UNITED STATES NUCLEAR REGULATORY COMMISSION

NUCLEAR POWER PLANT SYSTEM SOURCEBOOK

NINE MILE POINT 1

50-220

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NUCLEAR POWER PLANT SYSTEM SOURCEBOOK

NINE MILE POINT 1

50-220

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CAUTION

The information in this report has been developed over an extended period of time based on a site visit, the Final Safety Analysis Report, system and layout drawings, and other published information. To the best of our knowledge, it accurately reflects the plant configuration at the time the information was obtained, however, the information in this document has not been independently verified by the licensee or the NRC.

NOTICE

This sourcebook will be periodically updated with new and/or replacement pages as appropriate to incorporate additional information on this reactor plant. Technical errors in this report should be brought to the attention of the following:

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Correction and other recommended changes should be submitted in the form of marked up copies of the affected text, tables or figures. Supporting documentation should be included if possible.

NINE MILE POINT 1
RECORD OF REVISIONS

REVISION	ISSUE	COMMENTS
0	11/89	Original report

NINE MILE POINT 1 SYSTEM SOURCEBOOK

This sourcebook contains summary information on the Nine Mile Point 1 nuclear power plant. Summary data on this plant are presented in Section 1, and similar nuclear power plants are identified in Section 2. Information on selected reactor plant systems is presented in Section 3, and the site and building layout is illustrated in Section 4. A bibliography of reports that describe features of this plant or site is presented in Section 5. Symbols used in the system and layout drawings are defined in Appendix A. Terms used in data tables are defined in Appendix B.

1. SUMMARY DATA ON PLANT

Basic information on the Nine Mile Point 1 nuclear plant is listed below:

- Docket number	50-220
- Operator	Niagara Mohawk Power Corporation
- Location	Scriba, NY
- Commercial operation date	12/69
- Reactor type	BWR/2
- NSSS vendor	General Electric
- Power (MWt/MWe)	1850/610
- Architect-engineer	Niagara Mohawk Power Corporation
- Containment type	Steel drywell and wetwell (Mark I)

2. IDENTIFICATION OF SIMILAR NUCLEAR POWER PLANTS

Nine Mile Point 1 has a General Electric BWR/2 nuclear steam supply system and a Mark I containment incorporating the drywell/pressure suppression concept. It also has a secondary containment structure of reinforced concrete. Oyster Creek is the only other BWR/2 plant in the United States.

3. SYSTEM INFORMATION

This section contains descriptions of selected systems at Nine Mile Point 1 in terms of general function, operation, system success criteria, major components, and support system requirements. A summary of major systems at Nine Mile Point 1 is presented in Table 3-1. In the "Report Section" column of this table, a section reference (i.e. 3.1, 3.2, etc.) is provided for all systems that are described in this report. An entry of "X" in this column means that the system is not described in this report. In the "FSAR Section Reference" column, a cross-reference is provided to the section of the Final Safety Analysis Report where additional information on each system can be found. Other sources of information on this plant are identified in the bibliography in Section 5.

Several cooling water systems are identified in Table 3-1. The functional relationships that exist among cooling water systems required for safe shutdown are shown in Figure 3-1. Details on the individual cooling water systems are provided in the report sections identified in Table 3-1.

Table 3-1. Summary of Nine Mile Point 1 Systems Covered in this Report

Generic System Name	Plant-Specific System Name	Report Section	FSAR Section Reference
Reactor Heat Removal Systems			
- Reactor Coolant System (RCS)	Same	3.1	5
- Isolation Condenser System	Emergency Cooling (EC) System	3.2	5.E
- Emergency Core Cooling Systems (ECCS)	Same		
- High-Pressure Injection & Recirculation	Feedwater Coolant Injection (FWCI) System	3.3	7.I
- Low-pressure Injection & Recirculation	Core Spray (CS) System	3.3	7.A
- Automatic Depressurization System (ADS)	Same	3.3	7.A
- Decay Heat Removal (DHR) System (Residual Heat Removal (RHR) System)	Reactor Shutdown Cooling System (single-mode)	3.8	10.A
- Main Steam and Power Conversion Systems	Main Steam System,	X	11.A
	Condensate Transfer System,	X	11.B.7
	Feedwater System,	X	11.B.9
	Circulating Water System	X	11.B.4
- Other Heat Removal Systems	Reactor Head Spray System	X	10.C
Reactor Coolant Inventory Control Systems			
- Reactor Water Cleanup (RWCU) System	Reactor Cleanup System	X	10.B
- ECCS	See above	-	-
- Control Rod Drive Hydraulic System (CRDHS)	Same	3.6	10.C

Table 3-1. Summary of Nine Mile Point 1 Systems Covered in this Report (Continued)

Generic System Name	Plant-Specific System Name	Report Section	FSAR Section Reference
Containment Systems			
- Primary Containment	Same (drywell and pressure suppression chamber)	X	6.A, 6.B
- Secondary Containment	Same (Reactor Building)	X	6.C
- Standby Gas Treatment System (SGTS)	None noted	-	-
- Containment Heat Removal Systems			
- Suppression Pool Cooling System	Same (an operating mode of the Containment Spray system)	3.4	7.B
- Containment Spray System	Same	3.4	7.B
- Containment Fan Cooler System	None noted	-	-
- Containment Normal Ventilation Systems	Containment Ventilation System	X	6.E
- Combustible Gas Control Systems	Containment Inerting System,	X	7.G.2
	Containment Atmospheric Dilution System	X	7.G.3
Reactor and Reactivity Control Systems			
- Reactor Core	Same	X	4
- Control Rod System	Control Rod Drive Mechanisms	X	4.B.6
- Chemical Poison System	Liquid Poison Injection System	X	7.C
Instrumentation & Control (I&C) Systems			
- Reactor Protection System (RPS)	Same	3.5	8.A
- Engineered Safety Feature Actuation System (ESFAS)	Instrumentation Systems, Regulating Systems	3.5	8.C 8.B

Table 3-1. Summary of Nine Mile Point 1 Systems Covered in this Report (Continued)

Generic System Name	Plant-Specific System Name	Report Section	FSAR Section Reference
Instrumentation & Control (I&C) Systems (continued)			
- Remote Shutdown System	Same	3.4	10.L
- Other I&C Systems	None	-	-
Support Systems			
- Class 1E Electric Power System	Same	3.6	9.B.4
- Non-Class 1E Electric Power System	Same	3.6	9.B.1
- Diesel Generator Auxiliary Systems	Same	3.6	9.B.4.1
- Component Cooling Water (CCW) System	Reactor Building Closed Cooling Water (RBCCW) System	X	10.D
- Service Water System (SWS)	Same	X	10.F
- Residual Heat Removal Service Water (RHRSW) System	Containment Spray Raw Water System, (CCRWS),	3.4	7.B
	RBCCW, see above	X	10.D
- Other Cooling Water Systems	Turbine Building Closed Loop Cooling Water System	X	10.E
- Fire Protection Systems	Same	X	10.K
- Room Heating, Ventilating, and Air-Conditioning (HVAC) Systems	Emergency Ventilation System	X	7.H

Table 3-1. Summary of Nine Mile Point 1 Systems Covered in this Report (Continued)

Generic System Name	Plant-Specific System Name	Report Section	FSAR Section Reference
Support Systems (continued)			
- Instrument and Service Air Systems	Same	X	10.I
- Refueling and Fuel Storage Systems	Fuel and Reactor Components Handling System	X	10.J
- Radioactive Waste Systems	Same	X	12.A
- Radiation Protection Systems	Same	X	12.B

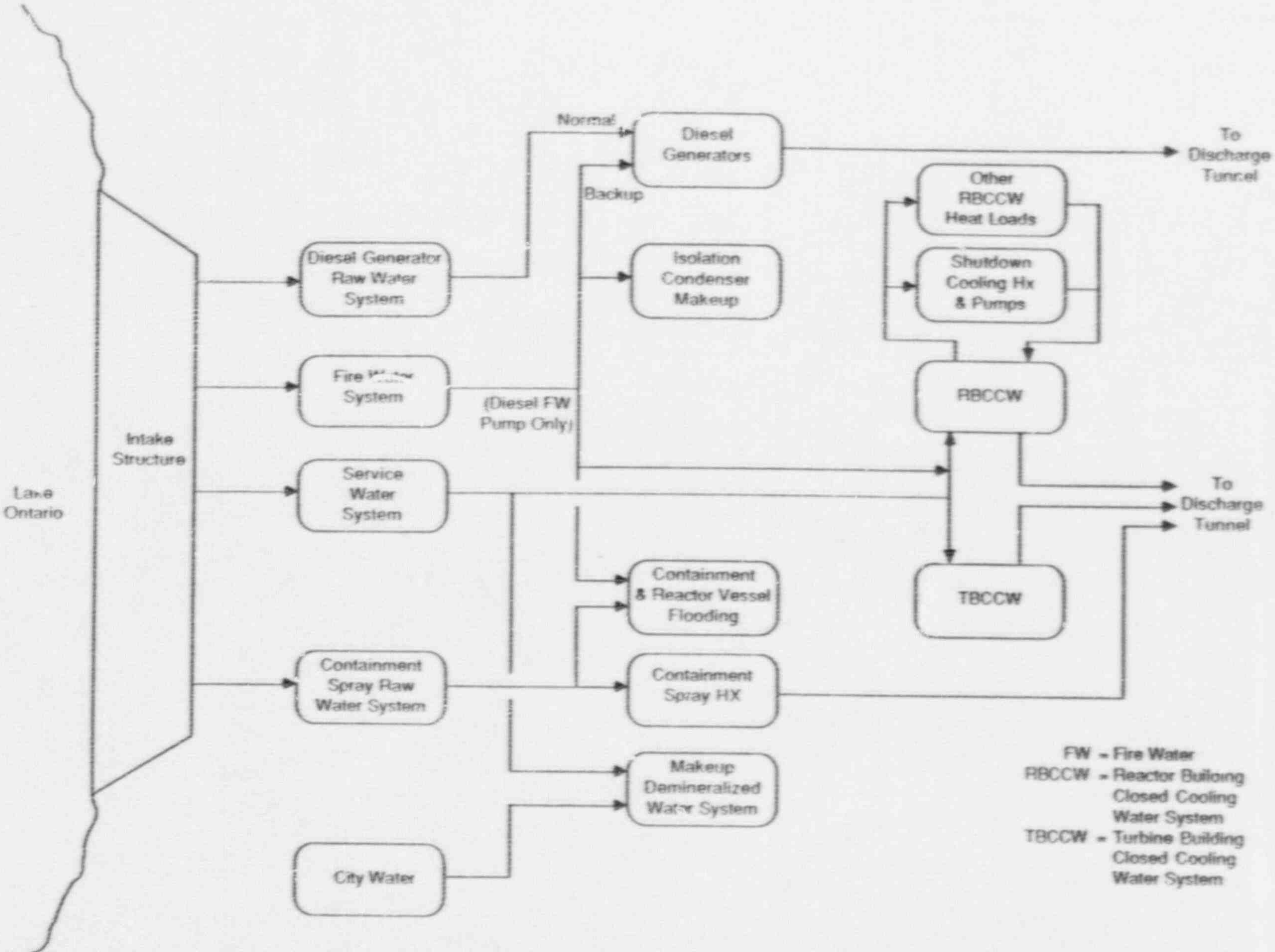


Figure 3-1. Cooling Water Systems Functional Diagram for Nine Mile Point 1

3.1 REACTOR COOLANT SYSTEM (RCS)

3.1.1 System Function

The RCS, also called the Nuclear Steam Supply System (NSSS), is responsible for directing the steam produced in the reactor to the turbine where it is used to rotate a generator and produce electricity. The RCS pressure boundary also establishes a boundary against the uncontrolled release of radioactive material from the reactor core and primary coolant.

3.1.2 System Definition

The RCS includes: (a) the reactor vessel, (b) five recirculation loops, (c) five recirculation pumps, (d) 6 solenoid-actuated relief valves, (e) 16 safety valves and (f) connected piping out to a suitable isolation valve boundary. Simplified diagrams of the RCS and important system interfaces are shown in Figures 3.1-1 and 3.1-2. A summary of data on selected RCS components is presented in Table 3.1-1.

3.1.3 System Operation

During power operation, circulation in the RCS is maintained by one recirculation pump in each of the five recirculation loops. The steam-water mixture flows upward in the core to the steam dryers and separators where the entrained liquid is removed. The remaining "dry" steam is piped to the turbine via the two main steam lines. Condensate from the turbine is returned to the RCS as feedwater.

Following a transient that involves the loss of normal heat transfer path, heat transfer from the RCS is accomplished by: (a) emergency condensers which vent directly to atmosphere and which have a heat removal capacity of 3 percent of maximum reactor steam flow (see Section 3.2); or (b) by opening the safety valves or automatic depressurization system (ADS) relief valves which discharge directly to the suppression pool (see Section 3.3).

A LOCA inside containment also dumps heat to the suppression chamber. In this case, the Core Spray system is used to prevent overheating of the fuel by supplying make-up water to the reactor vessel at low pressure (see Section 3.3). Heat removal from the containment is provided by the Containment Spray system (see Section 3.4). Actuation systems provide for automatic closure of the main steam isolation valves (MSIVs) and isolation of other lines connected to the RCS.

3.1.4 System Success Criteria

The RCS success criteria can be described in terms of LOCA and transient mitigation as follows:

- An unmitigatable LOCA is not initiated.
- If a mitigatable LOCA is initiated, then LOCA mitigating systems are successful.
- If a transient is initiated, then either:
    - RCS integrity is maintained and transient mitigating systems are successful, or
    - RCS integrity is not maintained, leading to a LOCA-like condition (i.e. stuck-open safety or relief valve, reactor coolant pump seal failure), and LOCA mitigating systems are successful.

3.1.5 Component Information

A. RCS
  1. Steam flow: 7.29 x 10^6 lb/hr
  2. Normal operating pressure: 1030 psig at 550° F.

B. Solenoid Actuated Relief Valves (6)
  1. Set pressure: 2 @ 1090 psig
     2 @ 1095 psig
     2 @ 1100 psig
  2. Relief Capacity: 600,000 lb/hr each at 1130 psig

C. Safety Valves (16)
  1. Set pressure: 1218 to 1254 psig
  2. Relief Capacity: 633,000 to 651,000 lb/hr

D. Recirculation Pumps (5)
  1. Rated flow: 7,200 to 36,000 gpm
  2. Type: Vertical centrifugal

3.1.6 Support Systems and Interfaces

A. Motive Power
The recirculation pumps are supplied with non-Class 1E power from variable frequency motor generator sets.

B. MSIV Operating Power
There are two MSIV's located on each main steam line, one air-operated wye-globe type valve (outside containment), and 1 AC motor-operated wye-globe type valve (inside of drywell). The instrument air system which supports the normal operation of the air-operated MSIV has not been determined. It is very likely that this valve's operation is controlled by an AC and a DC solenoid pilot valve. It is assumed that both solenoid valves must be de-energized to cause MSIV closure. In other similar BWRs, this design prevents spurious closure of an MSIV if a single solenoid valve should fail. The AC motor-operated MSIV's are powered by Class 1E power from the 600 VAC power boards 161 or 171.

C. Recirculation Pump Cooling
The Reactor Building Closed Loop Cooling Water System provides cooling water to the recirculation pump coolers.

3.1.7 Section 3.1 References

1. Final Safety Analysis Report, Nine Mile Point 1 Nuclear Power Station, Section V.

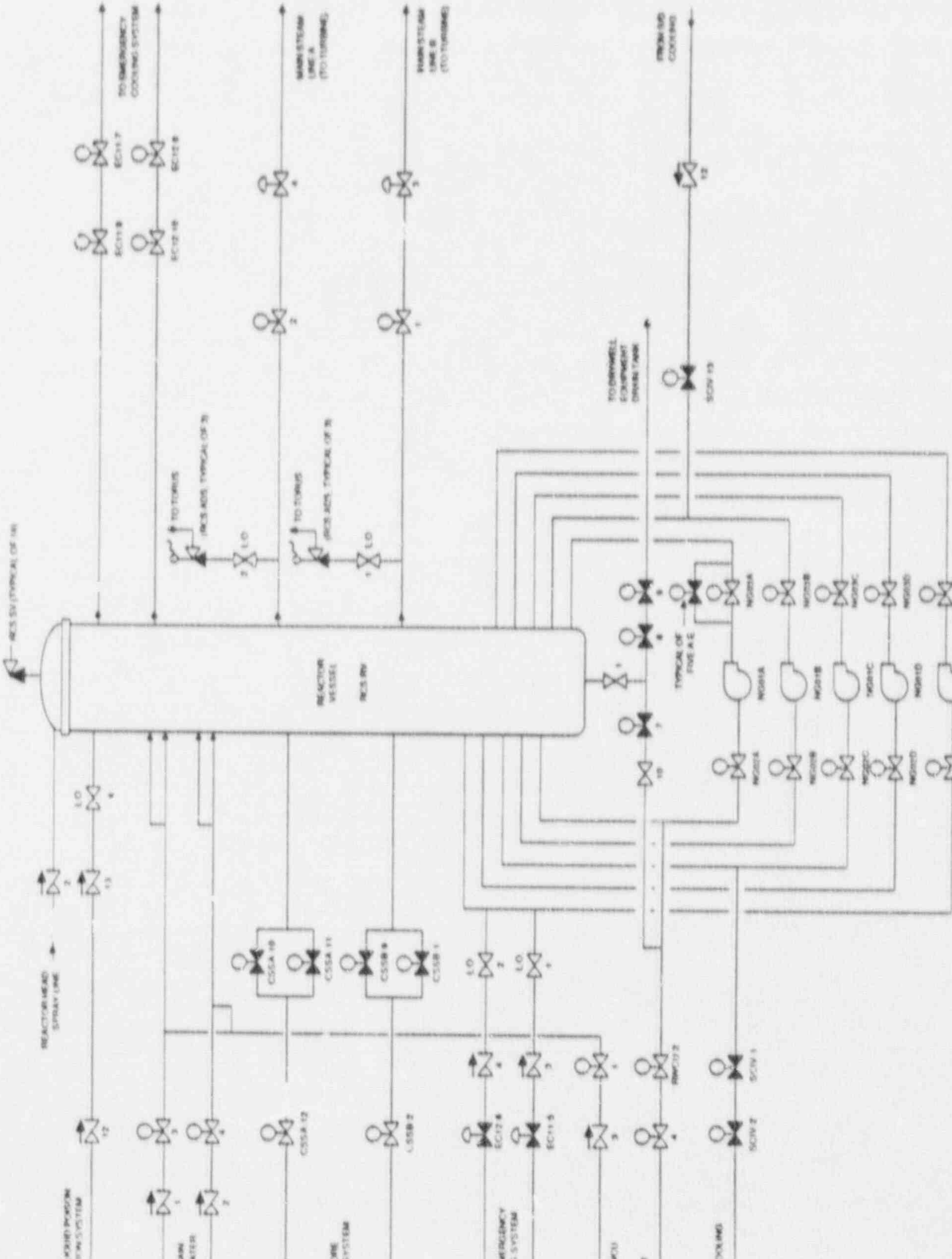


Figure 3.1-1. Nine Mile Point 1 Reactor Coolant System

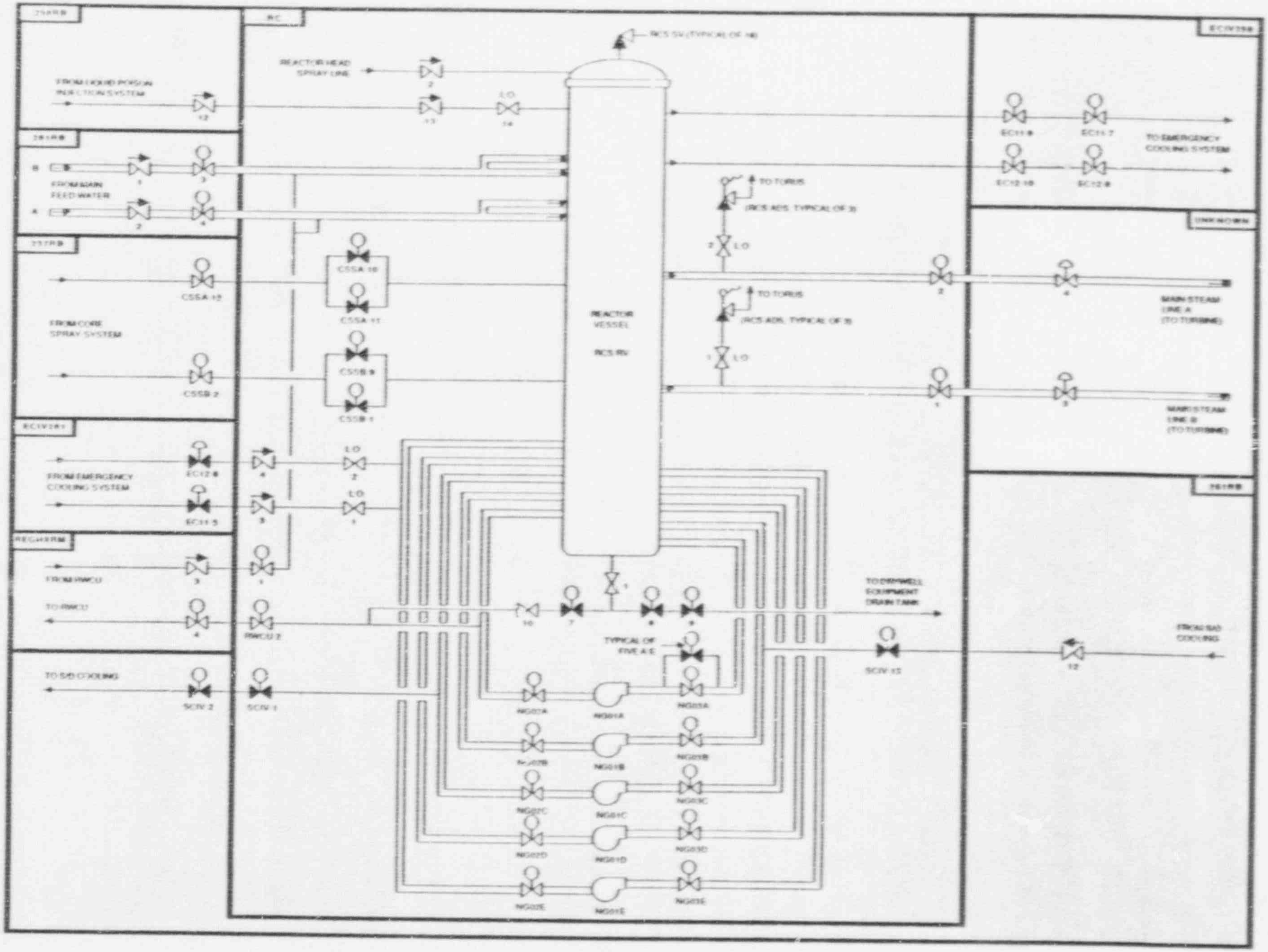


Figure 3.1-2. Nine Mile Point 1 Reactor Coolant System Showing Component Locations

Table 3.1-1. Nine Mile Point 1 Reactor Coolant System Data Summary for Selected Components

COMPONENT ID	COMP. TYPE	LOCATION	POWER SOURCE	VOLTAGE	POWER SOURCE LOCATION	EMERG. LOAD GRP.
CSSA-10	MOV	RC	EP-BS-171	600	261RB	AC/103
CSSA-11	MOV	RC	EP-BS-161	600	261RB	AC/102
CSSB-1	MOV	RC	EP-BS-161	600	261RB	AC/102
CSSB-9	MOV	RC	EP-BS-171	600	261RB	AC/103
EC11-7	MOV	ECIV298	EP-VB12	125	DCVB12	DC/12
EC11-9	MOV	ECIV298	EP-BS-161	600	261RB	AC/102
EC12-10	MOV	ECIV298	EP-BS-171	600	261RB	AC/103
EC12-8	MOV	ECIV298	EP-VB12	125	DCVB12	DC/12
RCS-ADS	PORV	RC				
RCS-SV	SV	RC				
RWCU-2	MOV	RC	EP-BS-167	600	PB167	AC/102
SCIV-1	MOV	RC	EP-BS-167	600	PB167	AC/102
SCIV-13	MOV	RC	EP-BS-167	600	PB167	AC/102
SCIV-2	MOV	261RB	EP-VB12	125	DCVB12	DC/12

3.2 EMERGENCY COOLING (EC) SYSTEM

3.2.1 System Function

The emergency cooling system, also known as the isolation condenser system, provides for decay heat removal from the reactor fuel in the event that reactor feedwater capability is lost and the main condenser is not available. Because of its role in emergency cooling, the EC system is considered part of the Emergency Core Cooling System (ECCS, see Section 3.3).

3.2.2 System Definition

The EC system is connected to the reactor and operates by natural circulation. Heat is transferred to the environment by boiling secondary-side water in the condensers and venting steam directly to atmosphere. The EC system consists of two independent emergency cooling loops with two condensers on each loop. One loop is adequate to handle the decay heat load following reactor isolation.

Simplified drawings of the EC system are shown in Figures 3.2-1 and 3.2-2. A summary of data on selected EC system components is presented along with other ECCS components in Table 3.3-1 of Section 3.3.

3.2.3 System Operation

Automatic operation of the emergency cooling system is initiated by high reactor pressure in excess of 1080 psig sustained for ten seconds. To assist in depressurization for small breaks, the system is initiated on sustained (10 seconds) low-low water level (-60 inches). In the standby condition, the steam inlet isolation valves are normally open so that the condenser tubes are continuously at reactor pressure. The system is placed into operation by opening the normally closed condensate return isolation valve (air operated). During operation, steam rises from the reactor vessel to the condenser tubes where it is condensed. As the water condenses, it returns by gravity flow to the suction of a reactor recirculating pump and, thus, to the reactor vessel. Sufficient make-up water is available from gravity feed makeup tanks to permit operation for at least eight hours.

The system may also be initiated manually, either from the main/auxiliary control room or independently from two remote shutdown panels. Moreover, even in the event of total a-c power loss, operation of the EC system will proceed normally.

3.2.4 System Success Criteria

For the EC system to be successful, at least one of the two emergency cooling loops must be functional. A loop is considered functional if no valves have failed close, the condensers are operable, and the makeup water for the secondary side of the condensers is available when needed.

3.2.5 Component Information

A. Emergency Condensers
   1. Number of Tube Bundles: 4
   2. System Capacity: 380 x 10^6 Btu/hr
   3. Design Pressure: 1250 psig

B. Emergency Condenser Makeup Tanks (2)
   1. Volume: 40,000 gallons each

3.2.6 Support Systems and Interfaces

A. Control Signals

1. Automatic

   The emergency cooling system is automatically initiated on sustained high reactor pressure or sustained low-low reactor water level. Initiation takes the form of opening the normally closed isolation valve (air operated) on the condensate return lines.

2. Remote Manual

   The emergency cooling system motor or air-operated valves can be actuated by remote manual means from the control room.

B. Motive Power

The EC system motor-operated valves are Class 1E AC or DC loads that can be supplied from the emergency diesel generators or station batteries, as described in Section 3.6.

C. Other

The diesel-driven fire water pump is a backup source of secondary makeup water for the isolation condensers.

3.2.7 Section 3.2 References

1. Final Safety Analysis Report, Nine Mile Point 1 Nuclear Power Station, Section V.E.

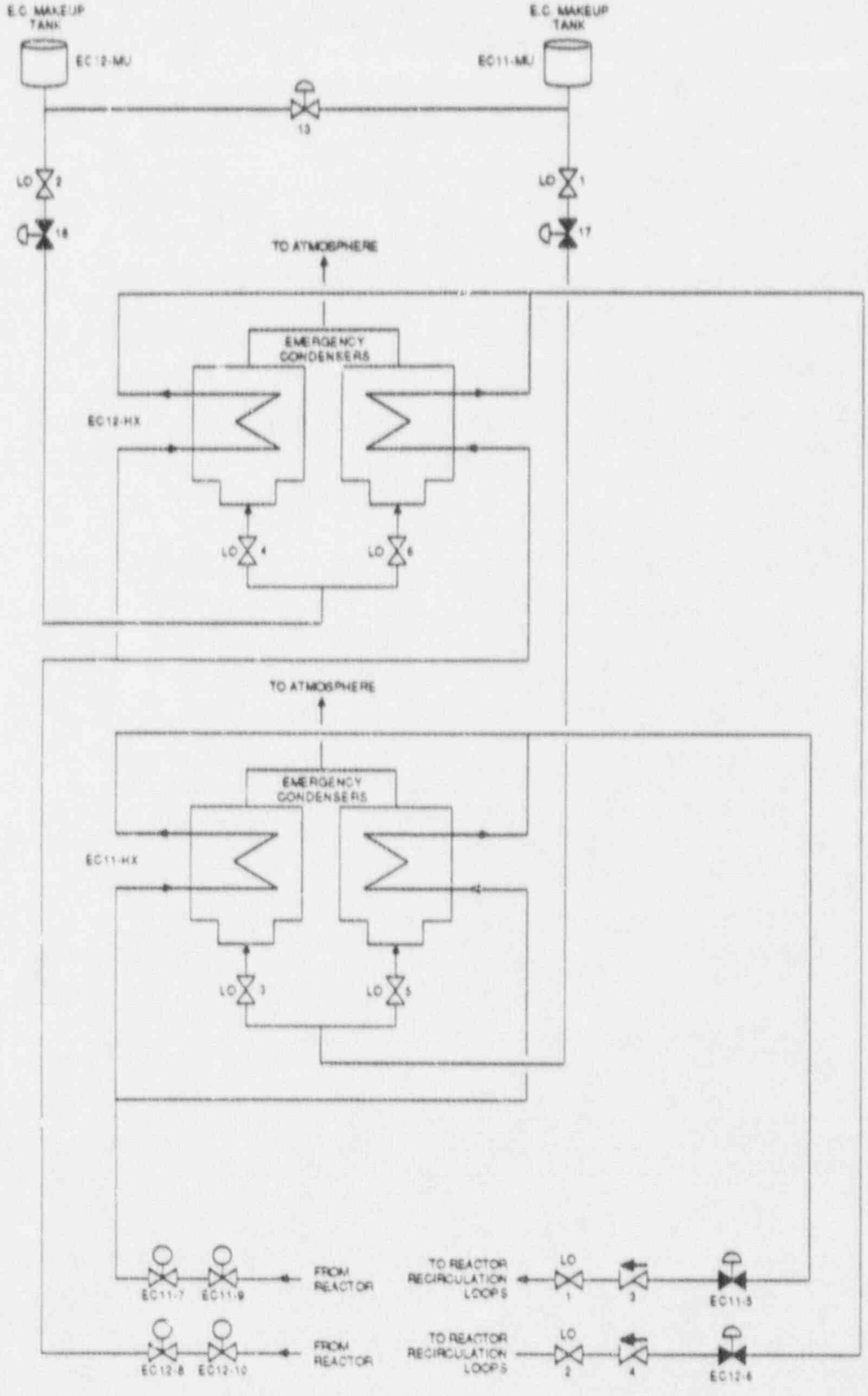


Figure 3.2-1. Nine Mile Point 1 Emergency Cooling System

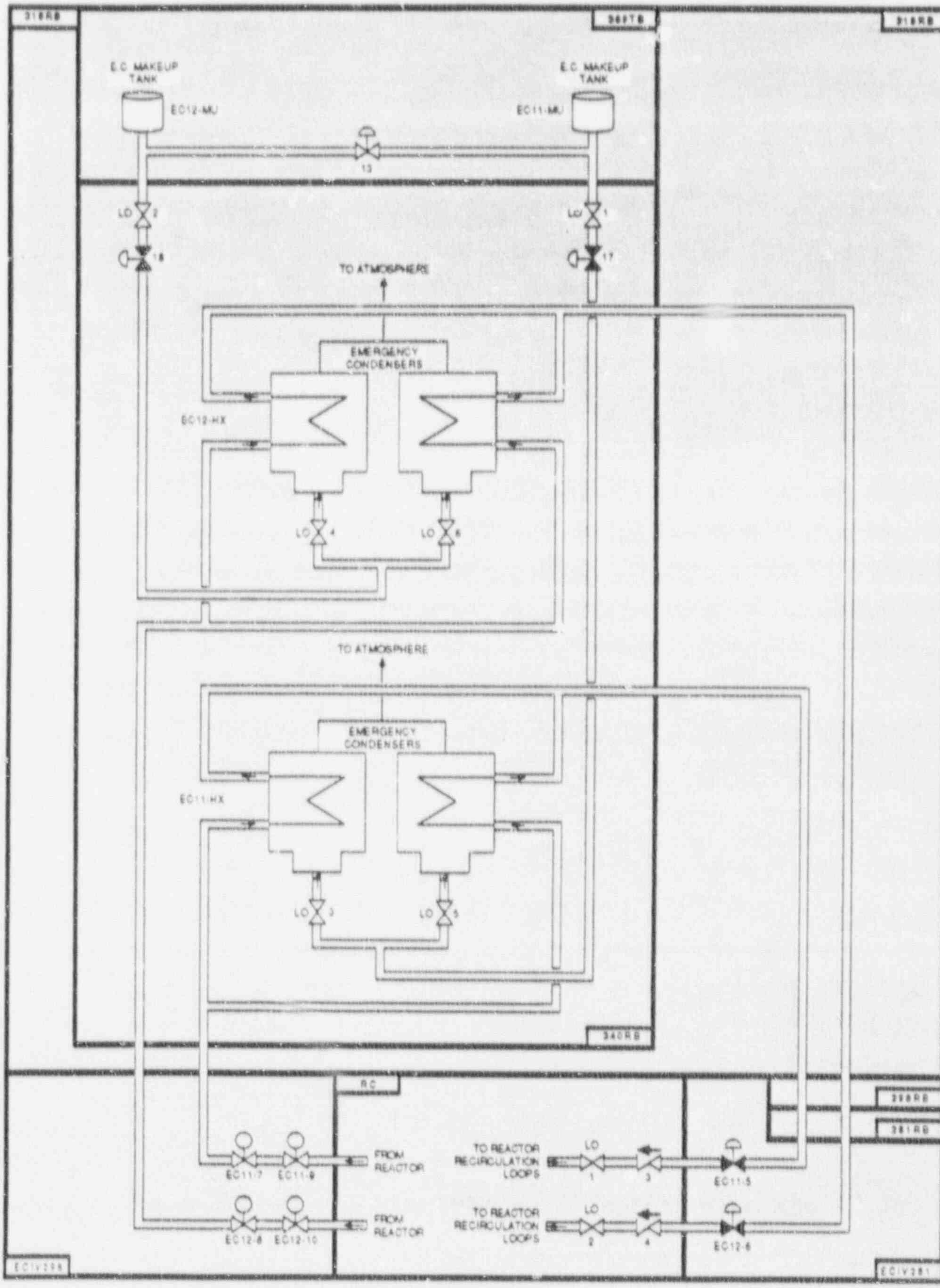


Figure 3.2-2. Nine Mile Point 1 Emergency Cooling System Showing Component Locations

3.3 EMERGENCY CORE COOLING SYSTEM (ECCS)

3.3.1 System Function

The ECCS is an integrated set of subsystems that perform emergency coolant injection and recirculation functions to maintain reactor core coolant inventory and adequate decay heat removal following a LOCA. The ECCS also has a capability for mitigating transients.

3.3.2 System Definition

The emergency core cooling system consists of the following subsystems:

- Core Spray (CS) System
- Feedwater Coolant Injection (FWCI) System
- Automatic Depressurization System (ADS)
- Emergency Cooling (EC) System (see Section 3.2)

The CS system consists of two separate and independent sub-systems to prevent overheating of the fuel following a LOCA by supplying suppression pool water to the reactor vessel at low pressure. Each system has redundant active components, and consists of four motor-driven pumps to supply water to a separate ring header located inside the reactor vessel above the core. After cooling the core, the water returns to the suppression chamber through the break, therefore, no additional coolant supplies are necessary.

The feedwater coolant injection system provides high-pressure injection in the event of a small reactor coolant line break which is not large enough to allow rapid depressurization for the CS system to be effective. It utilizes existing feedwater and condensate components, namely condensate pumps and demineralizers; feedwater booster pumps, heaters, and pumps; piping and valves; and the two CST's. Limited offsite power is required for this system to be functional.

The automatic depressurization system provides for automatic depressurization of the reactor vessel to permit timely operation of the core spray system in the event of small system ruptures. The ADS utilities at least 3 of the 6 solenoid-actuated relief valves that discharge the high pressure steam into the suppression pool.

The emergency cooling system removes residual and decay heat from the reactor vessel in the event that the main condenser is not available, or if a high-pressure condition exists. The system employs natural circulation as the driving head through the emergency condenser tubes. A more detailed description of the EC system is provided in Section 3.2.

Simplified drawings of the core spray system are shown in Figures 3.3-1 and 3.3-2, and the feedwater coolant injection system is shown in Figure 3.3-3. A summary of data on selected ECCS components is presented in Table 3.3-1.

3.3.3 System Operation

All ECCS systems are normally in standby. The manner in which the ECCS operates to protect the reactor core is a function of the rate at which coolant is being lost from the RCS. The core spray pumps start automatically when a low-low reactor water level or high drywell pressure condition exists. However, the normally closed condensate return isolation valves for the EC system do not open until the low reactor pressure permissive switches are tripped. Thus, for small LOCAs, the ADS system may need to be actuated to allow makeup from the low-pressure CS system to reach the RCS.

The FWCI system is initiated by a turbine trip which will signal the motor-driven feedwater pumps to start. As a backup, low reactor water level will also signal the motor-driven pumps to start. Thus, there will be a continuous uninterrupted supply of feedwater to the reactor. However, both condensate and feedwater booster pumps require off-site power to operate; and, as a result, the FWCI system would be non-functional if no off-site power was available.

Actuation of the ADS system occurs if low-low-low reactor water level exists and high-drywell pressure is sustained for 120 seconds. Three of the six solenoid-actuated relief valves are required to discharge properly for the ADS system to be successful. The ADS system may also be actuated manually from the main control room.

Automatic operation of the emergency cooling system is initiated by high reactor pressure in excess of 1080 psig sustained for ten seconds. To assist in depressurization for small breaks, the system is initiated on sustained (10 seconds) low-low water level (-60 inches). In the standby condition, the steam inlet isolation valves are normally open so that the condenser tubes are continuously at reactor pressure. The system is placed into operation by opening the normally closed condensate return isolation valve (air operated). During operation, steam rises from the reactor vessel to the condenser tubes where it is condensed. As the water condenses, it returns by gravity flow to the suction of a reactor recirculating pump and, thus, to the reactor vessel. Sufficient make-up water is available from gravity feed makeup tanks to permit operation for at least eight hours.

The system may also be initiated manually, either from the main/auxiliary control room or independently from two remote shutdown panels. Moreover, even in the event of total AC power loss, operation of the EC system will proceed normally.

3.3.4 System Success Criteria

LOCA mitigation requires that the emergency coolant injection (ECI), emergency coolant recirculation (ECR), and reactor scram functions be accomplished. The success criteria are not clearly defined in the Nine Mile Point 1 FSAR but can be inferred from pump capacities that are defined based on certain design basis accidents that are considered in the licensing office. On this basis, the system success criteria for a large LOCA is the following:

- At least one core spray main pump and one corresponding core spray topping pump (of the four sets available) take suction on the suppression pool and inject into the reactor vessel.

The ECI system success criteria for a small LOCA are the following:

- Three of the six relief valves of the ADS open (i.e. ADS is functional) and the core spray system is functional, as described above under large LOCAs.

For transients, the requirements for adequate core cooling involve the following:

- Either successful operation of the emergency cooling (EC) system, or
- The ADS is functional (see above) and large LOCA mitigating systems are functional (see above).

Note that the EC system provides core cooling only. A high pressure makeup system eventually will be needed to maintain adequate core coolant inventory.

3.3.5 Component Information

A. Core Spray Pumps: Main and Topping (4 sets)
   1. Rated flow per set: 3400 gpm @ 697 ft. head (302 psid)
   2. Rated Capacity: 100%
   3. Type: Centrifugal

B. Emergency Condensers
   1. Number of tube bundles: 4
   2. System capacity: 380×10^6 Btu/hr
   3. Design Pressure: 1250 psig

C. Emergency Condenser Makeup Tanks (2)
   1. Volume: 40,000 gallons

D. Suppression Pool (Torus)
   1. Normal operating water volume: 89,000 ft^3
   2. Maximum operating water temperature: 90°F
   3. Design temperature: 205°F
   4. Design pressure: 56 psig

E. Solenoid Actuated Relief Valves (6)
   1. Set pressure: 2 @ 1090 psig
      2 @ 1095 psig
      2 @ 1100 psig
   2. Relief capacity: 600,000 lb/hr each at 1130 psig

F. Feedwater Pumps (3)
   1. Rated flow: 3800 gpm each
   2. Rated capacity: 100%
   3. Type: Centrifugal

3.3.6 Support Systems and Interfaces

A. Control Signals
   1. Automatic
      a. The core spray pumps are started upon low-low water level or high drywell pressure.
      b. Once reactor pressure drops below a certain threshold and the core spray pumps are operating, the four parallel normally closed isolation valves (two in each injection line) are opened.
      c. The ADS system is actuated upon coincident signals of low-low-low reactor water level and sustained high drywell pressure (120 seconds).
      d. The normally closed condensate return isolation valves of the EC system are opened by high reactor pressure (1080 psig) sustained for ten seconds.
      e. The motor-driven feedwater pumps are automatically actuated on a turbine-trip signal or low reactor water level.

   2. Remote Manual
      ECCS pumps and valves and the ADS can be actuated by remote manual means from the main control room.

B. Motive Power
   1. The CS motor-driven pumps and motor-operated valves are Class 1E AC and DC loads that can be supplied from the emergency diesel generators or station batteries. The FWCI system needs offsite power to operate.
   2. The ADS solenoid-actuated pressure relief valves are powered from the station battery systems.

C. Other
  1. Lubrication and cooling of the ECCS pumps are all supplied locally.
  2. Details on equipment room cooling provisions have not been determined.
  3. The containment spray raw water system (CSRWS, see Section 3.4) and the diesel-driven fire water pump are backup sources of core and containment flooding.

3.3.7 Section 3.3 References

1. Final Safety Analysis Report, Nine Mile Point 1 Nuclear Power Station, Sections VII and XV.

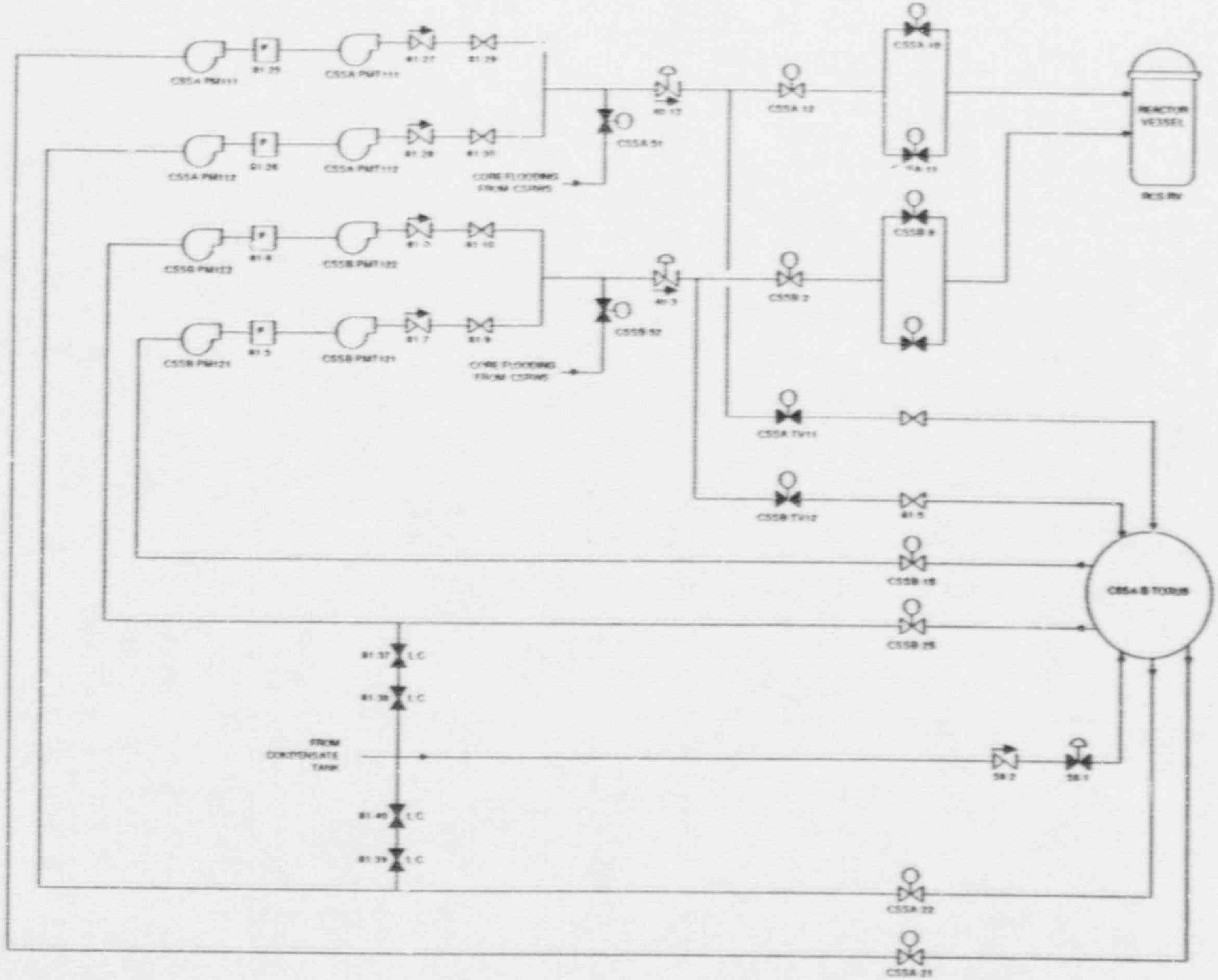


Figure 3.3-1. Nine Mile Point 1 Core Spray System

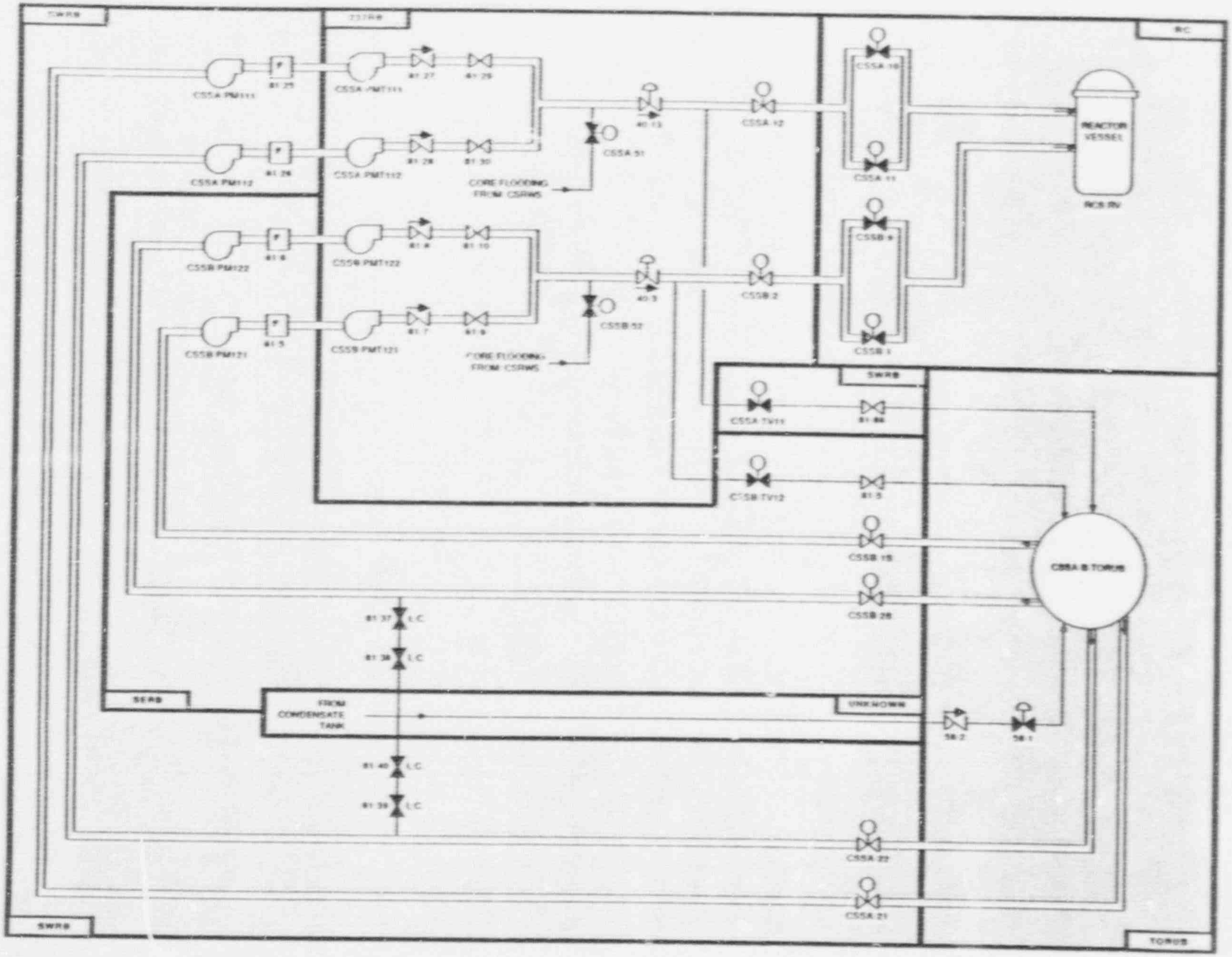


Figure 3.3-2. Nine Mile Point 1 Core Spray System Showing Component Locations

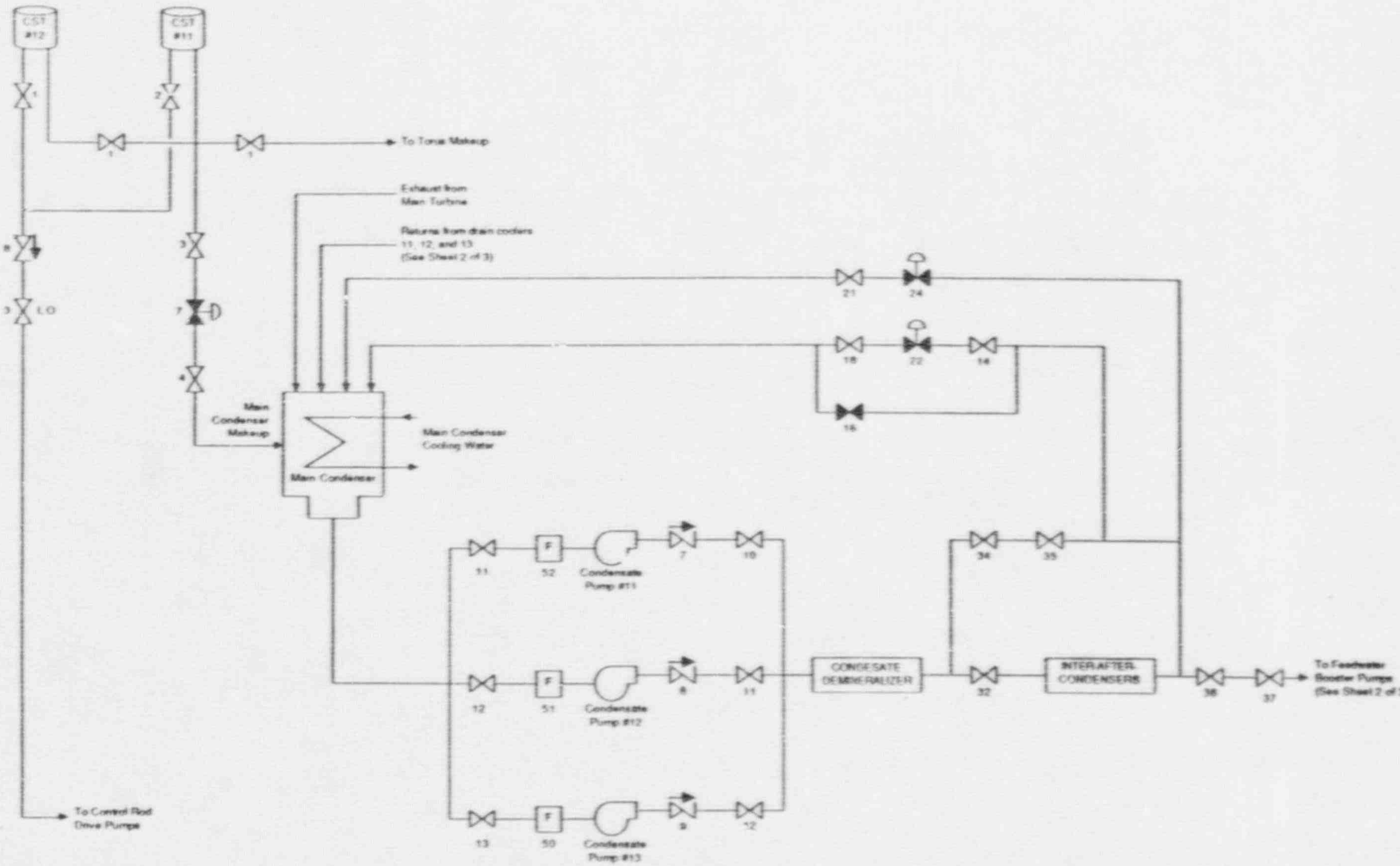


Figure 3.3-3. Nine Mile Point 1 Feedwater Coolant Injection System (Sheet 1 of 3)

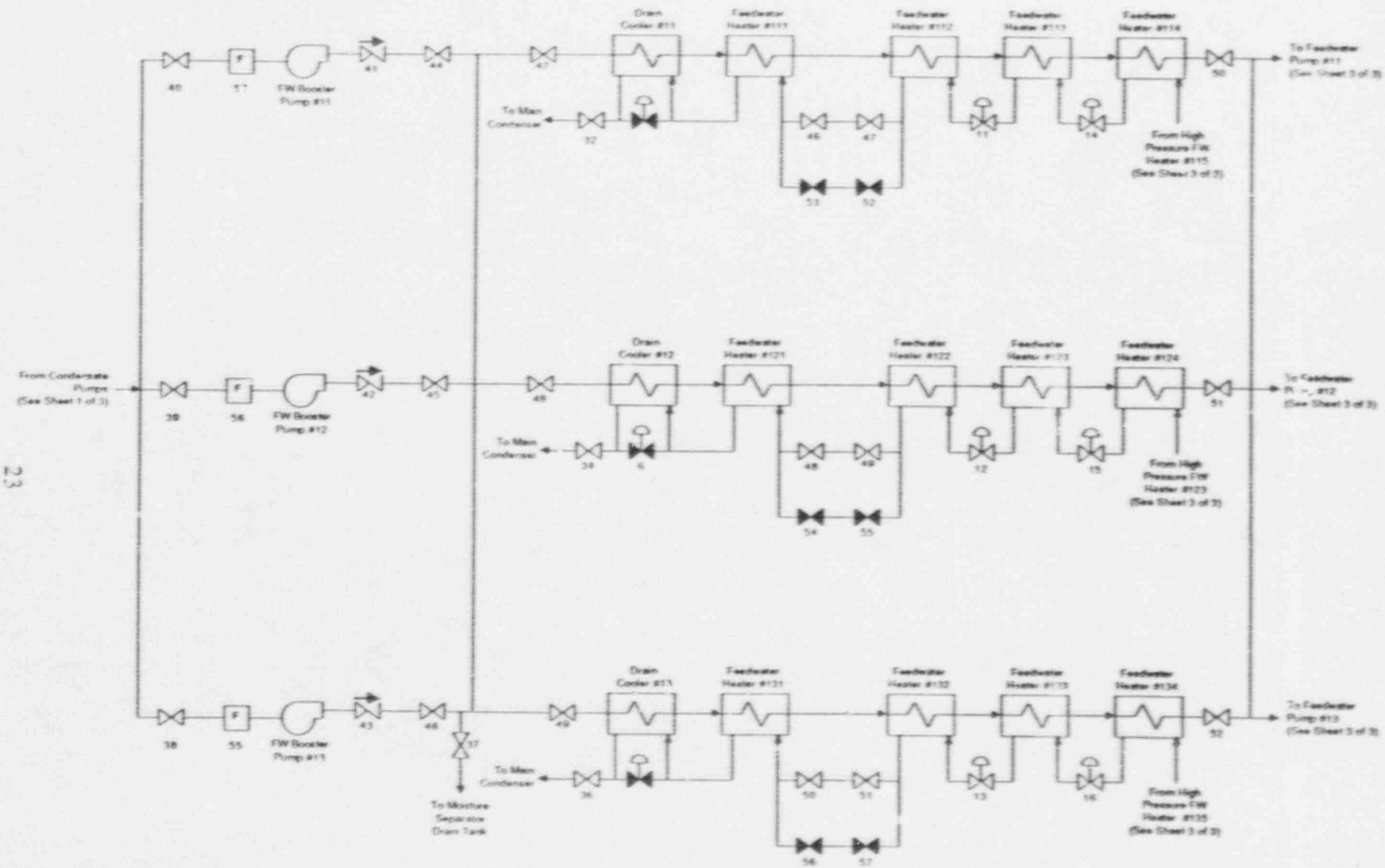


Figure 3.3-3. Nine Mile Point 1 Feedwater Coolant Injection System (Sheet 2 of 3)

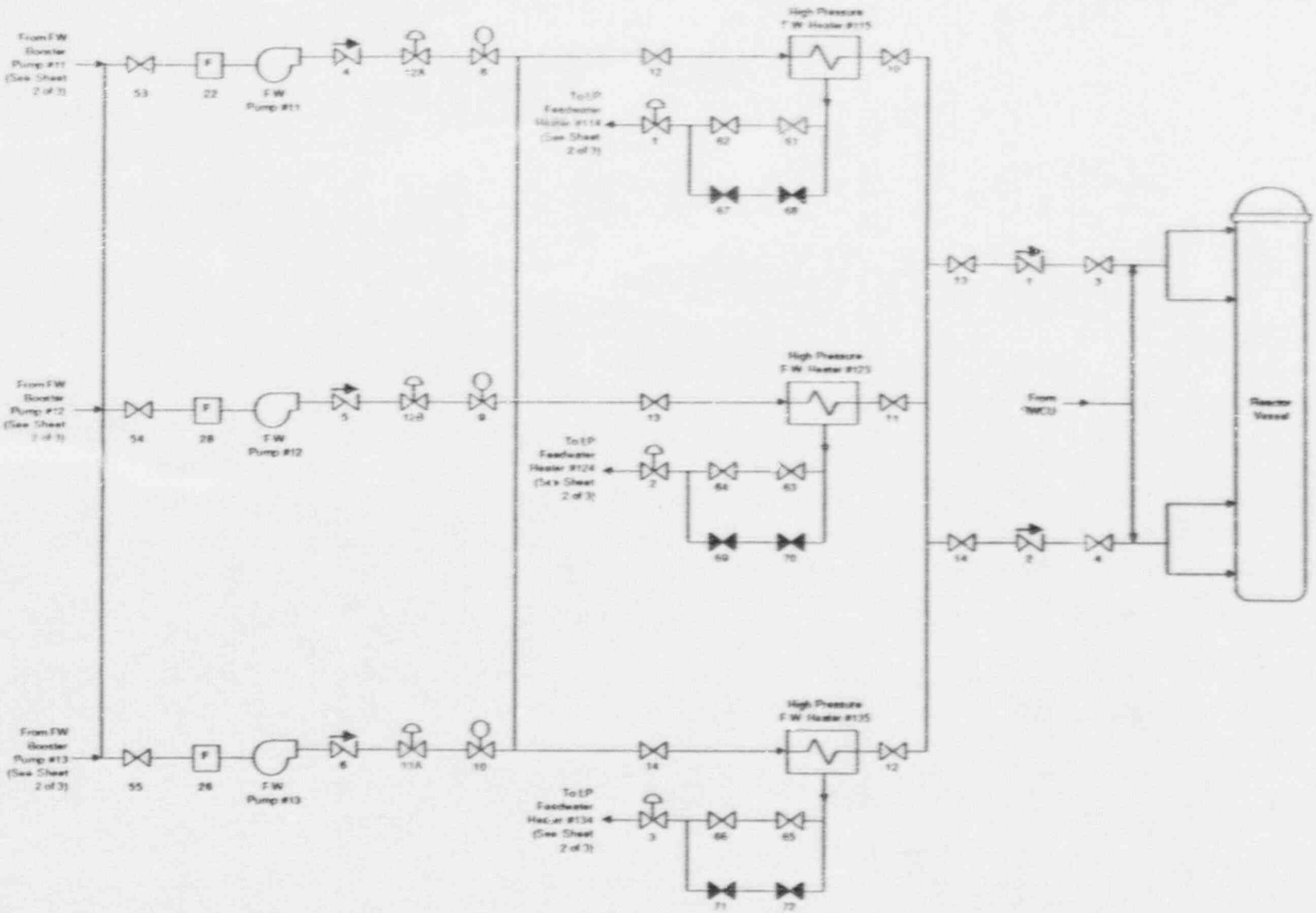


Figure 3.3-3. Nine Mile Point 1 Feedwater Coolant Injection System (Sheet 3 of 3)

Table 3.3-1. Nine Mile Point 1 Emergency Core Cooling System Data Summary for Selected Components

COMPONENT ID	COMP. TYPE	LOCATION	POWER SOURCE	VOLTAGE	POWER SOURCE LOCATION	EMERG. LOAD GRP.
CSSA-10	MOV	RC	EP-BS-171	600	261RB	AC/103
CSSA-11	MOV	RC	EP-BS-161	600	261RB	AC/102
CSSA-12	MOV	237RB	EP-BS-167	600	PB167	AC/102
CSSA-21	MOV	SWRB	EP-BS-161	600	261RB	AC/102
CSSA-22	MOV	SWRB	EP-BS-171	600	261RB	AC/103
CSSA-B-TORUS	TOR	TORUS				
CSSA-B-TORUS	TOR	TORUS				
CSSA-PM111	MDP	SWRB	EP-BS-102	4160	PB102	AC/102
CSSA-PM112	MDP	SWRB	EP-BS-103	4160	PB103	AC/103
CSSA-PMT111	MDP	237RB	EP-BS-102	4160	PB102	AC/102
CSSA-PMT112	MDP	237RB	EP-BS-103	4160	PB103	AC/103
CSSA-TV11	MOV	SWRB	EP-BS-167	600	PB167	AC/102
CSSB-1	MOV	RC	EP-BS-161	600	261RB	AC/102
CSSB-1S	MOV	SERB	EP-BS-161	600	261RB	AC/102
CSSB-2	MOV	237RB	EP-BS-167	600	PB167	AC/102
CSSB-2S	MOV	SERB	EP-BS-171	600	261RB	AC/103
CSSB-9	MOV	RC	EP-BS-171	600	261RB	AC/103
CSSB-PM121	MDP	SERB	EP-BS-102	4160	PB102	AC/102
CSSB-PM122	MDP	SERB	EP-BS-103	4160	PB103	AC/103
CSSB-PMT121	MDP	237RB	EP-BS-102	4160	PB102	AC/102
CSSB-PMT122	MDP	237RB	EP-BS-103	4160	PE103	AC/103
CSSB-TV12	MOV	SERB	EP-BS-167	600	PB167	AC/102
EC11-5	NV	ECIV281				
EC11-7	MOV	ECIV298	EP-VB12	125	DCVB12	DC/12
EC11-9	MOV	ECIV298	EP-BS-161	600	261RB	AC/102
EC11-HX	HX	340RB				

Table 3.3-1. Nine Mile Point 1 Emergency Core Cooling System Data Summary for Selected Components (Continued)

COMPONENT ID	COMP. TYPE	LOCATION	POWER SOURCE	VOLTAGE	POWER SOURCE LOCATION	EMERG. LOAD GRP.
EC11-MU	TANK	369TB				
EC12-10	MOV	ECIV298	EP-BS-171	600	261RB	AC/103
EC12-6	NV	ECIV281				
EC12-8	MOV	ECIV298	EP-VB12	125	DCVB12	DC/12
EC12-HX	HX	340RB				
EC12-MU	TANK	369TB				
RCS-RV	RV	RC				

3.4 CONTAINMENT SPRAY SYSTEM (CNS) AND CONTAINMENT SPRAY RAW WATER SYSTEM (CSRWS)

3.4.1 System Function

The CNS, in conjunction with the Containment Spray Raw Water System, is designed to reduce containment temperature and pressure by delivering water from the suppression pool, through heat exchangers, to spray headers in the drywell and torus. The spray water condenses steam and water returns by gravity to the suppression chamber for recirculation. The CNS and the CSRWS establish a heat transfer path between the suppression pool and the ultimate heat sink. The CSRWS also is capable of providing low pressure makeup to the RCS in an emergency.

3.4.2 System Definition

The CNS is composed of four separate and independent containment spray subsystems, each capable of removing all the decay heat at a rate which will prevent containment pressures and temperatures from exceeding their design values. Each of the four independent spray systems is provided with a pump, strainer, heat exchanger and associated piping. Moreover, each heat exchanger is independently supplied with raw cooling water from the Containment Spray Raw Water System (CSRWS).

The CSRWS is a composite of four independent supply systems each consisting of a pump, strainer, piping and valves. In addition, water from the CSRWS can also be diverted to the Core Spray System.

Simplified drawings of the containment spray system are shown in Figures 3.4-1 and 3.4-2, and the CSRWS is shown in Figure 3.4-3 and 3.4-4. A summary of data on selected CNS and CSRWS components is presented in Table 3.4-1

3.4.3 System Operation

During normal operation, the CNS is in standby. However, following a LOCA (or combination of low-low pressure vessel water level and drywell pressure ≥ 3.5 psig), an actuating signal is sent to the four containment spray pumps which sequentially start them. Water is pumped from the suppression pool header, through heat exchangers and then back to the containment through spray headers.

The CSRWS pumps raw water from the condenser circulating water intake tunnel to the CNS heat exchangers. In each CSRWS loop, the water passes through a strainer to the containment spray heat exchangers which cool the spray water supply from 140°F to 100°F, while the raw water side has a temperature rise of 40°F.

In the event of a total loss of the containment spray primary water source (a loss of suppression chamber water below the containment spray pump suction level), raw lake water can be delivered to the containment spray nozzles to provide an alternate source of containment cooling.

3.4.4 System Success Criteria

The CNS pumps and heat exchangers are 100% capacity components. Therefore, system success is achieved when at least one CNS pump takes suction from the suppression pool and delivers water to the CNS spray headers though its associated CNS heat exchanger. The corresponding CSRWS loop must also operate successfully in order to complete the heat transfer path to the ultimate heat sink.

3.4.5 Component Information

A. Containment Spray Pumps (4)
   1. Rated Flow : 3000 gpm @ 165 psig
   2. Type : centrifugal
   3. Rated Capacity : 100%

B. Containment Spray Heat Exchangers (4)
   1. Rated Duty : 60×10^6 Btu/hr
   2. Type : unknown

C. CSRWS Pumps (4)
   1. Rated Flow : 3000 gpm
   2. Type : centrifugal
   3. Rated Capacity : 100%

3.4.6 Support Systems and Interfaces

A. Control Signals
   1. Automatic
      The CNS is actuated automatically on coincidence of high drywell pressure and low reactor water level.
   2. Remote Manual
      The CNS pumps can be actuated by remote manual means from the control room.

B. Motive Power
   The CNS and CSRWS pumps are Class 1E AC loads that can be supplied from the emergency diesel generators as described in Section 3.6

3.4.7 Section 3.4 References

1. Final Safety Analysis Report, Nine Mile Point 1 Nuclear Power Station, Sections VI, VII, and X.

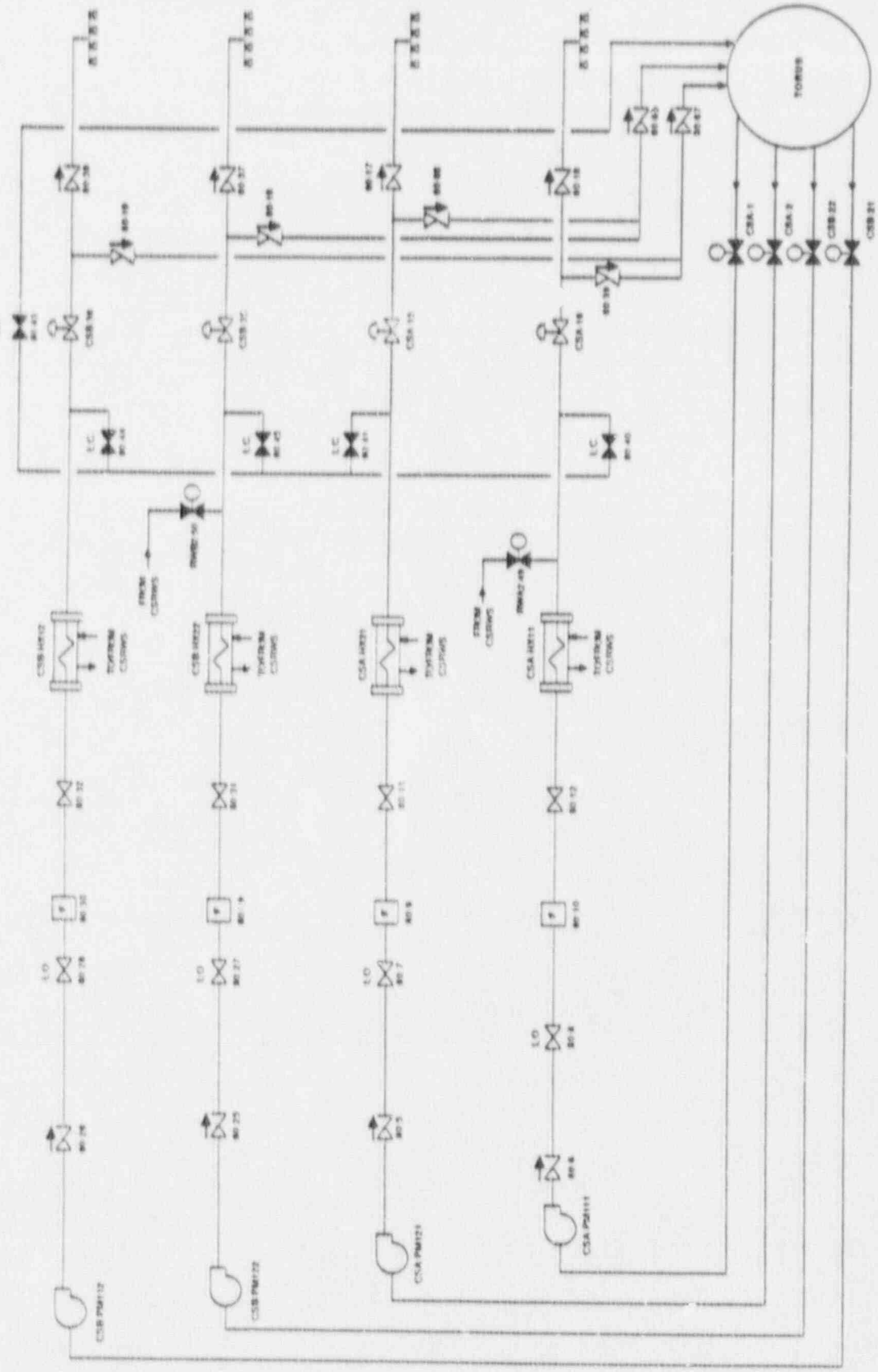
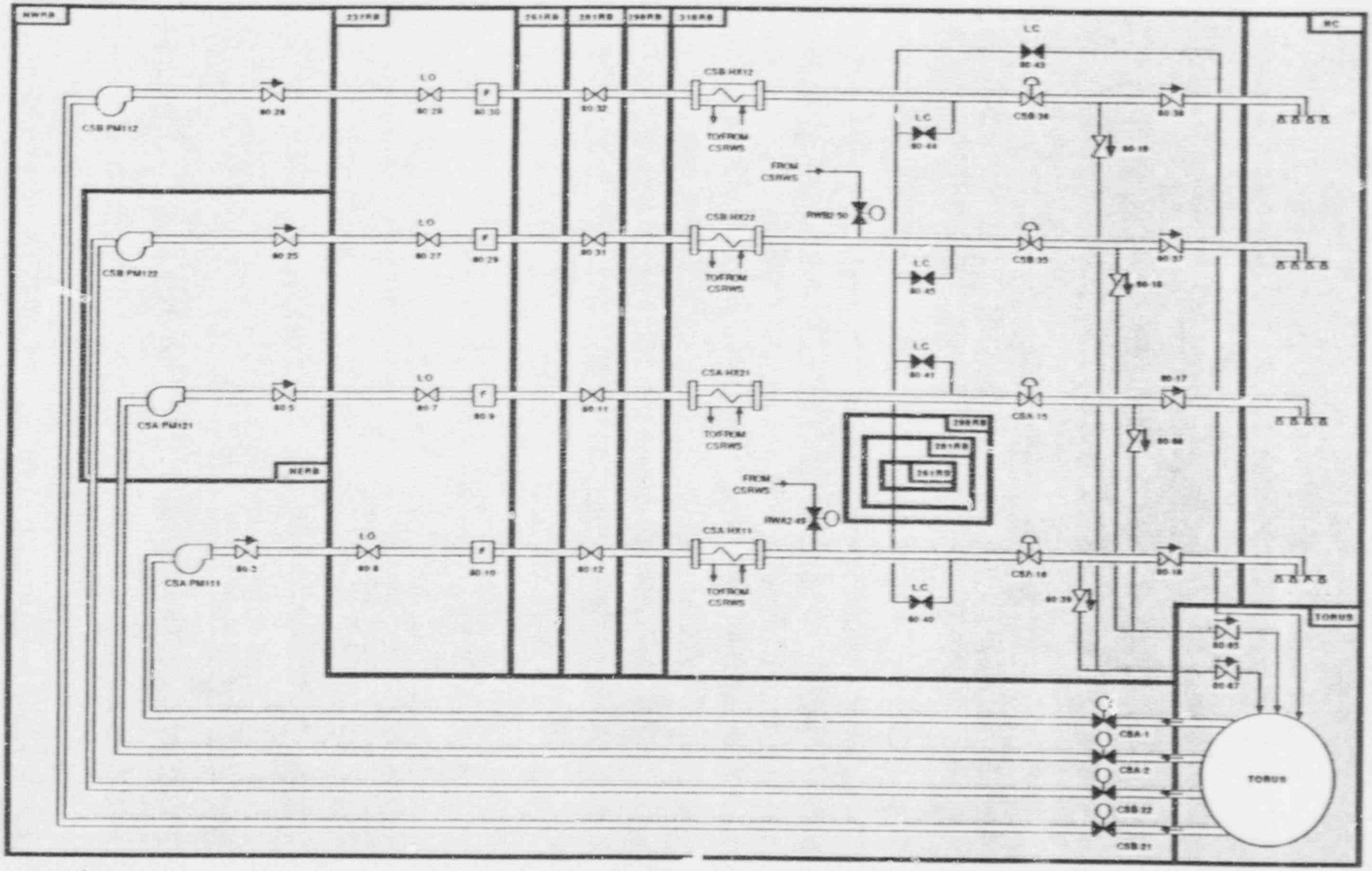


Figure 3.4-1. Nine Mile Point 1 Containment Spray System



NOTE: LOCATION OF PUMP DISCHARGE VALVES IS UNCERTAIN

Figure 3.4-2. Nine Mile Point 1 Containment Spray System Showing Component Locations

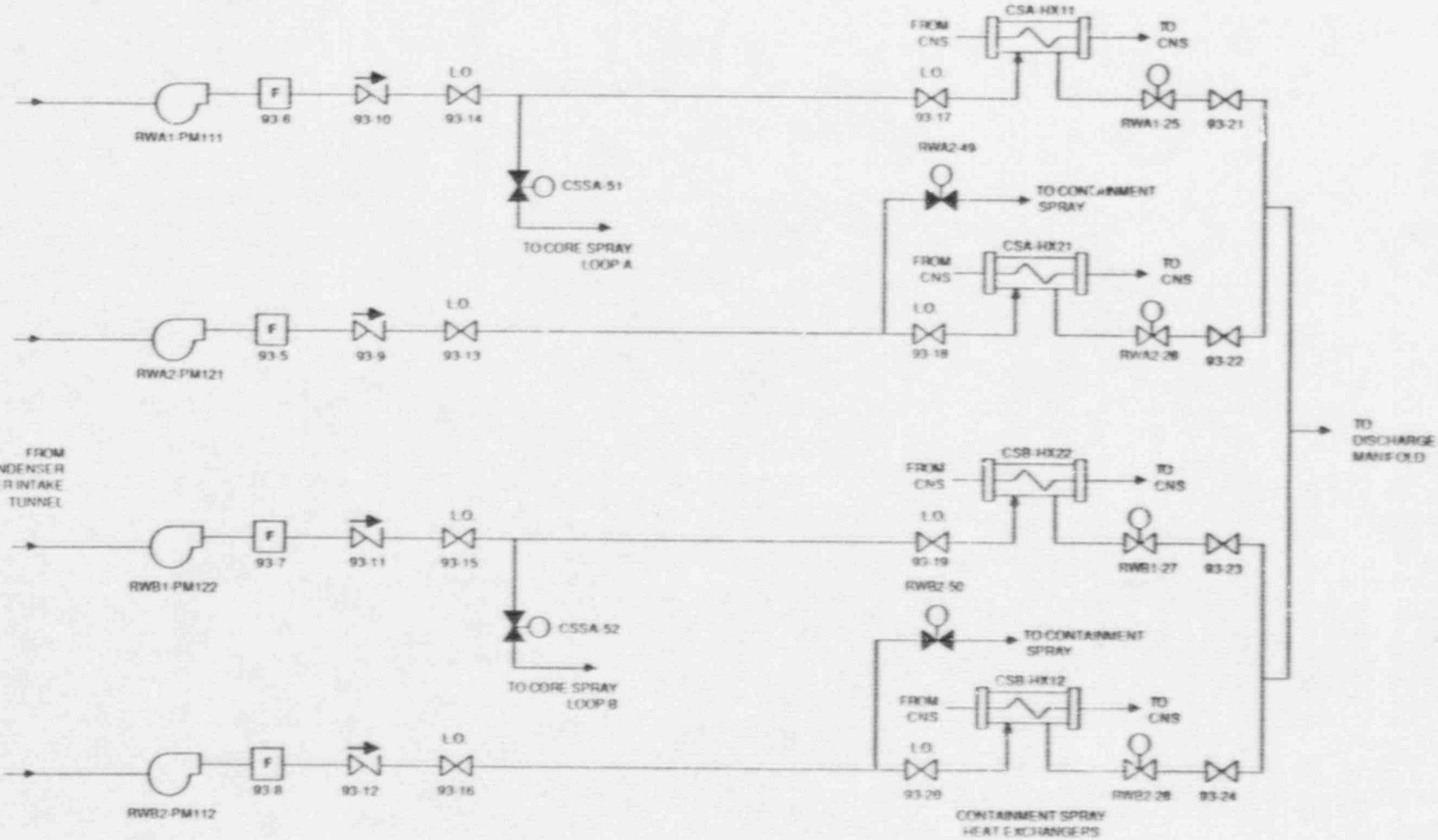


Figure 3.4-3. Nine Mile Point 1 Containment Spray Raw Water System

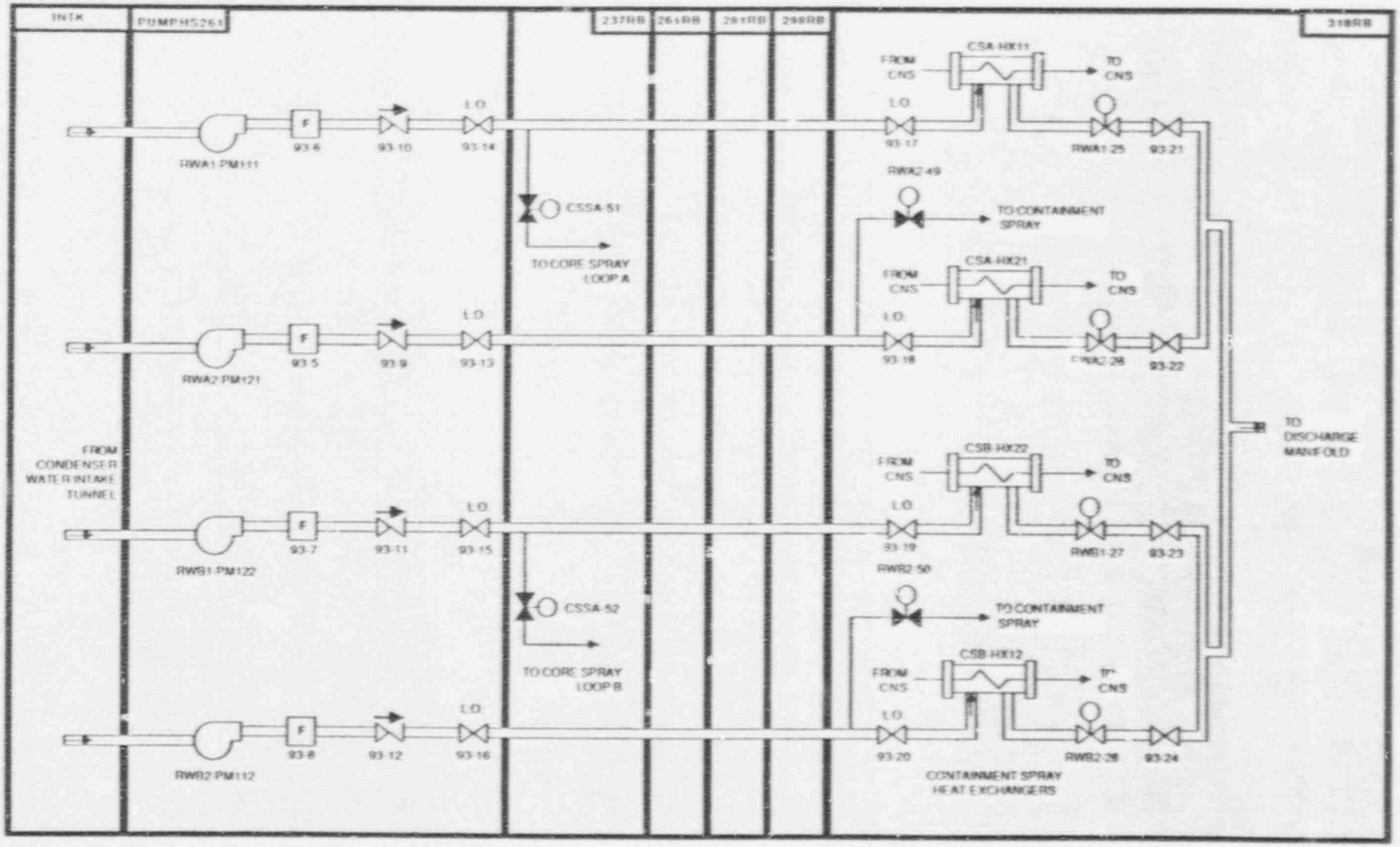


Figure 3.4-4. Nine Mile Point 1 Containment Spray Raw Water System Showing Component Locations

Table 3.4-1. Nine Mile Point 1 Containment Spray System Data Summary for Selected Components

COMPONENT ID	COMP. TYPE	LOCATION	POWER SOURCE	VOLTAGE	POWER SOURCE LOCATION	EMERG. LOAD GRP.
CSA-1	MOV	NWRB	EP-BS-161	600	261RB	AC/102
CSA-15	NV	318RB				
CSA-16	NV	318RB				
CSA-2	MOV	NWRB	EP-BS-171	600	261RB	AC/103
CSA-PM111	MDP	NWRB	EP-BS-102	4160	PB102	AC/102
CSA-PM121	MDP	NERB	EP-BS-103	4160	PB103	AC/103
CSB-21	MOV	NWRB	EP-BS-161	600	261RB	AC/102
CSB-22	MOV	NWRB	EP-BS-171	600	261RB	AC/103
CSB-35	NV	318RB				
CSB-36	NV	318RB				
CSB-PM112	MDP	NWRB	EP-BS-102	4160	PB102	AC/102
CSB-PM122	MDP	NERB	EP-BS-103	4160	PB103	AC/103
RWA-25	MOV	318RB	EP-BS-161	600	261RB	AC/102
RWA-26	MOV	318RB	EP-BS-171	600	261RB	AC/103
RWA-PM111	MDP	PUMPHS261	EP-BS-102	4160	PB102	AC/102
RWA-PM121	MDP	PUMPHS261	EP-BS-103	4160	PB103	AC/103
RWB-27	MOV	[illegible]8RB	EP-BS-171	600	261RB	AC/103
RWB-28	MOV	[illegible]B	EP-BS-161	600	261RB	AC/102
RWB-PM112	MDP	PUMPHS261	EP-BS-102	4160	PB102	AC/102
RWB-PM122	MDP	PUMPHS261	EP-BS-103	4160	PB103	AC/103

3.5 INSTRUMENTATION AND CONTROL (I&C) SYSTEMS

3.5.1 System Function

The instrumentation and control systems consist of the Reactor Protection System (RPS), Engineered Safety Features (ESF), and systems for the display of plant information to the operators. The RPS monitors the reactor plant, and alerts the operator to take corrective action before specified limits are exceeded. The RPS will initiate an automatic reactor trip (scram) to rapidly shutdown the reactor when plant conditions exceed one or more specified limits. The ESF systems will automatically actuate various safety systems based on the specific limits or combinations of limits that are exceeded. Equipment is provided outside the main control room for remote shutdown in the event that the control room must be evacuated.

3.5.2 System Definition

The RPS includes sensor and transmitter units, logic units, and output trip relays that interface with the control circuits for components in the scram portion of the Control Rod Drive Hydraulic System (see Section 3.7). Other actuation and control systems include independent sensor and transmitter units and relay units that interface with the control circuits of many different components in safety systems. Operator instrumentation display systems consist of display panels that are powered by 120 VAC power (see Section 3.6).

3.5.3 System Operation

A. RPS

The RPS consists of two independent logic channels (11 and 12), each of which consists of two identical subchannels of tripping devices. One channel in each of the two trip systems must trip to cause a full scram (one-out-of-two twice logic). The pilot scram valves are solenoid operated and normally energized. The solenoids de-energize to cause a scram. The system is therefore considered "fail safe" on loss of power. RPS inputs are listed below:

- High neutron flux
- High reactor pressure
- Low water level in reactor vessel
- High Containment [illegible]
- Closure of main steam [illegible] on valves
- Low vacuum in condenser
- Turbine trip
- High level in scram discharge volume
- High radiation in main steam lines
- Manual

B. Anticipated Transient Without Scram (ATWS) Mitigation System

ATWS event mitigation is achieved by means of a recirculation pump trip system that initiates a trip of all recirculation pumps if any of the following conditions occur (Ref. 1):

- high reactor vessel dome pressure (1135 psig) with a 3-second delay
- low-low reactor vessel water level with a 9-second delay

When actuated, the recirculation pumps are tripped, reducing flow through the core to a rate appropriate for natural circulation. This creates increased voiding in the core, thus reducing reactor power.

C. ESF
The ESF systems cause the various safety systems to be started, stopped or realigned as needed to respond to abnormal plant conditions. Details regarding actuation logic are included in the system description of the actuated system.

D. Remote Shutdown
The remote shutdown system is designed to provide the plant operators with hot shutdown capability independent of the main and auxiliary control rooms. It consists of two independent panels, and each are located in a separate fire area relative to each other and the main and auxiliary control rooms.

Each panel controls one set of emergency condenser return isolation, inlet isolation, and makeup level control valves. Both panels must be manned to achieve full scram, but once a full scram has been achieved, only one panel is required to bring the reactor safely to hot shutdown. The remote shutdown system was designed primarily in the event that the control room become uninhabitable (e.g., fire).

3.4 System Success Criteria

A. RPS
The RPS uses hindrance logic (normal = 1, trip = 0) in both the input and output logic. Therefore, a channel will be in a trip state when input signals are lost, when control power is lost, or when the channel is temporarily removed from service for testing or maintenance (i.e. the channel has a fail-safe failure mode). A reactor scram will occur upon loss of control power to the RPS. A reactor scram is implemented by the scram pilot valves in the control rod drive hydraulic system (see Section 3.6). Details of the RPS for Nine Mile Point 1 have not been determined.

B. ATWS Mitigation System
Success criteria for the ATWS mitigation system have not been determined.

C. ESF Actuation Systems
A single component usually receives a signal from only one actuation system output train. Trains A and B must be available in order to automatically actuate their respective components. Actuation systems other than the RPS typically use hindrance input logic (normal = 1, trip = 0) and transmission output logic (normal = 0, trip = 1). In this case, an input channel will be in a trip state when input signals are lost, when control power is lost, or when the channel is temporarily removed from service for testing or maintenance (i.e. the channel has a fail-safe failure mode). Control power is needed for the actuation system output channels to send an actuation signal. Note that there may be some actuation subsystems that utilize hindrance output logic. For these subsystems, loss of control power will cause system or component actuation, as is the case with the RPS. Details of the other actuation systems for Nine Mile Point 1 have not been determined.

D. Manually-Initiated Protective Actions
When reasonable time is available, certain protective actions may be performed manually by plant personnel. The control room operators are capable of operating individual components using normal control circuitry, or operating groups of components by manually tripping the RPS or an actuation subsystem.

The control room operators also may send qualified persons into the plant to operate components locally or from some other remote control location (e.g., a motor control center). To make these judgements, data on key plant parameters must be available to the operators.

3.5.5 Support Systems and Interfaces

A. Control Power

1. RPS
   The RPS is powered from the 120 VAC Class 1E electric power system.
2. ESF system
   The control power interfaces for the ESF actuation systems at Nine Mile Point 1 have not been identified.
3. Operator instrumentation
   Operator instrumentation display systems are powered from the 120 VAC Class 1E electric power system.

3.5.5 Support Systems and Interfaces

1. Final Safety Analysis Report, Nine Mile Point 1 Nuclear Power Station, Section VIII.A.2.2

3.6 ELECTRIC POWER SYSTEM

3.6.1 System Function

The electric power system supplies power to various equipment and systems needed for normal operation and/or response to accidents. The onsite Class 1E electric power system supports the operation of safety class systems and instrumentation needed to establish and maintain a safe shutdown plant condition following an accident, when the normal electric power sources are not available.

3.6.2 System Definition

The onsite Class 1E AC electric power system consists of two diesel generators which provide emergency power directly to two 4160 VAC power boards, PB102 and PB103. PB102 and PB103, in turn, distribute power to five 600 VAC power boards through step-down transformers. These standby generators are completely separate and self-contained within the station, and each has adequate capacity to start and carry all of the loads required during a maximum emergency-power-requirement periods.

The Class 1E DC system consists of two separate and independent station batteries each connected to a 125 VDC battery board. The batteries have a nominal rating at 125-volt, 1500 - ampere-hour. Each battery has its own charging equipment in the form of a three unit motor generator set.

Simplified diagrams of the AC electric power distribution system at Nine Mile Point 1 are shown in Figures 3.6-1 to 3.6-4. The DC distribution system is shown in Figure 3.6-5 and 3.6-6. A summary of data on selected electric power system components is presented in Table 3.6-1. A partial listing of electrical sources and loads is presented in Table 3.6-2.

3.6.3 System Operation

During normal operating conditions, the Class 1E AC power system is supplied from the 115 kV offsite power system via the 115 kV reserve bus (which is fed by two 115 kV transmission lines from remote generating systems) and Service Transformers 101S and 101N. These transformers, in turn, energize the 4160 VAC Power Boards 101, 102, and 103, as shown in Figure 3.5-1. Thus, the loss of offsite power would result in Power Boards 101, 102, and 103 becoming de-energized.

Power Boards 102 and 103 have as alternate supply sources diesel-generators 102 and 103 which start automatically upon low or degraded voltage at the buses to which they are connected. These diesel generators are completely separate and self-contained within the station. Power Boards 102 and 103 feed the post-accident cooling pumps and, through step-down transformers, 600 VAC Power Boards 16 and 17. Power Boards 16 and 17 feed the low-voltage auxiliaries that are required for station safety and are vital to safe shutdown under accident conditions.

Two separate and independent station batteries are provided for the Class 1E DC power system, with each battery feeding into its own battery board. Each DC power division provides startup and control power for its division dedicated diesel generator and control power for the breakers within its associated AC division distribution system. Each battery has its own charging equipment in the form of a three-unit motor generator set: an AC motor driving the DC battery charger and an AC generator for emergency lighting. Continuous power supply to the computer is also provided by using a third three-unit motor generator set. In an emergency, the chargers will be supplied from diesel-generators. With no AC power, the batteries will meet all reactor safety load requirements for an unknown period of time.

Redundant safeguards equipment such as motor driven pumps and motor operator valves are supplied by different power boards. For the purpose of discussion, this equipment has been grouped into "load groups." Load group AC/102 contains components powered either directly or indirectly from diesel-generator 102 through 4160

VAC Power Board 102. Load group AC/103 contains components powered either directly or indirectly from diesel-generator 103 through 4160 VAC Power Board 103. Components receiving DC power are assigned to load groups DC/11 or DC/12, based on the battery power source. Selected loads and components supplied by the Nine Mile Point 1 Class 1E electric power system are listed in Table 3.6-2.

3.6.4 System Success Criteria

Basic system success criteria for mitigating transients and loss-of-coolant accidents are defined by front-line systems, which then create demands on support systems. Electric power system success criteria are defined as follows, without taking credit for cross-ties that may exist between independent load groups:

- Each Class 1E DC load group is supplied initially from its respective battery (also needed for diesel starting)
- Each Class 1E AC load group is isolated from the non-Class 1E system and is supplied from its respective emergency power source (i.e. diesel generator)
- Power distribution paths to essential loads are intact
- Power to the battery chargers is restored before the batteries are exhausted

3.6.5 Component Information

A. Standby diesel generators 102, 103
   1. Continuous power rating: 2950 kW
   2. Rated voltage: 4160 VAC
   3. Manufacturer: General Motors

B. Station batteries 11, 12
   1. Type: lead-calcium
   2. Rated voltage: 125 VDC
   3. Design capacity: unknown

3.6.6 Support Systems and Interfaces

A. Control Signals
   1. Automatic
      The standby diesel generator are automatically started upon loss of voltage at their associated 4160 VAC power board.

   2. Remote manual
      A manual start control with dead line pick-up of the diesel generators is provided from the main control room.

B. Diesel Generator Auxiliary Systems
   1. Cooling
      Each diesel generator is cooled by a separate diesel cooling raw water system. Details on this system have not been determined, however, this system is separate from the service water system. The diesel-driven fire water pump can be used as a backup source of cooling water for either diesel generator (Ref. 1, Section X.K)
   2. Fueling
      The capacity of the diesel-generator day tanks is unknown. However, the long term fuel supply for these diesel-generators is contained in two 10,000-gallon storage tanks which are cross-connected (outside the diesel-generator building) so that either unit can take oil from either tank. These

tanks have the capacity to permit one diesel unit to operate for four days at full load.

3. Lubrication
   Each diesel generator has a self-contained lubrication system.
4. Starting
   Power to start each unit using air motors for cranking is provided by an air system consisting of five air tanks and two compressors. Each air system stores sufficient air for five starts. A separate system exists for each diesel generator.
5. Control power
   Each diesel generator is dependent on 125 VDC power from a station battery for control power.
6. Diesel room ventilation
   Diesel room ventilation fans provide room cooling during diesel operation.

C. Switchgear and Battery Room Ventilation
Ventilation capabilities for the essential switchgear rooms and battery rooms could not be determined

3.6.7 Section 3.6 References

1. Final Safety Analysis Report, Nine Mile Point 1 Nuclear Power Station.

2. NUREG/CR-2989, "Reliability of Emergency AC Power Systems at Nuclear Power Plants, " Oak Ridge National Laboratory, July 1983.

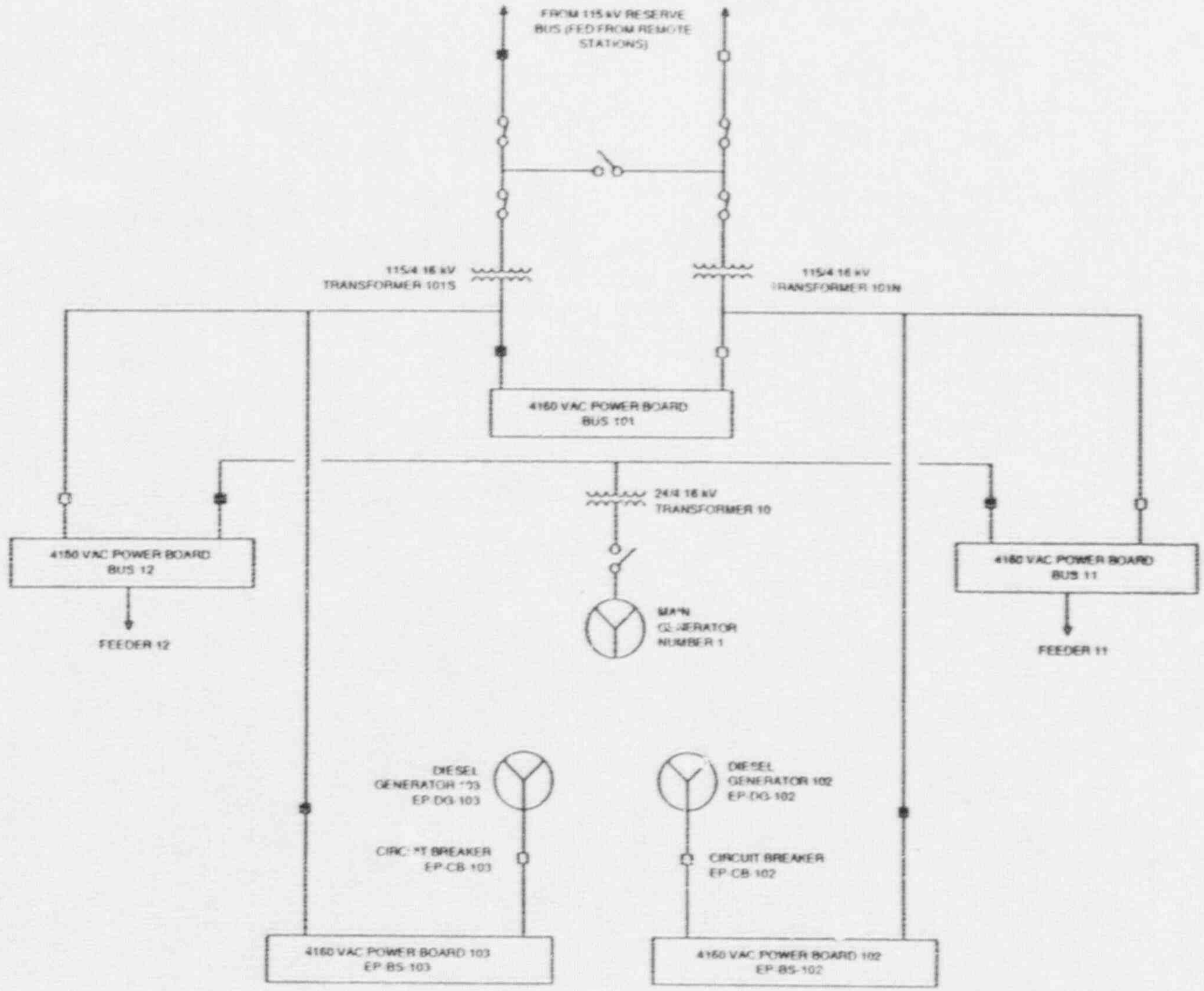
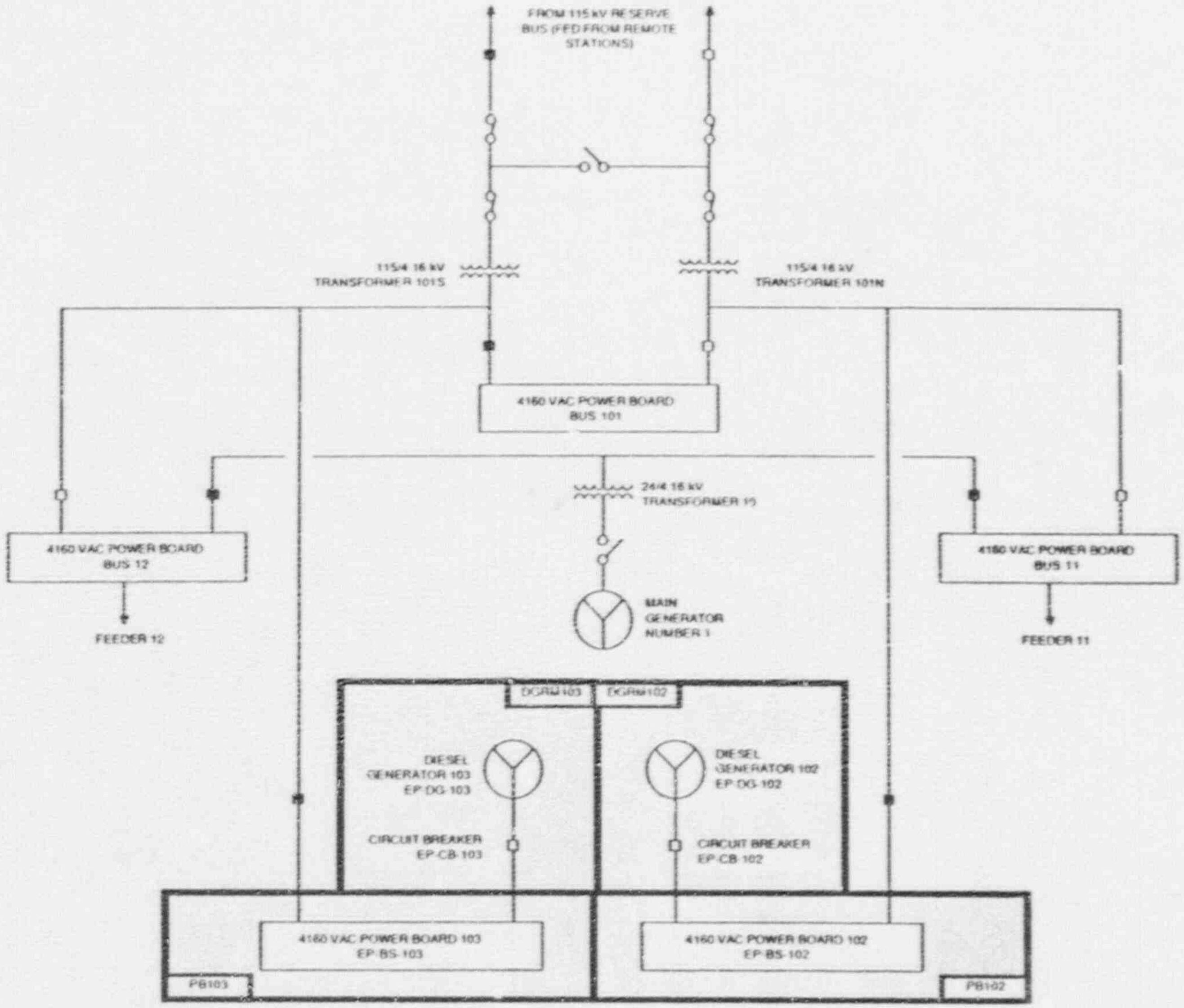


Figure 3.6-1. Nine Mile Point 1 4160 VAC Electric Power Distribution System



NOTE

LINES MAY NOT REPRESENT TRUE CABLE ROUTING BETWEEN ROOMS

Figure 3.6-2. Nine Mile Point 1 4160 VAC Electric Power Distribution System Showing Component Locations

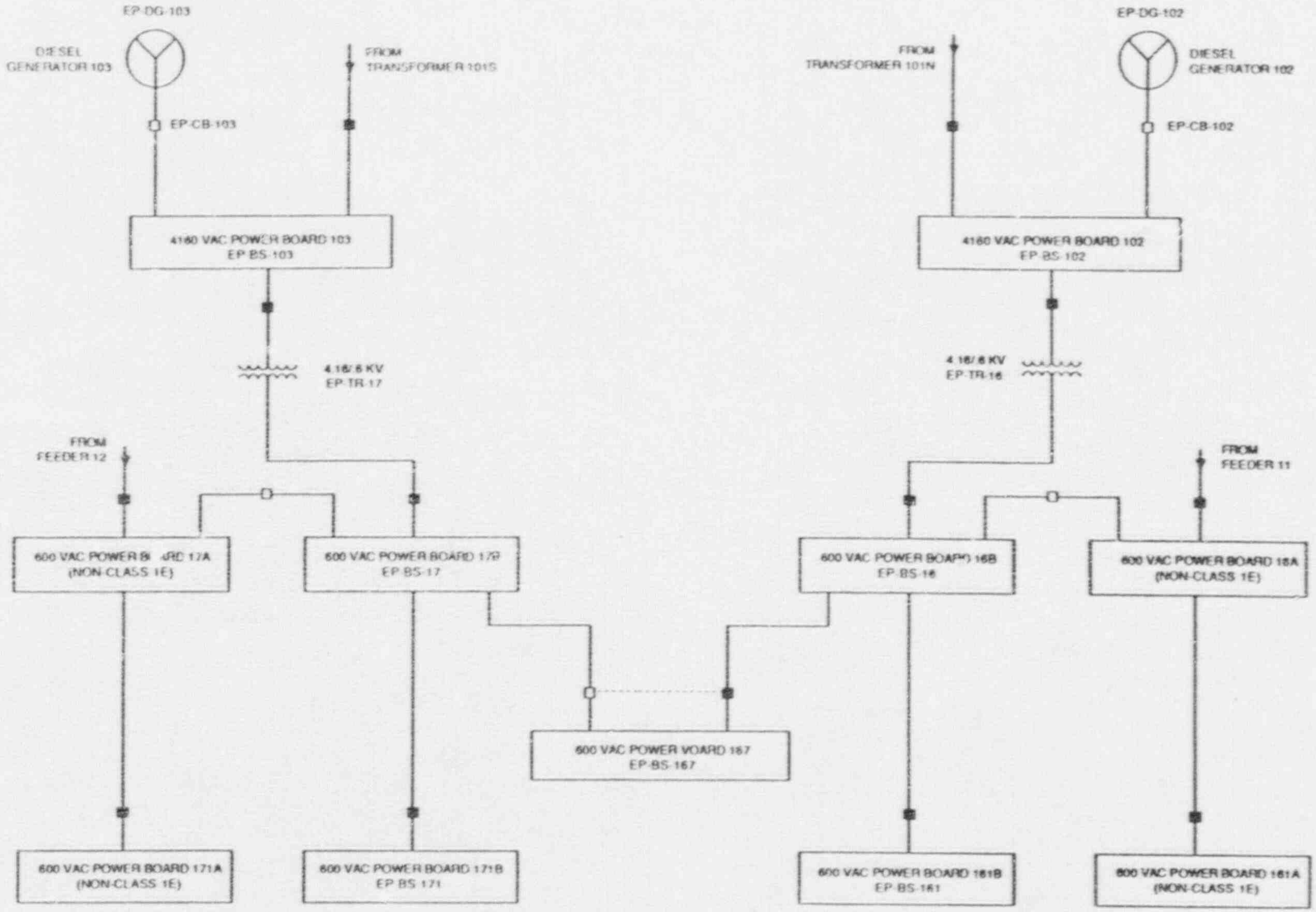
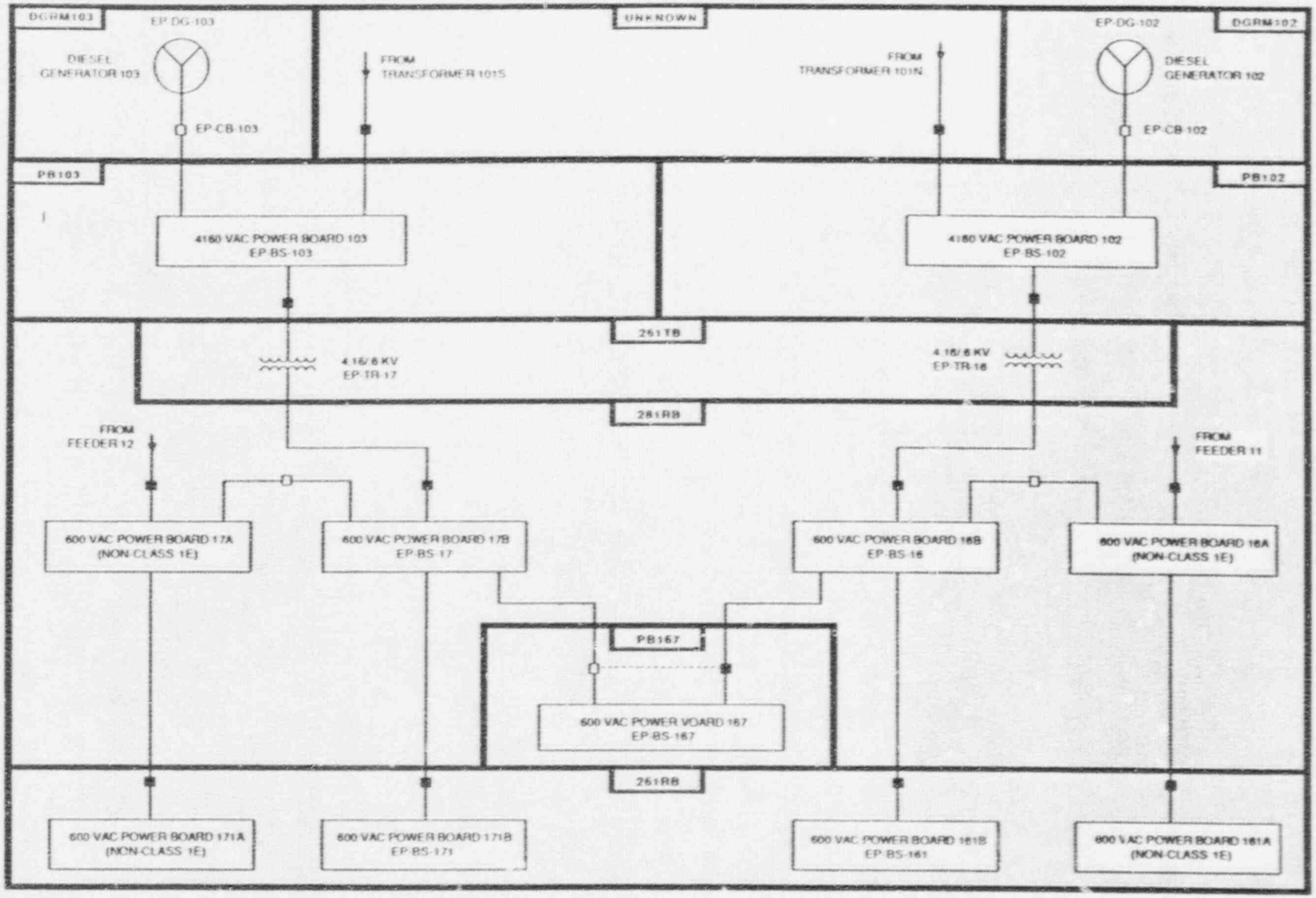


Figure 3.6-3. Nine Mile Point 1 600 VAC Electric Power Distribution System



NOTE
LINES MAY NOT REPRESENT TRUE CABLE ROUTING BETWEEN ROOMS

Figure 3.6-4. Nine Mile Point 1 600 VAC Electric Power Distribution System Showing Component Locations

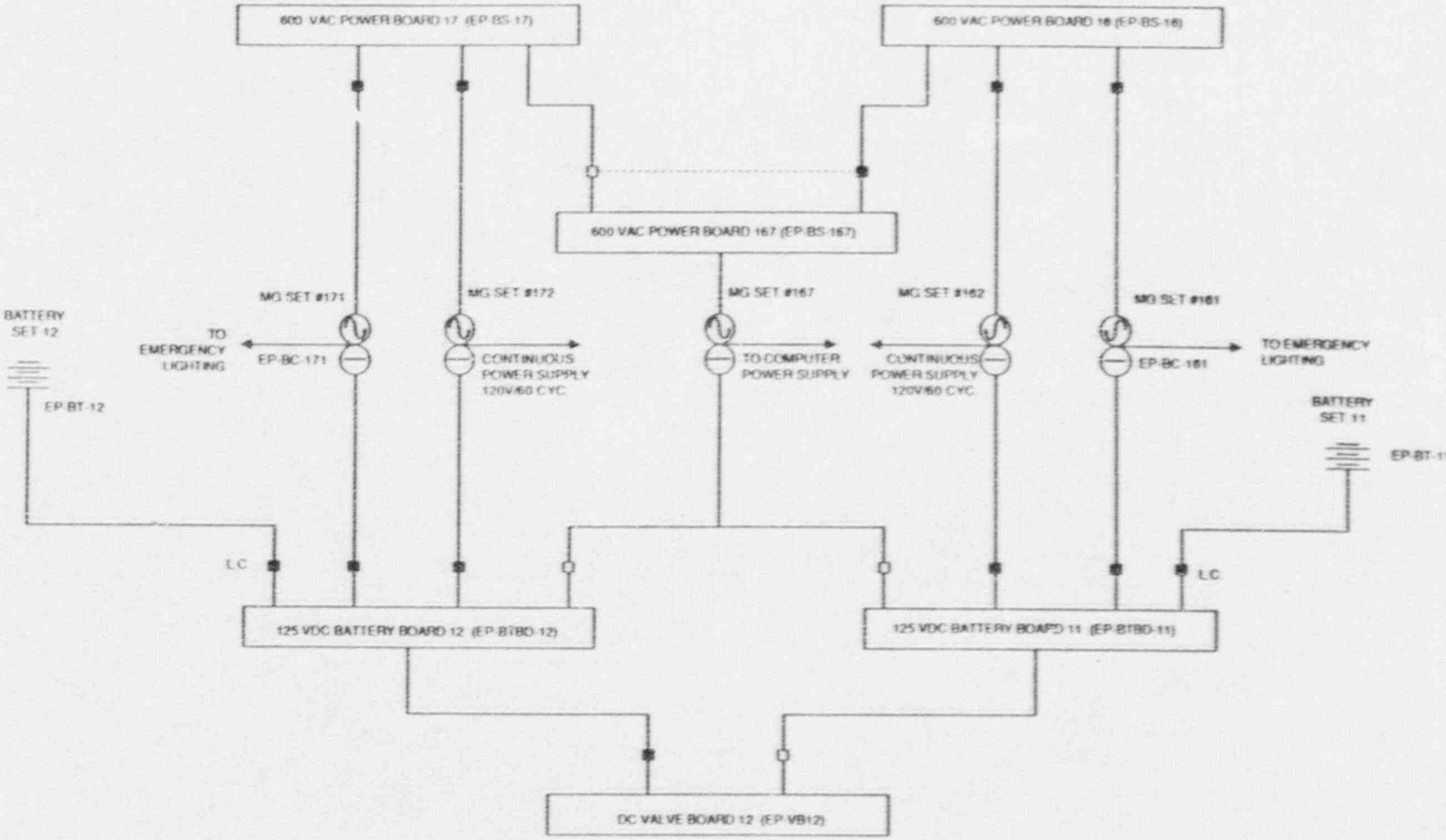
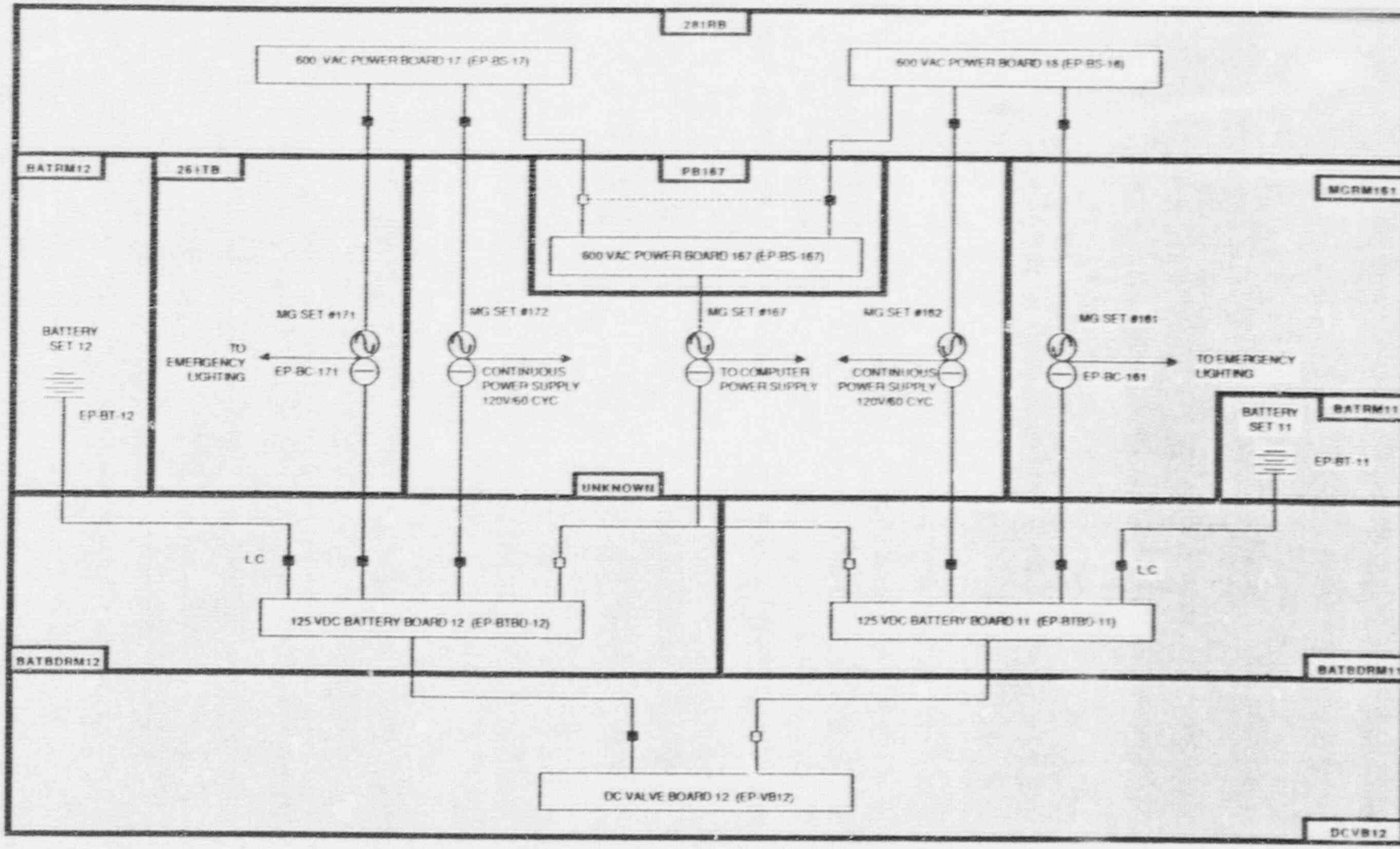


Figure 3.6-5. Nine Mile Point 1 125 VDC Electric Power Distribution System



NOTE

LINES MAY NOT REPRESENT TRUE CABLE ROUTING BETWEEN ROOMS

Figure 3.6-6. Nine Mile Point 1 125 VDC Electric Power Distribution System Showing Component Locations

Table 3.6-1. Nine Mile Point 1 Electric Power System Data Summary for Selected Components

COMPONENT ID	COMP. TYPE	LOCATION	POWER SOURCE	VOLTAGE	POWER SOURCE LOCATION	EMERG. LOAD GRP.
EP-BC-161	BC	MGRM161	EP-BS-16	600	281RB	AC/102
EP-BC-171	BC	261TB	EP-BS-17	600	281RB	AC/103
EP-BS-102	BUS	PB102	EP-CB-102	4160	PB102	AC/102
EP-BS-103	BUS	PB103	EP-CB-103	4160	PB103	AC/103
EP-BS-16	BUS	281RB	EP-TR-16	600	261TB	AC/102
EP-BS-161	BUS	261RB	EP-BS-16	600	281RB	AC/102
EP-BS-167	BUS	PB167	EP-BS-17	600	281RB	AC/103
EP-BS-167	BUS	PB167	EP-BS-16	600	281RB	AC/103
EP-BS-17	BUS	281RB	EP-TR-17	600	261TB	AC/103
EP-BS-171	BUS	261RB	EP-BS-17	600	281RB	AC/103
EP-BT-11	BATT	BATRM11		125		DC/11
EP-BT-12	BATT	BATRM12		125		DC/12
EP-CB-102	CB	PB102	EP-DG-102	4160	DGRM102	AC/102
EP-CB-103	CB	PB103	EP-DG-103	4160	DGRM103	AC/103
EP-DG-102	DG	DGRM102		4160		AC/102
EP-DG-103	DG	DGRM103		4160		AC/103
EP-TR-16	TRAN	261TB	EP-BS-102	600	PB102	AC/102
EP-TR-17	TRAN	261TB	EP-BS-103	600	PB103	AC/103
EP-VB12	BUS	DCVB12	EP-BTBD-12	125	BATBDRM12	DC/12

Table 3.6-2. Partial Listing of Electrical Sources and Loads at Nine Mile Point 1

POWER SOURCE	VOLTAGE	EMERG LOAD GRP	POWER SOURCE LOCATION	LOAD SYSTEM	LOAD COMPONENT ID	COMP TYPE	COMPONENT LOCATION
EP-BC-161	600	AC/102	MGRM161	EP	EP-BTBD-11	BUS	BATBDRM11
EP-BC-161	600	AC/102	MGRM161	EP	EP-BTBD-11	BUS	BATBDRM11
EP-BC-171	600	AC/103	261TB	EP	EP-BTBD-12	BUS	BATBDRM12
EP-BS-102	4160	AC/102	PB102	CNS	CSA-PM111	MDP	NWRB
EP-BS-102	4160	AC/102	PB102	CNS	CSB-PM112	MDP	NWRB
EP-BS-102	4160	AC/102	PB102	CNS	RWA-PM111	MDP	PUMPHS261
EP-BS-102	4160	AC/102	PB102	CNS	RWB-PM112	MDP	PUMPHS261
EP-BS-102	4160	AC/102	PB102	ECCS	CSSA-PM111	MDP	SWRB
EP-BS-102	4160	AC/102	PB102	ECCS	CSSA-PMT111	MDP	237RB
EP-BS-102	4160	AC/102	PB102	ECCS	CSSB-PM121	MDP	SERB
EP-BS-102	4160	AC/102	PB102	ECCS	CSSB-PMT121	MDP	237RB
EP-BS-102	600	AC/102	PB102	EP	EP-TR-16	TRAN	261TB
EP-BS-103	4160	AC/103	PB103	CNS	CSA-PM121	MDP	NERB
EP-BS-103	4160	AC/103	PB103	CNS	CSB-PM122	MDP	NERB
EP-BS-103	4160	AC/103	PB103	CNS	RWA-PM121	MDP	PUMPHS261
EP-BS-103	4160	AC/103	PB103	CNS	RWB-PM122	MDP	PUMPHS261
EP-BS-103	4160	AC/103	PB103	ECCS	CSSA-PM112	MDP	SWRB
EP-BS-103	4160	AC/103	PB103	ECCS	CSSA-PMT112	MDP	237RB
EP-BS-103	4160	AC/103	PB103	ECCS	CSSB-PM122	MDP	SERB
EP-BS-103	4160	AC/103	PB103	ECCS	CSSB-PMT122	MDP	237RB
EP-BS-103	600	AC/103	PB103	EP	EP-TR-17	TRAN	261TB
EP-BS-155	600	N/A	281RB	CRD	CRD-18	MOV	237RB
EP-BS-155	600	N/A	281RB	CRD	CRD-40	MOV	237RB
EP-BS-16	600	AC/102	281RB	CRD	CRD-PM11	MDP	237RB
EP-BS-16	600	AC/102	281RB	EP	EP-BC-161	BC	MGRM161
EP-BS-16	600	AC/102	281RB	EP	EP-BS-161	BUS	261RB
EP-BS-16	600	AC/103	281RB	EP	EP-BS-167	BUS	PB167
EP-BS-161	600	AC/102	261RB	CNS	CSA-1	MOV	NWRB
EP-BS-161	600	AC/102	261RB	CNS	CSB-21	MOV	NWRB
EP-BS-161	600	AC/102	PB161	CNS	CSSA-51	MOV	237RB
EP-BS-161	600	AC/102	261RB	CNS	RWA-25	MOV	318RB
EP-BS-161	600	AC/102	261RB	CNS	RWA2-49	MOV	318RB
EP-BS-161	600	AC/102	261RB	CNS	RWA2-49	MOV	318RB

Table 3.6-2. Partial Listing of Electrical Sources and Loads at Nine Mile Point 1 (Continued)

POWER SOURCE	VOLTAGE	EMERG LOAD GRP	POWER SOURCE LOCATION	LOAD SYSTEM	LOAD COMPONENT ID	COMP TYPE	COMPONENT LOCATION
EP-BS-161	600	AC/102	261RB	CNS	RWB-28	MOV	318RB
EP-BS-161	600	AC/102	261RB	ECCS	CSSA-11	MOV	RC
EP-BS-161	600	AC/102	261RB	ECCS	CSSA-21	MOV	SWRB
EP-BS-161	600	AC/102	PB161	ECCS	CSSA-51	MOV	237RB
EP-BS-161	600	AC/102	261RB	ECCS	CSSB-1	MOV	RC
EP-BS-161	600	AC/102	261RB	ECCS	CSSB-1S	MOV	SERB
EP-BS-161	600	AC/102	261RB	ECCS	EC11-9	MOV	ECIV298
EP-BS-161	600	AC/102	261RB	RCS	CSSA-11	MOV	RC
EP-BS-161	600	AC/102	261RB	RCS	CSSB-1	MOV	RC
EP-BS-161	600	AC/102	261RB	RCS	EC11-9	MOV	ECIV298
EP-BS-167	600	AC/102	PB167	ECCS	CSSA-12	MOV	237RB
EP-BS-167	600	AC/102	PB167	ECCS	CSSA-TV11	MOV	SWRB
EP-BS-167	600	AC/102	PB167	ECCS	CSSB-2	MOV	237RB
EP-BS-167	600	AC/102	PB167	ECCS	CSSB-TV12	MOV	SERB
EP-BS-167	600	AC/102	[illegible]	RCS	RWCU-2	MOV	RC
EP-BS-167	600	AC/102	PB167	RCS	SCIV-1	MOV	RC
EP-BS-167	600	AC/102	PB167	RCS	SCIV-13	MOV	RC
EP-BS-17	600	AC/103	281RB	CRD	CRD-PM12	MDP	237RB
EP-BS-17	600	AC/103	281RB	EP	EP-BC-171	BC	261TB
EP-BS-17	600	AC/103	281RB	EP	EP-BS-167	BUS	PB167
EP-BS-17	600	AC/103	281RB	EP	EP-BS-171	BUS	261RB
EP-BS-171	600	AC/103	261RB	CNS	CSA-2	MOV	NWRB
EP-BS-171	600	AC/103	261RB	CNS	CSB-22	MOV	NWRB
EP-BS-171	600	AC/103	PB171	CNS	CSSB-52	MOV	237RB
EP-BS-171	600	AC/103	261RB	CNS	RWA-26	MOV	318RB
EP-BS-171	600	AC/103	261RB	CNS	RWB-27	MOV	318RB
EP-BS-171	600	AC/103	261RB	CNS	RWB2-50	MOV	318RB
EP-BS-171	600	AC/103	261RB	CNS	RWB2-50	MOV	318RB
EP-BS-171	600	AC/103	261RB	ECCS	CSSA-10	MOV	RC
EP-BS-171	600	AC/103	261RB	ECCS	CSSA-22	MOV	SWRB
EP-BS-171	600	AC/103	261RB	ECCS	CSSB-2S	MOV	SERB
[illegible] BS-171	600	AC/103	261RB	ECCS	CSSB-52	MOV	237RB
EP-BS-171	600	AC/103	261RB	ECCS	CSSB-9	MOV	RC

Table 3.6-2. Partial Listing of Electrical Sources and Loads at Nine Mile Point 1 (Continued)

POWER SOURCE	VOLTAGE	EMERG LOAD GRP	POWER SOURCE LOCATION	LOAD SYSTEM	LOAD COMPONENT ID	COMP TYPE	COMPONENT LOCATION
EP-BS-171	600	AC/103	261RB	ECCS	EC12-10	MOV	ECIV298
EP-BS-171	600	AC/103	261RB	RCS	CSSA-10	MOV	RC
EP-BS-171	600	AC/103	261RB	RCS	CSSB-9	MOV	RC
EP-BS-171	600	AC/103	261RB	RCS	EC12-10	MOV	ECIV298
EP-BT-12	125	DC/12	BATRM12	EP	EP-BTBD-12	BUS	BATBDRM12
EP-BTBD-11	125	DC/11	BATBDRM11	ECCS	RCS-ADS	PORV	RC
EP-BTBD-11	125	DC/11	BATBDRM11	RCS	RCS-ADS	PORV	RC
EP-BTBD-12	125	DC/12	BATBDRM12	ECCS	RCS-ADS	PORV	RC
EP-BTBD-12	125	DC/12	BATBDRM12	EP	EP-VB12	BUS	DCVB12
EP-BTBD-12	125	DC/12	BATBDRM12	RCS	RCS-ADS	PORV	RC
EP-CB-102	4160	AC/102	PB102	EP	EP-BS-102	BUS	PB102
EP-CB-103	4160	AC/103	PB103	EP	EP-BS-103	BUS	PB103
EP-DG-102	4160	AC/102	DGRM102	EP	EP-CB-102	CB	PB102
EP-DG-103	4160	AC/103	DGRM103	EP	EP-CB-103	CB	PB103
EP-TR-16	600	AC/102	261TB	EP	EP-BS-16	BUS	261RB
EP-TR-17	600	AC/103	261TB	EP	EP-BS-17	BUS	261RB
EP-VB12	125	DC/12	DCVB12	ECCS	EC11-7	MOV	ECIV298
EP-VB12	125	DC/12	DCVB12	ECCS	EC12-8	MOV	ECIV298
EP-VB12	125	DC/12	DCVB12	RCS	EC11-7	MOV	ECIV298
EP-VB12	125	DC/12	DCVB12	RCS	EC12-8	MOV	ECIV298
EP-VB12	125	DC/12	DCVB12	RCS	SCIV-2	MOV	261RB

3.7 CONTROL ROD DRIVE HYDRAULIC SYSTEM (CRDHS)

3.7.1 System Function

The CRDHS supplies pressurized water to operate and cool the control rod drive mechanisms during normal operation. This system implements a scram command from the reactor protection system (RPS) and drives control rods rapidly into the reactor. The CRDHS also can provide makeup water to the RCS.

3.7.2 System Definition

The CRDHS consists of two high-head, low-flow pumps, piping, filters, control valves, one hydraulic control unit for each control rod drive mechanism, and instrumentation. Water is supplied from condensate and from the condensate storage tank. The CRDHS also includes scram valves, scram accumulators, and a scram discharge volume (dump tank).

Simplified drawing of the CRDHS are shown in Figures 3.7-1 and 3.7-2. Details of the scram portion of typical BWR CRDHS are shown in Figure 3.7-3 (adapted from Ref. 1). A summary of data on selected CRDHS components is presented in Table 3.7-1.

3.7.3 System Operation

During normal operation the CRDHS pumps provide a constant flow for drive mechanism cooling and system pressure stabilization. Excess water not used for cooling is discharged to the RCS. Control rods are driven in or out by the coordinated operation of the direction control valves. Insertion speed is controlled by flow through the insert speed control valve. Rod motion may be either stepped or continuous.

A reactor scram is implemented by pneumatic scram valves in the CRDHS. An inlet scram valve opens to align the insert side of each control rod drive mechanism (CRDM) to its scram accumulator. An outlet scram valve opens to vent the opposite side of each CRDM to the dump tank (or discharge volume). This coordinated action results in rapid insertion of control rods into the reactor.

Although not intended as a makeup system, the CRDHS can provide a source of cooling water to the RCS during vessel isolation. It is noted in NUREG-0626 (Ref. 2), that this function is particularly important for some BWR/1 and BWR/2 plants for which the CRDHS is the primary source of makeup on vessel isolation. In later model BWR plants, RCS makeup at high pressure is performed by other systems. For Nine Mile Point 1, each pump can deliver a nominal 50 gpm water makeup to the reactor vessel.

3.7.4 System Success Criteria

For the scram function to be accomplished, the following actions must occur in the CRDHS:

- A scram signal must be transmitted by the RPS to the actuated devices (i.e., pilot valves) in the CRDHS
- The pneumatic inlet scram valve and outlet scram valve must open in the hydraulic control units (HCUs) for the individual control rod drives. This is accomplished by venting the instrument air supply to each valve as follows:
  - Both scram pilot valves in each HCU must be deenergized, or
  - Either backup scram pilot valve must be energized.
  - A high-pressure water source must be available from the scram accumulator in each HCU.
  - A hydraulic vent path to the scram discharge volume must be available and sufficient collection volume must exist in the scram discharge volume.
  - A specified number of control rods must respond and insert into the reactor core (specific number needed is not known).

3.7.5 Component Information

A. Control rod drive pumps (11, 12)
   1. Rated capacity: 100% (for control rod drive function)
   2. Flow rate: 85 gpm
   3. Type: centrifugal

B. Condensate Storage Tank
   1. Minimum Volume: unknown

C. Scram Accumulator
   1. Normal pressure: 1500 psig

D. Scram Disc...ge Volume
   1. Normal pressure: Atmospheric

3.7.6 Support Systems and Interfaces

A. Control Signals
   1. Automatic
      The RPS transmits scram commands to solenoid pilot valves which control the pneumatic scram valves.
   2. Remote Manual
      a. A reactor scram can be initiated manually from the control room, the auxiliary control room, and from the remote shutdown panels.
      b. The CRDHS can be operated manually from the control room to insert and withdraw rods, or to inject water into the RCS.

B. Motive Power
   The CRDHS pumps are Class 1E AC loads that can be powered from the diesel generators as described in Section 3.6.

3.7.7 Section 3.8 References

1. NEDO-24708A, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," General Electric Company, December 1980.

2. NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant-Accidents in GE-designed Operating Plants and Near-term Operating License Applications," USNRC, January 1980.

3. Final Safety Analysis Report, Nine Mile Point 1 Nuclear Power Station, Section X.C.

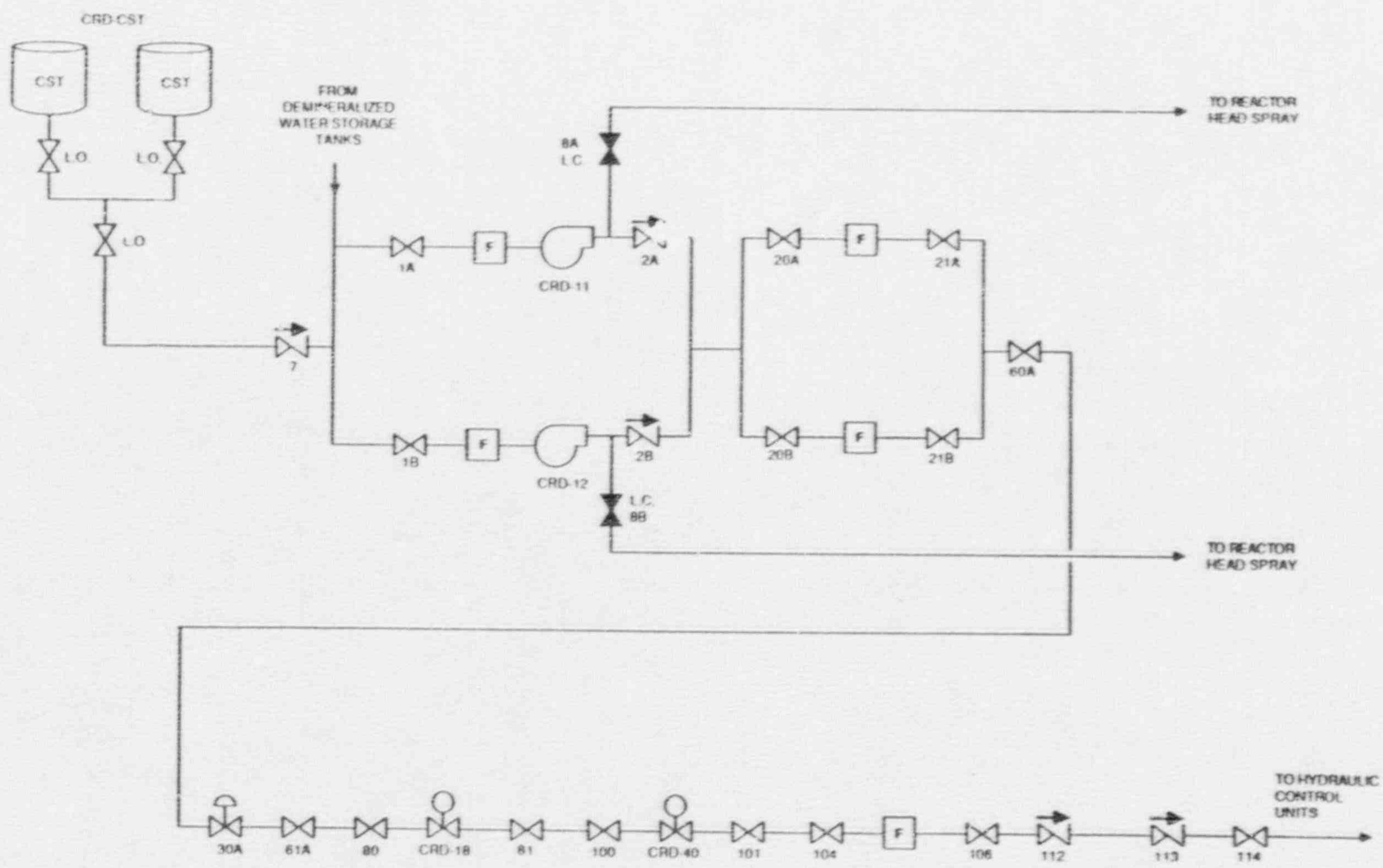


Figure 3.7-1. Nine Mile Point 1 Control Rod Drive Hydraulic System

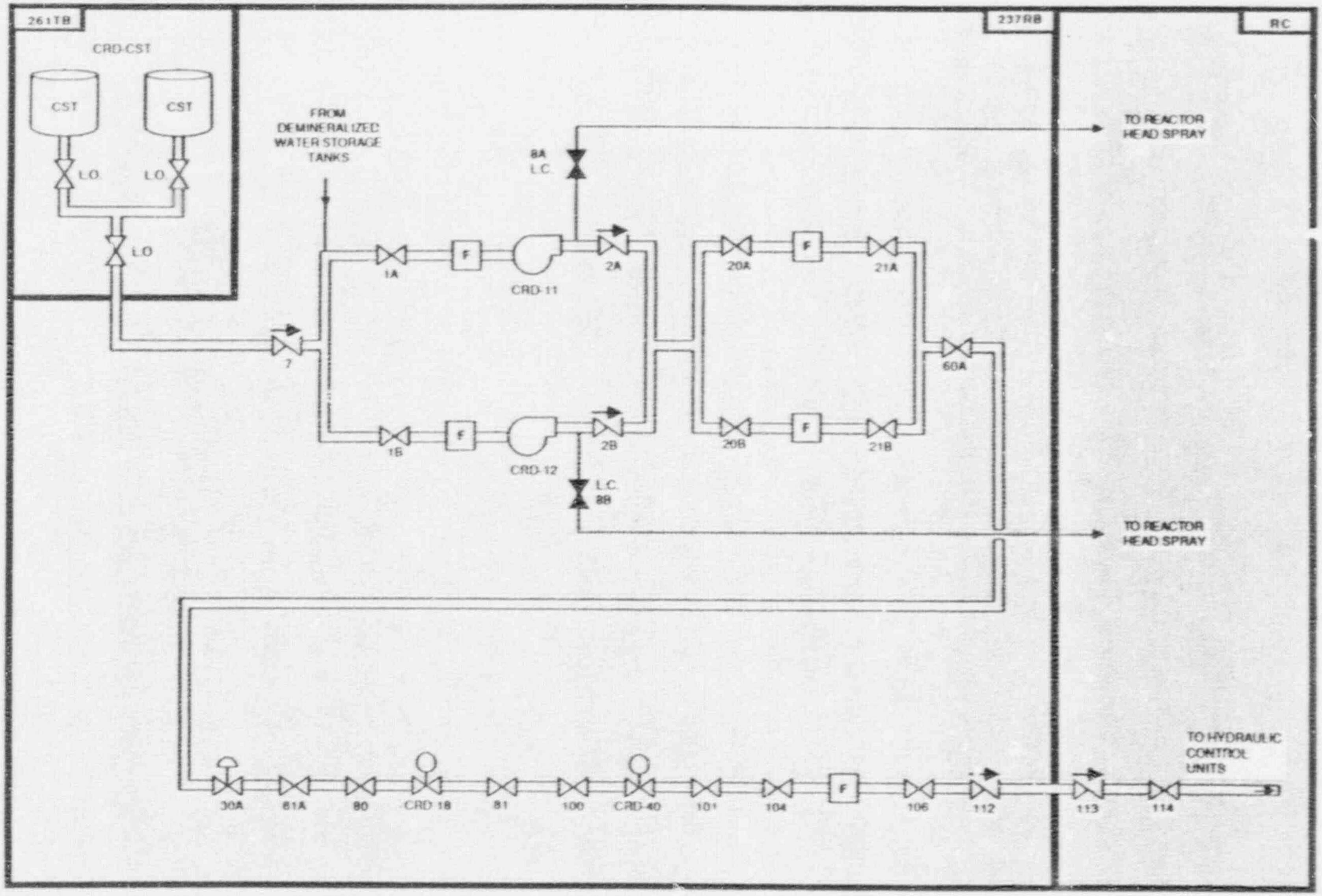


Figure 3.7-2. Nine Mile Point 1 Control Rod Drive Hydraulic System Showing Component Locations

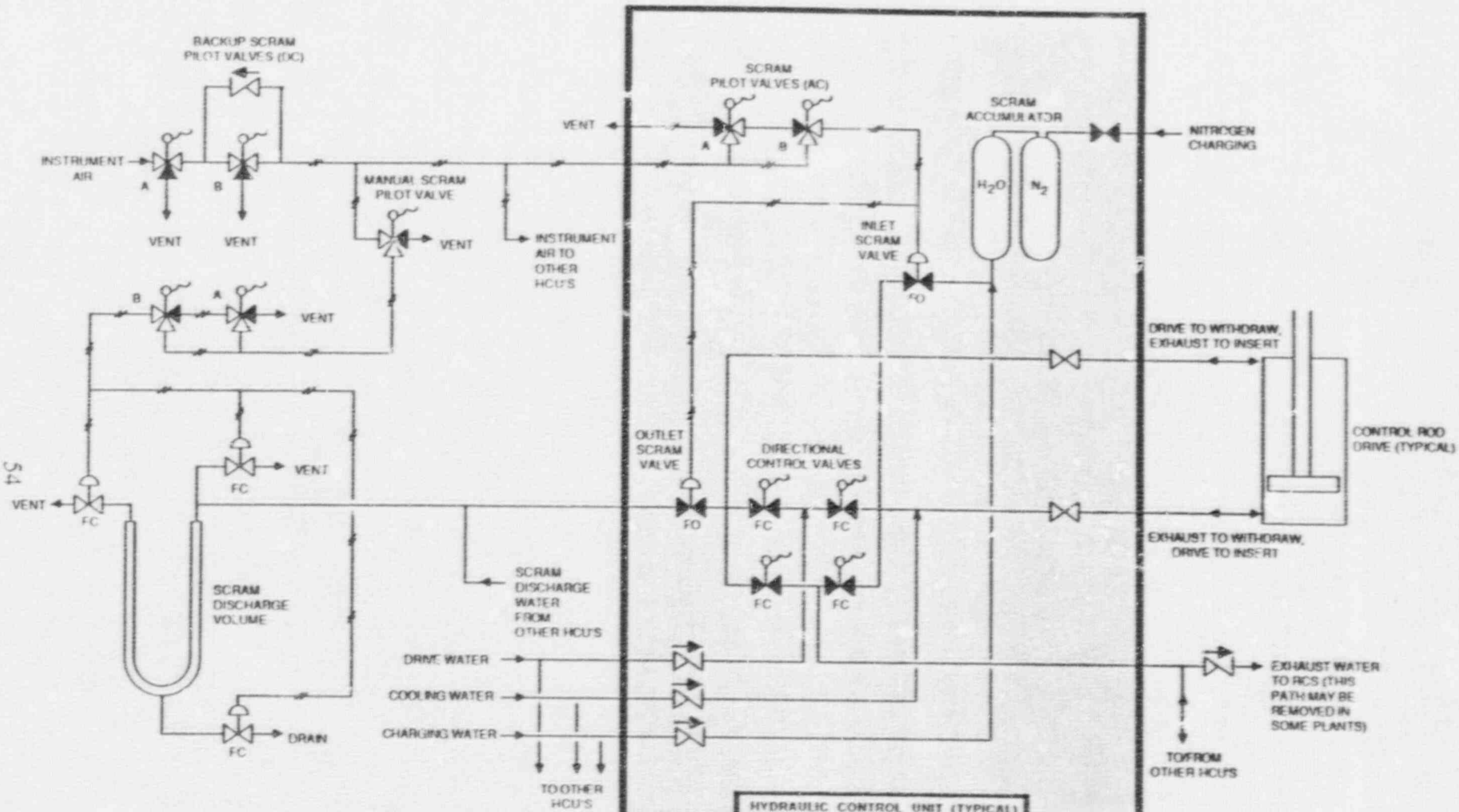


Figure 3.7-3. Simplified Diagram Of Portions Of The Control Rod Drive Hydraulic System That Are Related To The Scram Function

Table 3.7-1. Nine Mile Point 1 Control Rod Drive Hydraulic System Data Summary for Selected Components

COMPONENT ID	COMP. TYPE	LOCATION	POWER SOURCE	VOLTAGE	POWER SOURCE LOCATION	EMERG. LOAD GRP.
CRD-18	MOV	237RB	EP-BS-155	600	281RB	N/A
CRD-40	MOV	237RB	EP-BS-155	600	281RB	N/A
CRD-CST	TANK	261TB				
CRD-PM11	MDP	237RB	EP-BS-16	600	281RB	AC/102
CRD-PM12	MDP	237RB	EP-BS-17	600	281RB	AC/103

3.8 SHUTDOWN COOLING SYSTEM

3.8.1 System Function

The shutdown cooling system (SCS) provid... for shutdown cooling of the reactor after the RCS has been depressurized to less than 120 psig and 350°F. The SCS transfers decay heat to the reactor building closed cooling water system. The SCS is not part of the Emergency Core Cooling System.

3.8.2 System Definition

The SCS is essentially a single train system with three parallel SCS pumps and three parallel SCS heat exchangers. The system takes a suction on RCS recirculation loop C and returns coolant to recirculation loop B. Simplified drawings of the shutdown cooling system are shown in Figures 3.8-1 and 3.8-2.

3.8.3 System Operation

During normal operation, the SCS is shutdown and isolated from the RCS. Following reactor shutdown, RCS cooling is provided by steaming on the turbine bypass system and providing makeup to the RCS with the main condensate and feedwater system. The SCS is manually actuated during RCS cooldown and depressurization when RCS pressure is less than 120 psig and temperature is less than 350°F. The SCS is designed with sufficient capacity to remove decay heat being generated by the core and hold RCS temperature at 125°F (Ref. 1).

3.8.4 System Success Criteria

Two SCS pumps and heat exchangers are capable of establishing a maximum allowable RCS cooldown rate of 100°F per hour (Ref. 1). Three SCS pumps and heat exchangers can be operated if needed.

3.8.5 Component Information

A. SCS pumps (3)
   1. Type: Centrifugal
   2. Capacity: Not determined

B. SCS heat exchangers (3)
   1. Type: Horizontal U-tube
   2. Heat removal capacity: 12.5×10^6 Btu/hr

3.8.6 Support Systems and Interfaces

A. Control Signals
   1. The SCS is controlled from the reactor control room
   2. An interlock prevents opening both isolation valves between the RCS and SCS when RCS pressure greater than 120 psig. A single isolation valve can be exercised at higher RCS pressure.
   3. The SCS isolation valves are automatically closed by a low-low RCS vessel water level signal from the Reactor Protection System; or a high SCS area temperature from temperature detectors intended to sense the effects of an SCS pipe break.
   4. The SCS pump permissive instrumentation prevents pump operation unless suction pressure is above 4 psig and reactor water temperature is below 350°F.

B. Motive Power
The power sources for the SCS pumps has not been determined.

C. Others
The reactor building closed cooling water (RBCCW) system provides cooling water to the SCS heat exchangers and to the SCS pump bearings.

3.8.7 Section 3.8 References

1. Final Safety Report, Nin Mile Point 1 Nuclear Power Station Section X-A.

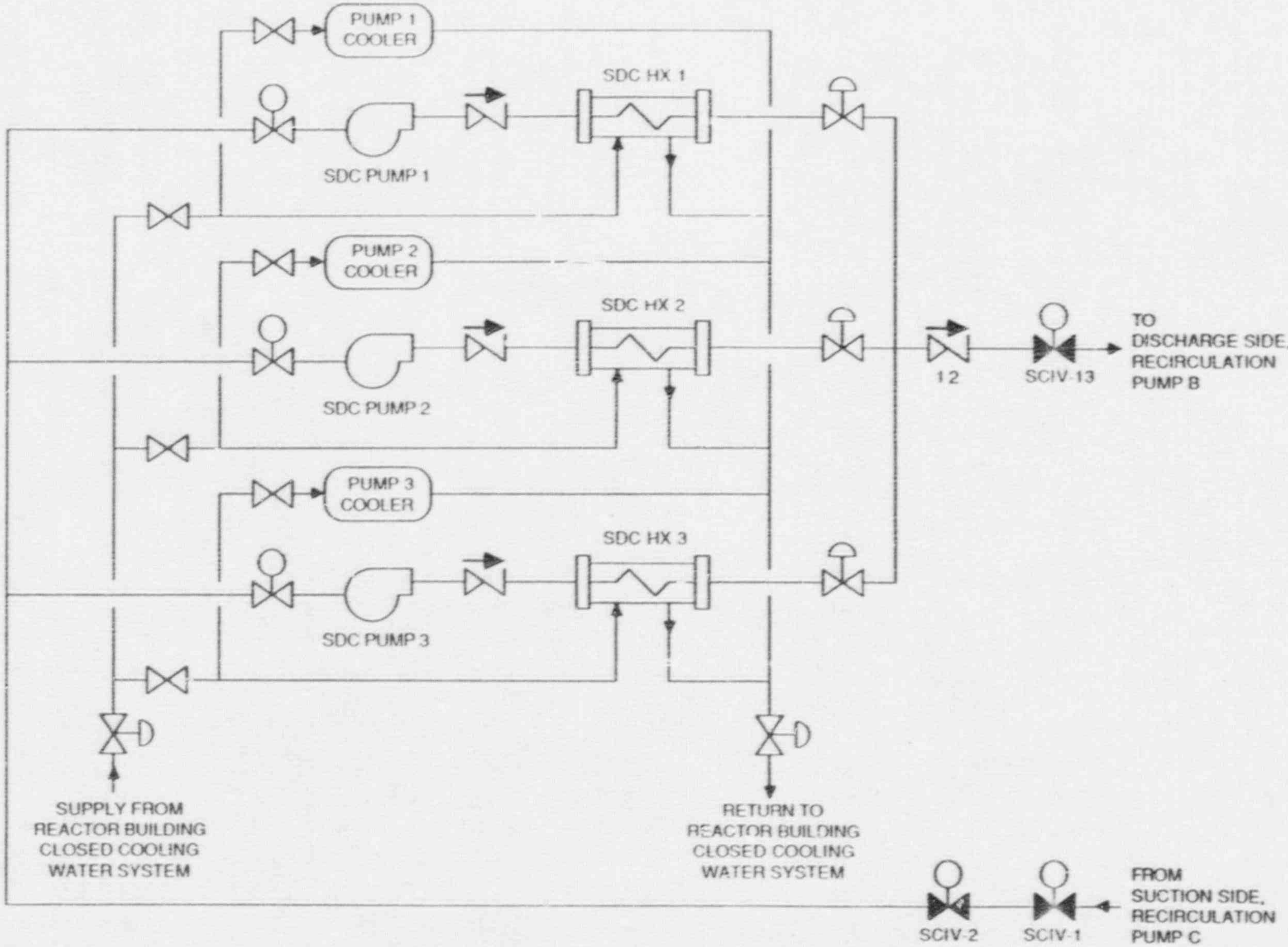


Figure 3.8-1. Nine Mile Point 1 Shutdown Cooling System

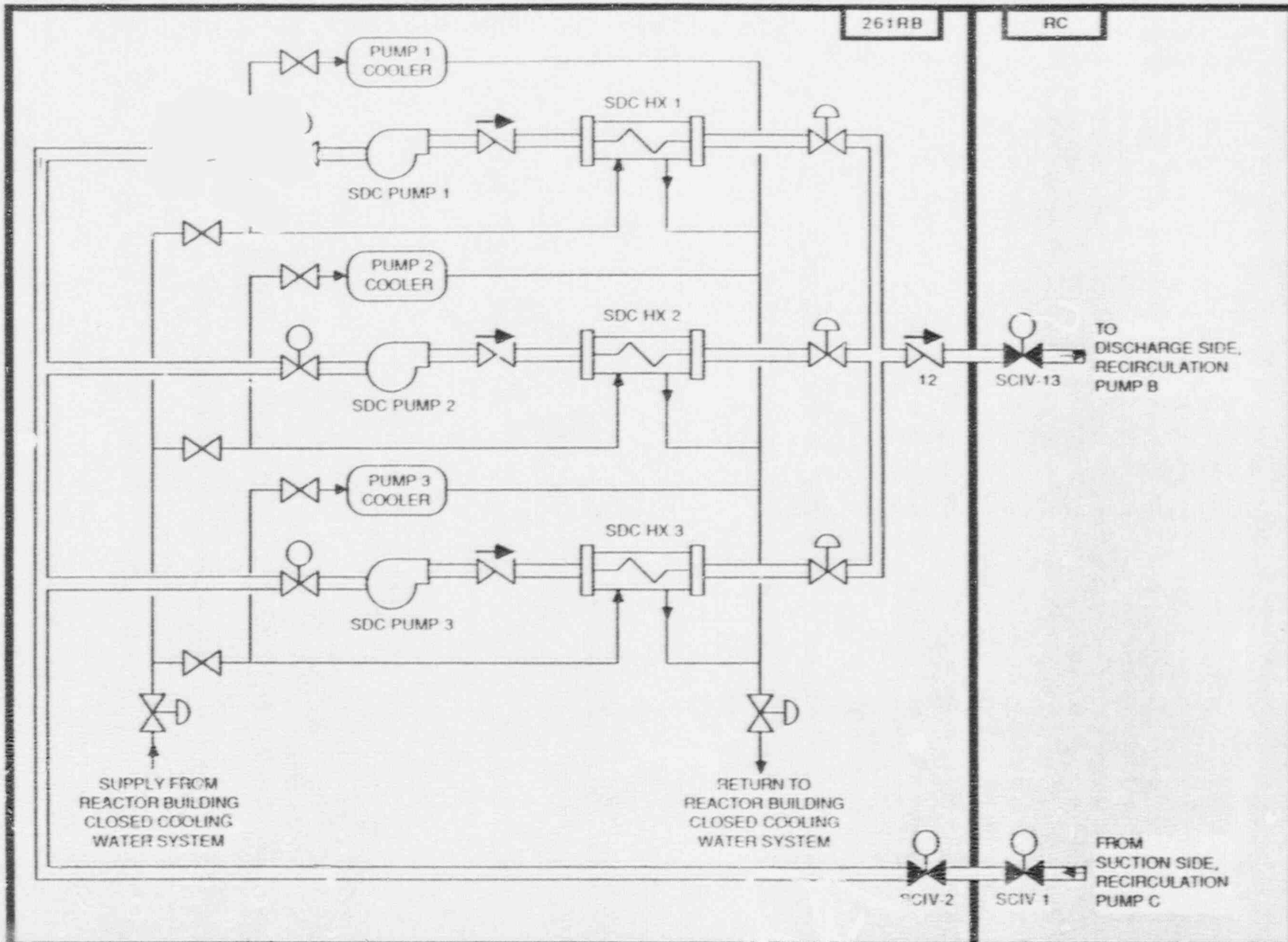


Figure 3.8-2. Nine Mile Point 1 Shutdown Cooling System Showing Component Locations

4. PLANT INFORMATION

4.1 SITE AND BUILDING SUMMARY

The Nine Mile Point 1 site is located on the Lake Ontario coast in the town of Scriba, New York. It is in the north central portion of Oswego County, approximately five miles north-northwest of the nearest boundary of the city of Oswego, and 230 miles northwest of New York City. A general view of the site is shown in Figure 4-1 (from Ref. 1) and a more detailed site plan is shown in Figure 4-2.

The reactor building is located between the turbine generator building and the screen and pump house. It houses the reactor vessel and several safety-related components for the CSS, CNS, and CRDHS. These components include pumps, valves and piping.

The turbine generator building is located adjacent to the reactor building toward the south south-west. It also houses important safety-related systems including the two diesel-generators, Class 1E power boards, and the control room. The screen and pump house building lies directly north of the turbine building.

4.2 FACILITY LAYOUT DRAWINGS

Figures 4-3 and 4-4 are section views of the Nine Mile Point 1 plant. Simplified building layout drawings for Nine Mile Point 1 are shown in Figures 4-5 to 4-11. A cutaway view of the upper floors of reactor building are shown in Figure 4-12 (from Ref. 2). Major rooms, stairways, elevators, and doorways are shown in the simplified layout drawings, however, many interior walls have been omitted for clarity. Labels printed in uppercase correspond to the location codes listed in Table 4-1 and used in the component data listings and system drawings in Section 3. Some additional labels are included for information and are printed in lowercase type.

A listing of components by location is presented in Table 4-2. Components included in Table 4-2 are those found in the system data tables in Section 3, therefore this table is only a partial listing of the components and equipment that are located in a particular room or area of the plant.

4.3 SECTION 4 REFERENCES

1. Heddleson, F.A., "Design Data and Safety Features of Commercial Nuclear Power Plants.", ORNL-NSIC-55, Volume 1, Oak Ridge National Laboratory, Nuclear Safety Information Center, December 1973.

2. Final Safety Analysis Report, Nine Mile Point 1 Nuclear Power Station.

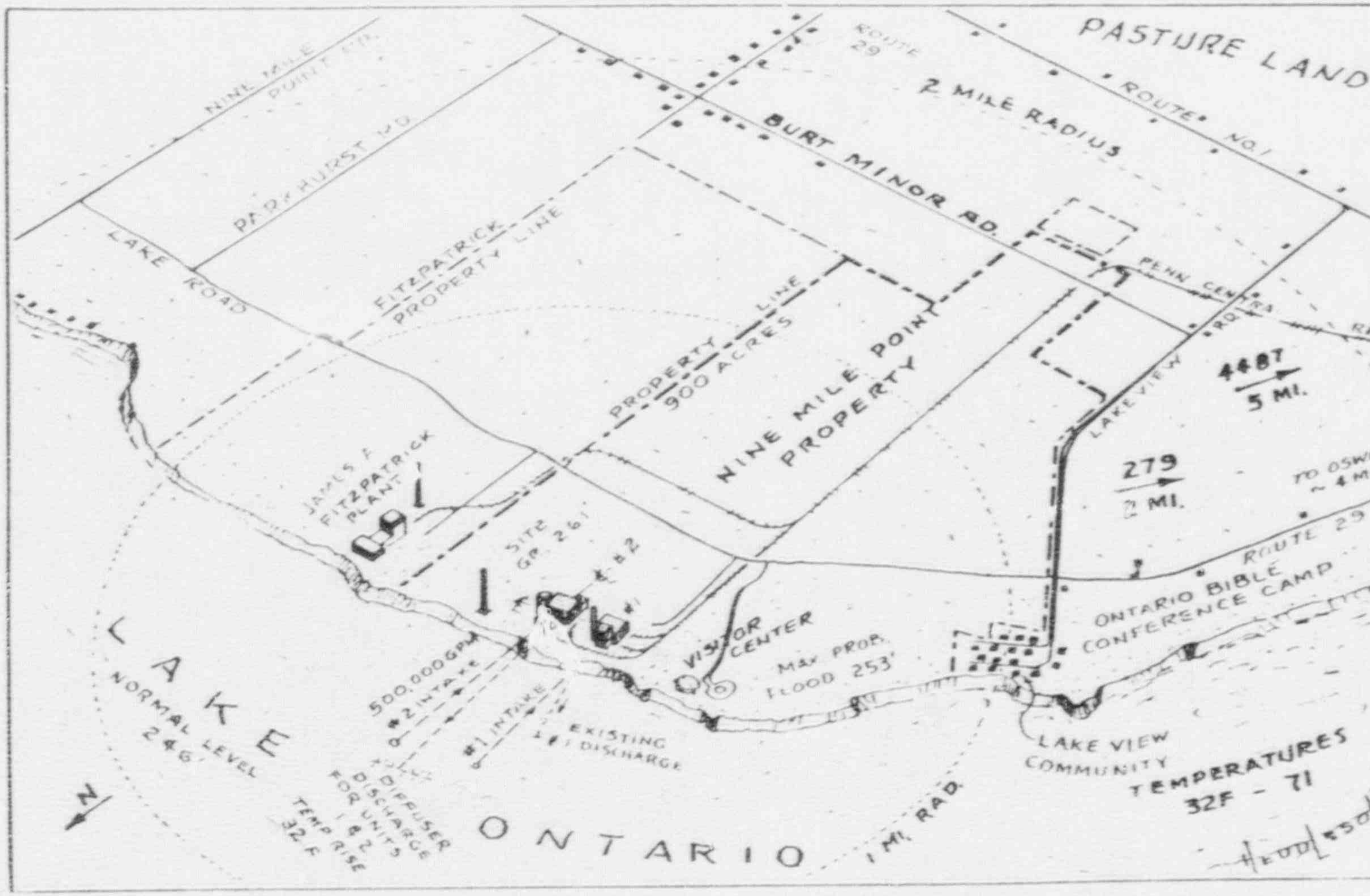


Figure 4-1. General View of the Nine Mile Point 1 Site and Vicinity

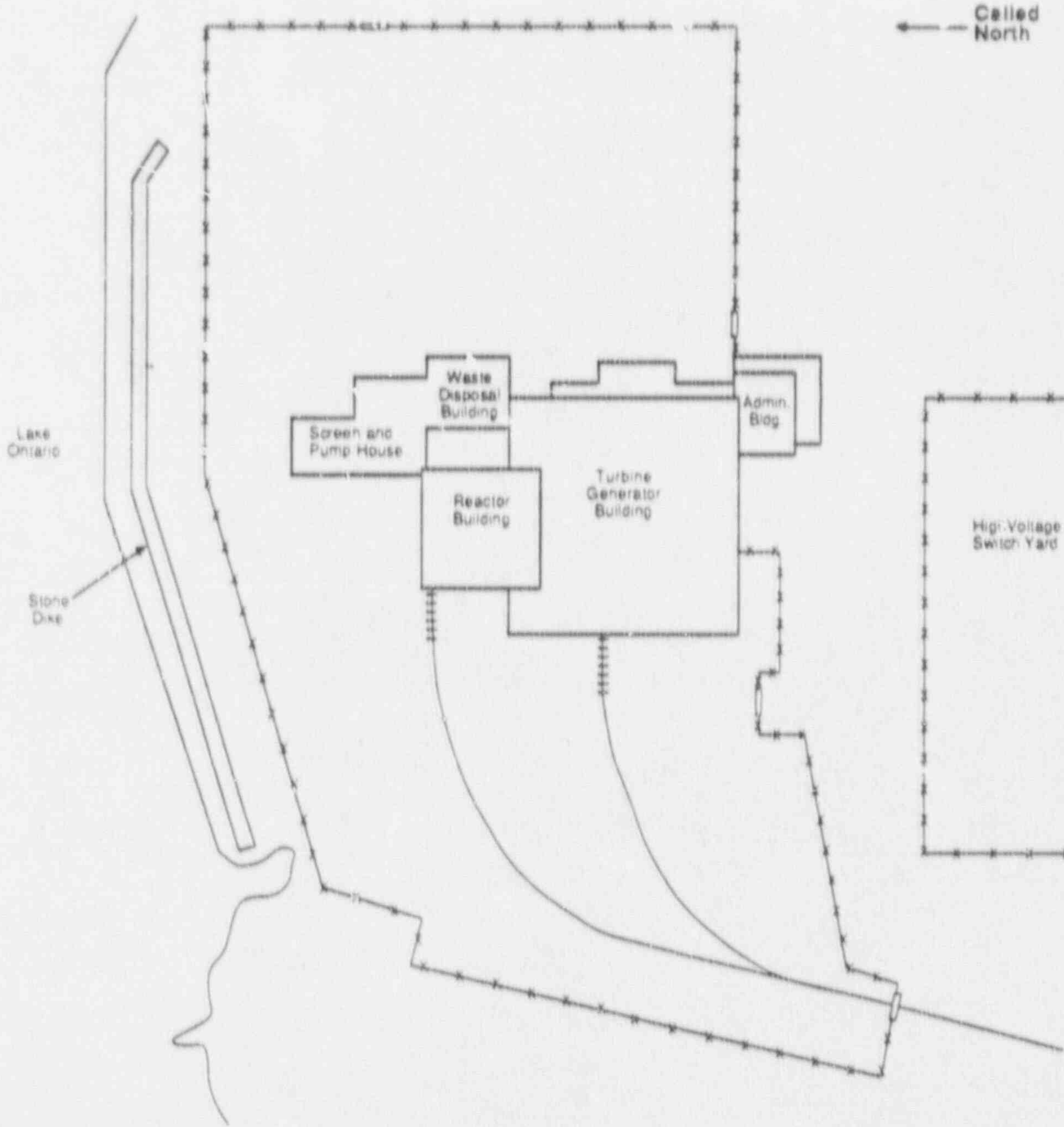


Figure 4-2. Nine Mile Point 1 Simplified Site Plan

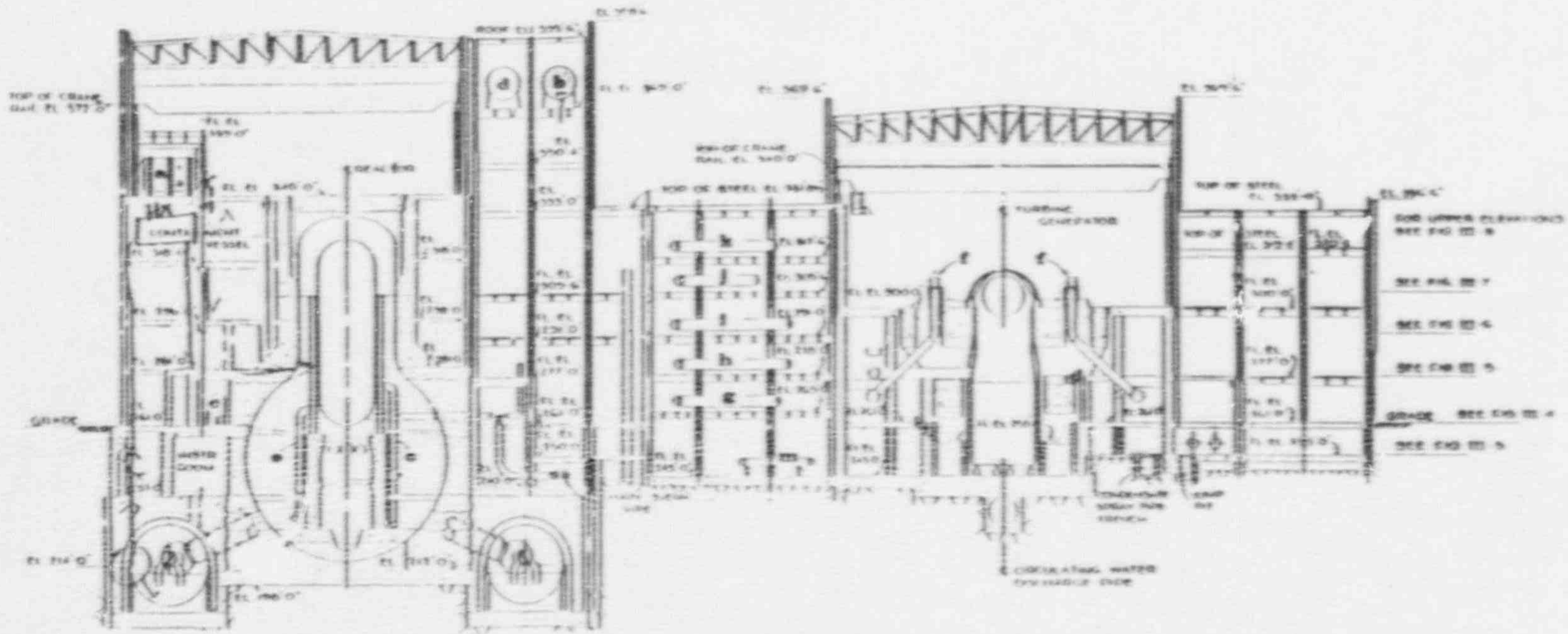


Figure 4-3. Nine Mile Point 1 Reactor and Turbine Buildings Section, Looking East

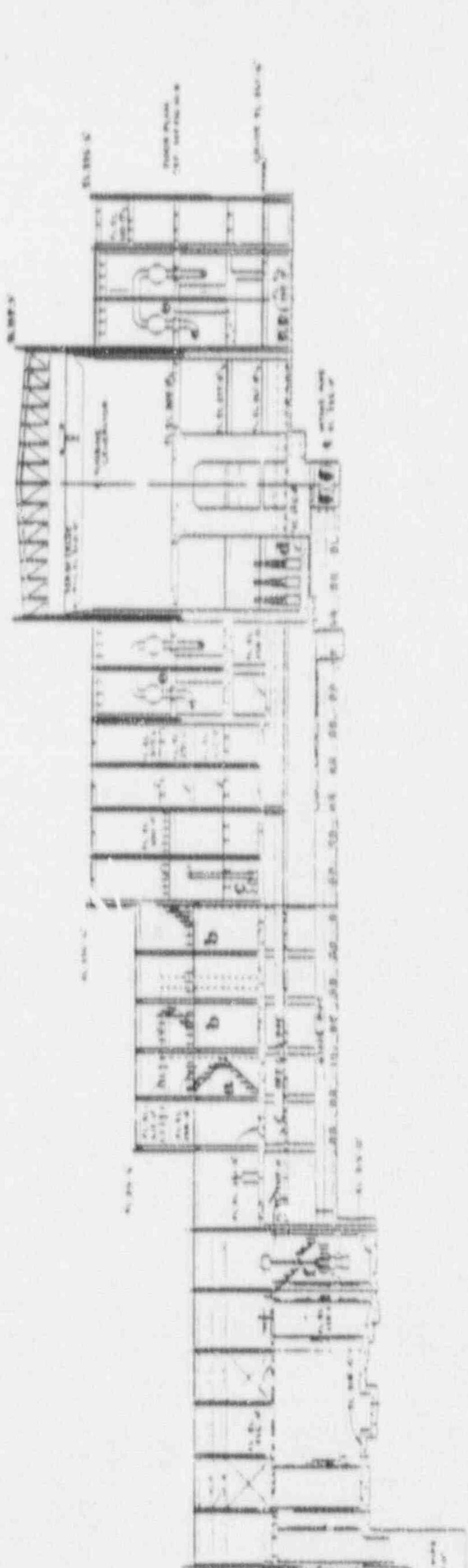


Figure 4-4. Nine Mile Point 1 Screen/Pump House and Turbine Buildings Section, Looking East

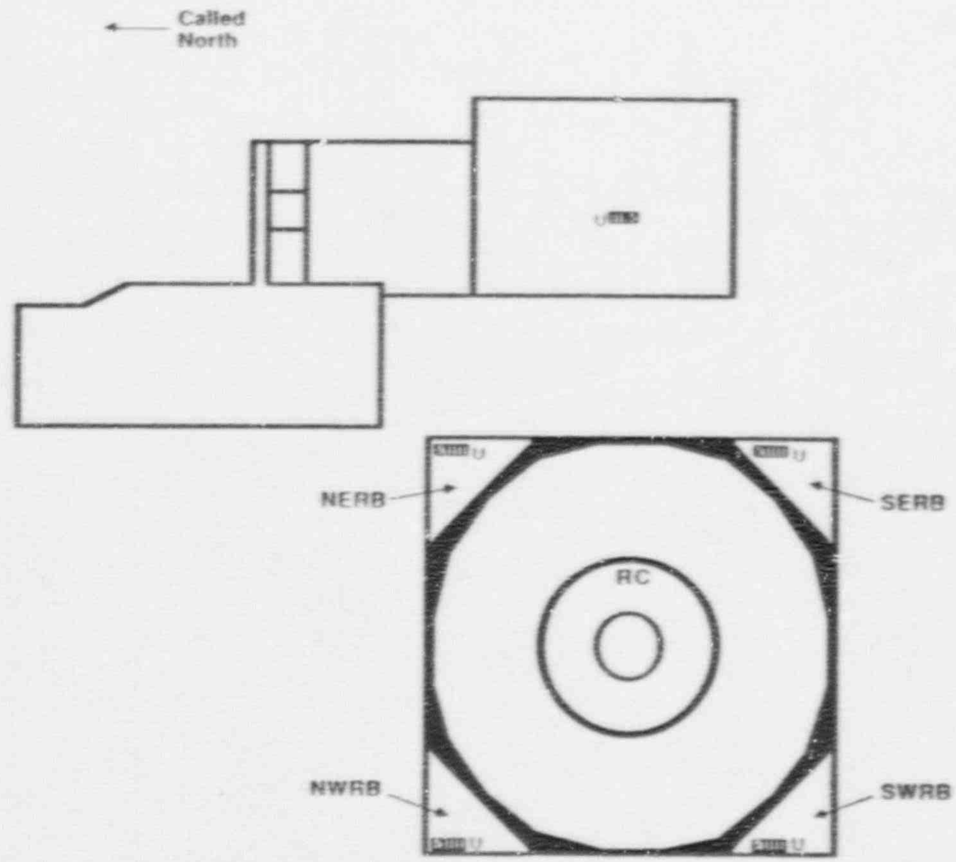


Figure 4-5. Nine Mile Point 1 Elevation 225'

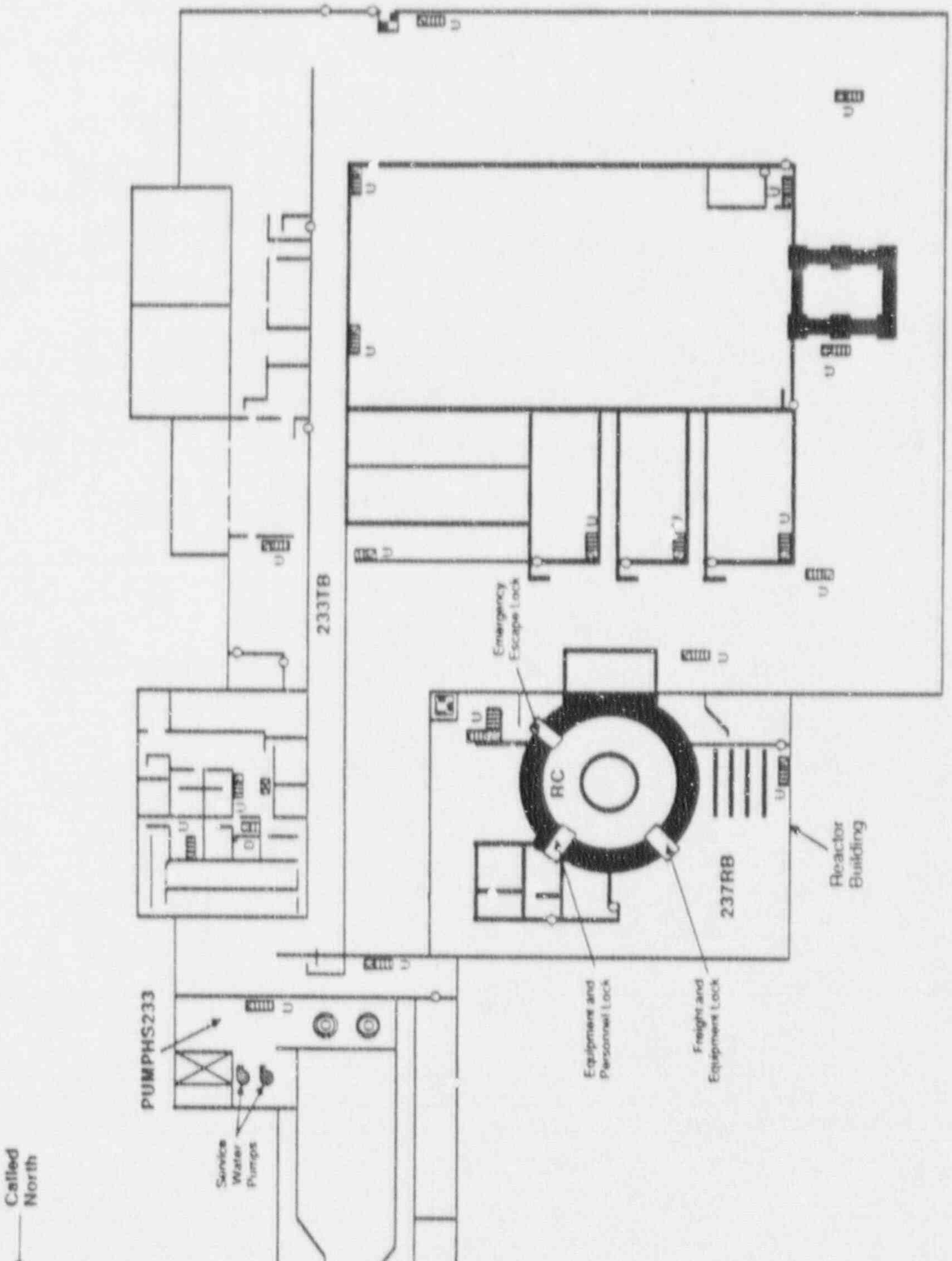


Figure 4-6. Nine Mile Point 1 Elevation 250'

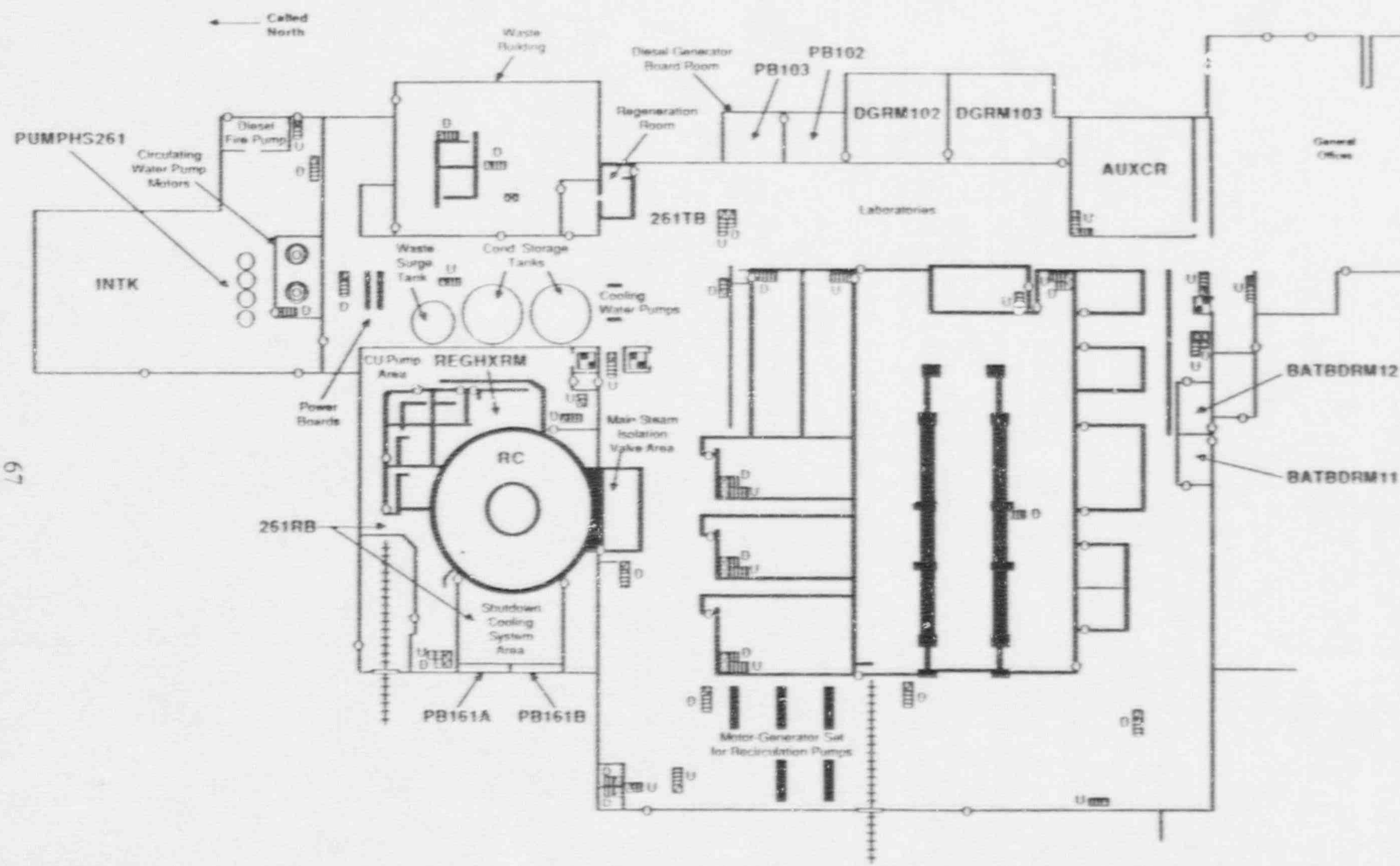


Figure 4-7. Nine Mile Point 1 Elevation 261'

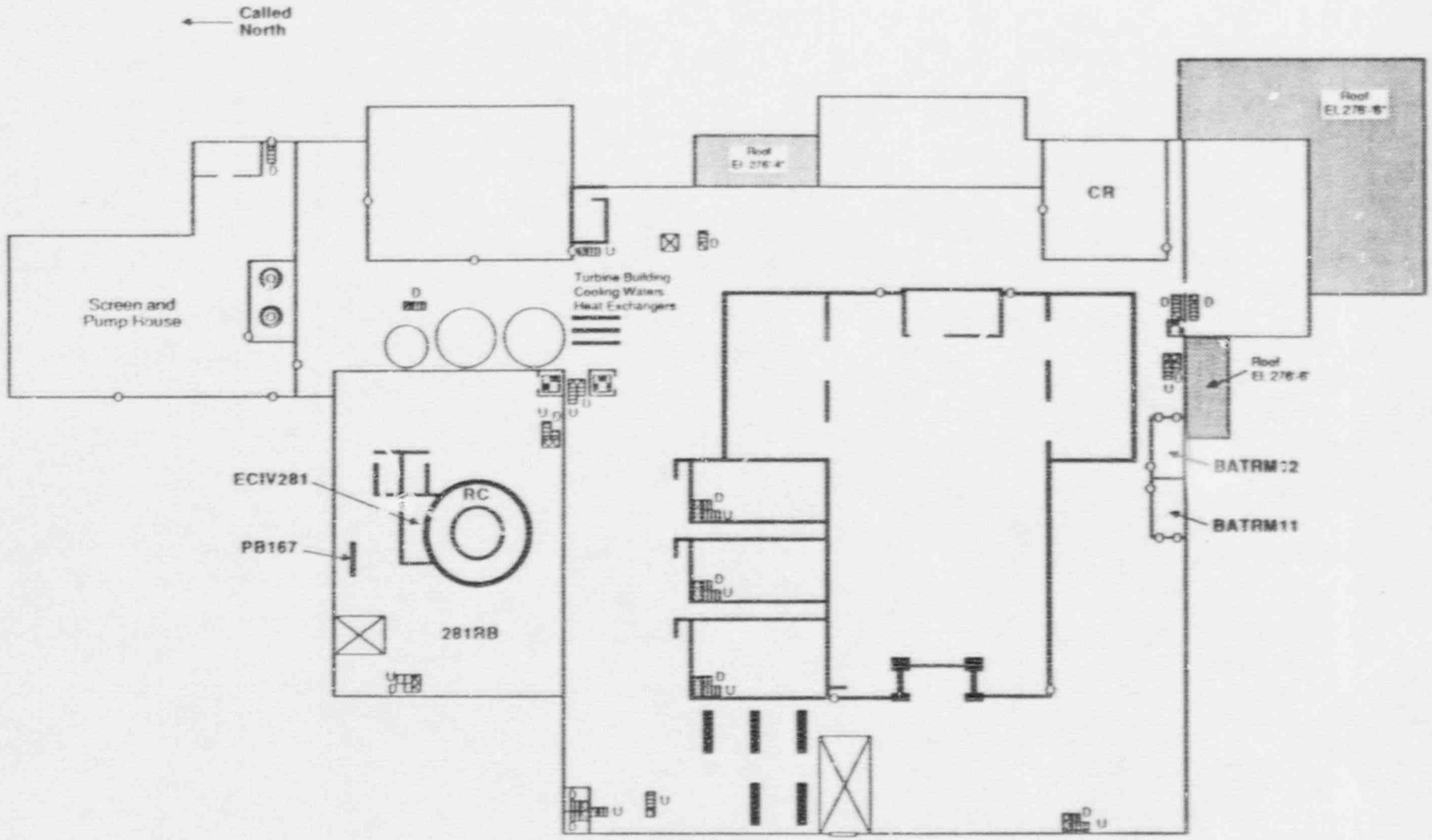


Figure 4-8. Nine Mile Point 1 Elevation 277'

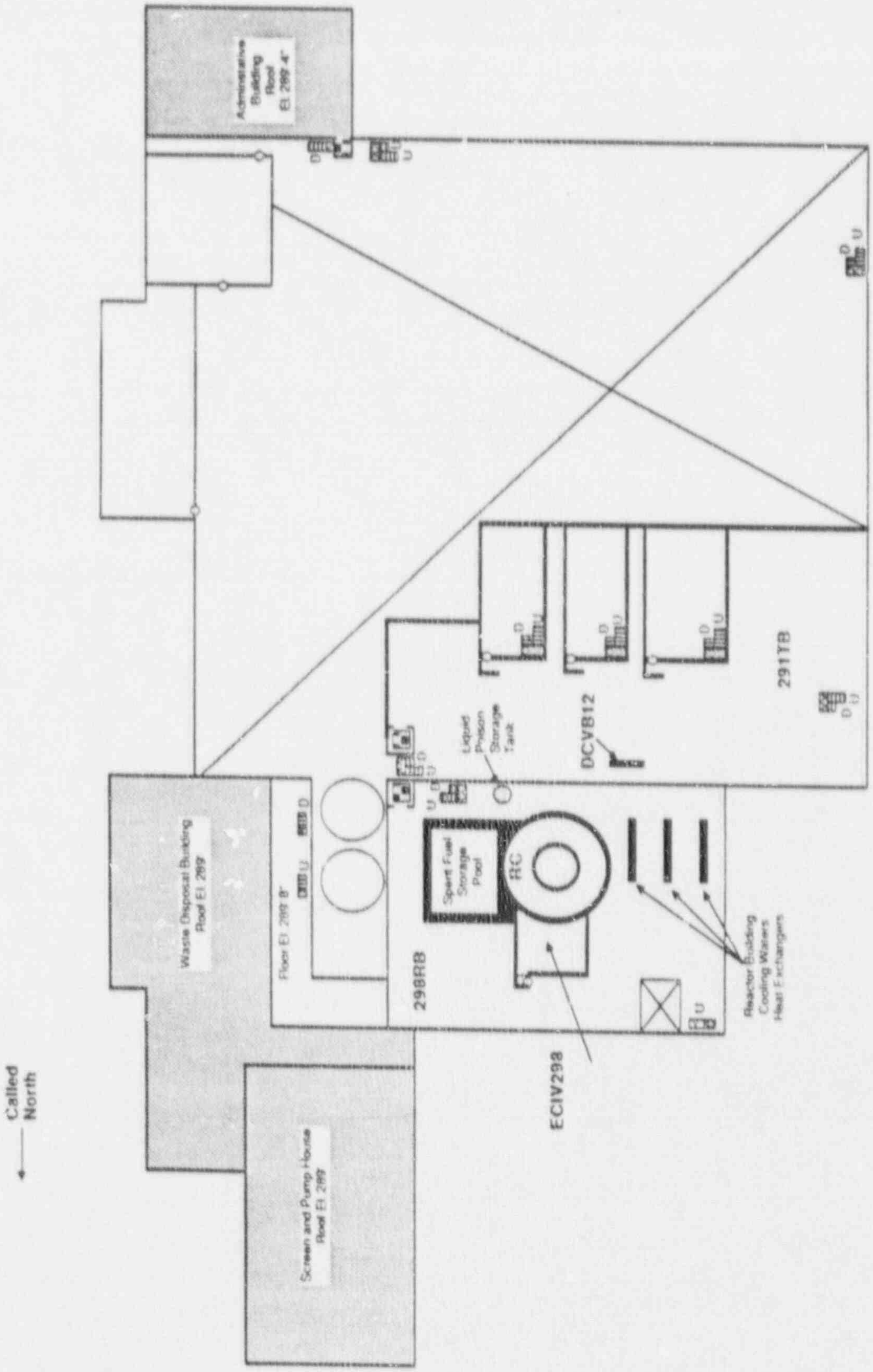


Figure 4-9. Nine Mile Point 1 Elevation 291'

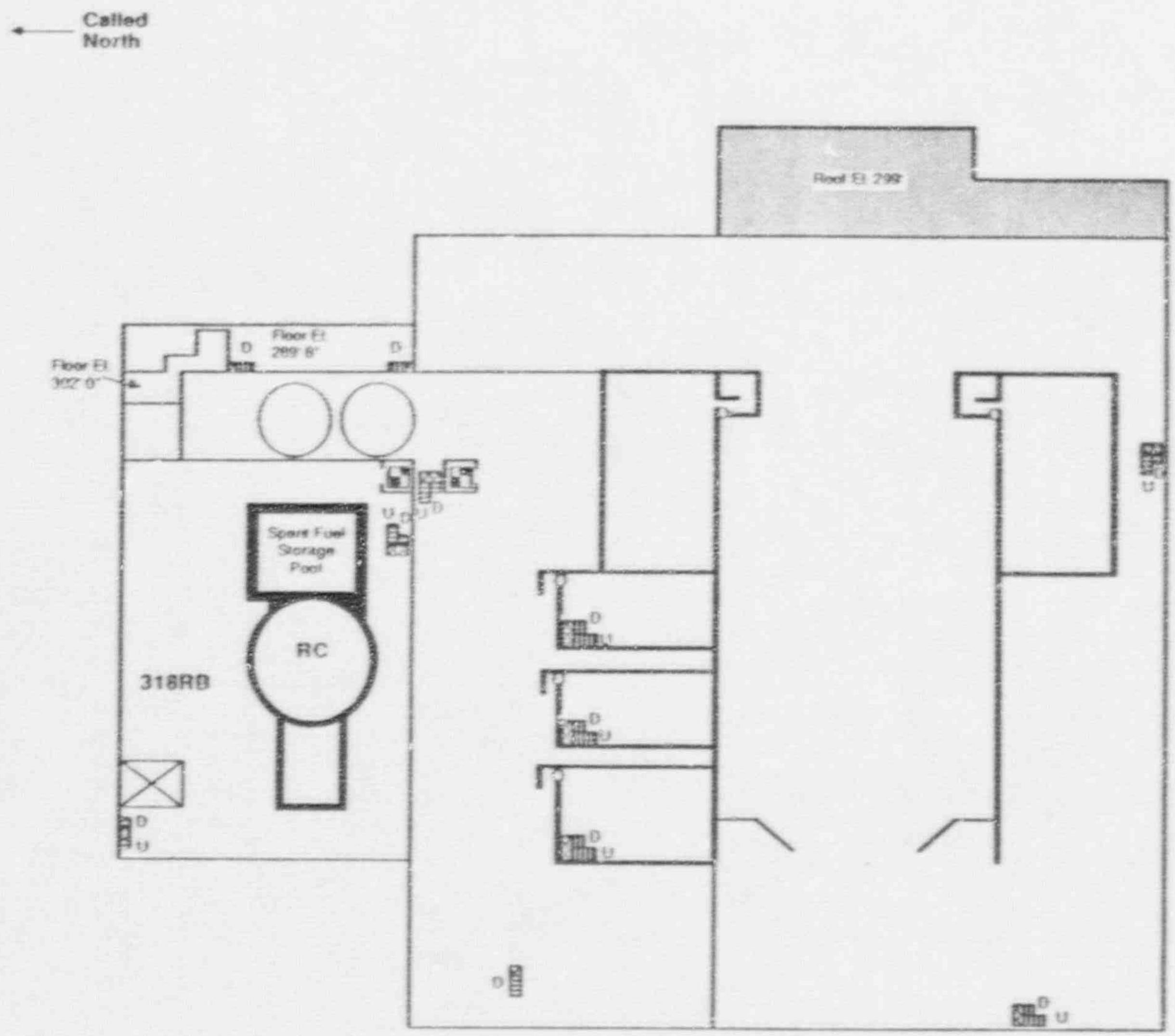


Figure 4-10. Nine Mile Point 1 Elevation 305'

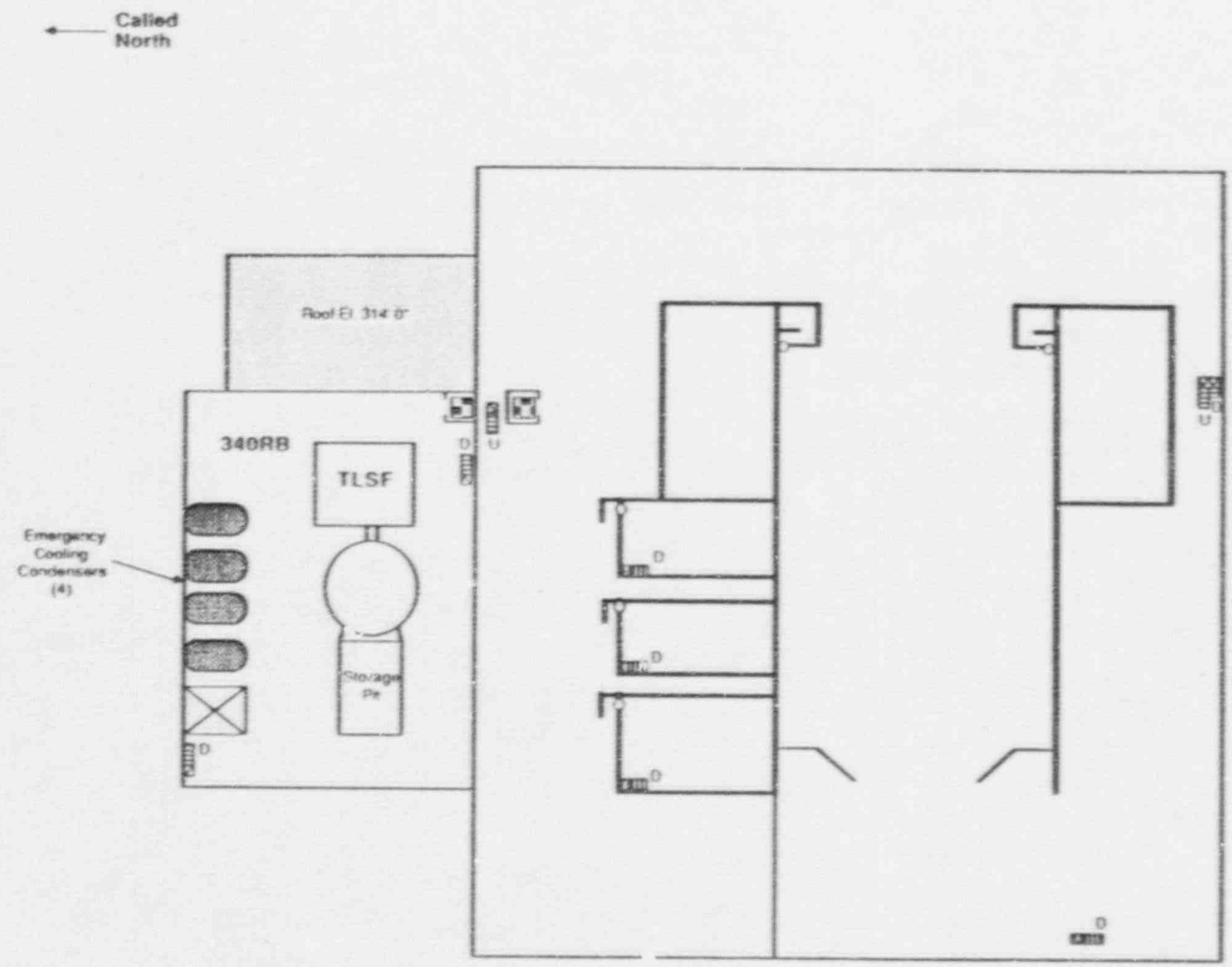


Figure 4-11. Nine Mile Point 1 Elevations 317' and 369'

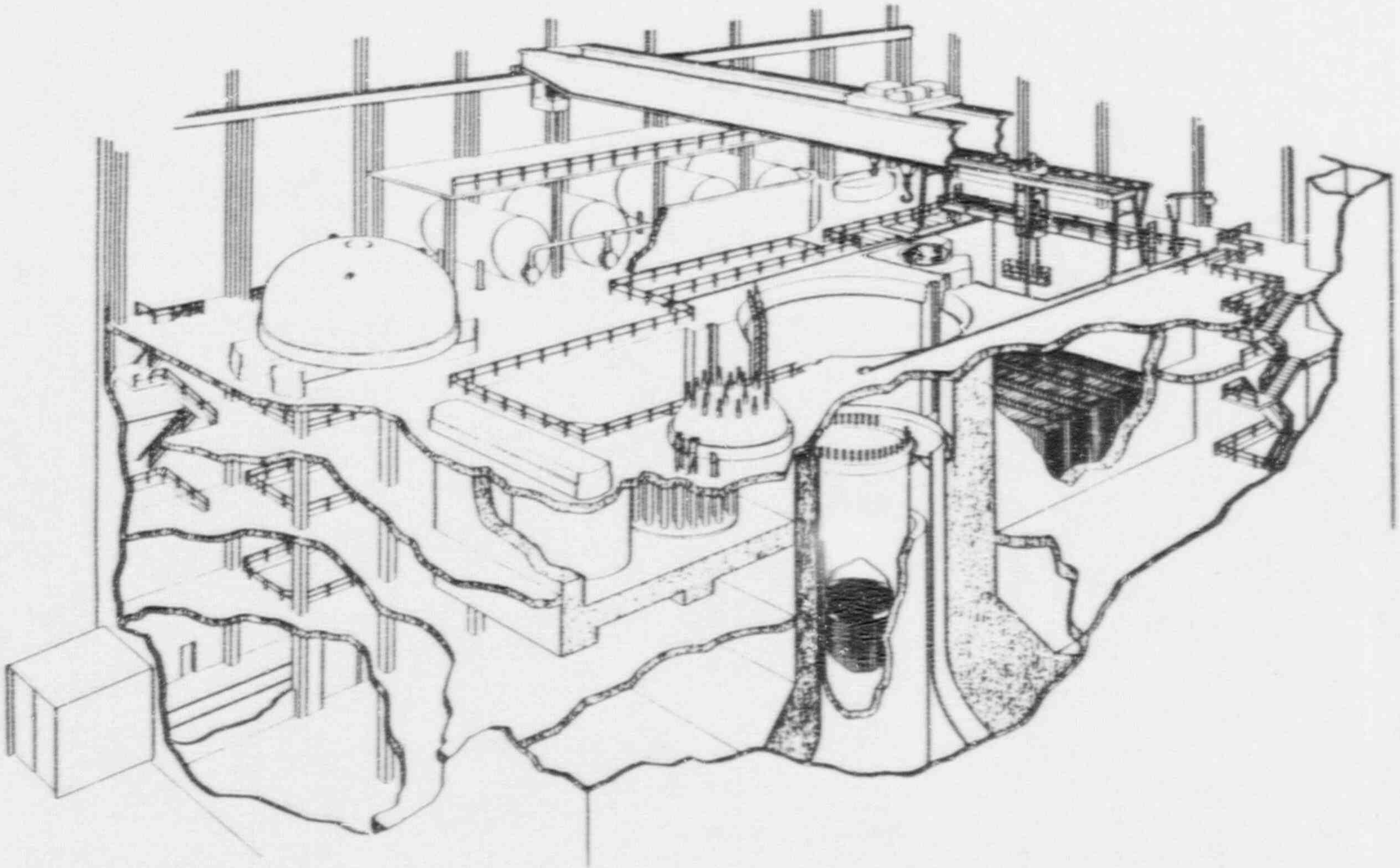


Figure 4-12. Cut-away View of the Upper Floors of the Nine Mile Point 1 Reactor Building

Table 4-1. Definition of Nine Mile Point 1 Building and Location Codes

	Codes	Descriptions
1.	233TB	233'0" - elevation of the Turbine Building
2.	237RB	237'0" - elevation of the Reactor Building
3.	261RB	261'0" - elevation of the Reactor Building
4.	261TB	261'0" - elevation of the Turbine Building
5.	281RB	281'0" - elevation of the Reactor Building
6.	291TB	291'0" - elevation of the Turbine Building
7.	298RB	298'0" - elevation of the Reactor Building
8.	318RB	318'0" - elevation of the Reactor Building
9.	340RB	340'0" - elevation of the Reactor Building
10.	AUXCP	Auxiliary Control Room, on the 261' elevation of the Turbine Building
11.	BATBDRM11	Battery Board Room #11 - located in the Turbine Building at elevation 261'0"
12.	BATBDRM12	Battery Board Room #12 - located in the Turbine Building at elevation 261'0"
13.	BATRM11	Battery Room #11 - located in the Turbine Building at elevation 277'0" south side
14.	BATRM12	Battery Room #12 - located in the Turbine Building at elevation 277'0" south side
15.	CR	Control Room - elevation 277'0"
16.	DCVB12	DC Valve Board #12 - located in the Turbine Building at elevation 261'0" south side
17.	DGRM102	Diesel Generator Room #102 on the 261' elevation of the Turbine Building
18.	DGRM103	Diesel Generator Room #103 on the 261' elevation of the Turbine Building

Table 4-1. Definition of Nine Mile Point 1 Building and Location Codes (Continued)

Codes	Descriptions
19. ECIV281	Emergency Cooling Isolation Valve Room - located in the Reactor Building at elevation 281'0"
20. ECIV298	Emergency Cooling Isolation Valve Room - located in the Reactor Building at elevation 298'0"
21. FT	Location of long-term fuel tanks for diesel generators (to be determined)
22. INTK	Intake structure
23. MGRM161	Location of motor-generator set #161 (to be determined)
24. NERB	Northeast corner of the Reactor Building at elevation 225'0"
25. NWRB	Northwest corner of the Reactor Building at elevation 225'0"
26. PB102	Power Board #102 - located in Turbine Building - elevation 277'0"
27. PB103	Power Board #103 - located in the Turbine Building at elevation 277'0"
28. PB167	Power Board #167 - located in the Reactor Building at elevation 281'0" north side
29. PUMPHS233	233'0" - elevation of the Pump House Building
30. PUMPHS261	261'0" - elevation of the Pump House Building
31. RC	Reactor Containment
32. REGHXRM	Regenerative Heat Exchanger Room - located in the Reactor Building at elevation 261'0"
33. SERB	Southeast corner of the Reactor Building at elevation 225'0"
34. SWRB	Southwest corner of the Reactor Building at elevation 225'0"
35. TLSF	Spent fuel operating floor on the 340' elevation of the Reactor Building
36. TORUS	Suppression Chamber - elevation 225'0"

Table 4-2. Partial Listing of Components by Location at Nine Mile Point 1

LOCATION	SYSTEM	COMPONENT ID	COMP TYPE
237RB	CNS	CSSA-51	MOV
237RB	CNS	CSSB-52	MOV
237RB	CRD	CRD-18	MOV
237RB	CRD	CRD-40	MOV
237RB	CRD	CRD-PM11	MDP
237RB	CRD	CRD-PM12	MDP
237RB	ECCS	CSSB-PMT121	MDP
237RB	ECCS	CSSB-PMT122	MDP
237RB	ECCS	CSSB-2	MOV
237RB	ECCS	CSSA-PMT111	MDP
237RB	ECCS	CSSA-PMT112	MDP
237RB	ECCS	CSSA-12	MOV
237RB	ECCS	CSSA-51	MOV
237RB	ECCS	CSSB-52	MOV
261RB	EP	EP-BS-161	BUS
261RB	EP	EP-BS-171	BUS
261RB	RCS	SCIV-2	MOV
261TB	CRD	CRD-CST	TANK
261TB	EP	EP-BC-171	BC
261TB	EP	EP-TR-17	TRAN
261TB	EP	EP-TR-16	TRAN
281RB	EP	EP-BS-17	BUS
281RB	EP	EP-BS-16	BUS
318RB	CNS	CSB-35	NV
318RB	CNS	CSB-36	NV
318RB	CNS	CSA-15	NV
318RB	CNS	CSA-16	NV
318RB	CNS	RWB-28	MOV
318RB	CNS	RWB-27	MOV
318RB	CNS	RWA-25	MOV
318RB	CNS	RWA-26	MOV
318RB	CNS	RWB2-50	MOV
318RB	CNS	RWA2-49	MOV

Table 4-2. Partial Listing of Components by Location at Nine Mile Point 1 (Continued)

LOCATION	SYSTEM	COMPONENT ID	COMP TYPE
318RB	CNS	CSB-HX12	HX
318RB	CNS	CSB-HX22	HX
318RB	CNS	CSA-HX11	HX
318RB	CNS	CSA-HX21	HX
318RB	CNS	CSA-HX11	HX
318RB	CNS	CSA-HX21	HX
318RB	CNS	CSB-HX22	HX
318RB	CNS	CSB-HX12	HX
318RB	CNS	RWA2-49	MOV
318RB	CNS	RWB2-50	MOV
340RB	ECCS	EC11-HX	HX
340RB	ECCS	EC12-HX	HX
369TB	ECCS	EC11-MU	TANK
369TB	ECCS	EC12-MU	TANK
BATBDRM11	EP	EP-BTBD-11	BUS
BATBDRM11	EP	EP-BTBD-11	BUS
BATBDRM12	EP	EP-BTBD-12	BUS
BATBDRM12	EP	EP-BTBD-12	BUS
BATRM11	EP	EP-BT-11	BATT
BATRM12	EP	EP-BT-12	BATT
DCVB12	EP	EP-VB12	BUS
DGRM102	EP	EP-DG-102	DG
DGRM103	EP	EP-DG-103	DG
ECIV281	ECCS	EC11-5	NV
ECIV281	ECCS	EC12-6	NV
ECIV298	ECCS	EC11-9	MOV
ECIV298	ECCS	EC11-7	MOV
ECIV298	ECCS	EC12-10	MOV
ECIV298	ECCS	EC12-8	MOV
ECIV298	RCS	EC11-9	MOV
ECIV298	RCS	EC11-7	MOV
ECIV298	RCS	EC12-10	MOV
ECIV298	RCS	EC12-8	MOV

Table 4-2. Partial Listing of Components by Location at Nine Mile Point 1 (Continued)

LOCATION	SYSTEM	COMPONENT ID	COMP TYPE
MGRM161	EP	EP-BC-161	BC
NERB	CNS	CSB-PM122	MDP
NERB	CNS	CSA-PM121	MDP
NWRB	CNS	CSB-21	MOV
NWRB	CNS	CSB-22	MOV
NWRB	CNS	CSB-PM112	MDP
NWRB	CNS	CSA-1	MOV
NWRB	CNS	CSA-2	MOV
NWRB	CNS	CSA-PM111	MDP
PB102	EP	EP-BS-102	BUS
PB102	EP	EP-CB-102	CB
PB103	EP	EP-BS-103	BUS
PB103	EP	EP-CB-103	CB
PB167	EP	EP-BS-167	BUS
PB167	EP	EP-BS-167	BUS
PUMPHS261	CNS	RWB-PM122	MDP
PUMPHS261	CNS	RWB-PM112	MDP
PUMPHS261	CNS	RWA-PM111	MDP
PUMPHS261	CNS	RWA-PM121	MDP
RC	CNS	RCS-RV	RV
RC	CNS	RCS-RV	RV
RC	ECCS	RCS-RV	RV
RC	ECCS	RCS-RV	RV
RC	ECCS	RCS-RV	RV
RC	ECCS	RCS-RV	RV
RC	ECCS	RCS-ADS	PORV
RC	ECCS	CSSA-11	MOV
RC	ECCS	CSSA-10	MOV
RC	ECCS	CSSB-1	MOV
RC	ECCS	CSSB-9	MOV
RC	ECCS	RCS-ADS	PORV
RC	RCS	RCS-RV	RV
RC	RCS	RCS-SV	SV

Table 4-2. Partial Listing of Components by Location at Nine Mile Point 1 (Continued)

LOCATION	SYSTEM	COMPONENT ID	COMP TYPE
RC	RCS	RCS-ADS	PORV
RC	RCS	RWCU-2	MOV
RC	RCS	RCS-ADS	PORV
RC	RCS	SCIV-1	MOV
RC	RCS	SCIV-13	MOV
RC	RCS	CSSA-10	MOV
RC	RCS	CSSA-11	MOV
RC	RCS	CSSB-1	MOV
RC	RCS	CSSB-9	MOV
RC	RCS	RCS-ADS	PORV
SERB	ECCS	CSSB-PM121	MDP
SERB	ECCS	CSSB-PM122	MDP
SERB	ECCS	CSSB-1S	MOV
SERB	ECCS	CSSB-2S	MOV
SERB	ECCS	CSSB-TV12	MOV
SWRB	ECCS	CSSA-PM111	MDP
SWRB	ECCS	CSSA-PM112	MDP
SWRB	ECCS	CSSA-21	MOV
SWRB	ECCS	CSSA-22	MOV
SWRB	ECCS	CSSA-TV11	MOV
TORUS	ECCS	CSSA-B-TORUS	TOR
TORUS	ECCS	CSSA-B-TORUS	TOR

5. BIBLIOGRAPHY FOR NINE MILE POINT 1

1. NUREG/CR-2746, Vol. 6, "Socioeconomic Impacts of Nuclear Generating Station: Nine Mile Point and Fitzpatrick Case Study," Social Impact Research, Inc., July 1982.

APPENDIX A
DEFINITION OF SYMBOLS USED IN THE SYSTEM AND LAYOUT DRAWINGS

A1. SYSTEM DRAWINGS

A1.1 Fluid System Drawings

The simplified system drawings are accurate representations of the major flow paths in a system and the important interfaces with other fluid systems. As a general rule, small fluid lines that are not essential to the basic operation of the system are not shown in these drawings. Lines of this type include instrumentation lines, vent lines, drain lines, and other lines that are less than 1/3 the diameter of the connecting major flow path. There usually are two versions of each fluid system drawing; a simplified system drawing, and a comparable drawing showing component locations. The drawing conventions used in the fluid system drawings are the following:

- Flow generally is left to right.
  - Water sources are located on the left and water "users" (i.e., heat loads) or discharge paths are located on the right.
  - One exception is the return flow path in closed loop systems which is right to left.
  - Another exception is the Reactor Coolant System (RCS) drawing which is "vessel-centered", with the primary loops on both sides of the vessel.
  - Horizontal lines always dominate and break vertical lines.

- Component symbols used in the fluid system drawings are defined in Figure A-1.
  - Most valve and pump symbols are designed to allow the reader to distinguish among similar components based on their support system requirements (i.e., electric power for a motor or solenoid, steam to drive a turbine, pneumatic or hydraulic source for valve operation, etc.)
  - Valve symbols allow the reader to distinguish among valves that allow flow in either direction, check (non-return) valves, and valves that perform an overpressure protection function. No attempt has been made to define the specific type of valve (i.e., as a globe, gate, butterfly, or other specific type of valve).
  - Pump symbols distinguish between centrifugal and positive displacement pumps and between types of pump drives (i.e., motor, turbine, or engine).

- Locations are identified in terms of plant location codes defined in Section 4 of this Sourcebook.
  - Location is indicated by shaded "zones" that are not intended to represent the actual room geometry.
  - Locations of discrete components represent the actual physical location of the component.
  - Piping locations between discrete components represent the plant areas through which the piping passes (i.e. including pipe tunnels and underground pipe runs).
  - Component locations that are not known are indicated by placing the components in an unshaded (white) zone.
  - The primary flow path in the system is highlighted (i.e., bold white line) in the location version of the fluid system drawings.

A1.2 Electrical System Drawings

The electric power system drawings focus on the Class 1E portions of the plant's electric power system. Separate drawings are provided for the AC and DC portions of the Class 1E system. There often are two versions of each electrical system drawing; a simplified system drawing, and a comparable drawing showing component locations. The drawing conventions used in the electrical system drawings are the following:

- Flow generally is top to bottom
  - In the AC power drawings, the interface with the switchyard and/or offsite grid is shown at the top of the drawing.
  - In the DC power drawings, the batteries and the interface with the AC power system are shown at the top of the drawing.
  - Vertical lines dominate and break horizontal lines.
- Component symbols used in the electrical system drawings are defined in Figure A-2.
- Locations are identified in terms of plant location codes defined in Section 4 of this Sourcebook.
  - Locations are indicated by shaded "zones" that are not intended to represent the actual room geometry.
  - Locations of discrete components represent the actual physical location of the component.
  - The electrical connections (i.e., cable runs) between discrete components, as shown on the electrical system drawings, DO NOT represent the actual cable routing in the plant.
  - Component locations that are not known are indicated by placing the discrete components in an unshaded (white) zone.

A2. SITE AND LAYOUT DRAWINGS

A2.1 Site Drawings

A general view of each reactor site and vicinity is presented along with a simplified site plan showing the arrangement of the major buildings, tanks, and other features of the site. The general view of the reactor site is obtained from ORNL-NSIC-55 (Ref. 1). The site drawings are approximately to scale, but should not be used to estimate distances on the site. As-built scale drawings should be consulted for this purpose.

Labels printed in bold uppercase correspond to the location codes defined in Section 4 and used in the component data listings and system drawings in Section 3. Some additional labels are included for information and are printed in lowercase type.

A2.2 Layout Drawings

Simplified building layout drawings are developed for the portions of the plant that contain components and systems that are described in Section 3 of this Sourcebook. Generally, the following buildings are included: reactor building, auxiliary building, fuel building, diesel building, and the intake structure or pumphouse. Layout drawings generally are not developed for other buildings.

Symbols used in the simplified layout drawings are defined in Figure A-3. Major rooms, stairways, elevators, and doorways are shown in the simplified layout drawings however, many interior walls have been omitted for clarity. The building layout drawings, are approximately to scale, should not be used to estimate room size or distances. As-built scale drawings for should be consulted his purpose.

Labels printed in uppercase bolded also correspond to the location codes defined in Section 4 and used in the component data listings and system drawings in Section 3. Some additional labels are included for information and are printed in lowercase type.

A3. APPENDIX A REFERENCES

1. Heddleson, F.A., "Design Data and Safety Features of Commercial Nuclear Power Plants.", ORNL-NSIC-55, Volumes 1 to 4, Oak Ridge National Laboratory, Nuclear Safety Information Center, December 1973 (Vol.1), January 1972 (Vol. 2), April 1974 (Vol. 3), and March 1975 (Vol. 4)

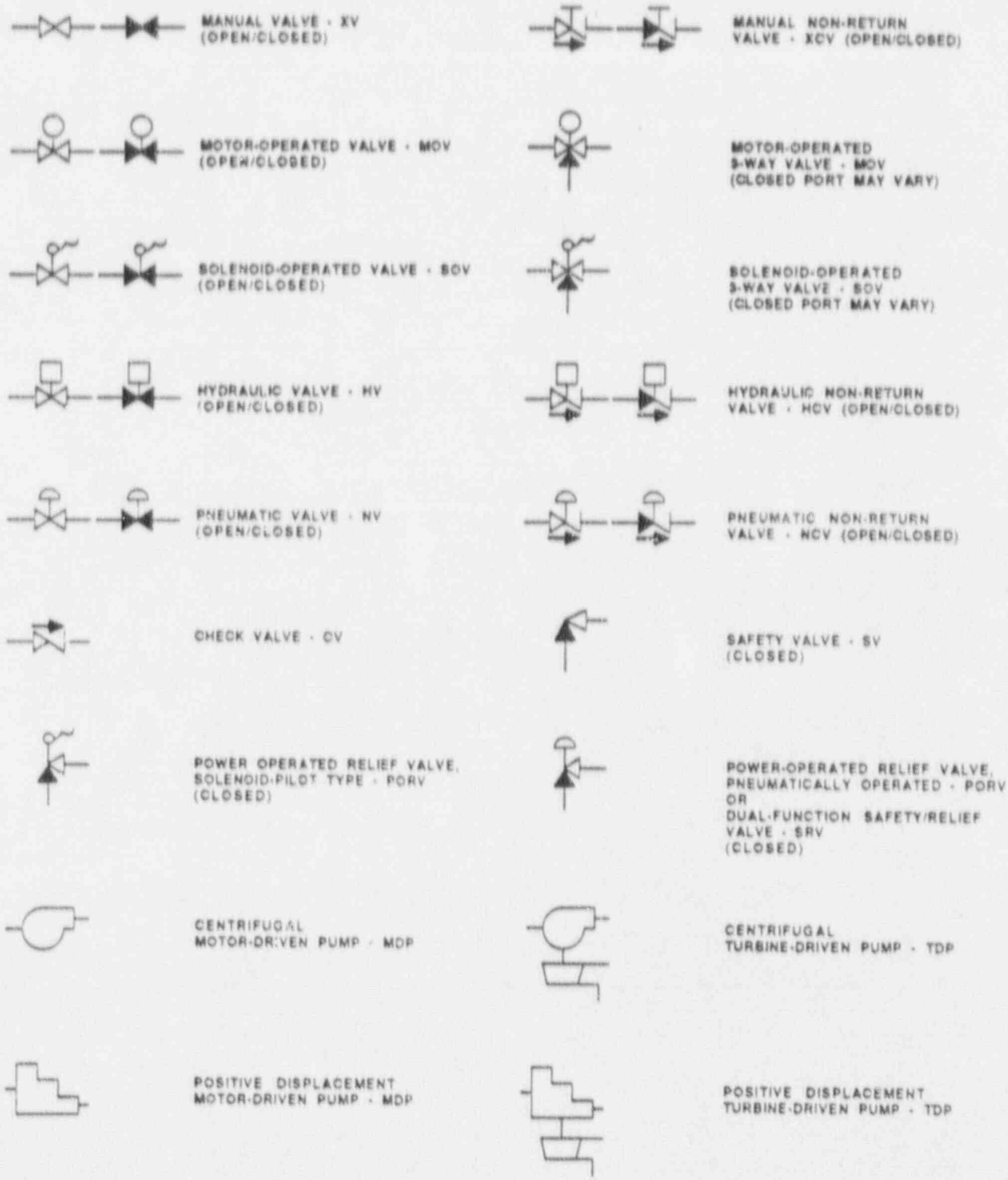


Figure A-1. Key To Symbols In Fluid System Drawings

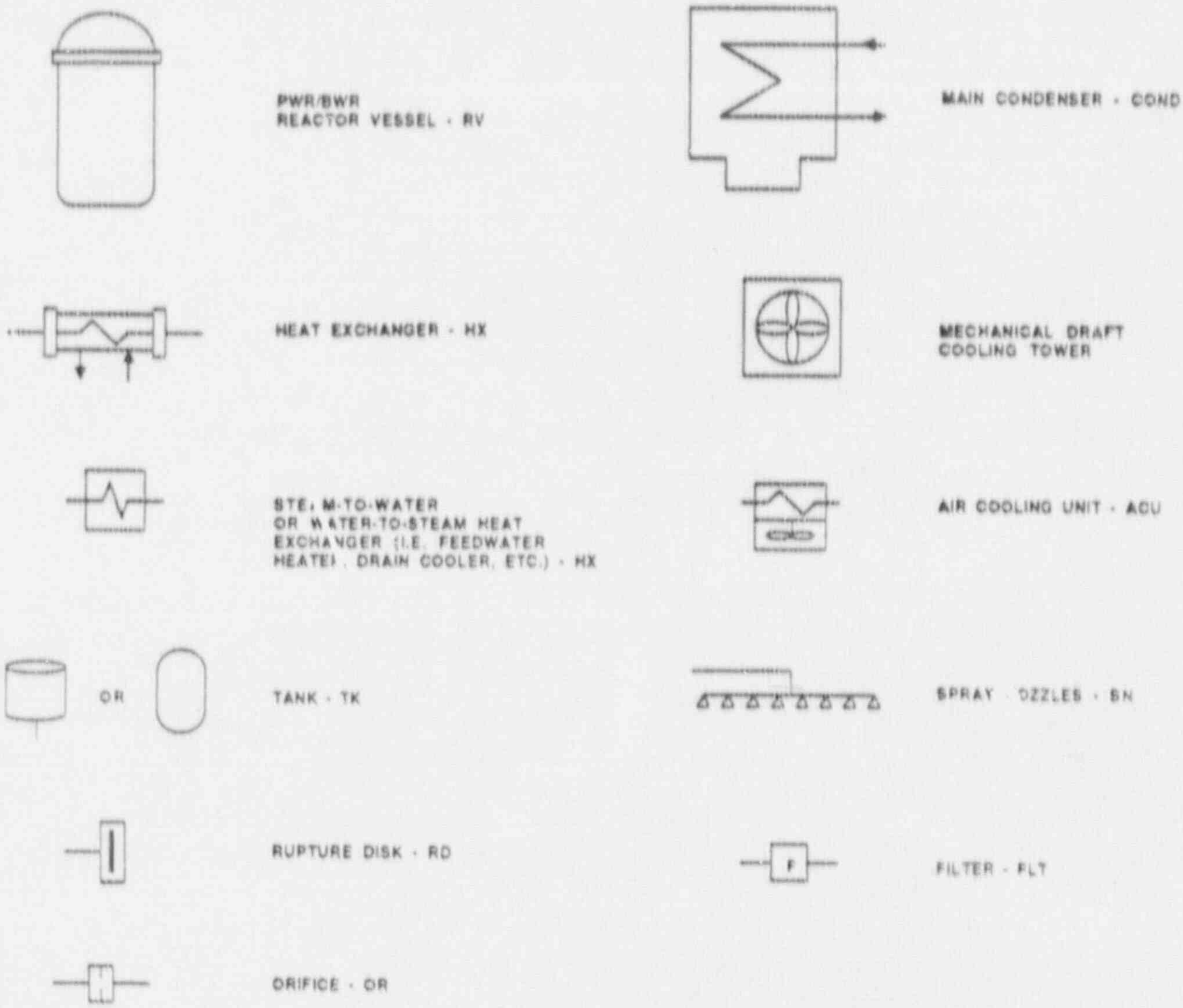


Figure A-1. Key To Symbols In Fluid System Drawings
(Continued)

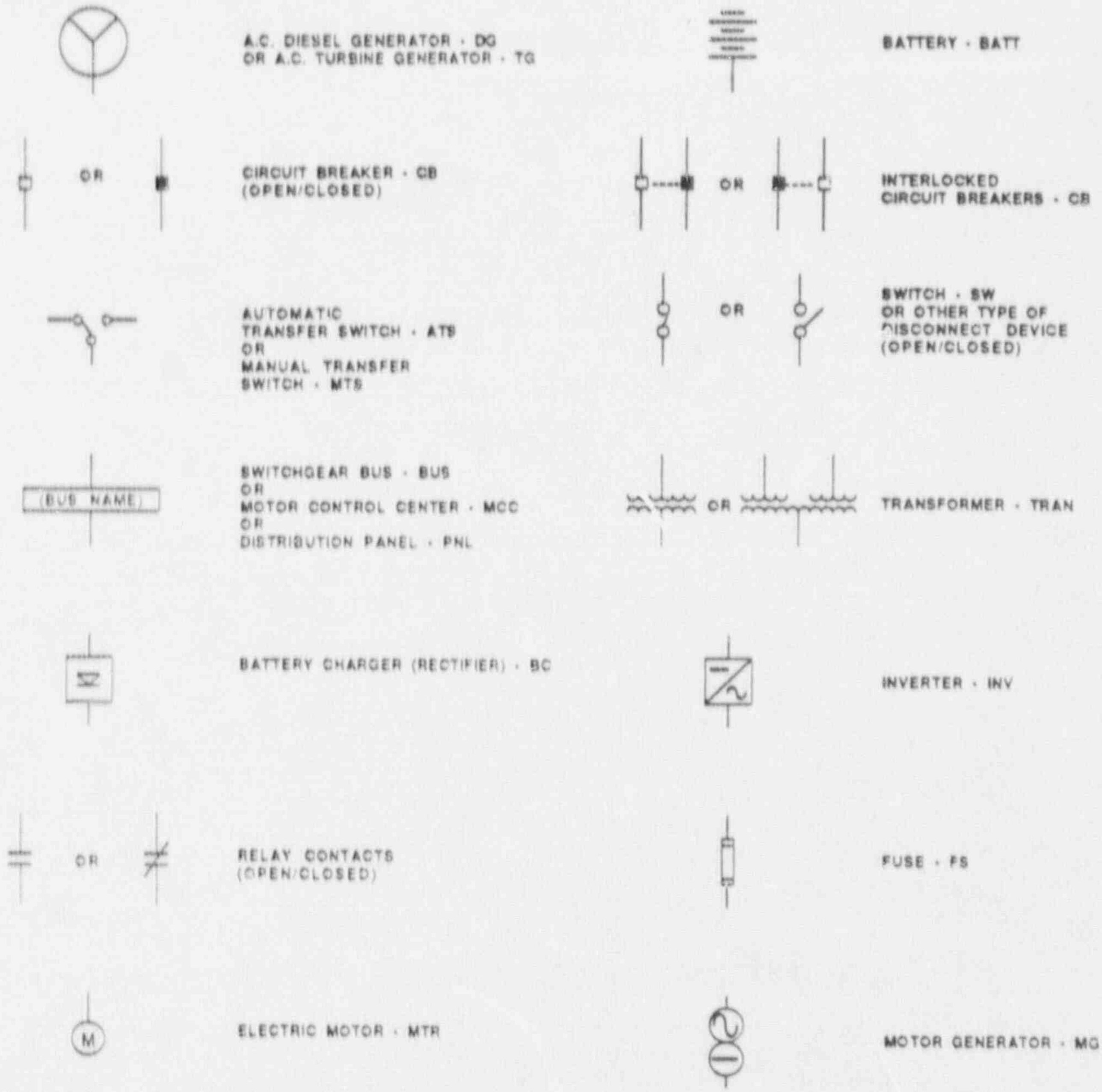


Figure A-2. Key To Symbols In Electrical System Drawings

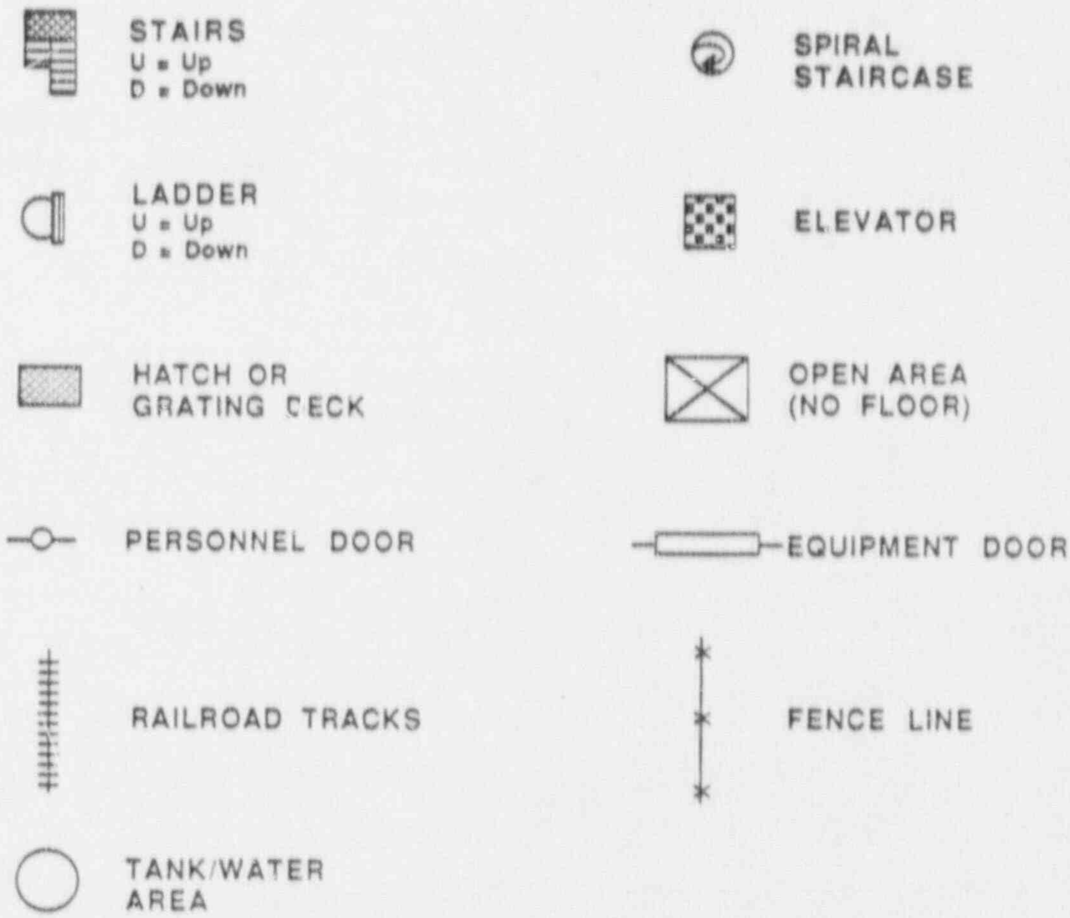


Figure A-3. Key To Symbols In Facility Layout Drawings

APPENDIX B
DEFINITION OF TERMS USED IN THE DATA TABLES

Terms appearing in the data tables in Sections 3 and 4 of this Sourcebook are defined as follows:

SYSTEM (also LOAD SYSTEM) - All components associated with a particular system description in the Sourcebook have the same system code in the data base. System codes used in this Sourcebook are the following:

Code	Definition
RCS	Reactor Coolant System
ECCS	Emergency Core Cooling Systems (including Core Spray, ADS and Isolation Condenser Systems)
CNS	Containment Spray System and Containment Spray Raw Water System
EP	Electric Power System
CRD	Control Rod Drive Hydraulic System

COMPONENT ID (also LOAD COMPONENT ID) - The component identification (ID) code in a data table matches the component ID that appears in the corresponding system drawing. The component ID generally begins with a system preface followed by a component number. The system preface is not necessarily the same as the system code described above. For component IDs, the system preface corresponds to what the plant calls the component (e.g. HPI, RHR). An example is HPI-730, denoting valve number 730 in the high pressure injection system, which is part of the ECCS. The component number is a contraction of the component number appearing in the plant piping and instrumentation drawings (P&IDs) and electrical one-line system drawings.

LOCATION (also COMPONENT LOCATION and POWER SOURCE LOCATION) - Refer to the location codes defined in Section 4.

COMPONENT TYPE (COMP TYPE) - Refer to Table B-1 for a list of component type codes.

POWER SOURCE - The component ID of the power source is listed in this field (see COMPONENT ID, above). In this data base, a "power source" for a particular component (i.e. a load or a distribution component) is the next higher electrical distribution or generating component in a distribution system. A single component may have more than one power source (i.e. a DC bus powered from a battery and a battery charger).

POWER SOURCE VOLTAGE (also VOLTAGE) - The voltage "seen" by a load of a power source is entered in this field. The downstream (output) voltage of a transformer, inverter, or battery charger is used.

EMERGENCY LOAD GROUP (EMERG LOAD GROUP) - AC and DC load groups (or electrical divisions) are defined as appropriate to the plant. Generally, AC load groups are identified as AC/A, AC/B, etc. The emergency load group for a third-of-a-kind load (i.e. a "swing" load) that can be powered from either of two AC load groups would be identified as AC/AB. DC load group follows similar naming conventions.

TABLE B-1. COMPONENT TYPE CODES

COMPONENT	COMP TYPE
VALVES:	
Motor-operated valve	MOV
Pneumatic (air-operated) valve	NV or AOV
Hydraulic valve	HV
Solenoid-operated valve	SOV
Manual valve	XV
Check valve	CV
Pneumatic non-return valve	NCV
Hydraulic non-return valve	HCV
Safety valve	SV
Dual function safety/relief valve	SRV
Power-operated relief valve (pneumatic or solenoid-operated)	PORV
PUMPS:	
Motor-driven pump (centrifugal or PD)	MDP
Turbine-driven pump (centrifugal of PD)	TDP
Diesel-driven pump (centrifugal of PD)	DDP
OTHER FLUID SYSTEM COMPONENTS:	
Reactor vessel	RV
Steam generator (U-tube or once-through)	SG
Heat exchanger (water-to-water HX, or water-to-air HX)	HX
Cooling tower	CT
Tank	TANK or TK
Sump	SUMP
Rupture disk	RD
Orifice	ORIF
Filter or strainer	FLT
Spray nozzle	SN
Heaters (i.e. pressurizer heaters)	HTR
VENTILATION SYSTEM COMPONENTS:	
Fan (motor-driven, any type)	FAN
Air cooling unit (air-to-water HX, usually including a fan)	ACU or FCU
Condensing (air-conditioning) unit	COND
EMERGENCY POWER SOURCES:	
Diesel generator	DG
Gas turbine generator	GT
Battery	BATT

TABLE B-1. COMPONENT TYPE CODES (Continued)

COMPONENT	COMP TYPE
ELECTRIC POWER DISTRIBUTION EQUIPMENT:	
Bus or switchgear	BUS
Motor control center	MCC
Distribution panel or cabinet	PNL or CAB
Transformer	TRAN or XFMR
Battery charger (rectifier)	BC or RECT
Inverter	INV
Uninterruptible power supply (a unit that may include battery, battery charger, and inverter)	UPS
Motor generator	MG
Circuit breaker	CB
Switch	SW
Automatic transfer switch	ATS
Manual transfer switch	MTS