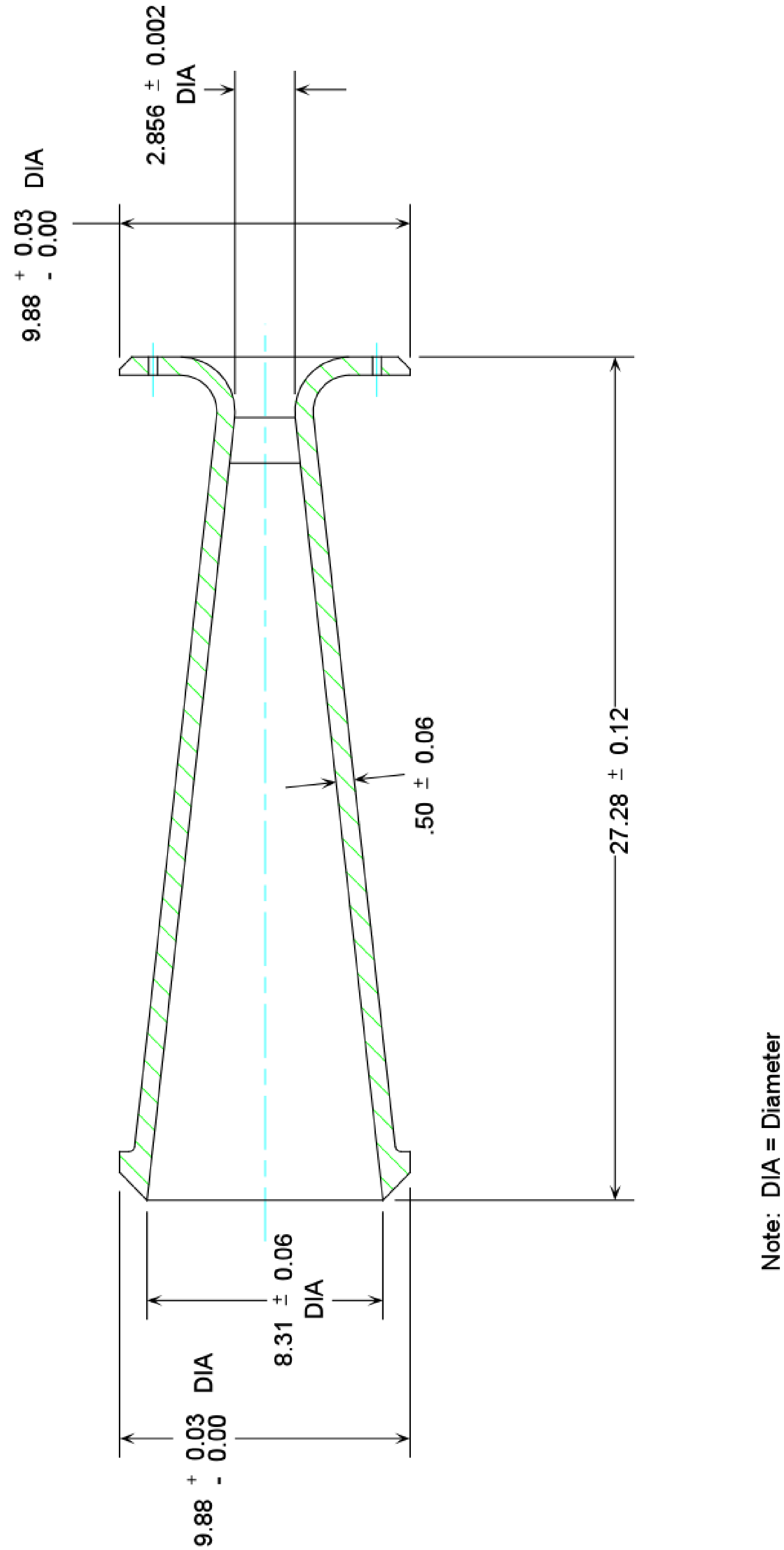

Figure 4-3 Cavitating Venturi

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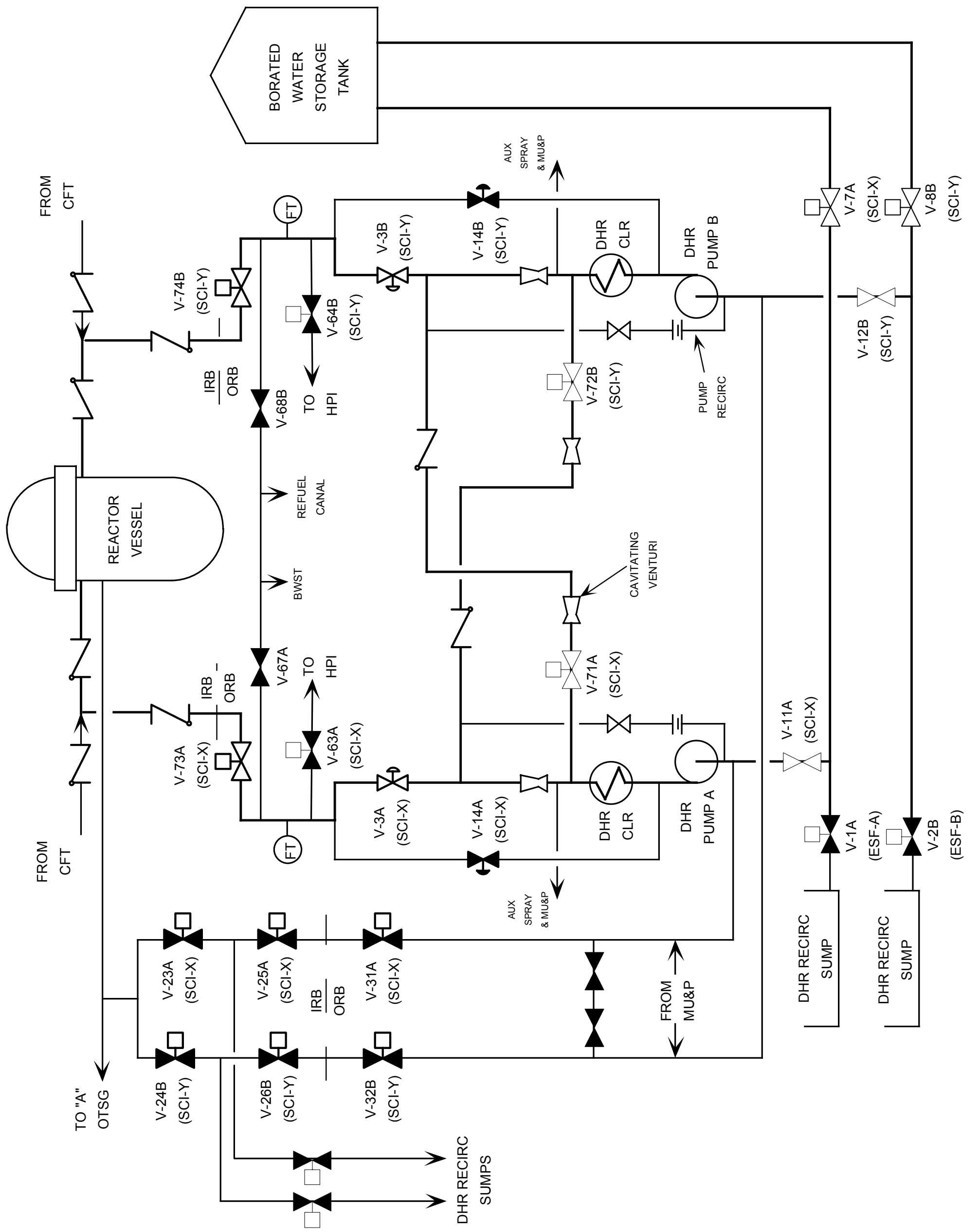

Figure 4-4 Decay Heat Removal System – Emergency Core Cooling Arrangement

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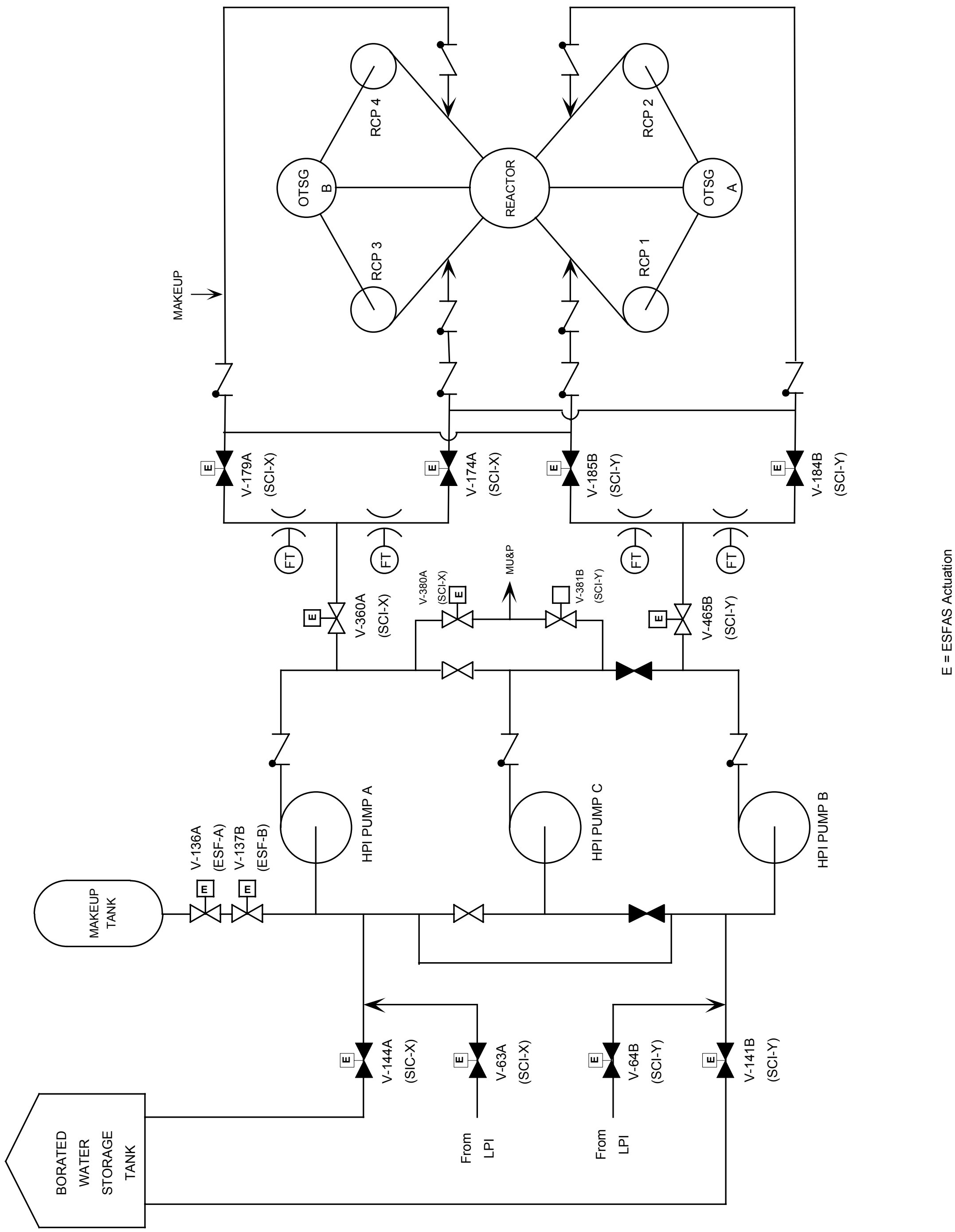

Figure 4-5 High Pressure Injection System

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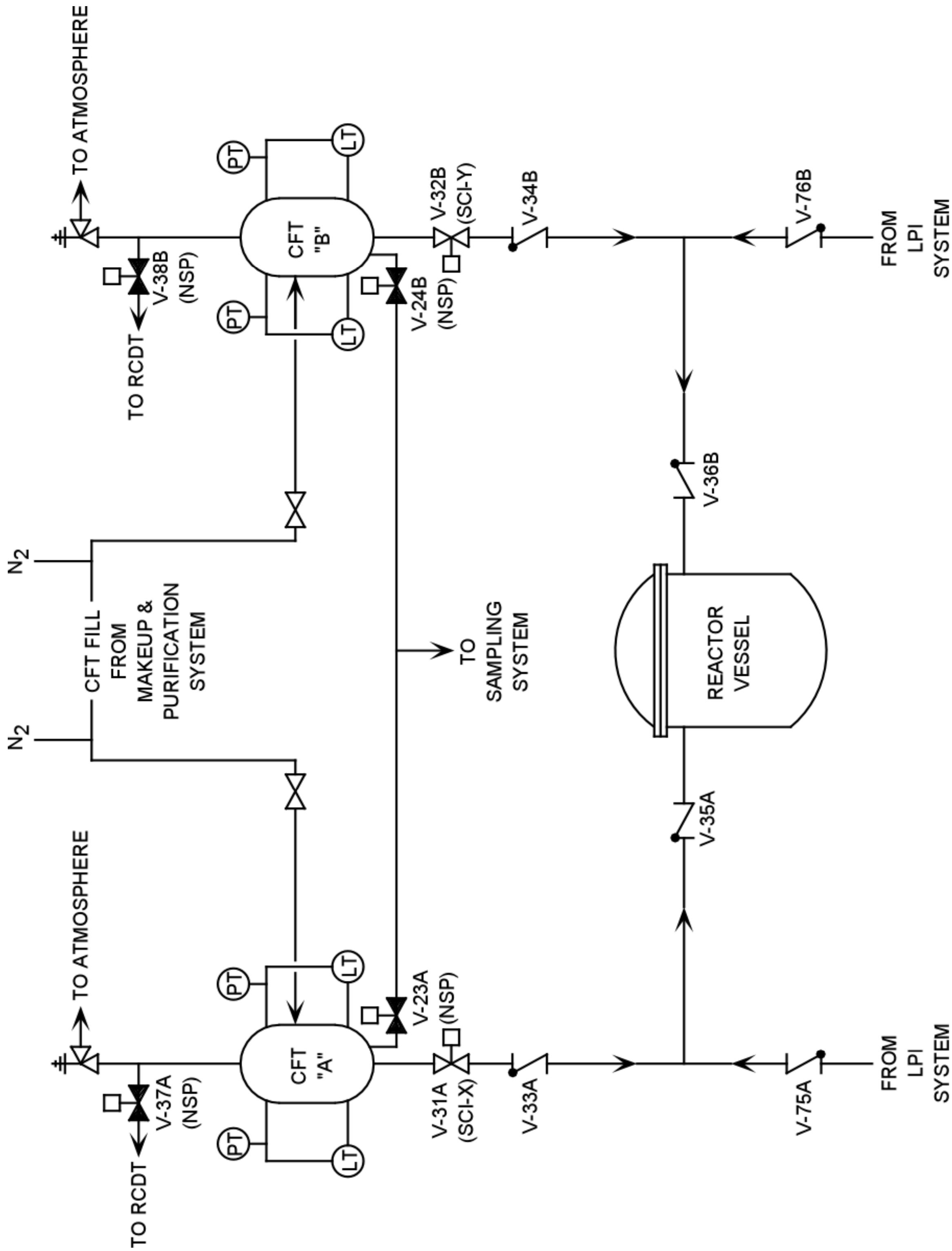

Figure 4-6 Core Flooding System

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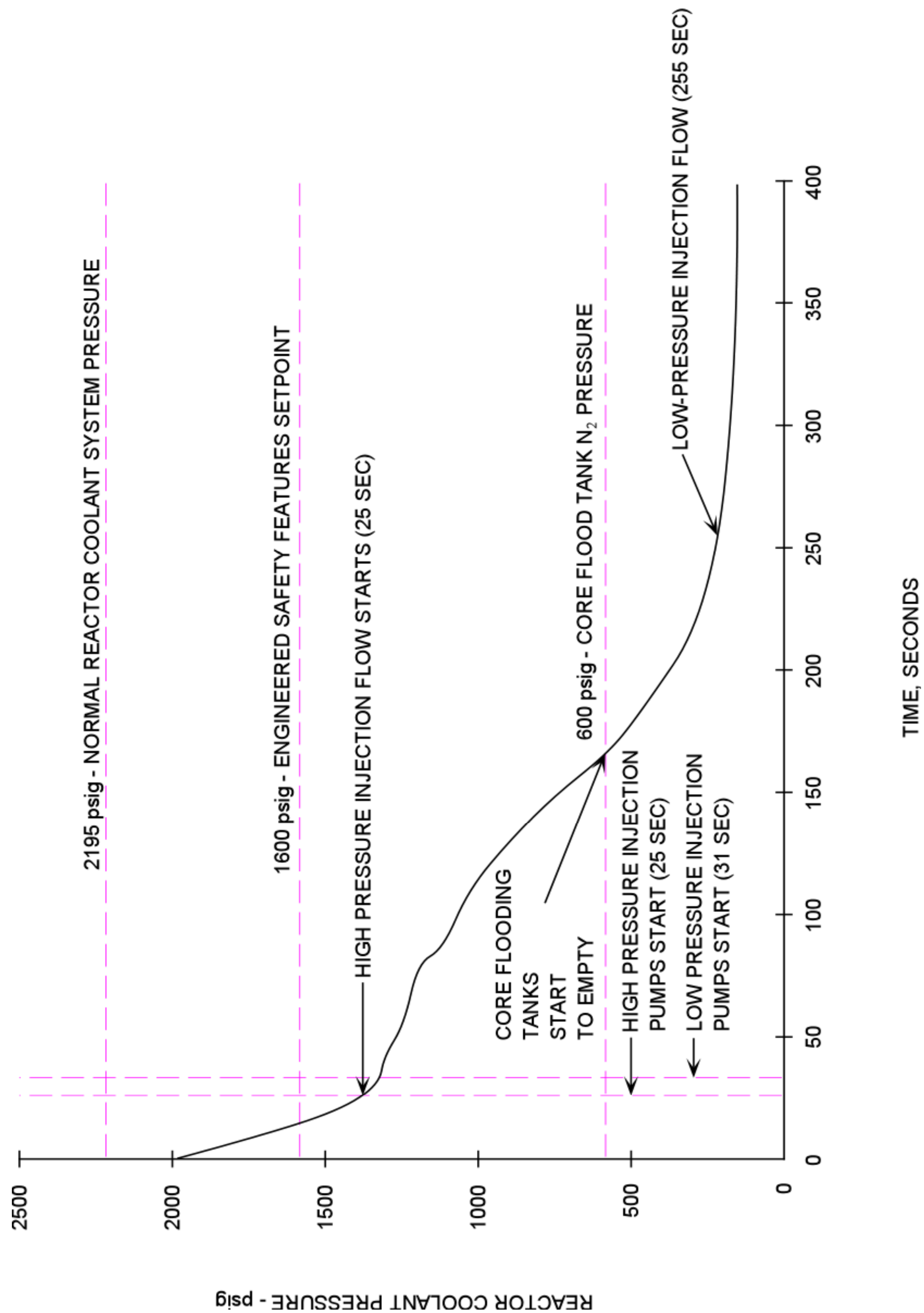

Figure 4-7 Emergency Core Cooling Actions

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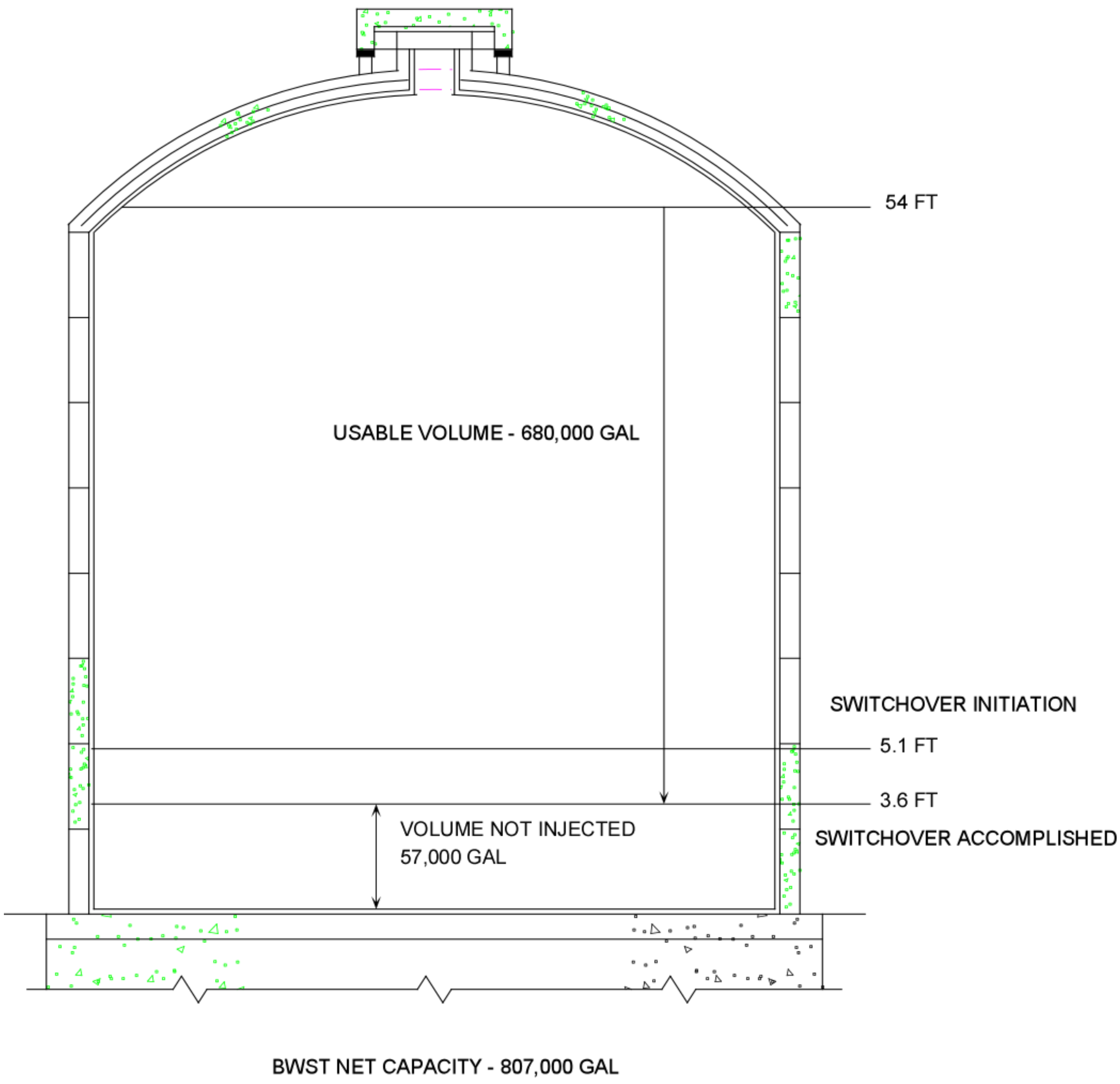

Figure 4-8 Borated Water Storage Tank Water Capacities

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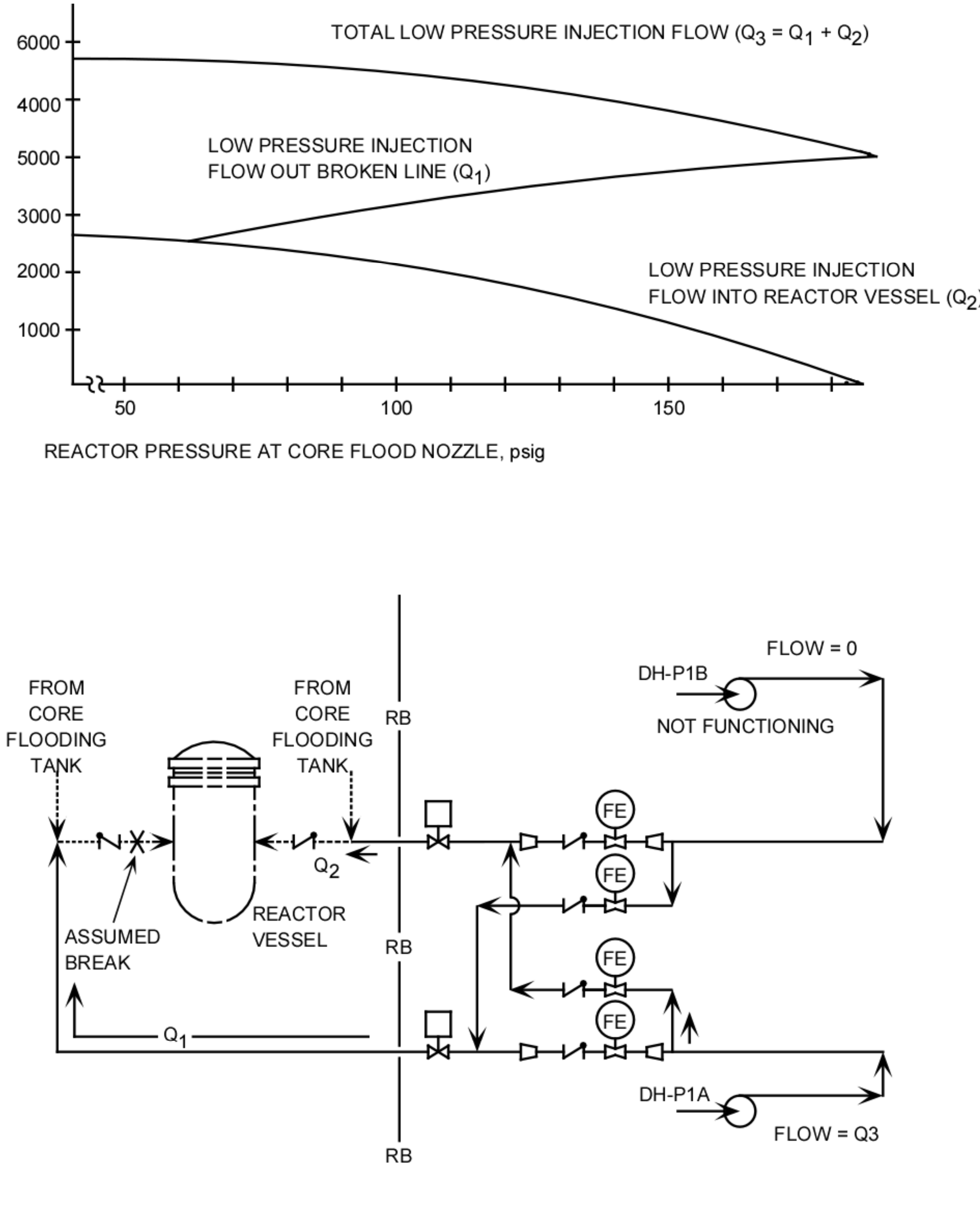

Figure 4-9 Low Pressure Injection Flow for Core Flood Line Break

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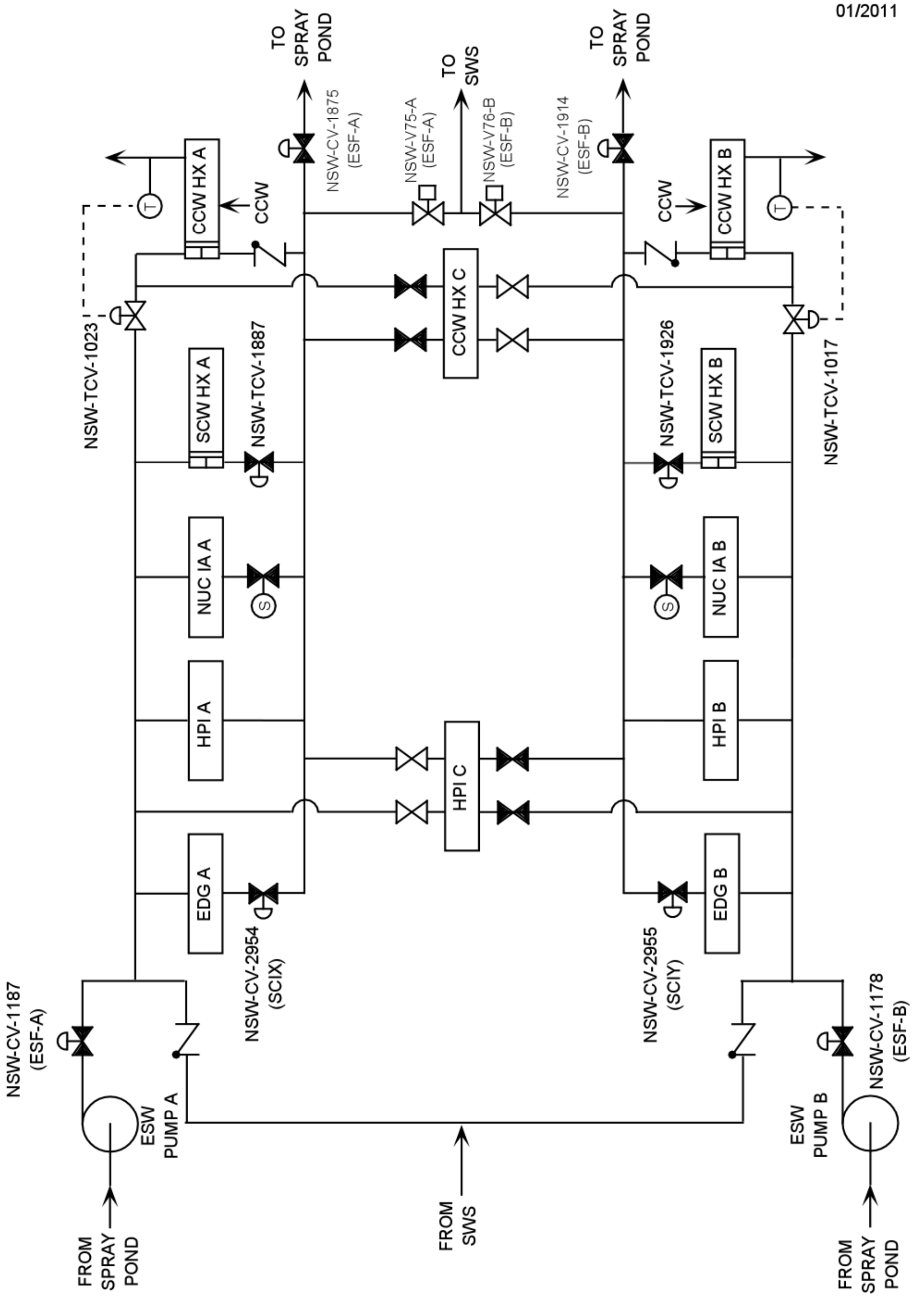

Figure 4-10 Nuclear Service Water

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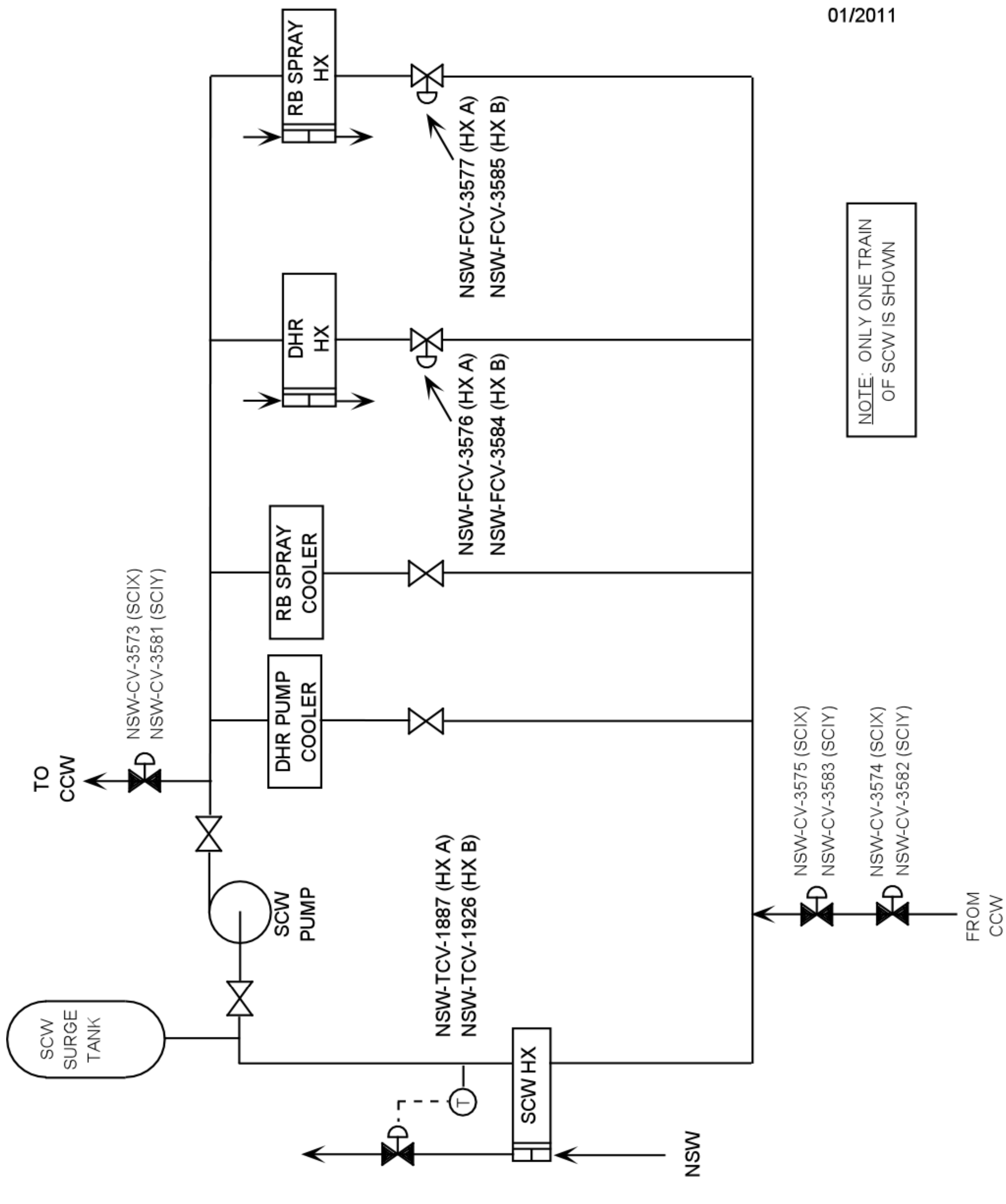

Figure 4-11 Shutdown Cooling Water

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