BROWNS FERRY NUCLEAR PLANT

UNITS 1 AND 2

EMERGENCY CORE COOLING SYSTEMS
LOW PRESSURE COOLANT INJECTION
MODIFICATIONS FOR PERFORMANCE IMPROVEMENT

OCTOBER 1977

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(recirculation pump discharge, LPCI injection, RHR pump minimum flow bypass

1.0 INTRODUCTION

Browns Ferry Emergency Core Cooling System (ECCS) design and performance for Units 1 and 2 have been the subject of a recent review. This review led to a change in the system, which provided a significant reduction in the peak cladding temperature following a postulated recirculation line break. This reduction in peak cladding temperature has been accomplished by elimination of the Low Pressure Coolant Injection (LPCI) System recirculation loop selection and keeping the Residual Heat Removal (RHR) cross-tie valve closed. A report on this previous modification was submitted to the Nuclear Regulatory Commission in a letter from J. E. Gilleland to Benard C. Rusche dated Fabruary 12, 1976. Portions of that previous report are presented here to give a coherent description and safety analysis.

The proposed additional modification changes the power supply to the recirculation pump discharge valves, LPCI injection valves, the RHR pump minimum flow bypass valves, and RHR test isolation valves. The change also modifies independent valve AC power supplies to eliminate NRC concerns on paralleling of AC power supplies.

Major areas of discussion in this report include the proposed independent power supplies and a detailed safety analysis of the modification.

2.0 BACKGROUND

With the advent of the Interim Acceptance Criteria, it became advisable to consider the simultaneous occurrence of spraying and flooding to meet the stringent new temperature limit of 2300°F. The thermal-hydraulic models were refined to permit an accurate calculation of coolant remaining in the vessel following the blowdown, and of spray coolant reaching the lower plenum after the boiloff which takes place as it passes through the active fuel region. These refinements permitted an accurate calculation of the flooding rate due to spray operation, and even with the new requirement of an active component failure anywhere in the ECCS, no jet pump BWR failed to meet the Interim Acceptance Criteria. ECCS modifications which might have been suggested by the new evaluation models were therefore unnecessary.

The final ECCS acceptance criteria adopted by the AEC are more conservative than the interim acceptance criteria. These new criteria reduce operating flexibility and could result in power level restrictions. To offset the effect of the new criteria, a modification has been added to Units 1 and 2 which takes advantage of the credit given for the flooding effect achieved through the availability of additional LPCI pumps under certain single-failure conditions. TVA commited to modify the power supply to the recirculation pump discharge valves, LPCI injection valves, and RHR pump minimum flow bypass valves before return to power operation following the second refueling outage of the respective units.

3.0 DISCUSSION.

3.1 Accident Description

The Design Basis Accident (DBA), Loss-of-Coolant Accident (LOCA), is one of several hypothesized events used to evaluate the ability of the plant to operate without undue hazard to the health and safety of the public.

The overall initial assumptions remain as described in Section 14.6.3.1 of the FSAR:

The reactor is operating at the most severe condition at the time of the LOCA, which maximizes the parameter of interest: primary containment response, fission product release, or core standby cooling system requirements.

A complete loss of normal AC power occurs simultaneously with the LOCA. This additional condition results in the longest delay time for the core standby cooling systems to become operational.

The LOCA assumes that a recirculation loop pipeline is instantly severed. This results in the most rapid coolant loss and depressurization with coolant discharged from both ends of the break.

3.2 Modification

Modification of the system requires the following hardware and wiring changes on Units 1 and 2:

> The auto-transfer feature of valve motive power for recirculation pump discharge, LPCI injection, RHR pump minimum flow bypass valves, and RHR test isolation valves will be isolated from redundant divisional power supplies by motor-generator (M/G) sets

3.2.1 Suction Line Break Effects

Figure 2 illustrates operation of the modified system for a break in the recirculation pump suction line. The break location producing the highest peak cladding temperature is, as before, at the nozzle on the pressure vessel. The other side of the postulated "double-ended" break is fed through the recirculation loop by the jet pump nozzles, whose small area limits flow to a low value and makes frictional losses negligible in the calculation.

The discharge valves of the recirculation loops will begin closing upon receipt of a permissive signal. The valves are capable of closing against a differential pressure of 200 psid. To assure the recirculation system discharge valve is not required to close with a differential pressure greater than 200 psid, valve closures are delayed until reactor vessel pressure has decreased to less than 225 psig. By the time the

recirculation discharge valve has stroked sufficiently that it could present a flow-limiting restriction, the vessel pressure will have decayed below 200 psig.

Valve closure is therefore effected in about 62 seconds, of which 29 seconds represents the reactor vessel pressure permissive and 33 seconds the maximum valve closure time. The effect is isolation of the break from the LPCI system injection point.

Approximately 46 seconds after the break, the LPCI startup sequence is complete and flow commences in both loops. Flow into the broken loop will not reach its expected value for an additional 16 seconds, when the recirculation discharge valve has fully closed. The RHR pumps go nearly to full runout flow, as limited by the additional resistance in the pump discharge line, because each pair of pumps is delivering flow to its own bank of jet pump nozzles rather than to one bank as would be the case of loop selection logic. Additional resistance has been added to the RHR pump discharge lines. This replaces the resistance lost when only one or two pumps are discharging into a system designed for three pump flow. The added resistance prevents insufficient Net Positive Suction Head (NPSH) in these modes of operation.

In analyzing the single failures for a suction line break, both AC and DC power failures are considered (see Figures 4 and 5). For AC power considerations the most significant single failure for the modified system is a 480-V Reactor Motor-Operated Valve (MOV) Board failure. This failure results in two RHR pumps

operating in one loop and four CS pumps operating in two CS systems.

The most significant DC power single failure would be loss of a battery. For a suction line break this failure results in two RHR pumps operating in one loop, one RHR pump operating in the alternate loop, and two CS pumps operating in one CS system. Table 1 shows the various pump combinations for postulated single failures.

The unique power arrangement at Browns Ferry Units 1 and 2 requires examination of an opposite unit spurious accident signal. For this event one RHR pump in each loop of each reactor and one core spray system (two pumps) plus all required valves are available.

The limiting single failure is that failure which results in the longest reflood time and consequently the highest peak cladding temperature (PCT). Sensitivity studies have been performed which demonstrate that a typical limiting failure in the modified system is the failure of the RHR injection valve in the unbroken loop. This failure results in four core spray pumps, two in each CS loop, and two RHR pumps in one loop providing ECCS flow to the core. This combination gives a longer reflooding time than one core spray system (two pumps) and one RHR pump in each loop which is available following an opposite unit spurious accident signal. This is due in part to the effects of counter current flow limiting (CCFL) on the amount of the core spray flow available for reflooding. The assumed occurrence of CCFL results in there

being only a slight improvement with four CS pumps when compared to two CS pumps. Additionally, the two RHR pumps feeding into one loop deliver significantly less than twice the flow delivered by a single pump feeding each loop due to the system orificing effects. Thus the availability of one RHR pump in each loop for the alternate unit spurious accident signal provides better reflood characteristics than two RHR pumps into one loop even when supplemented by two additional CS pumps.

3.2.2 Discharge Line Break Effects

Figure 7 illustrates the operation of the modified system with a break in the recirculation pump discharge line.

When the RHR startup sequence is complete, the RHR flow in the broken loop is lost through the break. With the modification, the worst-case single failures are failure during opening of the RHR injection valve opposite the break, failure during opening of the RHR pump minimum flow bypass valve serving the RHR pumps intended for injection into the unbroken loop, and failure of a 480-V Reactor Motor-Operated Valve (MOV) Board. Table 1 and Figures 8-11 show the pump combinations which result from the postulated single failures.

The suction line break remains the design basis accident for the modified system, but with a lower calculated peak cladding temperature.

A typical limiting single failure for the discharge line break is the RHR injection valve failure. This failure results in four core spray pumps available for core reflooding. This condition results in a longer reflood time than the opposite unit spurious accident signal in which two core spray and one RHR pumps are available for reflooding. As previously discussed one RHR pump provides faster reflooding and, consequently lower PCT than two additional CS pumps.

The present Browns Ferry Units 1 and 2 system utilizes two power supplies for the electrical distribution system providing power to the RHR valves. Figure 12 shows the arrangement of the buses and the valves fed from these buses. Figure 13 shows the modified system which isolates the auto-transfer feature from the electrical distribution system by use of motor-generator sets.

3.3 Model Application

The core heatup calculations are performed using the approved Appendix K emergency core cooling evaluation models.

3.4 Safety Analysis

The proposed modification has been analyzed and evaluated to assure the changes do not introduce adverse effects to the overall plant. The areas evaluated are discussed in the balance of this section.

3.4.1 Equipment Capability to Perform as Analyzed

The major components of the proposed modification are unchanged, except for the power supplies to selected valves. Each major element is considered below:

3.4.1.1 Emergency Diesel-Generators

The proposed modification adds the running load of the M/G sets to the operating requirements of the diesel generators. This addition does not adversely affect diesel operation and diesel loading remains within design specifications.

3.4.1.2 RHR Pumps

The operating modes of the RHR pumps were changed by the previous modification such that two pumps discharge to each injection header thereby changing the discharge flow characteristics from that previously established. Prior to reactor startup after the previous modification, flow tests were conducted to establish the pump discharge path characteristics from which pump flow curves were developed. This information was used to determine the additional resistance to be added on the discharge side of each pump to ensure satisfaction of pump Net Positive Suction Head (NPSH) requirements.

3.4.1.3 Control Circuitry

All standards for engineered safeguards control equipment are maintained. Additional relays and wiring have been added to assure single-failure capability.

3.4.1.4 Recirculation Loop Equalizer Valve and RHR System Cross-Tie Valve

Inadvertent opening of these valves could negate the RHR system injection when needed, therefore one equalizer valve and the cross-tie valve were closed and motive power removed by the previous modification. An annunciator was added to provide redundant indication when the RHR cross-tie valve and/or equalizer valve are not fully closed.

3.4.1.5 Recirculation Pump Discharge Valves

Closure of the recirculation pump discharge valves is of importance to the proper application of the proposed modification. Three aspects of valve compatibility have been investigated:

3.4.1.5.1 Environment

As reported in Section 5.2 of the Browns Ferry FSAR, the recirculation system valves are designed to operate under the environmental conditions associated with the DBA-LOCA.

3.4.1.5.2 Break Effects

A study of the drywell geometry was performed prior to the previous modification to determine the effects of jet impingement resulting from a postulated recirculation line break. For the break effects study, breaks were assumed at all terminals, branch lines, and at other locations based upon stress. Breaks were assumed at all locations where pressure plus dead load plus thermal plus earthquake stresses exceed $0.8(1.2S_h+S_A)$. Additionally, in piping runs where no stresses occur in excess of $0.8(1.2S_h+S_A)$, a minimum of two intermediate breaks were postulated based upon the highest total stresses combined as above.

For the suction line break, re-routing of cable has been provided, to ensure recirculation discharge valve operation. Valve closure at the time of a discharge line break is not considered in the ECCS analysis. Also, closure of the discharge valve does not change the RHR system input capability during a discharge line break (See Figure 7).

3.4.1.5.3 Valve Differential Pressure

Recirculation pump discharge valve closure requires both a LOCA initiation signal and a decrease in reactor pressure to the permissive setting. With valve closure initiation delayed until reactor pressure has decayed to less than 225 psig (approximately 29 seconds) the differential pressure across the closed valve will always be less than the maximum 200 psid. The sensor and

permissive circuitry are designed to satisfy all requirements for engineered safeguards control systems.

3.4.1.6 RHR Pump Minimum Flow Bypass Valve

RHR pump minimum flow bypass valves will be provided with redundant power supplies and control logic to assure maximum pump protection under postulated accident conditions.

3.4.1.7 Batteries

The proposed modification does not change any of the operating requirements on the station batteries.

3.4.1.8 Motor-Generator (M/G) Sets

Qualified motor-generator (M/G) sets will be used as isolation devices for the power feeds to the 480-V Reactor MOV boards with auto-transfer feature (see Figure 13). Although only one M/G set will normally supply power to the Reactor MOV board, both M/G sets will run at all times to assure readiness of the alternate M/G to accept load, if required. M/G sets will be sized to accept valve loads at any time during the initiating event (62 secs.). Fire protection to the M/G sets will be evaluated and appropriate measures taken to assure adequate fire protection.

3.4.1.8.1 Seismic Qualification

The operability of the motor-generator sets and all the appurtenances vital to their operation during and after a SSE is verified in accordance with IEEE 344 as applicable to the plant.

3.4.1.9 RHR Test Isolation Valves

The RHR test isolation valves will be provided with redundant power supplies and closing control logic to protect against those postulated occurrences which would leave the valve open and route an unacceptable amount of flooding water away from the core. Redundant controls are wired as specified in 3.4.2.3.

3.4.2 Equipment Interfaces

The effects of the proposed change on the various operating modes of the equipment have been evaluated and found to be acceptable, as described below:

3.4.2.1 Emergency Diesel-Generators

The proposed modification introduces no new or different interfaces for this equipment.

3.4.2.2 Valve Power

Motor operated valve (MOV) boards for those valves necessary for automatic operation for LPCI injection (recirculation pump discharge, LPCI injection, RHR pump minimum flow bypass, and RHR test isolation) will interface with divisionalized 480 VAC shutdown boards with M/G sets as isolation devices. Two 480-V AC Reactor MOV boards will be divisionalized and supply power to those valves necessary for the RHR function (see Figure 13). These valve boards will have an auto-transfer feature for redundant power but will be protected from the redundant divisional source by using M/G sets as isolation devices. The auto transfer feature will be eliminated from all valve boards not protected by M/G sets.

3.4.2.3 Valve Motor Control

To ensure that a malfunction in the individual valve controller does not couple back to the other valve control circuits, the redundant A and B circuits were provided separate relays and contacts in the logic panels on a previous modification. This separated, redundant arrangement has been applied to the RHR and recirculation system valves needed for operation as described. System interfacing and protection as related to the valve motor control centers are unchanged.

3.4.2.4 DC Control Power

As shown in Figure 14 and Browns Ferry FSAR Figure 8.6-3, 250 VDC from the station batteries provides control power to RHR logic panels. After the proposed modification the same equipment receives power from this source as in the original design. Failure of any one station battery does not cause interactions that exceed the limiting case for core cooling capabilities.

3.4.2.5 RHR Logic Panels

To provide the necessary redundancy required on the previous modification, changes were made to the RHR logic panels. To preclude valve-to-valve interface, redundant and separate relays and contacts were provided for each RHR and recirculation system. Each of the added redundant relays was provided full separation from all others by enclosure in a metal box. The wiring from redundant contacts between the two logic panels was provided separation by enclosure in flex conduit and termination in metal junction boxes. This logic scheme will be maintained in the new modification.

3.4.2.6 Motor-Generator (M/G) Sets

Two M/G sets will interface between each board with the auto-transfer feature and shutdown boards. These M/G sets provide isolation of the boards from redundant sources.

3.4.3 Functional Interface

The RHR system, as discussed in this report, performs as a short-term post-LOCA core cooling function. The system also provides a long-term containment cooling function which is described in Sections 4.8.6.2 and 14.6.3.3.2 of the FSAR. The effects of the proposed change to the core cooling function on the containment cooling function were evaluated and found to be acceptable after modification as described below.

In analyzing single failures which might influence long-term suppression pool cooling, both AC and DC control and emergency power failures as well as component failures in the RHR and RHRSW (cooling water) systems were considered. The worst case single failure (Reactor MOV Board loss) with the modified system still leaves two RHR heat exchangers, two RHR pumps, and two RHR Service Water pumps and associated valving available for suppression pool cooling. The suppression pool temperature versus time response for this combination of equipment is shown by curve C in FSAR Figure 14.6-12.

3.4.4 Satisfaction of Appropriate Standards

The proposed modification directly affects as Engineered Safeguards System and has been designed to Class I system standards. The standards and guides which were applicable to the original design have been reviewed to assure the modified system design, equipment, and installation meet or exceed the qualifications of the unmodified system.

3.4.5 Quality Assurance and Control

Quality assurance and control will be applied to this modification as detailed in Appendix D of the Browns Ferry FSAR. Appendix D incorporates the requirement of 10CFR50, Appendix B.

4.0 SUMMARY AND CONCLUSIONS

The proposed modification involves some physical changes to the plant to permit divisional isolation of the auto-transfer concept and adoption of the total system availability of the new design.

The analytical methods used reflect the most recent determinations of NRC staff and reactor suppliers for modeling the performance of Emergency Core Cooling Systems.

The application of the proposed modification adds to the overall capability of the plant to continue operation in a manner that ensures the health and safety of the public while providing benefit in the production of electrical energy.

5.0 REFERENCES

1. Interim Policy Statement, USAEC, dated June 19, 1971; Subject: AEC Adopted Interim Acceptance Criteria for Performance of ECCS for Light-Water Power Reactors.

2. Letter from J. E. Gilleland (TVA) to Benard C. Rusche (NRC) dated February 12, 1976.

TABLE 1

ECCS PUMP CONFIGURATION

Suction Side Break	Pumps Available**
No Failures	4 Core Spray, 2 RHR in each Loop
Opposite Unit Spurious Accident Signal	2 Core Spray, 1 RHR in each Loop
RHR Injection Valve Failure*	4 Core Spray, 2 RHR in one Loop
RHR Minimum Valve Failure*	4 Core Spray, 2 RHR in one Loop
Recirculation Discharge Valve Failure-Break Side*	4 Core Spray, 2 RHR in one Loop
480 V Reactor MOV Board*	4 Core Spray, 2 RHR in one Loop
Diesel Failure	2 Core Spray, 2 RHR in one Loop, 1 RHR in other Loop
Battery Failure	2 Core Spray, 2 RHR in one Loop, 1 RHR in other Loop

Discharge Side Break	Pumps Available**
No Failures	4 Core Spray, 2 RHR in one Loop
RHR Injection Valve Failure*	4 Core Spray
RHR Minimum Flow Valve Failure*	4 Core Spray
Diesel Failure	2 Core Spray, 1 RHR
Battery Failure	2 Core Spray, 1 RHR
Opposite Unit Spurious Accident Signal	2 Core Spray, 1 RHR
480 V Reactor MOV Board*	4 Core Spray

*Limiting Single Failure
**In Unbroken Loop

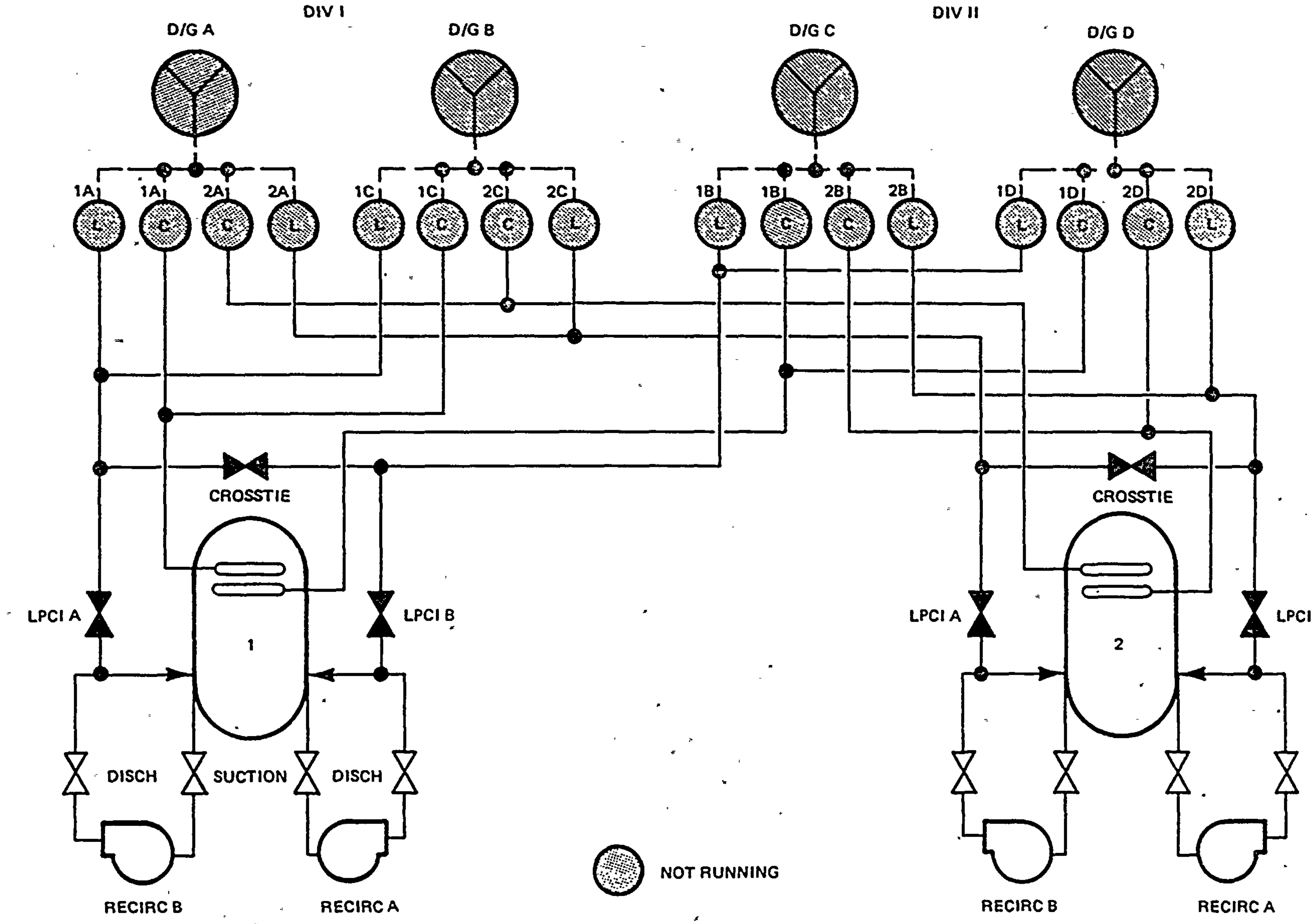


Figure 1 *System Normal Operation*

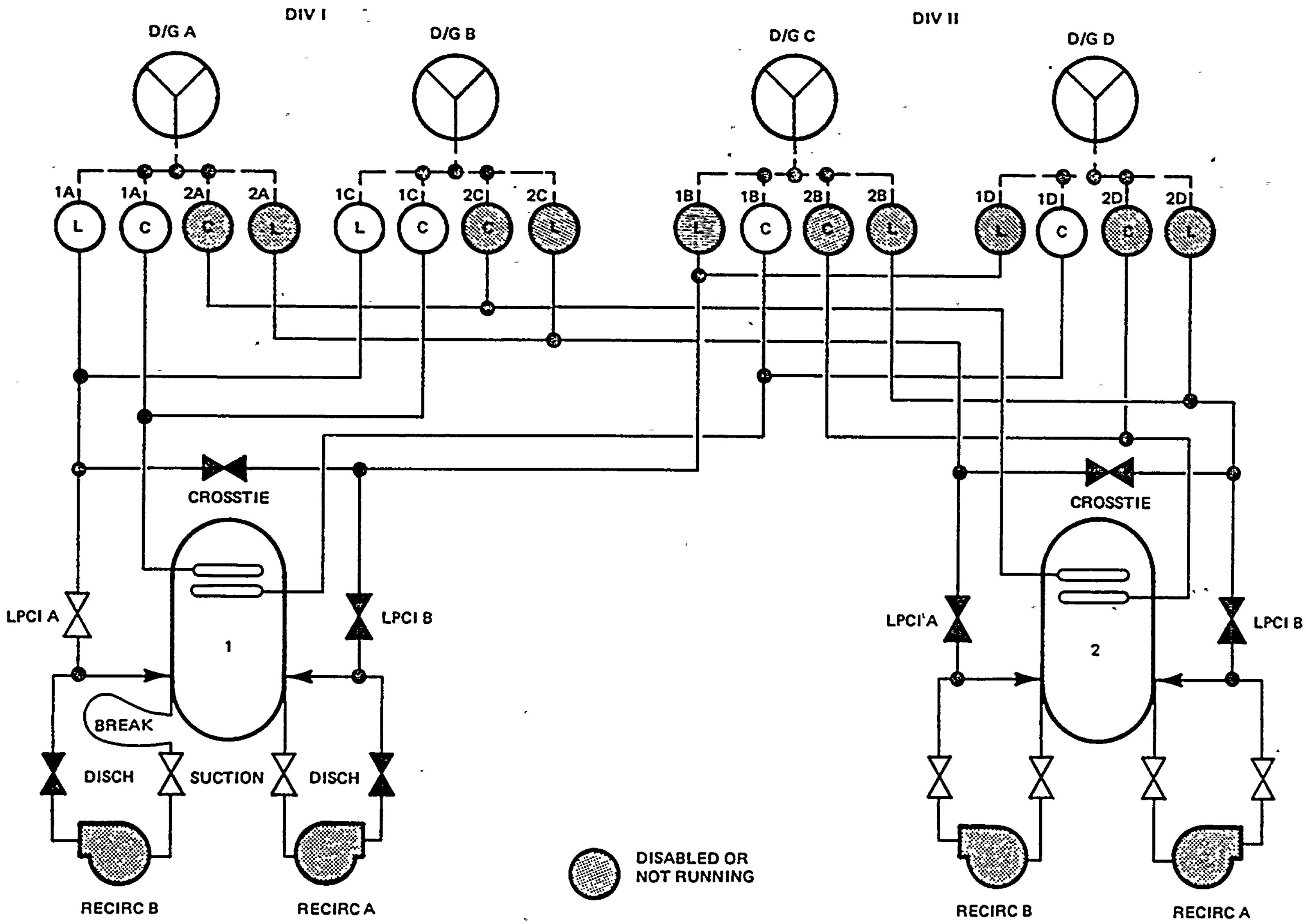


Figure 3 System Mode of Operation During Unit 1 LOCA (Suction Line Break) – LPCI Injection Valve Failure

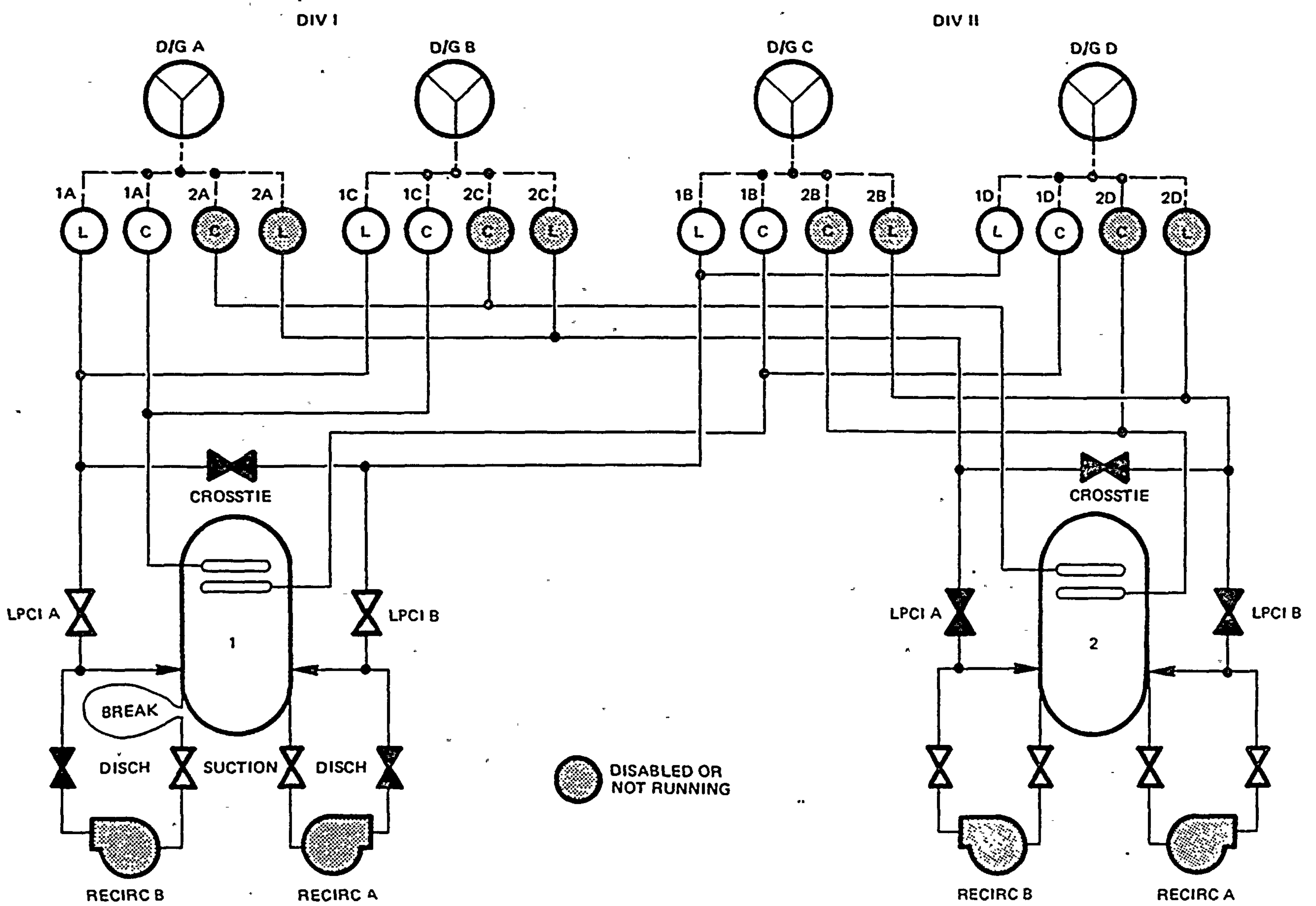


Figure 2 System Mode of Operation During Unit 1 LOCA (Suction Line Break) – No Failures

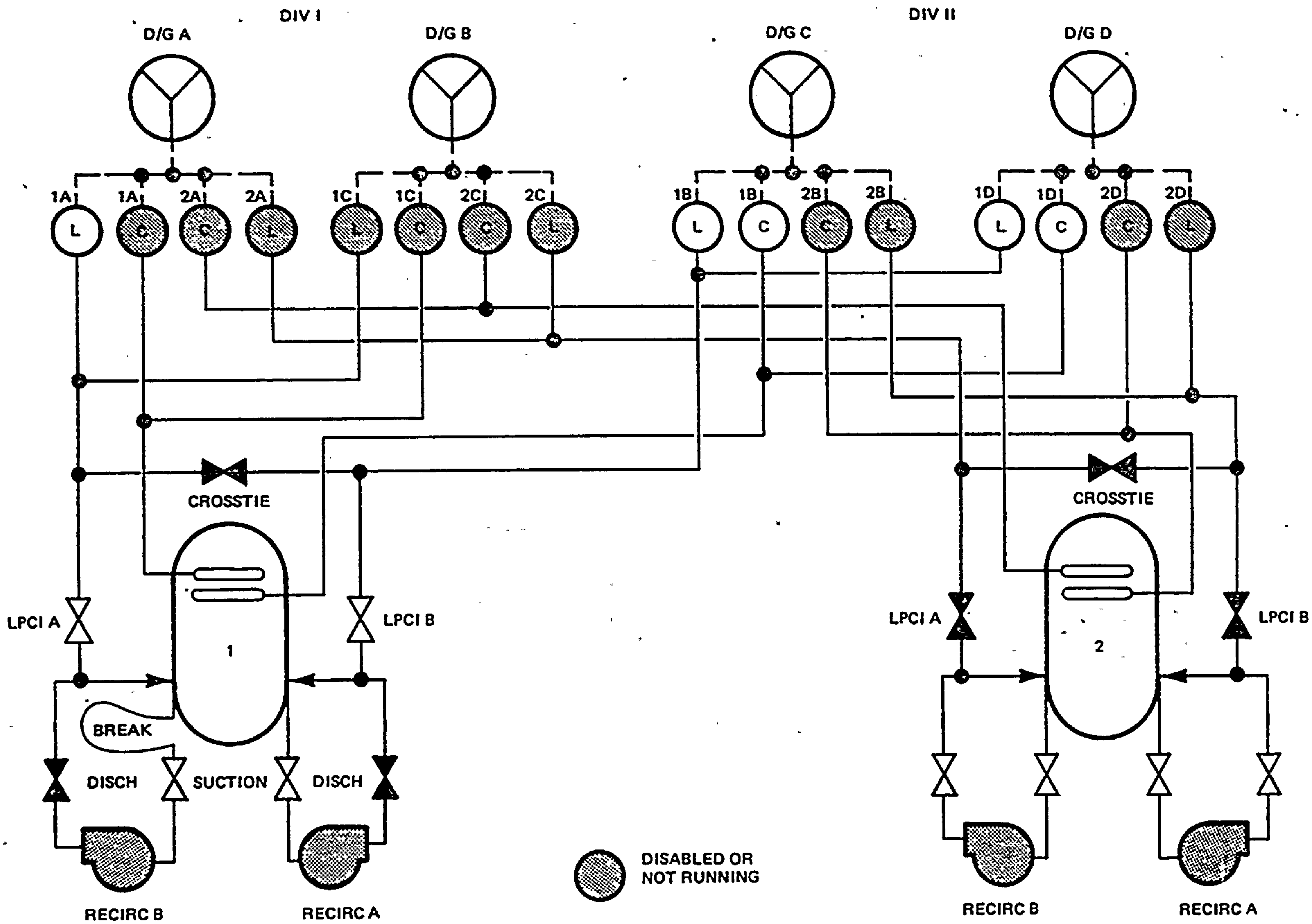


Figure 5 *System Mode of Operation During Unit 1 LOCA (Suction Line Break) — Battery Failure*

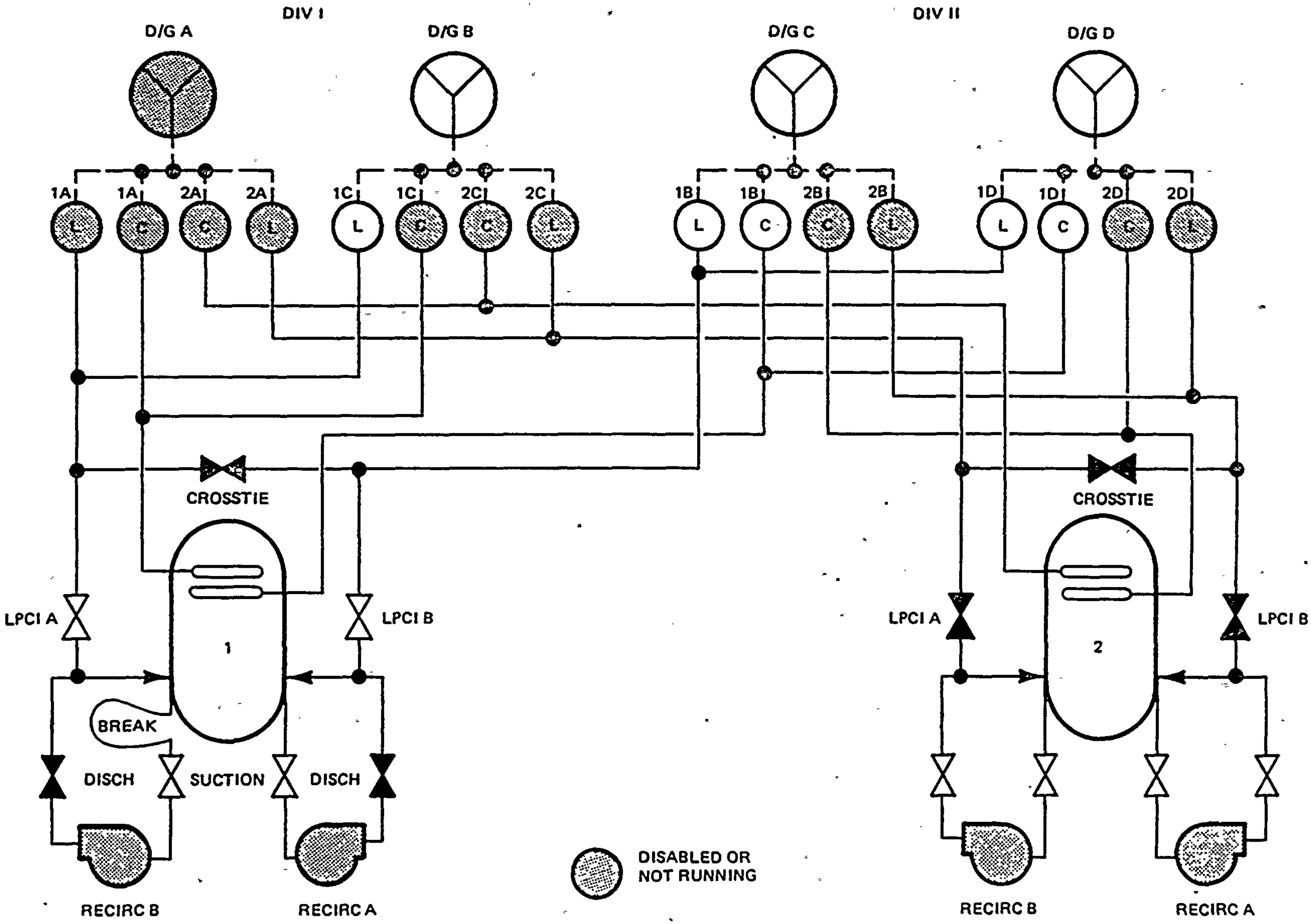


Figure 4 *System Mode of Operation During Unit 1 LOCA (Suction Line Break) — Diesel Failure*

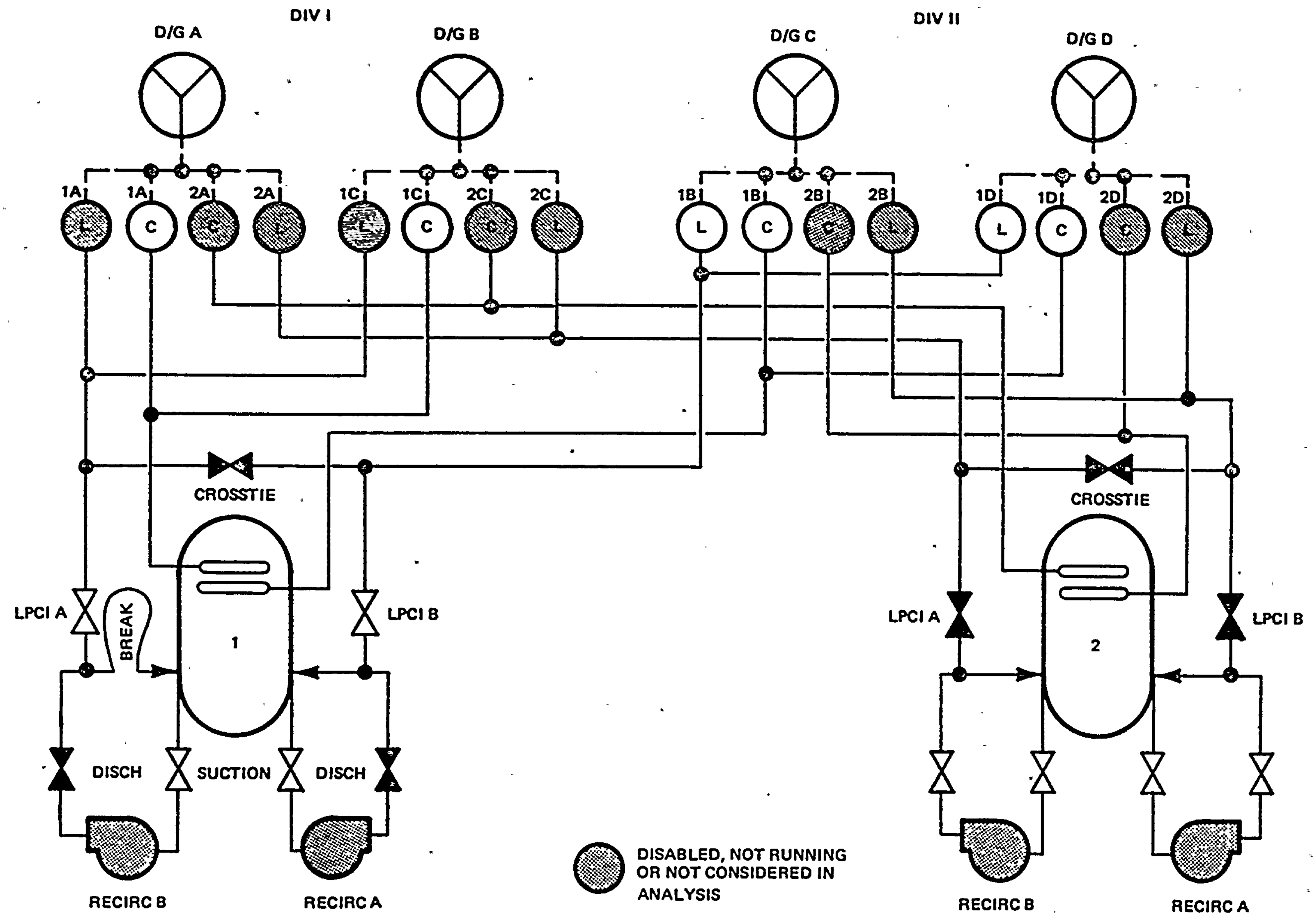


Figure 7 System Mode of Operation During Unit 1 LOCA (Discharge Line Break) – No Failures

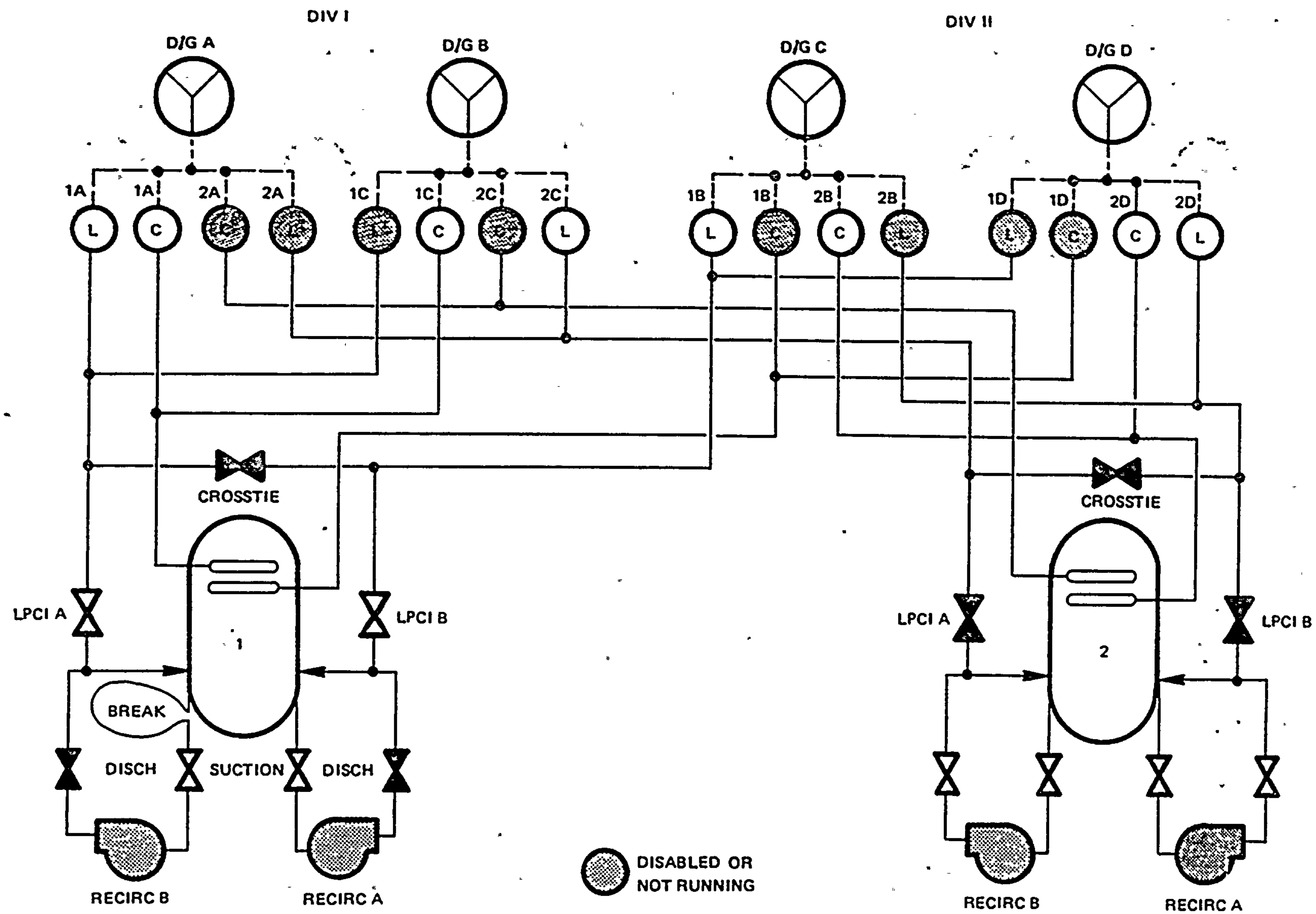


Figure 6 System Mode of Operation During Unit 1 LOCA (Suction Line Break) — Opposite Unit Spurious Accident Signal

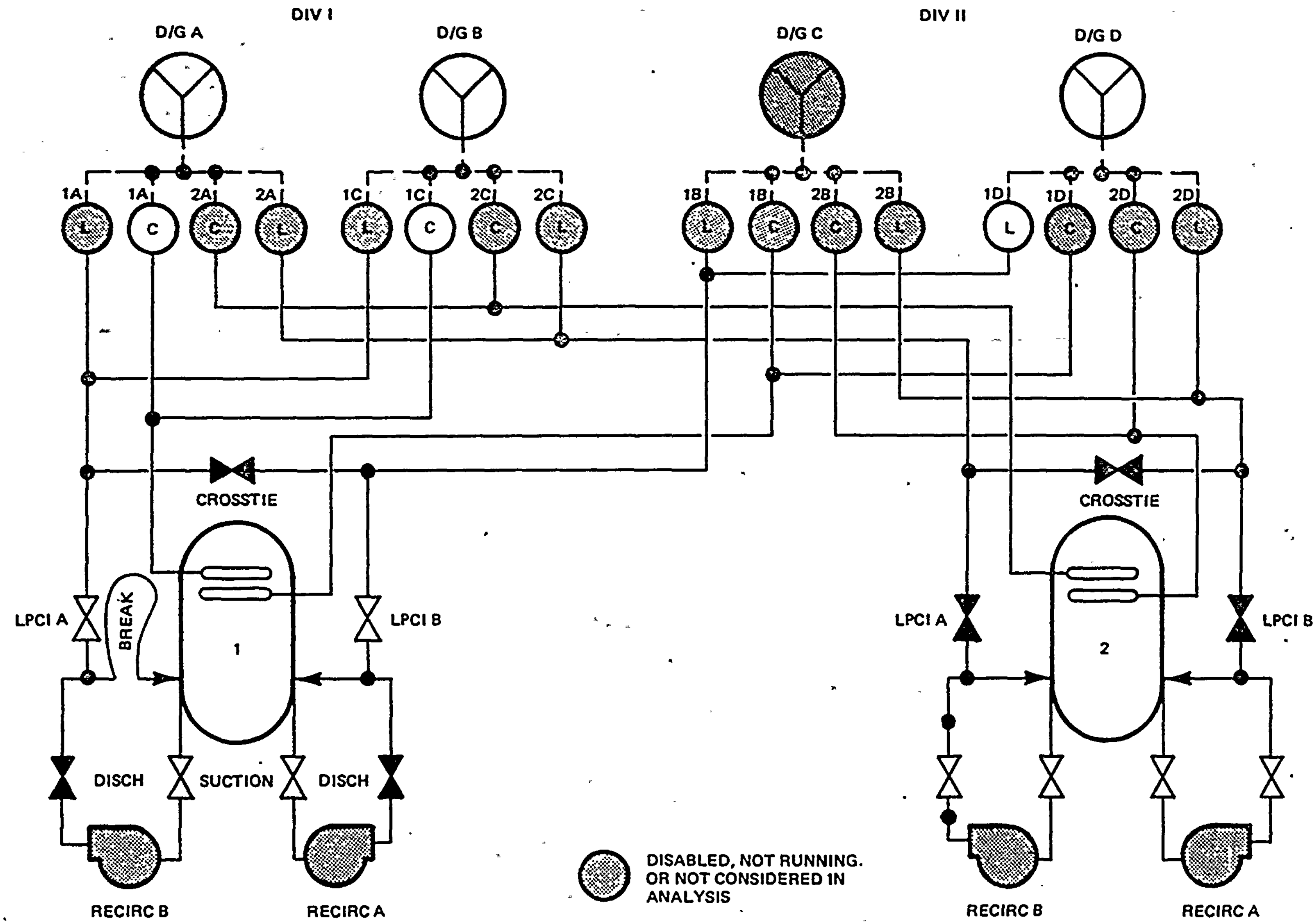


Figure 9 System Mode of Operation During Unit 1 LOCA (Discharge Line Break) – Diesel Failure

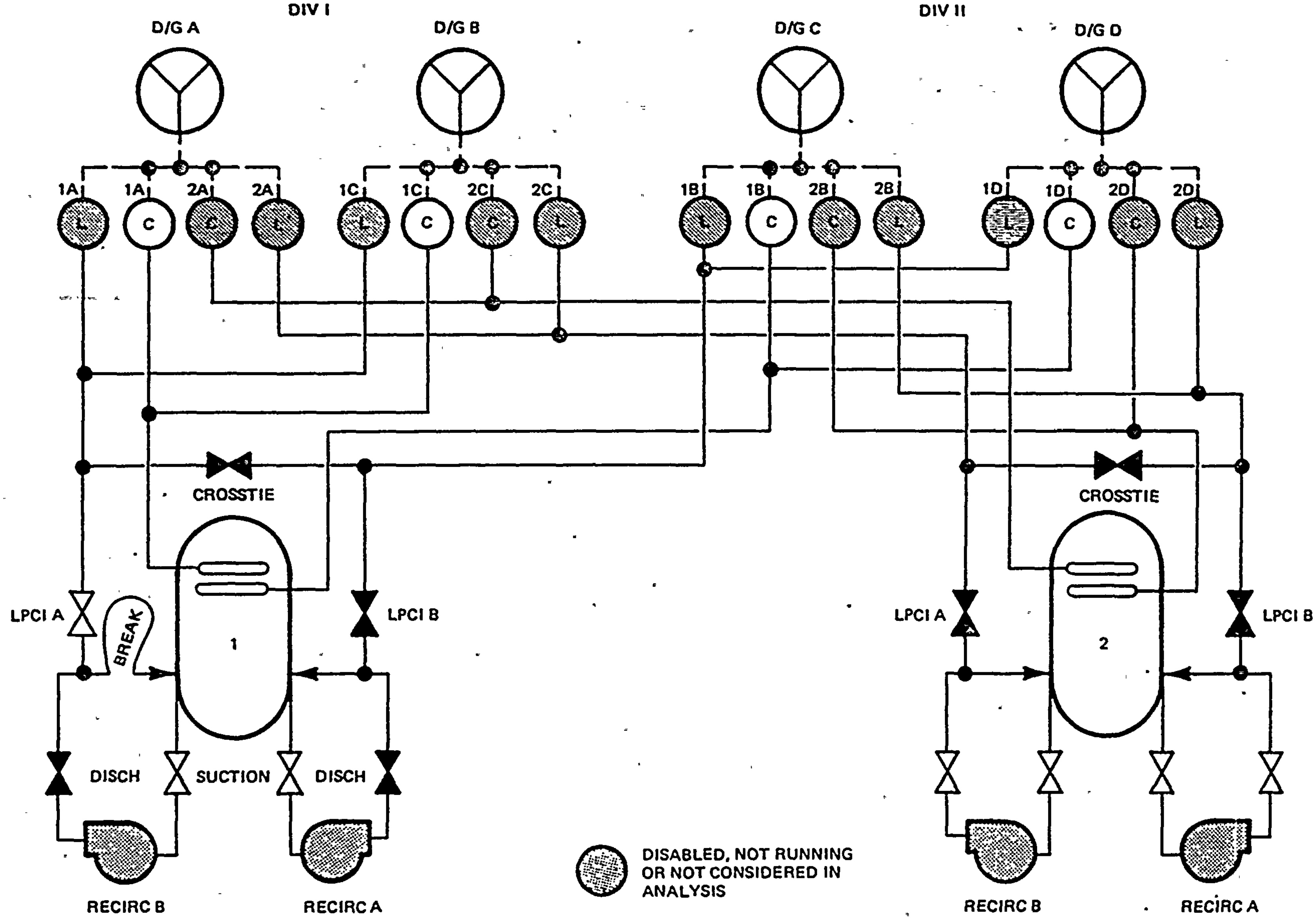


Figure 8 *System Mode of Operation During Unit 1 LOCA (Discharge Line Break) — LPCI Injection Valve Failure*

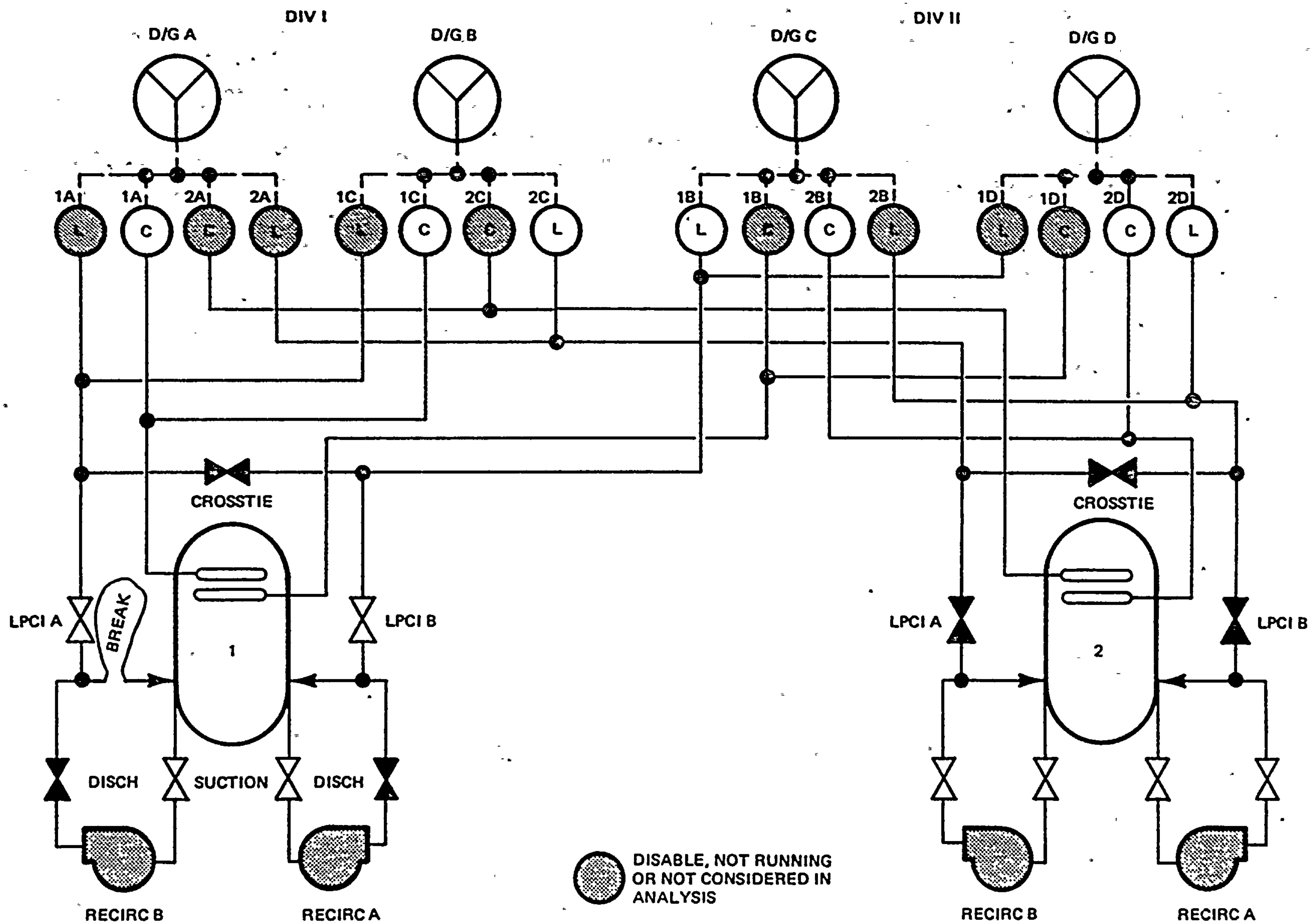


Figure 11 *System Mode of Operation During Unit 1 LOCA (Discharge Line Break) – Opposite Unit Spurious Accident Signal*

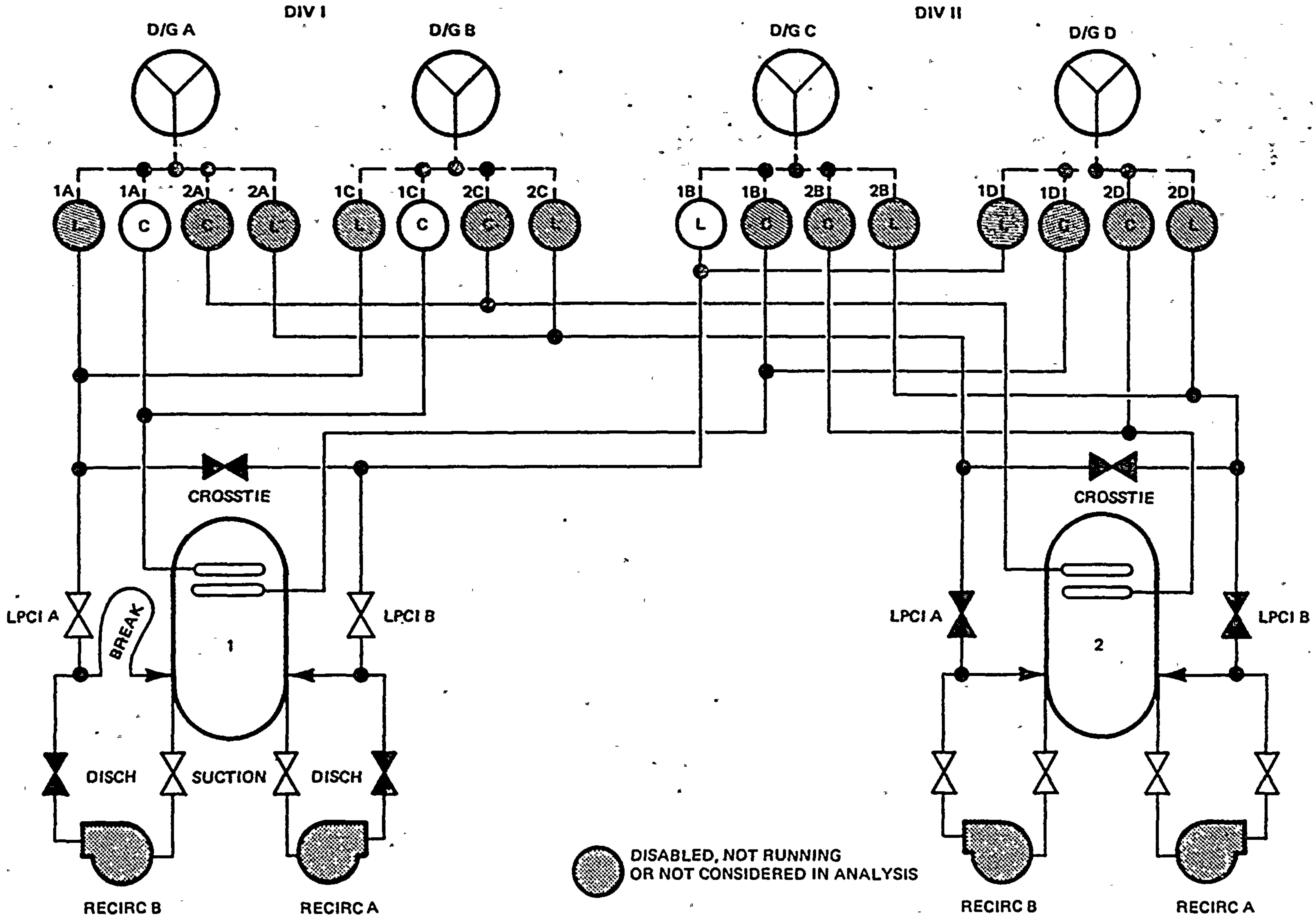


Figure 10 *System Mode of Operation During Unit 1 LOCA (Discharge Line Break) — Battery Failure*

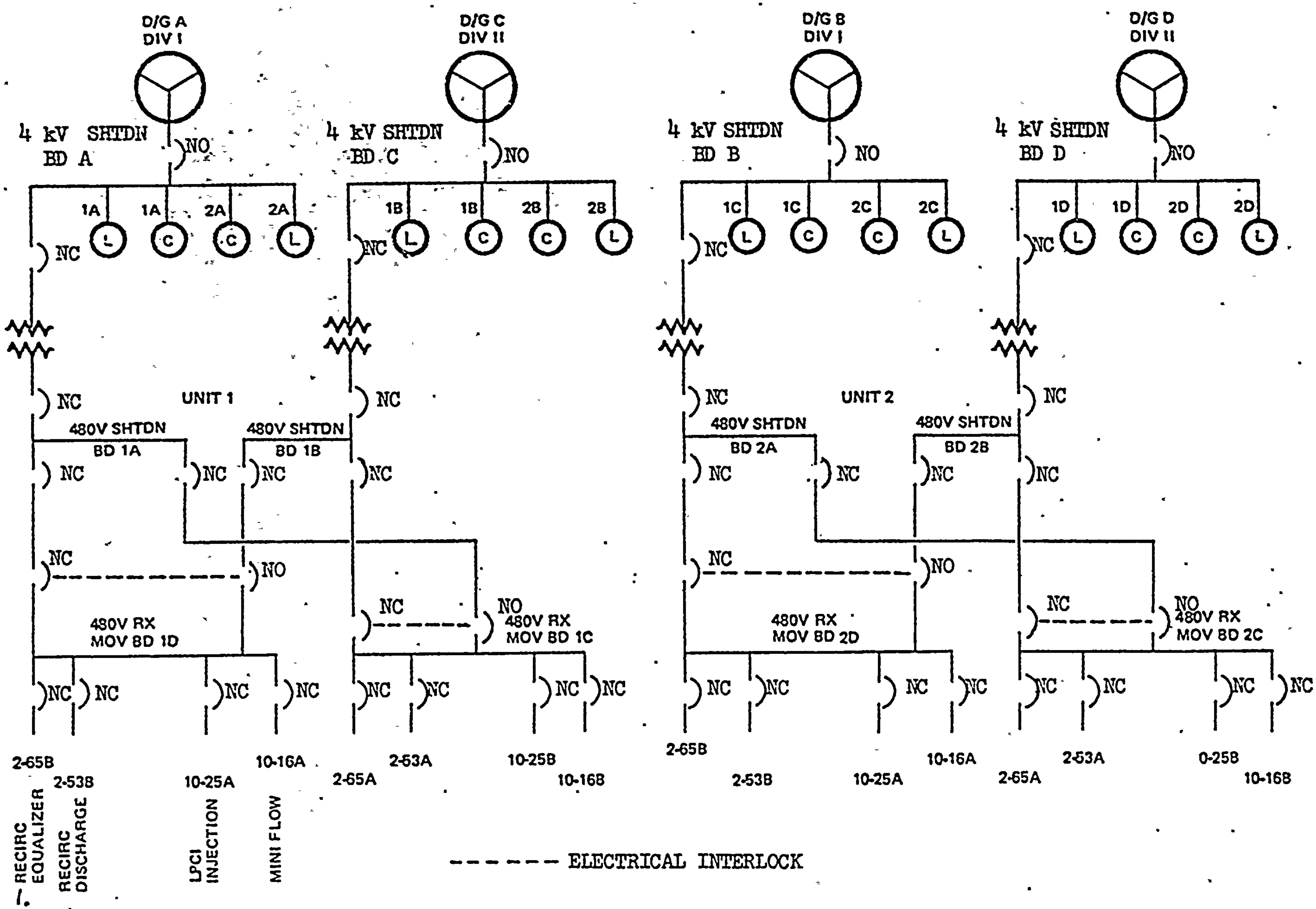


Figure 12 Existing System Valve Bus Arrangement

1. Valve closed and motive power removed.

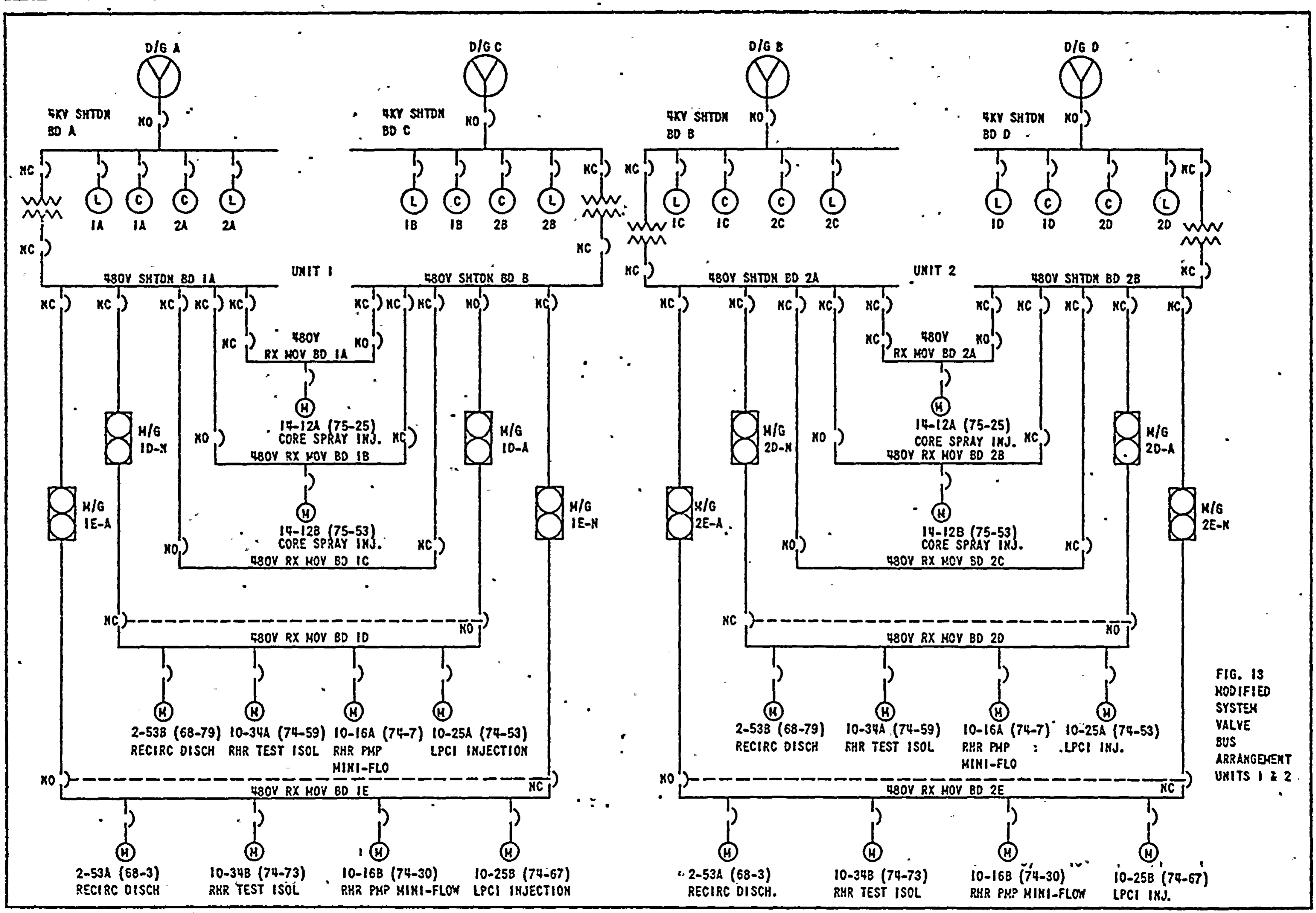
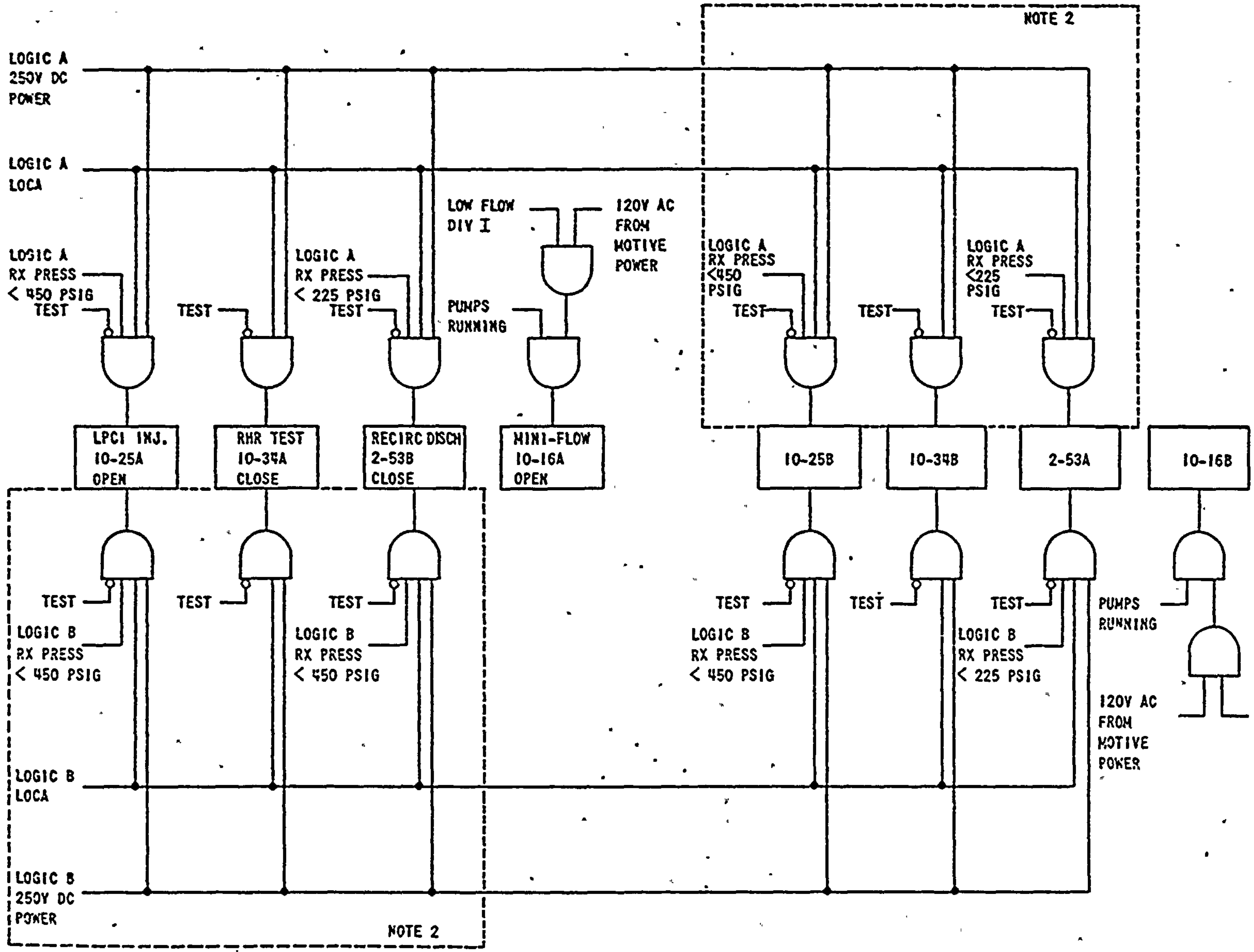


FIG. 13
MODIFIED
SYSTEM
VALVE
BUS
ARRANGEMENT
UNITS 1 & 2



NOTES:

1. THIS DIAGRAM IS FOR REPORT CLARIFICATION ONLY AND IS NOT INTENDED TO REPRESENT THE ENTIRE CONTROL POWER ARRANGEMENT. SEE APPROPRIATE RHR ELEMENTARY FOR COMPLETE WIRING LOCATIONS AND INTERFACES.
2. GUARDED CIRCUITS OF SEPARATE RELAYS AND CONTACTS IN LOGIC PANELS - THE ONLY AREA WHERE CIRCUITS INTERFACE.

   RELAYS ARE ENCASED IN SHEET STEEL; CABLING IS ROUTED IN FLEX CONDUIT; TERMINATIONS ARE IN METAL JUNCTION BOX.
3. ADMINISTRATIVE CONTROL ON TEST FUNCTIONS.

FIGURE 14 SYSTEM VALVE CONTROL POWER ARRANGEMENT