Pressurized Water Reactor
B&W Technology
Crosstraining Course Manual

Chapter 11.0

ANO-1 Partial Loss of Flow

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11.0 ANO-1 PARTIAL LOSS OF FLOW

Learning Objectives:

1. Explain the actions of the rapid feedwater reduction (RFR) circuits on the integrated control system and the feedwater system components.

2. Explain the cause of the overcooling of the reactor coolant system.

3. Explain how reverse flow occurred from the reactor coolant system into a cross connect of high pressure injection lines outside the reactor building.

4. State the concern over the reactor coolant flow into the high pressure injection system.

11.1 Introduction

On January 20, 1989, ANO-1 experienced an event which included a partial loss of flow in the reactor coolant system, leakage of reactor coolant through a check valve into a high pressure injection pipe, and a failure of the main feedwater system to decrease flow as designed after the reactor trip. The plant was operating at full power when main generator problems caused a generator trip. The turbine and reactor tripped as designed, but the failure of a bus transfer resulted in loss of power to two of the reactor coolant pumps. This partial loss of flow set up a differential pressure across the high pressure injection connections to the reactor coolant system. The failure of a check valve to reseat caused reactor coolant to flow through a cross connect pipe in the high pressure injection system outside the reactor building. Several failures in the feedwater and related systems allowed overfeeding of one of the steam generators, causing a slight overcooling of the reactor coolant system. A post reactor trip walkdown of the reactor building revealed a small leak in a weld where a drain line connects to a reactor coolant pump suction pipe. A detailed sequence of events is given in the Appendix.

11.2 Main Generator Failure

The turbine trip and subsequent reactor trip were caused by a trip of the main generator. The failure of the generator was a result of a loss of the generator field. An electrical connection on the exciter broke from what appeared to be a stress induced crack, possibly due to the weight of the lead wire and constant low level vibration during operation. The exciter is not a safety-related system.

11.3 Failure of Fast Bus Transfer

After a main generator trip, power to plant loads shift from the unit auxiliary transformer to the startup transformer. Power to the unit auxiliary transformer is supplied by the main generator, while power to the startup transformer is from the offsite power system. There are four non-vital buses that transfer to startup transformers, two of which supply reactor coolant pump motors and two which supply vital buses. One bus that supplies two reactor coolant pump motors failed to fast transfer. A slow transfer did occur, but the pumps had already tripped. This failure did not affect any vital buses. If the transfer had affected a vital bus, the bus would have been isolated and then supplied by a diesel generator. The licensee believes the failure to be related to the reset time on a synchronism check relay.

11.4 Spurious Actuation of EFIC Channels

Two redundant channels of the emergency feedwater initiation and control (EFIC) system actuated spuriously immediately after the turbine trip. The actuation signal was low OTSG level, although the levels were above the setpoint for actuating emergency feedwater. No equipment actually started because the channels that tripped did not satisfy the actuation logic. The actuation was caused by pressure oscillations in the steam lines induced by the closing of turbine stop valves. The time delays built into the system are either too short or did not function properly. If there had been an actuation of emergency feedwater, it would have added to the overcooling problem caused by other failures.

11.5 Main Feedwater System Failures

During normal operations, feedwater flow is controlled by the integrated control system (ICS) using startup control valves (SUCVs) from 0% to 15% power, low load control valves (LLCVs) from 15% to 50% power, and feed pump speed control from 50% to 100% power. Following a reactor trip, the rapid feedwater reduction (RFR) circuits provide signals to the integrated control system (ICS) to close SUCVs and LLCVs and runback the speed of the main feedwater pumps. Figure 11-1 shows the relationships between the RFR circuits, the ICS, and the feedwater pumps and valves.

In the ANO-1 event, the SUCVs and the LLCVs remained open and one of the feedwater pumps did not run back. In addition, the main feedwater block valve associated with the failed pump did not close. These failures all contributed to the overfeeding of an OTSG which caused an overcooling of the reactor coolant system. The failure of the SUCVs and LLCVs was due to a wiring error during installation of the RFR circuits. The ICS operated as designed. The output signals from the ICS and RFR circuits to decrease feedwater pump speed and close the main feedwater block valve were correct. The feedwater pump problem may have been a failure in the pump controller or the turbine steam supply. The main feedwater block valve started to close as designed, but stopped on mechanical overload. A change in the torque switch setting may solve the problem in the future.

11.6 Check Valve Failure

The overcooling caused by the feedwater system malfunctions made the pressurizer level decrease, and the operators manually started high pressure injection. After two minutes, high pressure injection was secured, but the check valve in the “B” injection line did not reseat. Due to the fact that two reactor coolant pumps were tripped and two were running, a differential pressure was created across high pressure injection lines. The “B” and “C” lines are cross connected, and with the “B” reactor coolant pump running and the “C” pump stopped, the check valve in the “B” injection line was needed to stop flow in the cross connect line. The failure of the check valve allowed hot reactor coolant to flow in a high pressure injection cross connect line and back into the reactor coolant system as shown in Figure 11-2. The discovery was made when a fire alarm activated in a penetration area. The “B” and “C” injection lines were found to be hot. This is of concern because the piping in the injection system is rated for reactor coolant system pressure, but it is not rated for reactor coolant system temperature. Analysis has shown that stress limits were exceeded in certain locations in the cross connect line during the event.

11.7 Reactor Coolant System Leakage

During a routine walkdown of the reactor building after the trip, a “pinhole” leak was found. The hole was on the weld of a drain line connected to the reactor coolant system. The leak was probably due to a weld defect and was not related to the event.

11.8 Summary

This event at ANO-1 began with a failure of the main generator which caused a turbine and reactor trip. The failure of a fast transfer to offsite power on one of the non-vital buses resulted in the trip of two reactor coolant pumps. When several feedwater related problems caused a slight cooldown of the reactor coolant system, the operators manually started high pressure injection. After a couple of minutes, high pressure injection was terminated, but one of the check valves did not reseat. With a differential pressure across high pressure injection lines caused by having two reactor coolant pumps running and two stopped, reactor coolant leaked back into a cross connect line in the high pressure injection system outside containment.

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APPENDIX - SEQUENCE OF EVENTS

Initial conditions: 100 percent power; normal operating temperature and pressure.

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20:30 (?)
(Exact time unknown) Perturbations in the form of voltage swings/spikes are observed by the operators on a control room meter that displays the output voltage from the main generator exciter voltage regulator (i.e., generator field voltage). The normal generator field voltage is 50 Vdc. The voltage spikes occur fairly regularly (at approximately four minute intervals) with peaks values of near 90 Vdc at first, and becoming more severe with time until the meter pegged high at 150 Vdc just prior to losing generator field voltage.

21:58:11
The automatic voltage regulator is placed in the off position by the control room operator. Loss of main generator field voltage occurs due to a failed electrical connection on an exciter field winding. This causes a generator lockout via the generator protection circuits (i.e., the generator field breakers and output breakers open, electrically disconnecting the generator).

The main turbine trips on generator lockout.

The reactor trips on main turbine trip via the safety related anticipatory reactor trip (ART) circuits. The plant operators proceeded to bring the unit to hot shutdown conditions.

The power source to nonsafety-related 6.9kV Bus H1 fails to automatically fast transfer from the unit auxiliary transformer (supplied from the main generator) to the startup transformer (supplied from the offsite power system). An automatic fast transfer does occur to provide power to the other nonsafety- related buses, 6.9kV Bus H2 and 4.16 kV Buses A1 and A2. All safety-related buses transfer as designed following the trip.

21:58:15
A trouble alarm is received in the control room on 6.9 kV Bus H1 loss of voltage.

Reactor Coolant Pumps "A" and "C" trip on undervoltage. These pumps are powered from Bus H1.

Two channels of the emergency feedwater initiation and control (EFIC) system spuriously trip upon sensing once through steam generator (OTSG) low level. Actual OTSG level was well above the emergency feedwater initiation setpoint.

The main feedwater (MFW) system startup flow control valves (SUCVs CV-2623 and CV-2673) and low load flow control valves (LLCVs CV-2622 and CV-2672) fail to close as designed following the reactor trip, allowing continued MFW flow paths to each OTSG.

The "B" MFW pump (P1B) fails to runback to minimum speed as designed following the reactor trip.

The "B" MFW block valve (CV-2675) fails to close as designed following the reactor trip. The valve starts to close, but stops when the valve torque switch actuates before the valve closes.

An automatic "slow dead bus transfer" occurs to provide power to 6.9kV Bus H1 from the startup transformer, restoring power to the bus. Reactor Coolant Pumps "A" and "C" remain shutdown.

SEQUENCE OF EVENTS (continued)

21:59:08
The operators manually start high pressure injection (HPI) System Pump P36A to provide additional makeup flow to maintain pressurizer level above the heater cutoff point. Pressurizer level was decreasing due to reactor coolant system cooldown from excessive MFW flow to OTSG “B” and the slight overcooling (reactor coolant temperature dropped about 11F).

21:59:25
OTSG “B” high level alarm at 92 percent is received in the control room.

21:59:38
The “A” MFW isolation valve (CV-2680) and the “B” MFW isolation valve (CV-2630) are manually closed by the operators from the control room.

21:59:40
The “B” MFW main block valve (CV-2675) is manually closed by the operators from the control room

The “B” MFW pump (P1B) runs back to minimum speed on its own.

21:59:44
Level in the “B” OTSG begins decreasing from a high value of 99 percent on the operating range. Level in the “A” OTSG is less than, and paralleling, the level in the “B” OTSG.

The MFW isolation valves are reopened by the operators from the control room.

22:00:25
“B” OTSG level begins increasing from 91 percent on the operating range.

22:01:33
HPI Pump P36A is secured.

“B” OTSG level increases to near 100 percent on the operating range (actual level may have gone slightly off scale above 100 percent).

22:01:40
“B” OTSG level begins decreasing from 100 percent.

22:02:33
The “B” MFW pump is secured, and Motor Operated Valve CV-2827 in the crosstie line between the discharge of the two turbine driven MFW pumps (P1A and P1B) is opened. MFW Pump “A” is now providing MFW flow to both OTSGs via the SUCVs and LLCVs.

22:03:53
The “A” and “B” MFW isolation valves are closed by the operators from the control room in response to the increasing level in the OTSGs.

Control of the SUCVs and the LLCVs for both OTSGs is transferred from automatic control to manual control, and these valves are manually closed by the operators.

SEQUENCE OF EVENTS (continued)

22: (?)
(Exact time unknown) A fire alarm is received in the control room. The alarm is activated from a smoke detector located in the upper north piping penetration room (UNPPR). There is no fire water system flow to the UNPPR which indicates no actual fire in the area. An operator is dispatched to the UNPPR to investigate the cause for the alarm.

22:31
The NRC Operations Center is notified of the reactor trip in accordance with 10 CFR 50.72. The call was initially misclassified by the licensee as a courtesy call instead of a required notification. The call was made by a shift administrative assistant, not an operator.

22:38 (?)
(Exact time unknown) The “A” and “B” MFW isolation valves (and SUCVs) are reopened by the operators.

22:50 (?)
(Exact time unknown) The operator dispatched to the UNPPR reports back to the control room by telephone that the temperature of the “B” and “C” HPI system injection lines, and the crossover line that connects them, is excessively high (more indicative of RCS temperature than the expected temperature of the borated water used for HPI). The smoke detector is believed to have been actuated when tape attached to the HPI piping began to melt and smolder/smoke. It was noted that this event could have gone undetected. The plant operators suspected that the high temperature in the HPI piping was caused by failure of Check Valve MU-34B to reseat after HPI Pump P36A was secured at 22:01:33. Check Valve MU-34B is located inside the reactor containment building. The leakage flow path was in the reverse direction through Check Valve MU-34B and outside the reactor containment building via the “B” HPI injection line, then through the crossover line to the “C” HPI injection line, and back inside the reactor containment building to the RCS. The upstream check valves (MU-1214 and MU-1215 in the “B’ and “C” HPI injection lines respectively) performed as designed to prevent further backflow of reactor coolant into the HPI system piping.

22:59:41
An EFIC system initiation of emergency feedwater (EFW) occurs upon sensing low level in the “B” OTSG. The operators were aware that OTSG levels were decreasing, and were beginning to increase MFW flow at the time of the EFIC system initiation of EFW.

23:00:00
EFW is secured. Very little, if any, EFW system flow was injected into the OTSGs.

23:05 (?)
(Exact time unknown) Auxiliary Feedwater Pump P75 (motor driven) is manually started by the operators from the control room.

23:15 (?)
(Exact time unknown) The “A” MFW pump is secured.

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1:00 (?)
(Exact time unknown) The oncoming shift waste control operator is dispatched to check the piping in the UNPPR but because of confusion he missed the hot HPI crossover piping. He erroneously reported to the control room that the pipe appeared to be cooling down.

SEQUENCE OF EVENTS (continued)

2:02
The NRC Operations Center is notified of the EFIC system initiation of EFW in accordance with 10 CFR 50.72. This notification was made by the unit shift supervisor.
3:00 (?)
(Exact time unknown) The waste control operator is again dispatched to the UNPPR with better instructions and he correctly determines that the HPI crossover piping is still hot.

3:53
Reactor Coolant Pumps “A” and “C” are restarted. It appears that restarting the pumps causes Check Valve MU-34B to reseat and/or removed the differential pressure which was driving the backflow, stopping the RCS back leakage into the HPI system piping. The waste control operator confirms that the piping was cooling off at this time.

5:00
A routine post reactor trip walkdown of the reactor containment building identifies possible reactor coolant system leakage.

12:56
The leakage is confirmed to be from an elbow weld in a 1 1/2-inch drain line off the “B” reactor coolant pump suction line. The leakage rate is believed to be small (approximately 10 to 20 ml per minute). An Unusual Event (UE) is declared by the licensee due to unisolable RCS pressure boundary leakage. (T.S. 3.1.6.3 requires cooldown to begin within 24 hours of identifying the leakage.)
13:06
Operators begin the process of taking the reactor to a cold shutdown condition.

13:10
The NRC Operations Center is notified of the declaration of an UE in accordance with 10 CFR 50.72. This notification is made by the shift administrative assistant.

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17:30
The reactor is at cold shutdown, and the UE is terminated.

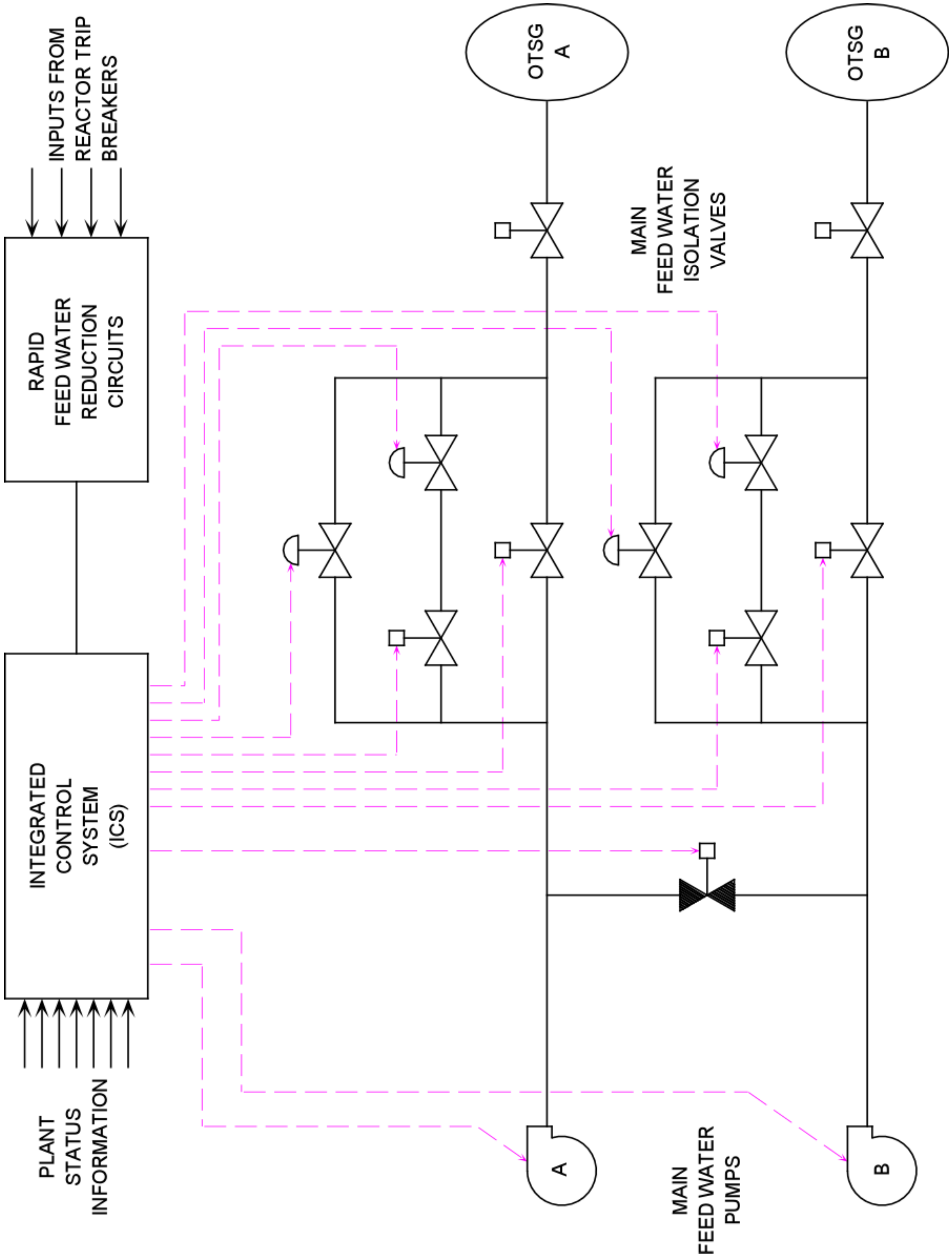

Figure 11-1 Rapid Feedwater Reduction Circuits Interface With ICS and Main Feedwater System

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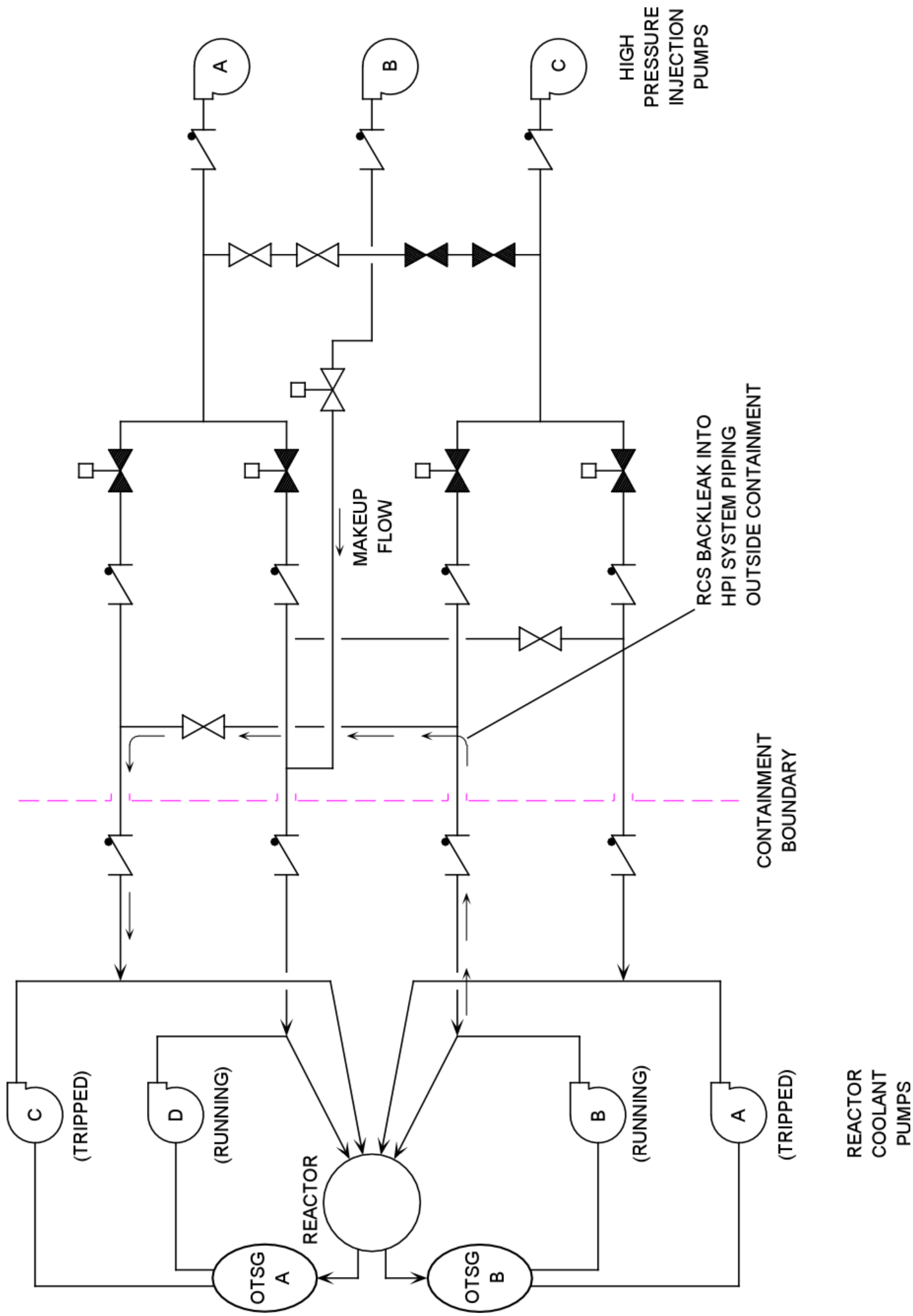

Figure 11-2 High Pressure Injection System

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