

General Electric Advanced Technology Manual

Chapter 7.1

Brown's Ferry Fire Log Summary

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7.1 BROWN'S FERRY FIRE LOG SUMMARY

Learning Objectives:

1. List and explain the items learned from the event.
2. List the three major fire fighting systems.
3. List the pumps capable of supplying water to the reactor vessel at normal operating pressure.

7.1.1 Introduction

The Brown's Ferry Nuclear Plant consists of three BWR/4 units, each designed to produce approximately 1100 megawatts of electrical power. Units 1 and 2 were both operating at the time of the fire. Unit 3 was still under construction. At approximately 1235pm, March 22, 1975, a fire was reported in the unit 1 reactor building. The immediate cause of the fire, unknown to the operators, was the ignition of the polyurethane foam which was being used to seal air leaks in the cable penetrations between the unit 1 reactor building and a cable spreading room located beneath the control room of units 1 and 2 (Figure 7.1-1). The material ignited when a candle flame, which was being used to test the penetrations for leakage, was drawn into the foam by air flow through the leaking penetration.

Following ignition of the polyurethane foam, the fire propagated through the penetration in the wall between the cable spreading room and the unit 1 reactor building. In the cable spreading room, the extent of burning was limited and the fire was controlled by a combination of the installed carbon dioxide extinguishing system and manual firefighting efforts. Damage to the cables in this area was limited to about 5 feet next to the penetration where the fire started. The major damage occurred in the unit 1 reactor building adjacent to the cable spreading room, in an area roughly 40 feet by 20 feet, where there is a high concentration of electrical cables.

Approximately 1600 cables were damaged. There was very little other equipment in the fire area and the only damage, other than cables, cable trays, and conduits, was the melting of a solder joint on an air line and some spalling of concrete.

The electrical cables shorted together and grounded to their supporting trays or conduits after the insulation burned off. Sufficient equipment remained operational throughout the event to shutdown both reactors and maintain the reactor cores in a cooled and safe condition, even though all of the emergency core cooling systems for unit 1 were inoperable. No release of radioactive material above the levels associated with normal plant operation resulted from the event.

7.1.2 Fire

Starting about 5 minutes after the control room was first notified of the fire, alarms were received on the unit 1 control panel which contains the controls and instrumentation for much of the ECCSs. Comparison between the indications (alarms) revealed discrepancies. For example, one panel indicated all the ECCS pumps were operating, whereas another indicated normal reactor parameters with no need for such emergency operation. Intermittent and apparently spurious alarms continued at a lesser rate. Reactor power had by this time decreased from 1100 MWe to almost 700 MWe due to a decrease in recirculating pump speed from a cause unknown to the operator. At 1251 the recirculation pumps tripped and the reactor was manually scrammed. The rod position information system was operating at the time and all rods were verified to be fully inserted. Unit 2 was manually scrammed a short time later due to similar indication and control problems.

The operators still did not know the extent of the fire and its location was only generally defined. The operators did verify that there was no immediate threat to the safety of the reactors, but that the fire was affecting the emergency core cooling systems.

Normal cooldown was interrupted when the main steam isolation valves closed on unit 1 less than fifteen minutes after scram and on unit 2 less than ten minutes after scram. Although isolated from the main condenser, the plants could remain in hot standby by using the Reactor Core Isolation Cooling System (RCIC) provided for this situation.

Operation of the RCIC System was initiated on unit 2, but the system on unit 1 was disabled by the fire. The unit 1 RCIC had started earlier automatically, but was not needed then and was shutdown. When required later, it could not be restarted because of power failure to the isolation valve in the RCIC steam line which prevented it from opening and admitting reactor steam to the turbine. However, RCIC could also be driven by steam from the plant auxiliary boilers. The system is not normally connected to the boilers and this connection must be accomplished by inserting a spool piece between the RCIC turbine steam admission line and the auxiliary boilers. The spool piece had been used for startup tests and was made available for usage within an hour. With this capability in mind, the operators started an auxiliary boiler, and it was ready for use by 1330. However, the spool piece was not installed, as discussed later.

About forty minutes after scram, an operator stated that he knew that the unit 1 reactor water level could not be maintained with the running CRD pump. Also the only other available pumps could not inject water into the reactor at reactor pressures above 350 psig. After realigning the necessary valves in the feedwater lines and determining that two of the three condensate pumps and one of the three condensate booster pumps were running, the four unit 1 relief valves that could be manually operated from the control room were opened to lower reactor pressure. During blowdown, the water level dropped to

about 48 inches above the top of the core and then began to rise as pressure fell below 350 psig and the condensate booster pump started injecting water into the reactor. Within two hours after scram, conditions in unit 1 had stabilized. Water level was maintained using a condensate booster pump and steam was vented to the suppression pool through the manually actuated relief valves.

Before the unit 1 relief valves were opened at 1330 to depressurize the reactor, and again after 1800 when the relief valves could not be opened, the steam generated in the reactor core caused reactor pressure to rise slowly. When pressure exceeded 350 psi, the operating condensate booster pump could not inject water into the reactor. That left a single CRD pump injecting somewhat more than 100 gpm of water as pressure increased.

At high reactor pressure, automatic makeup is normally provided by the feedwater system. Backup is provided by either the steam driven HPCI or RCIC systems. On unit 1, the HPCI and RCIC were rendered inoperable by the fire shortly after their initial operation and were unavailable during the remainder of the event.

7.1.3 Source of High Pressure Makeup Water

Besides the CRD pump on unit 1, other installed sources of high pressure makeup were the CRD pump on unit 2, a shared spare CRD pump and standby liquid control (SLC) pumps. The CRD pumps, while performing normal functions associated with the control rod drive system, also provide water to the vessel at high or low pressure. One CRD pump per unit is normally in operation and the pump for unit 1 operated continuously throughout the course of the incident. In addition, the SLC pumps are each capable of providing approximately 56 gpm of water at pressures up to reactor coolant system design pressure. The SLC pumps were not required as a backup reactivity shutdown system since the control rods functioned normally.

An analysis of available evidence suggested that there was a period of up to three hours following initiation of the fire during which the SLC pumps were not available due to loss of power. However, the power for at least one pump is known to have been available at 1800 and the other was readily available or could have been made available, if needed, within 1 hour.

The operating CRD pump was part of units 1 and 2 system which consisted of three CRD pumps. One pump was normally operating in each unit and the third pump was available for use in either unit. Also, subsequent examination of the actual piping configuration confirmed that it was also possible to align the unit 2 pump to provide water to unit 1. Means also existed to increase the output of a CRD pump by valving in the pump test bypass line. It was estimated that by opening that single valve it would have been possible to provide approximately 225 gpm and maintain the core covered throughout

the course of the incident. No other systems would have been required to provide water to the reactor vessel and depressurization would not have been necessary. The 225 gpm flow could have been increased to 300 gpm by placing an additional CRD pump in service.

An additional source of high pressure water, mentioned previously as being unavailable due to fire damage, was the unit 1 RCIC system. The RCIC pump would have been capable of providing sufficient makeup flow (600 gpm) to the reactor vessel through the entire course of the incident if the decision had been to make it available. It appears that this system could have been in service within an hour after making the decision. The source of steam for the RCIC system would have been an auxiliary boiler which was used for testing the RCIC prior to plant operation. Two procedures were necessary to provide the steam path. First, the auxiliary boiler must be put into operation. Full steam pressure from this source can be obtained in less than one hour. The operators actually put the auxiliary boiler into operation by 1330, and it was available during the time the relief valves could not be opened. The second procedure installs a spool piece into the steam flow path from an auxiliary boiler to the RCIC turbine. This could have been accomplished in less than one hour. The operation of the RCIC would then have been possible from the backup control panel. However, the system was not actuated. Instead, actions to restore relief valve operability was accomplished in approximately 3-1/2 hours following which the reactor vessel pressure was once again reduced to below the shutoff head of the condensate booster pump and it was not used to provide makeup.

7.1.4 Possible Operator Actions

There were other courses of action which might have been taken by the operator in the event that remote manual operation of the relief valves was lost. Reactor pressure could have been allowed to increase to the set points of the relief valves with subsequent steam relief to the suppression pool. Adequate makeup was available from the CRD pump. Considerable time became available for other operator actions as decay heat subsided: two hours at 1330; at least 8 hours at 1800 p.m. The alternative sources of high pressure makeup water were available even if control air to the relief valves could not be reestablished.

Calculations, however, indicate that after 1900 no augmentation of CRD pump flow was necessary to maintain the plant in a safe condition. This was due to the availability of a depressurization and heat removal path via the main steam line drain valves to the condenser. Both of these valves were electrically inoperable as a result of fire damage. The operators, however, decided to return draining capability to the main steam lines and this was achieved at approximately 1900.

It was calculated that the quantity of steam being removed from the pressure vessel through main steam drain lines was great enough that reactor pressure would have leveled prior to reaching the relief valves set points. That lower equilibrium pressure condition would have reduced the head on the operating CRD pump such that the pump would have provided sufficient makeup flow to keep the core covered throughout the remainder of the incident.

7.1.5 Enforcement Items

The following apparent items of noncompliance were identified during the investigation:

7.1.5.1 Failure to Comply with 10CFR50.59

10 CFR 50.59, requires, in part, that records be maintained of changes to the facility to the extent that such changes constitute changes to the facility as described in the Safety Analysis Report. It further requires that these records shall include a written safety evaluation which provides the bases for the determination that the change does not involve an unreviewed safety question. The Browns Ferry FSAR Section 5.3.3.5 specifies, in part, that all electrical penetrations are sealed with sealant around conductors.

Contrary to this requirement, a safety evaluation was not made of the "change to the facility as described in the Safety Analysis Report" which was constituted by operation of the reactor with containment penetrations unsealed while concurrently sealing and testing the penetrations.

This infraction had the potential for causing or contributing to an occurrence related to health and safety.

7.1.5.2 Failure to Comply With Technical Specifications

The Technical Specifications, Sections 6.3.A and 6.3.B state, in part:

1. Detailed written procedures, including applicable check off lists covering items listed below shall be prepared, approved and adhered to:
 - a. Emergency conditions involving potential or actual release of radioactivity.
 - b. Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.
 - c. Surveillance and testing requirements.
2. Written procedures pertaining to those items listed above shall be reviewed by PORC and approved by the plant superintendent prior to implementation. . . Such changes

shall be documented and subsequently reviewed by PORC and approved by the plant superintendent." Contrary to these Technical Specifications requirements, the following problems were identified:

- a. The leak testing, sealing and inspection of the penetrations were being accomplished; but detailed written procedures approved by the plant superintendent and reviewed by PORC had not been developed for the control of this work.
- b. Persons discovering the fires on March 20 and March 22, 1975, did not adhere to the provisions of the Emergency Procedure in that they did not initiate the fire alarm.
- c. The Browns Ferry Emergency Procedure was not adhered to in that the Shift Engineer did not delegate on scene responsibility for fire fighting to an assistant shift engineer when he departed the fire area.

There were also problems identified with Technical Specifications compliance with plant procedures. For example, the Browns Ferry Standard Practices Manual specified, in part (in Standard Practice BFS3) that: "Plant fire protection systems shall be fully operational at all times. Removal of a plant fire protection system from service for any reason other than as required in a test procedure requires the approval of the plant superintendent. Removal of a system from service for more than seven days requires a review by PORC."

Contrary to these requirements, the fire protection system for the cable spreading room was not fully operational in that metal plates had been installed under the glass in the manual stations during the construction of the plant and had not been removed. The approval of the installation of the plates had not been documented prior to or subsequent to the issuance of the operating license and the installation had not been reviewed by PORC. Additionally, the CO₂ manual automatic initiation system had been electrically disabled by the construction workers without documented approval of the Plant Superintendent.

This infraction had the potential for causing or contributing to an occurrence related to safety.

7.1.5.3 Failure to Comply with Appendix B to 10 CFR 50

Criterion XVI of Appendix B to 10 CFR 50 and the related commitments in the FSAR, Appendix D.4, "Operational Quality Assurance Program Plan," Section D.4.2.4.7 specifies, in part, that measures be established to assure that conditions adverse to quality are promptly identified and corrected; that measures assure that causes of conditions be determined and action taken to preclude repetition; and that the corrective actions are documented and reported to appropriate levels of management.

Contrary to these requirements, during the penetration sealing operations, the conditions adverse to quality were not promptly identified and corrected; the causes of conditions were not determined and actions taken to preclude repetition; and the required documentation was not supplied in the two instances that fires were reported to management. This infraction had the potential for causing or contributing to an occurrence related to safety.

Criterion X of Appendix B to 10 CFR 50, requires, in part, that a program for inspection of activities affecting quality be established to verify conformance with documented instructions, procedures, and drawings; and that persons assigned the responsibilities for such inspections shall be independent of individuals directly responsible for work performance. Related commitments are spelled out in the FSAR, Appendix D.4., Sections D.4.2.3.1 and D.4.2.1.1, respectively.

Contrary to these requirements, inspections of the sealing of cable penetrations were not conducted so as to assure conformance with drawings; and plant inspectors were involved in the work activities for which they had inspection responsibilities.

Criterion XVIII of Appendix B to 10 CFR 50, and the related commitments set forth in the FSAR, Appendix D.4 "Operational Quality Assurance Program Plan," specifies, in part, that a comprehensive system of planned audits be carried out to assure compliance with all aspects of the quality assurance program.

Contrary to these requirements, a review of the records of the audits conducted at Browns Ferry and discussions with responsible individuals indicated that no audits had been conducted of the penetration installation.

Criterion III of Appendix B to 10 CFR 50, and the related commitments set forth in the FSAR, Appendix D.2, "TVA Quality Assurance Plan for Design and Construction," Section D.2.4.3.4., specifies, in part, that certain basic design drawings, such as single line diagrams, are reviewed to determine that they meet the design bases, design criteria and other design input requirements. The FSAR, Amendment 25, "Response to AEC Question 7.5," states, in part, that cables for the Engineered Safeguards Systems are separated into two redundant divisions (Division I or Division II) such that no single credible event could damage the cables of redundant counterparts. This section further states that power cables from the 4160-Volt Shutdown Boards are installed in separate conduits. It further states that the electric circuits of one of the two loops of the Core Spray System including the pump motors and electrically operated valve, are in Division I; and the circuits of the other loop are in Division II. Additionally, it states that the electric circuits associated with pumps A and C, and their valves, of the LPCI system are in Division I; and the electrical circuits of pumps B and D, and their valves are in Division II.

Contrary to this requirement, the power cable supplying 480 Volt Shutdown Board 1B from 4KV Shutdown Board C (Division II) was routed in the same tray as the power cable supplying 480 Volt Shutdown Board 2A from 4KV Shutdown Board B (Division I).

Contrary to this requirement, RHR Pump 1C and Core Spray Pump 1C were supplied from 4KV Shutdown Board B, and their associated valves were supplied from 4KV Shutdown Board C. Shutdown Board B is in Division I and Shutdown Board C is in Division II.

7.1.6 Lessons Learned

Lessons were learned from the Browns Ferry fire and changes have been implemented at BFNP and throughout the industry related to fire protection. Some of the lessons learned are listed below:

- Improve method for air leakage test.
- Provide better fire protection systems, equipment, and establish equipment storage areas.
- Establish fire protection procedures.
- Establish fire watch procedures.
- Establish better QA and documentation of cable arrangement.
- Provide better power supply separation.
- Establish procedures to delineate the person in charge of control room and fire fighting.
- Establish better regulations concerning fire protection and prevention.
- Standardize fire protection equipment.

Table 7.1-1 Sequence of Events

TIME	SUMMARY
1200	Units 1 and 2 were operating at 100% power.
1235	A fire was reported in the unit 1 reactor building. Plant personnel were informed of fire location. The fire brigade began fire fighting activities.
1240	Some ECCS annunciators alarmed and all diesel generators started despite normal vessel water level, steam pressure, and drywell pressure.
1242	The RHR System aligned itself in LPCI mode with all four pumps running. All four Core Spray System pumps aligned and started. The unit operator manually secured all eight pumps but could not reset annunciators.
1244	The RHR and CS pumps restarted for no apparent reason. The unit operator was unable to stop the pumps from the control panels.
1248	The unit operator was able to stop RHR and CS pumps from the control panels. Both recirculation pumps ran back for no apparent reason causing unit power to go from 1100 MWe to 700 MWe. Electrical boards started losing power. Half of RPS was lost. Remote manual control of seven SRVs was lost. Numerous annunciators alarmed on several control panels.
1251	The unit operator started lowering recirculation pump speed. Both recirculation pumps tripped for no apparent reason at >20% loading. The unit operator manually scrambled the reactor.
1253	The unit operator confirmed that all rods were fully inserted. The unit operator tripped all but one RFP, secured extra condensate and condensate booster pumps, and manually started RCIC as a backup.
1254	The assistant shift engineer (ASE) tripped the turbine, opened the generator field breaker, and opened the motor operated disconnects. HPCI initiated automatically although the MSIVs did not close. Vessel water level increased to the normal range. Operators secured both HPCI and RCIC.
1255	120VAC unit preferred power was lost. Among other things control rod position indication and all neutron monitoring indication was lost.
1256	<p>Numerous electrical boards were lost:</p> <ul style="list-style-type: none"> • 250Vdc Reactor MOV Board 1A • 250Vdc Reactor MOV Board 1B • 480Vac Reactor MOV Board 1A (CS, RHR, HPCI, RHRSW valves, RPS A) • 480Vac Reactor MOV Board 1B (CS RHR, RHRSW, RCIC, EECW valves, RPS B) • 480Vac Reactor MOV Board 1C (EECW, RHR valves) • 480Vac Shutdown Board 1A • 480Vac Shutdown Board 1B <p>At this point the MSIVs closed placing the unit in isolation from the main condenser and securing steam to the RFP turbine. Also all ECCS was lost with the exception of four SRVs which could be operated from panel 9-3.</p>

Table 7.1-1 Sequence of Events

TIME	SUMMARY
1258	Reactor pressure rapidly increased; SRVs began opening and closing to maintain pressure between 1080 and 1100 psig. An unsuccessful attempt was made to open MSIVs from the backup control center. The unit operator manually opened SRVs for which he had control and reclosed them after pressure had decreased to <850 psig. Pressure began increasing immediately.
1259	Reactor water level was decreasing due to almost constant SRVs blowdown to the torus. HPCI and RCIC were both inoperable due to the previous loss of valve controls. The only pump left with the capability to overcome a pressure above 350 psig was the CRD hydraulic pump. Torus cooling became vital due to heat added as a result of SRVs blowdown; but the RHR system was not available as a result of electrical board losses.
1300	4kV shutdown bus 2 lost power. This caused shutdown boards C and D to transfer to D/G's C and D.
1320	4kV shutdown bus 1 lost power. This caused shutdown boards A and B to transfer to D/Gs A and B.
1321	The process computer was lost.
1330	The decision was made to depressurize the reactor by SRV blowdown. The unit operator manually opened four SRVs. Reactor pressure and water level began decreasing. At this time a condensate booster pump was running and the RFP bypass valve was open.
1334	4kV shutdown bus 2 was energized from unit 2.
1345	Unit preferred power was restored from Unit 2. Reactor pressure had decreased to 350 psig allowing the condensate booster pump to start injecting. Reactor water level had dropped from a normal 201 inches above the top of the active fuel to 43 inches above the top of the active fuel. Reactor water level began to increase.
1355	Reactor water level was approaching normal but the operator lost the ability to control the RFP bypass valve.
1357	An operator was able to manually close the RFP bypass valve.
1400	The unit operator was maintaining reactor pressure at <200 psig using SRVs and vessel level about normal using the CRD hydraulic pump and a condensate booster pump. 480VAC shutdown boards A and B were restored.
1500	Attempts were made to align one RHR system for torus cooling and the other for shutdown cooling.
1600	RHR System I was aligned for torus cooling; but the decision was made not to start the RHR pumps in this condition since it was not established that the system was charged with water.
1630	480VAC reactor MOV board 1A was re-energized. This allowed placing the main turbine on the turning gear, restoring power to ECCS valves fed from that board, and energizing RPS A which restored half of the process monitoring.

Table 7.1-1 Sequence of Events

TIME	SUMMARY
1800	SRVs previously operable from the control room became inoperable due to loss of instrument and control power to a solenoid in the air supply to a diaphragm valve in the air header to the primary containment. The drywell air compressor was started but its isolated discharge prevented air flow to the primary containment.
1900	Power was restored to the main steam line drain valves.
1945	The fire was reported out.
2008	High torus level was reached due to SRV blowdown. The assistant unit operator aligned and started the RHR drain pump to the main condenser hotwell.
2040	Drywell venting via SBGTS was started with 2.5 psig drywell pressure.
2150	A control air supply to primary containment equipment was established making four SRVs operable from the control room. The operator began lowering reactor pressure again starting from 580 psig.
2200	Reactor pressure was 340 psig and decreasing slowly.
2250	Secondary containment was established. Reactor pressure was 200 psig with vessel water level being controlled at 36" by a condensate booster pump.
0000	The need existed to flush RHR System II prior to placing it in shutdown cooling. A temporary flushing procedure had to be developed since the existing procedure could not be used.
0100	Two source range monitors were placed in temporary service on the reactor side of the fire. This established the capability to monitor the core with 10 cps indicated on each SRM.
0130	Torus cooling was initiated using RHR System I
0245	Restoration of equipment had progressed to the point that core spray pumps A and C were operable from the control room.
0410	Shutdown cooling was established using RHR System II.
0930	The unit reached the cold, shutdown condition. Drywell temperature was 120°F. Torus temperature was 130°F and level +4".

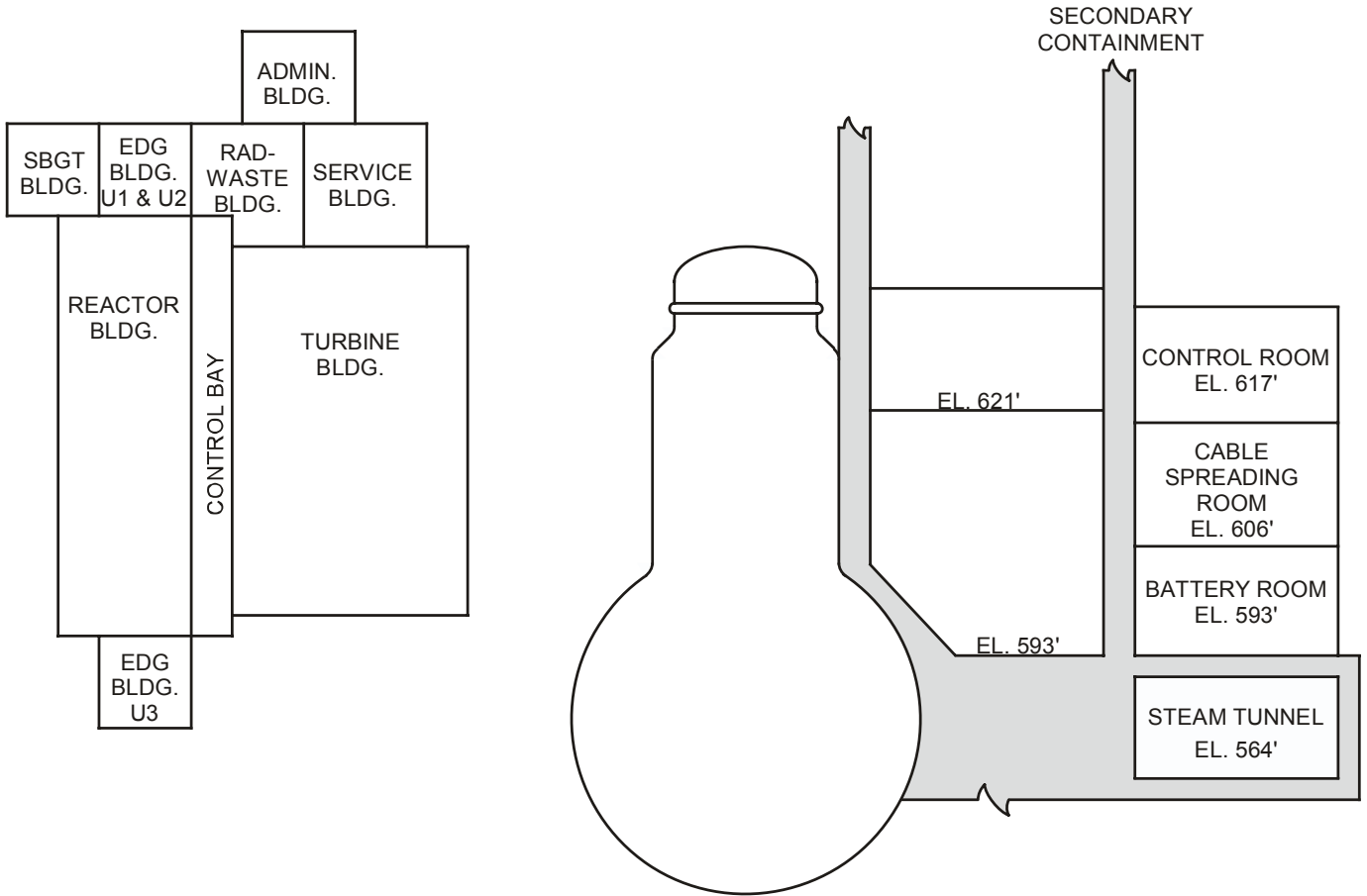
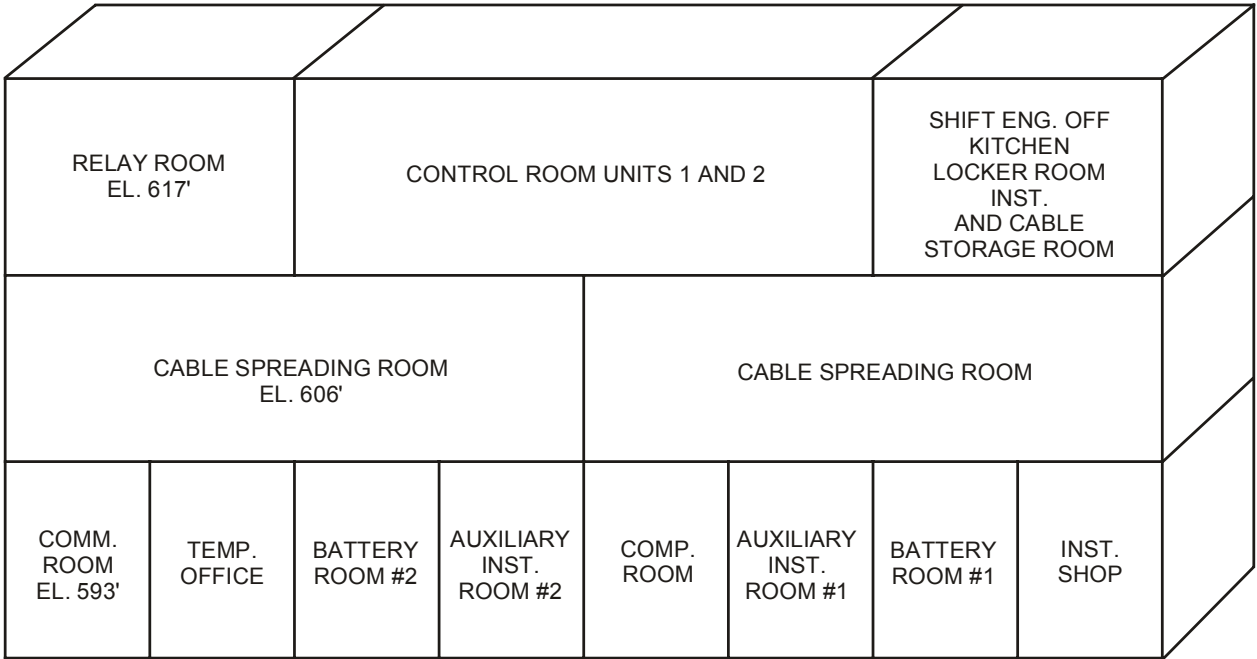
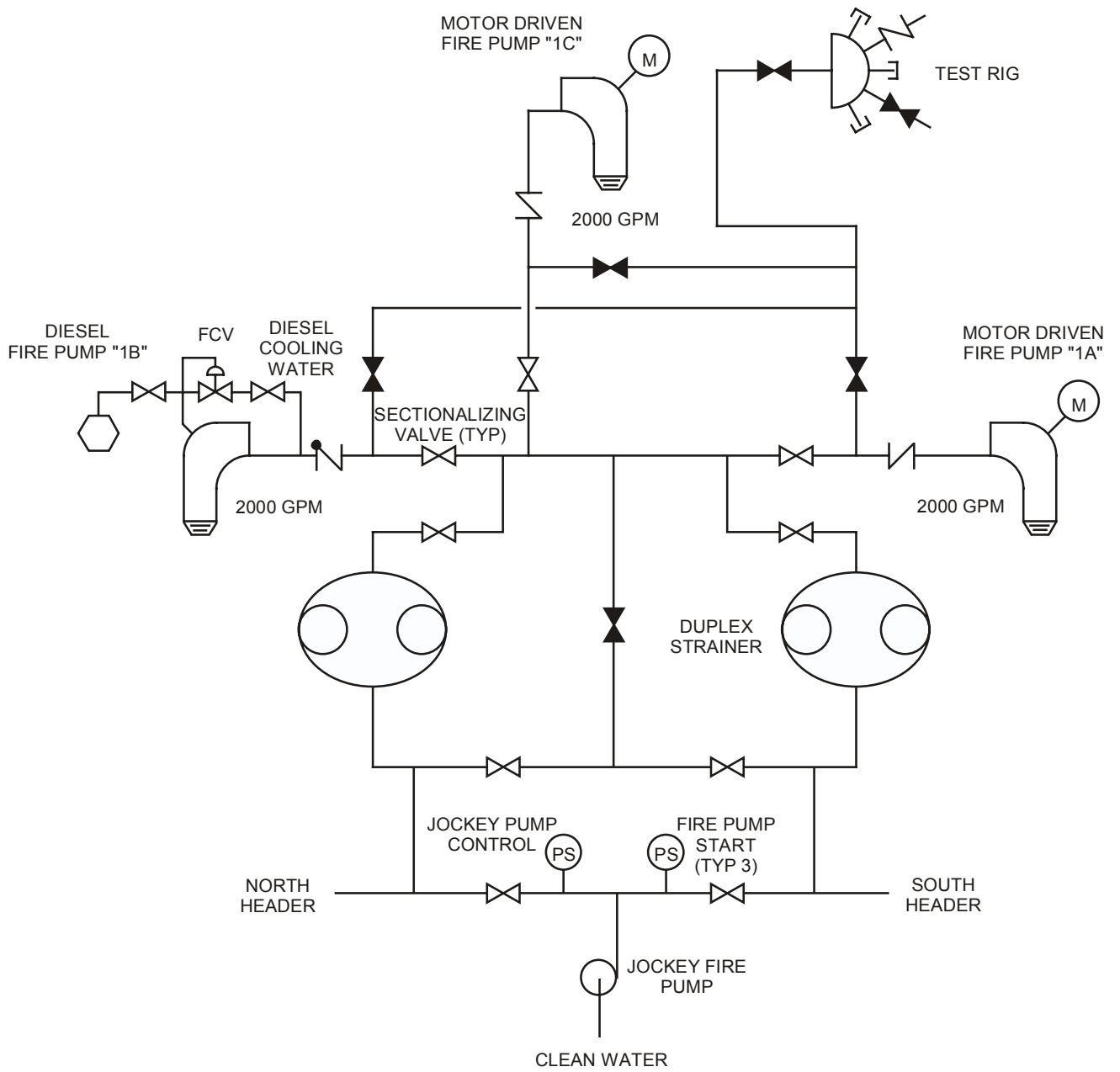


Figure 7.1-1 BFN Control Bay Arrangement



TYPICAL START SEQUENCE

JOCKEY PUMP—100 PSIG

FIRE PUMP 1A —80 PSIG

FIRE PUMP 1B —75 PSIG

FIRE PUMP 1C —68 PSIG

Figure 7.1-2 Fire Water Header

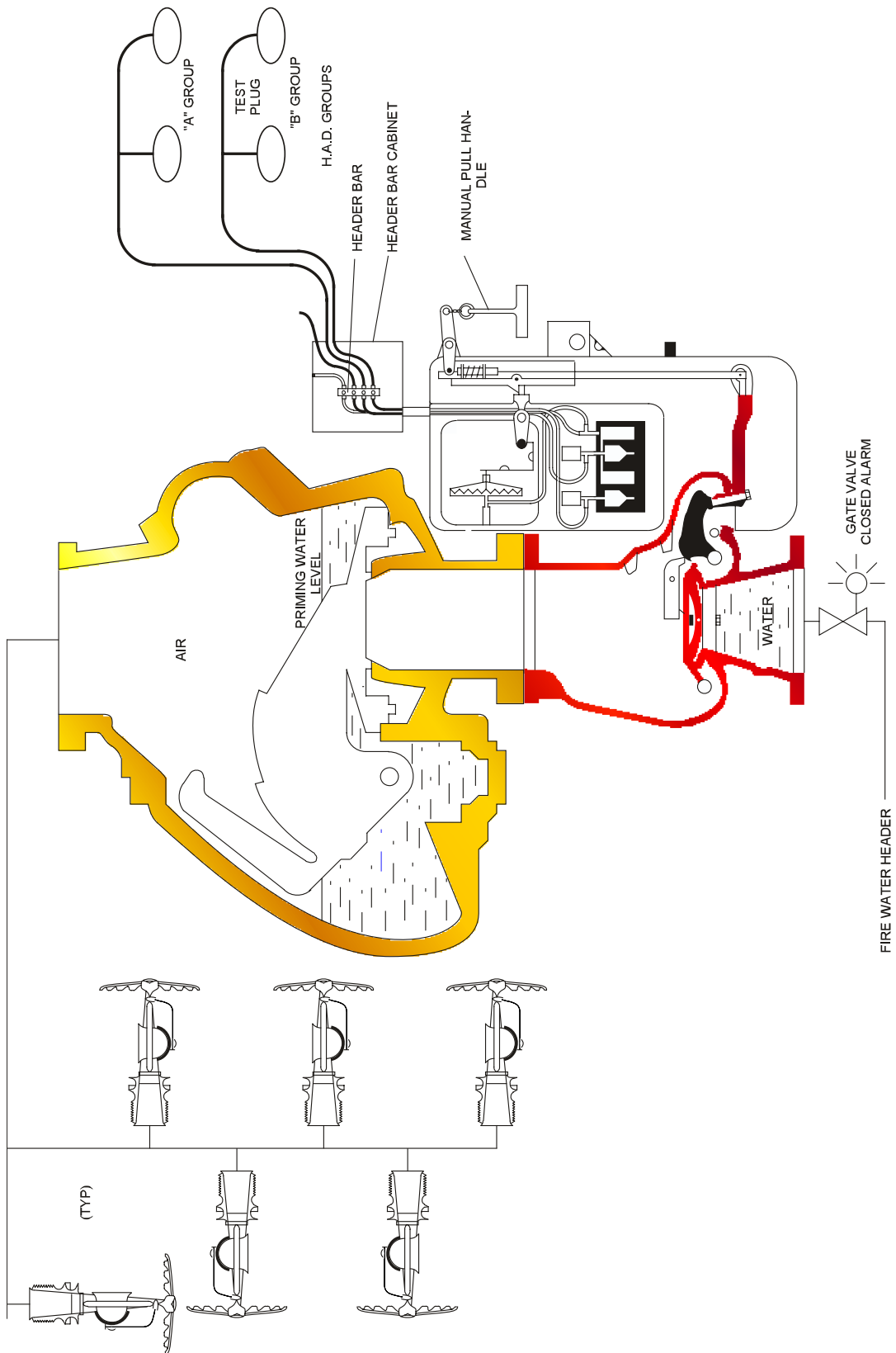
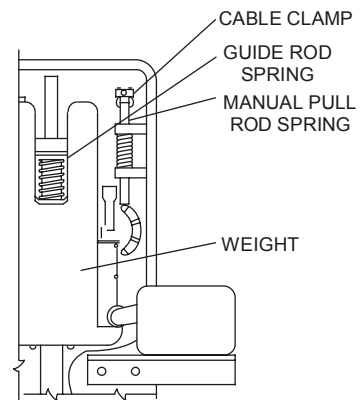
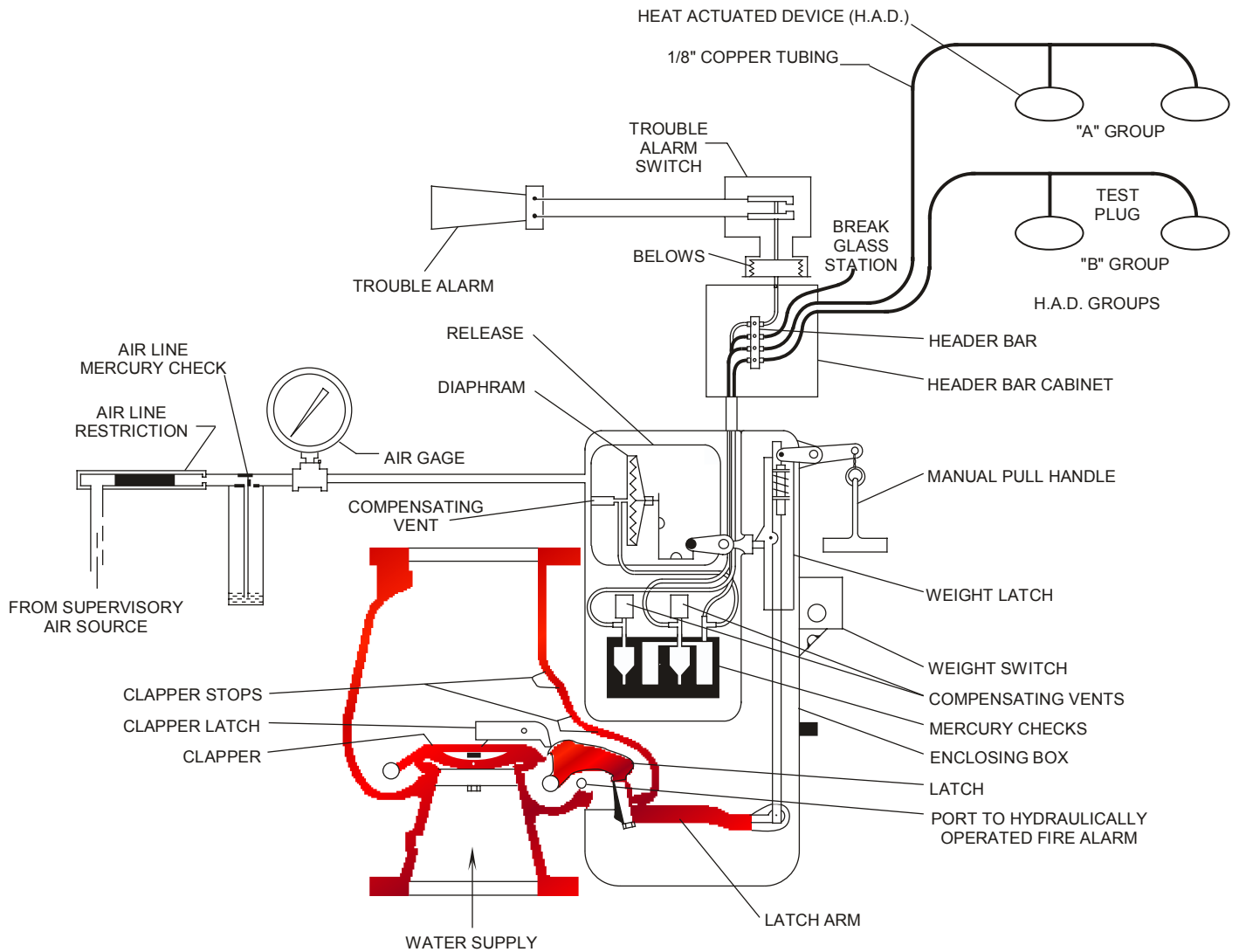
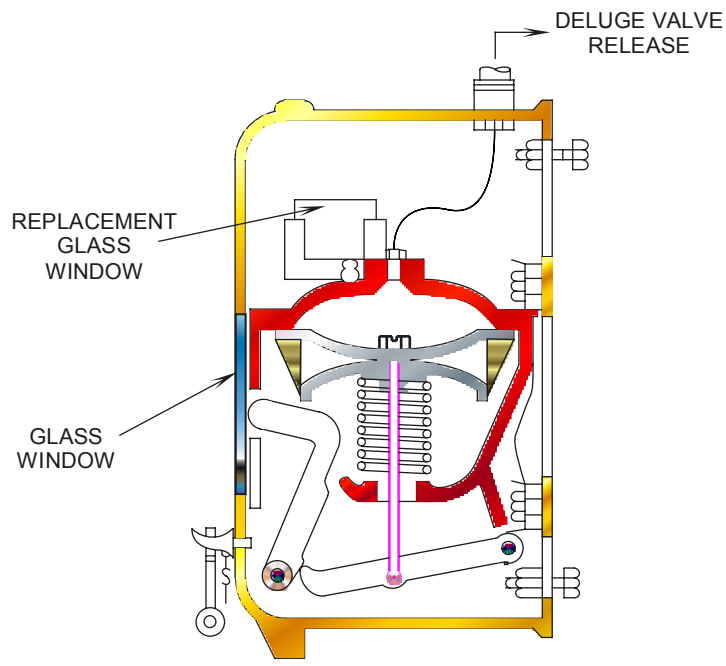


Figure 7.1-3 Preaction Sprinkler System

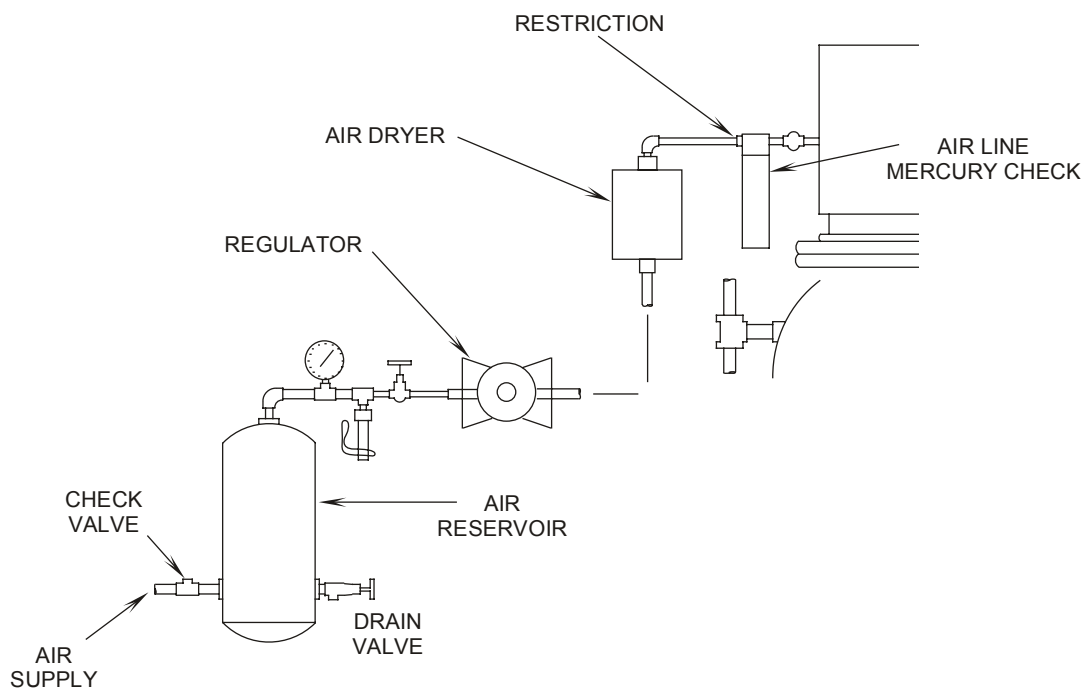


SECTION THRU WEIGHT SWITCH

Figure 7.1-4 Preaction System Deluge Valve



BREAK GLASS STATION



SUPERVISORY AIR SUPPLY

Figure 7.1-5

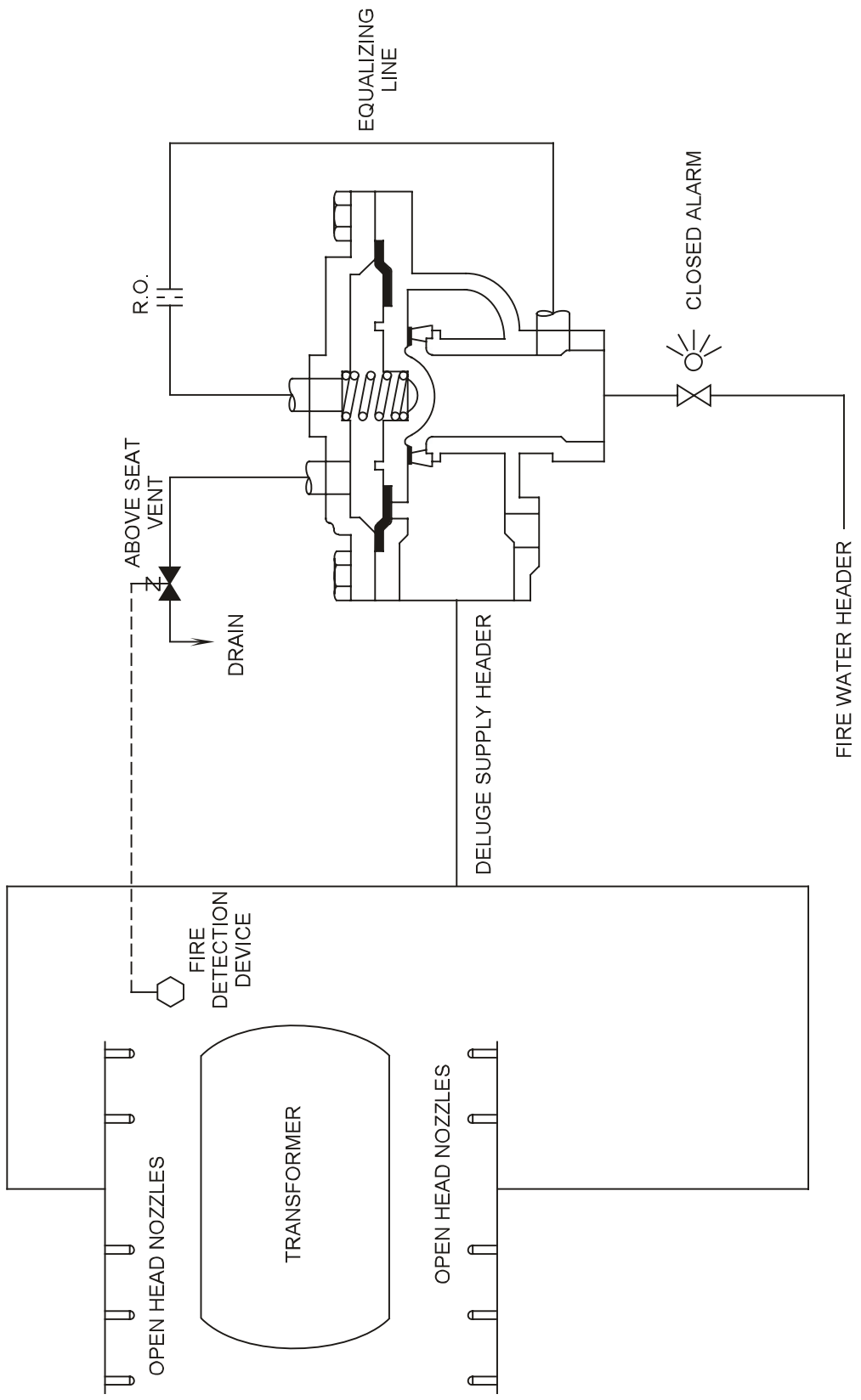


Figure 7.1-6 Deluge System

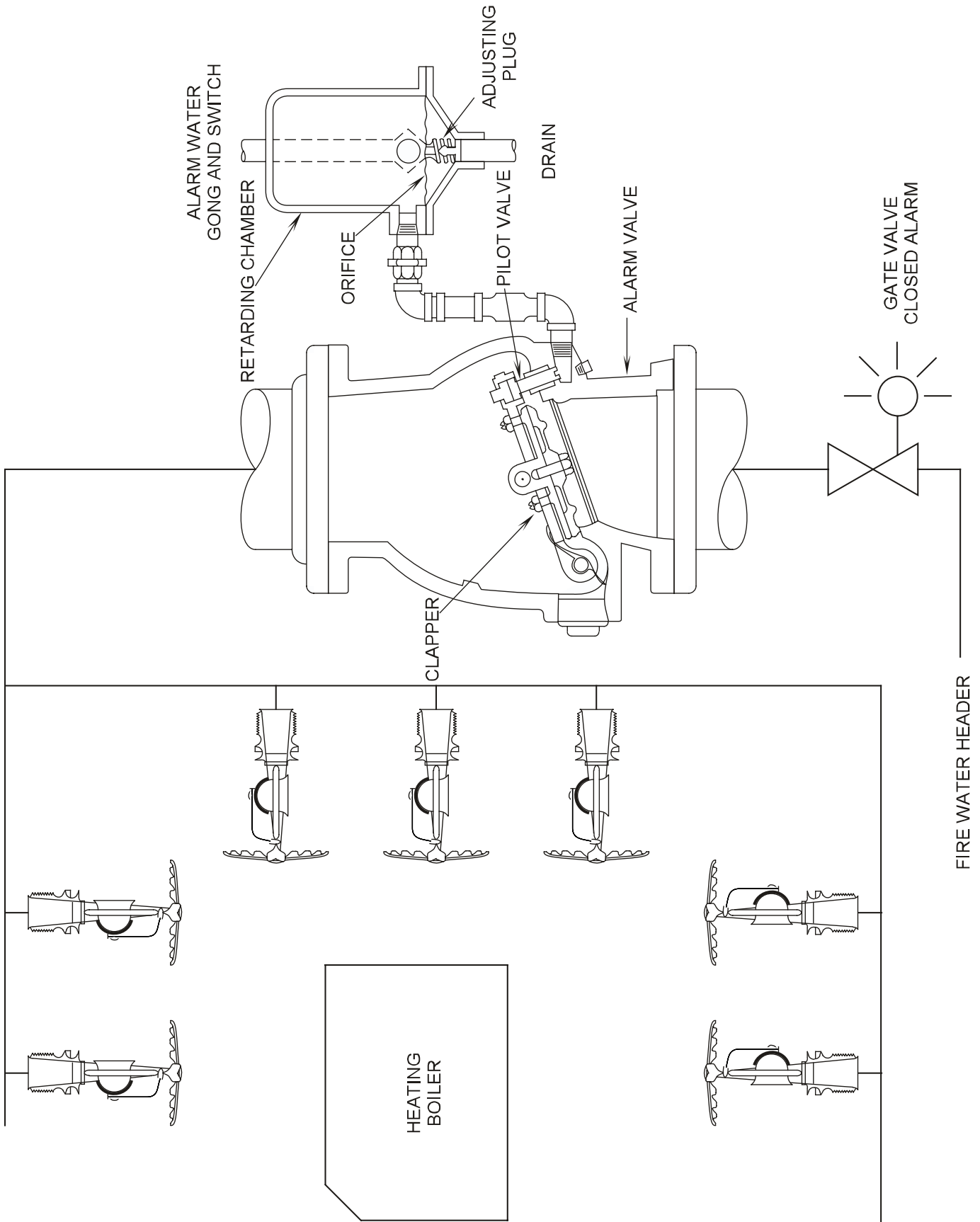


Figure 7.1-7 Wet-Pipe Sprinkler System

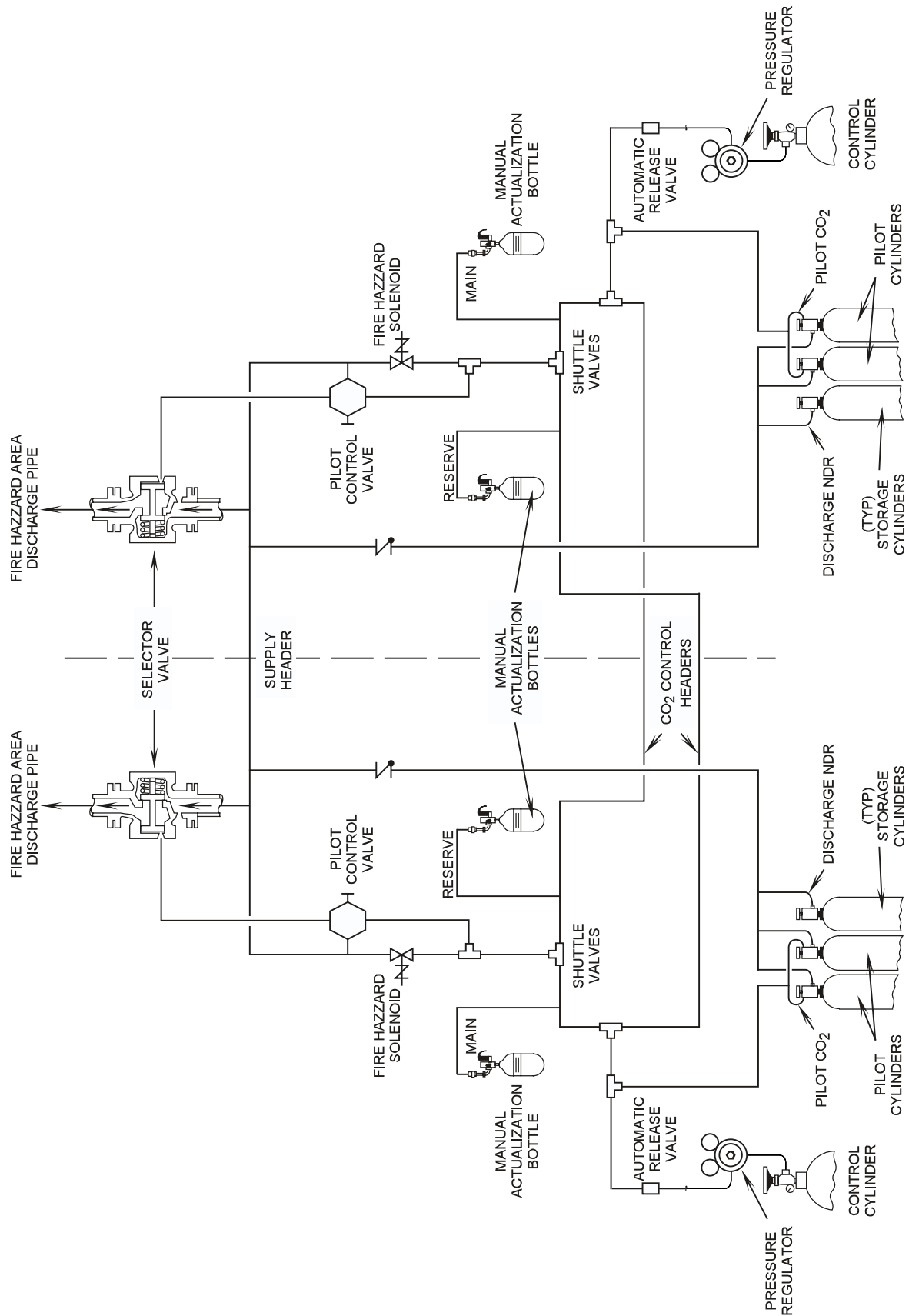


Figure 7.1-8 High Pressure Carbon Dioxide System

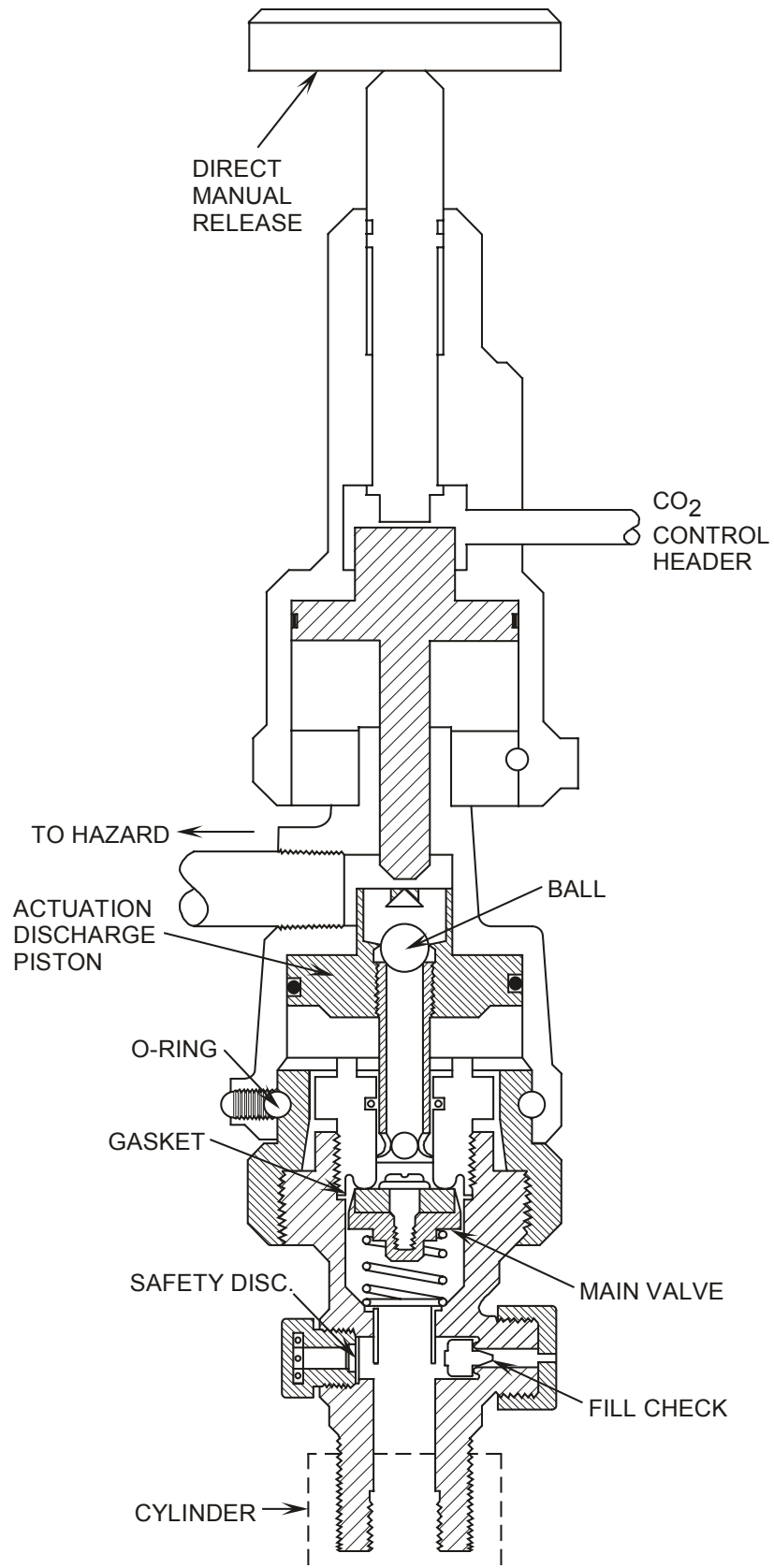


Figure 7.1-9 Pilot Control Cylinder Discharge Head

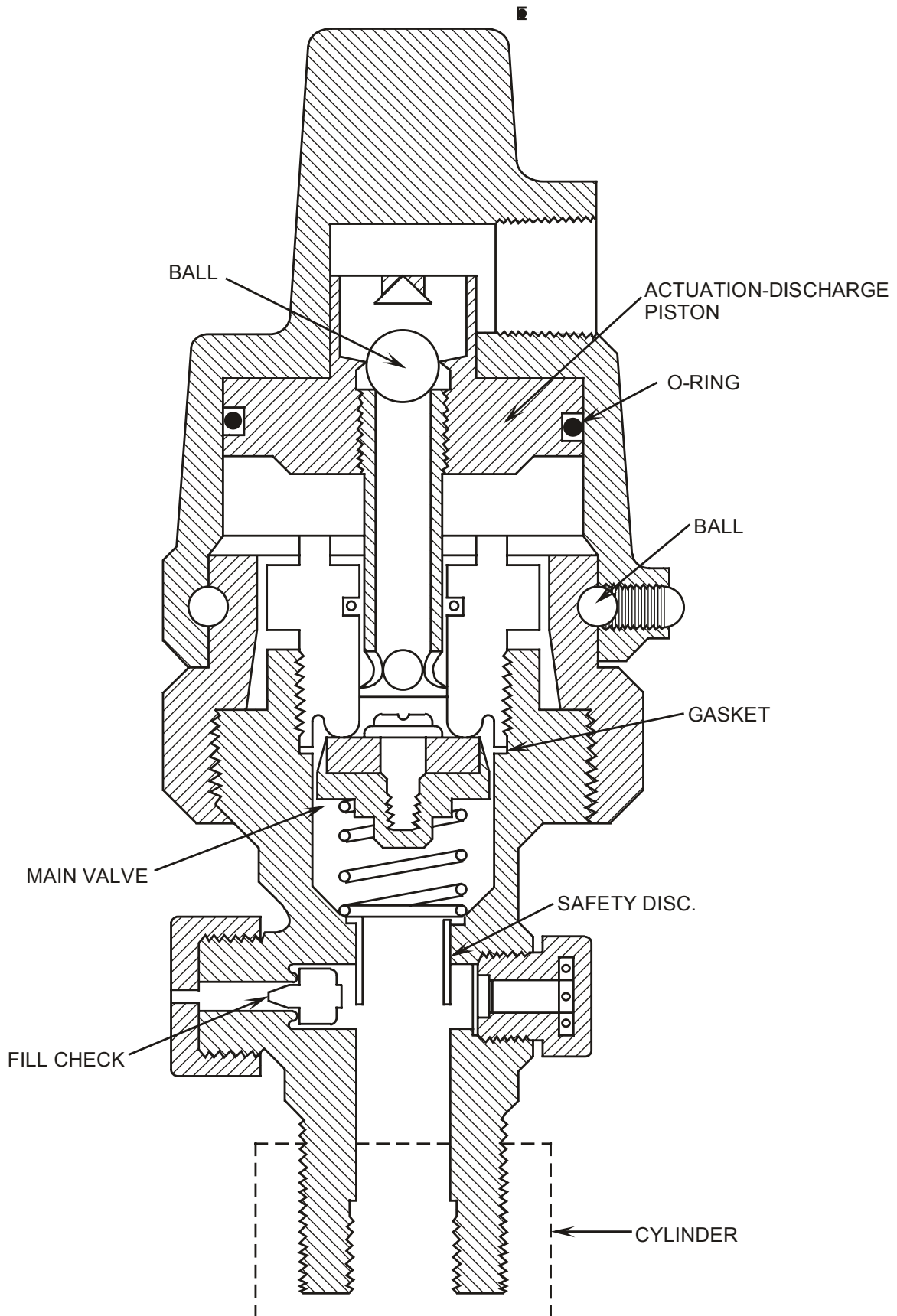


Figure 7.1-10 Actuation/Discharge Head Assembly

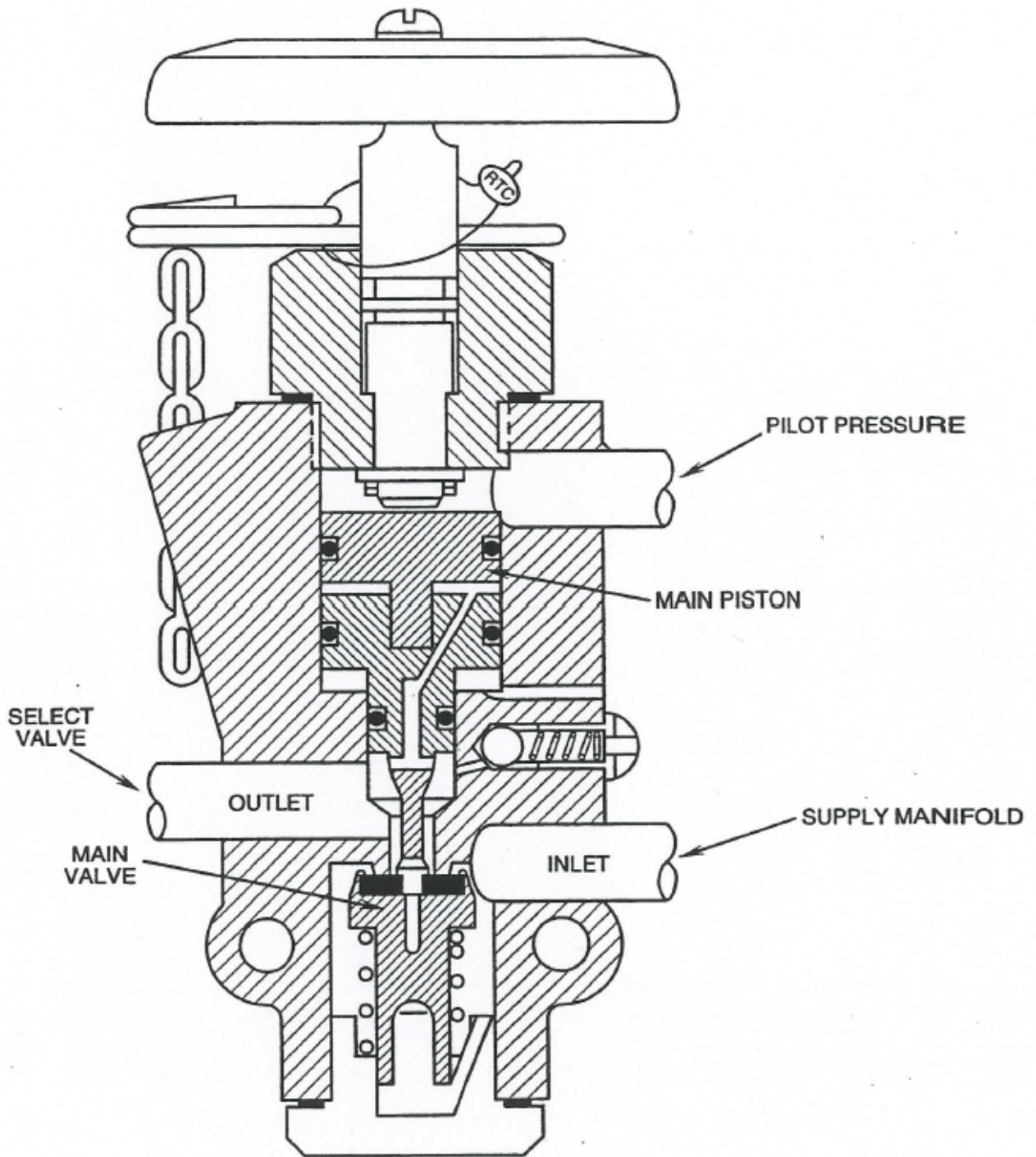


Figure 7.1-11 Pilot Control Valve

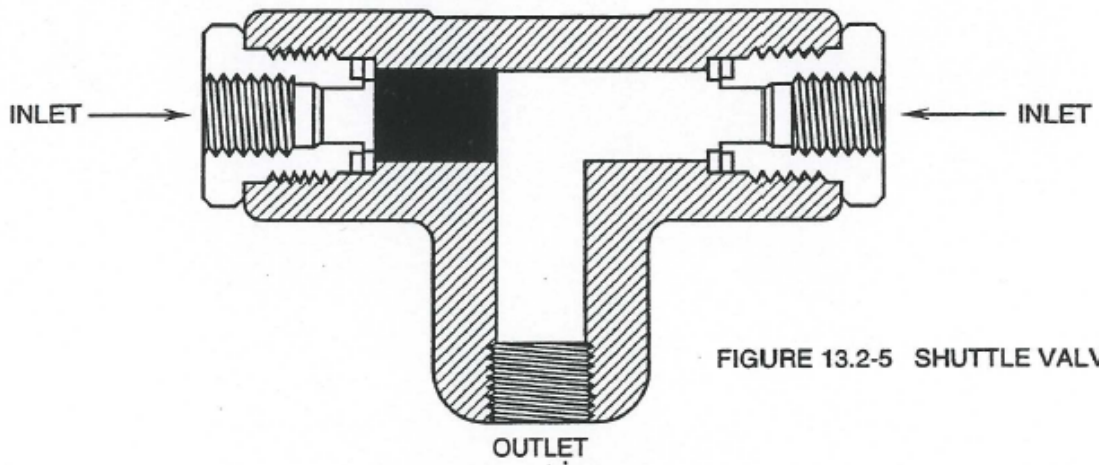


FIGURE 13.2-5 SHUTTLE VALVE

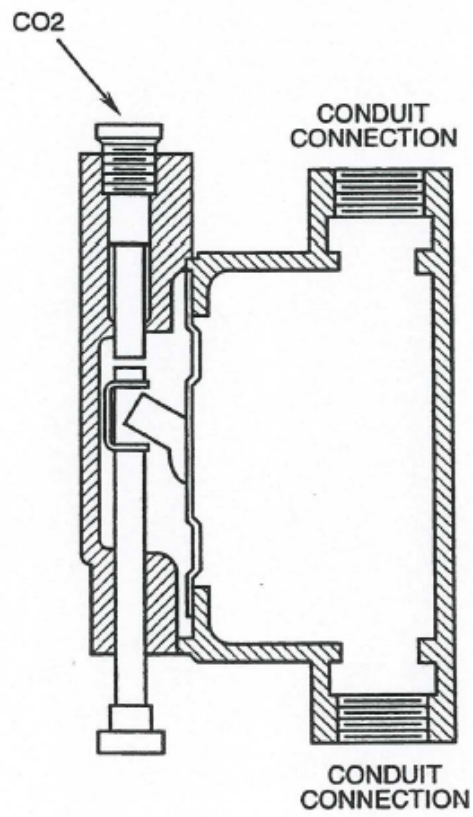


Figure 7.1-12 Cardox Pressure Switch

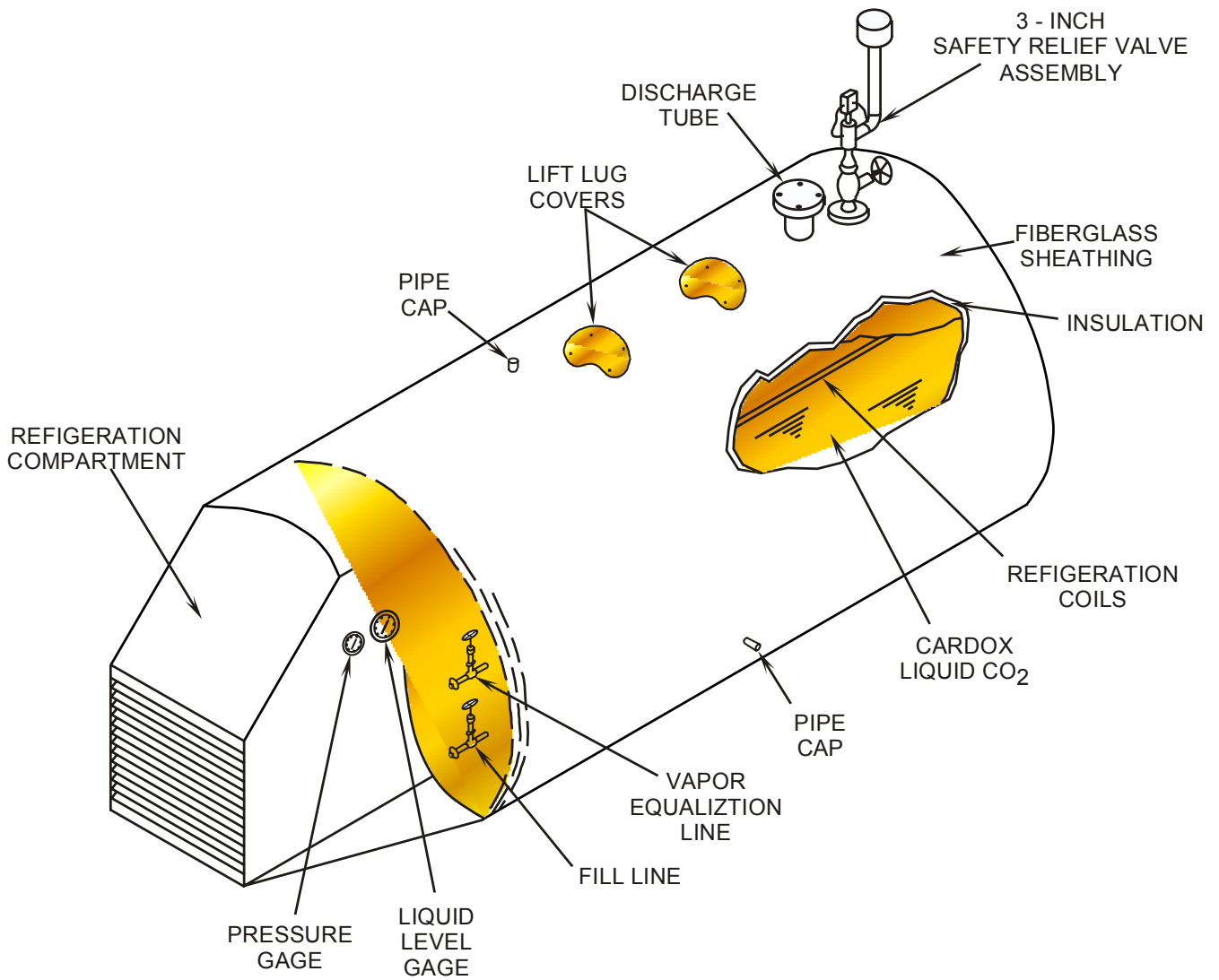


Figure 7.1-13 Low Pressure Carbon Dioxide Storage Unit

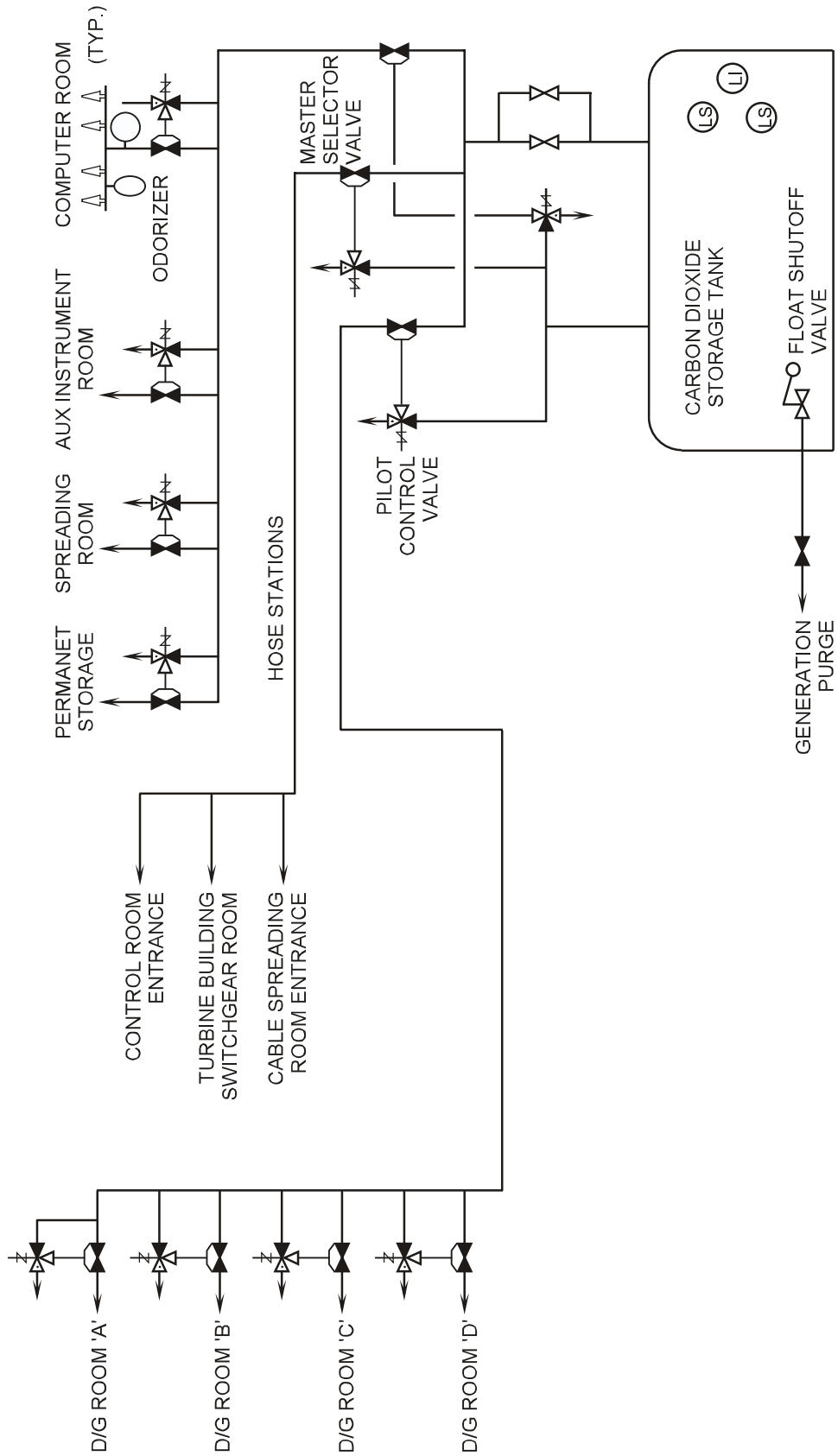


Figure 7.1-14 Low Pressure Carbon Dioxide System

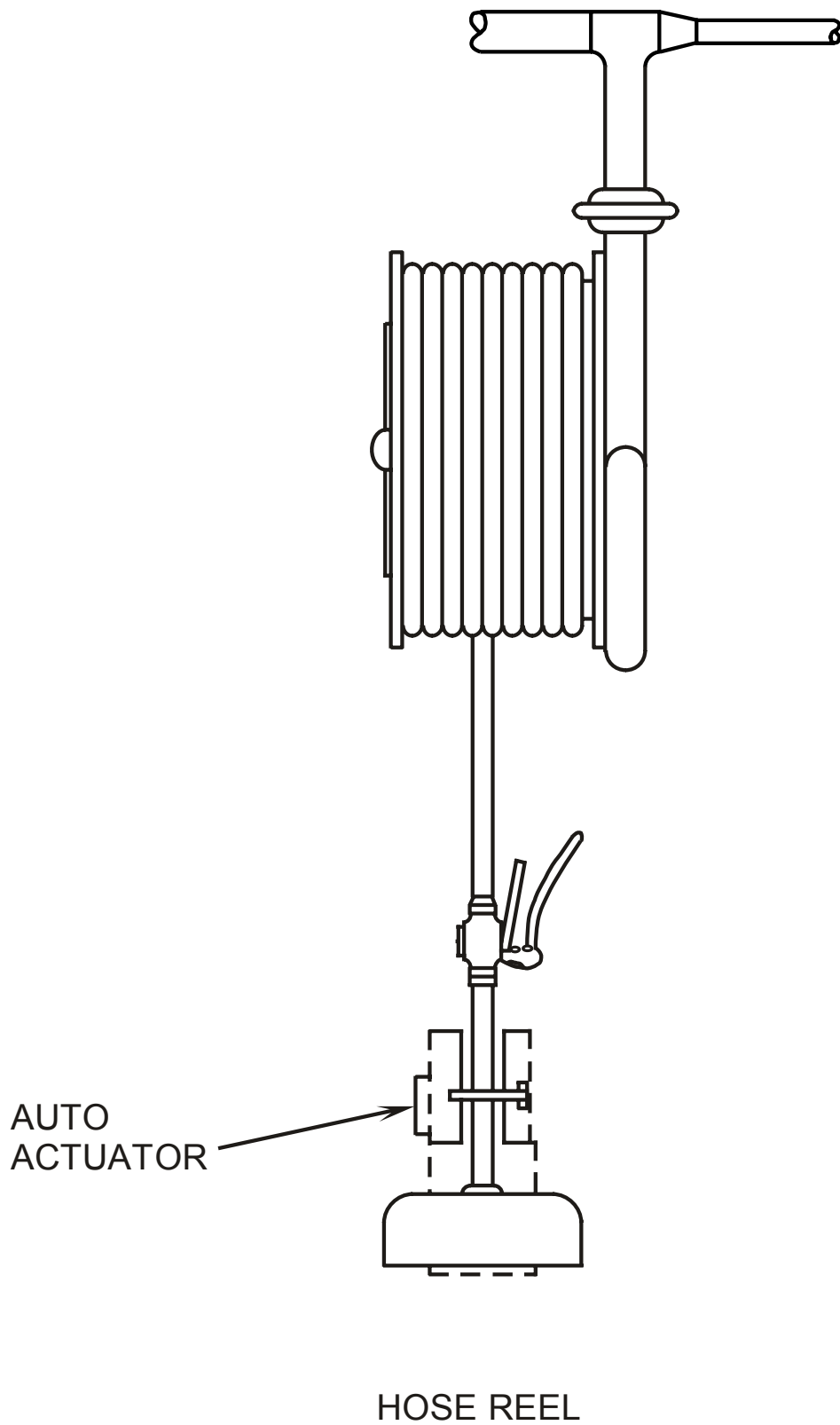


Figure 7.1-15 Hose Reel Station