Fukushima - a review 10-years after

Boiling Water Reactors cf Pressure water Reactors

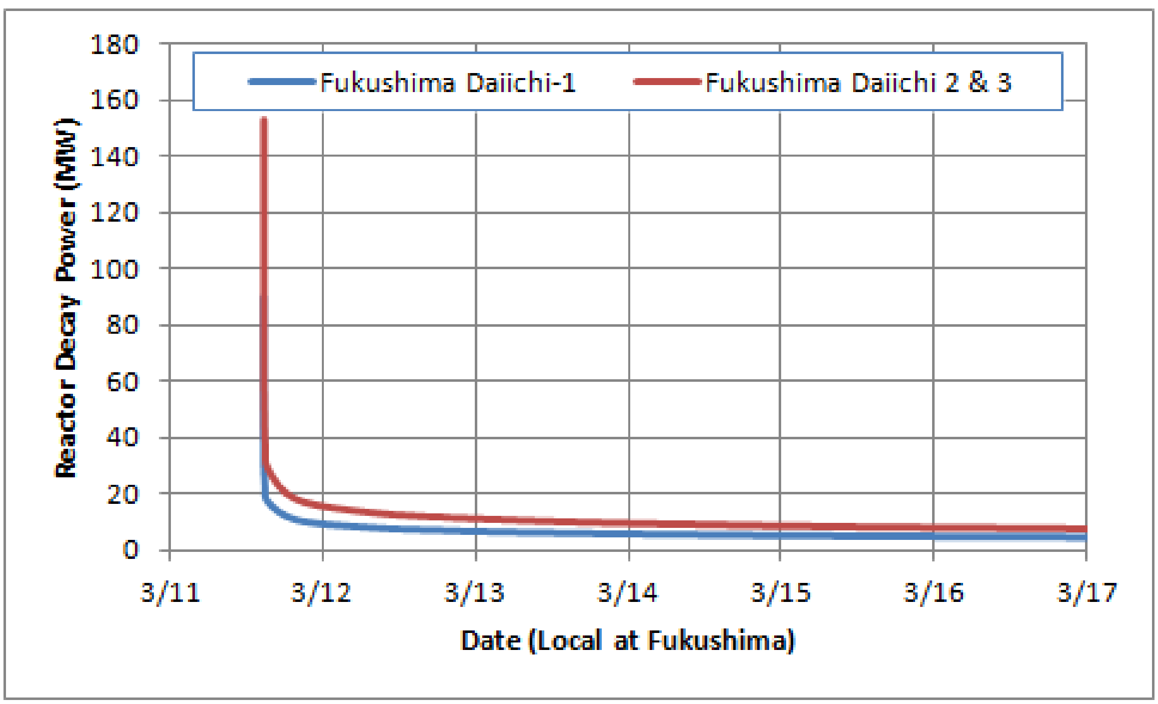
The reactors in Fukushima Daiichi were Boiling Water Reactors (BWRs) are different to the Three Mile Island Pressure Water reactors (PWRs) in that they have no steam generators. So, the top half of the reactor vessels has the steam, which is sent directly to the generator turbines. This means that the control rod drive mechanisms (CRDMs) are located at the bottom of the reactor vessels. The control rods are driven up against the reactor pressure by nitrogen pressurised accumulators and water scram accumulators, individually mounted below each of the CRDMs. The number of CRDMs varies with the BWR version, but the diagram examples found for this analysis have 137.

Fukushima Reactor Units 1, 2 and 3

Reactor units 1,2 and 3 successfully shut down automatically when the earthquake struck on 11 March 2011 at 14.16 local time and external power was lost. Emergency diesel generators started up automatically, but with the arrival of the tsunami at 15.36 stopped working. The batteries, which provided power for the station controls on Unit 1 ran out on 11 March, but remained active for Units 2 and 3 until three days later on 14 March.

With the insertion of the control rods, which in a BWR are raised from the bottom of the reactor vessel, the fission power would have been reduced to just 7% of the full power and then would have slowly decayed. See Figure 1

Figure 1



There was residual heat removal for an hour and 20 minutes before the standby generators stopped, while further removal by steam turbine-driven pumps in Units 2 and 3 until the controls were lost when the batteries ran out.

Fukushima compared to TMI 2

At Three Mile Island Unit 2 (TMI 2) the lack of removal of the residual heat resulted in the partial meltdown of the core, while at Fukushima although access to inspect was lost by lethal, high levels of radiation, it is believed that around 80% of the cores melted and went through the bottoms of the reactor vessels, while the buildings were destroyed by hydrogen explosions.

In comparison with TMI 2 the scale of damage at Fukushima was hugely out of proportion to that caused by just the curtailing of the residual heat removal. At Fukushima, after the station blackout (SBO) and the running out of the batteries, the pressure in the reactors rose unacceptably and, which when relieved, led to the hydrogen explosions. The destruction of Unit 1 occurred on 11 March, while that of Units 2 and 3 occurred on 14 March shortly after the batteries ran out.

Rising pressure explained?

It has been generally assumed by Fukushima commentators. that the remaining, but decayed, residual heat was sufficient to boil off sufficient water in the reactor vessels to progressively uncover the fuel cans leading to the melting of the cores. It was also assumed that the remaining residual heat was sufficient to raise the reactors' pressure so greatly that it had to be relieved by manually opening automatically-controlled relief valves. But it may be that the production of steam was so great from restored fission by the forced withdrawal of the control rods by the rising pressure, that it overwhelmed the capacity of the normal relief valves and the relief of the pressure needed augmentation by the manual opening of the automatic relief valves.

Did the control rods stay raised in the shutdown position?

The earthquake at 14.46 local time on 11 March 2011 led to shutting down of the operating reactors, while the standby generators started automatically with the loss of off-site power, so that there was indication in the control rooms that Units 1, 2 and 3 had been successfully shut down. Has it never occurred to question whether the control rods stayed in the shut-down position? The BWR CRDMs have had a chequered history of mechanical and chemical contamination (crud), but mainly due to their positioning on the underside of a reactor with the demanding conditions for the operators doing maintenance work.

However, the lowering of the control rods by the increasing pressure or from the failure of standby generators and the battery DC supply needs consideration.

Loss of standby generators and battery power

Each CRDM has a Hydraulic Control Unit (HCU) with a scram nitrogen-pressured accumulator and a scram water accumulator with doubly sufficient stored energy to drive the control rods up between the fuel cans against the reactor internal pressure. However, after the loss of the 480 volts ac power from the standby generators to the condensate water pumps supplying the HCUs, the water content in the CRDM scram water accumulators may not have been restored. The loss of the standby generators would have also failed the power to the instrument air compressors for the pneumatic control valve operations. The nitrogen pressure in the nitrogen scram accumulators may have lost restoration if it depended on the operation of pneumatic control valves.

The CRDMs were controlled by 120 volts ac via transformers from the 480 volt ac (also from the standby generators) and by 125 volts dc from the batteries, which also supported the 120 volts ac by DC/AC inverters. While the batteries' supply remained, there was some cooling of the residual heat for Units 2 and 3 until 14 March, and for just 1 hour 20 minutes for Unit 1 on 11 March, after which its associated battery ran out.

Reactor Protection System (RPS)

Figure 7.3-5 appended shows the electrical, hydraulic and pneumatic services to the CRDMs.

There are two Notes:-

1. All relays and solenoids are shown in the normal non-scram condition.
2. Three-port Solenoid valves operation. (When the solenoid changes state, the dark port opens and the port with the dot closes).

The shut-down operation was performed by the insertion of the control rods by the CRDMs, when the RPS was in its second state (Note 2). The scram inlet and outlet valves opened, but when the standby generators and the batteries failed, the subsequent loss of power to the condensate water pumps and to the instrument air compressors would have lost the hydraulics to the hydraulic control units and the pneumatics and the electrics to the control valves, which would all have returned to the "non-scram" condition (Note 1), creating the potential for the withdrawal of the control rods.

Hydraulic Control Units (HCUs)

There is one Hydraulic Control Unit for each of the multiple control rod drive mechanisms, consisting of a scram N2 accumulator and a scram water accumulator. The scram water enters the control rod drive above the ball check valve and to the underside of the drive piston, discharging the water above the piston as it moves upward. The control rods are rapidly forced upwards against the reactor pressure to shut down the reactor. The scram water under the collet piston compresses the collet spring, which pushes the collet fingers into the notches in the index tube, locking the control rods in the shut-down position

However, when the relays and solenoids with the loss of DC power revert to the non-scram condition, the scram water is discharged, the collet springs expand, releasing the collet fingers and unlocking the control rods from the shutdown position.

It is the supposition of this review that, the loss of the 120 volts ac power from the standby generators and the subsequent loss of the 125 volts dc power supply from the batteries to the CRDMs' control panel, caused the withdrawal of the control rods. This raised the fission heat from the core, which raised the pressure in the reactor vessel, which in turn pushed down on the tops of the control rods, the lowering then increasing the fission heat and the pressure, continuing in a positive feedback manner, progressively withdrawing the control rods from the reactor vessels in a continuum.

Pressure balancing

When the scram water pressure lowers below it, the reactor vessel pressure is applied below the ball check valve to balance the pressure below the piston with that pressing on the spuds mounted on the tops of the control rod drives. However, the cross-section of the piston annulus is only 13% greater than the cross-section of the spud mounting. With rising pressure in the reactor vessel the balancing flow may be impeded by a build-up of crud in the multiple, narrow annuluses between the mechanism outer tube and the inside of the reactor branch.

The level of damage to the Fukushima reactor vessels, is commensurate with a progressive build up of pressure by an increasing fission heat starting from the top of the core with the progressive lowering of the control rods, while raising the surface temperature of the fuel cans. The rise in the temperature of the fuel can surfaces led to the melting of the fuel in the core. The ion exchange between the zirconium in the can alloy and the steam, created the hydrogen, which when released above its auto-ignition temperature of 585°C into the buildings within the explosive air/hydrogen content limits, exploded.

Conclusion

The size and scale of the damage to the Fukushima Daiichi boiling water reactors cannot be attributed just to the lack of removal of residual heat, which by the time of the meltdown events would have decayed considerably.

It can only be that the control rods, initially fully applied, had by some means lowered, exposing more of the core to uncontrolled fission and heat. This, perhaps subject to positive feedback, resulted in the raising of the reactor pressure, which when relieved resulted in an hydrogen explosion, while melting an increasing part of the cores.

As there are still BWRs and ABWRs in service and, if indeed the above scenario is correct, then the whole structure of the bottom-entry control rod drive mechanisms, their energy supplies and the control logic needs examination and possibly re-engineered.

John Busby 12 June 2021

The diagrams from General Electric BWR Systems Technology Manual Chapter 2.3 Control Rod Drive System are appended. Also Simplified Reactor Protection System Figure 7.3-5 from Chapter 7.3 of the Control Rod Drive System

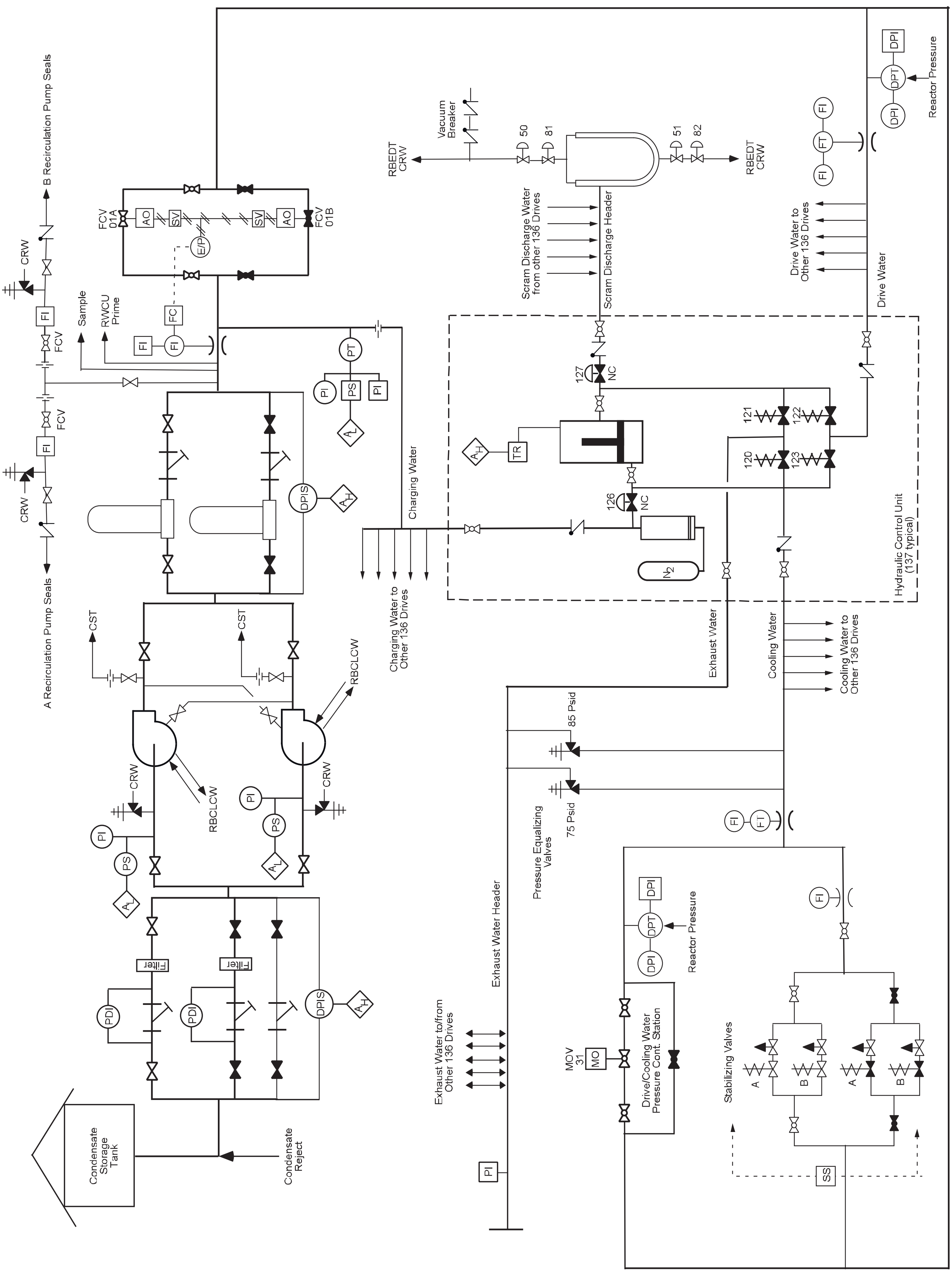


Figure 2.3-1 Control Rod Drive Hydraulic System

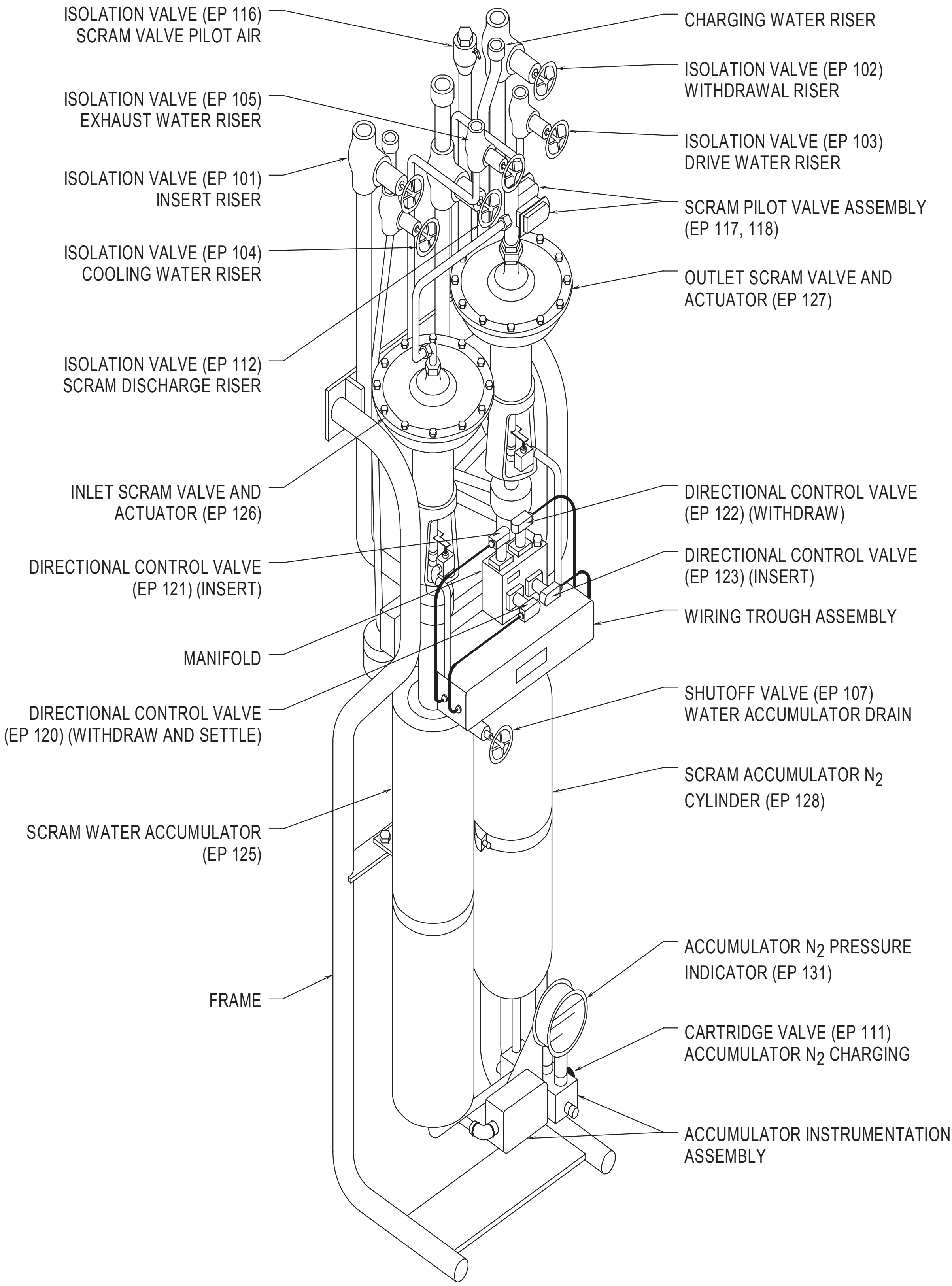


Figure 2.3-2 Hydraulic Control Unit

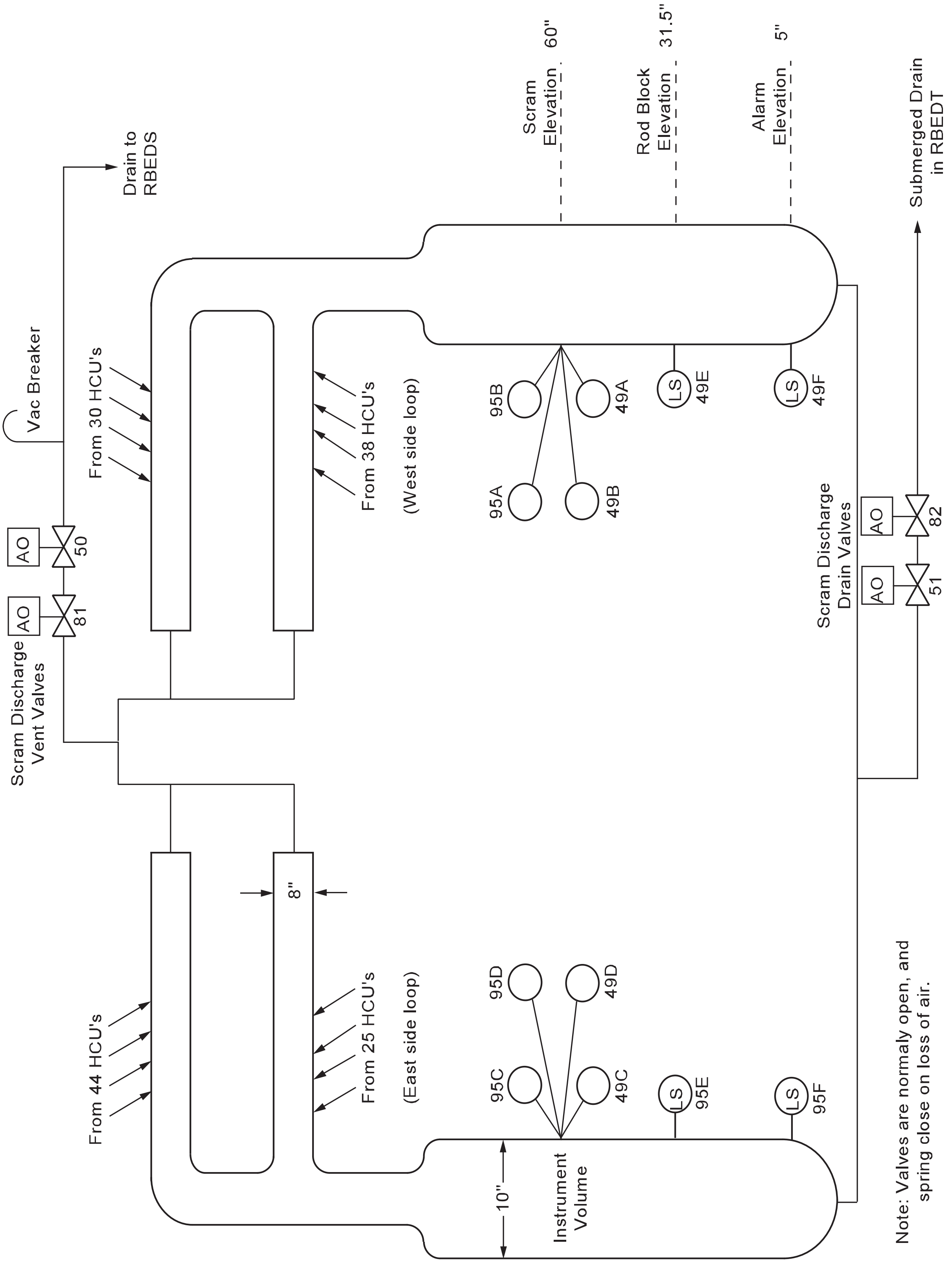


Figure 2.3-3 Scram Discharge Volume

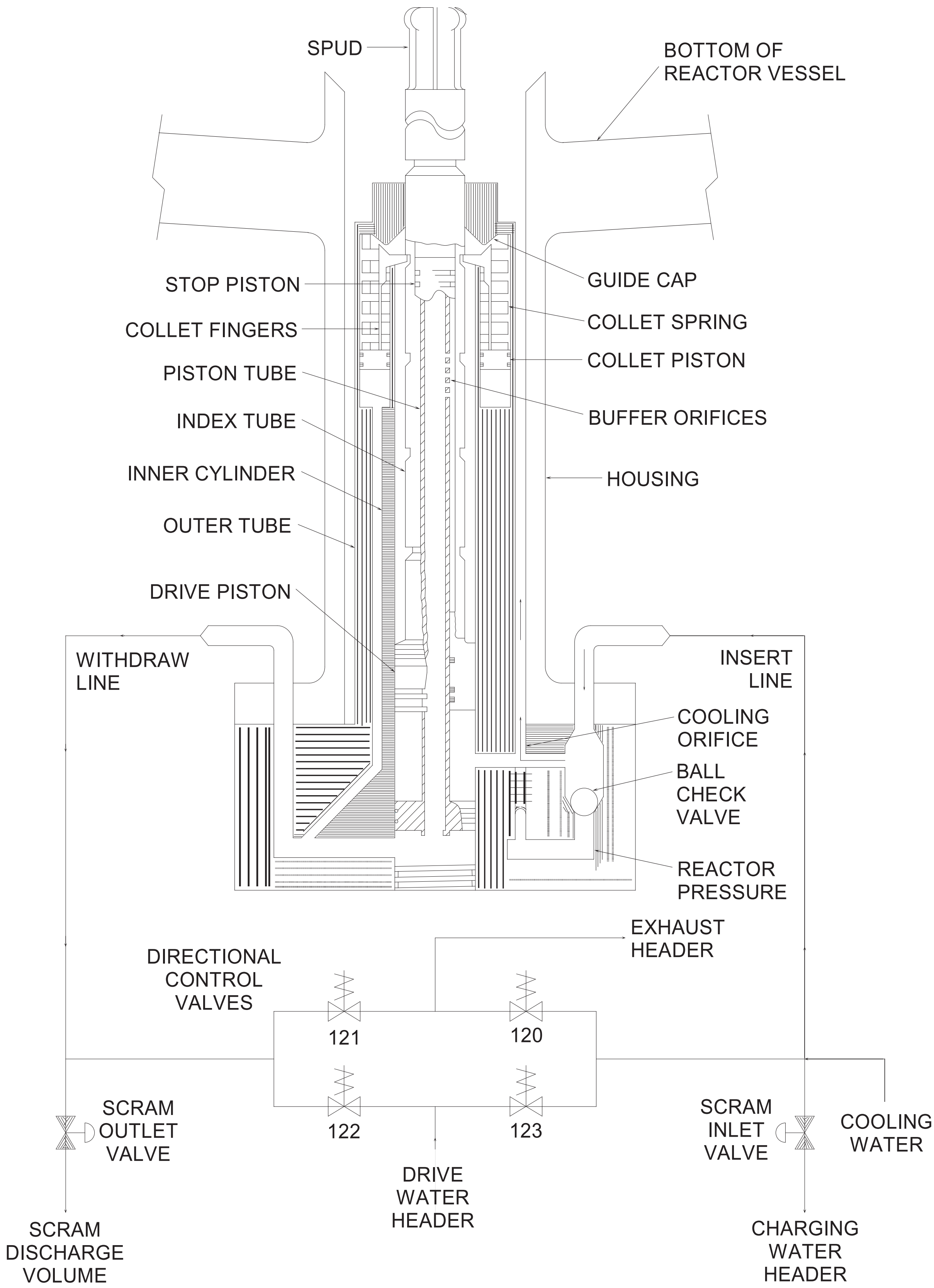


Figure 2.3-4 CRDM

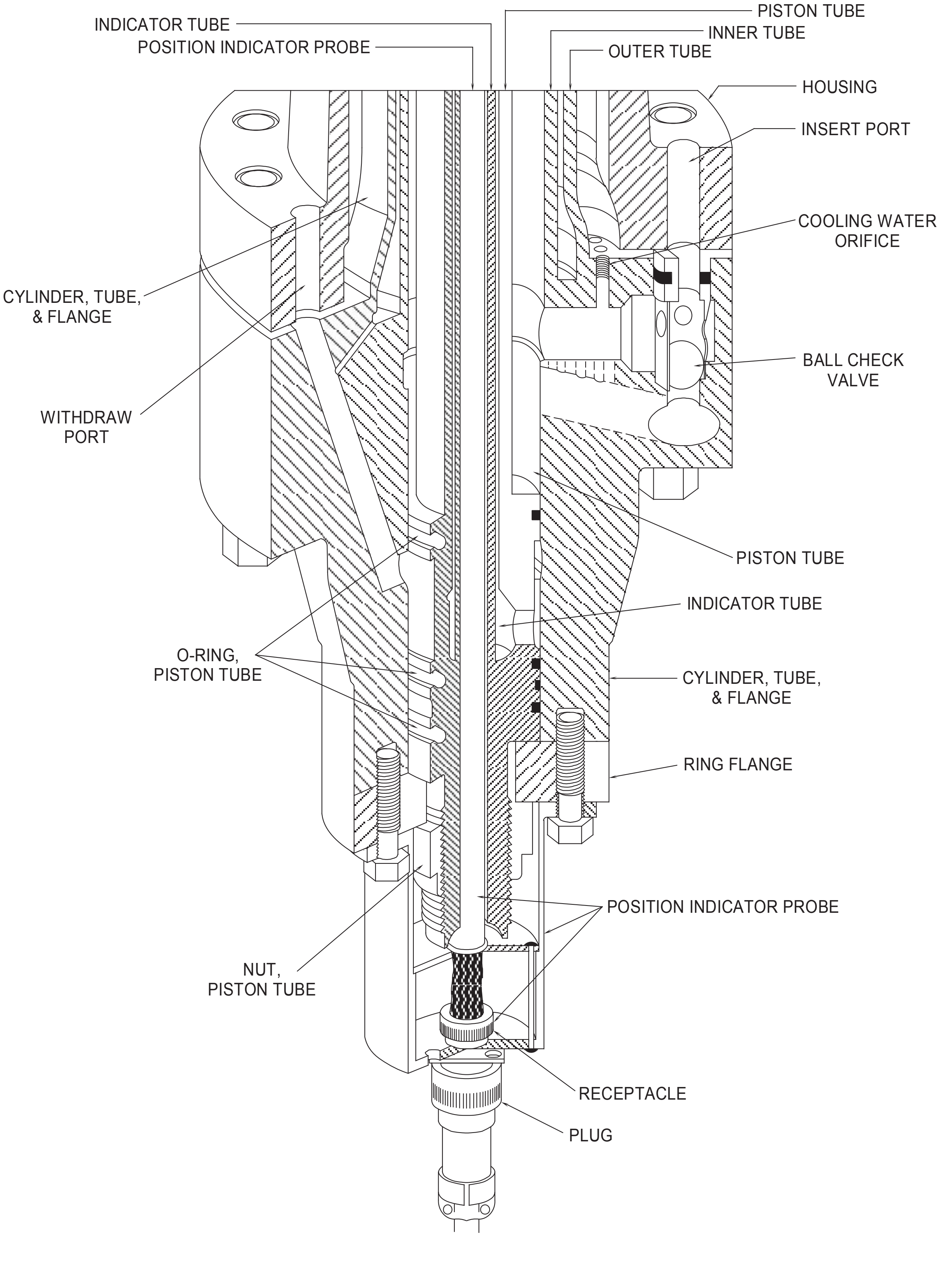


Figure 2.3-5 Enlarged Cutaway of Lower CRD

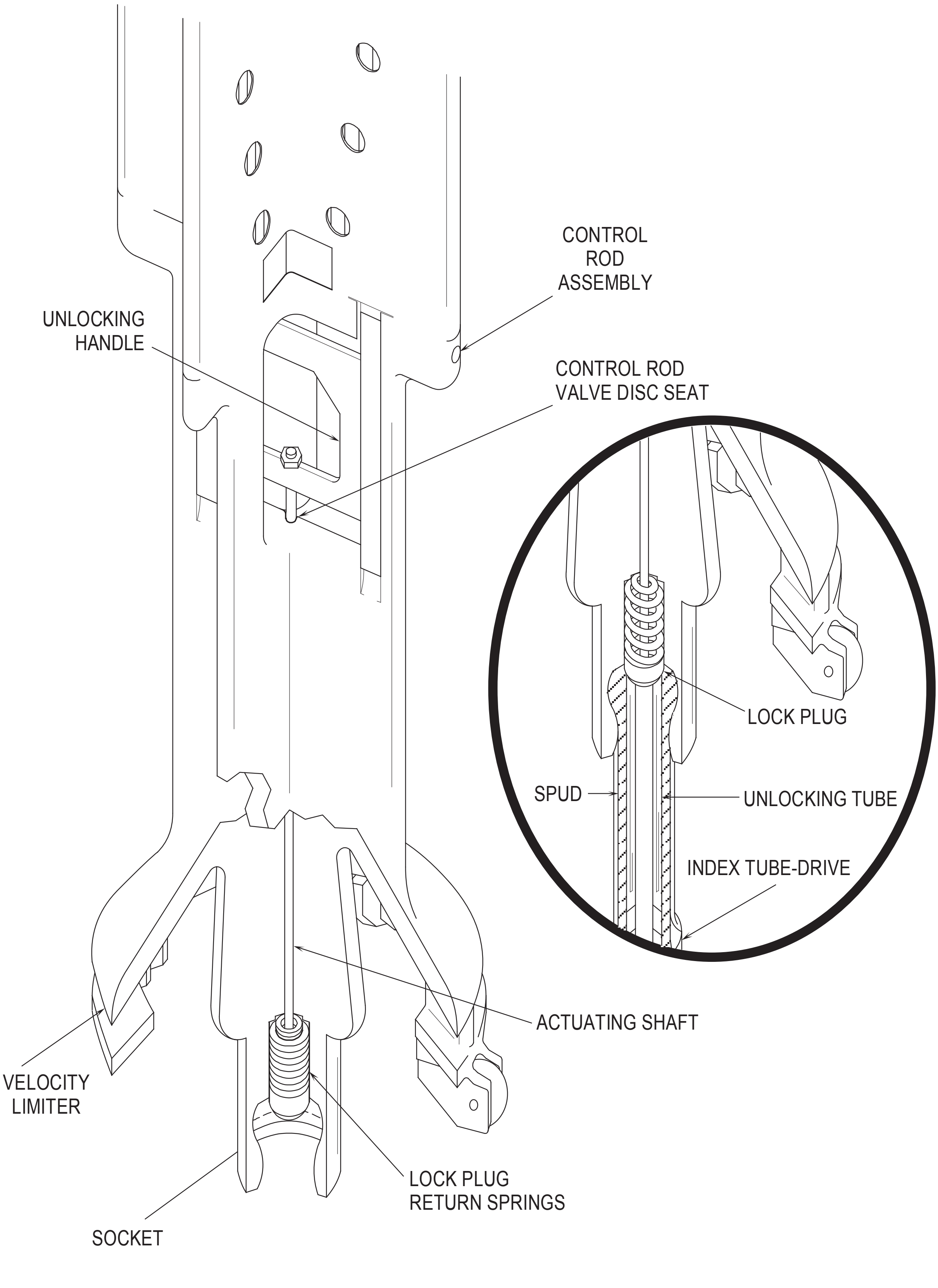


Figure 2.3-6 Control Rod Velocity Limiter and Coupling Mechanism

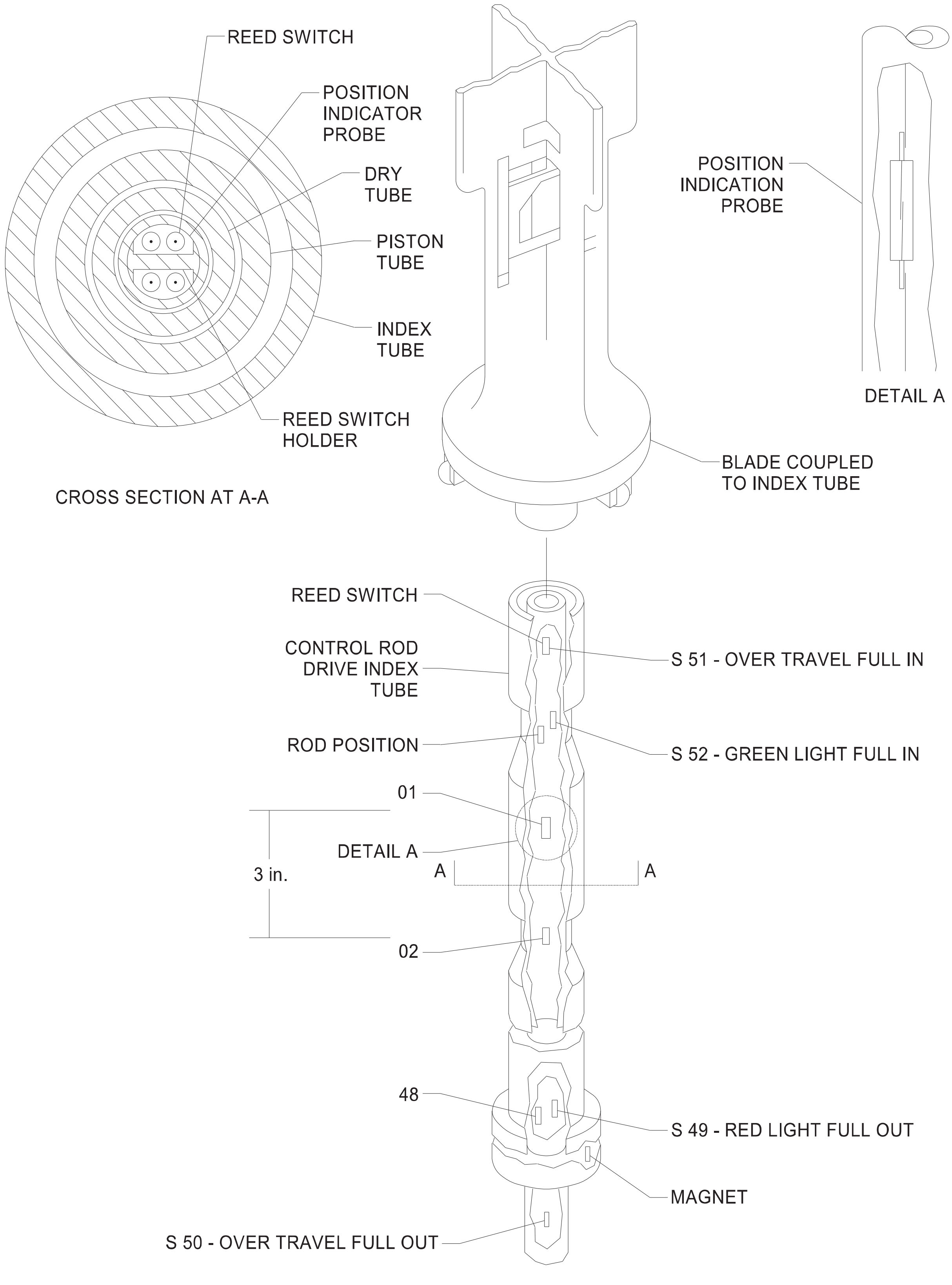


Figure 2.3-7 Control Rod Position Indication

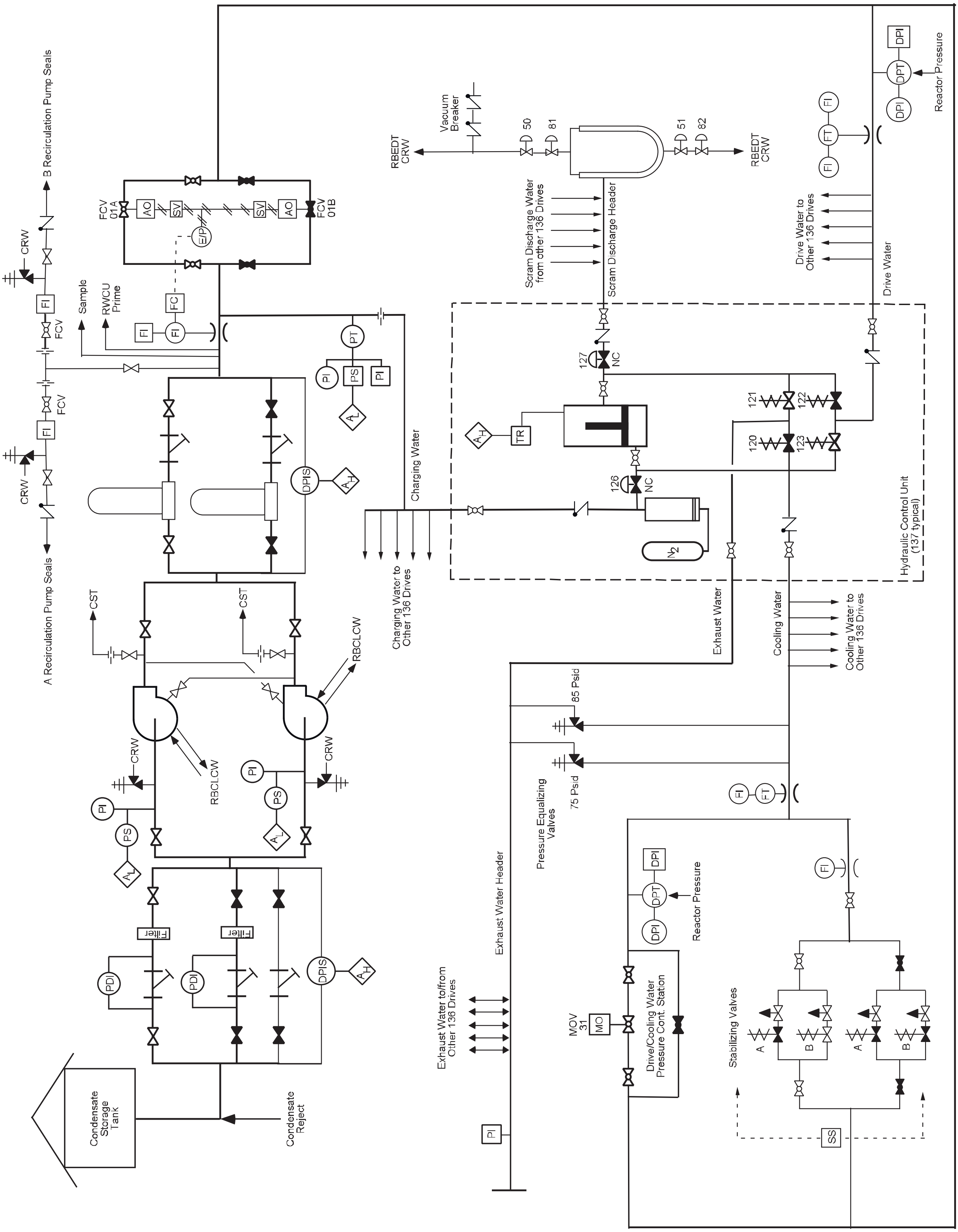


Figure 2.3-8 Control Rod Drive Hydraulic System Insert Operation

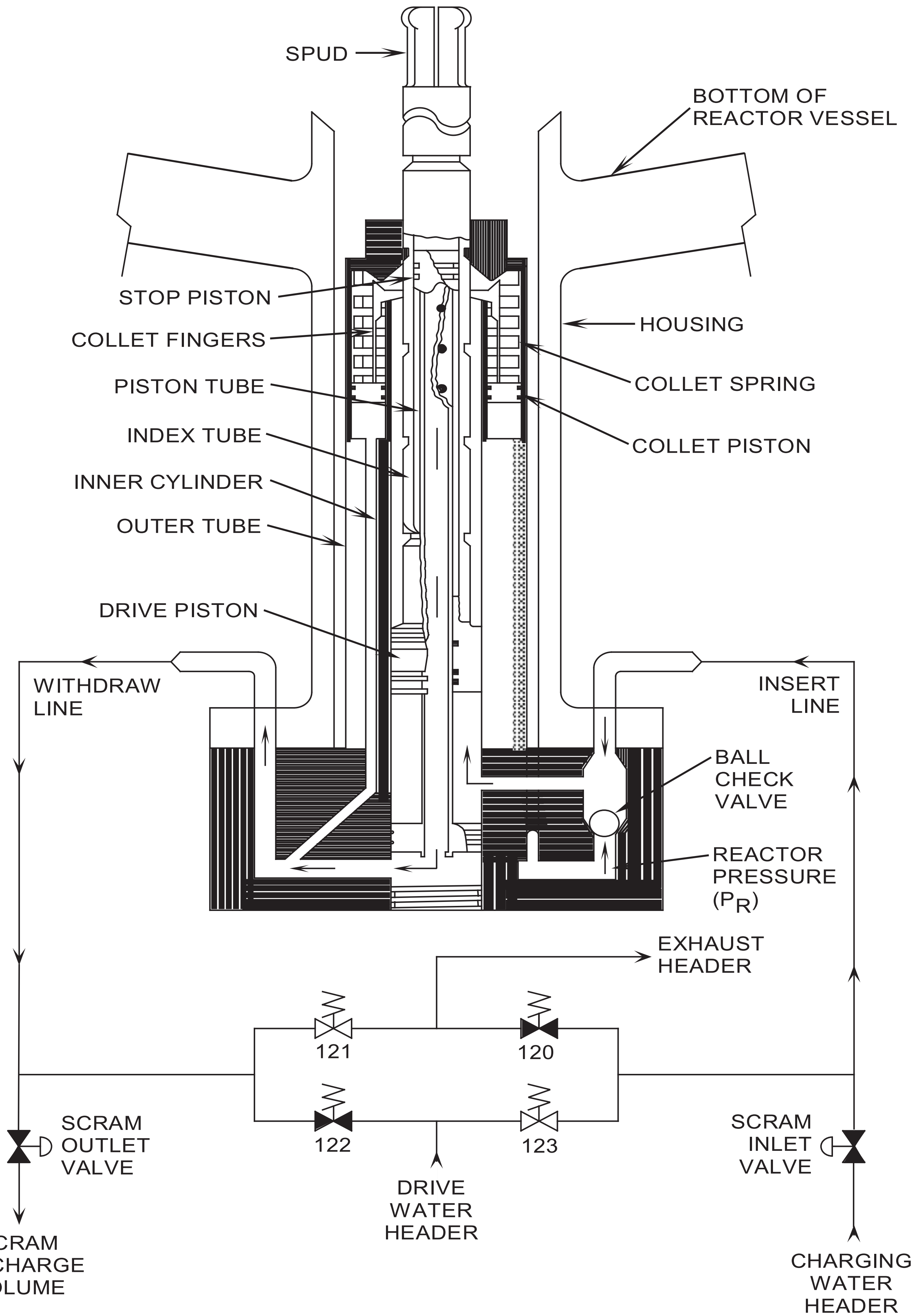


Figure 2.3-9 CRDM Operation (Insert)

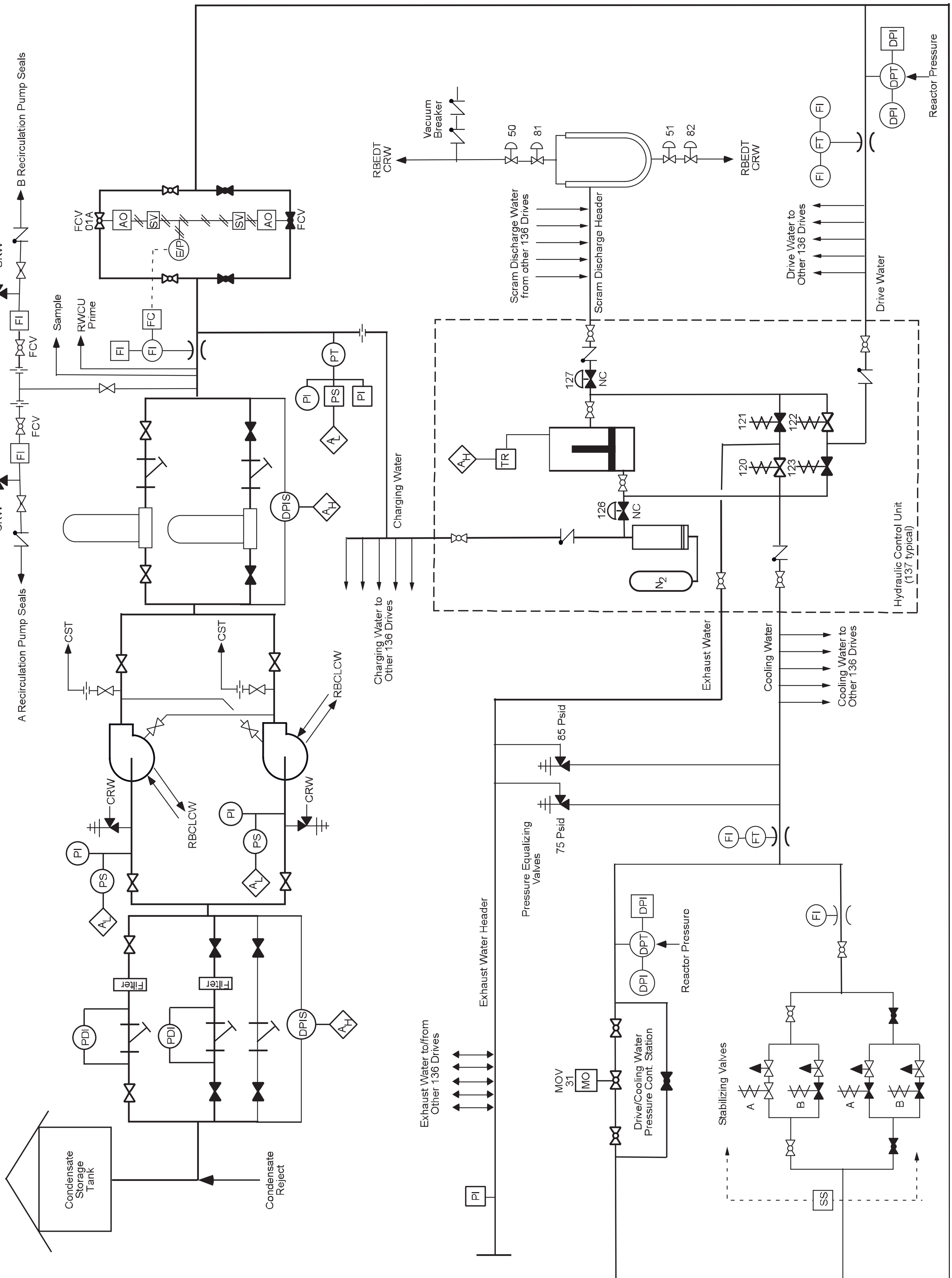


Figure 2.3-10 Control Rod Drive Hydraulic System Withdraw Operation

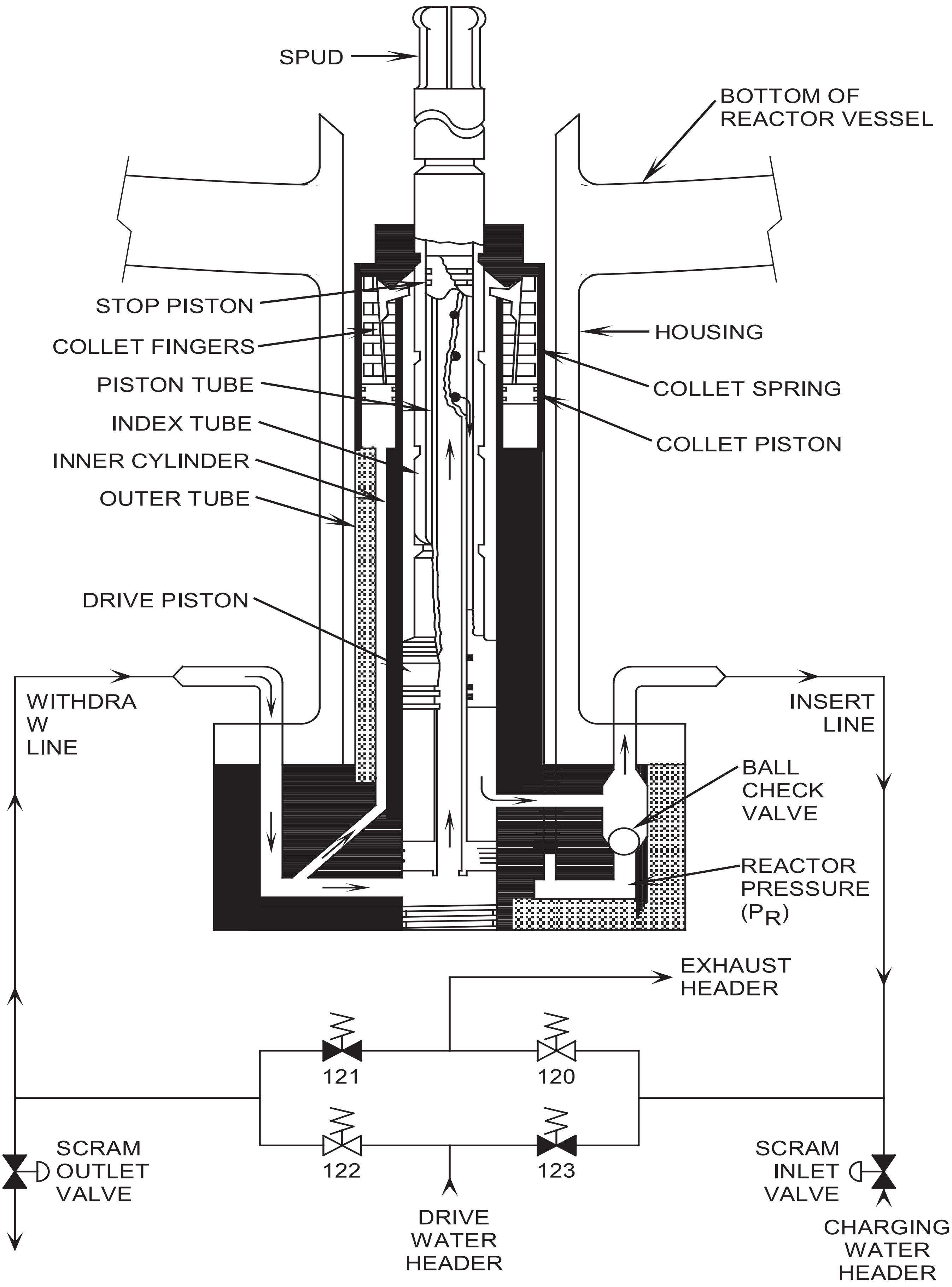


Figure 2.3-11 CRDM Operation (Withdraw)

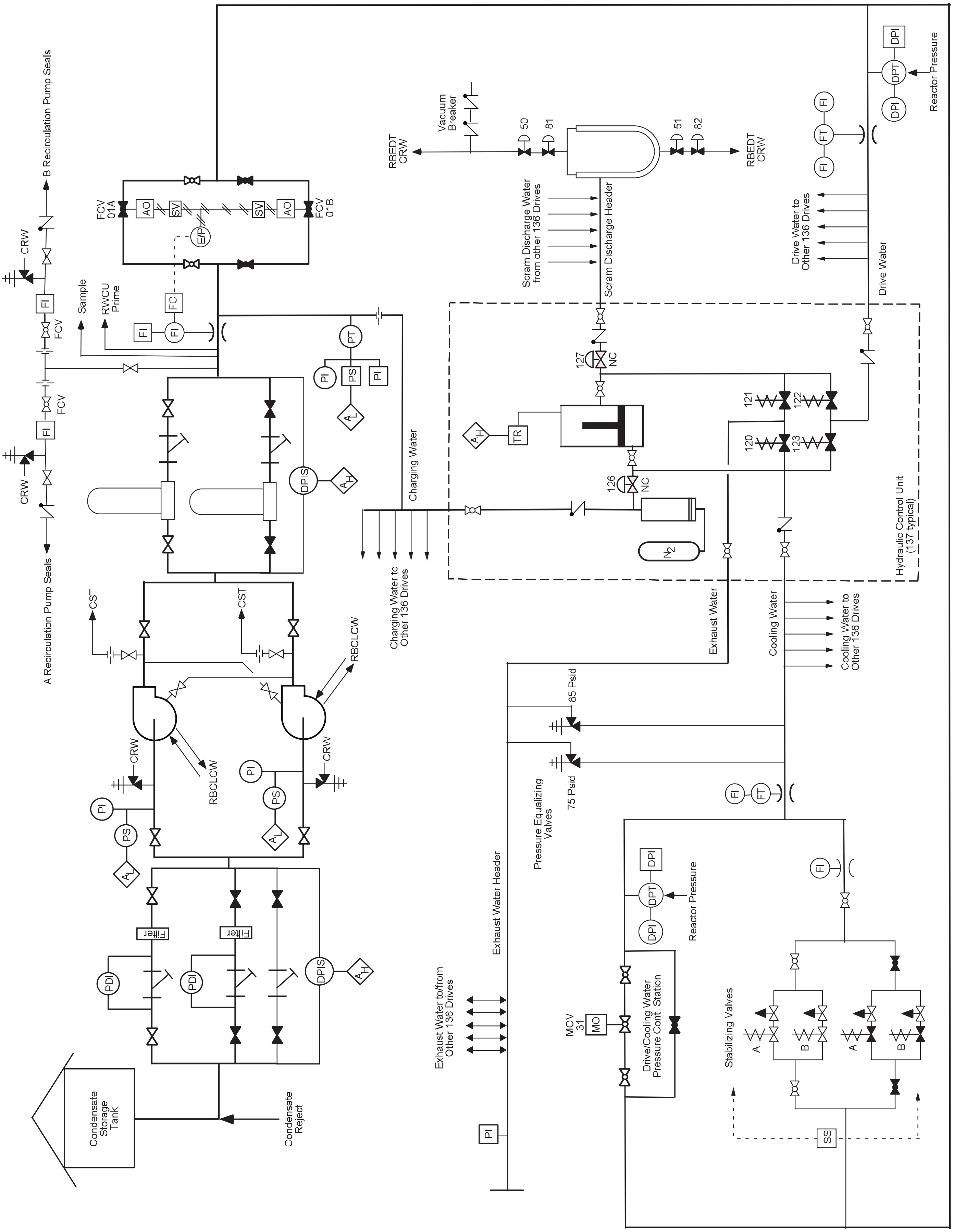


Figure 2.3-12 Control Rod Drive Hydraulic System Reactor Scram

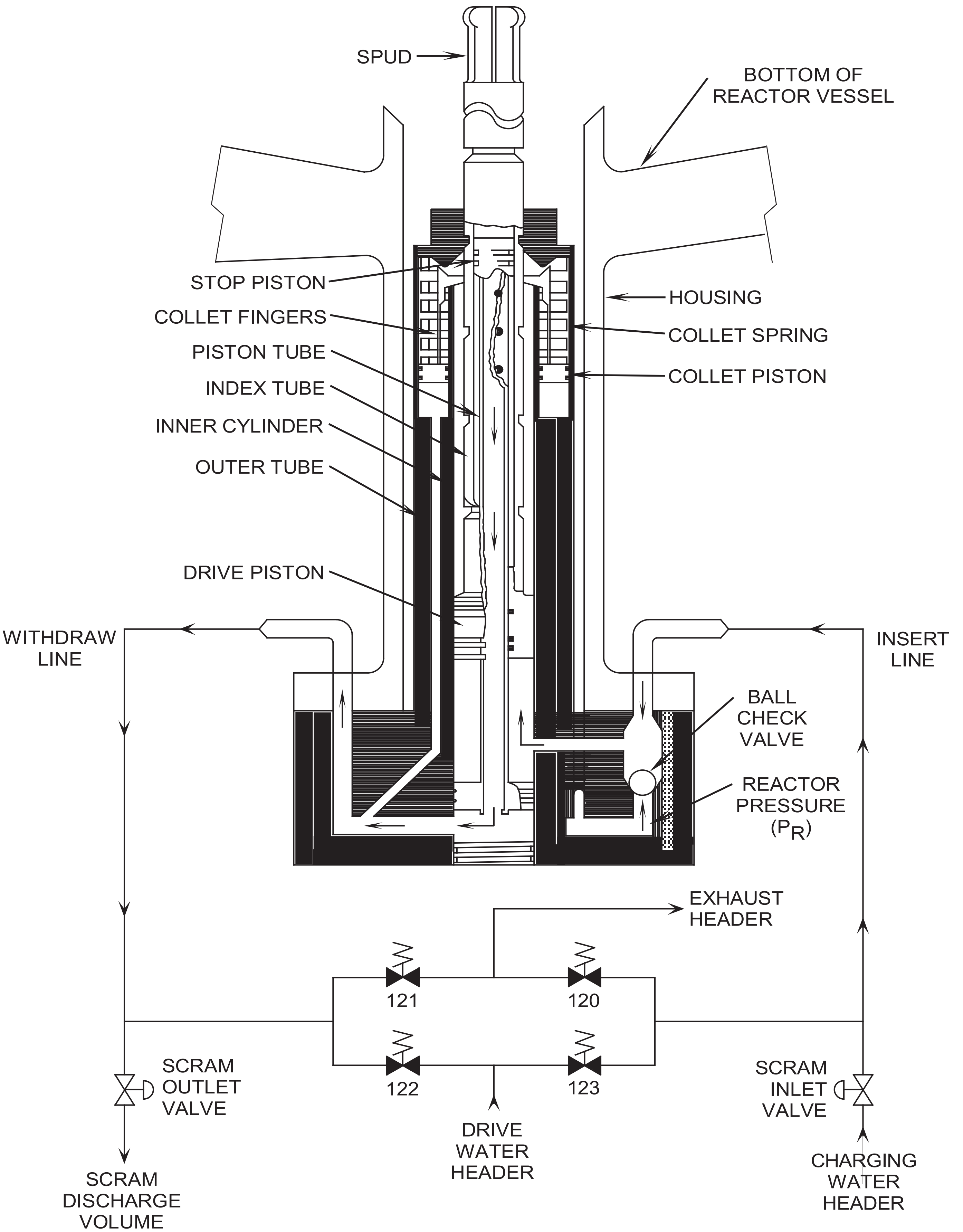


Figure 2.3-13 CRDM Operation (Scram)

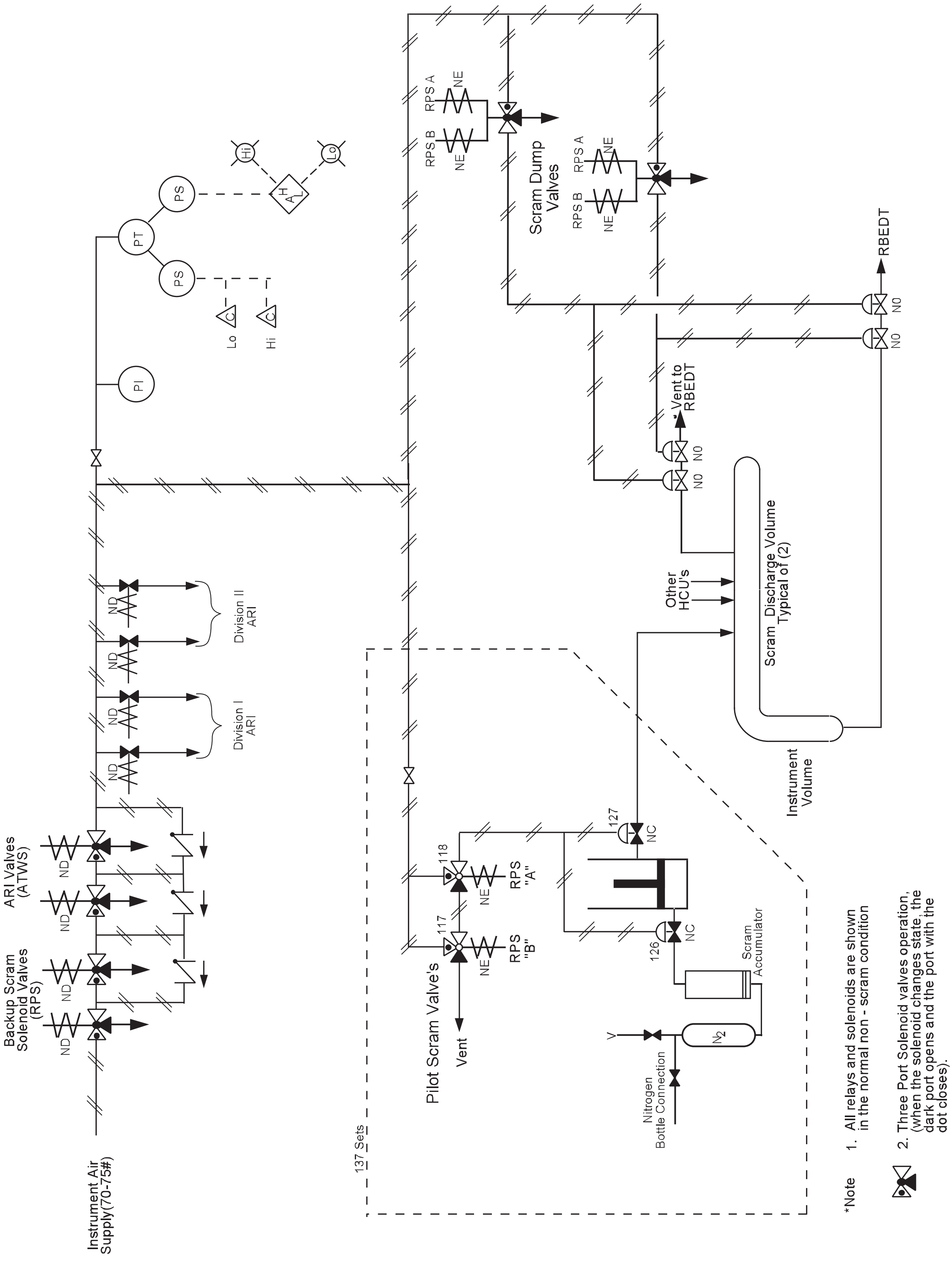


Figure 2.3-14 Scram Pilot Air Header

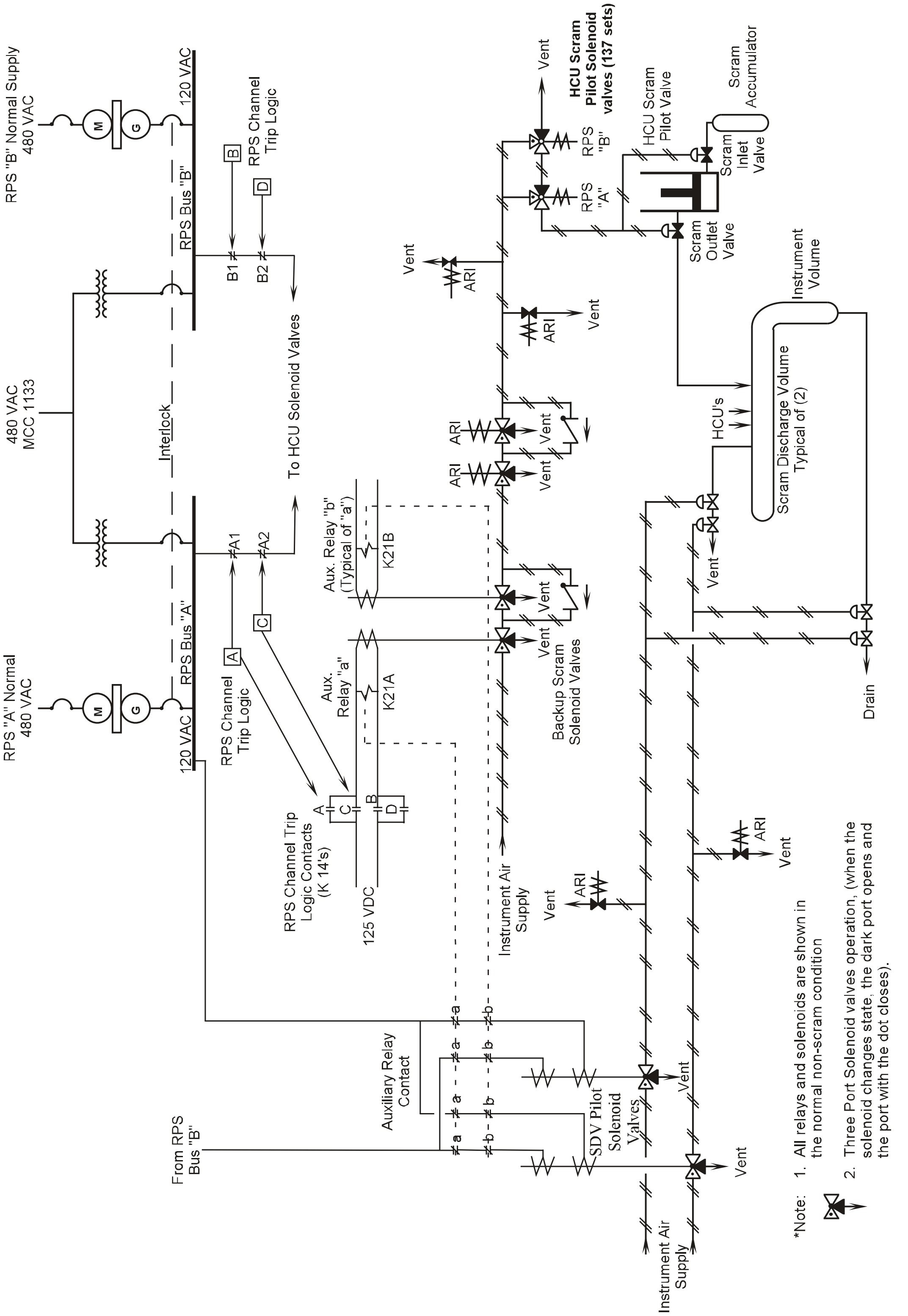


Figure 7.3-5 Simplified Reactor Protection System