- Fire protection booster pump
- Isolation valves in seismically designed standpipes
- Butterfly valves in fire pump test/relief return line
- Charcoal filter fire detection or system.
- Globe valve in the fire protection booster pump discharge test connection.

9.5.1.2 System Description

a. Fire Prevention

The plant fire protection system utilizes design aspects which employ separation criteria, noncombustible material, fire barrier divisions, fire rated penetrations for conduit, cable, piping and ductwork, as well as fire dampers. Fire barrier floors and walls, including all penetrations, have a design fire rating commensurate with the hazard. Physical separation or fire barriers are provided between redundant systems or equipment. In addition, fire stops are provided in long vertical cable tray runs to further ensure the non-propagational properties of the cables. These fire stops are provided where no other fire barriers exist. Electrical separation criteria between divisions is described in Subsection 8.3.1.4.

Plant equipment location and separation to limit fire-related damage is discussed in detail in the report "Seabrook Station Fire Protection System Evaluation and Comparison to Branch Technical Position 9.5-1, Appendix A" and "Fire Protection of Safe Shutdown Capability (10 CFR 50, Appendix R)."

b. Detection Systems

Fire detection devices are provided in areas which are judged to contain sufficient combustibles to present a fire hazard.

Fire detectors are installed consistent with the type of fire anticipated. A minimum of two detectors of any type are provided in each fire zone or fire area. Failure of one detector will not affect the operability of any other detector. The detectors are positioned within the zone or areas so that the flow of air or pressure differences will not affect proper operation of the detector.

The fire detection system contains supervisory panels to monitor the detector status. Fire detectors alarm at the control console in the control room to provide rapid identification of the location of any fire so that corrective action can be initiated.

Table 9.5-2 identifies the fire detector types for buildings and structures. Technical Requirement 12, located in the Technical Requirements Manual, lists the minimum number of operable detectors in each fire area.

Charcoal filter fire detection systems sense carbon monoxide to provide an early warning of a fire within the charcoal filter bed being monitored. Each charcoal filter located outside of containment is monitored by sample probes which are located both upstream and downstream of the charcoal beds. Control modules process signals from the sample probes and initiate alarms to the Fire Detection System upon detection of a high carbon monoxide concentration. Within containment, filter CAH-F-8 is monitored by a self-contained Sample System which draws a sample from the downstream side of the charcoal filter. The Sample System initiates an alarm to the Fire Detection System upon detection of a high carbon monoxide concentration. Alarms are initiated by the Fire Detection System on the fire control panel located in the control room.

c. Suppression Systems

Fire suppression capability is provided by installed systems which include water supply, pumps, valves and piping that supply hose stations, wet and preaction sprinklers, and deluge spray systems. Portable fire extinguishers are provided, where appropriate, and installed gas suppression systems are used where water would cause a hazard to equipment or personnel.

1. Water Supply

The fire protection system is shown schematically in Figures 9.5-1 through 9.5-8. The water supply for the plant fire protection system is obtained from two 500,000-gallon heated water storage tanks, of which 300,000 gallons in each tank is reserved for fire protection. Water for fire protection is supplied to the system by one 1500-gpm motor-driven centrifugal fire pump and one 1500-gpm diesel engine-driven centrifugal pump which provide the system design capacity. A second 1500-gpm diesel engine-driven centrifugal fire pump is provided as a spare. Each pump is capable of taking suction from either tank.

Two 25-gpm motor-driven centrifugal pressure maintenance jockey pumps maintain fire system pressure, and prevent unnecessary starting of the main fire pumps.

The fire pumps and jockey pumps are housed in a pumphouse adjacent to the fire tanks. The pumphouse is heated and ventilated to maintain suitable ambient conditions for pump

- Cable tunnels from Control Building to Primary Auxiliary Building
- Electrical penetration areas outside containment
- Primary Auxiliary Building at elevation 25'-0" and the electrical chase
- Diesel Generator Building fuel oil storage tank rooms and the fuel oil piping trenches.

Preaction sprinkler systems contain valve actuation provisions from fire detectors to charge the system with water, which will then discharge from any sprinkler head fused-open by a fire. Fire detection is annunciated at the control console in the control room and on a local control panel.

7. <u>Manually Operated Pre-Action Sprinkler Systems</u>

Manually operated sprinkler systems are provided for the following areas: (a) turbine generator bearings, (b) lube oil piping from bearings to guard pipe and (c) diesel generator rooms.

Fire detectors in the area annunciate a fire condition at the control console in the control room and on a local control panel.

8. <u>Standpipe Systems</u>

The Turbine Generator Building, the Administration and Service Building, Containment, Control Building, Primary Auxiliary Building, Fuel Storage Building, Waste Process Building, RHR equipment vault, Diesel Generator Building and emergency feedwater pump area are provided with fire hose stations at approximately 100-foot intervals around or within the building or stairwells to provide coverage, using 100 feet of hose. Each hose station consists of 1½ inch hose with Factory Mutual approved accessories.

The Turbine Generator Building hose stations are supplied from two looped building mains fed from two branch lines supplying the building from separate sections of the 12-inch yard fire main.

Two branch lines from separate sections of the yard fire main, backed up by a branch line from the safety-related plant service water system and booster pump, supply water to the standpipe hose stations in the RHR equipment vault, Primary Auxiliary Building, Fuel Storage Building, Diesel Generator

Building, Control Building, and emergency feedwater pump area. These systems are designed to be operational following an SSE.

To provide increased reliability for cooling safety-related components, a crossconnect from the Fire Protection and Demineralized Water systems to the PCCW System is included in the system design. This crossconnect can be used to provide cooling water to the charging pump lube oil coolers or provide emergency makeup water to safety-related portions of the PCCW System. This crossconnect is backed up by a seismic Category I Service Water System and booster pump makeup source.

Standpipes in safety-related areas are designed and supported as seismic Category I systems to prevent pipe failure and subsequent pipe whip. This feature also applies to deluge water spray and preaction sprinkler systems installed in safety-related areas.

Table 9.5-2 identifies the areas provided with hose stations.

9. Portable Fire Extinguishers

Portable fire extinguishers are located throughout the plant as the primary fire-fighting provisions in those areas determined to have negligible fire hazard, and as secondary defense in areas containing fixed fire protection systems. Portable fire extinguishers were selected on the basis of the most suitable type for the hazard present, with the radiological, metallurgical, physical and chemical compatibility of the extinguishing agents with plant components in mind. The types of portable extinguishers provided are pressurized water, Halon 1211, dry chemical and CO_2.

The extinguishers are conveniently located and conspicuously marked. Table 9.5-2 identifies the type of extinguishers provided in the plant.

10. Halon 1301 Fire Extinguishing Systems

A Halon 1301 fire extinguishing system is installed in the following nonsafety-related area:

- Main computer room (in Control Building)

Halon 1301 systems contain valve actuation provisions from fire detectors to discharge the gas for total flooding of the area experiencing a fire. Fire detection is annunciated at the control console in the control room and on a local control panel. The detection system also contains provisions to close all doors, and to close dampers in the air supply and ducts to the rooms, thus isolating the affected area from adjacent rooms.

j. The cooling tower switchgear rooms and the Service Water Pumphouse electrical control rooms are provided with ionization detectors. The areas are protected with portable extinguishers since the magnitude of a design-basis fire does not warrant fixed fire protection in these areas. In the unlikely event of a continuing fire, yard hydrants and hoses are available for backup suppression.

k. Areas not containing sufficient quantities of combustible materials to warrant installation of fire detectors are identified in the PSNH report "Seabrook Station Fire Protection Program Evaluation and Comparison to BTP APCSB 9.5-1, Appendix A."

l. All fire protection systems in areas containing safe shutdown equipment with the exception of the systems listed in Section 4.0, are preaction systems to preclude inadvertent system trip. Piping in the safe shutdown equipment areas is seismically supported. Drains are provided in these areas to convey any fire protection water away from the fire zone.

m. The status of all fire detection circuits is provided at the control console in the control room and on a local control panel. Alarm, detector malfunction, or detector removal are annunciated for operator action.

n. The plant communication system is available to alert personnel of a fire, its location, and remedial action required.

o. A failure modes and effects analysis for the systems and components is described in Table 9.5-3.

p. The seismically designed Lube Oil Collection System for the four reactor coolant system pumps has been designed with two collection tanks, with two pumps draining to each tank. Each of the two tanks has been sized to contain 125 percent of the oil inventory of one pump. A seismically designed dike has been provided around each tank. Each tank in combination with its associated dike has been sized to contain the entire inventory of two pumps. The tanks and the dikes have been located so that the excess oil does not present a fire hazard to any safety-related equipment. Additionally, there is no ignition source near the diked area.

9.5.1.4 Inspection and Testing Requirements

a. Preoperational Testing

1. Automatic systems (wet pipe sprinkler, preaction sprinkler, deluge water spray) are inspected and tested using the general guidelines of NFPA-13 and 15.

2. Yard piping, standpipes and hose stations (excluding the hoses) are hydrostatically tested to a pressure of 200 psig for a period of 2 hours using the general guidelines of NFPA-13 and 14. Fire hoses are tested and maintained using the general guidelines of NFPA-1962.

3. Fire pump field acceptance tests are performed using the general guidelines of NFPA-20.

4. Halon 1301 systems are tested and inspected using the general guidelines of NFPA-12A.

5. The presence of the NEIL representative is requested for the final inspection and tests of completed installations.

b. Surveillance

Inspections of fire protection equipment are made with filled out reports reviewed by Plant Engineering in accordance with the work control process and filed for examination by a NEIL representative.

9.5.1.5 Personnel Qualification and Training

a. Overall Responsibility for Nuclear Plant Fire Protection

The ultimate responsibility for the overall fire protection program rests with the Executive Vice President and Chief Nuclear Officer. The responsibility for the fire protection program has been assigned to the Director - Engineering. The program responsibilities have been delegated to:

1. Manager, Design Engineering (Electrical) - responsible for the technical adequacy of the Fire Protection Program and the licensing and design of fire protection systems and components. The corporate fire protection engineer is responsible for establishing and coordinating the implementation of the program under the Manager, Design Engineering (Electrical).

2. Station Director - responsible for the implementation of the fire protection program within the protected area, as well as the Fire Pumphouse, tanks and yard piping outside the protected area.

3. Director of Support Services - responsible for the implementation of the fire protection program for those areas outside the protected area not within the scope of the Station Director.

United Engineers & Constructors performed the design and selection of the fire protection systems for Seabrook Station, assisted by

Yankee Atomic Electric Company. United Engineers was responsible for the construction of the systems; Yankee Atomic Electric Company was responsible for the preoperational inspections and tests. The qualifications of those persons responsible for the re-evaluation of the fire protection program proposed for Seabrook Station against the guidelines provided in Appendix A to Branch Technical Position APCSB 9.5-1 and Appendix R to 10 CFR 50 are contained in the Seabrook Station Fire Protection System Evaluation and Comparison report.

The Station Director had been authorized to implement the fire protection program for the Executive Vice President and Chief Nuclear Officer using the station staff organization.

A general description of the station staff responsibilities for fire protection is as follows:

1. <u>Station Fire Protection Supervisor</u> - responsible for implementation for the station of the fire protection program, as directed by the Station Director.

2. <u>Manager Nuclear Training</u> - responsible for the fire fighting training program for employees.

3. <u>Station Fire Brigade</u> - responsible for fire fighting in the station.

In addition, the station staff is very active in fire protection and safety activities. Station personnel receive some training in manual fire fighting techniques, and are continually reminded of the importance and methods of fire prevention. Regular safety meetings are held for station personnel, and regular training sessions and drill sessions are held for the station fire brigade.

The station insurers, Nuclear Electric Insurance Limited, is considered to be an integral part of the station's fire prevention program. Frequent routine inspections of the station are performed by NEIL. Their comments and suggestions are carefully considered by the station staff, and changes are made in the fire protection program or in fire protection systems if they are needed.

b. <u>Fire Protection Training</u>

A training program and schedule have been established for Seabrook Station to develop and maintain an organization fully qualified to be responsible for the fire protection program at the station. The training program schedule is such that a fully trained and qualified fire brigade are available in the necessary numbers required to ensure the fire protection needed for safe and efficient operation of the facility. A continuing program is used for the training of replacement personnel and for any requalification training necessary to ensure that personnel remain proficient. The training program that each one can provide reliable offsite communications in all cases of emergencies. All systems can operate independently of each other. Failure of one system will not affect the others.

The telephone system has various offsite connections. These include the trunk lines to the public network, private network tie-lines and long distance carrier lines.

The microwave link has two transceivers, one active and one in hot standby mode. Its AC power source is backed up by the equipment's own DC batteries for continued operation for a minimum period of eight hours in case of loss of all AC power.

The radio transceivers can operate independently of all other systems. They are all backed up by their own batteries for continued operation in case of loss of all AC power.

9.5.2.4 Inspection and Testing Requirements

All communications systems are inspected and tested at the completion of the installation to ensure proper coverage and audibility under maximum plant noise levels during various operating conditions. Since the communications systems are used on a daily basis, periodic testing is not required.

9.5.3 Lighting System

The lighting system consists of the normal lighting system, the essential lighting system and the emergency lighting system.

9.5.3.1 Design Basis

a. The normal lighting system is designed to provide sufficient illumination to permit normal plant operation and maintenance functions.

b. The essential lighting system is designed to permit orderly plant shutdown following loss of offsite power. Reduced lighting is provided in control locations.

c. The emergency lighting system is designed in accordance with the requirements of 10 CFR 50, Appendix R, Section III.J, with deviations as noted in Subsection 9.5.3.2c. The emergency lighting system provides adequate lighting for continued operation in those areas of the plant that may need to be manned for safe shutdown operations and in access and egress routes to and from all such fire areas following the loss of the normal and essential lighting systems. Portions of the emergency lighting system, not associated with Appendix R requirements, provide egress lighting for the balance-of-plant areas.

d. The lighting systems are not Class 1E; however, in seismic Category I buildings the mounting of lighting transformers and panels and lighting fixtures is seismically analyzed to ensure that their failure could not damage safety-related equipment.

9.5.3.2 System Description

a. Normal Lighting System

The normal lighting system is fed from local 120/240 volt lighting distribution panels located in the various buildings. These local lighting panels are fed from locally mounted distribution transformers which are connected to the respective building motor control centers. Receptacle circuits are fed from the local lighting panels as required.

The 480-volt feeders to the local transformers are routed in the Plant Raceway System as Train A associated circuits. Branch circuits use aluminum sheath cable (ALS) throughout the plant except in the Guard House, Administration Building and the containment, where branch circuits use cable in electrical metallic tubing (EMT) and rigid steel conduit, respectively.

Incandescent lamps are used in the Containment Building and areas of the PAB and WPB where mercury is restricted. Aluminum fixtures are also restricted from the containment. However, high pressure sodium vapor lamps (containing a mercury-sodium amalgam) which have a double, water impermeable barrier, may be used in containment and the FSB during refueling outages, or if SFP fuel movement/inspection is needed during the fuel cycle. These high intensity lamps provide improved lighting with negligible possibility of contaminants reaching reactor water or components, when used in a temporary capacity as described here.

Normal lighting intensity levels, in general, are in accordance with the guidelines of the Illuminating Engineering Society handbook.

b. Essential Lighting System

The essential lighting system is generally fed from local 120/240 volt AC lighting distribution panels. These panels are fed from locally mounted distribution transformers which are connected to motor control centers. These motor control centers are energized from the diesel generator following a loss of offsite power.

The essential lighting system provides a reduced but adequate illumination for operation in the control room, the emergency switchgear rooms (including the remote shutdown locations), the diesel generator rooms, emergency feedwater pump room and the first aid area in the Administration Building. A minimum lighting level is provided in other selected areas for egress or minimum access.

capability of refilling the tanks is assured, and alternate means of venting can be provided if necessary.

The portion of the vent lines inside the Diesel Generator Building is designed to seismic Category I, Class 3, requirements. The portion of the vent lines outside the Diesel Generator Building is not protected from damage by tornado missiles; damage to this piping is unlikely to affect operation of the diesel generators. In the unlikely event that the storage tank vent lines are damaged, temporary provisions for venting can be provided during refilling.

Should all fill and vent connections external to the building be damaged, filling can still be accomplished via the spare 4" connection inside the building, and venting could be accomplished by unbolting the manway cover. Fire protection controls, in accordance with the Fire Protection Plan, would be implemented under these conditions.

There are no high or moderate energy lines or nonseismic Category I items located close to the fuel oil system whose failure could affect the operation of the fuel oil systems of both diesels.

The results of a failure modes and effects analysis are given in Table 9.5-5.

9.5.4.4 Tests and Inspections

During the preoperational test program, the diesel generator fuel oil system is tested for integrity in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 3 requirements. Preoperational tests are performed to verify proper system operation.

During plant operation, the diesel generator fuel oil system integrity and operability will be demonstrated during periodic tests of the diesel generator, as required by the Technical Specifications. The system will be inspected in accordance with ASME Code, Section XI requirements.

Tests of new fuel are performed per requirements stated in Regulatory Guide 1.137 except as follows:

- A clear and bright test per ASTM D4176-82 may be performed as an acceptable measure of the water and sediment test.
- An ASTM D2274-70 test for distillate fuel oil accelerated method is not performed.
- All ASTM D975 test results other than water and sediment, viscosity, flash point and API gravity are completed within 31 days instead of 2 weeks.

- The 10-year interval pressure test of the fuel oil system will be conducted in accordance with the inservice test program as specified in Technical Specification 4.0.5 in lieu of a 110% pressure test (authorized by License Amendment 54).

The monthly tests of the oil stored in the fuel storage tanks are

- Accumulated Water

- Total particulate per ASTM D2276-78

Required testing of both new and stored fuel oil is controlled by the diesel fuel oil testing program contained in Technical Requirements Manual.

Fuel oil samples are tested on a periodic basis for algal and bacterial growth. If they are detected, a suitable microbiocide additive, such as Biojar J. F. or Vancide 51, may be used.

Every 10 years, or earlier if necessary, the fuel oil will be removed and the tank cleaned using a sodium hypochlorite solution or equivalent, as required by Regulatory Guide 1.137.

9.5.4.5 Instrumentation

A safety-related level sensor at each day tank functions to operate the fuel oil transfer pump with separate level sensors at the tank, alarm low level in division wall. There are no high energy lines in either building whose failure could affect the cooling water piping. Adequate drainage is provided in these buildings to prevent flooding caused by a crack in the cooling water piping, or other adjacent moderate energy piping.

The buried piping has been coated and wrapped prior to installation with Tapecoat-20, applied in accordance with the manufacturer's recommendations and standard industry practice. An impressed current system for cathodic protection has also been provided.

When the diesel generator is operating, removal of heat from the cooling water is accomplished by circulating cooling water through the shell side of the main heat exchanger which is located in the Primary Auxiliary Building, with service water (sea water) circulating through the tubes.

The heat exchanger drain, vent, and relief valve discharge lines are connected to the floor and equipment drains downstream of normally closed valves. The lines are normally empty, and are connected to the floor and equipment drain system as a convenience to avoid puddles on the floor during system maintenance and startup operations. During normal plant operation, the valves in these lines are closed to maintain cooling water system integrity. A failure of these lines will not affect operation of either diesel generator. There is no flood hazard in the PAB due to the size (1" and 3/4") of these lines.

The engine-driven jacket coolant pump discharges cooling water to the engine cylinder walls and turbo-charger prior to being returned through the main heat exchanger. For low coolant water temperature, the coolant water bypasses the main heat exchanger through a temperature-controlled bypass valve and is directed to the suction side of the jacket coolant pump.

The engine-driven air cooler pump discharges cooling water to the air cooler, generator bearing and the lube oil cooler prior to being returned through the main heat exchanger. The air cooler pump is piped in parallel with the jacket coolant pump.

The Cooling Water System will dissipate the heat transferred to the diesel generator coolers in accordance with the following design parameters:

	Heat Removal Rate, Btu/hr.	Cooling Water Flow, gpm	Temp. Diff.	Design Margin
Jacket Water Cooler	15,550,400	1800	17°F	55%
Lube Oil Cooler	1,991,800	1060	4°F	N/A
Engine Air Cooler	5,395,135	1060	10°F	N/A

The noncode (manufacturer's standard) motor-driven auxiliary coolant pump is located off-skid and is piped in parallel with both the jacket coolant pump and the air cooler pump with Class 3 piping and isolation valves. The auxiliary coolant pump starts automatically in the event of failure of either or both of the engine-driven pumps. The auxiliary coolant pump can also be started manually by the operator.

The valves connecting the auxiliary coolant pump to the coolant piping for jacket coolers and air coolers are pneumatic cylinder operated. The valves open automatically on low coolant pressure. For the jacket coolers, valves V11 and V12 are opened on low outlet coolant pressure. For the air coolers, valves V9 and V13 are opened on low coolant pressure. Ref. Dwg. 503486 (Updated FSAR Section 1.7). The supply air to the valve operators is controlled by solenoid valves which are activated by pressure switches. The air is supplied through a reducing valve connecting to the on-skid air start piping. A failure of these valves or air supply will not affect operation of the engine-driven pumps or the cooling capability of the system.

When the diesel generator is not operating, the engine block is maintained in a warmed condition to provide reliable starting. This is accomplished by maintaining the cooling water at a temperature recommended by the manufacturer. The cooling water is pumped through the jacket coolant heater and back into the jacket by the jacket coolant standby circulating pump. In the standby condition, the only portion of the Cooling Water System that requires operation is the circulating pump, heater, and associated piping to and from the engine jacket.

When the diesel generator is operating, the jacket coolant heater and jacket coolant standby circulating pump will stop operating. The three-way temperature control valves in the on-skid piping will automatically mix heated cooling water from the engine with cold cooling water from the heat exchangers and associated piping, to prevent thermal shock on the engine during startup.

The cooling water temperature is controlled between 170°F and 180°F at the engine outlet. The control valves admit cold water when the engine outlet temperature reaches 170°F and thereafter admit sufficient cold water to maintain that setpoint. The engine is capable of operating for three minutes without any flow of service water to the heat exchanger.

A corrosion inhibitor and antifreeze compound is mixed with demineralized water in accordance with manufacturer's specifications. To maintain the proper quantity of water within the system, an expansion tank is located at the highest point in the Cooling Water System. The cooling water expansion tank has a design capacity of 290 gallons and is located 46 feet above the engine skid at elevation 67'-6". This location assures that the pump NPSH requirements are maintained. Pump shaft seals, valve stem packing and other components are checked for zero leakage during routine engine testing. The expansion tank capacity can allow a leak of 1.7 gph for seven days without loss of contents. The tank is replenished manually from the demineralized

The prelube and filter pump, P-116, may be test run by positioning the control switch, mounted at the motor control center, to "Run." This switch is normally key-locked in the "auto" position.

The schedule and scope of instrumentation calibration and testing is in accordance with applicable requirements of the Technical Specification and other recommendations of the vendor's technical manuals. Calibration frequencies will generally be on a refueling interval or as relative to the importance of the specific instrument.

Upon receiving system alarms, the operators will take corrective action as required by the particular Alarm Response Procedure.

9.5.7.5 Instrumentation

The motor-driven prelube and filter pump is designed to run continuously. When the pump is running, lube oil temperature is monitored by temperature switch TS-OHT (see Figure 9.5-13). The lube oil temperature is maintained at a temperature recommended by the manufacturer. This assures prelubrication of the engine with warm lube oil.

When the diesel generator is running, lube oil is pumped through a water cooled heat exchanger E-41. Temperature control valve, V29, determines the volume of oil that is directed through the heat exchanger. The remainder is bypassed back to the engine header. Lube oil temperature is monitored at the lube oil pump outlet header and high lube oil temperature is alarmed locally and at the computer. High lube oil temperature is also an input to the engine trouble shutdown logic.

Normally, when the diesel generator is in operation, lube oil is pumped by the engine-driven pump. Lube oil pressure is monitored by four pressure switches PS-OPL1, PS-OPL2, PS-OPL3, and PS-OPL4. PS-OPL1 will close at 70 psi decreasing, and reset at 75 psi increasing. PS-OPL2 will close at 65 psi decreasing and reset at 70 psi increasing; PS-OPL3 and PS-OPL4 will close at 60 psi decreasing and reset at 65 psi increasing.

With diesel generator running at greater than 375 rpm and the alarm permit logic satisfied, PS-OPL1 or PS-OPL2 will start the auxiliary lube oil pump. "Auxiliary Lube Oil Pump Running" is alarmed at this local control panel and at the computer. If the alarm permit logic is satisfied, the detection of low pressure by any one of the four pressure switches will be alarmed locally and at the computer. Two out of three low pressure signals from PS-OPL2, PS-OPL3, and PS-OPL4 will result in an engine trouble shutdown.

High level in the rocker arm lube oil reservoir is alarmed locally by LS-KLHA and, provided the engine speed is greater than 375 rpm, is an input to the DG system trouble alarm at the computer.

Low pressure at the discharge of the rocker arm lube oil filter is alarmed locally by PS-KPLA and on the computer, provided the engine speed is greater than 375 rpm.

Level in the engine sump is monitored by level switch LS-OLLA, and low level is alarmed locally and at the computer. A pressure switch is provided to alarm a loss of vacuum.

Lube oil temperature in the engine sump is monitored by temperature switch TS-OTLA. Low temperature is alarmed locally and at the computer.

Differential pressure across the lube oil strainer is monitored by pressure differential switch PDS-OSHD. High differential pressure is alarmed locally and is an input to the DG system trouble alarm at the computer.

Lube oil high temperature and low lube oil pressure trips are provided. In accordance with BTP ICSB-17, the lube oil high temperature trip is bypassed under accident conditions. The lube oil low pressure trip is not bypassed under accident conditions because the required coincident logic is provided in the low lube oil trip circuit.

9.5.8 Diesel Generator Combustion Air Intake and Exhaust System

9.5.8.1 Design Basis

The diesel generator combustion air intake and exhaust system is capable of supplying adequate combustion air and disposing of resultant exhaust products to permit continuous operation of the diesel engine.

The system design is based on the following requirements:

a. Each diesel engine is provided with an independent air intake and exhaust system.

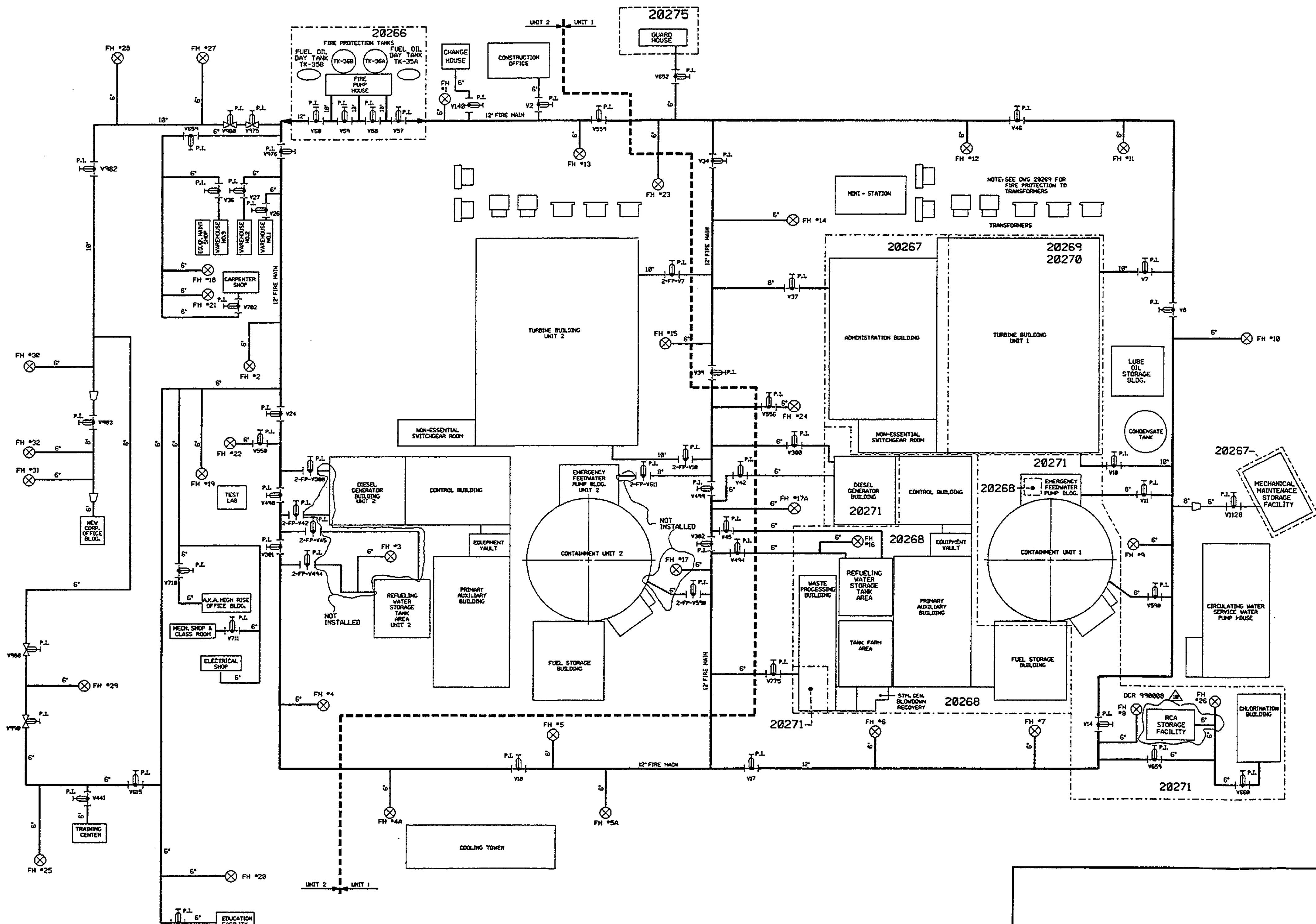
b. Components of the system are located in a seismic Category I structure which provides protection from external missiles, natural phenomena, and contaminating substances.

c. The consequences of a single active failure in the system will not result in the loss of function of more than one diesel generator.

d. The system components are located to avoid the effects of pipe whip or jet impingement forces resulting from high and moderate energy pipe breaks.

REV. 07

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2



FOR YARD PIPING REFER TO FP-20274

NOTES:
1. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION ARE PREFIXED "FP", UNLESS NOTED OTHERWISE.
2. WORK THIS DRAWING WITH DRAWINGS 20266 THRU 20275
3. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS
4. △ INDICATES REVISION LEVEL
5. DELETED
6. N.I. INDICATES PIPING AND EQUIPMENT NOT INSTALLED.
7. F.P. STRAINERS ARE LOCATED INSIDE BUILDING

PID- 1-FP-B20264

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	FIRE PROTECTION OVERVIEW	
	1-FP-B20264	FIGURE 9.5-1

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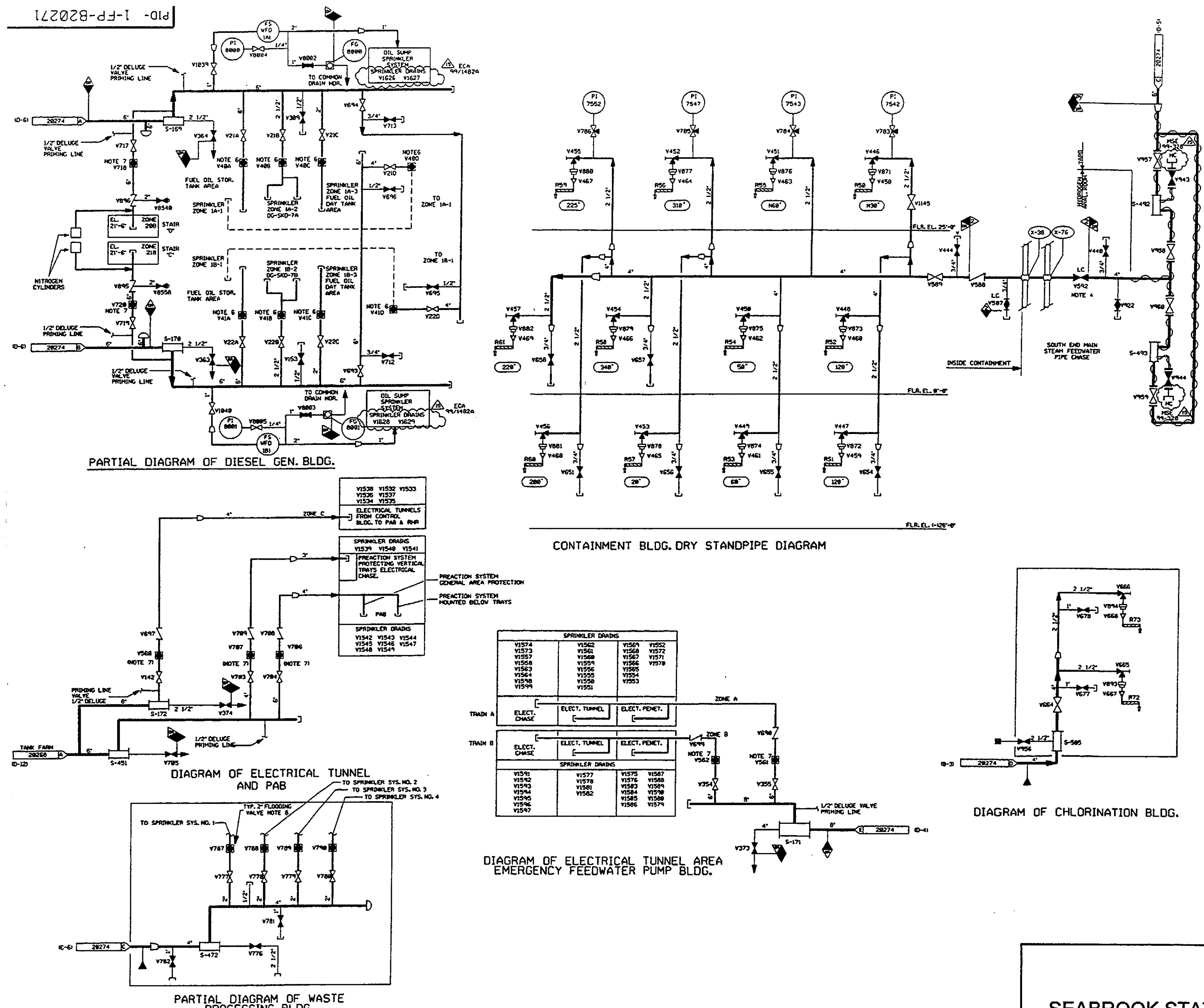
REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

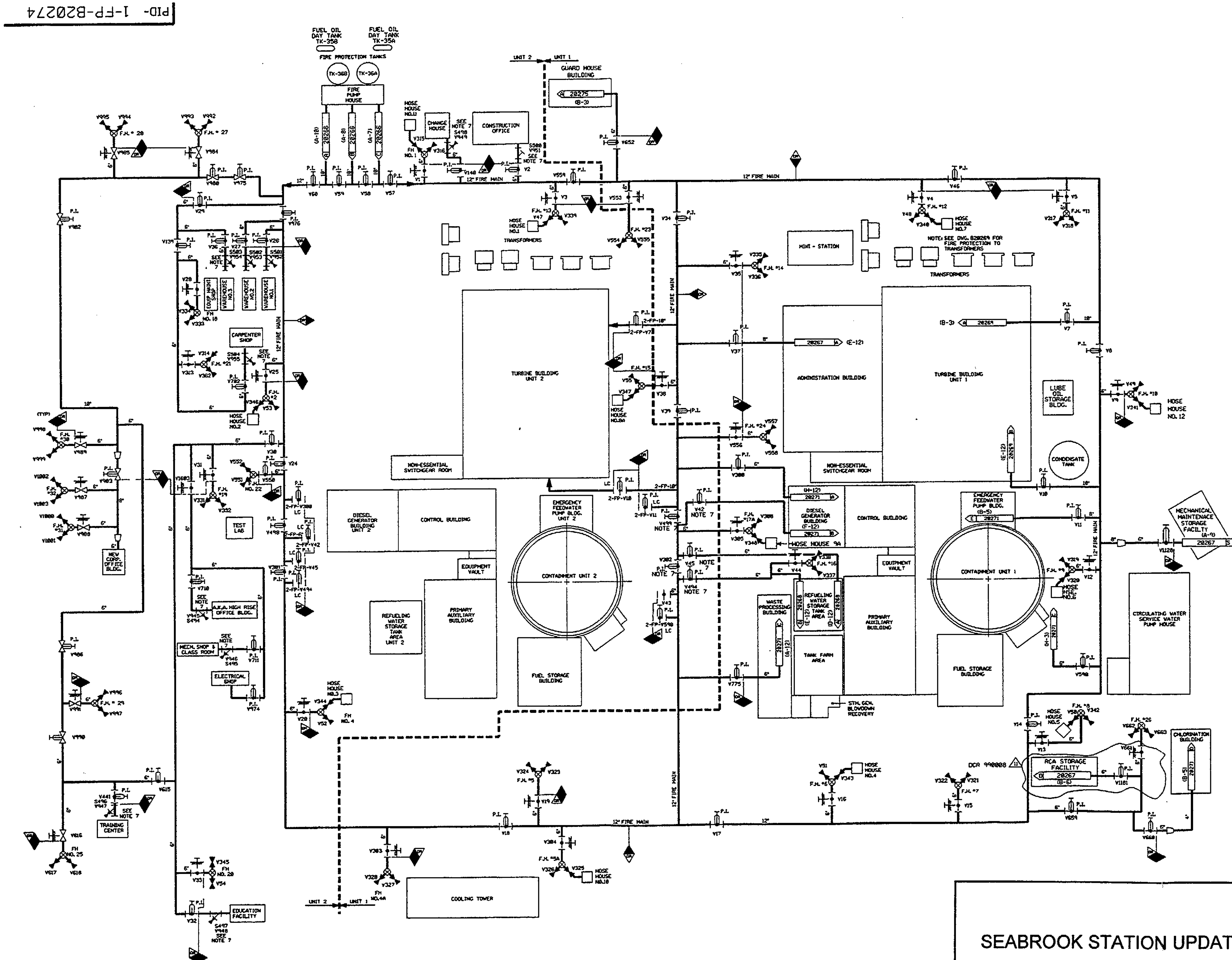
1. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX FP, UNLESS NOTED OTHERWISE.
2. DELETED
3. DELETED
4. STANDPIPE SYSTEM WITHIN CONTAINMENT SHALL BE DRAINED PRIOR TO OPERATION OF UNIT. VALVE MUST BE CLOSED IN MODE 1,2,3,4 AND CAN BE OPENED IN MODE 5.
5. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS, UNLESS NOTED OTHERWISE.
6. SEE DETAIL A, DWG. 20272.
7. SEE DETAIL B, DWG. 20272.
8. SEE DETAIL C, DWG. 20272.
9. △ INDICATES REVISION LEVEL.
10. DELETED
11. DELETED
12. DELETED
13. DELETED
14. DELETED

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	FIRE PROTECTION DETAILS
	1-FP-B20271 / FIGURE 9.5-2

REV. 07



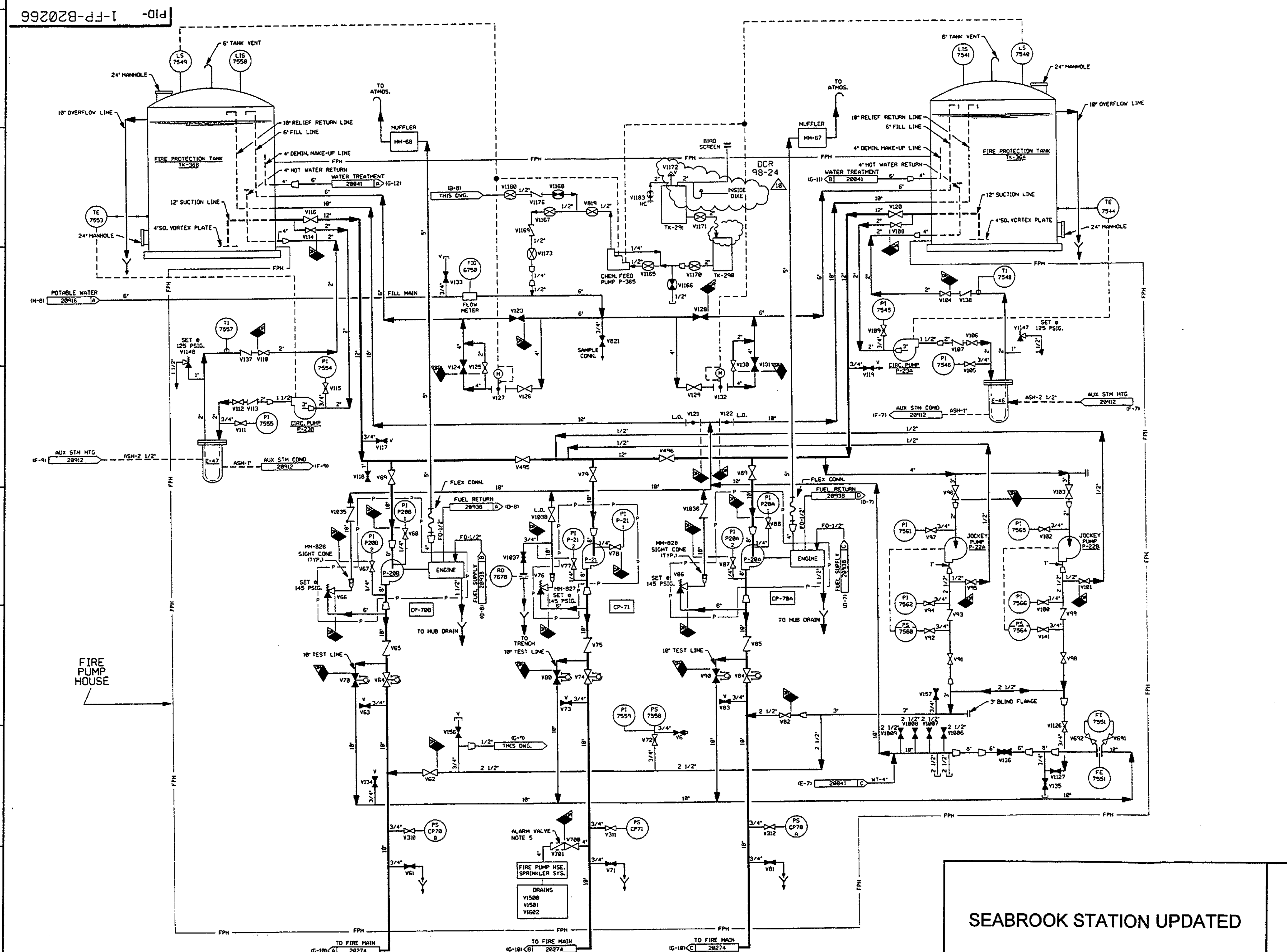
FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION ARE PREFIXED "FP". UNLESS NOTED OTHERWISE.
2. WORK THIS DRAWING WITH DRAWINGS 20264, 20266 THRU 20275.
3. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS
4. △ INDICATES REVISION LEVEL
5. N.I. INDICATES PIPING AND EQUIPMENT NOT INSTALLED.
6. F.P. STRAINERS ARE LOCATED INSIDE BUILDING, EXCEPT CONTAINMENT.
7. THESE DEVICES ARE PART OF UNIT 1.

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	FIRE PROTECTION YARD PIPING
	1-FP-B20274 / FIGURE 9.5-4

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID-LEGEND 1 AND P&ID-LEGEND 2.

NOTES:

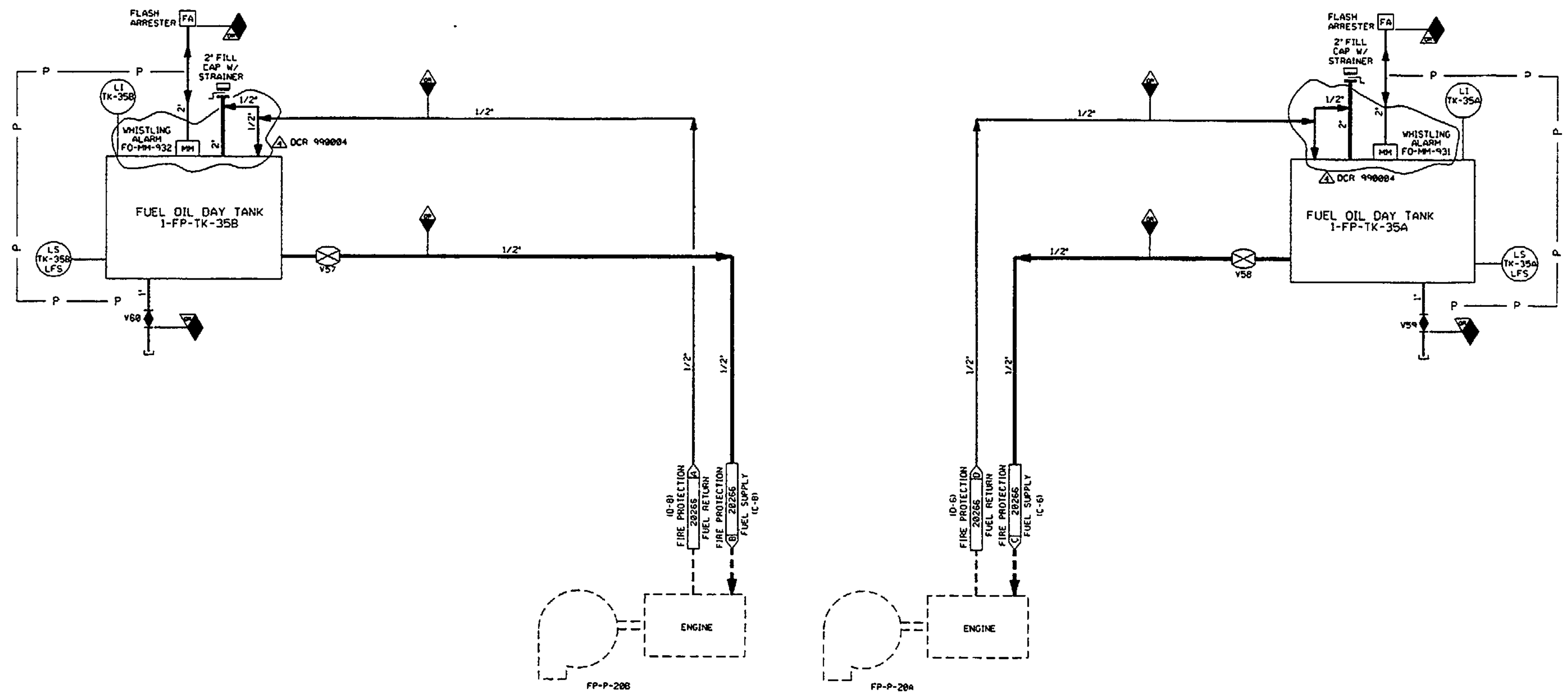
1. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION HAVE SYS. PREFIX FP, UNLESS NOTED OTHERWISE.
2. WORK THIS DRAWING WITH DRAWINGS 20264 AND 20267 THRU 20272.
3. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS, UNLESS NOTED OTHERWISE.
4. DELETED
5. SEE DETAIL "D" DWG. 20272.
6. △ INDICATES REVISION LEVEL.
7. DELETED
8. -OR- CALIBRATION ONLY
9. VENTS & DRAIN TAP OR BOUNDRY ARE AT THE DOWN STREAM END OF THE OUTER ISOLATION VALVE.

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	FIRE PROTECTION FIRE PUMP HOUSE DETAIL
1-FP-B20266	FIGURE 9.5-5

PID- 1-FO-B20938

REV. 07

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2



NOTES:

1. THE FO IS A ONE DWG. SYSTEM AND AN OVERVIEW IS NOT REQUIRED.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYTEM PREFIX FO UNLESS NOTED.
3. ALL LINES, EQUIPMENT, COMPONENTS, AND INSTRUMENTS ARE SAFETY CLASS NNS UNLESS NOTED.

9763-F-604069	16	YARD & FIRE PUMPHOUSE FIRE PROTECTION
REF. DRAWING NO.	REV	TITLE

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT

FUEL OIL SYSTEM

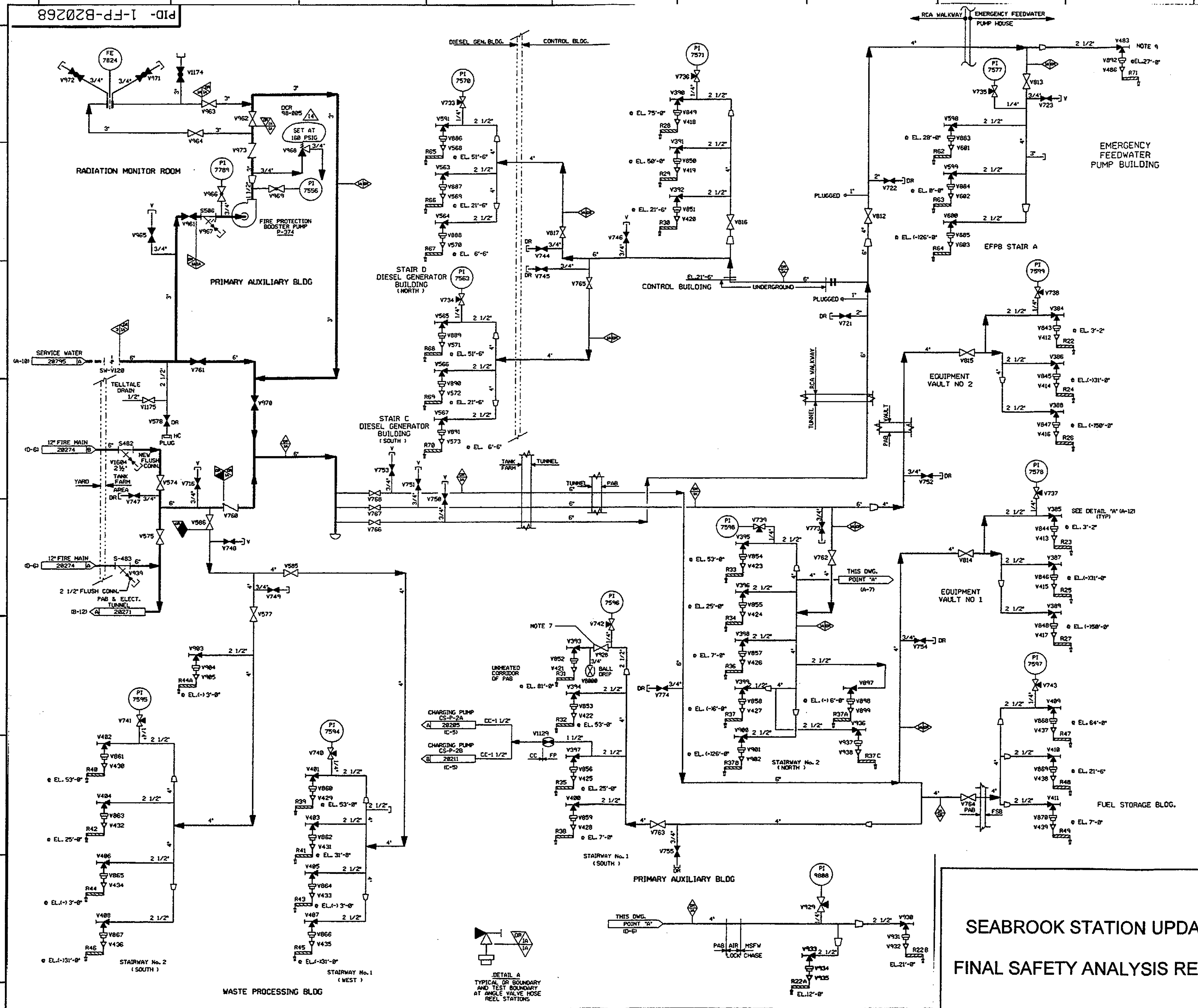
1-FO-B20938

FIGURE 9.5-6

04/07/00 REV. 04

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

REV. 07



NOTES:

1. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION ARE PREFIXED "FP". UNLESS NOTED OTHERWISE.
2. WORK THIS DRAWING WITH DRAWINGS 20264, 20266, 20267 & 20269 THRU 20275.
3. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS. UNLESS NOTED OTHERWISE.
4. △ INDICATES REVISION LEVEL.
5. DELETED
6. DELETED
7. VALVE 928 TO BE NORMALLY CLOSED DURING PERIODS OF FREEZING TEMP. HOSE RACK 31 IS DRY STANDPIPE DURING THESE PERIODS.
8. VENTS, DRAINS & INSTRUMENT TAPS OR BOUNDARY ARE AT THE DOWNSTREAM END OF THE OUTER ISOLATION VALVE.
9. CONNECTION FOR EMERGENCY MAKE-UP TO CST USING TEMPORARY HOSE (CO-20426 G-7)

REF. DRAWING NO.	REV	TITLE
9763-F-604146	10	FIRE PROTECTION DIAGRAM - STAND PIPE

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FINAL SAFETY ANALYSIS REPORT

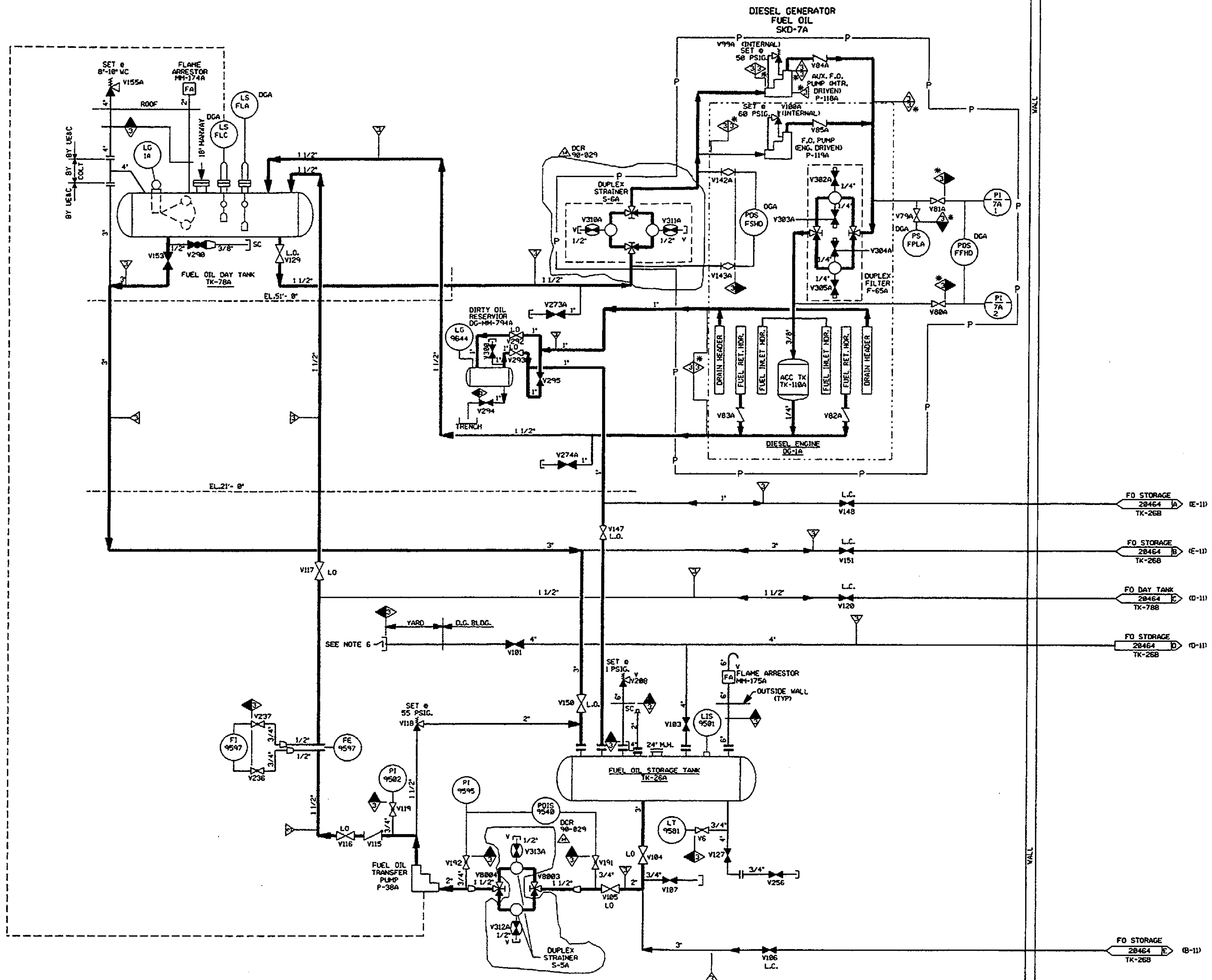
FIRE PROTECTION STANDPIPE DETAIL

1-FP-B20268

FIGURE 9.5-8

PID- 1-DG-B20459

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20456, 20457, 20458, 20460, 20461, 20462 & 20464.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION HAVE SYSTEM PREFIX DG, UNLESS NOTED OTHERWISE.
3. DELETED.
4. DELETED.
5. △ -INDICATES REVISION LEVEL.
6. NPT QUICK DISCONNECT FITTING PROVIDED BY OWNER FOR DIESEL FUEL CONNECTION.
7. VENT, DRAINS AND TEST CONN. SAFETY CLASS AND CODE BREAKS ARE AT THE DOWNSTREAM END OF THE OUTER ISOLATION VALVE PER 1-NHY-805111, UNLESS NOTED OTHERWISE.

REF. DRAWING NO.	REV	TITLE
FP 20669	9	EMERG. DIESEL GEN. JACKET WTR. EXP. TANK
FP 21070	9	EMERG. DIESEL GEN. FUEL OIL DAY TANK
9763-F-202102	16	DG FUEL & LUBE OIL P&I DIAGRAM

SEABROOK STATION UPDATED

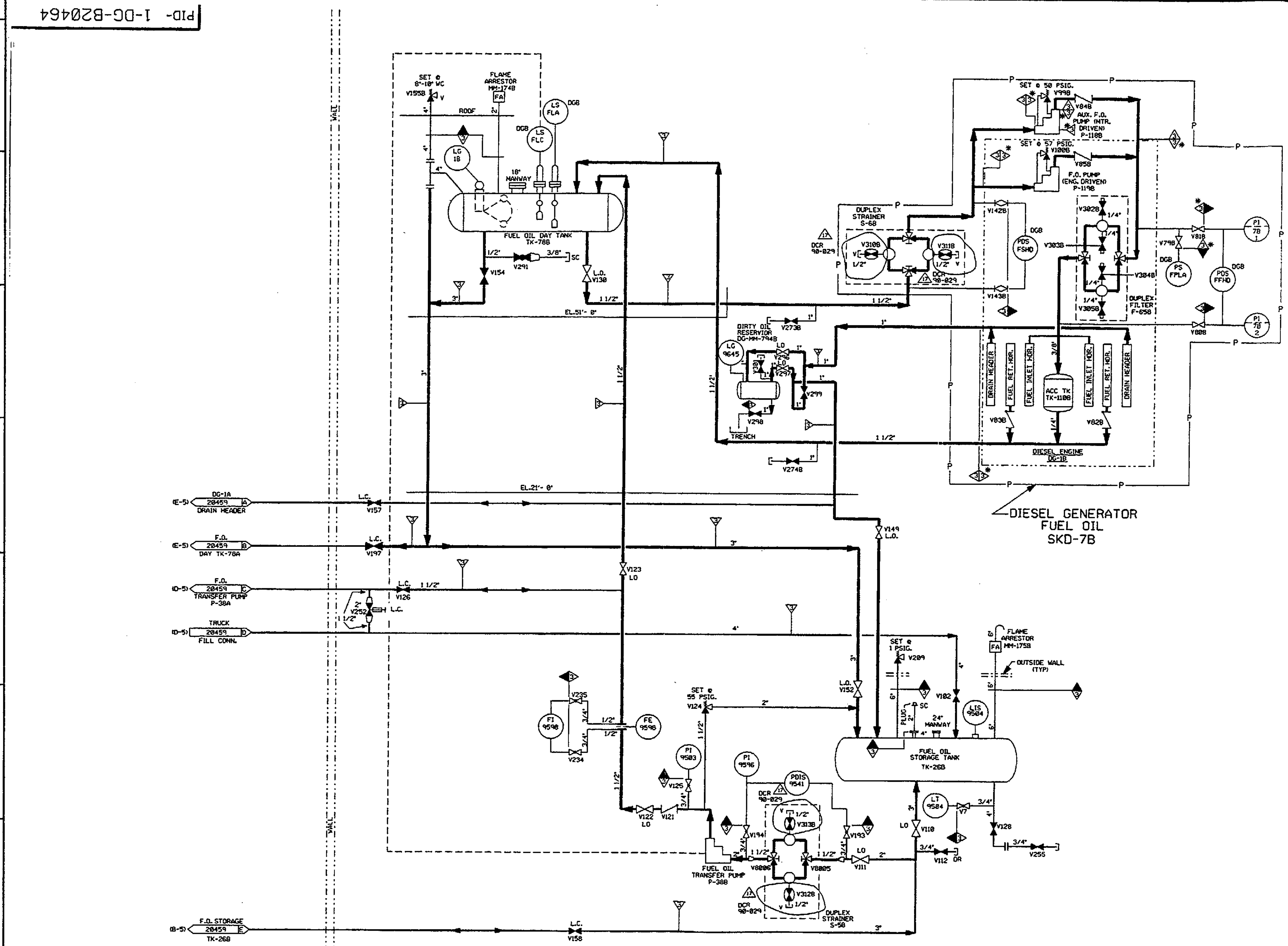
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR
FUEL OIL SYSTEM TRAIN 'A'
DETAIL

1-DG-B20459

FIGURE 9.5-10 SH 1

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID-LEGEND 1 AND P&ID LEGEND 2.

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20456, 20457, 20459, 20463, 20465, 20466 & 20467.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION HAVE SYSTEM PREFIX DG, UNLESS NOTED OTHERWISE.
3. DELETED.
4. DELETED
5. △ INDICATES REVISION LEVEL.
6. VENT, DRAINS AND TEST CONN, SAFETY CLASS AND CODE BREAKS ARE AT THE DOWNSTREAM END OF THE OUTER ISOLATION VALVE PER 1-NHY-805111, UNLESS NOTED OTHERWISE.

REF. DRAWING NO.	REV	TITLE
FP 21070	9	EMERG DIESEL GEN FUEL OIL DAY TANK
FP 20669	9	EMERG DIESEL GEN JACKET WTR EXP TANK
9763-F-202102	16	D.G. FUEL & LUBE OIL P&I DIAGRAM

SEABROOK STATION UPDATED

FINAL SAFETY ANALYSIS REPORT

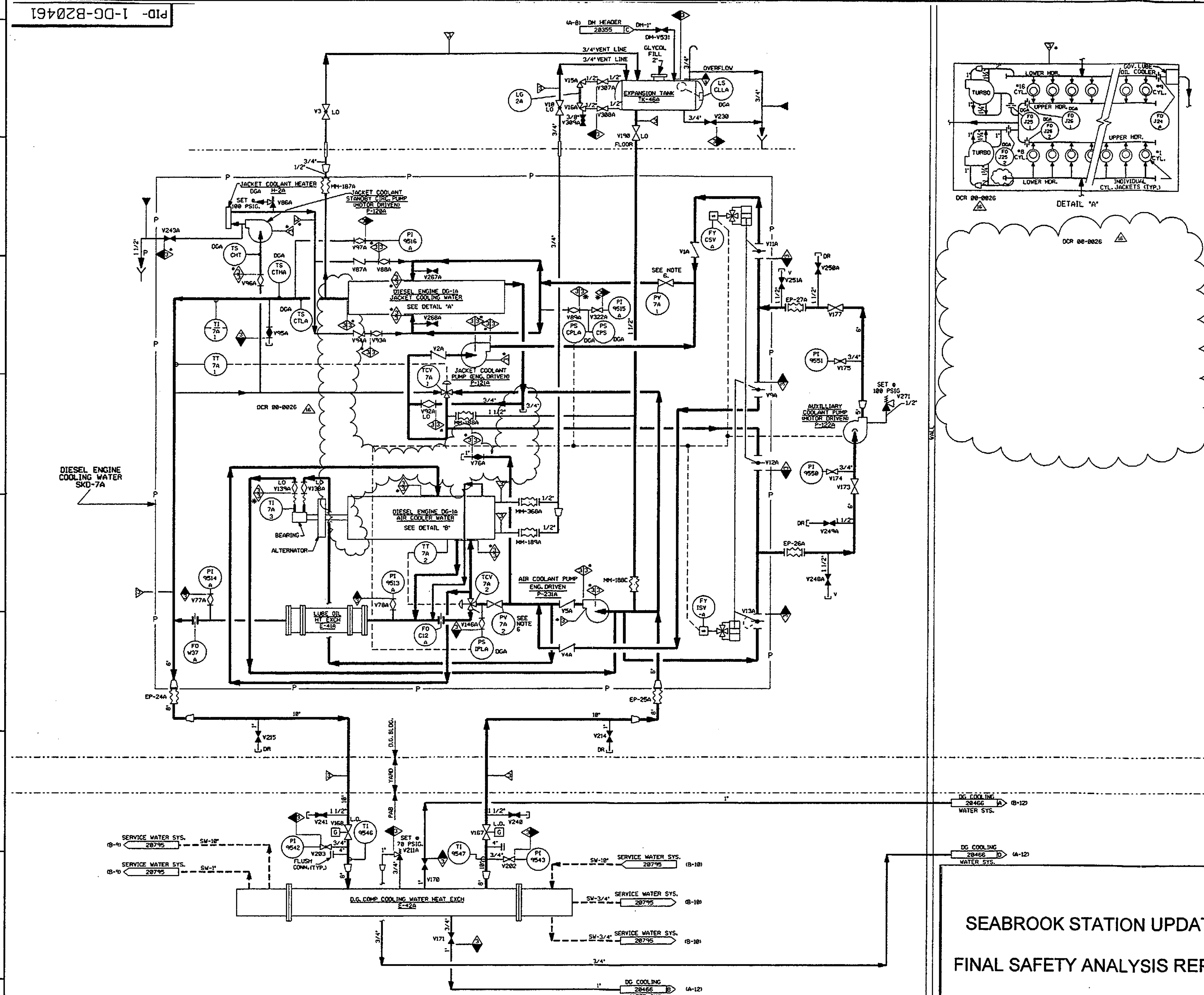
DIESEL GENERATOR
FUEL OIL SYSTEM TRAIN 'B'
DETAIL

1-DG-B20464

FIGURE 9.5-10 SH 2

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

REV. 07



NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20456, 20457, 20458, 20459, 20460, 20462 & 20466.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION HAVE SYSTEM PREFIX DG. UNLESS NOTED OTHERWISE.
3. DELETED
4. VENT & DRAIN CODE AND SAFETY CLASS BREAKS ARE AT THE DOWNSTREAM END OF THE OUTER ISOLATION VALVE PER 1-NHY-805111, UNLESS NOTED OTHERWISE.
5. △ INDICATES REVISION LEVEL.
6. THIS VALVE HAS BEEN STAKED IN A PRE-DETERMINED THROTTLED POSITION.

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

FP 21175	4	DG COOLANT BACKUP JW PMP. & MTR. OUTLINE
FP 20699	9	EMERG. DG SKID GEN. ARRGM'NT
FP 21069	7	EFWP PUMP ASSEMBLY
9763-F-202103	14	DIESEL GENERATOR COOLING WATER - P&ID
REF. DRAWING NO.	REV	TITLE

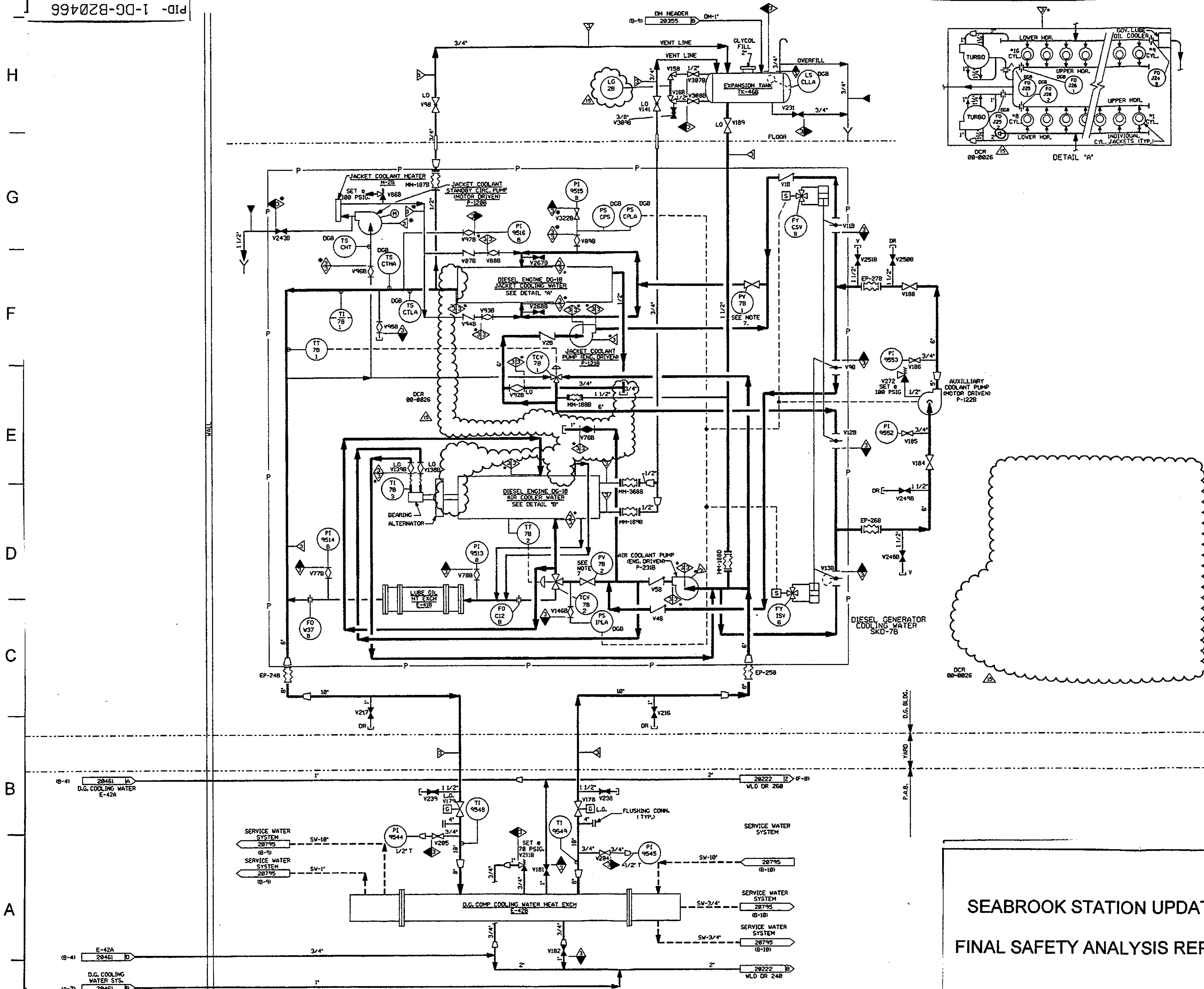
SEABROOK STATION UPDATED

FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR COOLING WATER SYSTEM TRAIN 'A' DETAIL

1-DG-B20461

FIGURE 9.5-11 SH 1



FOR P&ID REFERENCE DRAWINGS. SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

REV. 07

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20456, 20457, 20461, 20463, 20464, 20465 & 20467.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION HAVE SYSTEM PREFIX DG, UNLESS NOTED OTHERWISE.
3. DELETED
4. VENT, DRAIN & TEST CONN. CODE AND SAFETY CLASS BREAKS ARE AT THE DOWN STREAM END OF THE OUTER ISOLATION VALVE PER 1-NHY-805111, UNLESS NOTED OTHERWISE.
5. △ INDICATES REVISION LEVEL.
6. DELETED
7. THIS VALVE HAS BEEN STAKED IN A PRE-DETERMINED THROTTLED POSITION.

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

9763-F-202103	15	DIESEL GENERATOR COOLING WATER P&ID
REF. DRAWING NO.	REV	TITLE

SEABROOK STATION UPDATED

FINAL SAFETY ANALYSIS REPORT

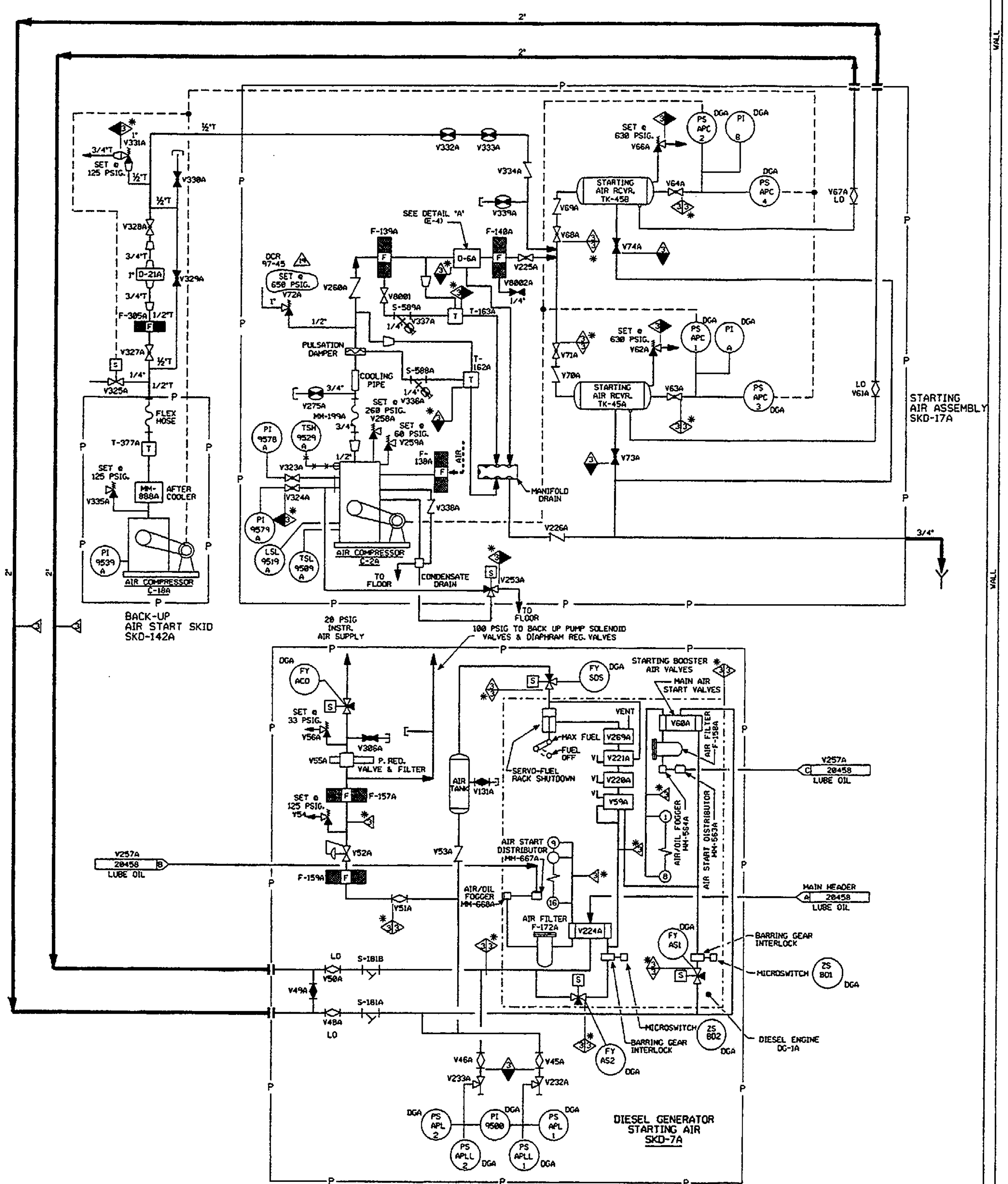
DIESEL GENERATOR COOLING WATER SYSTEM TRAIN 'B' DETAIL

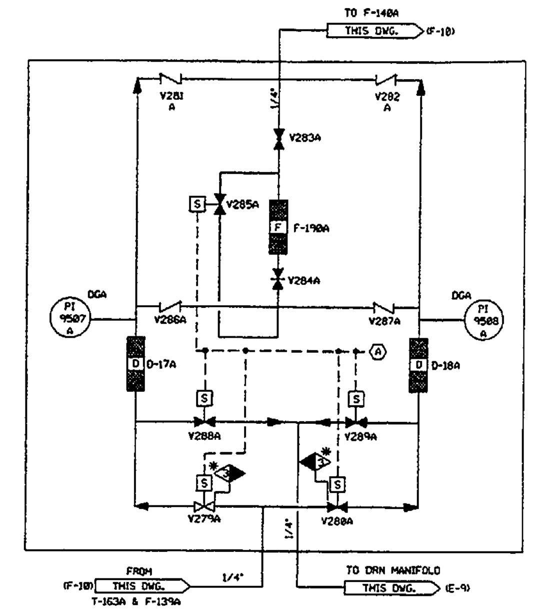
1-DG-B20466

FIGURE 9.5-11 SH 2

PID- 1-DG-B20460

REV. 07





AIR DRYER 1-DG-D-6A DETAIL 'A' (F-10)

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20456, 20457, 20458, 20459, 20461, & 20462.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION HAVE SYSTEM PREFIX DG, UNLESS NOTED OTHERWISE.
3. DELETED
4. △-INDICATES REVISION LEVEL.
5. VENTS, DRAINS AND TEST CONN. CODE AND SAFETY CLASS BREAKS ARE AT THE DOWN STREAM END OF THE OUTER ISOLATION VALVE PER 1-NHY-805111, UNLESS NOTED OTHERWISE.
6. INSTRUMENT REFERENCE

   (A) TO CONTROL PANEL ON SKD-17A, FP-22574.

REF. DRAWING NO.	REV	TITLE
FP-20883	12	EMERG.DIESEL STARTING AIR ASSEMBLY SKD-17A
FP-20699	13	DIESEL GENERATOR STARTING AIR SKD-7A
9763-F-202101	13	DIESEL GENERATOR-AIR SYSTEM P&ID

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

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FINAL SAFETY ANALYSIS REPORT

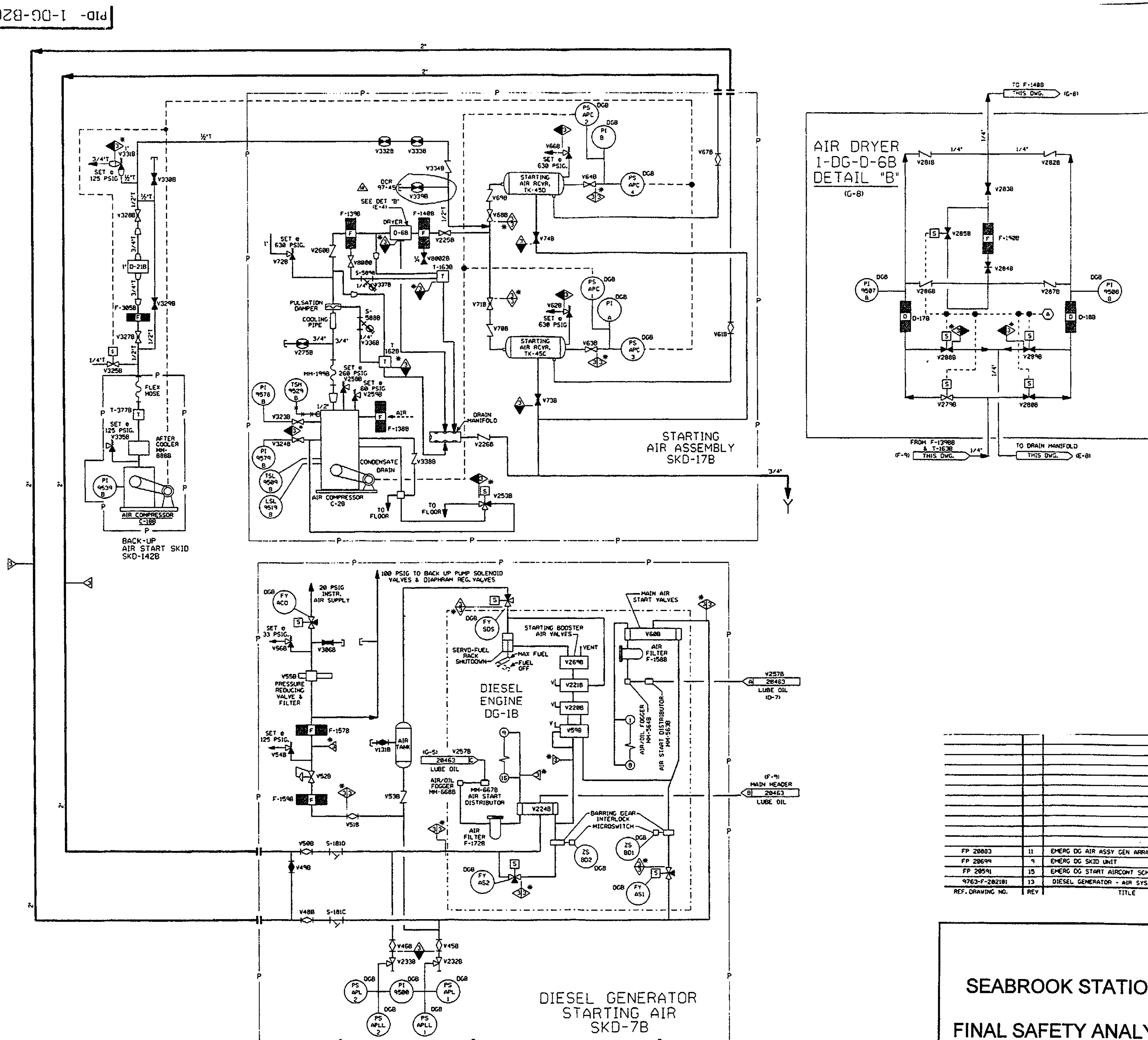
DIESEL GENERATOR STARTING AIR SYSTEM TRAIN 'A' DETAIL

1-DG-B20460

FIGURE 9.5-12 SH 1

06/12/00 REV. 19

REV. 07



NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20456, 20457, 20463, 20464, 20466, AND 20467
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION HAVE SYSTEM PREFIX DG. UNLESS NOTED OTHERWISE.
3. DELETED
4. △ INDICATES REVISION LEVEL.
5. VENTS, DRAINS AND TEST CONN. CODE AND SAFETY CLASS BREAKS ARE AT THE DOWNSTREAM END OF THE OUTER ISOLATION VALVE PER 1-NHY-805111, UNLESS NOTED OTHERWISE.
6. INSTRUMENT REFERENCES:
   (A) TO CONTROL PANEL ON SKD-178 FP-22575

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

REF. DRAWING NO.	REV	TITLE
FP 20883	11	EMERG DG AIR ASSY GEN ARRANGEMENT
FP 20699	9	EMERG DG SKID UNIT
FP 20591	15	EMERG DG START AIRCONT SCHEM AND LEGEND
9763-F-202101	13	DIESEL GENERATOR - AIR SYSTEMS P&ID

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FINAL SAFETY ANALYSIS REPORT

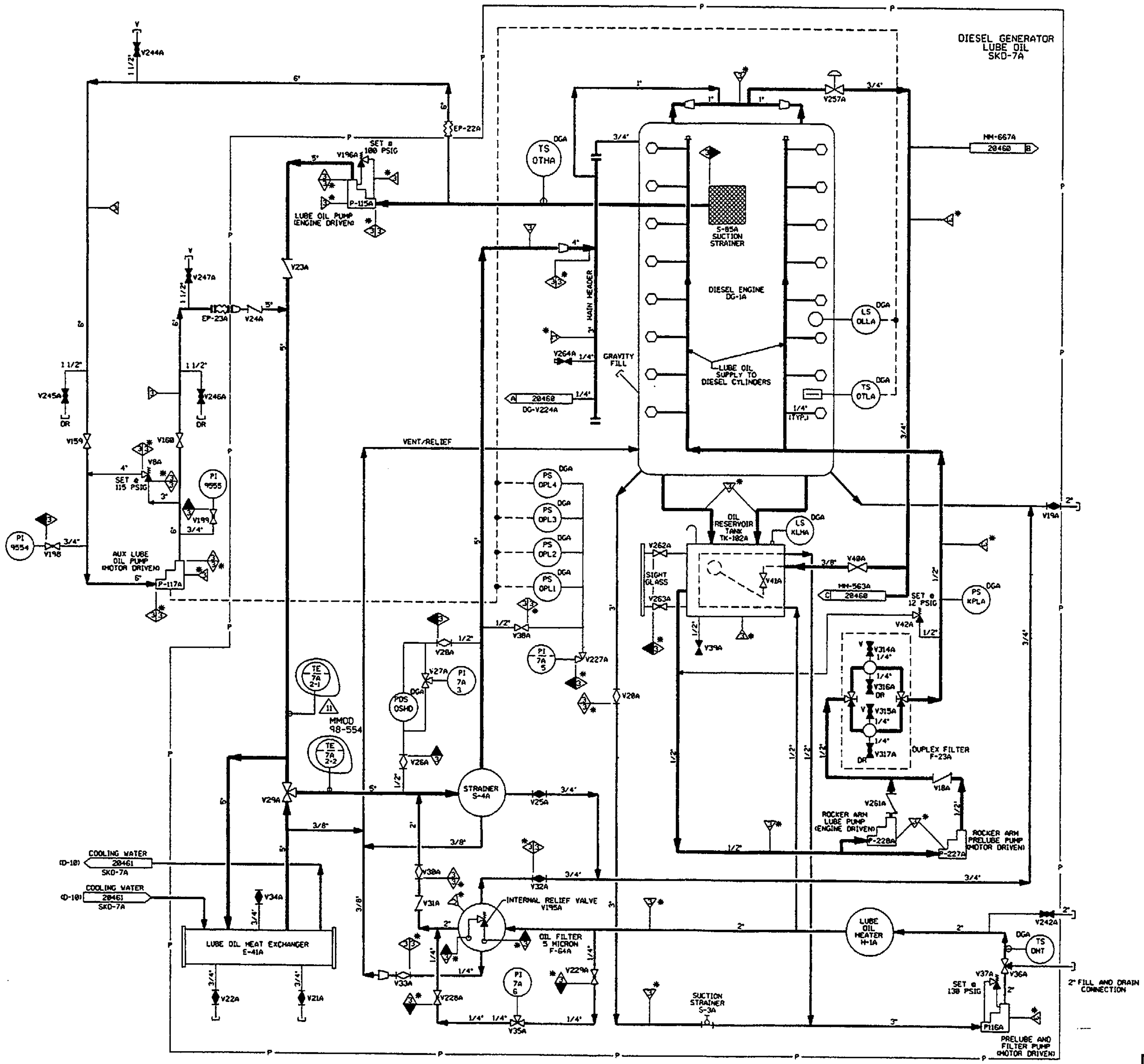
DIESEL GENERATOR STARTING AIR SYSTEM TRAIN 'B' DETAIL

1-DG-B20465

FIGURE 9.5-12 SH 2

PID- 1-DG-B20458

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20456, 20457 & 20459 THRU 20462.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION HAVE SYSTEM PREFIX DG, UNLESS NOTED OTHERWISE.
3. DELETED.
4. VENT AND DRAIN CODE AND SAFETY CLASS BREAKS ARE AT THE DOWNSTEAM END OF THE OUTER ISOLATION VALVE PER DESIGN STANDARD 1-NHY-805111, UNLESS NOTED OTHERWISE.
5. △ INDICATES REVISION LEVEL

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

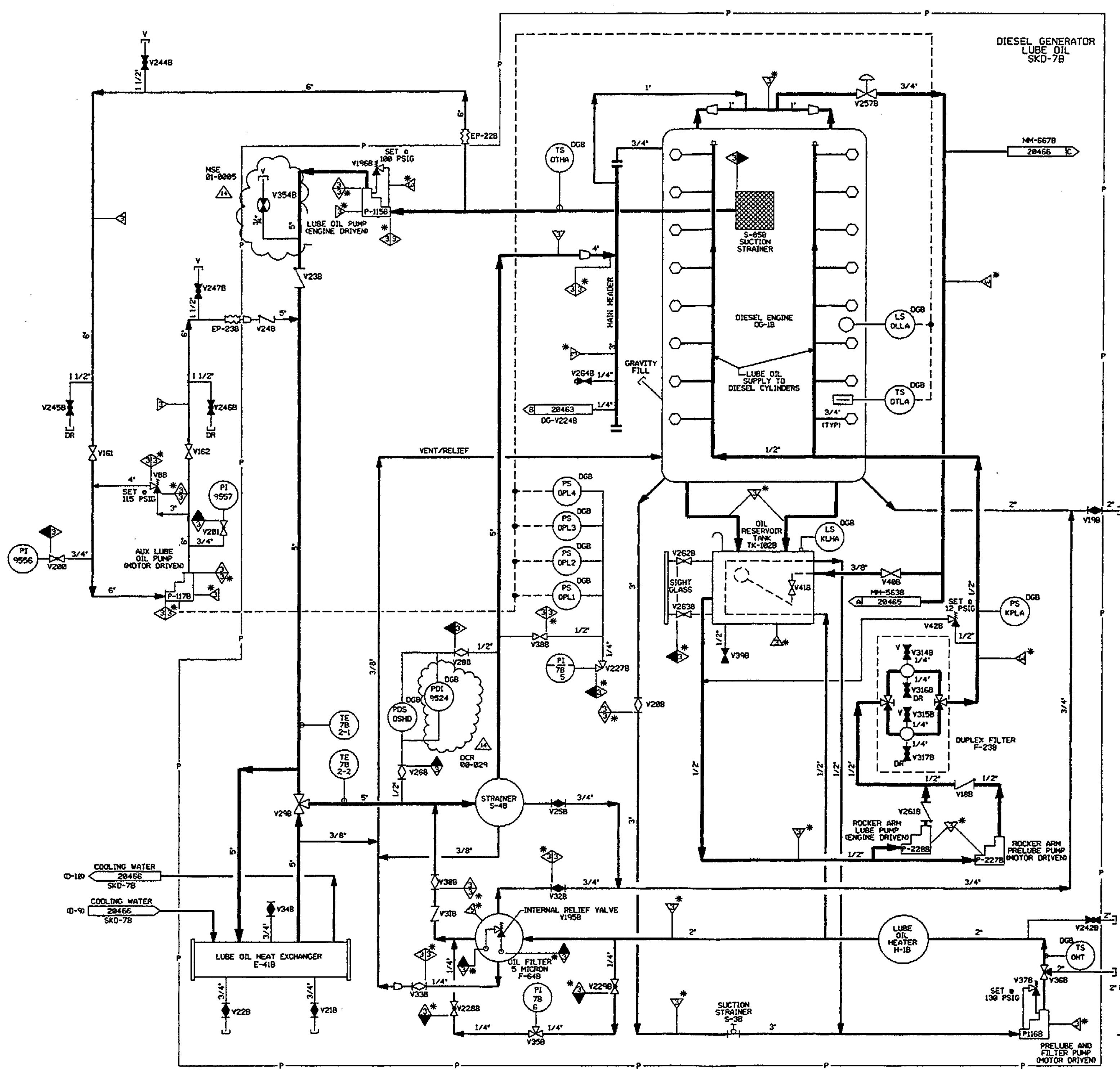
REF. DRAWING NO.	REV	TITLE
9763-F-202102	16	DIESEL GENERATOR FUEL & LUBE OIL P&ID

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	DIESEL GENERATOR LUBE OIL SYSTEM TRAIN 'A' DETAIL	
	1-DG-B20458	FIGURE 9.5-13 SH 1

PID- 1-DG-B20463

REV. 07

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2



NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20456, 20457, 20464, 20465, 20466, & 20467.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION HAVE SYSTEM PREFIX DG, UNLESS NOTED OTHERWISE.
3. DELETED
4. VENT AND DRAIN SAFETY CLASS AND CODE BREAKS ARE AT THE DOWNSTEAM END OF THE ISOLATION VALVE PER DESIGN STANDARD 1-NHY-805111, UNLESS NOTED OTHERWISE.
5. △ INDICATES REVISION LEVEL

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

REF. DRAWING NO.	REV	TITLE
FP. 20590	13	EMERG. DG LUBE OIL SYS. SCHEMATIC
FP. 20699	9	EMERG. DG SKID UNIT GEN. ARRGMNT.
9763-F-202102	16	D.G. FUEL & LUBE OIL P&I DIAGRAM

SEABROOK STATION UPDATED

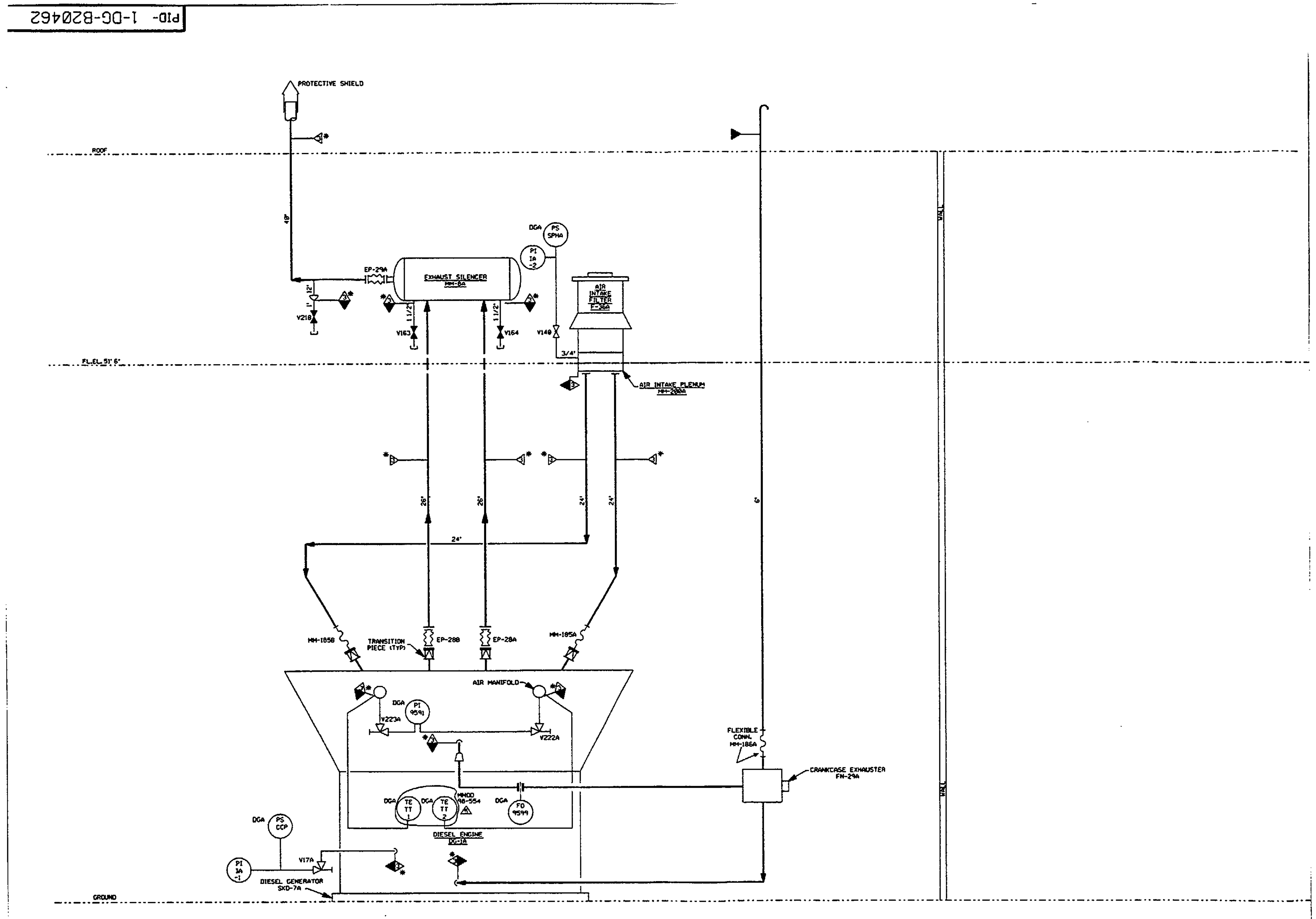
FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR LUBE OIL SYSTEM TRAIN 'B' DETAIL

1-DG-B20463

FIGURE 9.5-13 SH 2

REV. 07



FOR P&ID REFERENCE DRAWINGS SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID-LEGEND 1 AND P&ID-LEGEND 2.

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20456, 20457, 20458, 20459, 20460, & 20461.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTATION HAVE SYSTEM PREFIX DG, UNLESS OTHERWISE NOTED.
3. DELETED
4. VENT AND DRAIN CODE AND SAFETY CLASS BREAKS ARE AT THE DOWNSTEAM END OF THE OUTER ISOLATION VALVE PER DESIGN STANDARD 1-NHY-805111, UNLESS NOTED OTHERWISE.
5. △ INDICATES REVISION LEVEL

REF. DRAWING NO.	REV	TITLE
FP20589	09	EMERG DG INTK, EXT 4 & CRKC SCHEM
FP20903	08	EMERG DG AIR INLET FILTER (DRY TYPE)
FP20840	07	EMERG DG EXHAUST SIL.(40 INCH) 26"INLETS
9763-F-202101	13	DIESEL GENERATOR AIR SYSTEM P&ID

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FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR INTAKE, EXHAUST & CRANKCASE VACUUM SYSTEM TRAIN 'A' DETAIL

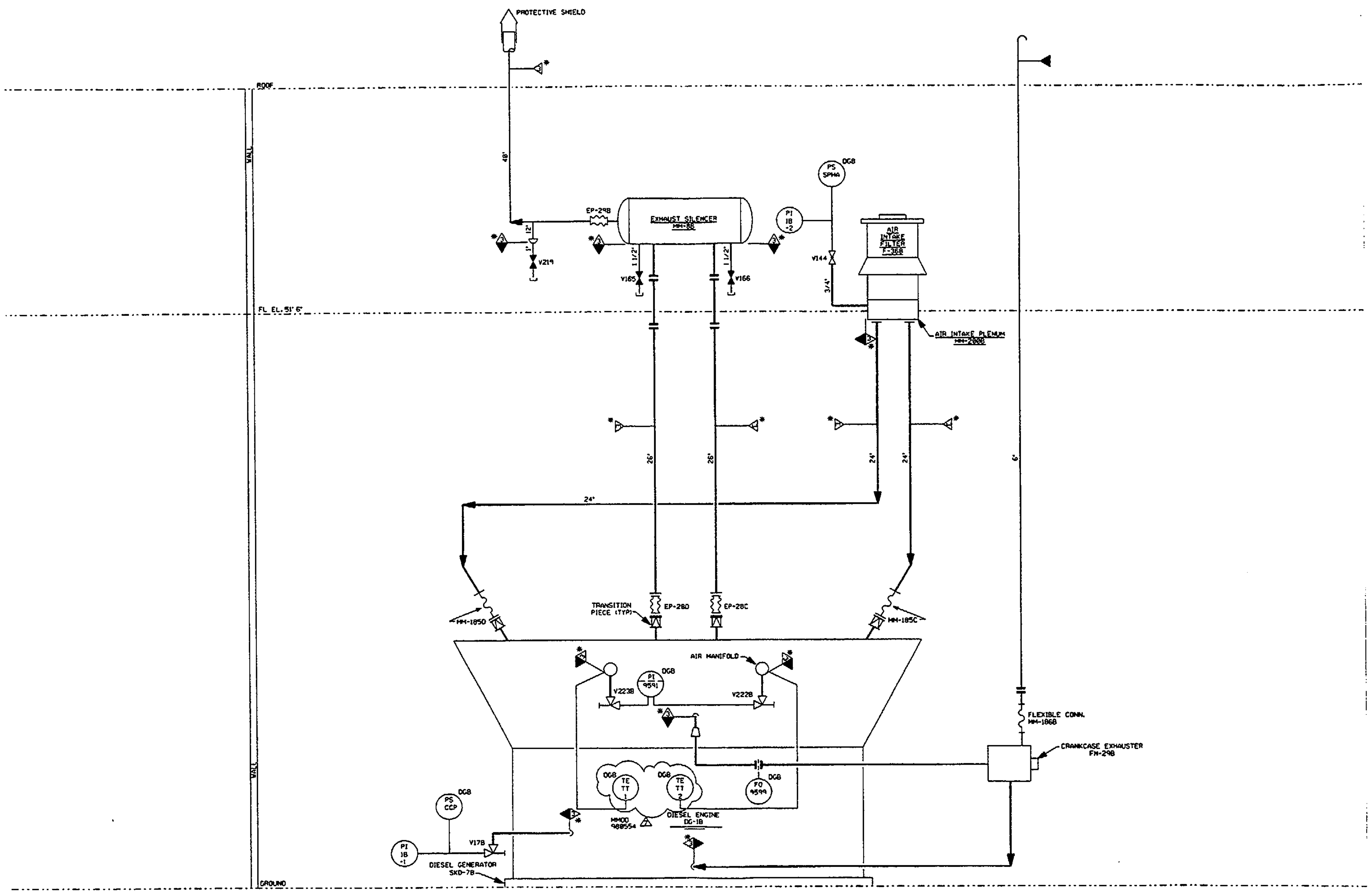
1-DG-B20462

FIGURE 9.5-14 SH 1

PID- 1-DG-B20467

REV. 07

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID-LEGEND 1 AND P&ID LEGEND 2.



NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20456, 20457, 20463, 20464, 20465, & 20466.
2. ALL LINES, EQUIPMENT, COMPONENTS, AND INSTRUMENTS HAVE SYSTEM PREFIX DG, UNLESS NOTED OTHERWISE.
3. DELETED.
4. △ INDICATES REVISION LEVEL
5. VENT AND DRAIN CODE AND SAFETY CLASS BREAKS ARE AT THE DOWNSTEAM END OF THE OUTER ISOLATION VALVE PER DESIGN STANDARD 1-NHY-805111, UNLESS NOTED OTHERWISE.

REF. DRAWING NO.	REV	TITLE
FP 20589	9	EMERG DG INTK, EXH & CRANKCASE VACUUM SYS
FP 20840	5	EMERG DG AIR INLET FILTER (DRY TYPE)
FP 20903	E	EMERG DG EXHAUST SILENCER(40 IN) 26" INLET
9763-F-202101	13	D.G. AIR SYS. P&ID

SEABROOK STATION UPDATED

FINAL SAFETY ANALYSIS REPORT

DIESEL GENERATOR INTAKE, EXHAUST & CRANKCASE VACUUM SYSTEM TRAIN 'B' DETAIL

1-DG-B20467

FIGURE 9.5-14 SH 2

08/25/99 REV. 07

Seabrook Station

North Atlantic

Updated Final Safety Analysis Report

Revision 7

CHAPTER 10

STEAM AND POWER CONVERSION SYSTEM

TABLES

CHAPTER 10

STEAM AND POWER CONVERSION SYSTEM

FIGURES

CHAPTER 10

STEAM AND POWER CONVERSION SYSTEM

10.1 SUMMARY DESCRIPTION

This section summarizes the steam and power conversion system provided for each of the two Seabrook units. The steam and power conversion systems are those portions of the plant designed to transmit and convert the thermal energy produced in the reactor into electrical energy.

The major components of the steam and power conversion system are located in the Turbine Building, and are not safety related. Design parameters of the steam and power conversion system components are presented in the applicable subsections which describe the major systems and components. The system components are shown on the general arrangement drawings in Section 1.2.

Steam at 1000 psia, 1191.3 H, 0.25 percent moisture is supplied from the outlet of four steam generators to drive a tandem-compound, six-flow exhaust, 1800-rpm turbine. Heat balances at guaranteed flow (100 percent rating) and maximum flow with valves-wide-open (VWO) are shown in Figures 10.1-1 and 10.1-2.

The turbine nameplate rating is 1,197,085 kW at 2.0 inches Hg absolute back pressure and zero percent makeup; the rating of the generator coupled to the turbine is 1,350,000 kVA at 75 psig H_2 pressure, 25,000 volts, 3 phase, 60 Hz and 0.92 power factor. The heat cycle results in a calculated T-G output of 1,198,408 kW at 100 percent load. Allowing for plant loads, the net plant output is approximately 1,148,000 kW.

During normal operation, main steam is taken ahead of the turbine stop valves to supply the single-stage reheaters and the turbine shaft sealing system. Crossover steam from the moisture separator-reheater outlets is supplied to drive two steam generator feedwater pump turbines. During startup or low load operation, main steam can be used to drive the steam generator feedwater pump turbines or the electric driven startup feed pump can be utilized.

Moisture separation with one stage reheat is provided between the high-pressure and low-pressure turbines for all steam entering the low-pressure turbines. Steam from the low-pressure turbines is condensed in a three-shell surface-type, two-pass condenser. Condensate is collected in condenser hotwells which are sized for a minimum of 3-minute storage capacity at full load. The condensate and Feedwater System returns feedwater to the steam generators through six stages of extraction feedwater heaters.

Circulating water (sea water) is pumped through the main condenser and returned to the ocean to dissipate the remaining unusable heat from the steam and power conversion system.

Disposal of heat from the Reactor Coolant System following sudden load rejections or trip of the unit is handled by the Steam Dump System. A detailed description of the Steam Dump System is contained in Subsection 10.4.4. The steam generators are also utilized as the heat sink for reactor heat decay, by absorbing heat from the Reactor Coolant System and producing steam. The steam may be bypassed around the main turbine and condensed in the condenser or vented to atmosphere if the condenser is not available.

The steam and power conversion system provides load-following capability at step load changes of ±10 percent and ramp load changes of ±5 percent per minute over the load range of 15 to 100 percent reactor output. The system can accept a 50 percent load reduction without a reactor trip. The Steam Dump System is capable of bypassing 40 percent of the full-load steam flow. This allows for shutdown from half-load or controlled reactor runback without atmospheric venting of steam through the steam generator relief valves. The Steam Dump System opens automatically to the extent necessary during rapid load reductions to remove excess heat from the Reactor Coolant System and closes as operating conditions stabilize at the new load.

Safety valves are provided on the main steam lines from steam generators, and the steam (shell) side of feedwater heaters and moisture separator-reheaters. Diaphragms are provided in the exhaust sections of the low pressure turbines for overpressure protection of the turbine exhaust sections and the main condensers.

The individual components of the steam and power conversion system are based on proven conventional design, acceptable for use in large central power generating stations. The turbine plant auxiliary equipment is selected to provide the optimum operating economy with maximum safety and reliability. All auxiliary equipment is specified for a design capability corresponding to turbine valves-wide-open condition (VWO). Adequate design margins are included as required for wear and system surges to provide dependable service.

10.2 TURBINE GENERATOR

10.2.1 Design Bases

The Turbine Generator System is designed to receive the steam output of the nuclear steam supply system and convert its thermal energy into electrical energy. The turbine generator has a guaranteed output of 1,197,085 kW with throttle steam conditions of 975 psia, 1191.3 Btu/lb and a back pressure of 2.0" Hg abs.

The turbine generator is intended for base load operation, but is capable of step load changes varying from 20 percent at one-quarter load to 67 percent at full-load, and ramp load changes up to 10 percent/minute increasing and 27 percent/minute decreasing. This is compatible with the nuclear steam supply operation limitations, which are ±10 percent maximum step load changes and ±5 percent/minute ramp load changes, over the load range from 15 percent to 100 percent power.

The turbine generator is equipped with instrumentation to continuously monitor and alarm all significant variables, such as speed, load, pressures, temperatures, valve positions, thermal expansion movements, and shaft eccentricity and vibration.

The Turbine Generator System is a nonnuclear system, with associated piping designed in accordance with either manfacturer's standards or the Power Piping Code, ANSI B31.1. Pressure-containing vessels are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII, Division 1.

10.2.2 Description

The turbine generator and related auxiliary equipment, piping and pressure vessels are shown on the turbine building general arrangement drawings in Section 1.2. Turbine generator general auxiliaries and other equipment in the Turbine Building are cooled by the Secondary Component Cooling Water System (see Figure 10.4-14).

10.2.2.1 Turbine

The turbine is an 1800 rpm, tandem compound with six flow, low pressure stages and 43-inch last stage blades. The turbine consists of one high pressure double-flow section that exhausts into four single-stage reheat MSRs then to three low pressure double-flow sections.

Steam from the main steam header is admitted to the turbine through four angle-body control valves. The main stop valves are welded directly to the inlet nozzles of the control valves. The stop valve below seat chambers are connected by an equalizing line. An internal bypass valve is provided in one of the main stop valves to provide pre-warming steam to the stop and control valves bodies, as well as the turbine. The bypass valve also provides pressure equalization across the stop valve seats, which is required prior to opening the stop valves.

Steam leaving the high pressure turbine has a moisture content of approximately 10 percent. To improve turbine efficiency and reduce low-pressure turbine exhaust moisture, thus minimizing maintenance on the low-pressure blading, the steam exhausted from the high pressure turbine is passed through four moisture separator-reheaters in parallel.

Within the moisture separator-reheater, the moisture is removed from the wet exhaust steam, and the dried steam is reheated to approximately 510°F by condensing high pressure steam in a tube bundle. The reheated steam is admitted to the three low-pressure cylinders of the turbine, from which it exhausts into three individual shells of the main condenser.

The steam and water contained in the moisture separator-heater and cross-around piping would, on loss of load, tend to accelerate the turbine. To prevent this occurrence, combined stop and intercept valves are provided in the cross-around lines, at the inlet of the low pressure turbines. Relief valves which discharge to the condenser are provided on the moisture separator-reheaters to prevent overpressure in the cross-around system.

The turbine and generator bearings are lubricated by a conventional pressurized oil system. A main lubricating oil pump, driven by the turbine shaft, provides the bearing lubricating oil during normal operation. During startup or shutdown, bearing lubricating oil is supplied by AC motor-driven pumps. In addition, a DC motor-driven emergency oil pump provides bearing lubricating oil in case of loss of site power.

The main steam and intermediate reheat valves are opened by a 1600 psig hydraulic fluid system which is totally independent of the bearing oil system. These valves are closed by springs and steam forces upon depressurization of the hydraulic fluid system. The valve actuation system is such that loss of hydraulic fluid pressure for any reason leads to valve closing and consequent unit shutdown.

The turbine valve closure times are as follows:

Turbine main steam stop valve	0.15 seconds
Turbine main steam control valve	0.19 seconds
Intermediate stop valve	0.20 seconds
Intercept valve	0.17 seconds

The turbine valves will be tested for proper operation for overspeed protection using the following procedures.

10.3.5 Secondary Water Chemistry

10.3.5.1 Description

Secondary side water chemistry is established and maintained within the steam generator supplier's specification by:

a. Using a deaerating condenser that removes oxygen from the condensate

b. Chemical addition

c. Continuous blowdown of the steam generator

d. Cleaning the condenser and the Condensate and Feedwater Systems before startup.

When the steam is condensed, undissolved gases are released. These gases, including oxygen, are taken out by the condenser evacuation system. The free oxygen content in the condensate leaving the condenser is less than 10 parts per billion.

Additional oxygen removal is accomplished by the addition of hydrazine at the condensate pump's discharge header or steam generator feedwater pump discharge piping. This hydrazine, by a chemical reaction, scavenges the oxygen in the feedwater. Maintaining a hydrazine residual near the steam generator feedwater inlet ensures that any dissolved oxygen is removed. Hydrazine also promotes the formation of a protective metal oxide film on carbon steel surfaces.

An approved pH additive is also added at the condensate pump's discharge header or steam generator feedwater pump discharge piping. It is used to establish and maintain an alkaline pH in the feed train and the steam generator. Alkaline conditions reduce corrosion at elevated temperatures and promote the formation of a protective metal oxide film. The technique of adding hydrazine and an approved pH additive is known as Zero Solids or All Volatile Treatment (AVT) because these chemicals are volatile and will not concentrate in the steam generator, thereby adding solids. With this type of treatment, the contaminants entering the steam generator must be kept at a minimum. This is accomplished by continuous monitoring of condenser in-leakage, the feedwater train, and the steam generators. Water chemistry monitoring is discussed in Subsection 9.3.2.

The Seabrook Station secondary pH program is based on optimizing the "at" temperature pH (pHt) in various portions of the secondary plant by the use of amines. All amines suitable for use as secondary system pH control agents are classified as weak base compounds. When dissolved in water, these compounds partially ionize forming the hydroxide ion which is responsible for their alkaline properties. Temperature has a marked effect on the ionization constant of each amine depending on the amine itself. As a result the application of a single amine at a constant application rate will result in varying pHt in different regions of the secondary system. It may be necessary to employ a mixture as close as possible to the target pHt for that region in order to maintain the solubility of iron in that region at the lowest level possible.

The concentration of any contaminants that do enter the steam generator is reduced by intermittent and continuous blowdown. This is discussed in Subsection 10.4.8. The plant is equipped with a filtering system that recirculates a portion of the condensate through filters and returns it to the condenser. This procedure removes solid particles from the condenser and the Condensate and Feedwater Systems during low power operation.

10.3.5.2 Effect of Water Chemistry on Iodine Partitioning

The formation of volatile iodine compounds in the steam generator is suppressed by the condition of the secondary water chemistry. The amount of iodine carryover is normally dependent upon the efficiency of the moisture separators within the steam generators.

An approved pH additive is used to regulate the condensate pH. The pH range for effectively eliminating iodine volatility is 8.5-11.0. The "at" temperature pH for condensate is 9-10, maintaining iodine almost completely in its non-volatile state. Thus, only minimal amounts of iodine would be exhausted through the condenser vacuum pumps.

The amount of iodine released to the environment in the event of a steam generator tube rupture is minimal. This is the result of the high degree of iodine partitioning in the steam generator and condenser, and the charcoal filter at the condenser vacuum pump discharge. Refer to Subsection 15.6.3 for further discussion.

10.3.5.3 Control Program

The EPRI Secondary Water Chemistry Guidelines are the basis for the secondary chemistry control program.

A summary of operative procedures which are used for the steam generator secondary water chemistry control and monitoring program is as follows:

a. Procedures are available for sampling for the critical chemical and other parameters and of control points or limits for these parameters for each mode of operation: normal operating, hot startup, cold startup, hot shutdown, cold wet layup. The sampling schedule is expected to closely follow the recommendation of the EPRI Secondary Water Chemistry Guidelines. Critical parameters and specifications for each mode of operation are in accordance with the EPRI Secondary Water Chemistry Guidelines.

b. Procedures for chemical analysis of critical parameters were developed using references such as

   1. American Public Health Association, Standard Methods for Examination of Water and Waste Water

   2. American Society for Testing Materials, Part 31 Water

c. Locations for process instrumentation and sample points are indicted on Updated FSAR Figure 9.3-15. The extensive process instrumentation which monitors critical parameters of the secondary system will result in continuous assessment of the secondary system. Grab samples are taken at critical points (i.e., blowdown, feedwater, condensate, and makeup) as additional verification of system chemistry control.

d. Procedures for recording and management of data are available. Seabrook Station will have qualified chemistry personnel on station at all times to interpret analysis data of critical parameters on a continuous basis, followed by additional technical review by chemistry department management during normal work hours. All analysis records will be maintained in accordance with station administrative procedures.

e. Procedures for defining corrective action are predicated on the need to maintain condenser integrity by using state-of-the-art techniques for leak detection such as helium-mass spectroscopy to identify air and seawater intrusion. Process instrumentation will result in rapid identification of leaks at Seabrook. Corrective action to identify the location of a verified condenser seawater intrusion in excess of the EPRI chemistry control parameters will be taken promptly. Actions to minimize degradation of steam generator tubes will be taken as described in the EPRI Secondary Water Chemistry Guidelines.

f. The Seabrook shift chemistry technician is the primary individual responsible to interpret operational chemistry data. Shift chemistry technicians will have completed all training required by the nonlicensed training program for chemistry technicians giving them the expertise to advise the Unit Shift Supervisor on operational chemistry occurrences.

10.3.6 Steam and Feedwater System Materials

10.3.6.1 Fracture Toughness

The test methods and acceptance criteria used to verify the fracture toughness of the ferritic materials used in Class 2 and 3 components of the Steam and Feedwater Systems are in accordance with the applicable requirements of Articles NC-2300 and ND-2300 in Section III of the ASME Boiler and Pressure Vessel Code, 1974 edition.

a. Fracture toughness properties of the steam generator pressure boundary materials are described in Subsection 5.2.3.3 of the

Seabrook Updated FSAR. Impact testing of these materials has been performed to verify compliance with ASME III.

b. The main steam and feedwater isolation valves, containment penetrations and the piping between containment penetrations and isolation valves have been reviewed for compliance with General Design Criterion 51, and found to be acceptable. (See SER for Containment Boundary.)

c. The main steam system pipe and fittings inside Containment were fabricated from the materials listed in Subsection 10.3.6.2 below. All welded joints were examined radiographically to ensure minimum weld defects. Impact testing of this material was not considered necessary, since the maximum nil ductility transition temperature for these materials (conservatively taken from NUREG-0577 as 97°F considering the thickness adjustment) was below the minimum service temperature of 100°F established for the hydrostatic test fluid temperature.

The feedwater system pipe and fittings inside Containment and outside Containment up to the check valve beyond the isolation valve were fabricated from the materials listed in Subsection 10.3.6.2 below. All welds were examined radiographically to ensure minimum defects. The piping material, SA-106, was heat-treated to improve impact properties. Impact tests were performed on seven of the eight heats of piping material and met code requirements at the minimum feedwater injection temperature of 50°F.

10.3.6.2 <u>Materials Selection and Fabrication</u>

All Class 2 and 3 pipe, valves and fittings used in the Steam and Feedwater Systems are fabricated from materials that are listed in Appendix I of Section III of the ASME Code.

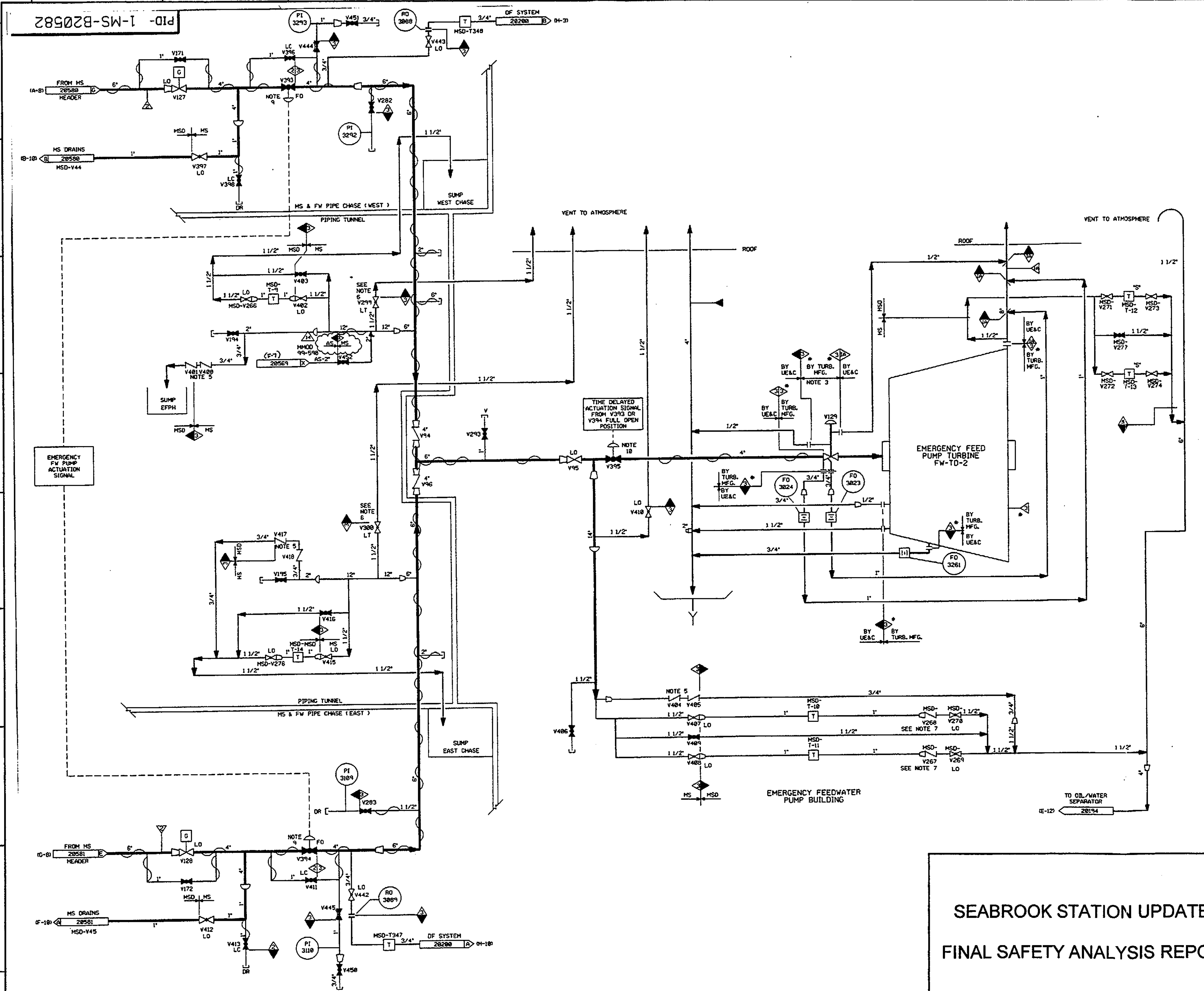
The following materials are used for Class 2 and 3 service:

Main Steam	Feedwater
SA-106, Grade B and C	SA-106, Grade B (normalized, fine grain)
SA-155, Grade KCF 70	SA-234
SA-234	SA-105
SA-105	SA-193, Grade B7
SA-193, Grade B7	SA-194, Grade 7, 2H, 4 or 3
SA-194, Grade 7, 2H 4 or 3	SA-312, Type 304

TABLE 10.3-1

DELETED

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

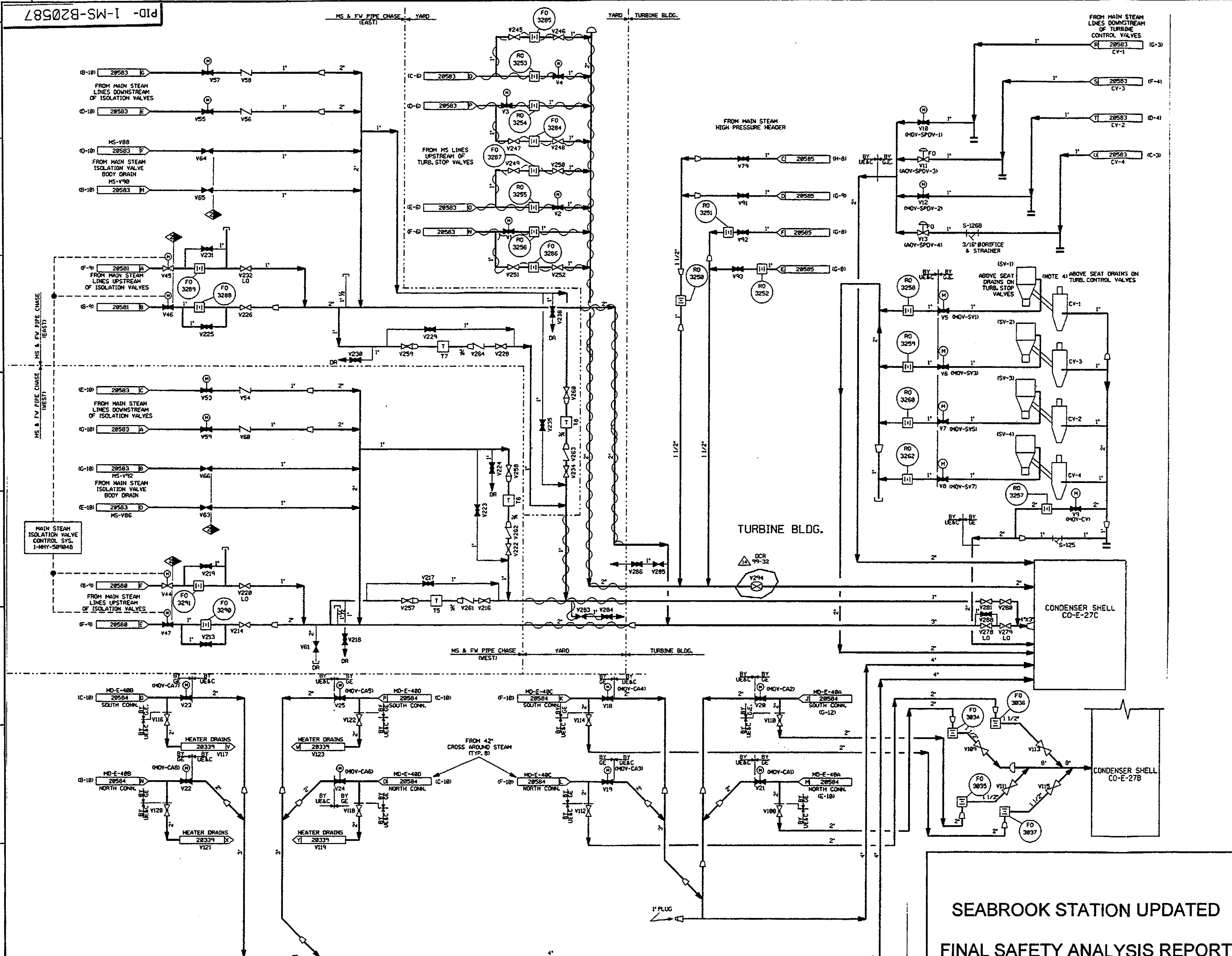
NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20579, 20580, 20581 & 20583 THROUGH 20587.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX "MS", UNLESS NOTED OTHERWISE.
3. VENTS, DRAINS AND TEST CONNECTION CODE BREAKS ARE AT THE DOWNSTREAM END OF THE OUTER ISOLATION VALVE PER 1-NHY-805111, UNLESS NOTED OTHERWISE.
4. DELETED
5. CHECK VALVES V417, V418, V401, V400, V404, & V405 ARE TO BE INSTALLED OPPOSITE FROM DIRECTION OF FLOW IN VERTICAL RUN OF PIPE SUCH THAT THEY HANG OPEN AND ALLOW DRAINAGE WHEN SYSTEM IS DEPRESSURIZED, THUS ENSURING CONDENSATE DRAINAGE & PREVENTING WATER HAMMER.
6. VALVES V299 & V300 ARE THROTTLED ¼ OPEN FROM FULL CLOSED POSITION.
7. THE INTERNALS FROM VALVES V267 & V268 HAVE BEEN REMOVED.
8. 3* INDICATES ANS SAFETY CLASS 3, NOT ASME CODE CLASS 3. (MANUFACTURE'S STANDARDS)
9. FOR INSTRUMENT AIR DETAILS SEE PID IA-B20647.
10. FOR INSTRUMENT AIR DETAILS SEE PID IA-B20644.

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	MAIN STEAM SYSTEM EMERGENCY FEEDWATER PUMP SUPPLY DETAIL	
	1-MS -B20582	FIGURE 10.3-3



REV. 07

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20579 THRU 20586.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX MSD. UNLESS NOTED OTHERWISE.
3. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS. UNLESS NOTED OTHERWISE.
4. MAIN STEAM CONTROL & STOP VALVES ARE SHOWN ON DRAWING 20583.
5. △ INDICATES REVISION LEVEL

9763-F-202118	2	MAIN STEAM PIPING DRAINS
9763-F-202885	7	HEATER MISC. VENTS AND DRAINS
9763-F-202075	14	MAIN STEAM SHT 2 OF 2 P & ID
9763-F-202074	21	MAIN STEAM SHT 1 OF 2 P & ID
9763-F-202086	14	MAIN TURBINE & STEAM PIPING DRNS P & ID
REF. DRAWING NO.	REV	TITLE

SEABROOK STATION UPDATED

FINAL SAFETY ANALYSIS REPORT

MAIN STEAM SYSTEM
MAIN STEAM DRAINS
DETAIL

1-MS-B20587

FIGURE 10.3-8

11/16/00 REV. 14

the unit. Though the isolated condenser shell is still exposed to the turbine exhaust steam, it is possible to inspect the tubes and perform minor maintenance and repairs while the unit is operating.

b. System Operation

During normal operation, exhaust steam from the low pressure turbines is directed into the shells of the main condenser. The following auxiliary flows are also discharged into the condenser:

1. Exhaust steam from steam generator feed pump turbines
2. Drains and vents from feedwater heaters
3. Condensate from gland seal system
4. Miscellaneous equipment drains
5. Steam generator blowdown from blowdown flash tank (vapor only) and from the blowdown demineralizers.

In addition to condensing the steam, the condenser also de-aerates the condensate. The noncondensible gases which accumulate in the condenser are removed by the air evacuation system (see Subsection 10.4.2).

Condenser hotwell level is monitored and necessary makeup is provided by the condensate storage facility (see Subsection 9.2.6). Hotwell conductivity is also monitored and alarmed in the main control room. The monitoring is arranged so the operator can determine which shell has the inleakage, so that the necessary steps to isolate or plug the leaking tubes can be taken.

The condenser has sufficient capacity to condense 40 percent of the full-load steam flow to the turbine during turbine steam load rejection without exceeding 5" Hg pressure.

Total hotwell capacity is 66,000 gallons at normal water level. This is a three-minute capacity, based on VWO (valves wide open) flow.

There is no provision for control of the circulating water flow except by taking a pump out of service. For a description of the Circulating Water System, see Subsection 10.4.5.

c. System Design Data

Design backpressure	2.0" HgA
Backpressure during full load turbine operation, range	1.5-2.8" HgA
Backpressure during steam dump	5" HgA Max.
Total Heat Load:	
Design (guaranteed flow)	$7.58x10^9$ Btu/hr
VWO	$7.90x10^9$ Btu/hr
Steam dump	$7.15x10^9$ Btu/hr
Tubes:	Titanium
Length	55'-3"
Size	1" OD, 22 gage
Circulating water flow (total)	399,000 gpm
Temperature rise	38°F
Water velocity	7 fps
Head loss	27 ft

10.4.1.3 Safety Evaluation

The main condenser has no safety-related design basis. Due to the plant design (PWR) there is negligible influence of condenser control functions on reactor coolant system operation, and negligible potential for hydrogen buildup in the condenser due to continuous gas removal.

In the event of a steam generator tube leak, radioactivity can be present in the secondary side. See Section 11.1 for expected activity due to steam generator tube leakage.

Due to the location of the condenser in the Turbine Building, any flooding resulting from condenser failure will not affect safety-related equipment.

10.4.1.4 Inspection and Testing

The main condenser shell, tubes, and waterboxes are hydrostatically tested to verify integrity prior to initial plant startup.

For service inspection, access manholes are provided on the outlet and turn-around waterboxes, on both ends of the hotwell, and in the steam dome. It is planned to perform a visual inspection of the condenser internals during each refueling outage as part of the normal station preventive maintenance activities.

10.4.1.5 Instrumentation

Condenser vacuum is indicated and recorded in the control room. Condenser vacuum pressure switches are used to (1) alarm pre-turbine trip vacuum, (2) trip the turbine with two-out-of-three coincidence logic and (3) block steam dump to the condenser, also with two-out-of-three coincidence.

Hotwell level indication and high, low and low-low level alarms are provided in the control room. Hotwell level will control the hotwell water inventory by admitting makeup water from the condensate storage facility.

Sea water inleakage to the condenser is monitored by conductivity cells at the catch trough and cation conductivity cells at the hotwell and is recorded and alarmed in the control room and at a local panel. Inleakage is also monitored and alarmed by measuring sodium ion concentration. The instrumentation is sensitive to leakage resulting in ppb concentration.

Monitoring of radioactive contamination is described in Subsection 10.4.2.5.

Additional instrumentation monitors the performance of the condenser by measuring circulating water inlet and outlet temperature and differential pressure. Display of circulating water outlet temperature at the main control board is used for temperature monitoring during backwash operations.

10.4.2 Main Condenser Evacuation System

10.4.2.1 Design Bases

The main condenser evacuation system, Figure 10.4-1, consists of two independent subsystems: the shell side evacuation subsystem and the waterbox priming subsystem.

The Main Condenser Evacuation System has no safety design basis; however, the following design criteria are applicable:

a. The shell side evacuation subsystem removes noncondensible gases and air inleakage from the steam space of the main condenser shells. This system is sized to achieve and maintain shell side vacuum in the condenser to permit plant startup and operation.

b. The waterbox priming subsystem removes the noncondensible gases from the condenser waterboxes (tube side) during startup and normal operation.

Refer to Subsection 10.4.1.1 for the applicable codes and standards for this system.

10.4.2.2 System Description

a. System Components

The air removal equipment consists of three mechanical vacuum pumps serving the three condenser shells, and two mechanical waterbox priming pumps serving the condenser waterboxes.

The vacuum pumps for both subsystems are of the rotary design and electric motor driven, with the shell side pumps being two stage and the waterbox priming pumps being a single stage. Each pump is skid-mounted with its own moisture separator located downstream of the pump discharge and seal water cooler.

The seals to each pump are provided with a closed cooling system using demineralized water. The seal system on the shell side pumps are cooled by circulating water, and on the waterbox pump by service water.

b. Operation

1. Shell Side Evacuation Subsystem

All three pumps will be operated to evacuate the condenser during startup. Condenser pressure can be reduced to approximately 2" HgA in approximately 76 minutes with zero air inleakage. Once vacuum is attained, one pump will be placed on standby to start automatically at approximately 26" Hg vacuum.

During normal plant operation, the noncondensible gases removed by the shell side evacuation system are piped to the Primary Auxiliary Building (PAB) where they are passed through a HEPA and charcoal filter prior to their discharge to the atmosphere. This is done to minimize the possibility of a radioactive discharge to the environment in the event of a steam generator tube leak. For the hogging or startup mode, the noncondensible gases are not expected to be radioactive, and are discharged directly through the Turbine Building vent to atmosphere. See Subsection 10.4.2.5 for discussion on monitoring of shell side discharge. Also, refer to Subsection 11.3.3 for anticipated release rates of radioactive materials.

Vacuum pump discharge flow is directed either to the filter or to the plant stack by diverting valves which are manually positioned by the operator at the main control board.

2. Waterbox Priming Pumps Subsystem

The waterbox priming pumps are used to remove noncondensible gases from the condenser waterboxes during startup, and to

The condenser is equipped with vacuum breaker valves which open within a two-minute period if two circulating water pumps trip, thus preventing the possibility of overpressure from water hammer.

Clogging of a circulating water traveling screen will trip its corresponding circulating water pump, thus preventing collapse of the screen. None of the controls or instrumentation relates to operations which could affect safety-related systems.

The expansion joints used in the CW system are made of fabric reinforced rubber with circumferential steel reinforcing rings. This makes the rupture of a joint in operation a very unlikely occurrence. To evaluate the consequences of a rubber joint failure, they can be divided into four groups as follows:

Group	Number of Joints	Size ID (in)	Pressure Normal Oper. (ft)	Max. (ft)	Location
A	4	102	20	55	Between intake transition structure and pumphouse.
	2	120	20	55	Between discharge transition structure and backwash conduits to the pumphouse.
B	4	120	15	50	At both ends of the discharge pipe section between the two transition structures.
C	6	84	50	130	CW pumps discharge.
D	12	84	50	130	Condenser connections.

Group A

Failure of a joint in this group will result in a flood in the valve pit around the transition structure in which the joint is installed, but the water will reach no higher than the water in the pumphouse. The worm gear boxes directly mounted on the butterfly valves will be submerged; however, the electric motors and controls, which are installed above grade, will always be above water. Since the valves remain operational, and no other equipment is affected by this failure, no immediate action will be necessary.

Group B

Failure of a joint in this group will also cause flooding of the valve pit around the transition structure in which the joint is installed, and the water leaking from the joint will drain into the valve pit, eventually fill it, and overflow in the ground unless the CW system of that unit is stopped. It is estimated that the time required to fill the pit will not be less than 45 minutes, even assuming the worst possible failure to be a 2"x24" gap at the lowest side of the joint. A water level alarm will warn the operator of the flooded condition in the valve pit. Failure of a joint in this group will not prevent the Service Water System from providing its function.

Group C

Failure of a joint in this group will flood the pump pit in the Circulating Water Pumphouse between elevations +3' and +20' MSL. A number of scuppers in the east wall of the Circulating Water Pumphouse will prevent the water from building up above the operating floor. The CW pump electric motors and the pump discharge butterfly valve electric motors will always be above water level, and will not be affected by such a failure. Assuming the worst possible failure to be a 2" gap all around, the CW pump pit would fill up in not less than 6.5 minutes. A water level alarm will warn the operator of the flooded condition. No safety-related equipment is affected by a failure of this equipment.

Group D

Failure of a joint in this group will flood the ground floor pit east of the condensers in the Turbine Building. Assuming the worst possible failure to be a 2" gap all around, the pit would fill up in about 3 minutes, unless prompt action by the operator is taken. There are two level switches in the condenser pit that provide sequential alarms in the control room to warn the operator of the flooded condition. No safety-related equipment is affected by a failure of this equipment provided operator action is taken within 30 minutes to mitigate the consequences of the flood.

Summary

The service water pumps are located at ground elevations and their motors are situated above ground elevation. There are no openings in the common wall between the service water pumps and CW pumps. Therefore, flooding which may spread to the open ground has little potential for entering the Service Pumphouse and no potential for interrupting the service water pump safety function.

Flooding caused by a flexible joint failure at the main condenser may spread over the turbine floor and from there to the open ground by passing through Turbine Building doors. However, sufficient physical separation is provided between the areas subject to such a flood and any safety-related equipment whose safety function could be impaired by such a flood.

bay. The outlet flow is manifolded and routed to two parallel low pressure heaters No. 5, also located in the turbine heater bay.

Low pressure heater No. 1 is a four-pass, and heaters 2, 3 and 4 are two-pass U-tube type with integral drain coolers, while heaters No. 5 are two-pass, U-tube without drain cooler. All feedwater heater strings are provided with block valves and bypass piping to take units out of service for maintenance. All heaters are provided with tube-side safety valves to provide protection against possible overpressurization caused by heating of water trapped between closed isolation valves.

Extraction steam from the main turbine and cross-under piping is the heat source for the heaters (see Figures 10.3-10 and 10.3-11). Automatic nonreturn valves are provided in the extraction steam lines for heaters that are not mounted in the condenser neck. The drains from low pressure heaters Nos. 1, 2, 3 and 4 are cascaded and eventually discharged to the condensers. The drains from low pressure heaters No. 5, which also receive the cascaded drains from high pressure heaters No. 6, are piped to the heater drain tank.

The condensate outlets from low pressure heaters No. 5 are manifolded in a common header, along with the discharge of the two vertical heater drain pumps, which account for about 30 percent of the total feedwater flow at full power. The heater drain pumps take suction from the heater drain tank. A branch line off the common condensate header, before the steam packing exhauster, connects to the individual heater drain pump suction lines to protect the heater drain pumps against low NPSH during transient conditions.

The common condensate header distributes the flow equally to the suction side of the two steam generator feed pumps, after passing through a flow measuring device located in each pump suction line. These are used to control the respective feedwater pump recirculation valves.

Condensate is obtained from the condensate storage tank for hotwell makeup upon receipt of a low level signal from the condenser hot-wells.

Demineralized water from the water treatment system, which serves both units, can be introduced into the three condenser hotwells via the condensate storage tank. The condensate storage tank is protected from freezing by a recirculation system which utilizes a heat exchanger and pump controlled by tank temperature. All condensate system connections to the condensate storage tank which are required for normal system operation are located above the tank level required for emergency plant shutdown (see Subsection 9.2.6).

Seabrook station has a steam generator design (Westinghouse Model F, see Subsection 5.4.2) that enables the feed ring to be uncovered without subsequent drainage because feedwater exits the feed ring from the top through inverted J-tubes. This arrangement insures a flooded feed ring for all level transients within the steam generator, thereby eliminating water hammer. Feedwater piping from the feedwater isolation valve to the steam generator is routed to be self-venting, with the steam generator being the high point of the system.

Any failure in the nonsafety class portion of the Condensate and Feedwater Systems has no effect on the safety of the reactor, which can be shutdown in an orderly manner. A source of feedwater supply to the steam generators is required for decay heat removal from the reactor following a unit shutdown. In the event that the Condensate and Feedwater Systems are not available, the Emergency Feedwater System (see Section 6.8) provides the required emergency supply of feedwater. Normal reactor cooldown is accomplished by dumping steam to the main condenser. When the reactor coolant system temperature and pressure are reduced to or below the design values for the Residual Heat Removal (RHR) System, steam dump can be secured. The RHR design conditions (600 psig and 400°F) correspond to a secondary system pressure of 233 psig.

If for any reason the normal cooldown mode cannot be utilized, the reactor can be cooled down using the Emergency Feedwater System (see Section 6.8) and the power-operated steam generator relief valves.

10.4.7.4 Inspection and Testing

During preoperational testing, the various components of the feedwater, condensate, and associated portion of the heater drain system are functionally tested to verify their performance to the extent practical. The systems are operated during hot functional testing at normal no-load conditions as a final check prior to plant operation. The specific testing is described in Chapter 14.

In-service inspection of the Class 2 feedwater piping will be performed in accordance with ASME B&PV Code, Section XI.

10.4.7.5 Instrumentation

The instrumentation employed for monitoring the Condensate and Feedwater System performance consists of the following:

a. Each condensate pump suction and discharge pressure is locally indicated, and discharge header pressure is indicated at the MCB. Pump controls are at MCB. Normally, two pumps are running with one in standby. The standby pump runs automatically on low discharge header pressure or on trip of any one of the running pumps. Each pump is protected from overheating by being interlocked to pump cooling water flow. Each pump is protected from damage by individual recirculation flow on low pressure. The steam packing exhauster recirculation valve opens on low flow in the condensate discharge header to provide minimum flow protection for the condensate pumps and closes on low header pressure to prevent pump run out and maintain feed pump NPSH.

Pump trip, bearing and motor winding high temperature, and low cooling water flow are alarmed in the control room.

b. The condenser hotwells are interconnected, and each hotwell is equipped with a level transmitter. The level signals from the three hotwells are processed by an auctioneering circuit which selects the lowest value for the hotwell level control system. The level is maintained within a preselected control band by admitting makeup water via the makeup control valve from the condensate storage tank. Low hotwell level is alarmed in the control room.

c. All feedwater heaters are provided with level controllers and drain control valves for normal drain disposition. High level drain control valves are provided for heaters Nos. 2, 3 and 4 to discharge into the condenser. Low, high and high-high levels are annunciated in the control room. High-high level actuates nonreturn and isolation valves in the extraction steam lines, or condensate isolation valves as applicable, to prevent water carryover to the turbine. Valve position monitoring lights are provided at the MCB for the feedwater heaters spill valves to the condenser and the extraction steam nonreturn and isolation valves.

Gauge glasses are provided for local direct observation of heater liquid levels.

Feedwater heater inlet, outlet and drain temperatures and shell side pressures are monitored and used for the performance computation.

d. The two heater drain pumps are controlled from the MCB. Suction and discharge pressure are indicated locally, and flow to the feed pump suction header is indicated in the control room. Instruments monitor heater drain tank level for indication, control, interlocks, and alarms. The heater drain tank is provided with level controllers which regulate the heater drain pump discharge, recirculation and condenser spill valves. The condenser spill valve is also opened by a high level switch. A high-high level switch initiates turbine water induction protection. Heater drain pump trip, motor bearing, and motor winding high temperatures are alarmed in the control room. The 10-inch spill valve opens on high level.

e. Steam generator feed pumps are provided with local suction and discharge pressure gauges. The indications of these pressures for the steam generator feed pumps are also provided at the MCB.

A flow element is provided in the feedline to each steam generator. Pairs of taps provide for two independent metering measurements from the flow element. The feedwater flow measurements are indicated in the control room and utilized in the steam generator level control system. The details of the safety-related display instrumentation are presented in Section 7.5. The details of the steam generator water level control system are presented in Subsection 7.7.1.7. Manual-auto stations for all the feedwater control valves are provided at the MCB.

For computer trending, separate flow measuring loops utilizing ultrasonic flow transmitters are provided in each feedline to the steam generators.

To ensure that minimum flow requirements are met, a recirculation valve has been provided for each pump, controlled by the suction flow measuring venturi.

f. The steam generator feed pump turbine provides variable speed feed pump operation. The variable speed feed pump control system is described in Subsection 7.7.1.7. Feed pump bearing high temperatures are alarmed in the control room.

10.4.8 Steam Generator Blowdown System

10.4.8.1 Design Bases

The Steam Generator Blowdown System is designed to limit the concentration of dissolved and suspended solids in the shell (secondary) side of the steam generators, which are introduced into the steam generators through the feedwater. Removal of these solids minimizes chemical deposition on steam generator tube surfaces, thus limiting the reduction in heat transfer capability, as well as reducing the rate of steam generator tube corrosion. The potential sources of solids in the steam generators can be any, or a combination of, the following:

a. Chemical additions to the secondary system for corrosion control

b. Reactor coolant boric acid due to primary to secondary leakage through the steam generator tubes

c. Secondary system corrosion products

d. Seawater, due to leakage through the condenser tubes into condensate returning to the steam generator

e. Impurities in condensate makeup water.

The unit has an independent steam generator blowdown and sampling system. Initial processing (depressurization and cooling) of steam generator blowdown liquid is accomplished using the flash tank and bottoms coolers. If the radioactivity in the blowdown liquid is insignificant, then the liquid is returned to the Condensate System after processing through demineralizers, or discharged directly to the Circulating Water System.

The method to process radioactive secondary liquid from the steam generators is to direct steam blowdown flash tank bottoms cooler discharge to the floor drain tanks. If no secondary pressure is available, the steam blowdown and wet lay-up pumps can be used. From the floor drain tanks, processing takes place using the installed vendor resin skid (WL-SKD-135) to the waste or recovery test tanks. (Reference Subsection 11.2.2.1.)

The unit has the continuous blowdown capability of 400 gpm (100 gpm per steam generator). This very high flow rate is used only when no primary to secondary leakage exists and when high feedwater solids concentrations require such a blowdown rate. This high rate of blowdown is expected to occur at startup, after a unit shutdown in which maintenance has taken place on the secondary side of the plant. A high rate of blowdown may also be required at other times to control chemistry conditions. In the flash tank, 30 percent of this blowdown is flashed to steam, leaving 70 percent to be processed by the demineralizers, or discharged directly to the Circulating Water System.

Blowdown may be discharged directly to the Circulating Water System when cleanup capabilities are unavailable, as long as the limits of 10 CFR 20 (Appendix B, Table 2, Column 2 - instantaneous release) and 10 CFR 50 Appendix I (average annual release) are not exceeded.

Isolation valves are provided outside the Containment to close all blowdown lines on a "T" signal, Auto emergency feed pump start, or on a HELB signal.

Piping and valves from the steam generator up to and including the containment isolation valves, are Safety Class 2, and are designed to ASME Section III, Code Class 2 (see Section 3.2). Other piping and equipment in the steam generator blowdown systems are nonnuclear safety class, and are designed to ANSI B31.1.B-1971 and ASME Section VIII.

Blowdown water chemistry control parameters are established using the EPRI Secondary Water Chemistry Guidelines.

10.4.8.2 System Description and Operation

a. Normal Operation

Figures 10.4-10 through 10.4-13 are flow diagrams of the system. Each of the four steam generators is provided with a bottom blowdown connection on the secondary side above the tube sheet. During normal operation, each steam generator undergoes continuous blowdown with the blowdown water passing through a containment isolation valve, flow meter, and system valves. A small quantity of blowdown is continuously drawn off automatically into the sample system through a sample heat exchanger for monitoring of the activity in the blowdown. If the activity in the blowdown discharge is higher than allowable (see Subsection 9.3.2), blowdown is automatically secured. The blowdown liquid then flows through a manual control valve which establishes the blowdown rate. Some of the liquid flashes upon passing through the control valve, and two-phase flow then enters the flash tank. There, approximately 30 percent of the blowdown flow exits the top of the tank as saturated steam. The remaining 70 percent exits the bottom of the tank as saturated water. The flash tank operates at 70 psia. Steam and water leaving the flash tank are directed as follows depending upon the existence and size of primary to secondary leakage.

1. If No Primary to Secondary Leakage Exists

Flash tank steam will normally be directed back to the No. 3 feedwater heater or to the main condenser for that unit, or can be exhausted to the atmosphere if the heater and condenser are not available. Liquid from the flash tank will be cooled in the flash tank bottoms cooler and directed through a radiation monitor and flow totalizer into demineralizers. The demineralizers will remove chemical contaminants and radioactivity (as explained later), and the liquid can be transferred back to the condenser. Filters upstream of the demineralizers prevent the transport of corrosion products into the beds. An alternate path to discharge the blowdown liquid to the environment is available, that is, into the service water system discharge line via the Waste Liquid System (see Section 11.2). Flow into the flash tank under these circumstances can vary from 20 gpm (5 gpm per steam generator) to 400 gpm (100 gpm per steam generator).

2. <u>If Primary to Secondary Leakage Exists</u>

The early indication of a primary to secondary leak may be observed as a result of routine grab sampling (at very low leak rates) or on any of the following radiation monitors:

- Condenser air removal, RM 6505
- Blowdown Flash Tank, RM 6519
- Individual Blowdown line, RM 6510-6513 (see Subsection 9.3.2).

Each of the setpoints on these radiation monitors is established, using station procedures, such that plant personnel can detect indications of small leaks, during the early phases of the leak.

If activity in excess of the alarm setpoint is measured on the Flash Tank RM, the blowdown system will automatically isolate. The individual monitors on the blowdown lines will provide indication of which steam generator is leaking. New RM setpoints may be established to continue system operation, once a steady state has been achieved.

EPRI PWR Primary to Secondary Leak Guidelines identifies the significance levels of primary to secondary leaks. The operational plan for how to continue plant operation when leakage exists is located in plant procedures for management overview as well as individual departmental procedures for increased monitoring.

These administrative and operational procedures dictate how the steam generator blowdown liquid will be processed during a primary to secondary leak. The following options are available to operations personnel:

- Continue to use the blowdown demineralizers as in normal operation. This requires additional radiation monitoring in the Demineralizer Building to ensure general area dose rates are within Zone II limits, as well as increased grab sampling frequency to closely monitor the leak rate change.

- Use the Waste Liquid System to treat all or part of the flashed liquid, bottoms or both, from the blowdown flash tank. This may require a reduction in blowdown flow rate.

- Discharge to the environment within the confines of 10 CFR 20 and Appendix I of 10 CFR 50 limits.

If the activity in the flash tank steam is low, the steam may be allowed to return to the No. 3 feedwater heater for reuse in the plant without processing. Flashed steam is not intentionally released to the atmosphere. If significant quantities of radionuclides are contained in the flash tank distillate, the liquid may be processed through the flash steam condenser to the waste test tanks. This limits the flow from the four steam generators to approximately 75 gpm as the cooler capacity is only 25 gpm.

Processing of the steam generator flash tank bottoms liquid through the Waste Liquid System will also necessitate a total blowdown flow limit of 71 gpm for all four steam generators (the vendor-operated Liquid Waste Treatment System capacity for process flow is 50 gpm). The basis for this flow is further discussed in Appendix 11A.

b. <u>Operation With the No. 3 Feedwater Heater or Main Condenser Not Available</u>

When the No. 3 feedwater heater or main condenser for the unit is not available, system flows are realigned as follows:

1. <u>If No Primary to Secondary Leakage Exists</u>

Steam from the flash tank may be directed to the flash steam condenser/cooler and then pumped to the waste test tanks in the Liquid Waste System (see Section 11.2). Although under these conditions this liquid would not contain radioactivity, the contents of the waste test tank would be sampled before discharge to the service water system discharge, since the tanks could contain other processed liquid waste.

If blowdown flow requirements result in steam flow from the flash tank above the capacity of the flash tank condenser/cooler (which can process steam only equivalent to a maximum of 25 gpm of water), an additional path is provided to discharge the flash tank steam via the atmospheric exhaust.

Water from the flash tank will be handled in the same manner as in Subsection 10.4.8.2a.1.

2. <u>If Primary to Secondary Leakage Does Exist</u>

Steam from the flash tank is handled in the same manner as explained in b.1. However, the blowdown capacity is limited as explained in Subsection 10.4.8.2a.2. Radioactive steam is not charged to the atmosphere (see Subsection 10.4.8.2a.2).

Liquid from the flash tank will be handled in the same manner as explained in Subsection 10.4.8.2a.2.

c. <u>Cooling Water Shortage Case</u>

During low heat removal capability of the Primary Component Cooling Water (PCCW) System, steam generator blowdown capacity may be limited. Such circumstances may arise during heat treatment of service water system tunnels and initial phases of plant cooldown for a short duration. Additionally, on a "T" signal, cooling water to the Waste Processing Building is isolated. However, the evaporators are automatically shutdown on a "T" signal.

10.4.8.3 <u>Component Design</u>

a. <u>Flow Control Valves</u>

Initial blowdown liquid flashing will occur due to the pressure drop across the flow control valve associated with each steam generator's blowdown line. The flow control valves are sized to pass zero to 100 gpm flow and are designed to minimize noise, vibration and erosion.

b. <u>Blowdown Flash Tank</u>

The blowdown flash tank is sized to permit 400 gpm (200,000 lbm/hr) two-phase flow. Flow enters the tank through four tangential nozzles. A stainless steel wear plate is used to prevent tank erosion. Steam exiting the tank must pass through a mesh style deentrainment separator (demister pad) to limit carryover. The vessel is carbon steel and will operate at 70 psia with overpressure protection provided to limit any pressure excursion while passing maximum steam flow.

Vessel operating pressure is used to force water flow to the demineralizers, or to the service water system discharge piping. Vessel operating pressure is maintained by pressure control valves in the steam line or cooling water line to the flash tank steam condenser, depending upon mode of system operation. Operating level is maintained by one of two level control valves in the liquid discharge line.

c. Flash Tank Bottoms Coolers

The flash tank bottoms coolers are shell and tube heat exchangers sized so that flashing will not occur downstream under maximum blowdown conditions. Each cooler will handle 50 percent maximum flow.

d. Flash Steam Condenser/Cooler

The flash steam condenser is a shell and tube heat exchanger sized to condense approximately 11,500 lb/hr steam. Cooling water flow (PCCW) through the condenser is regulated to maintain 70 psia in the flash tank. The subcooling section at the bottom of the condenser cools the condensate to approximately 120°F.

e. Flash Tank Distillate Pumps

Two 50 gpm (nominal) centrifugal pumps are provided to pump liquid from the flash steam condenser to the main condenser or waste test tanks depending upon the system mode of operation. These pumps can also be used to evacuate the flash tank through a separate valve (normally kept closed) if necessary.

f. Steam Generator Blowdown Evaporators

Each of the calandria blowdown evaporators has the process liquid on the tube side and heating steam on the shell side. The condensers and coolers with the evaporators subsystem are similar to those with the Boron Recovery System (Subsection 9.3.5) and the Liquid Waste System (Section 11.2). However, the heating element is contained within the evaporators for the Steam Generator Blowdown System, whereas it is separate for the Boron Recovery System.

g. Demineralizers

The unflashed liquid from the flash tank is cooled and directed to demineralizers. The three beds are configured to be compatible with secondary system chemistry. The effluent quality will be in accordance with EPRI Secondary Water Quality Guidelines.

h. Booster Pumps

The booster pumps are required to pump the liquid from the flash tank through heat exchangers and demineralizers into the Condensate System. The booster pumps are especially required to overcome the hydraulic resistance in the downstream circuit at high flow rates.

i. Steam Generator Blowdown Recovery (SGBR) Heat Exchanger

The heat exchanger cools down the unflashed liquid from the flash tank to less than 110°F before it enters the demineralizers. This temperature is required to maintain the characteristics of the demineralizer resins.

j. Iron Filters

The iron filters are located upstream of the demineralizer beds and serve to prevent the transport of corrosion products into the beds, which can reduce bed efficiency. The filters comprise two disposable cartridge housings connected in a duplex piping arrangement. This configuration allows for continuous SB operation during cleaning of a filter.

Table 10.4-2 lists the design and operating conditions of the steam generator blowdown system components.

10.4.8.4 Safety Evaluation

The Steam Generator Blowdown System has no safety function, nor is its performance required during or after an accident. Accordingly, the system is designed as nonnuclear safety (NNS), non-Category I. However, from the connection on the steam generator to the containment isolation valves, just outside the Containment, the system is Safety Class 2 and seismic Category I. Some parts of the system may contain radioactive fluids, depending on the presence of steam generator tube leakage. Closure of the blowdown lines is accomplished by air-operated valves that close on high pressure or level in the flash tank or startup of the emergency feed pumps signal. These valves are closed on loss of air pressure or electrical power to the solenoids, thus assuring the performance of the safety function under all failure conditions. Liquid discharge from the flash tank is automatically terminated on a high radiation signal in the discharge line or in the sample withdrawn from each steam generator.

Electrical power is provided at 460 volts, 3 phase, 60 Hz. Emergency electrical power is not provided. Each combination motor-starter incorporates thermal elements to protect against overloads and a magnetic molded case circuit-breaker to protect against faulted conditions.

Monitoring devices are provided to measure conditions of pressure, temperature, radiation, conductivity, flow, and liquid levels to ensure that the system is operated safely and within design limits. The design bases listed in Subsection 10.4.8.1 are met using the flash tank, demineralizer, and Waste Liquid System capabilities. A failure analysis is presented in Table 10.4-3.

10.4.8.5 Test and Inspections

Prior to initial plant startup, the Steam Generator Blowdown System is tested to verify proper operation of system equipment.

During normal plant operation, calibration of the radiation monitors (see Section 11.5) and surveillance testing of the containment isolation valves will be performed in accordance with Technical Specification requirements.

10.4.8.6 Instrumentation and Control

a. Flash Tank Subsystem

1. Containment Isolation Valves

These valves are controlled from the MCB. The outboard valves close automatically on a "T" signal, emergency feed pumps running, or a HELB signal (see Subsection 7.6.10). Flow of individual blowdown lines is indicated at the MCB and locally near the blowdown throttle valves.

A separate set of block valves inside the Containment will automatically close on high level or pressure in the flash tank, or on a HELB signal (see Subsection 7.6.10).

2. Flash Tank Instrumentation

Level and pressure are indicated locally and at the MCB. Temperature of the tank is indicated at the MCB. High and low level, as well as hi and hi-hi pressure is alarmed at the MCB.

3. Flash Tank Control

(a) Pressure Control

In normal operation, the pressure of the tank is maintained at 70 psia by throttling the pressure control valve in the steam line to the condenser. During this time the line to the atmosphere is kept closed. When the main condenser or No. 3 feedwater heater is not available, the flash tank is aligned to the flash steam condenser. Pressure control is then achieved by throttling the cooling water valve at the outlet of the flash steam condenser. In case the main condenser, No. 3 feedwater heater, and flash steam condenser are not available, the flash tank steam is processed to atmosphere in a controlled manner.

(b) Level Control

In normal plant operation the level of the tank is controlled by throttling the level control valve at the common discharge header of the flash tank bottoms coolers. During the operation of the Steam Generator Blowdown Recovery (SGBR) System, the level control valve at the common discharge header of the flash tank bottom coolers is maintained fully open and the level control valve at the outlet of SGBR demineralizer skid is throttled to control the level of the flash tank.

When the flash tank distillate is aligned to the main condenser or waste test tank (WTT), which may be required to drain the steam generators, the level of the tank is controlled by throttling the level control valve at the common discharge header of the flash tank distillate pumps. High level will alarm and then isolate the discharge from the steam generators to the flash tank.

4. Flash Steam Condenser

The level of the flash steam condenser is maintained by throttling the control valve at the common discharge header of the flash tank distillate pumps. Temperature of the distillate is monitored by the plant computer. High and low level are alarmed in the control room.

5. Flash Tank Distillate Pumps

These pumps are used in two different stages of plant operation:

(a) Draining the steam generators via the flash tank, transferring flash tank concentrates to waste test tanks (WTT).

(b) Transferring the distillate from flash steam condenser to WTT or main condenser.

These pumps are controlled from the MCB. In both modes of operation, only one pump is run at a time. In mode (a) the pump is started manually and is stopped automatically on low-low flash tank level. In mode (b), the starting of the aligned pump is automatically initiated by high level in the distillate condenser and stopped manually. When pressure in the flash steam condenser reaches normal operating condition, the distillate circuit is manually aligned to the main condenser (or WTT) and the pump is stopped and isolated. Pump trip is alarmed in the control room. Rotation of pump duty is administratively controlled.

6. Flash Steam Condenser Vent Valve

The valve remains normally closed, and opens automatically at predetermined pressure and closes on high-high pressure to prevent ingress of steam in the vent gas system. The valve can be manually opened from the MCB by overriding pressure interlocks.

7. Radiation Monitoring

The flash tank liquid discharge to the environment is measured continuously and recorded and totalized at the MCB. Totalized flow reading is taken every month and then manually reset. This stream is continuously monitored for radioactivity and is isolated on high radiation and on high flash tank concentrates discharge temperature. High radioactivity is alarmed locally and at MCB. Additionally, high radioactivity in the blowdown sample lines isolates this flash tank liquid discharge stream. All these high radioactivity interlocks, with the exception of high radiation and temperature flash tank concentrate discharge stream, can be manually overridden. This flash tank liquid discharge radiation monitor is reset by flushing the sampling line and draining the trapped liquid to the floor drain tanks. The high temperature interlock is reset by re-establishing cooling water flow through the flash tank bottoms cooler.

b. Evaporator Subsystem

The evaporator subsystem, of the Steam Generator Blowdown System, is not immediately available for use. Prior to any anticipated startup of this subsystem, plant management would be notified to plan for any training, procedure updates, and pre-operational testing required.

Control and instrumentation for all these evaporators are located at the waste management system (WMS) control panel in the Waste Processing Building. Each evaporator can be individually controlled. Normally, the evaporators operate automatically. The initial starting is, however, manual via a master selector switch.

The prime objective of the evaporator control system is to maintain the level constant by manipulating the feed and the auxiliary steam. This is achieved by controllers at the WMS control panel. Controllers also maintain auxiliary control loops, such as evaporator pressure, distillate and concentrates cooling water temperature, at stable values. In case any one of these controllers are lost, a manual control with aid of backup instrumentation maintains evaporator operation. Additionally, instrumentation for feed, temperature, pressure and level of evaporation is provided at the WMS control panel.

The following conditions are alarmed on the local control panel:

1. High, high-high, low pressure and low-low level in the evaporators
2. High and low level in the distillate condensers
3. High vent temperature in the distillate condensers
4. High and low temperature of concentrate outlet at bottoms coolers
5. High distillate cooler outlet temperature and conductivity
6. High auxiliary steam flow
7. Pumps trip
8. Evaporator bottoms pump seal water pressure low.

All low, low-low level, temperature and pressure alarms are suppressed in shutdown mode.

c. <u>Demineralizer Subsystem</u>

The demineralizer subsystem, along with its regenerative system and controls, is located in a separate room next to the Waste Processing Building. Cooled unflashed liquid is directed to the demineralizer. If the liquid temperature is higher than a predetermined value (120°F), the service valve will be closed and the liquid will bypass the demineralizer. The influent and effluent conductivities of each demineralizer are measured by conductivity cells and recorded by a multipoint recorder.

Regeneration is initiated by push-button and carried forward automatically until completed. Units are returned to service

TABLE 10.4-3

STEAM GENERATOR BLOWDOWN SYSTEM MALFUNCTION/FAILURE ANALYSIS

Component	Accident or Malfunction	Comments and Consequences
System	LOCA/LOOP**	The system does not have to be operational during accident conditions. Therefore it can be shutdown.
Pressure Vessels	Overpressure	Automatic controls and safety relief valves are provided.
System	Failure to Function	If the flash tank is out of service for repairs, the blow-down will have to be stopped. Evaluation of secondary chemistry will need to be used for outage time; this might eventually result in a unit shutdown.
Tanks & Piping	Rupture	The safety relief valves on the pressurized systems are set at pressures below the design pressures considering reasonable transients in the system. In spite of this, should a rupture occur, safety related structures and equipment will not be flooded. The portion of the piping within the Containment is designed to safety class 2 and is designed/supported for the corresponding seismic and other loads.
Instrumen-tation	Malfunction	Two level instruments, one for process control and indication and the other for indication and the other for indication and alarm, are provided on all the essen-tial equipment of the process. Moreover, the I&C are located outside the boron concentration areas to provide easy access during operation.
Containment isolation valves	Air or Electrical power failure to solenoid valve	Valves are designed to fail closed.
	Failure of one train "T" signal	Dual solenoid valves are provided. Each independently receive a "T" signal from the A and B Train respectively.

** LOOP = Loss of Offsite Power.

PID- 1-AR-B20744

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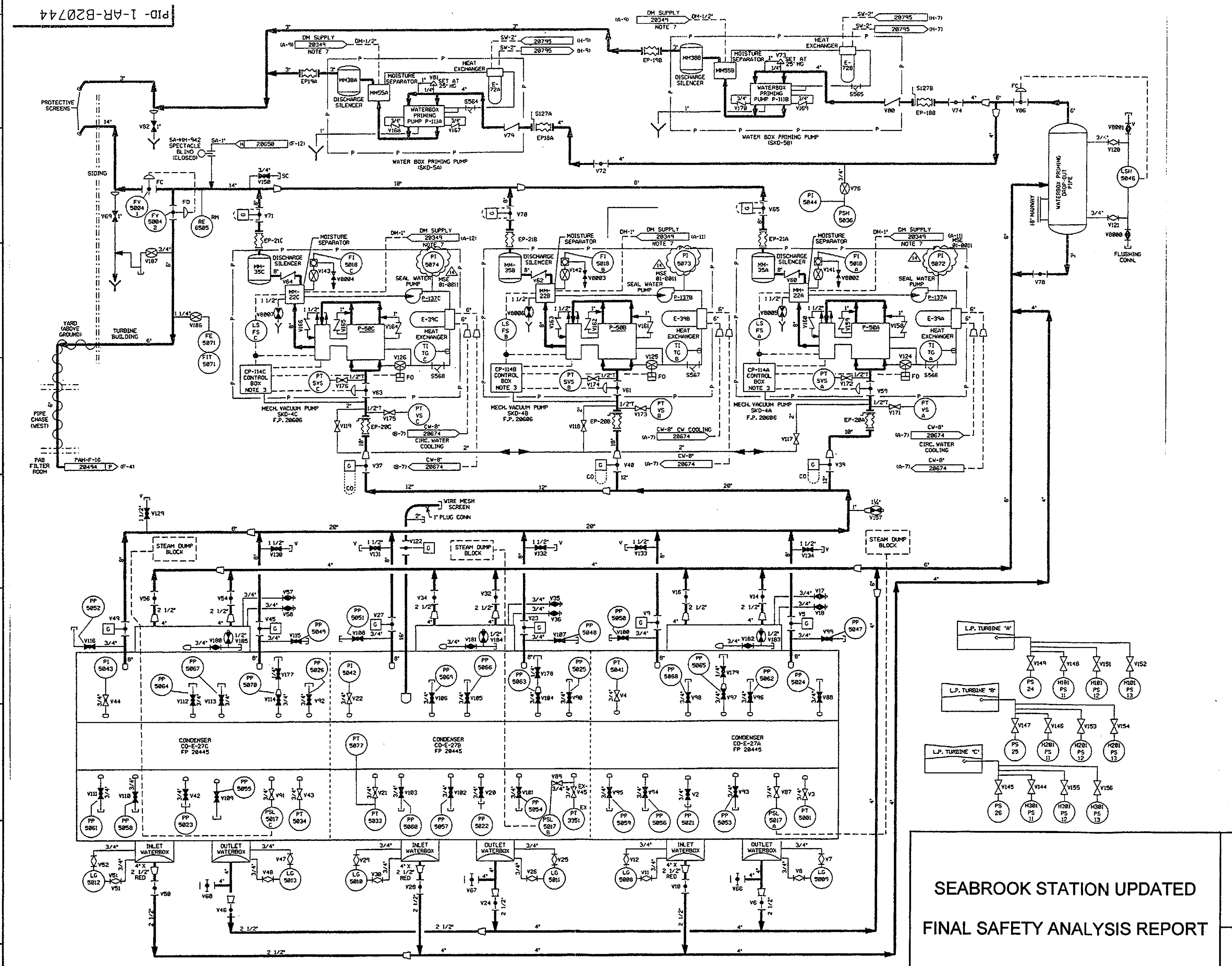

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS
REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1. AR IS A SINGLE DRAWING SYSTEM. AN OVERVIEW IS NOT REQUIRED.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX AR, UNLESS NOTED OTHERWISE.
3. REFER TO FP 20606 AND 20607 FOR DETAILS.
4. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS, UNLESS NOTED OTHERWISE.
5. △-INDICATES REVISION LEVEL.
6. FOR DEMINERALIZED WATER SUPPLY DETAILS TO MOISTURE SEPARATORS AR-MM-22A, 22B, 22C, 55A & 55B, SEE DWG 1-DM-20349

GENERAL REFERENCES:
SD-2, CONDENSER AIR EVACUATION SYSTEM DESIGN DESCRIPTION.
LOGIC DIAGRAM M-503140
LOOP DIAGRAM M-506049

REF. DRAWING NO.	REV	TITLE
9763-F-202084	14	HEATER MISC. VENTS AND DRAINS P&ID
FP-20445	15	CONDENSER PENETRATIONS
9763-F-202093	10	COND. AIR EVAC. SYSTEM P&ID

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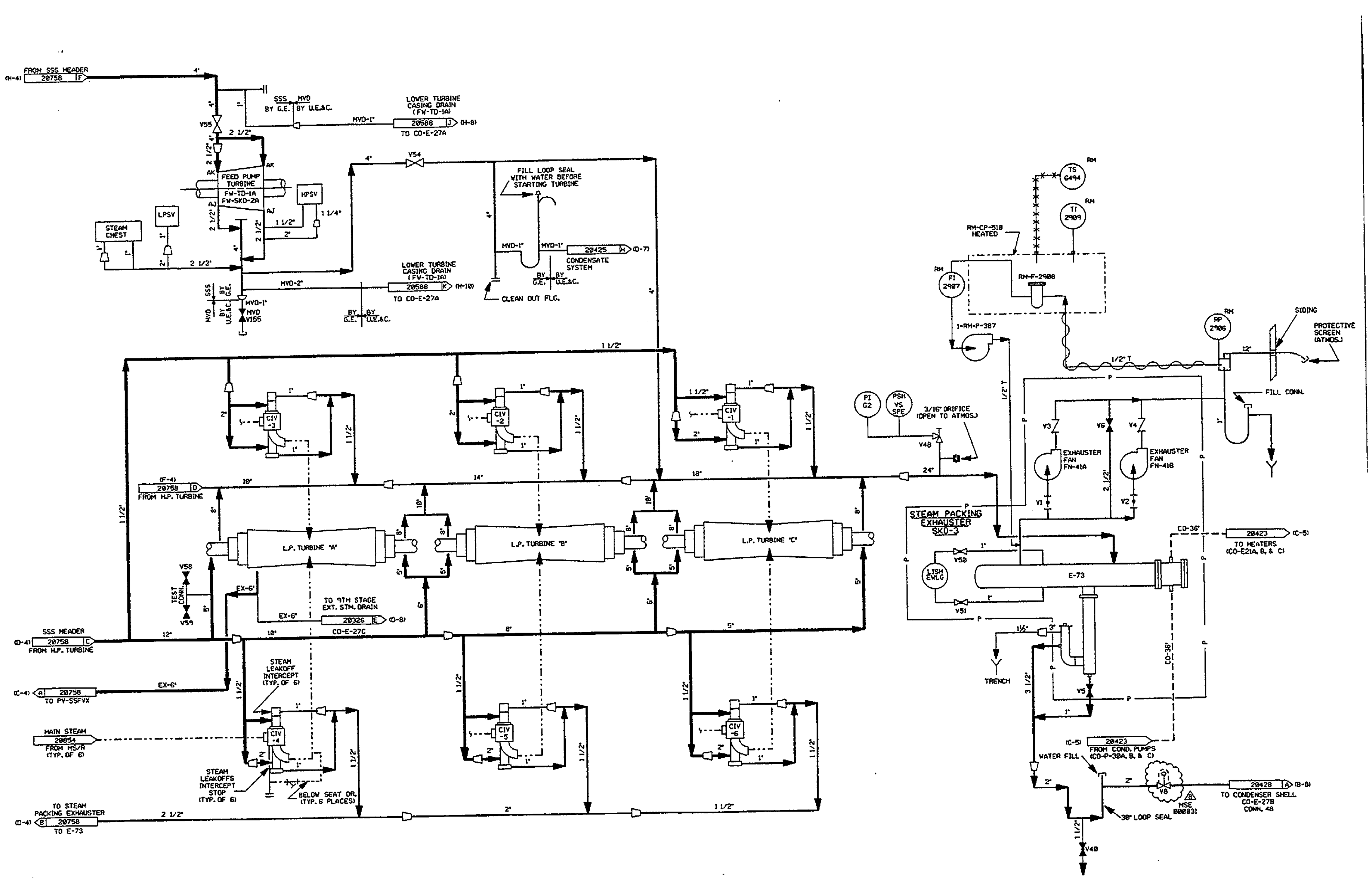
CONDENSER AIR EVACUATION SYSTEM P&ID

1-AR-B20744

FIGURE 10.4-1

PID- 1-SSS-B20759

REV. 07



FOR P&ID REFERENCE DRAWINGS. SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20756 & 20758.
2. ALL LINES, EQUIPMENT, COMPONENTS, AND INSTRUMENTS HAVE SYSTEM PREFIX 1-SSS, UNLESS NOTED OTHERWISE.
3. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS "NNS" UNLESS NOTED OTHERWISE.
4. INDICATES REVISION LEVEL.
5. DELETE
6. ---- DOTTED & DASHED LINES DENOTE MAIN STEAM PIPING.

REF. DRAWING NO.	REV	TITLE
9763-FP-20735	8	FEED PUMP TURBINE
9763-FP-20416	1	STEAM PACKING EXHAUSTER
9763-FP-20415	10	T-G STEAM SEAL SYSTEM DIAGRAM
9763-F-202080	11	EXTRACTION STEAM-P & I DIAGRAM
9763-F-202077	15	CONDENSATE SYSTEM-P & I DIAGRAM
9763-F-202084	14	HEATER MIST. VENTS & DR.-P & I DIAGRAMS
9763-F-202087	8	FEED PUMP TURBINE DRAINS-P & I DIAGRAMS
9763-F-202075	14	MAIN STEAM SHT 2 OF 2 P&ID
9763-F-202090	5	STEAM SEAL SYSTEM-P & I DIAGRAMS

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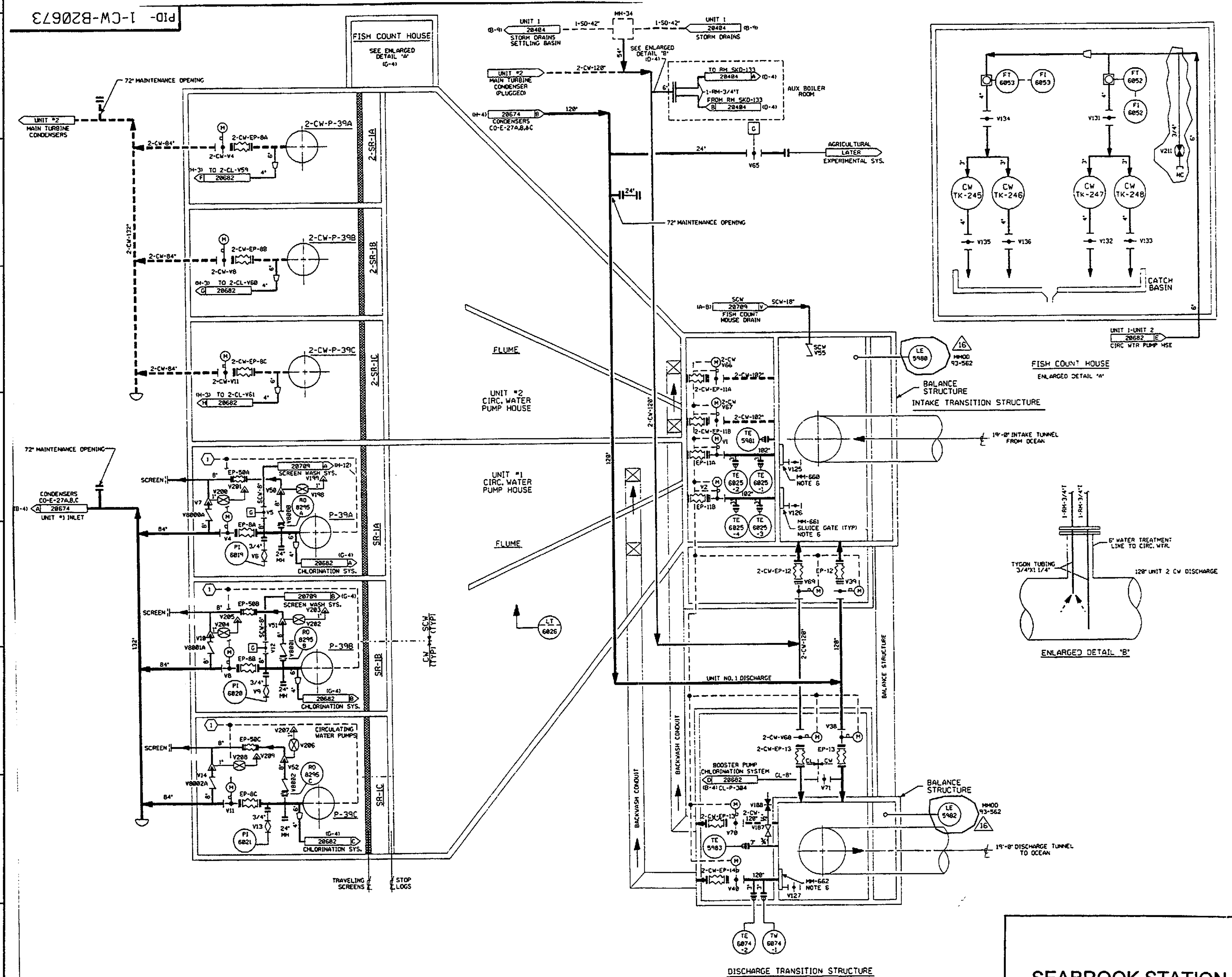
TURBINE STEAM SEAL SYSTEM DETAIL

1-SSS-B20759

FIGURE 10.4-2

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1.) WORK THIS DRAWING WITH DRAWINGS 20672, 20674 & 20675.
2.) ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX 1-CW, UNLESS NOTED OTHERWISE.
3.) ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS, UNLESS NOTED OTHERWISE.
4.) V66 THRU V70 ARE PHYSICALLY LOCATED IN UNIT #2, BUT CARRY UNIT #1 DESIGNATION SINCE THEY ARE CONTROLLED AND POWERED BY UNIT #1.
5.) 1 LUBE WATER FLOW NORMAL, SCREEN DIFFERENTIAL NORMAL AND CONDENSER WATER BOX VACUUM BREAKER VALVES CLOSED.
6.) SLUICE GATES ARE DESIGNED TO BE USED INTERCHANGEABLY IN ANY SET OF GUIDES OF THE SPECIFIED SIZE.
7.) DASH LINES DESIGNATE UNIT #2 PIPING.
8.) △ - INDICATES REVISION LEVEL
9.) DELETE
10.) VALVE V187 IS ON A COUPLING WELDED TO THE THERMOWELL SLEEVE. REF. SPEC. 252-8 SH. 41

9763-F-202481	6	CIRCULATING WATER P & ID
REF. DRAWING NO.	REV	TITLE

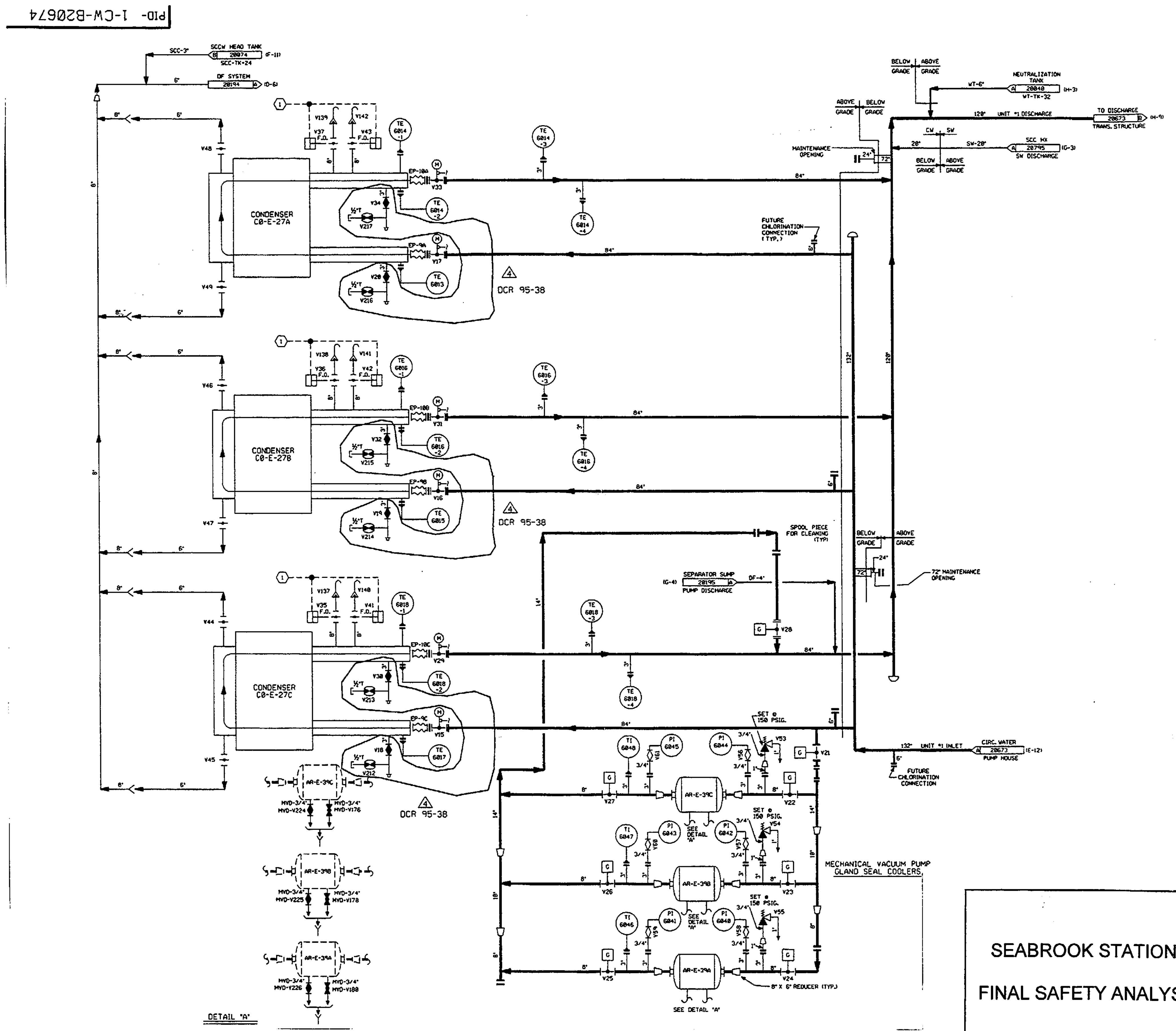
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT

CIRCULATING WATER DETAIL

1-CW-B20673

FIGURE 10.4-4 SH 1

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1.) WORK THIS DRAWING WITH DRAWINGS B20672, B20673 & B20675.

2.) ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX 1-CW, UNLESS NOTED OTHERWISE.

3.) ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS UNLESS NOTED OTHERWISE.

1 LUBE WATER FLOW NORMAL, SCREEN DIFFERENTIAL NORMAL AND CONDENSER WATER BOX VACUUM BREAKER VALVES CLOSED.

4. △ INDICATES REVISION LEVEL.

REF. DRAWING NO.	REV	TITLE
9763-F-202088	9	MISC. VENTS AND DRAINS
9763-F-202481	6	CIRCULATING WATER P & ID

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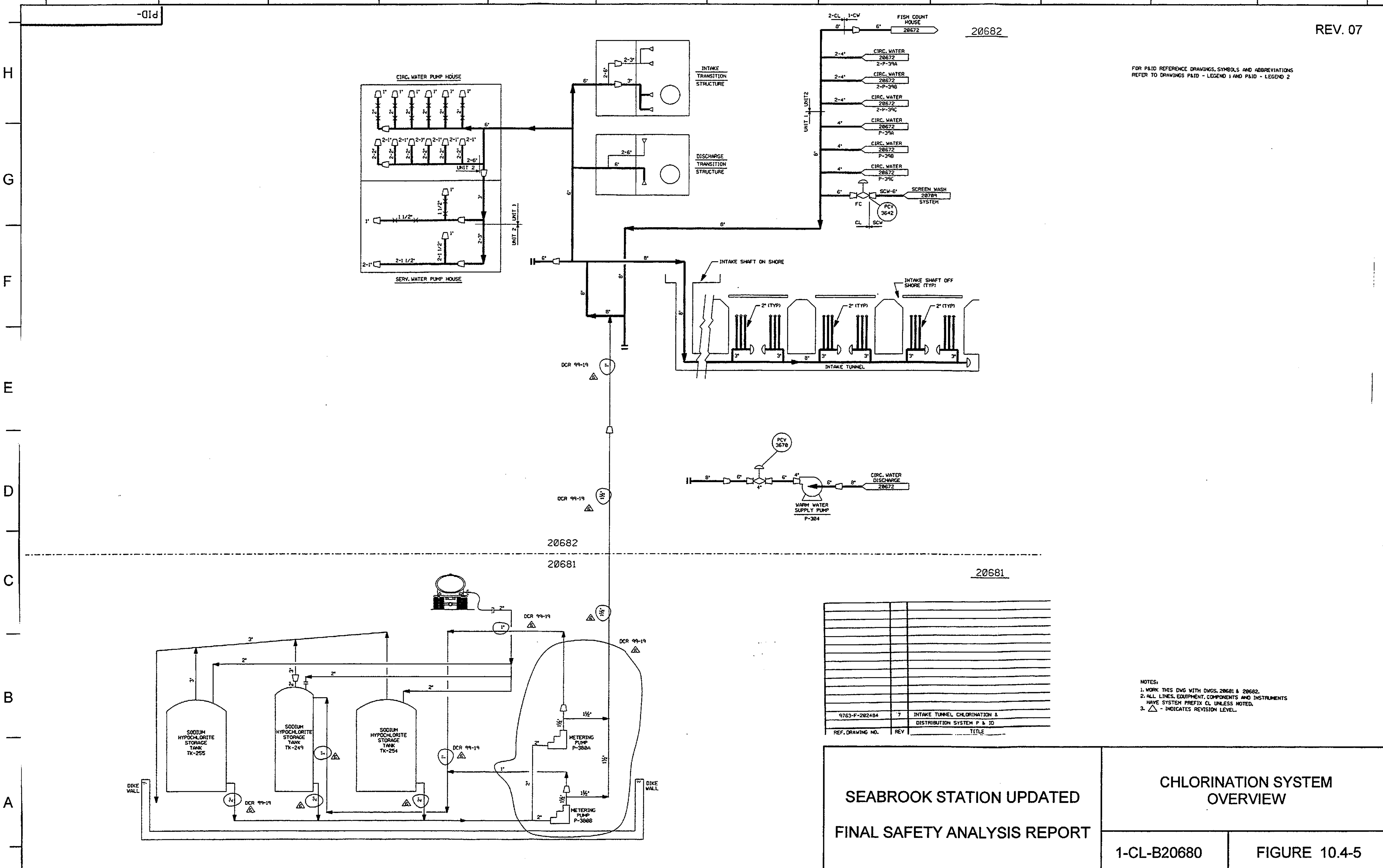
CIRCULATING WATER DETAIL

1-CW-B20674

FIGURE 10.4-4 SH 2



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FINAL SAFETY ANALYSIS REPORT

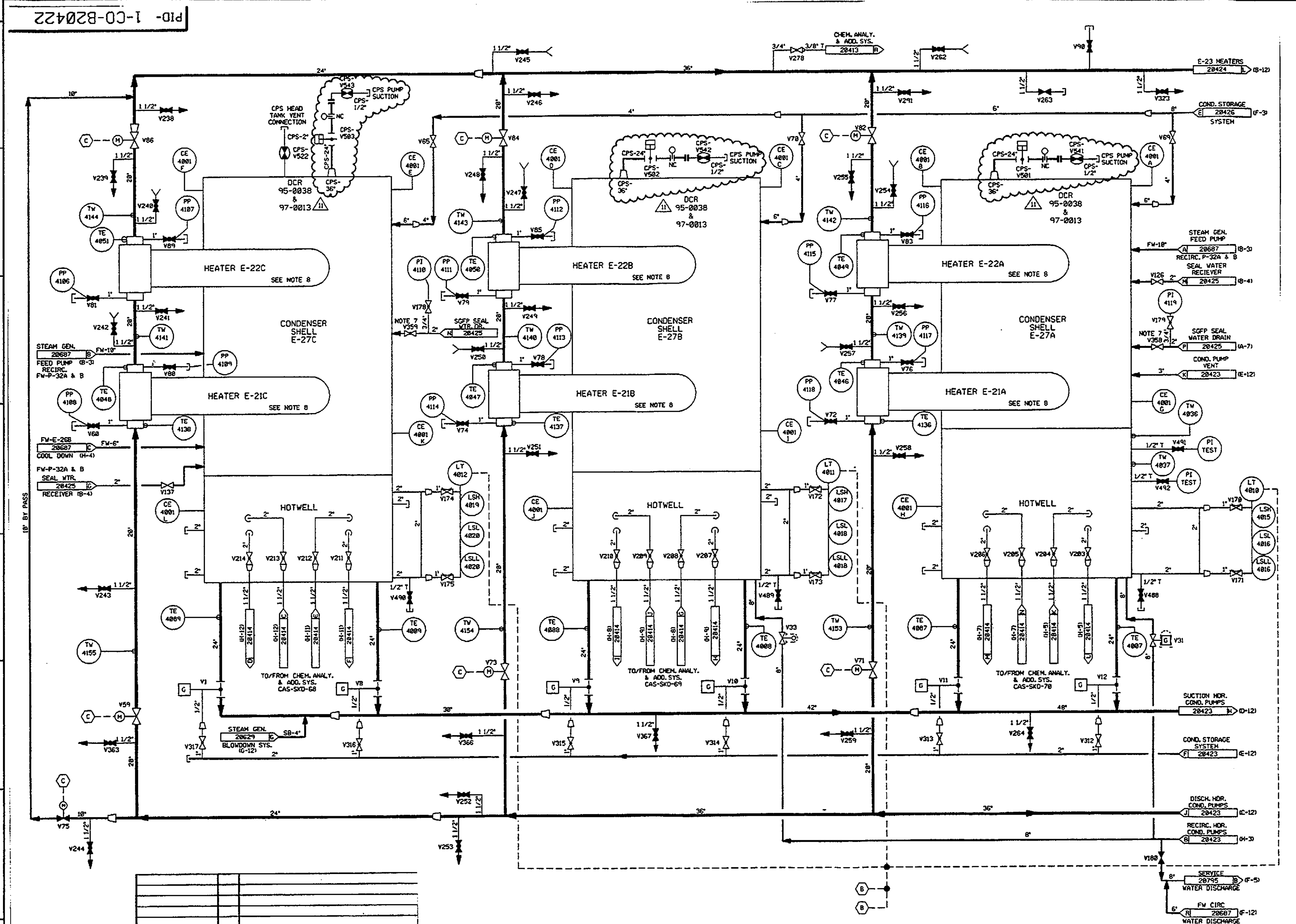
CHLORINATION SYSTEM OVERVIEW

1-CL-B20680

FIGURE 10.4-5

03/27/00 REV. 06

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID-LEGEND 1 AND P&ID-LEGEND 2.

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20420, 20423 THRU 20427.
2. ALL LINES, VALVES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX CO. UNLESS NOTED OTHERWISE.
3. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS. UNLESS NOTED OTHERWISE.
4. INSTRUMENTATION REFERENCES:
   B TO COND. STORAGE SYSTEM, 1-CO-LV-4014A & B ON DWG CO-20426.
   C HIGH-HIGH LEVEL TO CLOSE RESPECTIVE HEATER COND. INLET TO NO. 1 & OUTLET OF NO. 2 WHILE OPENING BYPASS VALVE 1-CO-V75.
5. DELETED
6. △ - INDICATES REVISION LEVEL
7. VALVES 358 & 359 TO BE LOCKED IN A THROTTLED POSITION TO PROVIDE A 40 PSIG NOMINAL BACK PRESSURE.
8. FOR OTHER NOZZLES ON HEATERS 1-CO-21A,B&C AND 1-CO-E-22A,B&C SEE PID'S
   CO-20428
   EX-20326
   EX-20327
   HD-20339
   HD-20340
   HD-20341

REF. DRAWING NO.	REV	TITLE
9763-F-202481	6	CIRCULATING WATER P & ID
9763-F-202077	18	COND. SYS. P&ID SHT 1

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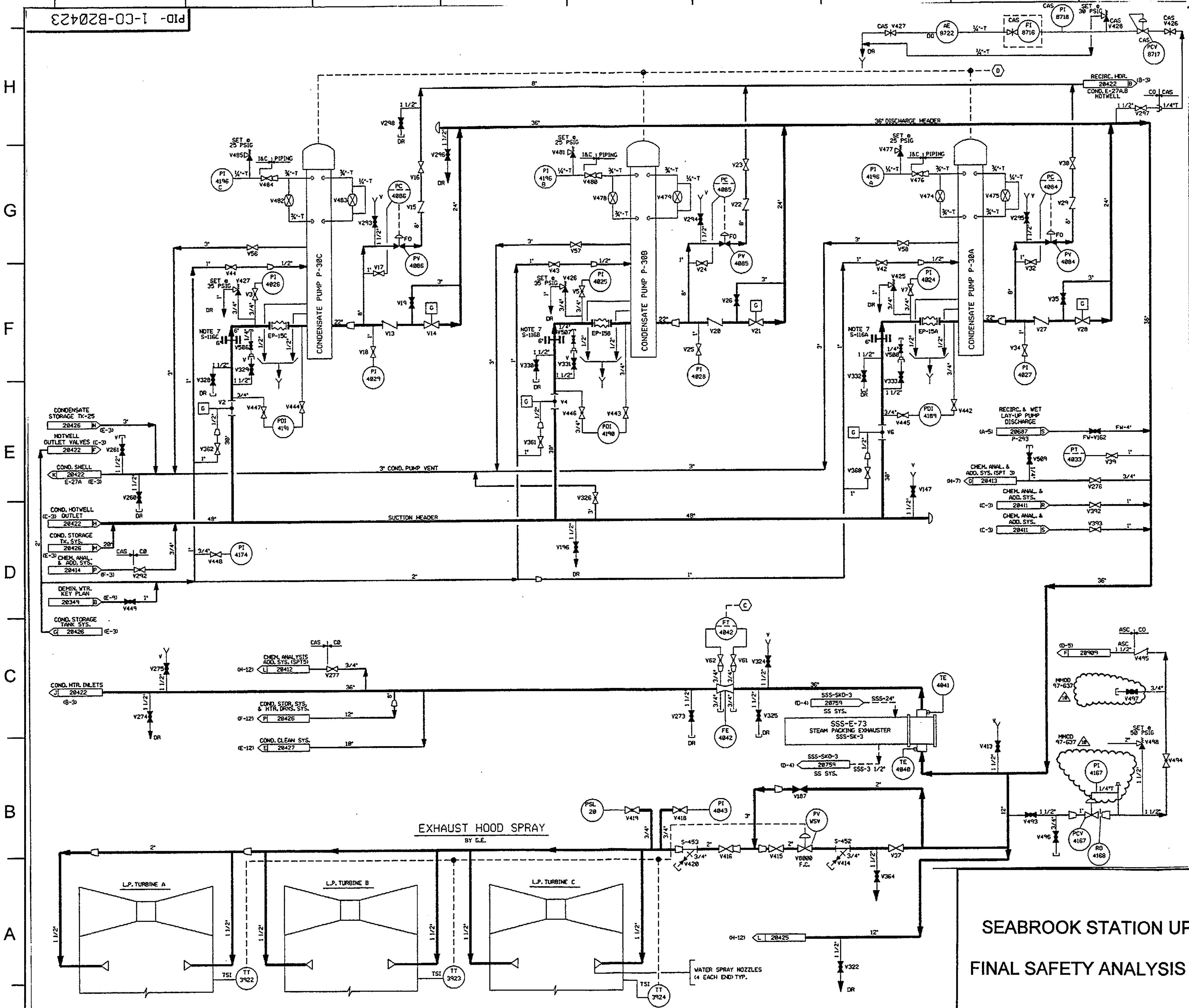
CONDENSATE SYSTEM DETAIL

1-CO-B20422

FIGURE 10.4-7 SH 1

PID- 1-CO-B20423

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20420, 20422 & 20424 THRU 20427.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX CO. UNLESS NOTED OTHERWISE.
3. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS. UNLESS NOTED OTHERWISE.
4. INSTRUMENTATION REFERENCES:
   - (C) TO COND. STORAGE SYS. 1-CO-FV-4042 ON DWG. 20426.
   - (D) TO FEEDWATER PUMP SUCTION 1-CO-PSL-4034 & 4035 DWG CO-20425.
5. △ - INDICATES REVISION LEVEL.
6. DELETED
7. PERMANENT SPOOL PIECE INSTALLATION IS SHOWN. SUCTION STRAINER IS USED DURING INITIAL OPERATION. STRAINER CAN BE REMOVED FOLLOWING CO SYSTEM CLEAN-UP.

REF. DRAWING NO.	REV	TITLE
9763-F-202077	18	COND. SYS. P&ID SHT. 1

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FINAL SAFETY ANALYSIS REPORT

CONDENSATE SYSTEM DETAIL

1-CO-B20423

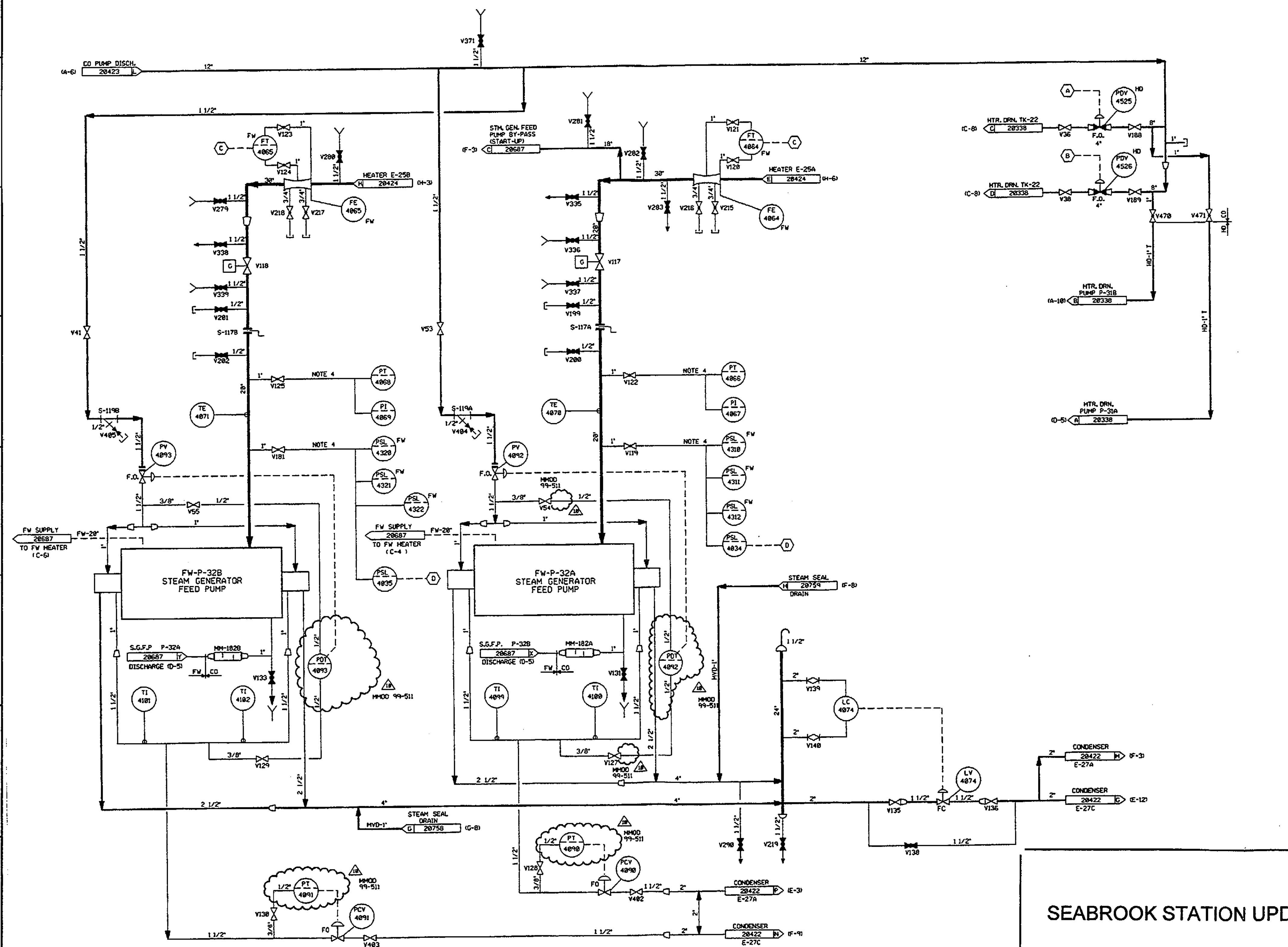
FIGURE 10.4-7 SH 2

07/26/00 REV. 10

PID- 1-CO-B20425

REV. 07

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2



NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20420, 20422, 20423, 20424, 20426 THRU 20430.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX CO UNLESS NOTED OTHERWISE.
3. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS. UNLESS NOTED OTHERWISE.
4. REFERENCE TEST LOOP DWGS. FOR VALVE I.D.
5. INSTRUMENTATION REFERENCES:
   - A FROM HD-PDY-4525-2 LOCATED IN HD-CP-551.
   - B FROM HD-PDY-4526-2 LOCATED IN HD-CP-551.
   - C TO RECIRCULATING VALVES FV-4064 & 4065.
   - D TO CONDENSATE PUMP MOTORS 1-CO-P-30A, B&C ON DWG. CO-20423.
6. △ - INDICATES REVISION LEVEL

REF. DRAWING NO.	REV	TITLE
9763-F-202083	14	HTR. DRN. SYS. P&ID SHT. 2 OF 2
9763-F-202089	10	COND. CLEANING & CHEM. FEED SYS. P&ID
9763-F-202078	11	COND. SYS. P&ID SHT. 2
9763-F-202077	16	COND. SYS. P&ID SHT. 1

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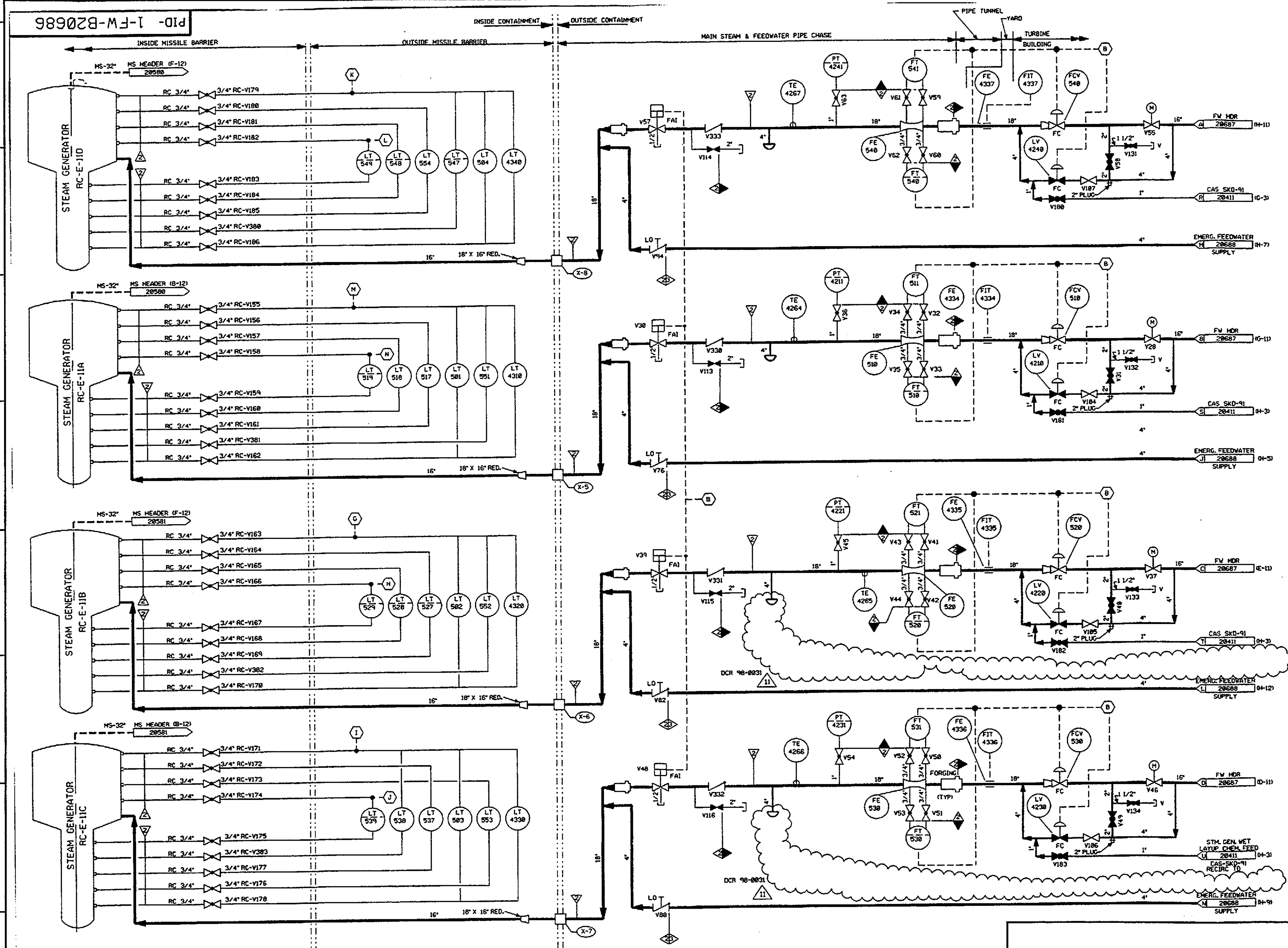
FINAL SAFETY ANALYSIS REPORT

CONDENSATE SYSTEM DETAIL

1-CO-B20425

FIGURE 10.4-7 SH 4

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID-LEGEND 1 AND P&ID-LEGEND 2.

NOTES:

1. PIPING, VALVES, EQUIPMENT & INSTRUMENT TAG NOS. ARE PREFIXED BY DESIGNATION "1-FW", UNLESS OTHERWISE NOTED.
2. WORK THIS DWG. WITH 20684, 20685, 20687 & 20688.
3. INSTRUMENT IDENTIFICATION
   - B PROTECTION SYS. FW ISOLATION SIGNAL.
   - G TO FT-523 ON DWG. 1-MS-20581
   - H TO FT-522 ON DWG. 1-MS-20581
   - I TO FT-533 ON DWG. 1-MS-20581
   - J TO FT-532 ON DWG. 1-MS-20581
   - K TO FT-542 ON DWG. 1-MS-20580
   - L TO FT-543 ON DWG. 1-MS-20580
   - M TO FT-512 ON DWG. 1-MS-20580
   - N TO FT-513 ON DWG. 1-MS-20580
4. VENTS, DRAINS AND TEST CONNECTION CODE BREAKS ARE AT THE DOWNSTREAM END OF THE OUTER ISOLATION VALVE PER 1-NHY-805111, UNLESS NOTED OTHERWISE.

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

REF. DRAWING NO.	REV	TITLE
9763-F-805006	18	REACTOR COOLANT SYS. LOOP-4 P & ID
9763-F-805005	15	REACTOR COOLANT SYS. LOOP-3 P & ID
9763-F-805004	16	REACTOR COOLANT SYS. LOOP-2 P & ID
9763-F-805003	19	REACTOR COOLANT SYS. LOOP-1 P & ID
9763-F-202079	18	FEEDWATER SYSTEM P & I DIAGRAM

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FINAL SAFETY ANALYSIS REPORT

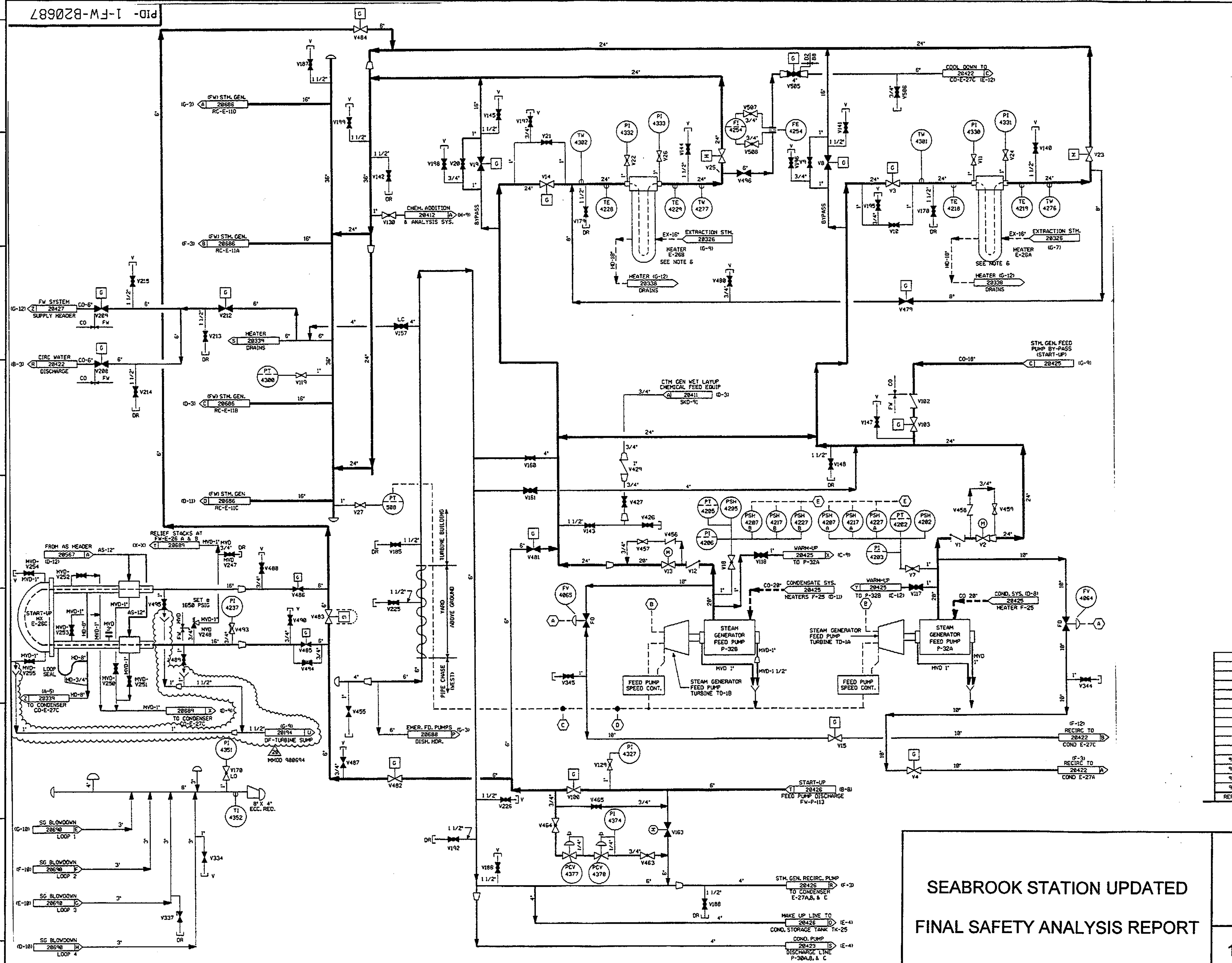
FEEDWATER SYSTEM DETAILS

1-FW-B20686

FIGURE 10.4-9 SH 1

PID- 1-FW-B20687

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1. WORK THIS DRAWING WITH 20684 THRU 20686 & 20688, 20689 & 20990.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX "FW", UNLESS OTHERWISE NOTED.
3. INSTRUMENT REFERENCES:
   - A FEEDWATER PUMP FLOW (LOW)
   - B PROTECTION SYS. FW ISOLATION SIGNAL
   - C MAIN STEAM HEADER PRESSURE
   - D TOTAL STEAM FLOW
   - E TRIPS FEEDWATER PUMPS ON 2/3 HIGH PRESSURE
4. ALL LINES, EQUIPMENT, COMPONENTS & INSTRUMENTS ARE SAFETY CLASS NNS, UNLESS OTHERWISE NOTED.
5. △ INDICATES REVISION LEVEL.
6. FOR OTHER NOZZLES ON HEATERS 1-FW-E-26A&B SEE PID'S:
   EX-20326
   FW-20689
   HD-20338
   MD-20475

REF. DRAWING NO.	REV	TITLE
9763-F-202078	12	CONDENSATE SYS. SHEET 2-2, P & ID
9763-F-202089	11	COND. CLEAN & CHEM. FEED SYS. P & ID
9763-F-202077	18	CONDENSATE SYSTEM P & ID
9763-F-202079	18	FEEDWATER SYSTEM P & ID

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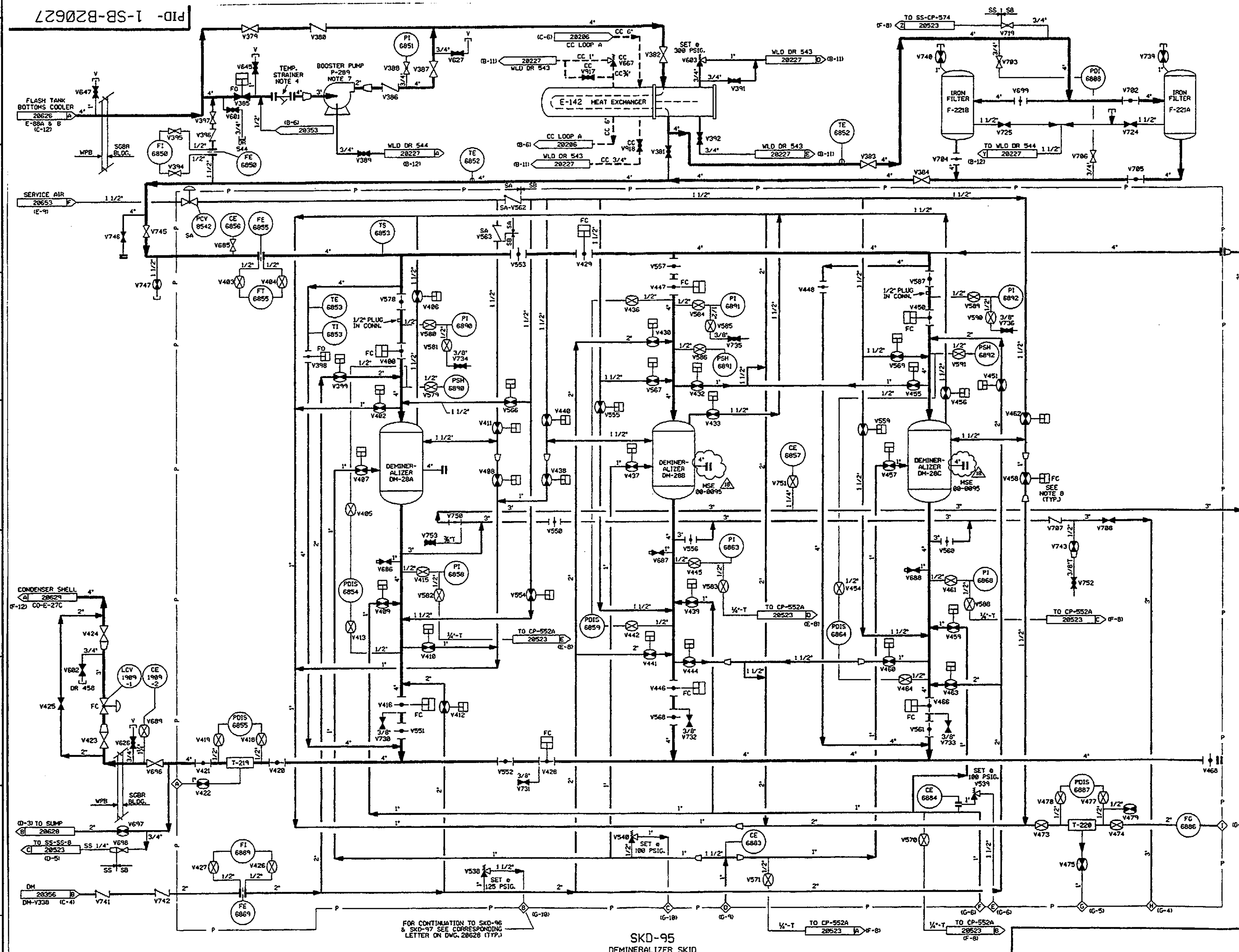
FINAL SAFETY ANALYSIS REPORT

FEEDWATER SYSTEM DETAILS

1-FW-B20687

FIGURE 10.4-9 SH 2



REV. 07

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABREVIATIONS REFER TO DRAWINGS P&ID-LEGEND 1 & P&ID-LEGEND 2.

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20625, 20626, & 20628 THRU 20631.
2. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX SB, UNLESS NOTED OTHERWISE.
3. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS ARE SAFETY CLASS NNS UNLESS NOTED OTHERWISE.
4. STRAINER TO BE REMOVED DURING NORMAL PLANT OPERATION.
5. △ - INDICATES REVISION LEVEL.
6. DELETED
7. SEE DRAWING DM-20353 FOR WATER SUPPLY TO PUMP DOUBLE MECHANICAL SEALS. ALSO FOR PUMP 2-P-289 (NOT SHOWN)
8. ALL AIR OPERATED BALL VALVES IN SKD-95 DENOTE (FC) FAIL CLOSED.

REF. DRAWING NO.	REV	TITLE
9763-F-805637	8	STM. GEN. BLDWN. RECOY. SYS. SHT 4 P&ID
9763-F-805638	4	STM. GEN. BLDWN. RECOY. SYS. SHT 5 P&ID

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FINAL SAFETY ANALYSIS REPORT

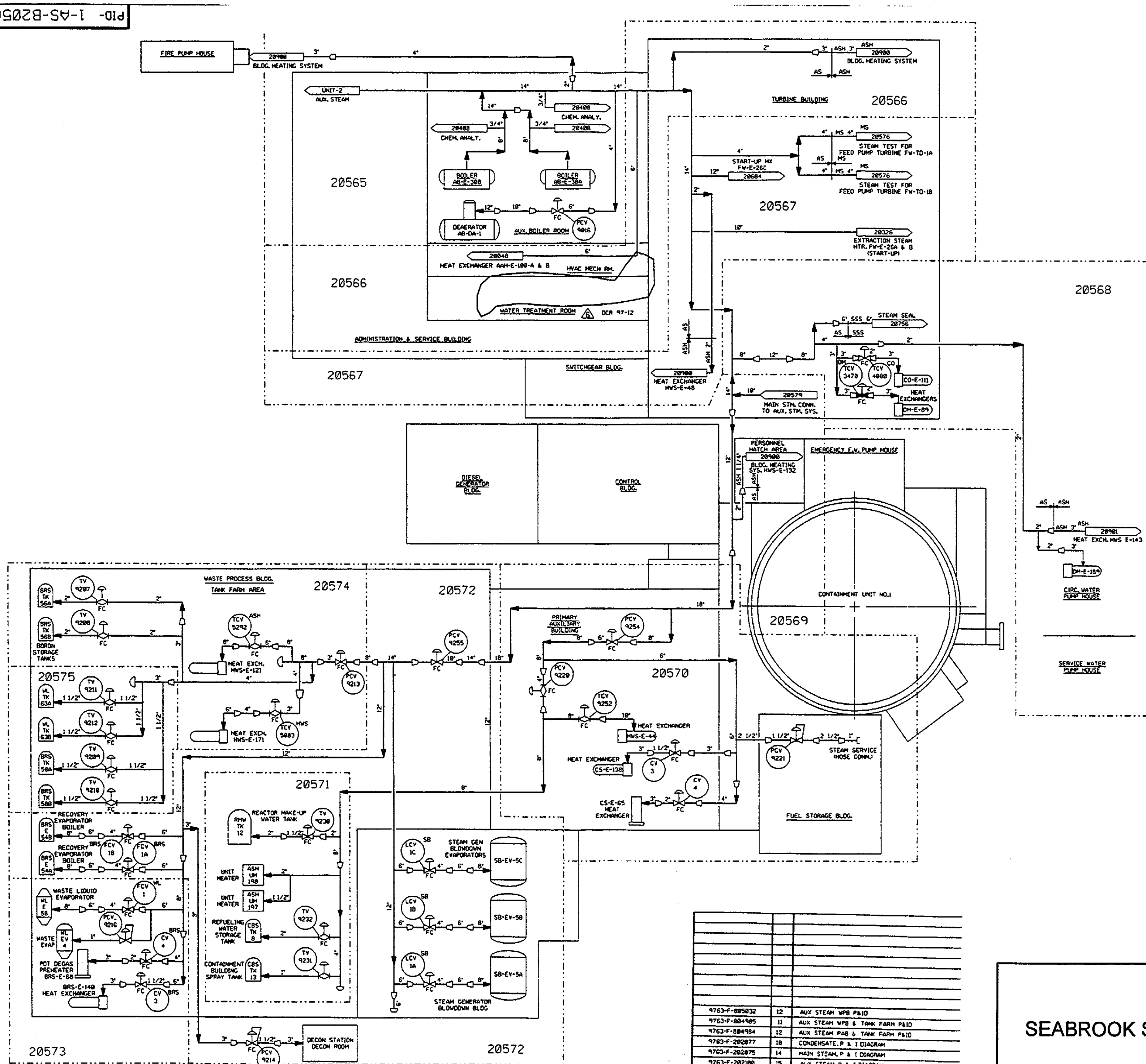
STEAM GENERATOR BLOWDOWN
BLOWDOWN RECOVERY
DETAIL

1-SB-B20627

FIGURE 10.4-13 SH 1

PID- 1-AS-B20564

REV. 07



FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1. ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX 'AS', UNLESS NOTED OTHERWISE.
2. WORK THIS DRAWING WITH DRAWINGS 20565 THRU 20575 & DRAWING 20576.
3. △ INDICATES REVISION LEVEL

REF. DRAWING NO.	REV	TITLE
9763-F-805032	12	AUX STEAM WPB P&ID
9763-F-804985	11	AUX STEAM WPB & TANK FARM P&ID
9763-F-804984	12	AUX STEAM PAB & TANK FARM P&ID
9763-F-202077	18	CONDENSATE, P & I DIAGRAM
9763-F-202075	14	MAIN STEAM, P & I DIAGRAM
9763-F-202100	15	AUX. STEAM, P & I DIAGRAM

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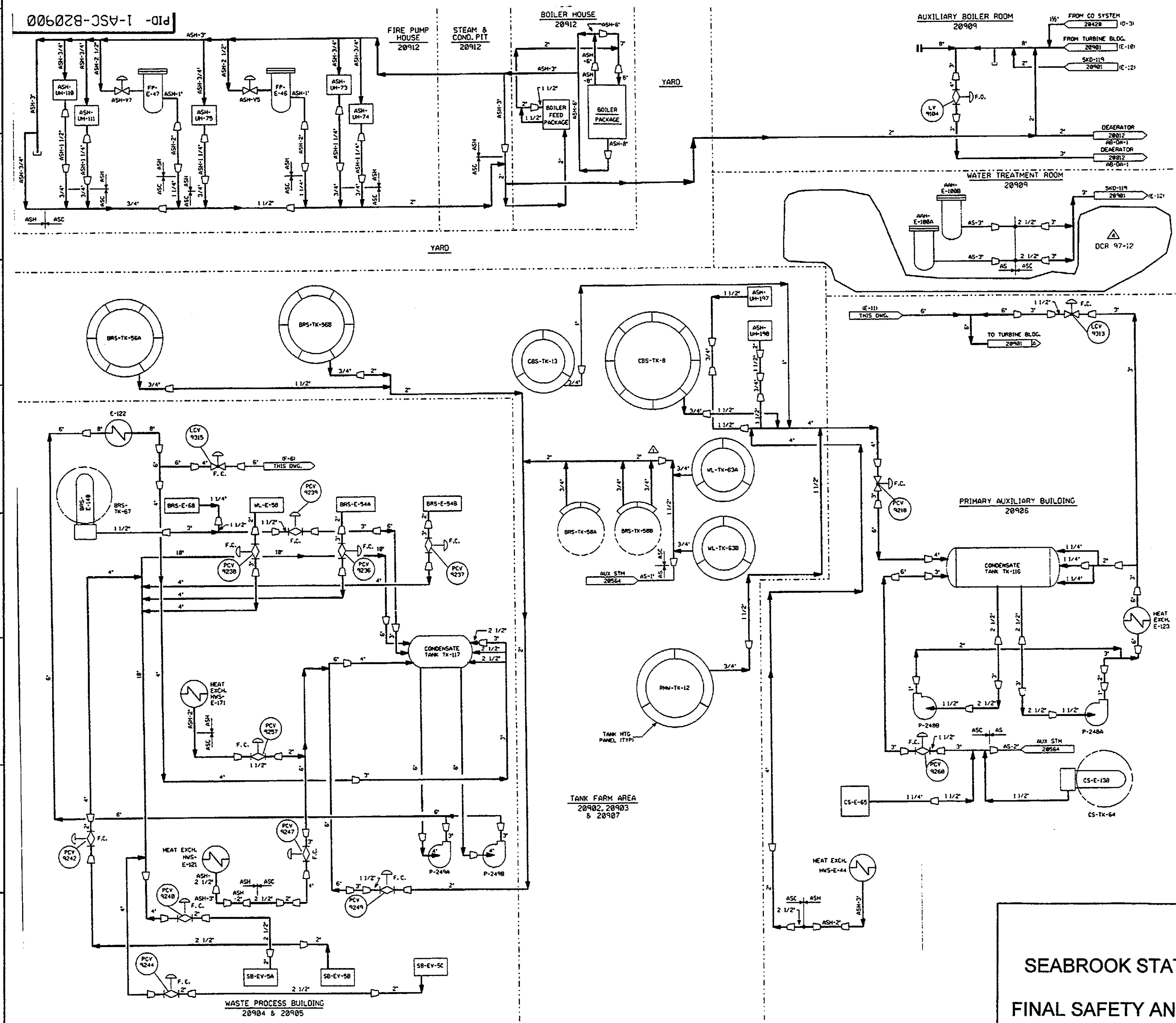
FINAL SAFETY ANALYSIS REPORT

AUXILIARY STEAM OVERVIEW

1-AS-B20564

FIGURE 10.4-16

REV. 07



NOTES:
1. WORK THIS DWG WITH DWGS 20901 THRU 20913, 20925 & 20926.
2. ALL LINES, EQUIPMENT, COMPONENTS & INSTRUMENTS HAVE SYSTEM PREFIX ASC UNLESS NOTED OTHERWISE.
3. △ INDICATES REVISION LEVEL

REF. DRAWING NO.	REV	TITLE
9763-F-604103	9	HOT WTR HTG SYS P&ID
9763-F-604005	9	STM HTG SYS. P&ID
9763-F-202004	14	HEATER MISC VENTS & DRNS P&ID
9763-F-202075	14	MAIN STM P&ID
9763-F-202098	10	AUX BOILER SYS P&ID
9763-F-202100	15	AUX STM & COND P&ID
9763-F-604059	13	FIREPUMP HOUSE HTG & VENT P&ID
9763-F-604519	3	WPB WASTE SOLIDIFICATION AREA HOT WTR HTG
9763-F-805618	10	BORON REC. SYS. PRIMARY DRN TK DEGAS P & ID
9763-F-805032	12	AUX STM & COND WPB P&ID
9763-F-804985	11	AUX STM & COND SYS P&ID
9763-F-604117	9	PAB & FUEL STO. BLDG HOT WTR HTG P&ID
9763-F-805037	11	CHEM & VOL. CONTROL SYS LETDN DEGAS P&ID
9763-F-804984	12	AUX STM COND PAB & TANK FARM P&ID

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FINAL SAFETY ANALYSIS REPORT

AUXILIARY STEAM CONDENSATE SYS OVERVIEW SHT. 1

1-ASC-B20900

FIGURE 10.4-17 SH 1

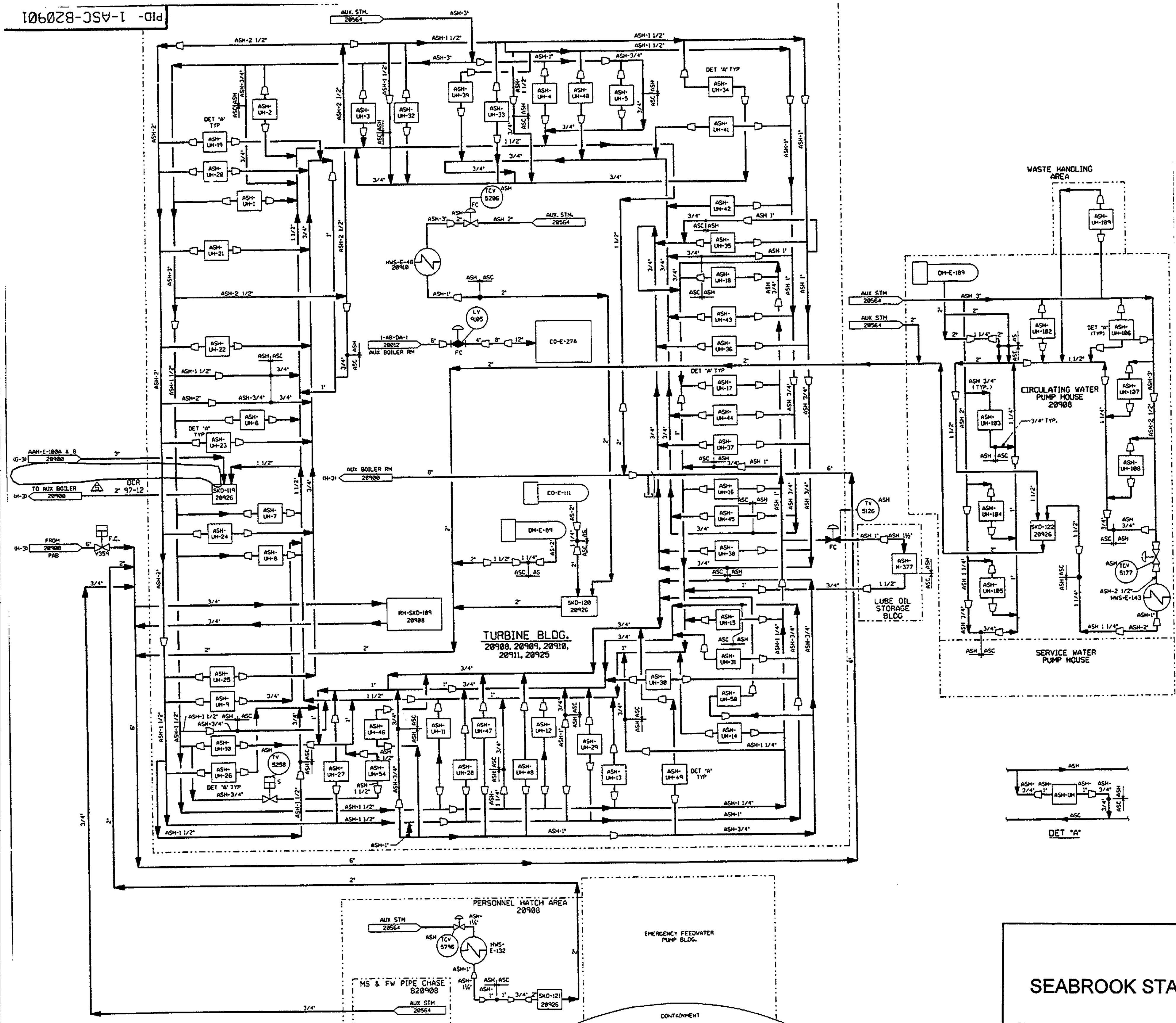
09/14/99 REV. 04

REV. 07

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS
REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1) WORK THIS DWG WITH DWGS 20900 & 20902 THRU 20913, 20925 & 20926.
2) ALL LINES, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX ASC UNLESS NOTED OTHERWISE.
3) △ INDICATES REVISION LEVEL



DET "A"

9763-F-604103	9	HOT WTR HTG SYS P&ID
9763-F-604005	9	STM HTG SYS P&ID
9763-F-202084	14	HEATER MISC VENTS & DRNS P&ID
9763-F-202075	14	MAIN STM P&ID
9763-F-202098	10	AUX BOILER SYS P&ID
9763-F-202100	15	AUX STM & COND P&ID
9763-F-604059	13	FIRE PUMP HOUSE HTG & VENT P&ID
9763-F-604519	3	WPB WASTE SOLIDIFICATION AREA HOT WTR HTG
9763-F-805618	10	BORON REC SYS PRIMARY DRN TK DEGAS P&ID
9763-F-805032	12	AUX STM & COND WPB P&ID
9763-F-804905	11	AUX STM & COND SYS P&ID
9763-F-604117	9	PAB & FUEL STO BLDG HOT WTR HTG P&ID
9763-F-805037	11	CHEM & VOL CONTROL SYS LETDN DEGAS P&ID
9763-F-804984	12	AUX STM COND PAB & TANK FARM P & ID
REF. DRAWING NO.	REV	TITLE

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FINAL SAFETY ANALYSIS REPORT

AUXILIARY STEAM CONDENSATE SYS OVERVIEW SHT. 2

1-ASC-B20901

FIGURE 10.4-17 SH 2

11.1.5 Effluent Releases from Other than Radioactive Waste Systems

This section discusses the effluent releases to the environment from other than radioactive waste systems during normal operations. Estimated releases from anticipated operational occurrences are discussed in Subsections 11.2.3 and 11.3.3.

11.1.5.1 Gaseous Effluent Releases

The Radioactive Gaseous Waste System has been designed to process the gaseous wastes generated in the plant, and is discussed in Section 11.3. Normally anticipated effluent releases from the system will be on a controlled basis. However, any leakage of primary coolant or the process stream either in the Containment or in the auxiliary buildings is collected in the buildings and vented through filtration systems to the environment. Any steam/water leakages in the Turbine Building are directly vented to the environment. The noncondensable gases will be also discharged through the main condenser vacuum system exhaust.

a. Containment Purges

Four purges are expected annually for shutdown, annual fuel loading, and planned maintenance. The duration of shutdown and pre-entry containment purges is expected to be 24 hours per purge. In addition, an online purge system is available for use during power operation. This system will be used to reduce containment airborne activity levels in anticipation of containment entry by plant personnel. To evaluate airborne releases from containment venting, a continuous 1000 scfm online purge rate is used.

b. Steam Generator Blowdown System

Steam Generator Blowdown System waste liquids are processed either through the blowdown demineralizer subsystem or the Liquid Waste Processing System. The normal method of processing blowdown fluids is through the blowdown demineralizer subsystem, which returns the liquid to the main condenser. This includes conditions of minor primary-to-secondary leakage. If necessary, this liquid may also be processed through the installed vendor resin skid (NL-SKD-135). Blowdown flash tank bottoms would be transferred to the floor drains tanks and then through this skid for treatment prior to discharge. The flash tank steam can be either returned to the No. 3 feedwater heater, or processed through the flash steam condenser cooler to the waste test tanks, prior to discharge. Noncondensable gases contained within the Secondary Coolant System are released via the main condenser vent system and as such are considered as part of the gaseous releases source term for the main condenser evacuation system.

c. Primary Auxiliary Building Ventilation

The activity from leakage into the PAB is assumed to be released directly to the environment through the filtration system. The partition factors are 0.0075 for iodines and 1.0 for noble gases. The leakages are mostly equipment leakages, and take place at

TABLE 11.1-3
(Sheet 2 of 2)

PRINCIPAL PARAMETERS USED IN ESTIMATING REALISTIC RELEASES OF RADIOACTIVE MATERIAL IN EFFLUENTS FROM SEABROOK

Liquid Waste Processing Systems

System	Input Flow Rate, gallons per day	Decontamination Factors Iodine	Cesium, Rubidium	Others
Miscellaneous Waste	1360	10^3	10^4	10^4
Equipment Drain	302	10^4	$2x10^4$	10^5
Turbine Building Sump Waste	7200	1	1	1
Boron Recovery	878	10^3	$2x10^3$	10^4
Steam Generator Blowdown (During a primary-to-secondary leak)	$1.1x10^5$	10^2	10^2	10^2

TABLE 11.1-4
(Sheet 1 of 3)

STEAM GENERATOR SECONDARY SIDE EQUILIBRIUM RADIONUCLIDE CONCENTRATIONS

A. Water

Radionuclide	Concentration ($\mu Ci/gm$) Expected Values(1) 0.12% Clad Defects	Design Values(2) 0.25% Clad Defects
H-3	1.00E-03*	1.4E-03
N-16	1.00E-06	-
I-130	1.45E-07	-
I-131	3.33E-05	1.6E-04
I-132	1.04E-05	1.1E-05
I-133	3.51E-05	1.8E-04
I-134	6.12E-07	2.9E-06
I-135	1.01E-05	5.8E-05
Br-83	1.41E-07	-
Rb-86	1.02E-08	-
Sr-89	4.91E-08	3.1E-07
Sr-90	1.20E-09	9.4E-09
Sr-91	3.81E-08	1.0E-06
Y-90	6.66E-10	9.9E-09
Y-91	7.34E-09	4.1E-07
Y-92	-	1.6E-08
Zr-95	7.33E-09	4.5E-08
Nb-95	7.43E-09	4.9E-08
Mo-99	9.88E-06	2.0E-05
Tc-99m	2.24E-05	-
Te-127m	2.18E-08	-
Te-127	1.29E-07	-
Te-129m	1.49E-07	-
Te-129	6.28E-07	-
Te-131m	2.09E-07	-
Te-132	2.54E-06	1.6E-05
Cs-134	2.88E-06	2.8E-05
Cs-136	1.30E-06	1.4E-05
Cs-137	1.92E-06	1.3E-04
Ba-140	2.35E-08	2.4E-07
La-140	3.04E-08	7.0E-08
Ce-144	4.82E-09	2.9E-08
Mn-54	4.82E-08	2.6E-07
Mn-56	-	2.0E-06
Co-58	1.71E-06	8.8E-06
Co-60	2.16E-07	2.6E-07
Fe-59	1.23E-07	3.5E-08
Cr-51	2.00E-07	3.3E-07

* 1.00E-03 = 1.00×10^{-3}

8. All sumps in contaminated plant areas, and the Administration and Service Building sump and the RCA walkway B&C sumps.

Prior to further processing, the pH of the liquid can be adjusted with the liquid waste chemical addition pump by recirculating the tank contents with the floor drain tank pump.

From the floor drain tanks, liquid is transferred by one of the two floor drain tank pumps to one of the following:

1. The waste evaporator for processing

2. The waste test tanks for direct discharge offsite, if the quality and radioactivity levels are within design limits

3. The boron recovery system evaporator (Subsection 9.3.5.2)

4. The waste feed tanks of the Waste Solidification System (Section 11.4)

5. The skid-mounted Waste Liquid Processing System

Overflow and recirculation lines are provided on the floor drain tanks. The operation is manual, except for the automatic shutdown of the floor drain tank pump on low liquid level in the tank.

b. <u>Evaporation</u>

The evaporation equipment of the Liquid Waste System is identical to the Boron Recovery System (see Subsection 9.3.5.2) except that only one liquid waste evaporator is employed, versus two for the BRS, and the concentration in the liquid waste evaporator is generally up to 12 percent TDS, as mentioned in the design bases.

c. <u>Testing and Demineralization</u>

The liquid waste evaporator distillate and the skid-mounted Waste Liquid Processing System effluent are the normal sources of liquid to the testing and demineralization subsystem. Other sources of liquid which enter the waste test tanks are listed below.

1. Liquid directly from the floor drain tanks, when that liquid does not require processing in the evaporator

2. Distillate from the boron recovery evaporator, when that evaporator is substituting for the waste evaporator

3. Flashed steam or bottoms from the steam generator blowdown flash tank and flash steam condensers, when that system must discharge liquid off site (see Subsection 10.4.8)

4. Liquid from chemical drain liquid tanks, when that liquid does not require processing in the evaporator.

A radiation element monitors the liquid entering the waste test tank. Liquid collected in the waste test tanks is transferred by one of two waste test tank pumps to the Circulating Water System intake and discharge transition structures for final disposal. Radiation levels and flow rates are also monitored on this line. If purification is required prior to discharge, the liquid is circulated through the waste demineralizer and filter. If reprocessing is required, the waste test tank contents are pumped back to the floor drain tanks. Overflow and recirculation lines are provided on the waste test tanks.

Liquid from the BRS testing and demineralization subsystem (Subsection 9.3.5.2d) and blowdown from the steam generator blowdown system flash tank (Subsection 10.4.8) can be transferred directly to the Circulating Water System intake and discharge transition structures for final disposal when water quality permits. Wide-range flow control is achieved by two parallel control valves in 3" and ¾" lines. The larger valve is used for a flow range between 10 and 120 gpm, and the smaller valve is used for a flow range between 0 and 15 gpm. One valve always remains closed when the other is open. The flow of filtered demineralized water which is discharged to the Circulating Water System is recorded as well as totalized at the Waste Management System (WMS) control panel.

Operation is essentially manual, with only two protective functions being automatic, i.e., securing of the waste test tank pumps on low test tank level, and termination of offsite discharge on high radiation levels.

11.2.2.2 Component Description

The detailed data is given in Table 11.2-1.

11.2.3 Radioactive Liquid Release

The amount of radioactivity projected to be released in liquid effluents from normal operation, including anticipated operational occurrences, is described below. The radiological impact from such releases is evaluated based on the dose models in Regulatory Guide 1.109 (Revision 1).

11.2.3.1 Normal Operation Release

The sources of radioactive wastes that are to be released are as follows:

- Boron Recovery System discharges for tritium control
- Nonrecyclable liquid releases from the Liquid Waste System
- Secondary system condensate leakage
- Nonrecyclable liquid releases from the steam generator blowdown waste holdup sump

The list of potential sources of liquid to be discharged is reduced to the above because of the processing system design principle to segregate, process, and recycle as much of the liquid extracted from the Reactor Coolant System as possible. The systems provided to carry out this design principle are the Boron Recovery System (Subsection 9.3.5) and the Liquid Waste System, described above. These descriptions illustrate the manner in which the collecting and handling of liquid letdown and leakage from the Reactor Coolant System are segregated from nonrecyclable liquids and sent, after processing, to the reactor makeup water storage tanks for reuse within the Reactor Coolant System.

a. Release Assumptions

The main assumptions and parameters used in estimating the magnitude of radioactive liquid releases are as follows:

1. The radionuclides and their concentrations within the Reactor Coolant System are as listed in Table 11.1-1 under the heading of "0.12 percent cladding defects."

2. The radionuclides and their concentrations within the secondary side of the steam generators are as listed in Table 11.1-4. The feed and condensate system activities are equivalent to the steam activities, excluding noble gases.

3. The decontamination factors (DFs) within the Boron Recovery System and the Liquid Waste System are given in Appendix 11A.

4. The times of radioactive decay between collection, processing and discharge are listed in Appendix 11A for each stream of liquid waste.

5. The reactor is assumed to be operating at 3654 MWt, with an 80 percent capacity factor.

b. Releases

1. Releases from Boron Recovery System

As described in Subsection 11.1.1.3, tritium control considerations anticipate the need for discharging reactor coolant letdown after processing by the Boron Recovery System. The expected volume required for this measure is 200,000 gallons per year. With the input liquid containing radionuclides at Table 11.1-1 values (0.12 percent clad defect), the processing DFs and the radioactive decay times in Appendix 11A, the annual release from this source is 0.033 Ci/year, except tritium. This release is shown by radionuclide in Table 11.2-2.

2. Nonrecyclable Releases from Liquid Waste System

The estimated average volumetric generation rates of nonrecyclable primary system leakage are 40 gallons/day inside the Containment at primary coolant activity (PCA), and 200 gallons/day in the Auxiliary Building at 0.1 PCA. This liquid is collected in the floor drain tanks, which are at the head end of the floor drain portion of the Liquid Waste System. It is also estimated that there are 400 gpd of liquid waste from laboratory drains at 0.002 PCA, 15 gpd from sampling drains at 1.0 PCA, and 700 gpd from miscellaneous waste at 0.01 PCA released into the Liquid Waste System. (There will be no liquid waste generated from laundry operations.) This liquid is collected in the chemical drain treatment tanks (refer to Updated FSAR Subsection 9.3.3.2). The total input rate to the floor drain tank is less than 1355 gallons/day with the effective composite activity concentration of 0.061 PCA. With the processing DFs and radioactive decay times in Appendix 11A, the annual release from this source is less than 0.035 Ci/year, except tritium. This release is shown by radionuclide in Table 11.2-3.

3. Steam Generator Blowdown

With reactor coolant radionuclide concentrations as listed in Table 11.1-1 and an estimated average leakage of 100 lbs/day of reactor coolant through the steam generator tubes, equilibrium secondary side steam generator radionuclide concentrations are calculated as listed in Table 11.1-4. The steam generator secondary side blowdown rate associated with this leakage is 75 gpm, total from all four steam generators.

Blowdown from the four steam generators is processed by the blowdown flash tank, where approximately 30 percent of the blowdown volume is flashed to steam. This steam is normally

routed to the No. 3 feedwater heater or to the main condenser. With no primary-to-secondary leakage, steam may be exhausted to the atmosphere if the heater and condenser are not available. The vent gases from the flash steam condenser are processed before discharge. Radioactive steam is not processed directly to the atmosphere. In the presence of a primary-to-secondary leak, flash tank steam may be sent to the No. 3 feedwater heaters or processed through the flash tank distillate coolers to the waste test tanks, prior to discharge.
The remaining volume of blowdown liquid (70 percent) can be released directly to the environment via the plant Circulating and Service Water System when no primary-to-secondary leakage exists or processed via the blowdown demineralizer subsystem. With significant primary-to-secondary contamination of the secondary side water, the primary method to process radioactive secondary liquid from the steam generators is to direct steam blowdown flash tank bottoms cooler discharge to the floor drain tanks. If no secondary pressure is available, the steam blowdown and wet lay-up pumps can be used. From the floor drain tanks, processing through the installed vendor resin skid (WL-SKD-135) to the waste test tanks may be performed (reference Subsection 11.2.2.1). Additional methods that could be used, with station management approval and planning, include the liquid volume within the flash tank being processed by the Blowdown Evaporator System (Subsection 10.4.8). Two evaporators in parallel are available to process a maximum of 50 gpm of blowdown liquid. Distillate from the evaporators is condensed and directed to the waste test tanks of the Liquid Waste System. Further processing is available within the Liquid Waste System, if required, prior to discharge to the environment via the Plant Circulating Water System. Bottoms from the blowdown evaporators is routinely released to the Solid Waste Processing System for processing and shipment offsite. No credit for collection and processing decay times has been assumed to calculate the liquid releases from this pathway. With the DFs presented in Appendix 11A for the installed vendor resin skid, the annual release from this source is 0.02 Ci/year. This release by radionuclide is presented in Table 11.2-4.

4. <u>Secondary System Condensate Leakage</u>

The estimated average liquid leakage rate of the secondary system is 7200 gallons/day. The leakage is assumed from liquid sources at main steam activity. The concentrations of the main steam activity are listed in Table 11.1-4. This liquid is collected in the Turbine Building floor drain and then is discharged from the plant unprocessed, which results in the annual release of 0.00658 Ci/year, except tritium. This release is shown by radionuclide in Table 11.2-5.

5. Summary of Radioactive Liquid Release From Normal Operation

The total estimated radioactivity to be released due to the generation and release of the liquid streams described above is shown in Table 11.2-6. The total annual release of 0.08 Ci except tritium and 730 Ci of tritium are the expected discharge levels due to normal operation. The tritium releases are discussed in Section 11.1.

11.2.3.2 Releases from Anticipated Operational Occurrences and Design Basis Fuel Leakage

The additional unplanned liquid release due to anticipated operational occurrences is estimated to be 0.15 Ci/year based on reactor operating data over a 2.5 year period, January 1973 through June 1975, representing 102 reactor-years of operation (NUREG-0017). These releases are assumed to have the same isotopic distributions for the calculated source term of the liquid wastes. The annual release from the anticipated operational occurrences is shown by radionuclide in Table 11.2-7.

Table 11.2-8 shows the total annual release by radionuclide from normal operation, including anticipated operational occurrences. The discharge concentrations are compared with (MPC)w, the concentration limits of 10 CFR Part 20, Appendix B, Table II, Column 2. Table 11.2-9 shows the total annual release by radionuclide from design basis fuel leakage (i.e., 1 percent failed fuel).

11.2.3.3 Release Points

The release routes of radioactive liquids generated in operation are shown in simplified flow sheet form in Figure 11.2-5. Liquid processed for discharge by the Boron Recovery System ultimately accumulates in the waste test tanks, located adjacent to the Waste Processing Building, as shown in Figures 1.2-22 and 1.2-25. After sampling for radioactivity analysis, the liquid is discharged from the waste test tank through a process radiation monitor to the Circulating Water System. The tie-in point with the Circulating Water System is as shown in Figure 11.2-4.

Nonrecyclable reactor coolant leakage is collected by the drain system within the Containment and Primary Auxiliary Building and is directed to the floor drain tank of the Liquid Waste Processing System. This tank is located within the Waste Processing Building as shown in Figure 1.2-29. This liquid is processed and directed to either of two waste test tanks. The liquid is then sampled and released as described above for boron recovery system releases.

Miscellaneous liquid wastes (decontamination water, laboratory, drains, etc.) are collected in the chemical drain treatment tank where they can be processed and directed to the waste test tanks for final sampling prior to discharge to the Circulating Water System.

Steam generator secondary side blowdown is directed to the flash tank. If steam generator blowdown activity is low, liquid from the flash tank may be transferred directly to the waste test tank or to the Circulating Water System. Otherwise, liquid from the flash tank is directed to the blowdown demineralizers or the installed vendor resin skid for treatment.

Secondary system condensate leakage is collected in the Turbine Building sumps. From here the liquid is directed to an oil separator and transferred to the circulating water system discharge. A radiation monitor is located on the sump effluent line which automatically isolates the sump at a predetermined radioactive concentration.

All these waste liquids, once released to the Circulating Water System, experience the same release path to the Atlantic Ocean. The radioactive liquid wastes then reach the environment via the circulating water discharge line.

11.2.3.4 Dilution Factors

The discharge route of radioactive liquid to the environment, as described in the above section, provides onsite dilution by the flow of the Circulating Water System, conservatively assumed to be 390,000 gpm (for a discussion of CWS operation, see Subsection 10.4.5). With the assumption of 80 percent operating capacity of the Circulating Water System, this flow dilutes the normal liquid radioactivity releases from 0.24 Ci/year except tritium and 730 Ci/year of tritium, to 3.9×10^{-10} μCi/ml except tritium and 1.2×10^{-6} μCi/ml of tritium.

Further dilution of the circulating water discharge plume will occur after leaving the discharge pipe. A near-field dilution factor of 8 was used for calculating the estimated doses reported in the next section. This dilution factor estimate is based on a multi-port, deep water discharge concept and is conservatively calculated for the tidal cycle surface area above the discharge pipe.

11.2.3.5 Estimated Doses

Radionuclides in liquid effluents from the site pose a potential environmental radiation source to certain individuals or segments of the general public. In general, many possible exposure pathways exist for liquid effluents, however, detailed consideration can be focused on the pathways that pose the greatest risk to public exposure - the "Critical Pathways." The critical pathways are considered to be the internal exposure by the ingestion of fin fish or other seafood harvested for the area affected by released radionuclides and the external exposures from recreational activities along the shoreline.

The radiological assessment of the ingestion pathways and the external exposure pathway is calculated based on the models, parameters, assumptions, and dose conversion factors in Regulatory Guide 1.109 (Revision 1). Table 11.2-10 shows the parameters used in the dose calculation.

The maximum annual doses from all pathways received by an individual in a particular age group due to the estimated liquid discharge radioactivities from normal operation (including anticipated operational occurrences) listed in Table 11.2-8 are shown in Table 11.2-11. Among three age groups, the highest maximum annual doses, 2.5×10^{-3} mrem/yr to the total body and 2.4×10^{-2} mrem/yr to the thyroid (the most critical organ), are small fractions of the numerical design dose objectives of Appendix I to 10 CFR 50.

11.2.4 Design Evaluation

a. General

The Liquid Waste System performs no safety function and is not required for the safe shutdown of the reactor. Accordingly, the system is designated NNS class. Also, because of the noncritical nature of this system, emergency electrical power is not provided.

All waste liquid generated within the plant is processed through the floor drainage system (Subsection 9.3.3), Boron Recovery System (Subsection 9.3.5) or Liquid Waste System, when necessary. Process discharge to the environment is only after testing of the effluent quality.

The evaporators are designed throughout with low-flow velocity and liquid vapor separators to reduce any entrainment of vapor. The evaporators are designed to yield a minimum decontamination factor of 10^4 for nonvolatiles. This will reduce the amount of radioactivity in the distillate to acceptable levels for disposal. The concentrated liquid is processed by the Solid Waste System before disposal offsite. Backup evaporator capacity from the BRS is available. All essential portions of the system are located away from any high energy lines. A negative pressure is always maintained in the vent header so that gases go into the vent header and not into the tank overflow lines. Relief valves and tank overflow lines have been provided to protect against overpressures. Should the high level point be exceeded, the tank overflow is channeled to the floor sump. Each floor sump includes one or more sump pumps which transfer the excess liquid to the floor drain tank. Pumps in the system have low level shutoffs. Filters and demineralizers have pressure indication upstream and downstream to indicate fouling.

Two sets of level instruments, one for process control and indication and the other for indication and alarm, are provided on all the essential equipment of the process. Moreover, the instrumentation and controls are located outside the boron concentrating areas to minimize radiation exposure to operating personnel.

The skid-mounted Waste Liquid Processing System is connected to permanent plant connections in the Waste Liquid System (located (-)3′ elevation WPB).

Radiological consequences from postulated failures of components in the Liquid Waste System are bounded by the results of the liquid waste system failure analyses presented in Subsections 15.7.2 and 15.7.3.

b. Equipment Redundancies

Only one of the two floor drain tank pumps, waste test tanks, waste test tank pumps, and floor drain filters is required at a time. If the liquid waste evaporator subsystem is not available at any time, the liquid can be concentrated in one of the two boron recovery system evaporators, or it can be discharged directly to the waste feed tanks.

TABLE 11.2-4

STEAM GENERATOR BLOWDOWN SYSTEM RELEASES**

Radionuclide	Annual Release (Ci/Year)
I-131	1.10E-3*
I-132	3.22E-3
I-133	3.23E-3
I-134	3.14E-3
I-135	5.27E-3
Mo-99	2.41E-4
Tc-99m	1.27E-4
Te-132	6.33E-5
Cs-134	3.65E-4
Cs-136	4.44E-5
Cs-137	4.86E-4
Ba-137m	9.68E-4
All Others	Negligible
TOTAL (Except Tritium)	1.83E-2

* 1.10E-3 = 1.10×10^{-3}

**Bases:
- Primary coolant activity per USNRC PWR Gale Code, Rev. 1 (1986)
- 100 lbm/day primary-to-secondary leakage
- 75 gpm blowdown processed
- WL DF value of 100 for listed radionuclides

TABLE 11.2-5

NORMAL SECONDARY SYSTEM CONDENSATE LEAKAGE RELEASES

Radionuclide	Annual Release (Ci/Year)
I-130	1.00E-05*
I-131	2.93E-03
I-132	1.90E-04
I-133	2.53E-03
I-135	5.40E-04
Mo-99	9.00E-05
Tc-99m	1.50E-04
Te-132	2.00E-05
Cs-134	3.00E-05
Cs-136	1.00E-05
Cs-137	2.00E-05
Ba-137m	2.00E-05
Co-58	2.00E-05
TOTAL (Except Tritium)	6.58E-03

* 1.00E-05 = 1.00×10^{-5}

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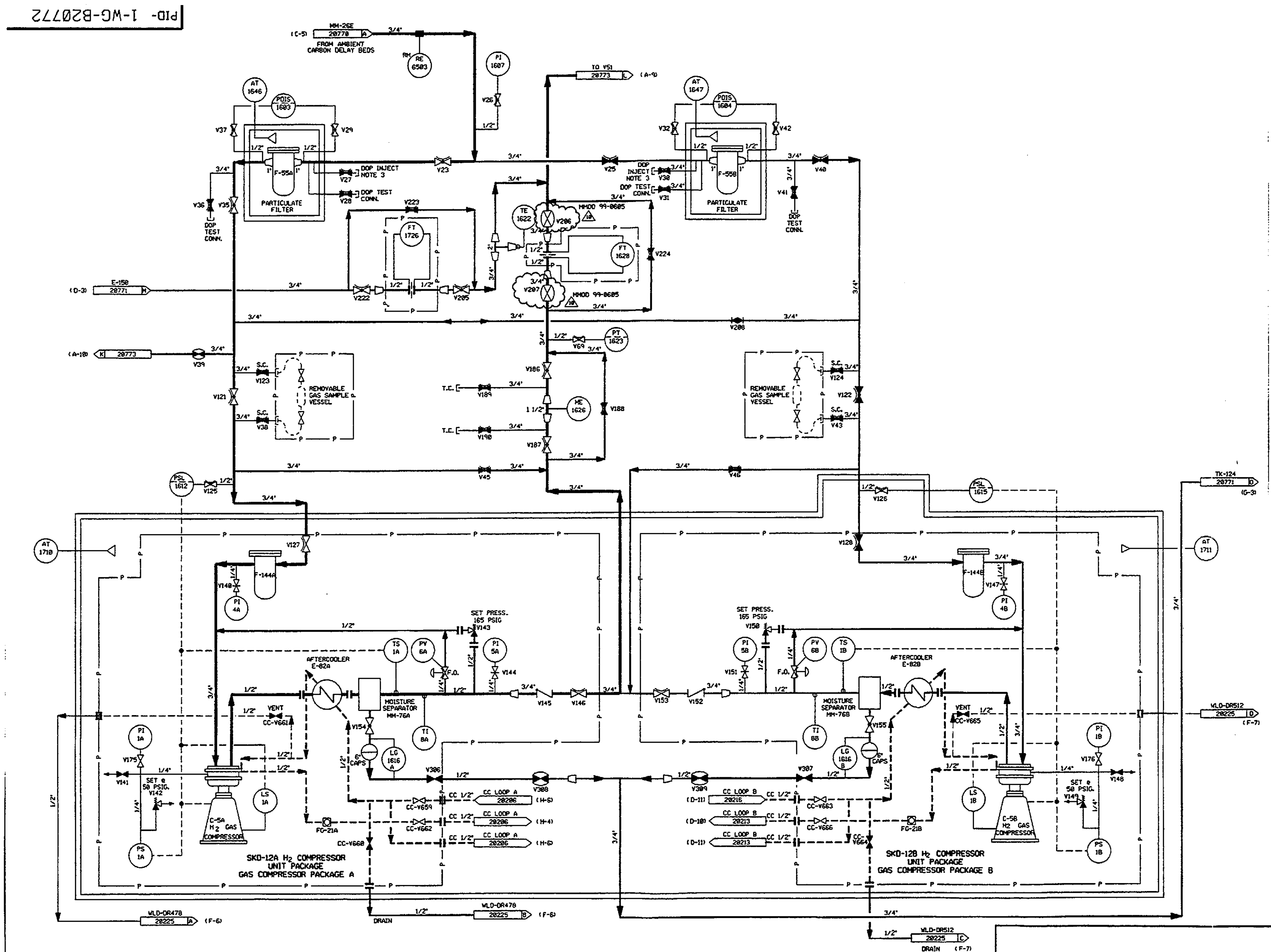

FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2

NOTES:

1. WORK THIS DRAWING WITH DRAWINGS 20768, 20770, 20771 & 20773.
2. ALL PIPING, EQUIPMENT, COMPONENTS AND INSTRUMENTS HAVE SYSTEM PREFIX 1-WG, UNLESS NOTED OTHERWISE.
3. DOP FILTER EFFICIENCY TEST PORTS.
4. ALL PIPING, EQUIPMENT, COMPONENT AND INSTRUMENTS ARE SAFETY CLASS NNS, UNLESS NOTED OTHERWISE.
5. △ -INDICATES REVISION LEVEL.
6. DELETED

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	WASTE GAS SYSTEM DETAIL
1-WG -B20772	FIGURE 11.3-2 SH 3

PID- 1-NG-B20135

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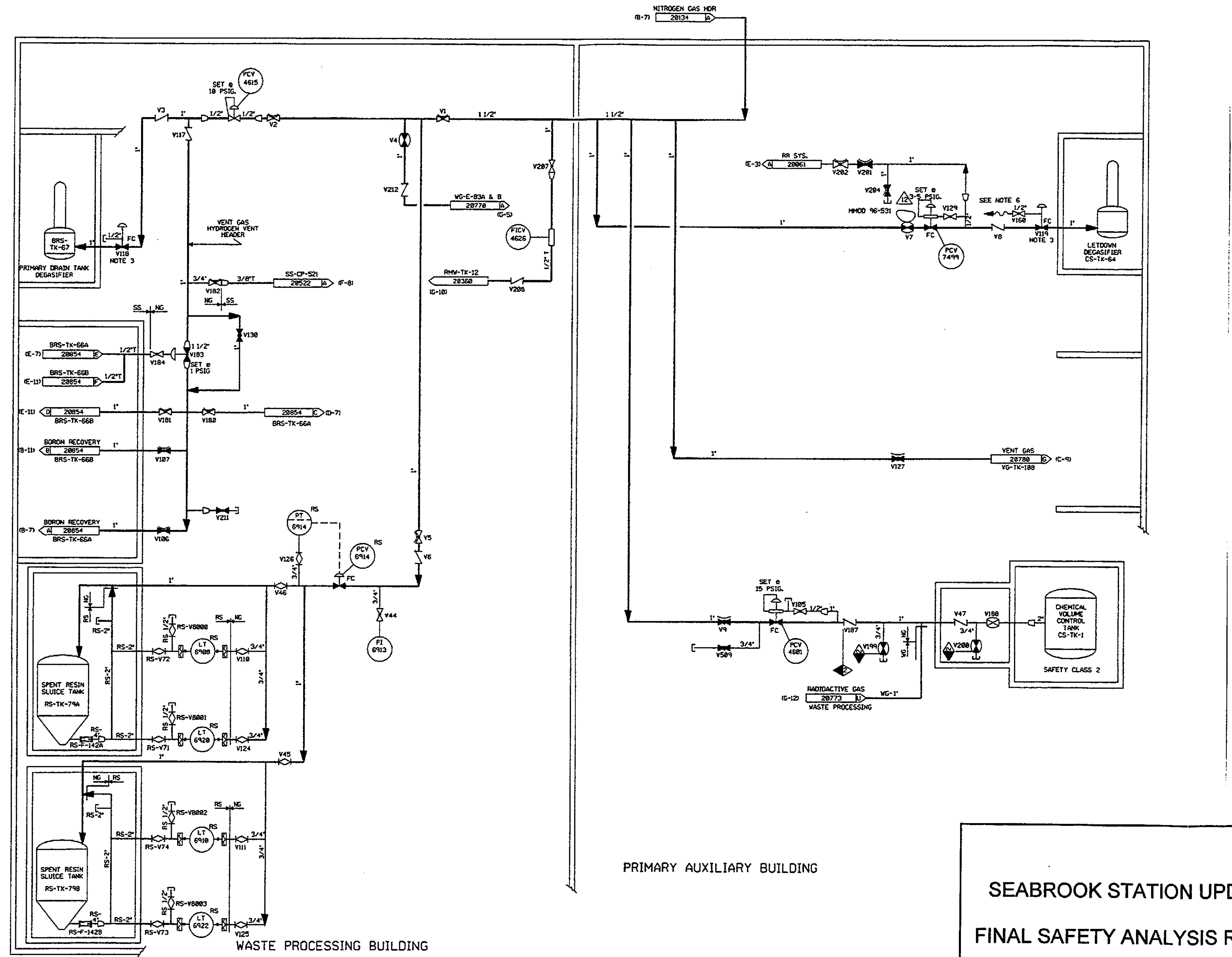
FOR P&ID REFERENCE DRAWINGS, SYMBOLS AND ABBREVIATIONS REFER TO DRAWINGS P&ID - LEGEND 1 AND P&ID - LEGEND 2



NOTES

1. WORK THIS DRAWING WITH DRAWINGS 20132, 20134, & 20136
2. ALL LINES, EQUIPMENT, COMPONENTS, AND INSTRUMENTS HAVE SYSTEM PREFIX NG, UNLESS NOTED OTHERWISE.
3. VALVE TO BE INSTALLED WITH DEGASIFIER LIQUID UNDER THE VALVE SEAT
4. ALL LINES, EQUIPMENT, COMPONENT AND INSTRUMENTS ARE SAFETY CLASS NNS, UNLESS NOTED OTHERWISE.
5. △ INDICATES REVISION LEVEL
6. REFER TO DRAWING VSL-20774 FOR VALVE LEAKOFFS.

PORTIONS OF THIS DRAWING ARE

NUCLEAR SAFETY RELATED

REF. DRAWING NO.	REV	TITLE
9763-F-805626	11	BORON RECOVERY SYS
9763-F-805618	10	BORON RECOVERY SYS
9763-F-805020	11	GAS SERVICE SYS NITROGEN NUCLEAR
9763-F-805613	12	SPENT RESIN SLUICING SYS
9763-F-805037	11	CHEM & VOLUME CONTROL SYS.

SEABROOK STATION UPDATED

FINAL SAFETY ANALYSIS REPORT

NITROGEN GAS DETAIL

1-NG-B20135

FIGURE 11.3-4

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system, in turn, is connected to an operator display/control console in the control room and the health physics checkpoint. The process and effluent radiation monitoring system instrument engineering diagram (Figure 11.5-1) shows an overview of the system, its components and location.

Table 11.5-1 lists the various process and effluent radiation monitoring channels provided and their pertinent design information, such as detector type, ranges, reference isotopes which the detectors are keyed to, and sensitivities.

Ranges and sensitivities have been selected using the following bases:

a. Maximum calculated concentrations during normal operations, anticipated operational occurrences and postulated accidents

b. State-of-the-art limitations of the commercially available detectors

c. Minimum concentrations that must be detected to permit timely automatic or operator manual responses tabulated in Table 11.5-2 and to avoid exceeding Technical Specification limits.

Shielding is provided on all the monitors to reduce the effect of background radiation, so that the minimum sensitivities specified are met. Table 11.5-2 shows the automatic system response and the operator response to annunciated radioactivity level limits.

A modular assembly with a microprocessor is provided in a locally mounted cabinet for each channel. The assembly converts pulse rate from the detectors to engineering units suitable for indication and recording. The following functions and components are included:

a. <u>Indication</u>

Radiation level is indicated by digital readout. Units are microcuries per cubic centimeter, milliroentgens per hour, counts per minute or roentgens per hour.

b. <u>Alarms</u>

Alarms on (1) rising signal (the setpoint is adjustable to any point on the scale), and (2) loss of signal implying circuit failure are provided.

c. <u>Functional Test and Calibration Test</u>

Each radiation monitoring channel automatically energizes a solenoid which exposes a radiation check source to the detector for a source check. This will cause an up-scale indication proving operation of the channel. The check source has a long half-life and an energy emission within the spectra of the radiation being monitored. The

Radiation Monitoring System computes printed records of the fact that a test is in progress. For ion chambers only, a calibration test is accomplished by inputting a pre-calibrated pulse signal to the channel. Proper indication at a meter will verify the calibration of the circuitry.

d. Indicating Lights

Indicating lights at the local radiation monitoring cabinets monitor individual channel high radiation alarms and circuit failures.

e. Power Supplies

Power supplies are mounted at the local radiation monitoring cabinets, and provide the voltages for the modular component circuitry, relays and alarm lights. The power supplies also supply high voltage for the detector. Internal battery backup is provided to prevent loss of stored information in the event of loss of AC power.

f. Fail-Safe

Fail-safe circuits in each monitoring channel indicate channel failure caused by signal or power failure.

11.5.2.1 Channel Descriptions

a. Waste Gas Processing Monitor - Channels 6502, 6503 and 6504

Radiogas monitors are located online at three points within the Radioactive Gaseous Waste System. Monitor 6502 is located up-stream of the carbon delay beds, 6503 is located downstream of the carbon delay beds, and 6504 is located downstream of the waste gas compressors. These monitors serve as indicators of carbon bed performance, with control room annunciation to alert station operators of abnormal operation or conditions. Remote indication and annunciation are provided on the control panel for the Radioactive Gaseous Waste System and in the control room.

A high radiation signal on 6504 terminates waste gas system discharges to the ventilation stack by automatic closure of the waste gas discharge valve.

b. Condenser Air Evacuator System Gas Monitor - Channel 6505

This channel monitors the discharge from the shell-side vacuum pump exhaust header of the condenser for gaseous radioactivity, which is indicative of a primary-to-secondary system leak. During normal plant operation, the gas discharge is routed to the Primary Auxiliary Building exhaust filter system.

During startup operation (hogging), the gases removed by the evacuation system are discharged to the atmosphere via the Turbine Building vent. A beta scintillator is used to monitor the gaseous radioactivity level. Remote indication and annunciation are provided locally and in the control room.

c. Boron Recovery System Monitors - Channels 6500 and 6501

Radiation monitors are located at two points within the Boron Recovery System (BRS). Monitor 6500 is located downstream of the boron recovery filters and upstream of the boron waste storage tanks. Monitor 6501 is located between the distillate cooler and the recovery test tanks. These monitors serve as indicators of BRS processing performance, with control room annunciation and indication to alert station operators of abnormal operation or conditions. Remote indication and annunciation are provided on the control board for the BRS.

d. Primary Component Cooling Liquid Monitors - Channels 6515 and 6516

These two channels continuously monitor trains A and B of the Primary Component Cooling System for radioactivity indicative of a leak from the Reactor Coolant System or one of the other radioactive systems which exchange with the Primary Component Cooling System. Indication and annunciation are provided locally and in the control room.

The gamma scintillation detectors are located in offline liquid samplers.

e. Waste Processing System Liquid Effluent Monitor-Channel 6509

All discharges from the station via the WL Test Tank discharge header are monitored by an online gamma scintillation detector. This includes all discharges from the Test Tank itself, as well as steam generator blowdown demineralizer regenerant solution from the waste holdup sump or the bottom of the demineralizer beds. (See Subsection 10.4.8.2.)

Automatic valve closure action is initiated by this monitor to prevent further release after a high-radiation level is indicated and alarmed. Control room and remote indication and annunciation are provided.

f. Steam Generator Blowdown Liquid Sample Monitor - Channels 6510, 6511, 6512, 6513 and 6519

These channels monitor the liquid phase of the secondary side of the steam generator for radioactivity concentrations, which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air evacuation system gas monitor. A sample from the bottom of each steam generator is continuously monitored by a scintillation counter mounted in line with an offline type sample assembly.

Monitor 6519 is an offline detector with pumping system which monitors the flash tank discharge.

High activity alarm indications are displayed at the detector location and in the control room.

In the event of a high activity alarm from any monitor, the isolation valve in the blowdown flash tank discharge closes.

g. Reactor Coolant Letdown Gross Activity - Channels 6520

The reactor coolant letdown monitoring system is in service whenever normal letdown is in service and has no automatic functions. This monitor provides indication of primary coolant radioactivity concentration over a wide range of operating conditions and assists in the detection of failed fuel.

The design utilizes an adjacent-to-line detector (Geiger Mueller type) positioned in a shield that provides a collimated view of the letdown line. The detector is placed far enough from the RCS to allow for sufficient N-16 decay. The monitor readout indication is in mR/hr and has a range of 10^{-1} to 10^{4} mR/hr. This detector has a minimum sensitivity of about 1×10^{0} μCi/cc in a 15 mR/hr background depending upon the isotope of interest. The detector exhibits good energy linearity for gamma rays between 210 keV and 1333 keV and provides acceptable response to the isotopes listed in Table 11.5-1. The upper detection limit of the monitor is about 1×10^{3} μCi/cc depending upon the isotope of interest thereby providing a range of 1 to 1000 μCi/cc.

h. Liquid Waste From Evaporators to Waste Test Tanks - Channel 6514

A scintillation detector in an online sampler continuously monitors the waste liquid transferred to the waste test tanks. Increasing radioactivity concentrations indicate a potential problem upstream or a need to filter the tank contents or add water to the tanks to reduce the radioactivity concentration. Control room and remote indication and alarms are provided. A high alarm level closes the waste test tank inlet valves.

i. Resin Sluicing Operation Monitors - Channels 6560, 6561 and 6564

Geiger-Mueller tubes clamped to the process pipe monitor the resin sluicing operation. They provide indication and alarm locally and at the waste management panel. They do not interface with the RDMS host computer system. The function of these detectors is to monitor filter failure and to indicate completion of sluicing operation.

j. Main Steam Line Radiation Monitors - Channels 6481-1, 6481-2, 6482-1, and 6482-2

Online gamma-sensitive detectors are located on each main steam line upstream of the safety relief valves. As required by NUREG-0737, these monitors provide a method of quantifying high-level releases of radioactive noble gasses after an accident. Control room indications and alarms are provided.

The monitors display steam line dose rates in "mr/hr." The noble gas release rate is calculated from the dose rate by using a procedure.

k. Plant Vent Monitor - Channels 6528-1, 6528-2, 6528-3, and 6495
The monitoring capability associated with the main plant vent is described in Subsection 12.3.4.

l. Fuel Storage Building Exhaust Monitor - Channel 6562

The monitoring capability associated with air exhaust is described in Subsection 12.3.4.

m. Turbine Building Sump Liquid Radiation Monitor - Channel 6521

This online gamma scintillation detector is located on the Turbine Building sump effluent line. At a pre-determined radioactive concentration, this monitor will alarm and automatically terminate the discharge and isolate the sump.

n. Containment Online Purge Monitor - Channels 6527A, 6527B

These detectors monitor the air exhausted via the containment purge. They utilize GM tubes sensitive to Xe-133. These detectors provide measurement of the activity of the containment purge and provide isolation on a high signal. The detectors and their associated microprocessors are Class 1E. Each monitor utilizes a two-out-of-two detector logic such that two detectors must be in alarm before the monitor initiates an isolation signal.

o. Auxiliary Condensate Monitor - Channel 6490

An offline, skid-mounted monitor draws a sample from the auxiliary condensate return line. In the event that the auxiliary steam should become contaminated, this monitor will automatically terminate the condensate return to the auxiliary steam boiler and isolate the return piping.

p. Storm Drains Monitor - Channel 6454

This is an offline, skid-mounted monitor which continuously samples the storm drainage. The monitor design is identical to the steam generator blowdown liquid sample monitor. An auxiliary pump provides the necessary sample flow. Indication and alarm are provided only locally. This monitor does not interface with the RDMS host computer system. On a high alarm a composite grab sample is automatically obtained. An automatic continuous operating composite sampler is also provided.

11.5.2.2 Alarm Setpoints

The alarm setpoints for the process and effluent radiation monitoring system are provided in Table 11.5-1.

In establishing the site boundary concentration, it is assumed for continuous releases that average annual meteorology exists for gaseous discharges and average annual circulating water flow exists for diluting liquid discharges. For intermittent or off-normal releases, short-term meteorology and actual dilution water flow is assumed for gaseous and liquid discharges, respectively.

11.5.2.3 Design Evaluation

a. The reactor coolant letdown gross activity monitor (RM6520) serves as a failed fuel advisory and provides a function independent of the

Alpha isotopic analysis is performed using the silicon surface barrier detector with the multichannel analyzer system, or by liquid scintillation.

Gamma spectrometry is used for isotopic analysis of liquid, gaseous and airborne particulate and iodine samples. A high efficiency, high-resolution HpGe detector is available, in conjunction with a multichannel analyzer, for resolving complex gamma spectra.

Effluent tritium samples are collected by various methods and analyzed by liquid scintillate.

11.5.2.6 Calibration and Maintenance of Effluent Radiation Monitors

A primary calibration is performed on a one-time basis, using typical isotopes of interest to determine proper detector response. Further primary calibrations are not required since the geometry cannot be significantly altered within the sampler. Calibration of samplers is then performed based on a known correlation between the detector responses and multiple secondary standards.

Secondary standard calibrations are performed with radiation sources of known activity. This calibration confirms the channel sensitivity. The secondary standard calibration is performed by placing the secondary standards on the sensitive area of the detector and comparing detector response to the detector response at the time of primary calibration.

The radiation monitoring system channels will be status checked at least daily, and calibrated at refueling intervals. If a monitor functionally tested quarterly provides a control function on release, it will be functionally tested prior to that release.

Calibration of the indicating channels is performed following any equipment maintenance which could result in reducing the accuracy of the instrument indication. It is also done any time use of the ion chamber precalibrated pulse test signal or the radioactive check source indicates instrument drift.

A burn-in test, operational test and isotopic calibration of the Complete Radiation Monitoring System are performed at the factory. Field calibration after system installation will be performed using calibration sources and their decay curves provided with the system. The sample chambers will be decontaminated in situ periodically and, if required, are easily replaceable.

11.5.3 Effluent Monitoring and Sampling

General Design Criterion 64 requires monitoring of effluent discharge paths. Compliance with requirements are as discussed in Subsection 11.5.2.

Airborne radioactivity monitoring is discussed in Subsection 12.3.4.

11.5.4 Process Monitoring and Sampling

Means are provided to control and monitor the release of radioactivity to the environment in accordance with the requirements of General Design Criteria 60 and 63 as discussed in Subsection 11.5.2.

TABLE 11.5-1
(Sheet 1 of 3)

PROCESS AND EFFLUENT RADIATION MONITORS

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DET. BACK-GRD. mr/hr	RANGE LOW-HIGH (μCi/cc)	(Note 5) ALARM SET POINT (μCi/cc)	REFERENCE ISOTOPE	DETECTOR QTY.	SAFETY CLASS	ENERGY* LEVEL	LOOP DIAG. 1-NHY	P&ID 1-NHY
6454	Storm Drains	Gamma Scint	0.5	10^{-6} 10^{-2}		Co^{58}, I^{131}, CS^{137}	1	Non 1E	Note 2	506765	SD-20404
6502	Waste Gas Inlet to Carbon Delay Beds	Gamma Scint	15.0	10^{-2} 10^{+2}		Xe^{133}	1	Non 1E	Note 1	506897	20772
6503	Waste Gas Compressor Inlet	Gamma Scint	15.0	10^{-3} 10^{+1}		Kr^{85}	1	"	Note 1	506898	20770
6504	H_2 Gas Compressor Disch.	Gamma Scint	15.0	10^{-3} 10^{+1}		Kr^{85}	1	"	Note 1	506899	20773
6505	Condenser Air Evac	Beta Scint	0.5	Note 3		Xe^{133}	1	"	Note 1	506055	20774
6500	Boron Recovery Stor. Tank Inlet	Gamma Scint	1.0	$10^{-5}$$10^{-1}$		Co^{58}, I^{131}, CS^{137}	1	"	Note 2	506105	20856
6501	Boron Recovery Test Tank Inlet	Gamma Scint	2.5	10^{-6} 10^{-3}		Co^{58}, I^{131}, CS^{137}	1	"	Note 2	506113	20861
6515,6516	Primary Component Cooling Water	Gamma Scint	2.5	10^{-7} 10^{-3}		Co^{58}, I^{131}, CS^{137}	2	"	Note 2	506190 506194	20211 20205
6509	Liquid Waste Test Tk Disch to CWS	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58}, I^{131}, CS^{137}	1	"	Note 2	506927	20831
6514	Waste Liquid From Evaporators	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58}, I^{131}, CB^{137}	1	"	Note 2	506931	20831
6510,6511 6512,6513	Steam Gen Blowdown Sample Loops 1,2,3,4	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58}, I^{131}, CS^{137}	4	"	Note 2	506815	20521
6519	Steam Gen Blowdown Flash Tank Drain	Gamma Scint	2.5	10^{-7} 10^{-3}		Co^{58}, I^{131}, CS^{137}	1	"	Note 2	506734	20626
6520	Reactor Coolant Gross Activity Monitor	GM	15	10^{-1} 10^{+4}mr/hr		Co^{58}, I^{131}, CS^{137}		"	Note 2	506269	20722
6481-1, 6482-1 6481-2, 6482-2	Main Steam Line Monitor	Gamma Scint	2.5	10^{-0} 10^{+5}mr/hr		Xe^{133}		" -	-	506551 -2, -3, -4	20580 20581
6490	Aux Steam Cond	Gamma Scint	0.5	$10^{-7}$$10^{-3}$		Co^{58}, I^{131}, CS^{137}	1	Non 1E	Note 2	507165	20908
6521	Turb. Bldg. Sump Liq. Monitor	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58}, I^{131}, CS^{137}	1	Non 1E	Note 2	506713 506716	20195

*See table 11.5-1 (Sheet 3) for notes.

TABLE 11.5-1
(Sheet 2 of 3)

PROCESS AND EFFLUENT RADIATION MONITORS

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DET. BACK-GRD. mr/hr	RANGE LOW-HIGH (μCi/cc)	(Note 5) ALARM SET POINT (μCi/cc)	REFERENCE ISOTOPE	DETECTOR QTY.	SAFETY CLASS	ENERGY* LEVEL	LOOP DIAG. 1-NHY	P&ID 1-NHY
6527A1,A2 B1,B2	COP Monitors	GM	2.5	10^1 -10^6cpm		Xe^{133}	4	1E	Note 1	506211	20504
6560	Resin Sluice Line	GM	100	Note 4		Xe^{133}	1	Non 1E		506694	20252
6561	Resin Transfer Line	GM	100	Note 4		Xe^{133}	1	Non 1E		506694	20735
6564	Sluice Pump Line	GM	100	Note 4		Xe^{133}	1	Non 1E		586692	20252

* See Table 11.5-1 (Sheet 3) for notes.

TABLE 11.5-1
(Sheet 3 of 3)

PROCESS AND EFFLUENT RADIATION MONITORS

Note	Isotopes	Max. Beta Energy (Mev)	Predominant Gamma Energy (Mev)
1	Xe^{133}	0.346	0.081
	Xe^{135}	0.92	0.249
	Kr^{85}	0.67	0.514
	Kr^{85m}	0.82	0.150
2	I^{131}	0.606	0.364
	I^{133}	1.27	0.53
	Cs^{134}	0.662	0.604
	Cs^{137}	0.514	0.662
	Co^{58}	0.474	0.81
	Co^{60}	0.314	1.17, 1.33

3 Condenser Air Evacuation Monitor to have output in counts per min (cpm) (10^1 to 10^6).

4 Monitors 6560, 6561, 6564 have output in mr/hr (10° to 10^5).

5 Radiation monitoring setpoints are varied during operation to follow station operating conditions. Setpoints are maintained within the bounds established in the Technical Specifications. The methodology for establishing the setpoints is found in the Station Offsite Dose Calculation Manual (ODCM) and/or other station operating procedures.

TABLE 11.5-2
(Sheet 1 of 2)

AUTOMATIC SYSTEM AND OPERATOR RESPONSES TO ANNUNCIATED RADIOACTIVITY LEVEL LIMITS

Monitor	Automatic Response to Radioactivity Level Limit	Operator Response to Radioactivity Level Limit
Waste Gas Processing (6504)	Closure of waste gas discharge valve	Request sampling and laboratory analysis
Condenser Air Evacuation System	No automatic response	Request sampling and laboratory analysis and/or observe steam generator liquid samplers.
Boron Recovery System	No automatic response	Request sampling and laboratory analysis - check operation of system.
Component Cooling Liquid	No automatic response	Request sampling and laboratory analysis
Waste Processing System Liquid Effluent	Close valve in effluent line	Terminate discharge of liquid effluents - request sampling and laboratory analysis of effluent.
Steam Generator Liquid Samples	Isolation valve in the blowdown flash tank discharge closes	Request sampling and laboratory analysis (observe Tech. Spec. 3/4.7.1.4 limit on secondary activity)
Reactor Coolant Gross Activity	No automatic response	Request sampling and laboratory analysis of reactor coolant samples - to detect failed fuel (observe Tech. Spec. 3/4.4.8 limit on reactor coolant activity)
Waste Test Tank Inlet	Close inlet valves	Correct cause of High Radioactivity Level in WL System, sample waste test tank

The radioactivity level for shim bleed and primary leakages is the same as the reactor coolant. Flow patterns for these sources are intermittent in nature. A combined flow rate of 4.16×10^2 lbs/hr (0.82 gpm) is estimated on an annual average basis.

To control the tritium level within the Primary Coolant System, 200,000 gallons of reactor coolant is expected to be discharged annually through the Boron Recovery System. Therefore, the release fraction amounts to 46 percent of reactor coolant processed through the BRS annually.

System decontamination factors (DF) are conservatively assumed to be 10^3 for iodines and 10^4 for other nuclides due to evaporation and demineralization.

Holdup time is calculated to be a minimum of 5 days on the basis of the capacities of two boron waste storage tanks (225,000 gals each) and two recovery test tanks (18,000 gals each).

3. <u>Secondary System</u>

a. The number and type of steam generators, the type of chemistry used and the carry-over factor used in the evaluation for iodine and nonvolatiles.

Response 3a. Four vertical, recirculating, inverted U-tube steam generators per unit. Volatile chemistry will be used to control secondary side chemistry.

Carry over factors: 1 percent for iodines

0.1 percent for nonvolatiles

b. The total steam flow (lbs/hr) in the secondary system.

Response 3b. 1.514×10^7 lbs/hr.

c. The mass of liquid in each steam generator (lb.) at full power.

Response 3c. 9.55×10^4 lbs.

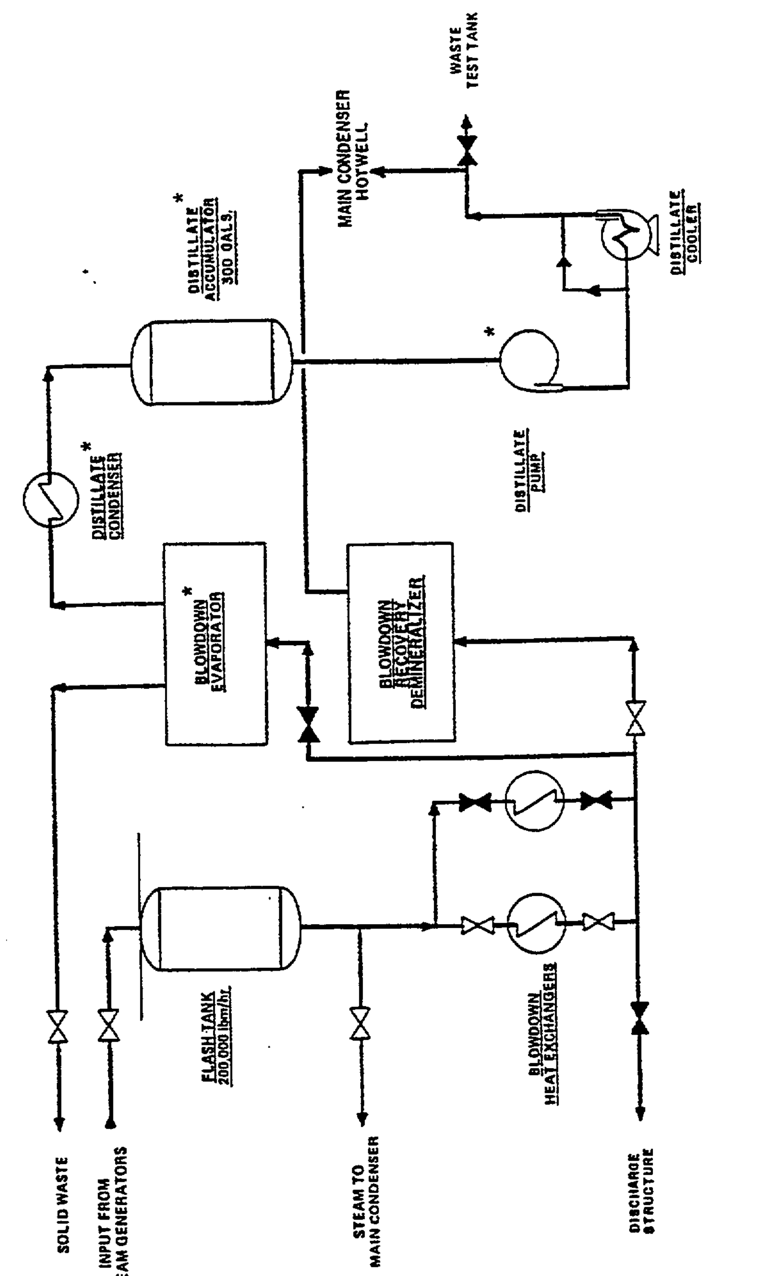
d. The primary-to-secondary leakage rate (lb/day) used in the evaluation.

Response 3d. 100 lb. per day

e. Description of the Steam Generator Blowdown and Blowdown Purification Systems. The average steam generator blowdown rate (lb/hr) is used in the evaluation.

Response 3e. When primary-to-secondary leakage as described in response 3d exists, the primary method to process radioactive secondary liquid from the steam generators is to direct steam blowdown flash tank bottoms cooler discharge to the floor drain tanks. If no secondary pressure is available, the steam blowdown and wet lay-up pumps can be used. From the floor drain tanks, processing through the installed vendor resin skid (WL-SKD-135) to the waste test tanks is the preferred method (reference Subsection 11.2.2.1). The DF assumed for all radionuclides by this method of treatment is 100. Approximately 30 percent of the blowdown volume flashes to steam in the flash tank. This steam is vented to the number 3 feedwater heater, or main condenser if they are available. With the number 3 feedwater heater or main condenser not available, steam from the flash tank will be directed to the flash tank condenser/cooler and then pumped to the waste test tanks in the Liquid Waste System (see Section 11.2). Further processing by the Liquid Waste System is available, if required, prior to release to the environment via the plant Service and Circulating Water System.

Average steam generator blowdown rate of 75 gpm (3.75×10^4 lbs/hr) is assumed for the analysis. The Steam Generator Blowdown System is described in Subsection 10.4.8 of the Updated FSAR. A schematic



*Evaporator Subsystem Not Immediately Available For Use.

G:\Word\Images_P\Images_P\UFSAR\11A2.ds4

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Blowdown System Schematic	
	REV. 07	FIGURE 11A-2

Seabrook Station

North Atlantic

Updated Final Safety Analysis Report

Revision 7

(g) Other Area Radiation Monitors - Channels 6534, 6537, 6538, 6539, 6541, 6543, 6544, 6545, 6546, 6547, 6549, 6550, 6551, 6552, 6553, 6554, 6555, 6556, 6557, 6558, 6559, 6570, and 6571

These channels use Geiger-Mueller detectors and monitor the ambient radiation at various points throughout the facility as listed in Table 12.3-14.

5. Calibration and Maintenance

Refer to Subsection 11.5.2.6 for calibration and maintenance details.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

a. Objectives and Design Basis

The ventilation airborne Radioactivity Monitoring System provides radiation measurements, indications, records, alarms and controls at selected locations to detect and control radiation levels within Containment, Service Building, Radwaste Building and the plant vent, and to verify compliance with applicable limits of 10 CFR 20 and 10 CFR 50, General Design Criteria 19, 63 and 64.

On May 21, 1991, a complete revision to 10 CFR 20 was issued. Several design bases reference 10 CFR 20 and specific terms or parts of 10 CFR 20. Design bases information provides a historical perspective of the information used to formulate a particular design. References to 10 CFR 20 when used in a historical or design bases context have not been changed to reflect the revised 10 CFR 20.

Monitored points within the station ventilation system are in areas where potential personnel exposure to radiation is most likely and in several ventilation exhaust ducts. A tabulation of the airborne Radiation Monitoring System is found in Tables 12.3-15 and 12.3-16 and Figure 12.3-20.

Those monitors which are Class 1E are listed in the above tables and further discussed in Subsection 12.3.4.1b.2.

The sensitivity of the airborne radioactivity monitors is such that they should be capable of detecting ten MPC-hours of particulate, iodine, and gaseous radioactivity in those plant areas that have contained sources of airborne radioactivity and which may be occupied by personnel. Typical airborne concentrations for various plant areas are given in Tables 12.2-31 through 12.2-37.

As discussed in Subsection 12.5.3.1, the Health Physics Program includes requirements to perform sampling and analysis for airborne radioactivity, routinely and during specific evolutions such as opening of the primary system. Sampling equipment includes portable continuous air monitors and portable samplers. The monitoring and sampling capabilities, when combined, provide sufficient information to permit adequate protection of personnel from exposure to airborne radioactivity.

b. System Description

Subsection 12.3.4.1b describes the digital computer based RDMS. Subsection 11.5.2 describes the local microprocessor provisions. The airborne Radioactivity Monitoring System consists of two basic types of monitoring systems:

- Particulate and gaseous monitors (with iodine sampling) which are skid-mounted and utilize pumping systems.
- Gross activity monitors which consist of detectors mounted directly in duct air stream.

A typical channel is shown in Figure 12.3-20.

1. Particulate, Iodine and Gaseous Monitors

Each airborne particulate, iodine and gaseous monitor has common equipment as follows:

(a) Isokinetic Sampler

Sampler and lines adhere to requirements of ANSI N13.1. Sample line sizes are one-half inch with flow rate designed for 2-3 scfm. All sample lines slope from high point (isokinetic sampler) to the low point (sample pump).

(b) Pumping System

(1) The flow control assembly includes a pump unit and selector valves that provide a representative sample (or a "fresh" sample) to the detector.

(2) The pump unit consists of:

a A pump to obtain the air sample

b A flowmeter to indicate the flow rate

TABLE 12.3-13
(Sheet 2 of 2)

COMPONENT SHIELDING THICKNESS

Reference Figure: 12.3-12, 12.3-13

Bldg: WPB Elevation: 53'-0"

COMPONENT	CONCRETE SHIELD THICKNESS (IN)					
	WALLS					
	NORTH	EAST	SOUTH	WEST	CEILING	FLOOR
Alternate Station Concentrates Feed Pump	30	30	12	36	12	24
Entrainment Separator	12	30	30	24	15	24
Crystallizer Condenser	30	30	12	36	12	24
Crystallizer Subcooler	30	30	12	36	12	24
Spent Resin Dewatering Pump	36	30	30	36	12	24
Crystallizer Reflux Pot	12	30	30	24	15	24

TABLE 12.3-14
(Sheet 1 of 2)

AREA RADIATION MONITORS

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH mr/hr	(Note 5) ALARM SET POINT mr/hr	DETECTOR QTY.	IEEE CLASS	UPDATED FSAR FIGURE REFERENCE
	Containment Structure							
6534	In-Core Instrument Seal Table	GM (Note 4)	15	10^{-1}-10^{4}		1	Non 1E	12.3-2
6535A,B (Note 1)	Manipulator Crane	GM (Note 4)	15	10^{-1}-10^{4}		2	1E	12.3-3
6536-1,2	Personnel Hatch (Post-LOCA)	Ion Chamber	2.5	10^{+1}-10^{9}		2	Non 1E	12.3-3
6529	Cavity Beneath Reactor Vessel	Ion Chamber	$3x10^{6}$ (Note 3)	10^{+1}-10^{7}		1	Non 1E	12.3-1
6576A,B	Containment (Post-LOCA)	Ion Chamber	25	10^{0}-10^{8} r/hr		2	1E	12.3-3
	Primary Auxiliary Building							
6537	Sampling Room	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-6
6538, 6539	RHR Pump Area	GM	>100	10^{-1}-10^{4}		2	Non 1E	12.3-4
6540	Volume Control Tank Area	Ion Chamber	$8x10^{4}$	10^{1}-10^{7}		1	Non 1E	12.3-7
6541	Lower Level	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-5
6543	Entrance	GM	>100	10^{-1}-10^{4}		1	Non 1E	12.3-5
6544	Entrance	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-6
6545, 6546 6547	Charging Pump Room	GM	110	10^{-1}-10^{4}		3	Non 1E	12.3-5
6508-1,2	PAB-HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	12.3-5
6563-1,2	PAB-HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	12.3-5
6517-1,2	RHR - Pump Vault HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	
	Fuel Storage Building							
6549	Spent Fuel Pool Area	GM	2.5	10-1-10^{4}		1	Non 1E	12.3-15
6518	Spent Fuel - HRAM	Ion Chamber	2.5	10^{-2}-10^{4} r/hr		1	Non 1E	12.3-15

Additionally, general performance guidelines have been identified for certain evolutions and operational tasks, such as those discussed below, to further promote personnel radiation exposure reduction.

Refueling procedures state the monitoring requirements, minimum fuel-to-water surface distance, and degassification of the Reactor Coolant System prior to removing the reactor vessel head for refueling operations. During refueling, the reactor coolant is filtered and cooled to help reduce airborne contamination caused by coolant evaporation.

Prior to initial handling of spent fuel, the equipment used is checked for possible damage and proper operation. During fuel movement the minimum top of fuel-to-water surface distance is maintained above a level as stated in procedures, and airborne activity will be monitored. During the initial use of the fuel transfer canal and tube, appropriate areas are barricaded, locked to access, and detailed radiation area surveys are conducted. Thereafter, radiation protection controls are used as necessary.

When shipping spent fuel, an approved cask is used. Fuel is loaded into the cask underwater, then the cask is capped or sealed, raised, surveyed, decontaminated, if necessary, and resurveyed as necessary for compliance with appropriate federal regulations prior to release.

Prior to performing in-service inspection (ISI), the personnel review prior inspections, verify proper equipment operation, and review general area radiation levels and hot spot locations in the vicinity of their work. Where practicable, remote ISI techniques are used.

Procedures for radwaste handling reflect the ALARA philosophy. Radwaste operators are usually stationed at a remote control console from which remote operations can be performed. Remote operations include moving, filling, and sealing containers. Prior to container usage, when possible, shielding and labeling operations are performed. Vehicles used for shipping radwaste are surveyed in accordance with Department of Transportation regulations and Seabrook Station procedures. A ventilation system with exhaust through a High Efficiency Particulate Air (HEPA) Filter System will be used at the dry waste compaction station.

During normal operation plant personnel will receive job-related training. A direct benefit of the training should be a reduction of exposure, which can be attributed to improved efficiency. Design features of the facility such as reach rods, remote activation controls, and control panels make it possible for most operations to be performed from a low radiation area.

Routine maintenance consists of activities such as scheduled and preventive maintenance. Maintenance procedures are reviewed to ensure completeness. Work on the RCA side of the plant requires a Radiation Work Permit (RWP) with its associated radiological information and requirements. Normally, when work is to be performed involving radiation and/or radioactive materials, the responsible group requests health physics to determine if a job specific RWP is required. Health physics personnel make the determination in accordance with procedures and, if necessary, investigate the actual and/or anticipated radiological conditions. If an RWP is necessary, Health Physics specifies radiological protection requirements and authorizes, by signature, the issuance of an RWP. Each individual working under the RWP shall initial it or sign an associated form signifying he/she has read and understands the conditions of work. (The control room is aware of work being performed in the Station through the work control procedures. Health Physics notifies the control room if significant changes in the physical or radiological status of the unit occur.)

Sample collection by chemistry of the reactor coolant system gas or water is normally obtained at one of the sample stations. The sample stations are, when required, equipped with a hood, ventilation system and HEPA filter. Shielded containers and long tongs are available for handling samples when this is necessary. A shielded area has been provided in the radio-chemistry laboratory for the storage of radioactive samples.

Some instrumentation requires in-place exposure to a radioactive source during calibration. The storage and handling of sources is described in Subsection 12.5.3.7. Transfer of the sources to and from the place of calibration is usually done by using a shielded container. During calibration, when possible personnel will use shielding, time, and distance to reduce their exposure.

12.5.3.3 Physical and Administrative Measures for Controlling Access

Access to the RCA is limited to those persons whose entry is requested by station supervisors and authorized by health physics personnel. Every area inside the RCA in which radioactive materials and radiation are present shall be surveyed and conspicuously posted with the appropriate radiation caution sign(s).

Access to high radiation areas is controlled through procedures developed by health physics in accordance with Technical Specification 6.11.

12.5.3.4 Contamination Control

The limits for surface contamination in the RCA are specified in procedures. When the contamination level limits are exceeded the area will be posted as a "Contaminated Area." Additionally, all personnel are required to wear protective clothing for entry to "Contaminated Areas." The contaminated area will, when possible, be decontaminated to minimize the spread of contamination.

Material and equipment will be given an unconditional release by Health Physics if they meet the criteria as specified by station procedures. Authorization for use or movement of radioactive material outside the RCA is obtained from health physics. Control over the packaging, labeling and movement of this material is provided by health physics personnel.

CHAPTER 13

CONDUCT OF OPERATIONS

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CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE

This section describes the organization of the North Atlantic Energy Service Corporation (North Atlantic). North Atlantic is a wholly-owned subsidiary of Northeast Utilities (NU). NU affiliates operate, and have ownership interests in, four other New England nuclear units besides Seabrook Station. They are the Haddam Neck plant maintained by Connecticut Yankee Atomic Power Company (CY) and Millstone Units 1, 2 and 3 which are operated by the Northeast Nuclear Energy Company (NNECO). The Northeast Utilities Service Company (NUSCO) is another wholly-owned subsidiary of NU which provides engineering, operational, administrative and general services to the NU system. North Atlantic has a service agreement with NUSCO through which it can obtain support services when requested. North Atlantic also maintains its relationship with an approved contractor to provide various engineering, licensing, environmental, quality assurance, and other related support services to Seabrook Station. Notwithstanding the services provided to Seabrook Station by NUSCO and the approved contractor, North Atlantic remains responsible, at all times, for the management, operation and maintenance of Seabrook Station. The corporate interrelationships of North Atlantic with the Joint Owners, the NU system, and the approved contractor are shown in Figure 13.1-1.

13.1.1 Management and Technical Support Organization

North Atlantic is responsible for the operation and maintenance of Seabrook Station. The Executive Vice President and Chief Nuclear Officer (Seabrook) has final site authority and responsibility for the overall safe operation and maintenance of Seabrook Station. The Station Director reports to the Executive Vice President and Chief Nuclear Officer (Seabrook) and is designated as the management official in overall charge of the station.

The following provides a description of the North Atlantic organization associated with the operation of Seabrook Station:

a. Executive Vice President and Chief Nuclear Officer (Seabrook)

   The Executive Vice President and Chief Nuclear Officer (Seabrook) has overall responsibility for Seabrook Station, including plant operation; maintenance, design/engineering, nuclear oversight, licensing and support services. He has assigned responsibility for the services to the Station Director, Director of Engineering, Director of Support Services, and the Nuclear Oversight Manager.

b. Station Director

The Station Director - Seabrook is responsible for the training, operation and operational support of Seabrook Station. Seabrook Station is headed by the Station Director who is responsible for the operation and administration of Seabrook Station.

c. Director of Support Services

The Director of Support Services, reporting to the Executive Vice President and Chief Nuclear Officer (Seabrook) is responsible for Station security services, emergency preparedness, community and public affairs, materials management and labor relations, information technology, performance improvement and business services and facilities. The Director of Support Services is also responsible for administering the overall corrective action program, trend analysis and failure prevention techniques.

d. Environmental, Government and Owner Relations Manager

The Environmental, Government and Owner Relations Manager, reporting to the Executive Vice President and Chief Nuclear Officer (Seabrook), is responsible for the overall direction of nonradiological environmental compliance, hazardous waste, government affairs, and industry relations at Seabrook Station.

e. Director of Engineering

The Director of Engineering, reporting to the Executive Vice President and Chief Nuclear Officer (Seabrook), is responsible for North Atlantic engineering design, engineering programs, reactor engineering, shutdown risk management, configuration management, plant engineering and regulatory programs, including NRC and generic licensing.

f. Concerns Resolution Manager

The Concerns Resolution Manager, reporting to the Nuclear Oversight Manager, is responsible for the administration of the nuclear safety concerns program.

The internal corporate organizational relationships are shown in Figure 13.1-2. Resumes of corporate management personnel are available upon request.

Westinghouse, United Engineers and Constructors (UE&C) and General Electric are three other organizations that had major responsibilities for the design and construction of the Seabrook project. Westinghouse was responsible for the design, fabrication and delivery of the Nuclear Steam Supply System, related auxiliary systems and the nuclear fuel. Technical direction for the installation of the equipment and technical assistance throughout the preoperational testing, initial core loading and power escalation testing programs were further responsibilities of Westinghouse. United Engineers and Constructors (UE&C) was responsible for construction phase engineering, design and certain construction activities of the station. Included in their services were furnishing the balance-of-plant systems and components, structures and switchyards so that a complete and integrated facility resulted. General Electric was responsible for the design, fabrication and delivery of the turbine-generator unit.

13.1.1.1 <u>North Atlantic Organization, Responsibilities and Authority</u>

The North Atlantic organization is under the overall direction of the Executive Vice President and Chief Nuclear Officer (Seabrook), who has responsibility for the operation and operational support for Seabrook Station. The Station Director reports to the Executive Vice President and Chief Nuclear Officer (Seabrook).

The responsibilities, training, organization and qualifications of the Station Staff are discussed in Subsection 13.1.2. The responsibilities of the Training Staff are discussed in Subsection 13.1.1.2.

13.1.1.2 <u>Seabrook Training Organization, Responsibilities and Authority</u>

North Atlantic has recognized the importance of training by establishing training facilities and by providing a Seabrook site-specific simulator. The Training Center is located on the Seabrook site outside of the protected area.

The training facilities contain classrooms, office space, a library, study areas, instructor material preparation rooms, a computer room, administrative areas, and a simulated Seabrook control room with a full-size main control board and various main control room panels. The simulator control board was manufactured by Link, a division of the Singer Company. Link has had extensive experience with nuclear simulators and a myriad of simulators for military applications. Seabrook represents the eighteenth simulator built by Link for the nuclear industry. The simulator control room is not only similar to the actual control room in appearance, but is also operated under the same working conditions as the actual main control room to provide a realistic atmosphere for operator training.

The Nuclear Training Manager reports to the Station Director as shown in Figure 13.1-2. The training organization is shown in Figure 13.1-3. Resumes of key training personnel are available upon request.

Operator training is performed under the cognizance of the Operations Training Supervisor who is responsible for implementation of the License Training and Requalification Training Programs as discussed in Subsection 13.2.1. The training program for instructors is identical in scope to the Seabrook operator training required for a Senior Operator license, except that instructors are provided additional training on simulator control functions to improve teaching skills.

Training for technical and management staff is performed by the Training Department. This function is further discussed in Subsection 13.2.2.

13.1.2 Operating Organization

13.1.2.1 Station Organization

The Seabrook Station organization chart is shown in Figure 13.1-4. The station organization includes all the technically trained personnel necessary to support all aspects of Unit 1 operation.

The key supervisory positions for the station organization were filled in 1979. Personnel to meet the operational requirements of Unit 1 were hired on a phased basis consistent with the training and licensing requirements of the individual positions.

The Unit 1 on-duty operating shift crews are composed as shown in Technical Specification Table 6.2-1, and meet the requirements outlined in Technical Specification Subsection 6.2.2 describing the plant organization. Manpower necessary to staff six shift crews is provided. Each member of the station organization meets, or exceeds, the minimum qualifications recommended for comparable positions in Regulatory Guide 1.8, Revision 2, except ANSI/ANS 3.1-1978 is used as the standard rather than ANS 3.1 ANSI 18.1-1971, and except for those positions specifically identified in Reg. Guide 1.8, Rev. 2, that comply with ANSI/ANS 3.1-1981.

A retraining and replacement licensed training program for the Station Staff shall be maintained under the direction of the Training Manager in accordance with the Seabrook Station Institute of Nuclear Power Operations (INPO) Accredited Programs.

The employees assigned to the station organization have been trained as described in Section 13.2.

13.1.2.2 Station Personnel Responsibilities and Authorities

a. Overall Station Management

The Station Director of the Seabrook facility is responsible for overall management of Unit 1. In his absence, the Assistant Station Director assumes this responsibility. In the event of contingencies of a temporary nature, occurring during the absence of these two directors, the Operations Manager will be responsible for overall station operations. The Shift Manager assumes the responsibility for overall management of Unit 1 when the station management is not within the station. In addition, the Station Director may designate in writing other qualified personnel to assume overall station responsibility in his absence.

The Station Director reports to the Executive Vice President and Chief Nuclear Officer (Seabrook) for all activities related to the station. The Assistant Station Director and the Nuclear Training Manager report to the Station Director. The Training Department will remain independent of the remainder of the Seabrook Station staff to ensure that training is able to maintain sufficient organizational freedom to allow for independency from operating pressures. Reporting to the Station Director or Assistant Station Director are the following:

1. Assistant Station Director

The Assistant Station Director reports directly to the Station Director and is responsible for

(a) being a member of the Station Operation Review Committee (SORC), and

(b) assuming the responsibilities of the Station Director in his/her absence.

2. Operations Manager

The Operations Manager reports directly to the Station Director and is responsible for

(a) operating equipment at the Station in compliance with Technical Specifications and other license requirements,

(b) assisting in the training of operations personnel to assure an adequate number of qualified employees for each task,

(c) preparing, reviewing, approving and implementing the operating procedures to be used for Station operations,

(d) directing actions, within the realm of the Operations Group, to perform the balance of the surveillance testing program required by the Station license, and

(e) maintaining a staff of fire protection personnel.

3. <u>Chemistry and Health Physics Manager</u>

The Chemistry and Health Physics Manager reports directly to the Station Director and is responsible for

(a) ensuring that the quality of steam and water is within specifications, and

(b) health physics, radiation protection and radwaste control programs at the Station.

4. <u>Maintenance Manager</u>

The Maintenance Manager reports directly to the Station Director and is responsible for

(a) performing the support functions that include the corrective and preventive maintenance programs, maintenance related surveillance activities and Station modification and repair actions, and

(b) scheduling the performance of work and controlling the material, personnel and processes involved.

5. <u>Work Control and Outage Manager</u>

The Work Control and Outage Manager reports directly to the Station Director and is responsible for

(a) long-range planning and scheduling of refueling and planned maintenance outage planning and scheduling and outage coordination.

(b) forced outage planning and scheduling,

(c) daily work scheduling,

(d) work status tracking, and Work Control Program trend reporting.

6. Nuclear Training Manager

The Nuclear Training Manager reports directly to the Station Director and is responsible for direction and control for the conduct of training at Seabrook Station. He has overall responsibility for administrative activities, program development and evaluation, record keeping and meeting accreditation requirements.

The Seabrook Station staff assumed several responsibilities in support of the Initial Test Program to increase staff awareness of the Preoperational Test Program details, provide staff contribution to the program implementation details and procedures, ensure staff readiness to accept the completed systems and derive as much training benefit as possible from the Initial Test Program effort. These support efforts included the loan of staff personnel to the Startup Test Department, the assumption of assigned preoperational testing responsibilities, the verification of design and installation, chemistry support, procedure review, evaluation of test data and operational as well as maintenance support of plant systems prior to final acceptance turnover of the systems.

The functions, responsibilities and authorities for station positions under the direct cognizance of these managers are defined below.

b. Operations

The Operations Manager is responsible for the operation of the station. He maintains close communications with the other managers regarding all activities at the station.

He is responsible for the safety and operation of the unit's equipment in accordance with written and approved station procedures. He has the authority to order the shutdown of the reactor, when in his judgment such action is required to protect the safety of the station or the health and safety of the public. The Operations Manager holds, or has held, a Senior Reactor Operator's License at Seabrook Station. He also supervises the Assistant Operations Managers and the Firefighter Supervisor.

The Assistant Operations Manager - Operations directs the activities of the Shift Managers. He reports to the Operations Manager and assumes the responsibilities of the Operations Manager in the former's absence. He is responsible for the safety and operation of the unit's equipment in accordance with written and approved station procedures. He has the authority to order the shutdown of the reactor, when in his judgment such action is required to protect the safety of the station or the health and safety of the public. The Assistant Operations Manager - Operations holds a Senior Reactor Operator's License.

The Assistant Operations Manager - Support directs the activities of the Operations Department Procedure and Technical Projects groups. He reports to the Operations Manager and assumes the responsibilities of the Operations Manager in his absence. He is responsible for the safety and operation of the unit's equipment in accordance with written and approved station procedures. He has the authority to order the shutdown of the reactor, when in his judgment such action is required to protect the safety of the station or the health and safety of the public. The Assistant Operations Manager - Support holds a Senior Reactor Operator's License.

1. Operating Shift Crew

An operating shift crew normally consists of a Shift Manager and one Unit Supervisor, two Control Room Operators and three Nuclear Systems Operators. The Shift Manager and Unit Supervisor possess Senior Reactor Operator's Licenses; Control Room Operators possess Reactor Operator's Licenses. The minimum shift crew composition for various modes of unit operation is shown in Technical Specification Table 6.2-1.

(a) Shift Manager (SM)

Each Shift Manager reports to the Assistant Operations Manager. The SM is responsible for the safety and operation of the station's equipment in accordance with written and approved station procedures. Each Shift Manager has the authority to order the shutdown of the reactor when in his/her judgment such action is required to protect the safety of the unit or health and safety of the public. The Shift Manager has in addition to a Senior Reactor Operator's License, the training and qualifications of a Shift Technical Advisor or a qualified Shift Technical Advisor will be assigned to his/her shift. The Shift Manager functions as the Shift Technical Advisor and provides requisite technical expertise to the Unit Supervisor in the event of any abnormal operational occurrences.

(b) Unit Supervisor (US)

The Unit Supervisor is responsible for ensuring all unit operations are conducted in accordance with appropriate station orders, procedures and Technical Specifications. The US is responsible for maintaining a record of all shift activities and establishing unit electrical load, as directed by the Shift Manager or as emergency conditions dictate. The US directs the Control Room Operators and the Nuclear Systems Operators in their daily activities. The US has the authority to order the shutdown of the reactor when in his/her judgment such action is required to protect the safety of the unit or the health and safety of the public. Each Unit Supervisor holds a Senior Reactor Operator's License.

(c) Control Room Operators (CRO)

The Control Room Operators monitor the unit's status and make adjustments, as needed, to maintain control of the various plant processes. Most CRO duties are confined to the control room although they may perform specific activities in other areas of the station under the direction of the Unit Supervisor. The Control Room Operators each hold a Reactor Operator's License.

(d) Nuclear Systems Operators (NSO)

The Nuclear Systems Operator performs routine inspections and surveillance activities in other areas of the unit. The NSOs maintain various logs and records as required by station procedures. They also perform routine or special radiation surveys commensurate with the duties of their job. During periods when the unit is shut down, NSOs conduct routine tests and clear/return equipment to service as directed by the Unit Supervisor. The Nuclear Systems Operators are unlicensed.

c. Chemistry and Health Physics

The Chemistry and Health Physics Manager is responsible for the coordination and direction of the Chemistry, Health Physics and Waste Services Departments.

1. Chemistry Department Manager

The Chemistry Department Manager has the direct responsibility for ensuring that the nuclear and steam portions of the station operate within the appropriate water quality specifications which includes water treatment and conditioning for specific station needs. He is responsible for verifying that all liquid, resin and gaseous wastes are properly analyzed and processed for station reuse or disposal.

2. Health Physics Department Manager

The Health Physics Department Manager is the Station Radiation Protection Manager and thus has the responsibility and authority to report to the Station Director (as shown on Figure 13.1-4) on any aspect of the Radiation Protection Program or its implementation as he deems necessary. He normally reports directly to the Chemistry and Health Physics Manager and is responsible for monitoring station activities for compliance with Health Physics-related regulations and programs. The entire station staff, from the Station Director on down, recognizes and honors this responsibility, thereby ensuring that the Health Physics Department Manager, together with the members of his appropriately trained and experienced staff, will fully implement the station radiation protection program.

3. Waste Services Department Manager

The Waste Services Department Manager is responsible for the operation of the Radioactive Waste Processing System and the collection, processing, packaging and loading of radioactive material. This individual provides decontamination services, shielding installation and labor support.

d. Maintenance

The Maintenance Manager is responsible for the coordination and direction of Instrumentation and Control (I&C), Mechanical Maintenance, and Electrical Maintenance, Building Maintenance and Modifications, and Maintenance Technical Departments.

1. Instrumentation and Control Department Manager

The Instrumentation and Control Department Manager is responsible for the maintenance of all instrument and control equipment associated with the reactor, its auxiliary systems and the conventional steam portions of the station. He maintains station process and control instrumentation in proper operating condition, as well as the maintenance of all station radiation monitoring equipment. He also maintains and directs the repair of all control circuitry associated with the reactor, turbine and auxiliary systems. He uses a program of preventative maintenance, corrective maintenance, surveillance testing and record keeping, as required by the station license, approved station procedures, and/or other station requirements.

2. Electrical Maintenance Department Manager

The Electrical Maintenance Department Manager is responsible for the maintenance of all electrical equipment associated with the operation of the station including all electrical distribution and its associated relay schemes. He uses a program of preventive maintenance, corrective maintenance, surveillance testing and record keeping, as required by station license, approved station procedures, and/or other station requirements.

3. Mechanical Maintenance Department Manager

The Mechanical Maintenance Department Manager is responsible for all station mechanical maintenance work required for safe, efficient and dependable service.

He uses a program of preventative maintenance, corrective maintenance, surveillance testing and record keeping, as required by the station license, approved station procedures, and/or other station requirements.

4. Building Maintenance and Modifications Department Manager

The Building Maintenance and Modifications Department Manager is responsible for completing Station Minor Modifications (MMOD) and Design Changes (DCR) to permanent plant systems, structures and components. Additional responsibilities include the upkeep of all buildings and the grounds within the protected area.

5. Maintenance Technical Department Manager

The Maintenance Technical Department Manager is responsible for the management of Maintenance administrative programs/manuals, the Corrective Action program, Self-Assessment program and the Human Performance program for the Maintenance group. Additional responsibilities include management of Key Performance Indicators, budget development and membership on several station steering committees.

e. Fire Protection Program Management

1. Fire Fighter Supervisor

The Fire Fighter Supervisor reports to the Operations Manager. He is responsible for implementation of the fire protection program on site. He develops procedures for testing and surveillance of all fire protection equipment and systems, and is responsible for all fire prevention and protection activities onsite.

2. Fire Brigade Leader

The Fire Brigade Leader is responsible for directing the fire brigade during a fire emergency on site. This duty is performed by a systems-trained individual who is qualified to fire fighting standards and responsible to the Fire Fighter Supervisor.

3. Fire Brigade (Fire Fighter Technicians)

Fire Fighter Technicians of the Fire Brigade are responsible for firefighting on site. Each shift fire brigade will have a minimum of five persons, including the Fire Brigade Leader. The responsibilities of the fire brigade members under normal conditions do not conflict with their responsibilities during a fire emergency.

13.1.2.3 Operating Shift Crews

The position titles, applicable operator licensing requirements, and the minimum numbers of personnel planned for each shift are described in detail in Subsection 13.1.2.2b and Technical Specification Subsection 6.2.2. During normal operations, an operating shift consists of five Nuclear Systems Operators, two Control Room Operators, a Unit Supervisor and a Shift Manager for the station.

During unit refueling operations, when the reactor core configuration is being altered, an individual having a Senior Reactor Operator's license directly supervises the refueling activities in the reactor containment.

Nuclear Systems Operators are trained in applicable station radiation protection procedures to perform routine or special radiation surveys commensurate with the duties of their job. They receive radiation worker training which includes the use of protective barriers and signs, protective clothing and breathing apparatus and limits of personnel exposure. The Shift Manager is responsible for the radiation protection program in the absence of the Health Physics Department Manager or his designated alternate. When fuel is in the reactor, a qualified health physics technician is assigned to the onsite shift to provide additional support to the Shift Manager.

When the unit is in operational modes 1 through 4, a chemistry technician qualified in primary and secondary chemistry analysis is assigned to the onsite shift to provide additional support to the Shift Manager.

13.1.3 Qualification of Nuclear Plant Personnel

13.1.3.1 Qualifications Requirements

The recommendations of Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," Revision 2, have been used as the basis for establishing minimum qualifications for all management, supervisory and professional-technical personnel in the station organization. See Section 1.8, "Regulatory Guide 1.8" for clarification.

The education, training and experience requirements for operators, technicians and mechanics equals or exceeds the qualifications for the positions stated in ANS 3.1-1978 and Regulatory Guide 1.8. A retraining and replacement licensed training program for the Station Staff shall be maintained under the direction of the Training Manager in accordance with the Seabrook Station Institute of Nuclear Power Operations (INPO) Accredited Programs. Established company training programs include documented academic and on-the-job training plus comprehensive qualification examinations applicable to the skill level of the position assignment. Where desirable, offsite facilities may be used for specialized training. Records of the scope, general content and level of accomplishment for each person attending offsite training are retained at the station.

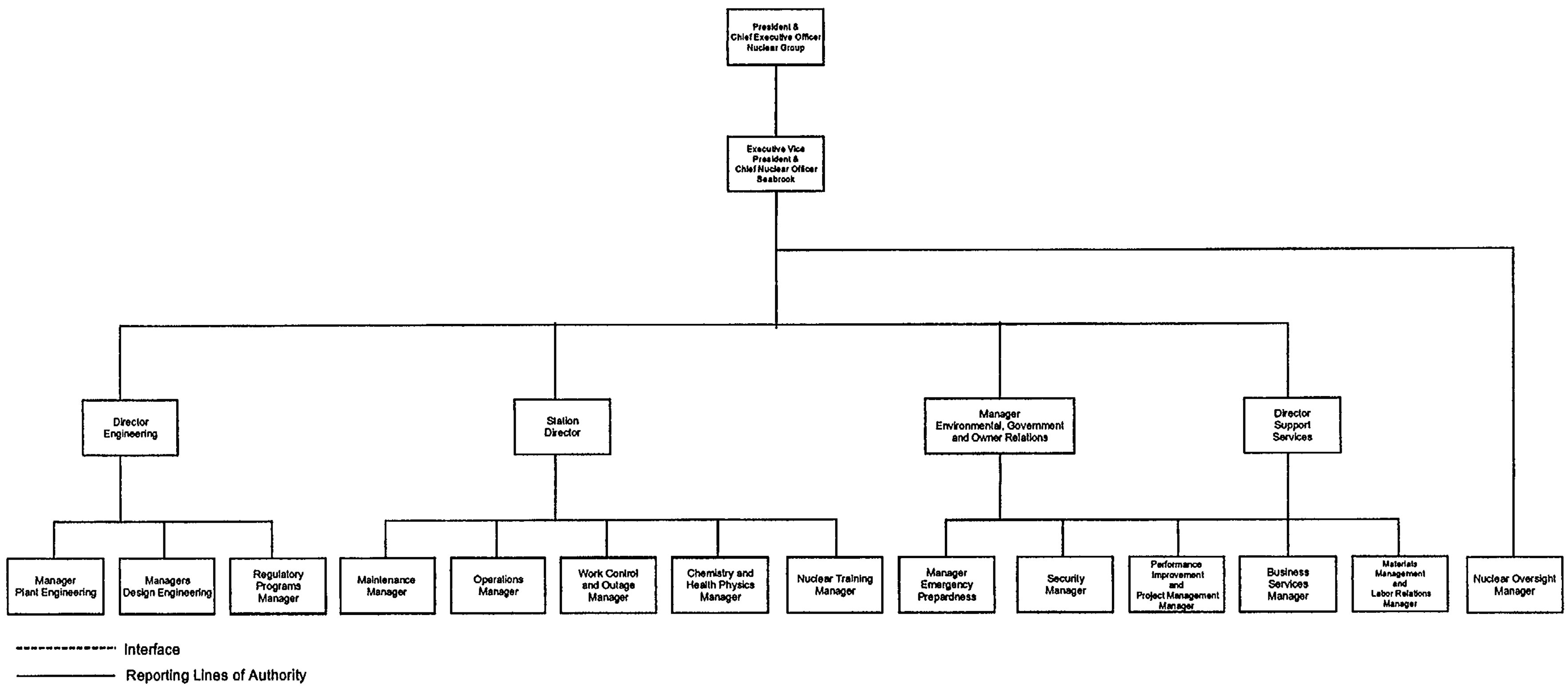
The titles of plant management and supervisory personnel who will meet the minimum requirements of ANS 3.1-1978 and Regulatory Guide 1.8 are listed below with their equivalent ANS 3.1-1978 title.

	Station Title	ANS 3.1 Title
a.	Station Director	Plant Manager
b.	Assistant Station Director	Plant Manager
c.	Operations Manager	Operations Manager
d.	Assistant Operations Manager - Operations	Operations Manager
e.	Assistant Operations Manager - Support	Operations Manager
f.	Maintenance Manager	Maintenance Manager
g.	Chemistry and Health Physics Manager	Supervisor without NRC License
h.	Shift Manager	Supervisor with NRC License
i.	Unit Supervisor	Supervisor with NRC License
j.	Chemistry Department Manager	Supervisor without NRC License
k.	Mechanical Maintenance Department Manager	Supervisor without NRC License
l.	Electrical Maintenance Department Manager	Supervisor without NRC License
m.	Instrumentation and Control Department Manager	Supervisor without NRC License
n.	Health Physics Department Manager	Supervisor without NRC License
o.	Waste Services Department Manager	Supervisor without NRC License
p.	Work Control and Outage Manager	Supervisor without NRC License
q.	Nuclear Training Manager	Supervisor without NRC License

r. Building Maintenance and Modifications Department Manager	Supervisor without NRC License
s. Maintenance Technical Department Manager	Supervisor without NRC License

13.1.3.2 Qualifications of Station Personnel

The key management, supervisory and technical positions in the station organization have been filled by individuals thoroughly trained in their specialty. In addition, most of the individuals have had extensive experience at operating nuclear power plants in their specialty. The nuclear experience of senior personnel at the time of startup was generally in the range of 8 to 20 years. Resumes for personnel holding key positions in the initial plant organization are available upon request. These personnel include the Station Director, Assistant Station Director, Operations Manager, Shift Managers, Instrumentation and Control Department Manager. Many of the key personnel had Senior Operator Licenses or Operator's Licenses at other operating plants or have had extensive nuclear submarine operational responsibilities.
Most of the Unit Supervisors and Control Room Operators and at least one individual in each of the major technical disciplines (nuclear engineering, chemistry, health physics, instrumentation and controls) have at least five years of similar experience.



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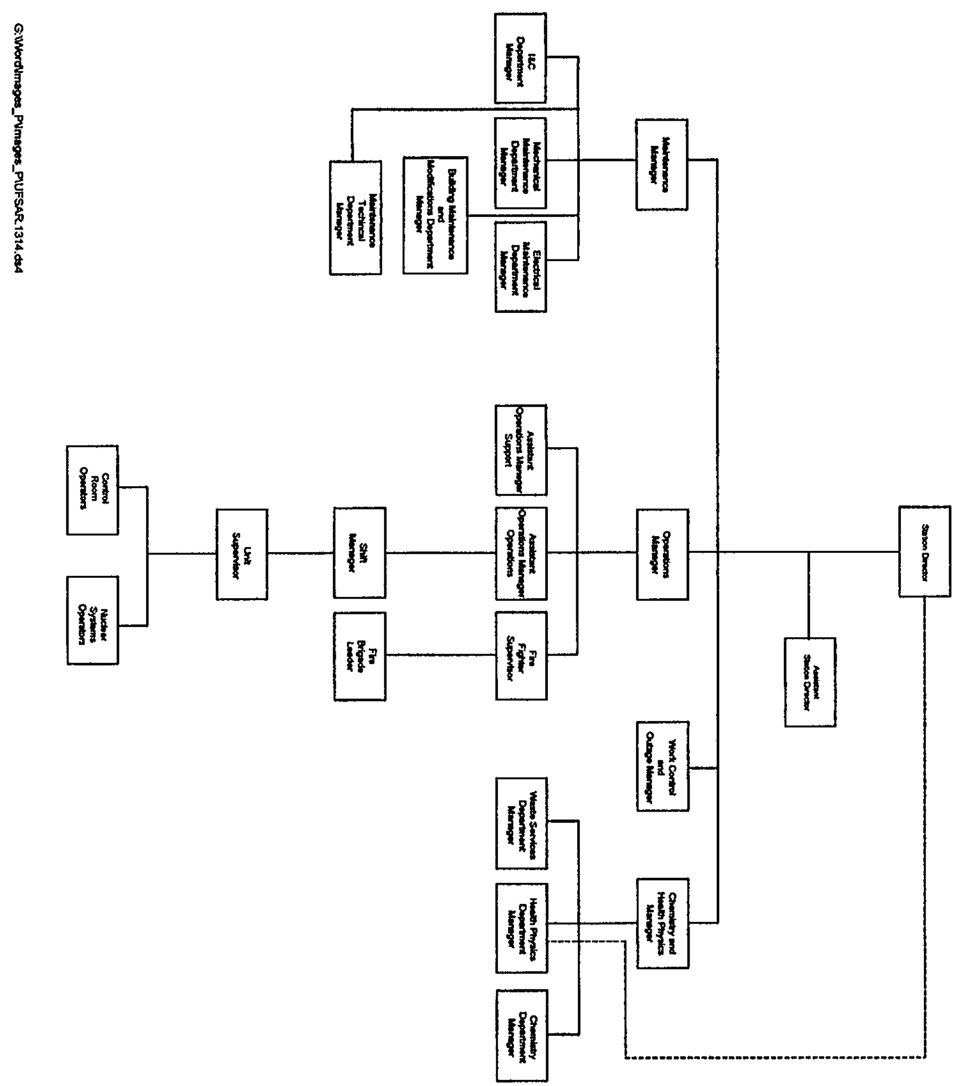
Seabrook Station Updated Final Safety Analysis Report	North Atlantic Organization	
	Rev. 07	Figure 13.1-2



Seabrook Station Updated Final Safety Analysis Report	Seabrook Station Organization	
	Rev. 07	Figure 13.1-4

- INPO Significant Event Reports (SERs) and Significant Operating Experience Reports (SOERs).

2. Retraining Lectures for License Holders

A formal classroom lecture series is conducted as part of the requalification program. The level of instruction will be consistent with the level of license held. This lecture series covers two general areas:

- Fundamentals and Systems Review
- Procedures and Administrative Controls

Fundamentals and Systems Review lectures present instruction based on information from standard reference sources relating to topics such as reactor theory, plant design, and radiation control. Procedures and Administrative Controls lectures cover topics involving essential plant operational guidelines such as Technical Specifications and administrative and operating procedures.

b. On-the-Job Training

On-the-job training is designed to ensure that all licensed personnel operate reactor controls and participate in major evolutions. On-the-job training is conducted throughout the term of the operator's license, and all required on-the-job training is completed prior to license renewal.

Seabrook Station's certified simulation facility is used to ensure that required control manipulations not performed in the plant are performed on the simulator during the term of the operator's license.

The simulator is used in requalification training to emphasize such areas as infrequently performed procedures, required responses to abnormal and emergency procedures, and significant operating events.

c. Requalification Examinations

An examination or quiz is administered in accordance with the training program description. These quizzes examine a combination of classroom and simulator objectives presented to date, and they will parallel, in content and degree of difficulty, segments of an annual requalification examination. Examinations and quizzes will be retained as a part of the training record.

A comprehensive written examination will be administered to all licensees at least every two years. These examinations incorporate many of the requirements of NUREG-1021, Operator Licensing Examination Standards for Power Reactors.

Each application for renewal of an RO or SRO license will be accompanied by a statement, signed by the Station Director, certifying that the applicant has satisfactorily completed the requalification program during the effective term of his or her current license, and that he or she has discharged license responsibilities competently and safely.

13.2.1.6 Requalification Training Program Records

Requalification training program records will be maintained for a minimum of six years from the date of the recorded event to document the participation of each licensed RO and SRO in the program. The records will include copies of written examinations administered, answers provided by the licensees, and results of evaluations.

13.2.1.7 Activation of Inactive License Program

An inactive license is defined as a license held by an individual who has participated fully in the Seabrook Station licensed operator requalification program, but has not actively performed the functions of a licensed RO or SRO for a minimum of seven, eight-hour shifts or five, twelve-hours shifts per calendar quarter.

For an individual with an inactive license to resume the functions authorized by the license, the conditions specified by Section f of 10 CFR 55.53 must be met.

13.2.1.8 Applicable Documents

The training programs listed under Items 1 through 4 in Subsection 13.2.1.a will be conducted in accordance with applicable requirements in the NRC section of Title 10 of the CFR, and they will meet the intent of applicable recommendations provided by Regulatory Guides and other publications. The recommendations and regulatory requirements are included in the program descriptions where applicable.

13.2.2 Training for Nonlicensed Personnel

a. General Discussion

The comprehensive training programs conducted for nonlicensed personnel comply with the provisions and intent of Regulatory Guides and other publications identified in the individual program descriptions. Subsections 13.2.2.1 through 13.2.2.11 have brief descriptions of the training each nonlicensed group receives. The programs meet or exceed the requirements of 10 CFR 50.120.

13.4 REVIEW AND AUDIT

Operating phase activities that affect nuclear safety are reviewed and audited through a comprehensive program. The review and audit program assures proper review and evaluation of proposed changes, tests, experiments and unplanned events. Regulatory Guide 1.33 and ANSI N18.7-1976/ANS 3.2 requirements for reviews and audits form the basis for the program.

13.4.1 Onsite Review

13.4.1.1 SORC

A Station Operation Review Committee (SORC) performs the onsite operational review responsibilities. The purpose of the SORC is to advise the station management on all matters related to nuclear safety. The function, composition, meeting frequency, responsibilities and authority of the SORC are contained in Technical Specification 6.4.1.

13.4.1.2 SORC Charter

A SORC Charter, approved by the Station Director, delineates the rules and procedures by which the SORC functions. The Charter contains the following information:

Name
Basis
Purpose
Authority
Composition
Meeting Frequency
Quorum
Committee Responsibilities
Records
Amendments

The qualification levels of station staff personnel and their alternates assigned to SORC membership meet or exceed those required by Section 4 of ANSI/ANS 3.1-1978.

To perform its duties the SORC established its own rules of practice that include:

a. When less than full membership is present, the quorum ensures that matters to be considered are limited to those that are within the technical competence of the members present.

b. Committee members ensure an appropriate interdisciplinary review of activities under discussion.

c. Subcommittees may be used at the discretion of the chairman. When used, consideration is given to the interdisciplinary composition of the subcommittee membership.

d. The chairman may authorize the use of experience from sources outside the SORC or outside the station staff where the particular matters under consideration cannot otherwise be reasonably resolved.

e. The minutes of each SORC meeting are official plant records and are retained in accordance with the station record retention procedures.

f. In addition to distribution of SORC meeting minutes as provided in the Technical Specification 6.4.1, copies are submitted to other appropriate management.

13.4.1.3 Operations Phase Reviews

The scope of SORC review matters include those noted in Section 4.4 of ANSI 18.7-1976/ANS 3.2 as endorsed by Regulatory Guide 1.33. The scope of SORC operational phase review is specified in Technical Specification 6.4.1.

13.4.1.4 Startup Phase Reviews

The SORC was activated prior to fuel load in order to conduct the following activities:

a. Review test results of the integrated system preoperational tests performed prior to fuel load of the unit.

b. Review and approve the test procedures and test results of initial startup tests.

13.4.2 Independent Review

13.4.2.1 NSARC

A Nuclear Safety Audit Review Committee (NSARC) was operational six months prior to Unit 1 fuel loading. The function, composition, qualification, meeting frequency, responsibilities and authority of the NSARC are contained in Appendix 17C. The goal of the NSARC is to provide management with an independent evaluation of station operations required to ensure safety. The NSARC performs its function through independent reviews and audits of all aspects of station safety.

13.4.2.2 NSARC Charter

An NSARC Charter, approved by the NSARC members and the Executive Vice President and Chief Nuclear Officer, delineates the mechanism for meeting Committee requirements. The Charter contains the following information:

Name
Basis
Function
Composition
Qualifications
Meeting Frequency
Quorum
Review
Audits
Authority
Records
NSARC Operations
Amendments
Attachments

The NSARC composition and qualifications are provided in Appendix 17C. The NSARC shall collectively have expertise and competence in the designated disciplines. Provisions are included in the charter to assure that individuals with appropriate expertise are in attendance at NSARC meetings to review the operational phase activities being discussed. The Charter also provides that no more than a minority of the quorum have line responsibility for the operation of the station. The NSARC members are generally selected from North Atlantic; however, members may also be selected from outside consultants or organizations. The minimum qualifications of all NSARC members meet or exceed those specified in ANS 3.1-1978. Resumes of the regular NSARC members and alternates are maintained by the NSARC chairman once committee members are selected.

The NSARC is advisory in nature, making recommendations to company management as deemed necessary. The NSARC has access to the station and station information at any time in carrying out its responsibilities. The charter establishes the powers of the chairman and the use of subcommittees in performing its function. It also establishes the requirements for distribution of reports and/or meeting minutes and its authorization for initiating work.

13.4.2.3 <u>Reviews</u>

Appendix 17C defines the Committee review requirements. These are conducted through a combination of document summary reviews, presentations at regularly scheduled meetings and special meetings held to review proposed changes to the Operating License or Technical Specifications. The NSARC will be required to review and approve all Operating License/ Technical Specification changes prior to their submittal to the NRC. Particular document review assignments may be made by the NSARC to either individual NSARC members or to subcommittees with the appropriate expertise. Their reports or summary documents are reviewed by the full Committee at regularly scheduled meetings. Certain activities, such as In-Plant Audits, are performed under the cognizance of the NSARC. In such cases, written reports are issued to the Committee and the results discussed at regularly scheduled meetings.

13.4.2.4 Audit Program

NSARC audits are considered management audits and are normally performed under the quality assurance audit program described in the Seabrook Operational Quality Assurance Program contained in Section 17.2.

13.4.3 Independent Technical Reviews

A Technical Review Program is established, implemented and maintained to encompass the following Technical Review Responsibilities and to address the reviews of NUREG-0737, Task I.B.1.2.

13.4.3.1 Function

The Technical Review Program responsibilities encompass the following:

a. NRC issuances, industry advisories, Licensee Event Reports, and other sources that may indicate areas for improving plant safety;

b. Internal and external operating experience information that may indicate areas for improving plant safety;

c. Plant operating characteristics, plant operations, modifications, maintenance and surveillance to verify independently that these activities are performed safely and correctly and that human errors are reduced as much as practical; and

d. Making detailed recommendations to the Senior Site Official for procedure revisions, equipment modifications or other means of improving nuclear safety and plant reliability.

The Technical Review Program utilizes several onsite personnel who are independent of the plant management chain to perform the reviews described above and specified in Technical Specification 6.2.3.1.

13.4.3.2 Records

Written records of technical reviews are maintained. As a minimum, these records include the results of the activities conducted and the status of recommendations made pursuant to Technical Specification 6.2.3.1 and an assessment of company operations related to the reviews performed.

(e) Lower Tier Procurement. Provisions for extending applicable requirements to lower tier subcontractors and suppliers, including purchaser's access to facilities and records.

q. Control of Purchased Material, Equipment and Services

Several procedures include the requirements of ANS 3.2, Subsection 5.2.13.2, i.e.:

1. Provide measures which assure that purchased items and services, whether purchased directly or through contractors, conform to the procurement documents. These measures shall include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor, inspection and audit at the source and examination of items upon delivery.

2. Measures for evaluation and selection of procurement sources include the use of historical quality performance data, source surveys or audits, or source qualification programs.

3. Source inspection or audit shall be performed as necessary to assure the required quality of an item. Source inspection or audit may not be necessary when the quality of the item can be verified by review of test reports, inspection upon receipt, or other means.

4. Where required by code, regulation, or contract requirements, documentary evidence that items conform to procurement requirements shall be available at the nuclear power plant site prior to installation or use of such items. This documentary evidence shall be retrievable and shall be sufficient to identify the specific requirements such as codes, standards and specifications met by the purchased item. Where not precluded by other requirements, such documentary evidence may take the form of written certifications of conformance which identify the requirements met by the items, provided means are available to verify the validity of such certifications.

5. The effectiveness of the control of quality shall be assessed by North Atlantic at intervals consistent with the importance, complexity and quality of the item or service.

r. Identification and Control of Materials, Parts and Components

Several procedures include the requirements of ANS 3.2, Subsection 5.2.13.3, i.e.:

1. Provide measures which identify and control materials, parts, and components including partially fabricated subassemblies.

2. Ensure that only correct and accepted items are used and installed, and that an item of production (batch, log, component, part) at any stage, from initial receipt through fabrication, installation, repair or modification is related to an applicable drawing, specification, or other pertinent technical document.

3. Physical identification shall be used to the maximum extent possible. Where physical identification is either impractical or insufficient, physical separation, procedural control or other appropriate means shall be employed. Identification may be either on the item or on records traceable to the item, as appropriate.

   Where identification marking is employed, the marking shall be clear, unambiguous and indelible, and shall be applied in such a manner as not to affect the function of the item. Markings shall be transferred to each part of an item when subdivided and shall not be obliterated or hidden by surface treatment or coatings unless other means of identification are substituted.

4. When codes, standards or specifications require traceability of materials, parts or components to specific inspection or test records, the procedures shall provide requirements for and methods to ensure such traceability.

s. <u>Handling, Storage and Shipping</u>

Several procedures include the following:

1. The requirements of ANS 3.2, Subsection 5.2.13.4, i.e.:

   (a) Provide measures to control handling, storage and shipping, including cleaning, packaging and preservation of material and equipment in accordance with established instructions, procedures or drawings, to prevent damage, deterioration and loss.

   (b) When necessary for particular items, special coverings, special equipment and special protective environments, such as inerted atmosphere, specific moisture content levels and temperature levels shall be specified, provided, and their existence verified.

   (c) For critical, sensitive, perishable or high-value articles, specific written procedures for handling, storage, packaging, shipping and preservation should be

3. The Technical Requirements, described in Section 16.3, for operation and surveillance of fire detection instrumentation, fire suppression systems and fire barrier penetrations.

4. The intent of the guidance in the 1977 NRC document entitled "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance."

aa. Communication

Several procedures include the following:

1. The requirements of Appendix A to Regulatory Guide 1.33, i.e., Item 1.m, "Communication Systems Procedure."

2. The requirements for normal and emergency communications both internal to the station and to external organizations such as North Atlantic Headquarters, Duke Engineering and Services, NRC and other federal, state and local organizations or commercial activities.

ab. Control of Radioactivity

Several procedures include the following:

1. The requirements of Appendix A to Regulatory Guide 1.33, i.e.; Item 7.e, "Radiation Protection Procedures" that limit materials released to the environment and limit personnel exposure. These procedures provide controls such as:

   (a) Access control to radiation areas including a radiation work permit system

   (b) Radiation surveys

   (c) Airborne radioactivity monitoring

   (d) Contamination control

   (e) Respiratory protection

   (f) Training in radiation protection

   (g) Personnel monitoring

   (h) Bioassay program

   (i) Implementation of ALARA program.

2. The requirements of Technical Specifications, i.e.:

(a) Item 6.10, "Radiation Protection Program."

(b) Item 6.12.2, "Process Control Program."

13.5.2 Operating Procedures

13.5.2.1 Conformance with Regulatory Guide 1.33

The operating procedures for Seabrook Station implement the recommendations of Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Rev. 2, dated February 1978. Specific alternatives, clarifications and exceptions to these requirements, and to ANSI N18.7-1976/ANS 3.2 are listed in Sections 1.8 and 17.2.

13.5.2.2 Preparation of Procedures

To provide adequate time for familiarization and training, operating procedures are completed and approved about six months prior to their anticipated need to accomplish nuclear safety-related activities. In those instances where meaningful procedures cannot be developed six months prior to use, they will be developed, reviewed, approved and training conducted prior to operation of the system or component.

Operating procedures and any changes to them are reviewed and approved by appropriate station supervisory and management personnel and are in accordance with Section 6.7 of the Technical Specifications and Updated FSAR Subsection 13.4.1.

13.5.2.3 Procedure Description

a. Control Room Operating Procedures

Operating procedures, which are performed by licensed operators in the control room, encompass those activities, systems and evolutions recommended in the applicable parts of Appendix A to Regulatory Guide 1.33 and Section 6.7 of the Technical Specifications. They are considered to fall into the following categories:

1. Station Operating Procedures

This category includes procedures for the integrated startup, operation and shutdown of the station. Table 13.5-2 presents a representative listing of Station Operating Procedures.

2. System Operating Procedures

This category includes individual system procedures and safety-related systems and major secondary plant systems and equipment. These procedures give instructions for energizing, filling, venting, draining, startup, shutdown, and changing