General Electric Advanced Technology Manual

Chapter 4.10

Shutdown Plant Problems

TABLE OF CONTENTS

4.10 SHUTDOWN PLANT PROBLEMS 4.10-1

4.10.1 Introduction 4.10-1

4.10.2 Binding of Gate Valves 4.10-2

4.10.2.1 Thermal Binding Phenomenon 4.10-3

4.10.2.2 Pressure Locking Phenomenon 4.10-3

4.10.2.3 Consequences of Locking.. 4.10-4

4.10.3 Mode 3/4 Event 4.10-4

4.10.4 Summary 4.10-5

LIST OF TABLES

4.10-1 Sequence of Events 4.10-6

LIST OF FIGURES

4.10-1 Accident Sequence Comparison

4.10-2 Pressure Locking Flexible-Wedge Gate Valve

4.10-3 RHR System Shutdown Cooling Mode

4.10 Shutdown Plant Problems

Learning Objectives:

1. List two major accident sequences identified at low power and shutdown plant conditions.

2. Describe the differences between full power and low power/shutdown major accident sequence classes.

3. List three systems and their components that have a history of becoming pressure locked.

4. Describe the alignment of the Residual Heat Removal System and Recirculation System when in shutdown cooling mode of RHR.

4.10.1 Introduction

In 1989 the Nuclear Regulatory Commission initiated a program to examine the potential risks presented during low power and shutdown conditions. Two plants, Surry (PWR) and Grand Gulf (BWR), were selected to be studied. These studies (NUREG/CR-6143) along with operational experiences indicated that the risk during low power and shutdown conditions may be significant.

The risk associated with Grand Gulf operating in modes 4 and 5 was shown to be comparable with the risk associated with full power operation, 10^{-6} range. While the risk is low, very few systems/features of the plant are required to be available to attenuate a release should it occur. Technical specifications permit more equipment to be inoperable during low power and shutdown conditions. In certain plant conditions, primary containment is not required.

Figure 4.10-1 presents a comparison of the mean core damage frequency percentages for the major classes of accidents from both the full-power NUREG-1150 and the low power and shutdown mode analyses NUREG/CR-6143. From this figure, obvious similarities and differences can be seen. The major similarity observed is that in both analyses the station blackout (SBO) class is important. In the full power analysis SBOs are dominate accident sequences due to the loss or degradation of multiple systems. In operating mode 3 and 4 SBOs also show up because they still cause loss or degradation of multiple systems. However, there are additional accidents that can cause loss or degradation of multiple systems because of considerations unique to those modes of operation.

The major differences in the accident progression associated with the SBOs are:

- Almost all low power and shutdown mode SBOs sequences lead to an interfacing system LOCA where as the full power sequences do not.
- The containment is always open at the start of the low power and shutdown accidents whereas it is isolated at the start of the full power accidents.
- The probability of arresting core damage in the vessel is greater for full power accidents than for low power and shutdown conditions.

The remaining classes of accidents indicate major differences between the two analyses. In the full power analysis, the anticipated transient without scram (ATWS) class is the second most important class while in the low power and shutdown analysis the second most important is SBO, with LOCA being number one. Given the plant conditions analyzed in the two studies, the first point that can be made is that ATWS sequences were simply not possible with the plant already in a shutdown state. On the other hand, since LOCAs were possible in both analyses, why did this class only show up in low power and shutdown results? While no detailed examination of this phenomenon was undertaken, the most likely reason for the appearance of LOCAs results is the intentional disabling of the automatic actuation of the suppression pool makeup system which is unique to the Mark III containment. Defeating automatic actuation of the suppression pool makeup is done for safety reasons. As a result, the continued use of injection systems during a LOCA requires operator intervention. The difference in reliability between automatic actuation and operator action generally accounts for the fact that LOCAs survived in the low power and shutdown analysis but not in the full power analysis.

4.10.2 Binding of Gate Valves

Thermal binding of double-disc and flexible wedge gate valves has been addressed by the NRC and the industry since 1977. Particularly, throughout the 1980's the industry issued a number of event reports concerning safety-related gate valve failures due to disc-binding. These failures were attributed to either pressure locking or thermal binding. Binding of gate valves in the closed position is of safety concern because gate valves have a variety of applications in safety-related systems and may be required to open during or immediately following a postulated event. During such events, valve performance is severely challenged by the rapid cooldown and depressurization rates which expose the disc to large differential pressures.

Generally valve operators are not sized to open a valve against binding forces. Pressure locking or thermal binding of gate valves represents a nonrevealing common-mode failure mechanism since normal surveillance tests may not detect or identify them.

Safety-related systems for a BWR in which valves have become pressure locked include:

- HPCI - Steam admission valve
- LPCS - Injection valve
- LPCI - Injection valve
- RCIC - Steam admission valve
- RR - Recirc pump discharge valve

A review of the events shows that there were two potential causes of pressure locking; liquid entrapment in the bonnet and high ΔP across the disc while in the closed position. Most of the events occurred during infrequent plant evolutions such as heat-up, cooldown, and testing. Pressure locking adversely affects operation of motor operated valves, and renders the associated system unavailable.

4.10.2.1 Thermal Binding Phenomenon

If a wedge gate valve is closed while the system is hot, thermal binding can occur as the system cools. The valve body and discs mechanically interfere because of their different thermal expansion and contraction characteristics. The difference in thermal contraction can cause the seats to bind the disc so tightly that reopening is extremely difficult or impossible until the valve is reheated. Several potential remedies have been suggested to alleviate this situation:

- Slightly opening and reclosing a valve periodically during a cooldown.

- Limiting valve actuator closing forces.

- Using compensating spring packs to reduce valve initial closing forces.

In general, neither ac nor dc valve motor operator sizing analyses account for the extra force needed to unseat a valve when it is thermally bound.

4.10.2.2 Pressure Locking Phenomenon

Pressure locking in flexible-wedge and double-disc gate valves generally develops because of valve design in combination with characteristics of the bonnet and specific local conditions at the valve (temperatures and pressures). The essential feature to develop pressure locking is the presence of fluid in the bonnet cavity, including the area between the discs. The fluid may enter the bonnet cavity during normal opening and closing valve cycle. Also, fluid may enter the bonnet cavity of a closed valve which has a ΔP across the disc. The pressure differential causes the disc to move slightly away from the seat, developing a flow path for fluid so that the bonnet cavity becomes filled with high pressure fluid. Whether these situations lead to a valve pressure locking scenario depends upon the pressure of the fluid that enters the bonnet cavity and the difference in pressure between the process fluid and bonnet cavity at the time the MOV is called upon to operate.

4.10.2.3 Consequences of Locking

These phenomena can delay the valve stroke time or cause the valve motor actuator to stall. Events at Susquehanna and FitzPatrick indicate that the RHR/LPCI and LPCS injection valves of a BWR are susceptible to pressure locking caused by bonnet cavity pressurization. In both systems the injection valve is normally shut and is require to automatically open upon receipt of an actuation signal. The testable check valve located between the reactor and injection valve is not a leak-tight valve. Leakage past the check valve can pressurize the piping between the valves and the injection valve cavity to reactor pressure. Near leak-tight seating surfaces of the injection valve may allow the valve cavity to remain pressurized and become subject to pressure locking when injection is needed during a LOCA. Under this condition, the bonnet pressure is greater than 1000 psig, while the downstream pipe suddenly depressurizes to between 400 and 500 psig. This high internal-to-external ΔP across both seating surfaces would result in double-disc drag forces, which if they exceed the available thrust of the actuator, will produce pressure locking.

When a valve disc becomes locked in the closed position due to pressure locking or thermal binding, actuation of the motor will result in locked-rotor current which will rapidly increase the temperature of the motor internals. Within 10 to 15 seconds, the heat buildup can degrade the motor's capability to deliver a specified torque, damage the motor, or both.

4.10.3 Mode 3/4 Event

Hope Creek is a BWR/4 plant rated at 3293 MWt and 1067 MWe with a Mark I containment. At the time of the event the plant was operating in an action statement requiring the plant to shutdown in seven days due to an inoperable control room ventilation component. The allowed operating time of seven days was approaching expiration so the plant had commenced a reactor shutdown. As part of the normal shutdown procedure the reactor was manually scrammed by placing the mode switch in the shutdown position. The plant entered operating mode 3 at 12:18 am on July 8, 1995. Table 4.10-1 lists the sequence of events and provides a detailed description of the event to conclusion.

By using the sequence of events, attached figures, and this text, answer learning objective 4 in this chapter.

4.10.4 Summary

The reader should be aware that the statistics presented herein are for Grand Gulf. As such, this information should not be generalized to other nuclear power plants without first considering all relevant factors. Complete details of the Grand Gulf statistics and insights can be found in SAND94-2949.

What can be generalized is the apparent change in dominant accident sequences from full power to low power and shutdown conditions. This is extremely important when you consider that technical specifications action statements usually require you to go to mode 3 or 4 within some time frame. The NRC felt so concerned about the apparent change in risk when entering modes 3 and 4 that they enlisted Sandia National Laboratories to evaluate the risk impact of the Limiting Conditions of Operation (LCOs) in the current Grand Gulf technical specifications. The results of the study were published in NUREG/CR-6166.

Table 4.10-1 Sequence of Events

July 8, 7:00 am	Operating Shift turnover
7:54	The B RHR pump was placed in service to establish shutdown cooling (SDC) in accordance with procedures. Indicated RHR flow was approximately 10,000 gpm.
7:54 to 9:40	The A and B recirculation pump discharge valves were stroked open and closed to prevent thermal binding in the closed position.
9:40	Nuclear controls operator unsuccessfully attempted to open the 'A' recirculation pump discharge valve.
9:50	Nuclear controls operator unsuccessfully attempted to open the 'A' recirculation pump discharge valve a second time. An action request was initiated to investigate and correct the valve failure.
10:57	Operating mode 4 is reached
11:00	The nuclear controls operator partially opened and left open the 'B' recirculation pump discharge valve to prevent thermal binding in the closed position.
11:52	Reactor pressure indicated zero pounds per square inch gage (psig) and the reactor vessel head vent valves to the equipment drain sump were opened in accordance with procedures.
12:59 pm	The electrical supply breaker for the reactor water cleanup supply line inside isolation valve (F001) was opened to support a corrective maintenance activity.
2:38	All high reactor pressure automatic isolation signals for the inboard and outboard shutdown cooling isolation valves were defeated. In addition, the isolation capabilty was defeated for the inboard valve. These signals were defeated in accordance with procedures, to prevent an inadvertent isolation and also in preparation for reactor protection system surveillance testing.
4:35	The shutdown cooling system was secured to facilitate manual operation of the RHR shutdown cooling isolation valves, per procedure, to verify that the valves could be closed manually. This is a precautionary step performed following defeat of the automatic signals.
5:09	Shutdown cooling was returned to service. The RHR heat exchanger inlet temperature promptly increased from 163 degrees to 182 degrees Fahrenheit.
5:30	Operators entered the drywell to perform outage activities, assess a drywell cooler leak and to investigate the reason for recirculation pump discharge valve failure.
5:54	Electrical supply breaker for the reactor water cleanup valves was reclosed.
6:45	Operators manually "cracked" open the 'A' recirculation pump discharge valve. Upon exiting the drywell, plant operators reported condensation on drywell surfaces and also fogging of their glasses. The nuclear controls operator opened 'A' recirculation pump discharge valve until he received an electrical dual indication.
7:00	Operating shift turnover

Table 4.10-1 Sequence of Events (Continued)

8:00 pm	The senior nuclear shift supervisor (SNSS) turnover completed, however, the on coming SNSS was involved with other activities and missed the shift briefing.
8:30	SNSS and NSS performed a control room panel walkdown and noted the 2000 gpm of recirculation system flow. They decided to shut the recirculation pump discharge valves.
8:45	The drywell primary containment instrument gas system was tagged out and depressurized in preparation for outage maintenance activities.
9:00	The nuclear controls operator closed the 'A' recirculation pump discharge valve after RHR heat exchanger inlet temperature decreased to 155 ^{o}F and the thermal binding limitation was no longer applicable. An attempt was made to also close the discharge valve for the 'B' pump, but was unsuccessful. The nuclear controls operator assumed this was due to some valve control interlock and decided to open the valve further and try again to close it.
10:00 to 11:00	The nuclear controls operator noted reactor pressure was indicating approximately 17 psig, but was not confident about the accuracy of the pressure indication at the low end of a 0 - 1500 psig meter. The electrical supply breaker for the RWCU F001 valve was opened in preparation for transferring the RPS system to its alternate power supply.
11:00	The operators noted that drywell floor drain leakage had increased to 1-2 gpm.
July 9, 00:30	RWCU valve F001 was returned to an operable status.
1:00 am	The nuclear controls operator noted that a shutdown cooling high pressure trip unit indicated 60 psig. The operator directed an instrument technician to accurately determine reactor pressure. The reading taken indicated pressure between 19 and 24 psig on all channels.
1:30	The operating crew decided to enter the drywell and identify the source of drywell leakage and to manually shut 'B' recirculation pump discharge valve.
4:29	The automatic isolation signals for shutdown cooling inboard and outboard suction valves were restored to normal.
4:49	The automatic isolation signals for shutdown cooling inboard and outboard isolation valves were again bypassed.
4:54	Shutdown cooling was secured and attempt was made to close 'B' recirculation pump discharge valve. In the attempt, the valve was fully opened.
5:00	SNSS and NSS discussed closing the 'B' recirculation pump suction valve as a contingency plan.
5:08	Shutdown cooling was restored. RHR heat exchanger inlet temperature increased approximately 7 ^{o}F.
5:50	'B' recirculation pump discharge valve was closed manually, RHR inlet temperature increased to 191 ^{o}F along with vessel bottom head temperature from 150 to 189 ^{o}F, in about 2 minutes. Reactor pressure started trending down toward zero.

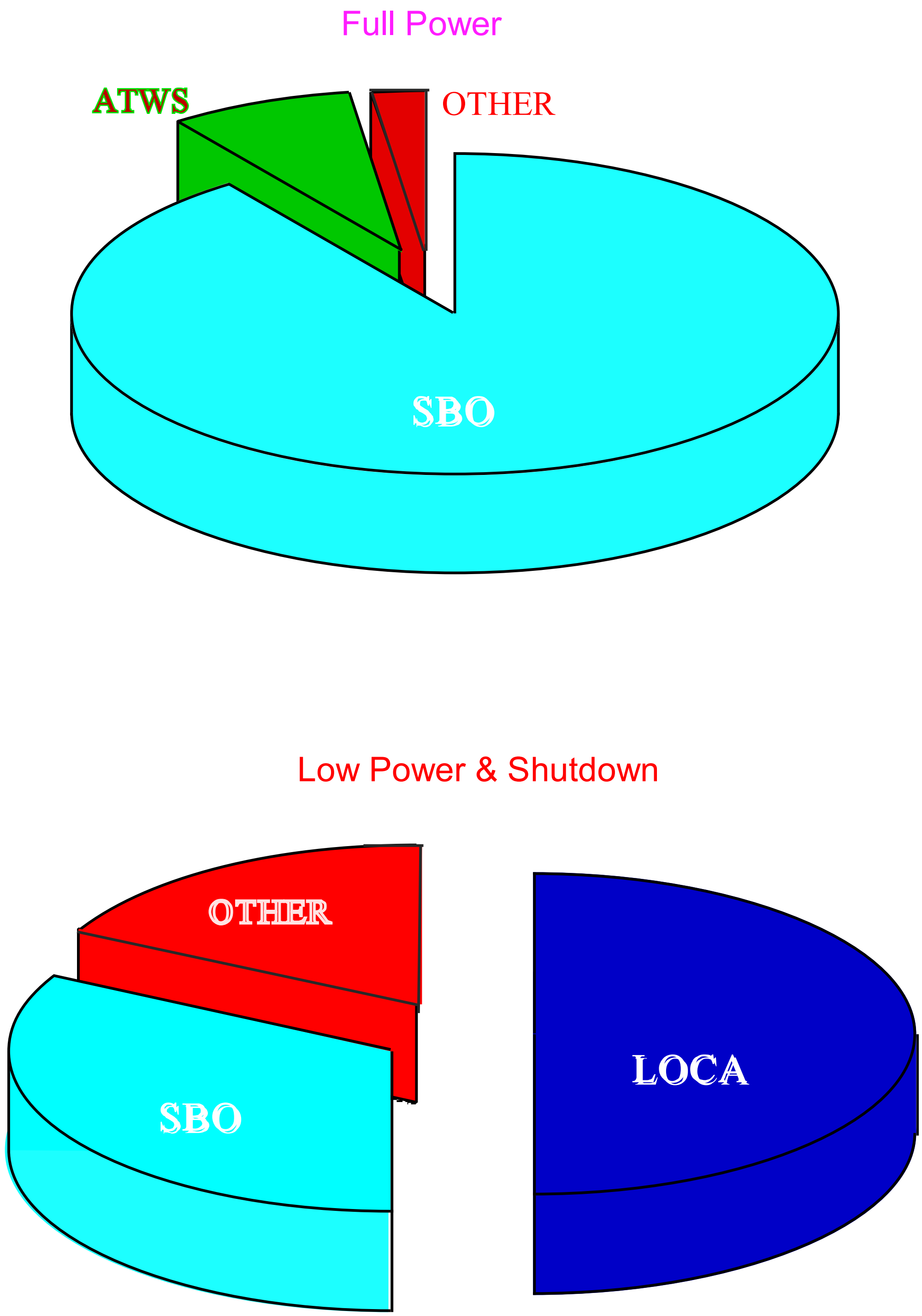

Figure 4.10-1 Accident Sequence Comparison

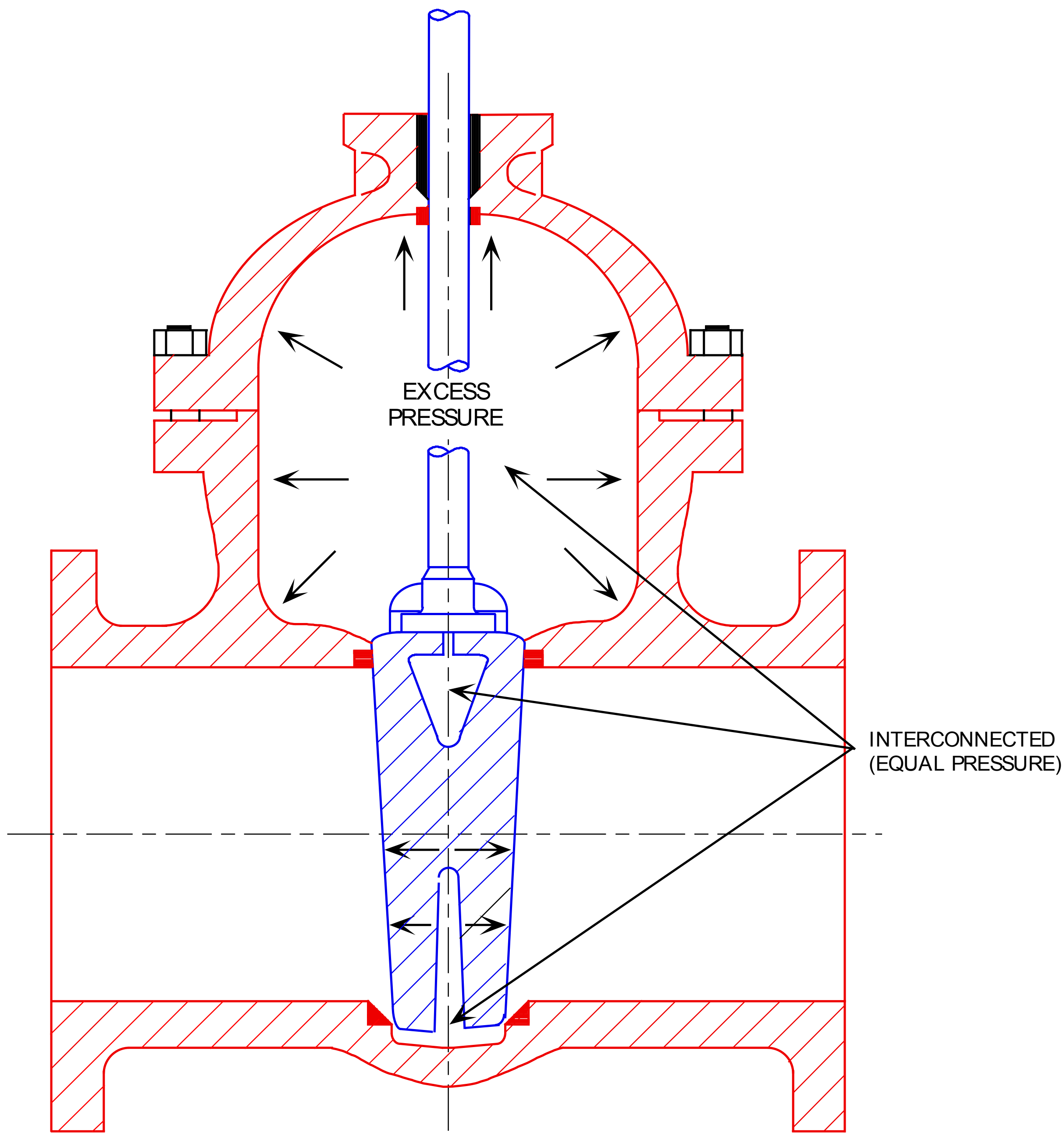

Figure 4.10-2 Pressure Locking Flexible-Wedge Gate Valve

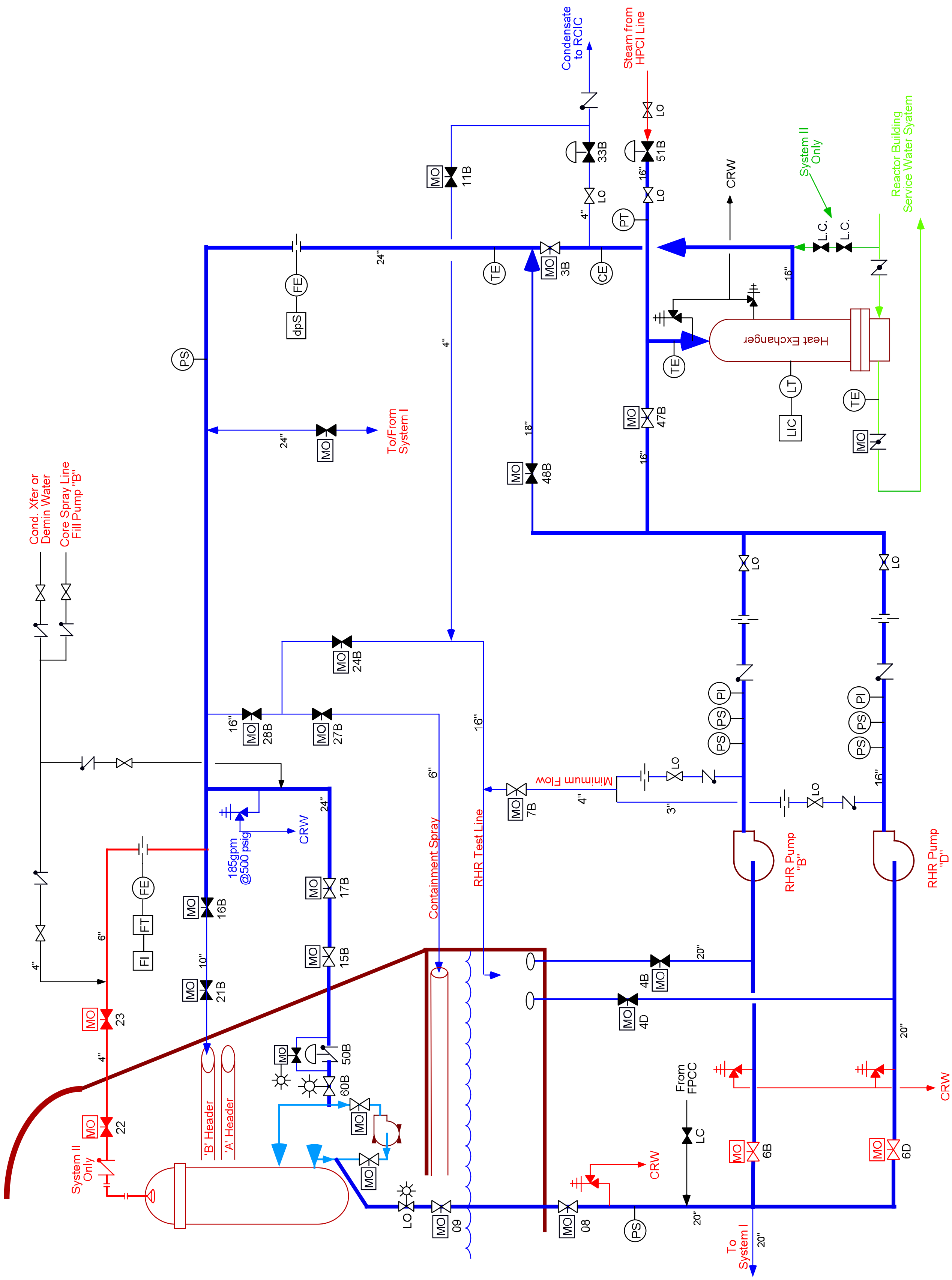

Figure 4.10-3 RHR System Shutdown Cooling Mode