Chapter 10

STEAM AND POWER CONVERSION SYSTEM

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STEAM AND POWER CONVERSION SYSTEM

10.1 SUMMARY DESCRIPTION

The steam and power conversion system is designed to produce electrical energy through conversion of a portion of the thermal energy contained in the steam supplied from the reactor, to condense the main turbine exhaust steam, and to return condensate to the reactor as heated feedwater with a major portion of its gaseous dissolved and particulate impurities removed.

The power conversion system uses the Rankine steam cycle with a closed regenerative feedwater heating cycle. It has the capability to accept 105% of the reactor's rated steam flow. Steam leaves the reactor vessel at 1035 psia. Steam enters the turbine at 999 psia with a 0.30% moisture content. The turbine is a tandem-compound turbine generator having a six-flow exhaust end. Steam is exhausted into a triple pressure condenser designed for a 2.4-in. Hg average backpressure and is condensed with circulating water cooled by mechanical draft cooling towers. Six stages of regenerative feedwater heating are provided, four heated with extraction steam from the low pressure turbines and two from the high pressure turbine. The final design feedwater temperature at normal full load is 423°F.

The major components of the steam and power conversion system are the turbine generator, main condenser, condensate pumps, condensate booster pumps, mechanical vacuum pumps, steam jet air ejectors, turbine gland sealing system (which includes gland seal steam evaporators and condenser), turbine bypass system, condensate filter demineralizers, turbine driven reactor feed pumps, feedwater heaters, and condensate storage facilities. The turbine cycle heat balance for rated 102.4% power is given in Figure 10.1-1. This figure is representative of the overall power conversion system.

The saturated steam produced by the boiling water reactor is passed through the high pressure turbine where the steam is expanded and is then exhausted to two moisture separator/reheaters (two reheat stages) arranged in parallel. The moisture separators remove the moisture content of the steam and superheat the steam before it enters the low pressure turbines where the steam is expanded further.

Steam for the first-stage reheater is taken from the first extraction point of the high pressure turbine while steam for the second-stage reheater is taken from the main steam header. From the low pressure turbines, the steam is exhausted into the main condenser where it is condensed and deaerated. The condensate pumps take suction from the condenser hotwell and deliver the condensate through the gland seal steam condenser, steam-jet air ejector condenser, offgas condenser, and condensate demineralizers to the condensate booster pump suction. The condensate booster pumps then discharge through the low pressure feedwater heater trains to

the reactor feedwater pumps. The reactor feedwater pumps supply feedwater through the high pressure feedwater heaters to the reactor. Steam for heating the feedwater in the heating cycle is supplied from turbine extractions. The drains from the feedwater heaters, the reheaters, and the moisture separators are cascaded to the next lower pressure feedwater heater and finally discharged to the condenser.

The ability of the plant to follow system loads depends on the adjustment of the reactor power level. The steam admission valves are controlled by the initial pressure regulator so that the turbine receives the proper amount of steam required for the load demand. The turbine speed governor, however, may override the initial pressure regulator to close the steam admission valves if an increase in system frequency or loss of generator load causes an increase in turbine speed. Reactor steam in excess of that which the admission valves will pass is bypassed directly to the main condenser through pressure controlled bypass valves. Load rejection in excess of bypass capacity causes the reactor safety/relief valves to open.

The main turbine, main condenser, and moisture separator/reheaters are located in a shielded area with controlled access to limit personnel exposure.

The portions of the power conversion system which constitute part of the reactor coolant pressure boundary are the main steam lines extending from the reactor pressure vessel to the outermost containment isolation valve.

Table 3.2-1 indicates the safety class, quality group classification, and seismic category of the power conversion system. Environmental design bases are discussed in Section 3.11.

Table 10.1-1 presents a summary of important design and performance characteristics of the steam and power conversion system.

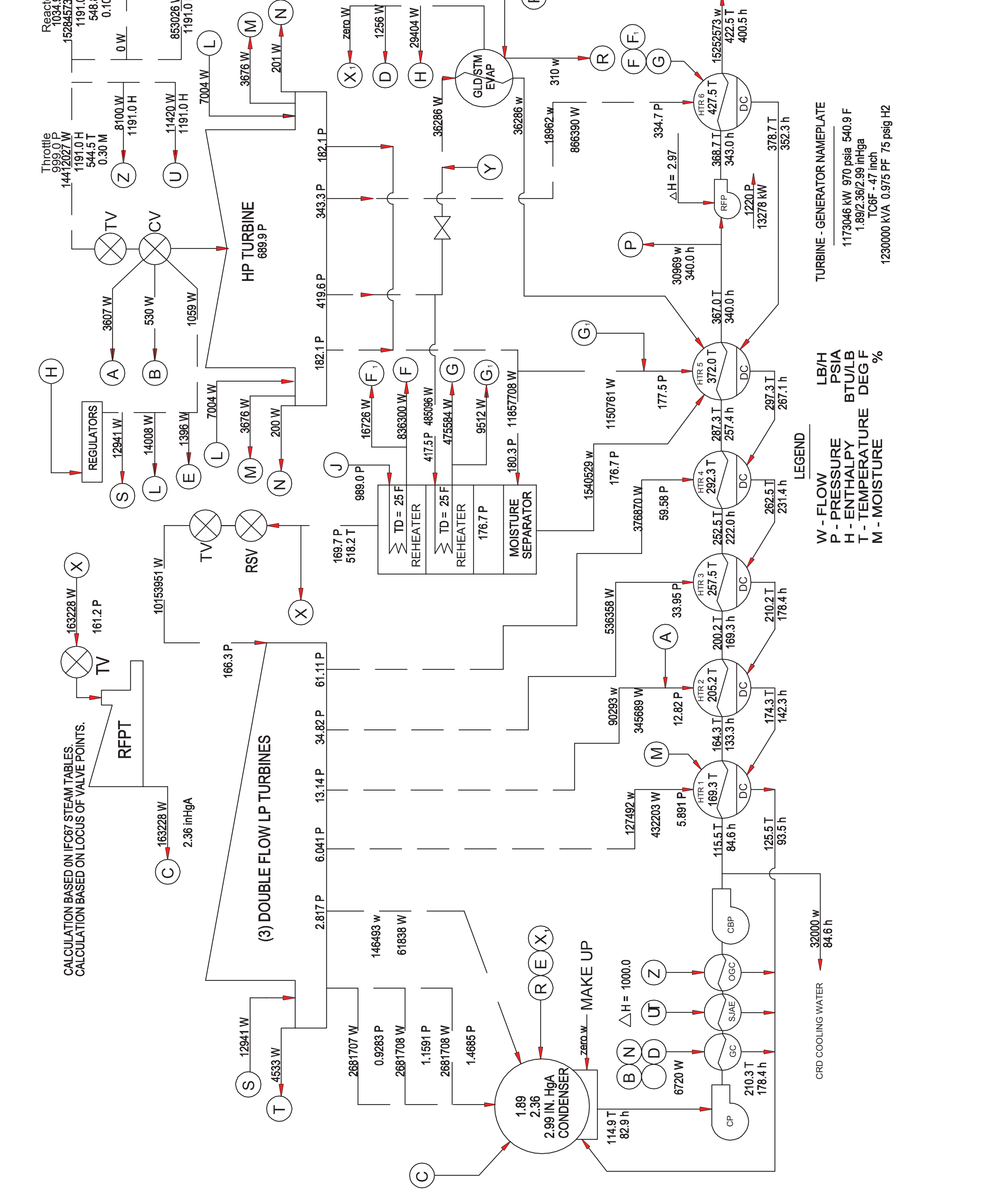
The design of various steam and condensate instrumentation systems are based on the need to monitor and control normal power generation system functions such as level, flow, pressure, and temperature. The instrumentation provides information that enables the control room operator to start up, operate, and shut down these systems.

Table 10.1-1

Design and Performance
Characteristics of Power Conversion System[a]

Turbine Data	
Manufacturer	Westinghouse
HPT building block	BB 296
LPT building block	BB 281R
LPT type/LSB length (in.)	TC6F/47
Number of casings	1-HP, 3-LP
Backpressure zones (in. Hg abs)	1.89/2.36/2.99 (2.4 average)
Generator	
Rating (kVA)	1,230,000
Gross output (MWe)	1230
Power factor	0.975
Voltage (volts)	25,000
Phase/frequency (Hz)	3/60
Hydrogen pressure (psig)	75-78
Steam conditions at throttle valve	
Flow (lb/hr)	14,412,027
Pressure (psia)	999
Temperature (°F)	544.5
Enthalpy (Btu/lb)	1191
Moisture content, maximum (%)	0.30
Turbine cycle heat rate (Btu/kW-hr)	9605
Final feedwater temperature (°F)	422.5
Turbine cycle arrangement	
Steam reheat stages	2
Number of feedwater heating stages	6
Feedwater heater in condenser neck	First stage
Type of condensate demineralizer	Powdered resin
Main steam bypass capacity (%)	23.6

[a] All data based on 100% load. See Figure 10.1-1.



1259001 kW Net Load Heat Balance

Columbia Generating Station
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Draw. No. 960222.24 | Rev. | Figure 10.1-1

Form No. 960690
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10.2 TURBINE GENERATOR

10.2.1 DESIGN BASIS

The turbine generator is designed to receive steam from the boiling water reactor, convert a portion of the thermal energy contained in the steam to electric energy, and provide extraction steam for feedwater heating.

The turbine generator, associated systems, and control characteristics, are integrated with the features of the reactor and associated systems to obtain an efficient and safe power generator unit. The turbine generator is designed to function only during normal plant conditions including startup, power generation, and shutdown. The turbine generator is not required for safe shutdown of the reactor nor to perform safety functions. The turbine generator equipment is in strict conformance with the latest edition in effect at the time of fabrication, of ANSI C.50.10, ANSI C.50.13, and the IEEE standards. The major portion of the manufacture was performed during 1975. The original LP turbine rotors were replaced with fully integral rotors during 1992. Safety class and seismic category are presented in Section 3.2.

The turbine generator design conditions are included in Table 10.1-1.

The turbine generator is intended for base load operation. Normal load swings are limited to the rate of change of power output of the nuclear steam supply system.

The turbine governor valves are capable of full stroke opening and closure within 7 sec for adequate pressure control performance. Normal governor valve closure is shown in Figure 10.2-1.

During events resulting in turbine throttle or governor valve fast closure, turbine inlet steam flow is not reduced faster than permitted by Figures 10.2-2 or 10.2-3.

10.2.2 SYSTEM DESCRIPTION

The main turbine is a tandem-compound unit, consisting of one double-flow high pressure turbine and three double-flow low pressure turbines (Figure 10.3-1), running at 1800 rpm with 47 in. last-stage blades. Exhaust steam from the high pressure turbine passes through two moisture separator/reheaters (two stage reheat) before entering the low pressure turbine inlets. The exhaust steam from the three low pressure turbines is condensed in the main condenser.

The generator is a three phase, 60 cycle, 25,000 V, 1800 rpm unit rated at 1,230,000 kVA at 0.975 power factor. The stator is water cooled and the rotor is hydrogen cooled. The hydrogen system is designed to minimize the hazard from fires or explosions as discussed in Appendix F.

The design of the system, Figures 10.2-4 and 10.2-5, and the specified operating procedures are such that explosive mixtures are not possible under normal operating conditions. The hydrogen gas supply system includes a storage trailer and storage cylinders used as backup if the trailer supply runs low. Pressure regulators are mounted on both the storage trailer and the bottle manifold for control of the hydrogen gas, and a circuit for supplying and controlling the carbon dioxide used in purging the generator during filling and degasing operations. To prevent hydrogen leakage by the generator shaft seals, a hydrogen seal oil system is provided. The hydrogen seal oil system, which includes pumps and controls, deaerates the oil before it is sent to the shaft seals.

The fundamental rule is that hydrogen and air should never be mixed. Carbon dioxide is used as an intermediate gas when changing either from air to hydrogen or from hydrogen to air. When changing from one gas to another, the generator is vented to the atmosphere. The valves, pressure gauges, regulators, and other equipment in the hydrogen gas supply system permit introducing hydrogen or prevent the flow of hydrogen into the generator and also provide means of controlling the gas pressure within the generator.

Steam is transported from the reactor by four main steam lines and flows through the turbine throttle valves and governor valves to the high pressure turbine. The steam lines are combined upstream of the throttle and governor valves. The turbine bypass valves are located upstream of the turbine throttle valves to permit steam bypass to the main condenser during transient conditions.

Two branch lines from the main steam heater supply steam to the two second-stage reheaters per moisture separator. The steam for the two first-stage reheaters per moisture separator is supplied by extraction lines from the high pressure turbine (see Figure 10.3-1). Moisture preseparator units remove moisture from the lower high-pressure turbine discharge exhaust steam as it exits the turbine. The moisture separator/reheaters remove the moisture from the high-pressure turbine exhaust steam and superheat the steam prior to admission to the low pressure turbines, thereby improving overall cycle efficiency. Extraction steam from the high pressure turbine is used in the first-stage reheater and for feedwater heating in heaters No. 5 and 6. Extraction steam from the low pressure turbines is used for the first, second, third, and fourth stage feedwater heaters. Moisture separator/reheater relief valves are provided to prevent overpressurizing the moisture separator/reheaters and the crossover lines in the event of crossover throttle or intercept valve closure; these relief valves discharge to the main condenser.

The turbine generator is equipped with a digital electrohydraulic (DEH) control system. See Section 7.7.1.5 for a detailed description of the turbine control system.

There are four methods of turbine overspeed control protection:

a. Digital Electro-Hydraulic (DEH) speed control,
b. Overspeed protection controller (OPC),
c. Digital control overspeed trip, and
d. Digital trip overspeed trip.

DEH Speed Control

The DEH speed control is designed to maintain turbine speed within 2-3 rpm of setpoint during startup; after the turbine generator has been synchronized to the grid, the grid frequency controls turbine speed. The DEH control system monitors turbine speed via three speed sensors. Upon detecting a separation from the grid and a resulting overspeed condition, the DEH speed control will rapidly close the governor valves via their servo-valves preventing an excessive overspeed condition from occurring.

Overspeed Protection Controller

The OPC primary function is to avoid excessive turbine overspeed such that a turbine trip is avoided. At 103% of rated speed, the OPC solenoids open, rapidly closing the governor and intercept valves to arrest the overspeed before it reaches the trip setting. When turbine speed falls below 101%, turbine speed control is returned to the DEH speed control mode.

Digital Control Overspeed Trip, Digital Trip Overspeed Trip and Quadvoter Hydraulic Trip Block

If the turbine accelerates further than 103% of rated speed, the digital control overspeed trip logic in the DEH control system will provide a trip signal that causes the quadvoter hydraulic trip block to de-energize and trip the turbine. Additionally, the digital trip overspeed trip has three redundant speed sensors and will initiate an independent trip of the quadvoter hydraulic trip block. Both the digital control overspeed trip and the digital trip overspeed trip use two out of three overspeed logic to initiate the trip signal prior to reaching 111% of rated speed. These signals cause the output module for the quadvoter to simultaneously de-energize, or trip, all of the quadvoter valves.

Redundant power supplies are auctioneered to assure loss of one power supply does not cause the quadvoter to trip. The quadvoter provides two channels, each with two solenoid valves in series, to depressurize the trip header and trip all the throttle, governor, intercept and reheat stop valves. The quadvoter design assures that a single failure of a quadvoter valve will neither cause the turbine to trip nor prevent the turbine from tripping if required. The DEH control system is designed to maintain the turbine speed below 120% of rated speed.

The quadvoter trip block assembly is a fail-safe design. Therefore a loss of all power or a loss of all signals to the quadvoter solenoids would cause a turbine trip as all of the quadvoter solenoid valves would de-energize. The turbine overspeed control equipment and electrical wiring may be destroyed by a postulated piping failure; however, this loss would result in a turbine trip based on the fail-safe design. A missile may destroy the electromagnetic speed pickups and associated electrical wiring, but the turbine will still trip on loss of all speed probe signals. Missile damage to the hydraulic lines for the trip block assembly would result in a loss of high pressure fluid thereby depressurizing the trip header and causing a turbine trip.

The operation of the DEH control system is continuously monitored during turbine generator operation. Detection of turbine speed variation is accomplished by the speed-control unit discussed in Section 7.7.1.5. The overspeed protection controller and two digital overspeed trips are tested during reactor startup from refueling outages. The turbine throttle, governor, interceptor, reheat stop valves and quadvoter solenoid valves are periodically tested during operation. Turbine throttle, governor, interceptor and reheat stop valves are periodically inspected. The manner and frequency of the inspection and testing will take into consideration the manufacturer's and others' recommendations and missile probability analysis (see Section 3.5.1.3) in conjunction with the plant generating requirements.

Instrumentation for the turbine generator is provided in the control room and is described in Section 7.7.1.5.

The turbine is equipped for normal operations with a shaft-driven lubricating oil pump and ac motor-driven lubricating oil pump for startup, shutdown, and turning gear, or for emergencies whenever oil pressure falls below set pressure. The turbine is also provided with a dc motor-driven lubricating oil pump with power supplied from storage batteries for emergency operation.

The turbine shaft is supplied with "clean" (essentially nonradioactive) sealing steam which prevents outleakage of steam from the high-pressure turbine and inleakage of air to the low pressure turbines. An evaporator generates essentially nonradioactive steam for turbine gland sealing (see Section 10.4.3).

Overpressure protection of the turbine exhaust hoods and the main condenser shell is provided by rupture diaphragms on the exhaust hoods.

The turbine incorporates protective devices including the exhaust hood relief diaphragms, exhaust hood temperature alarm, pilot dump valve for closing the extraction steam nonreturn valves, low vacuum alarm, thrust bearing wear alarm, and low bearing oil pressure alarm.

In addition, the following tabulation is a list of the turbine generator protective trips:

Mechanical Faults	Electrical Faults
Low vacuum	Generator power differential
Thrust bearing wear	Generator underfrequency
Low oil pressure	Generator differential current
Overspeed	Generator stator ground
Manual	Generator loss of excitation
Anti-motoring	Generator negative sequence
Low DEH pressure	Generator overcurrent during starting
Low EH (electrohydraulic) fluid level	Generator stator ground during starting
Reactor high water level	Generator overexcitation
RCIC-V-13 and 45 open	Generator/transformer overall differential
Moisture separator reheater shell side high level	Manual
Loss of DEH control power	Unit lockout
Both Throttle Valves on a steam chest closed	Unit overall lockout

The main steam throttle and governor valves are located in the steam chest assembly which is parallel to the axis of the high pressure turbine. The nominal closure time for a fully open throttle or governor valve is 0.15 sec. A failure of one governor valve causes the other valves to increase or decrease their opening to compensate for that valve. If one valve fails open at low load condition, the other valves close. If the closing of the other valves is not enough to compensate for that valve, the turbine load increases proportional to steam flow.

The reheat stop and intercept valves are in-line valves located in the crossover piping between the moisture separator/reheater and low pressure turbine. The closure time upon depressionization of the trip header for a fully open valve is 0.15 sec.

The valves described above are periodically tested as required by the Licensee Controlled Specifications (LCS) by using the DEH control system during power operation. Pressure variations caused by closing a governor valve cause the other governor valves to open. Therefore, testing must be done at a reduced power level to provide sufficient margin for pressure control. Details of the pressure control system are discussed in Section 7.7.1.5. In addition, one of each valve will have its internals periodically inspected as required by the LCS.

Each of the extraction steam lines has a reverse current valve and a gate valve, with the exception of the extraction lines to low pressure number 1 heaters. These valves are located near the condenser. On turbine trip the reverse current valves close immediately on reverse flow. Because of the fast closure and the short distance between these valves and the extraction points at the turbine, the amount of steam in these lines does not affect the turbine coastdown following a turbine trip.

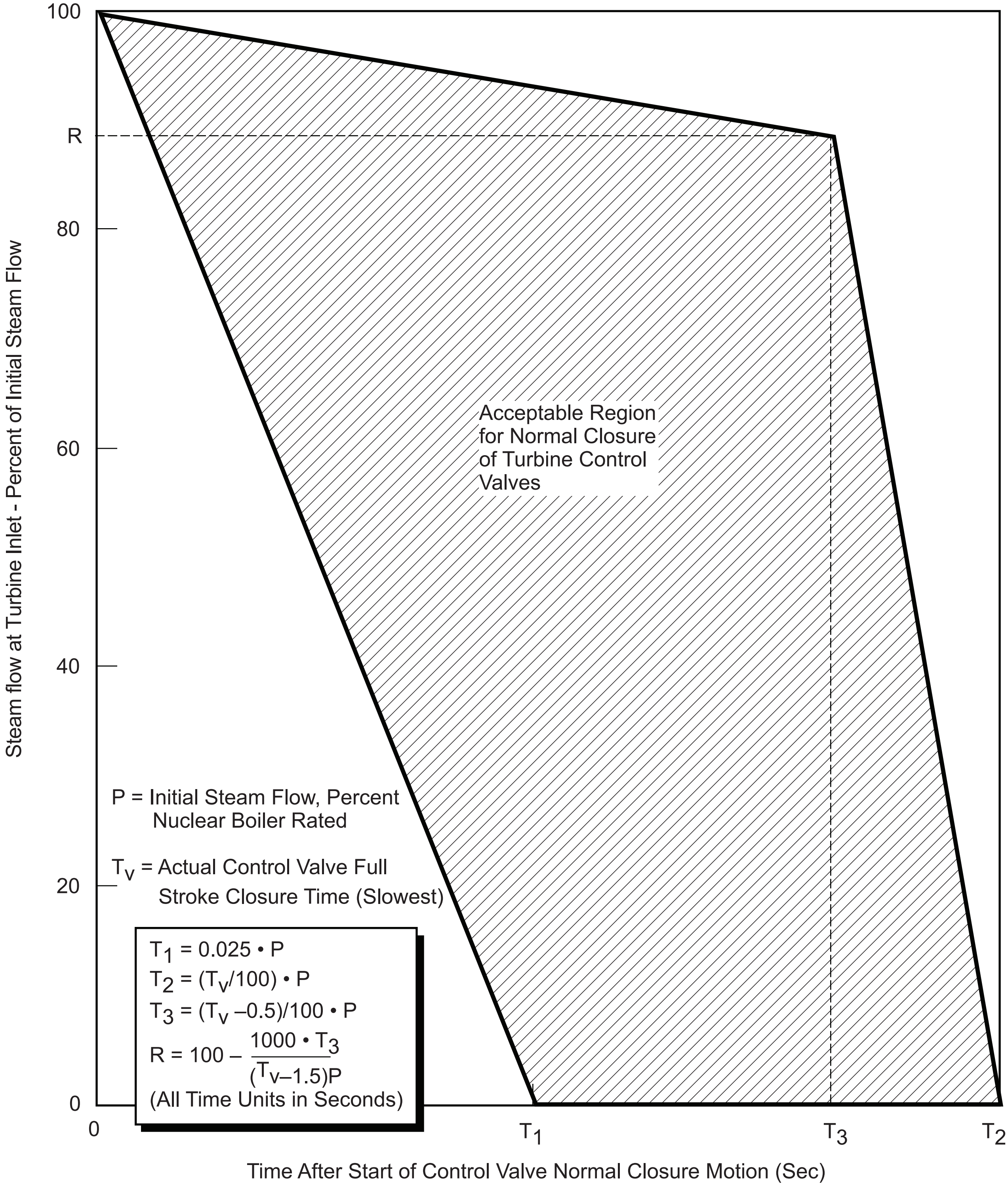
10.2.3 TURBINE DISK INTEGRITY

Analysis of potential turbine missile hazards and drawings showing the orientation of the turbine with respect to important structures are presented in Section 3.5.1.3.

10.2.4 SAFETY EVALUATION

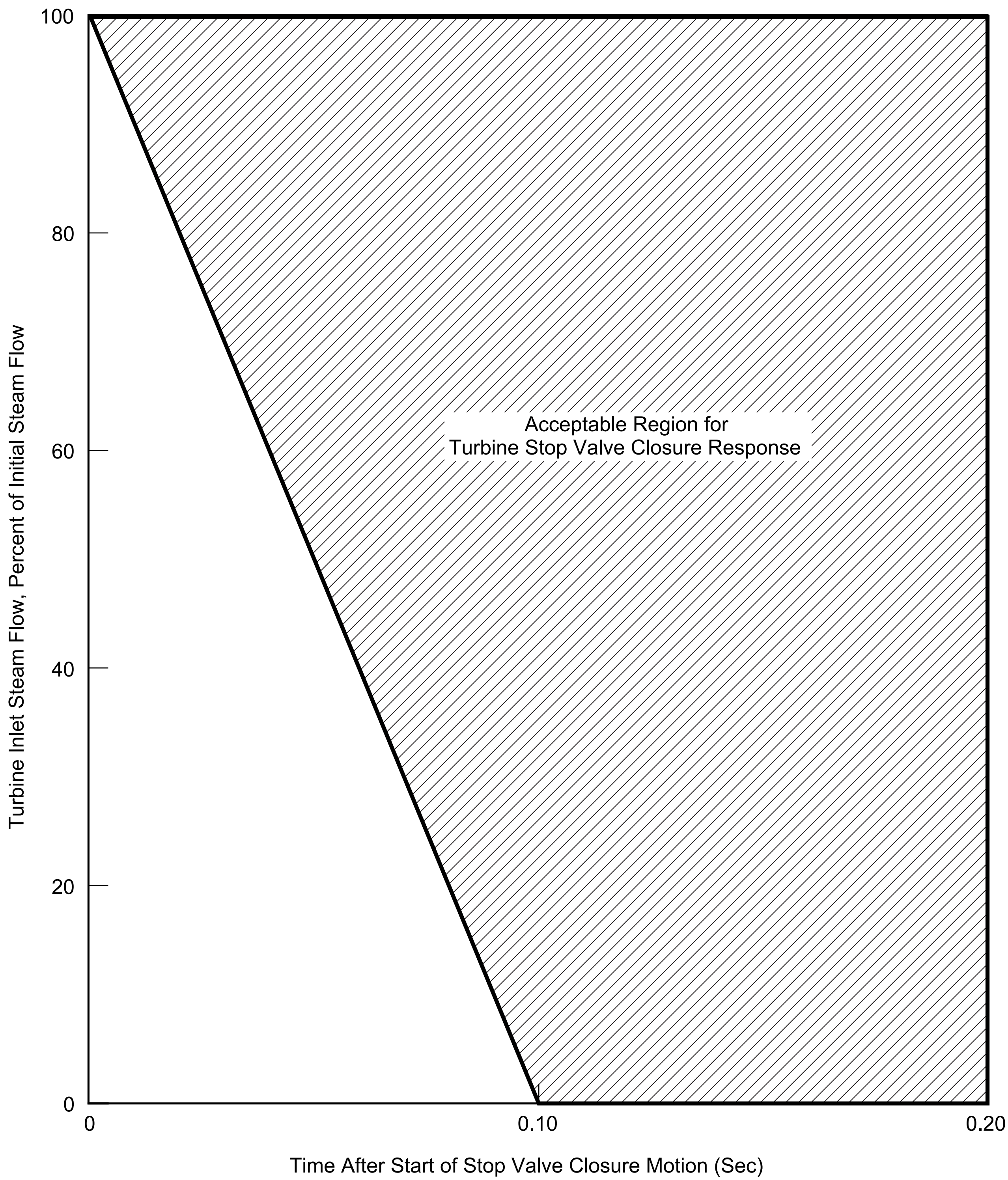
The steam entering the high pressure turbine may contain fission, coolant activation, and activated corrosion products. The anticipated concentration of nitrogen-16, which is the dominant radionuclide entering the high pressure turbine, is discussed in Section 12.2. Moisture separation and transit time between the high pressure and low pressure turbines reduces the concentration of radionuclides in the steam prior to entering the low pressure turbine. Most of the gaseous radioactivity is removed by the steam-jet air ejector and routed to the offgas system (see Section 11.3). The condensate in the condenser hotwell contains significantly less radioactive material than the inlet steam.

Access to the turbine area is controlled. Radiation levels associated with turbine components are described in Section 12.2 and shielding requirements are discussed in Section 12.3.



Acceptable Range for Control Valve Normal Closure Motion

Draw. No. 950021.39	Rev.	Figure 10.2-1



Columbia Generating Station
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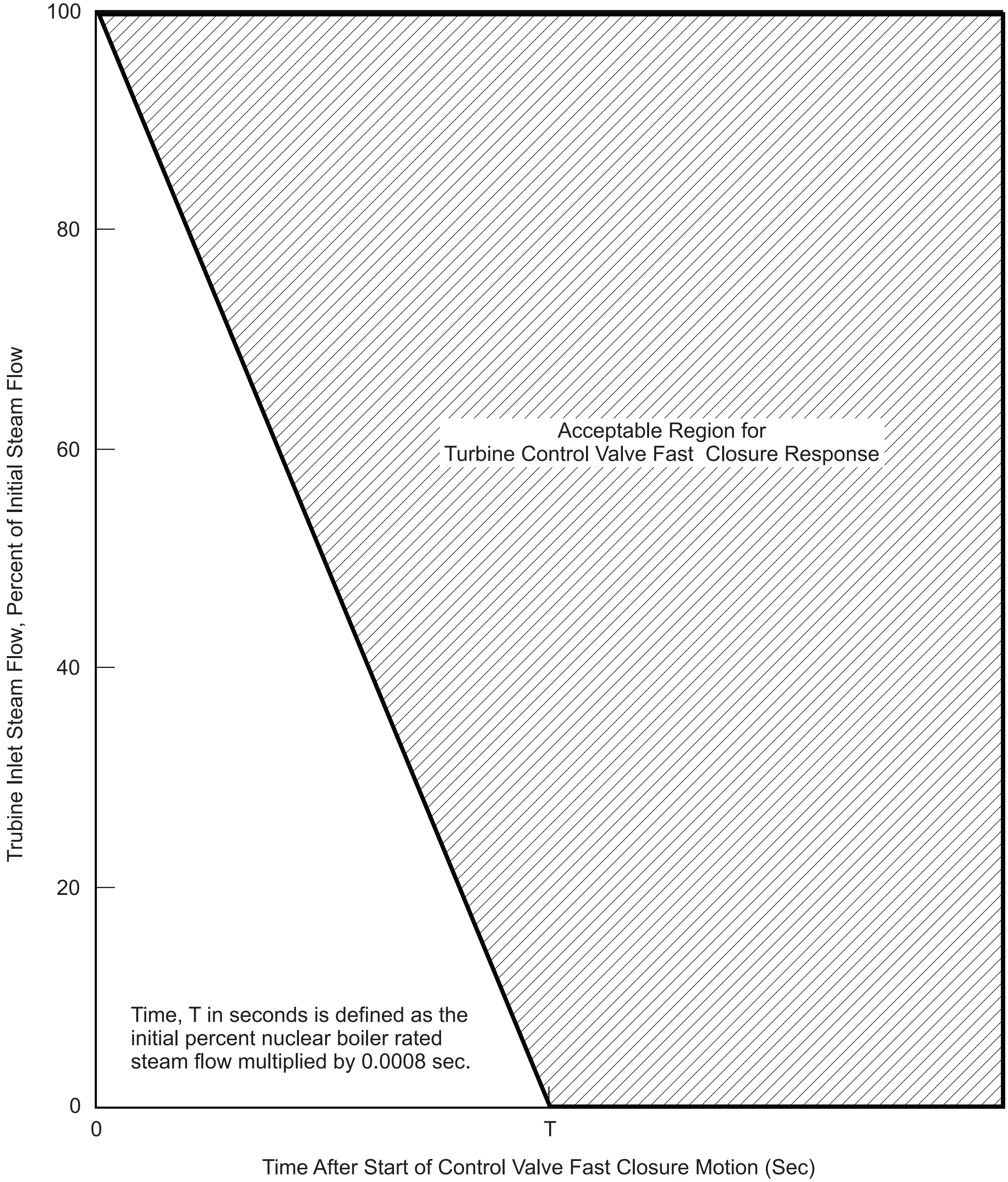
Turbine Stop Valve Closure Characteristic

Draw. No. 950021.40	Rev.	Figure 10.2-2



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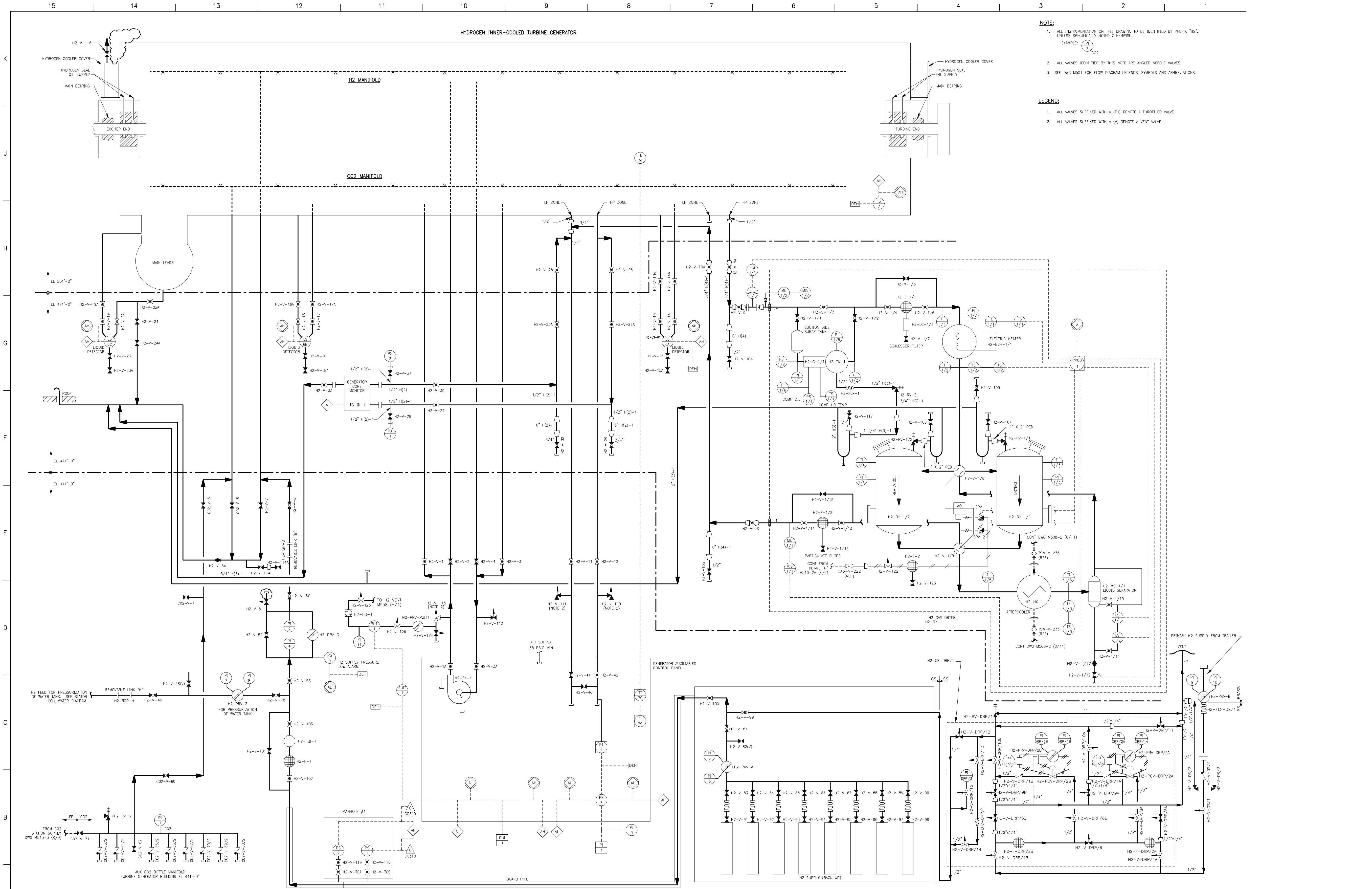
Turbine Control Valve Fast Closure Characteristic

Draw. No. 950021.41	Rev.	Figure 10.2-3

Amendment 64
December 2017



Columbia Generating Station
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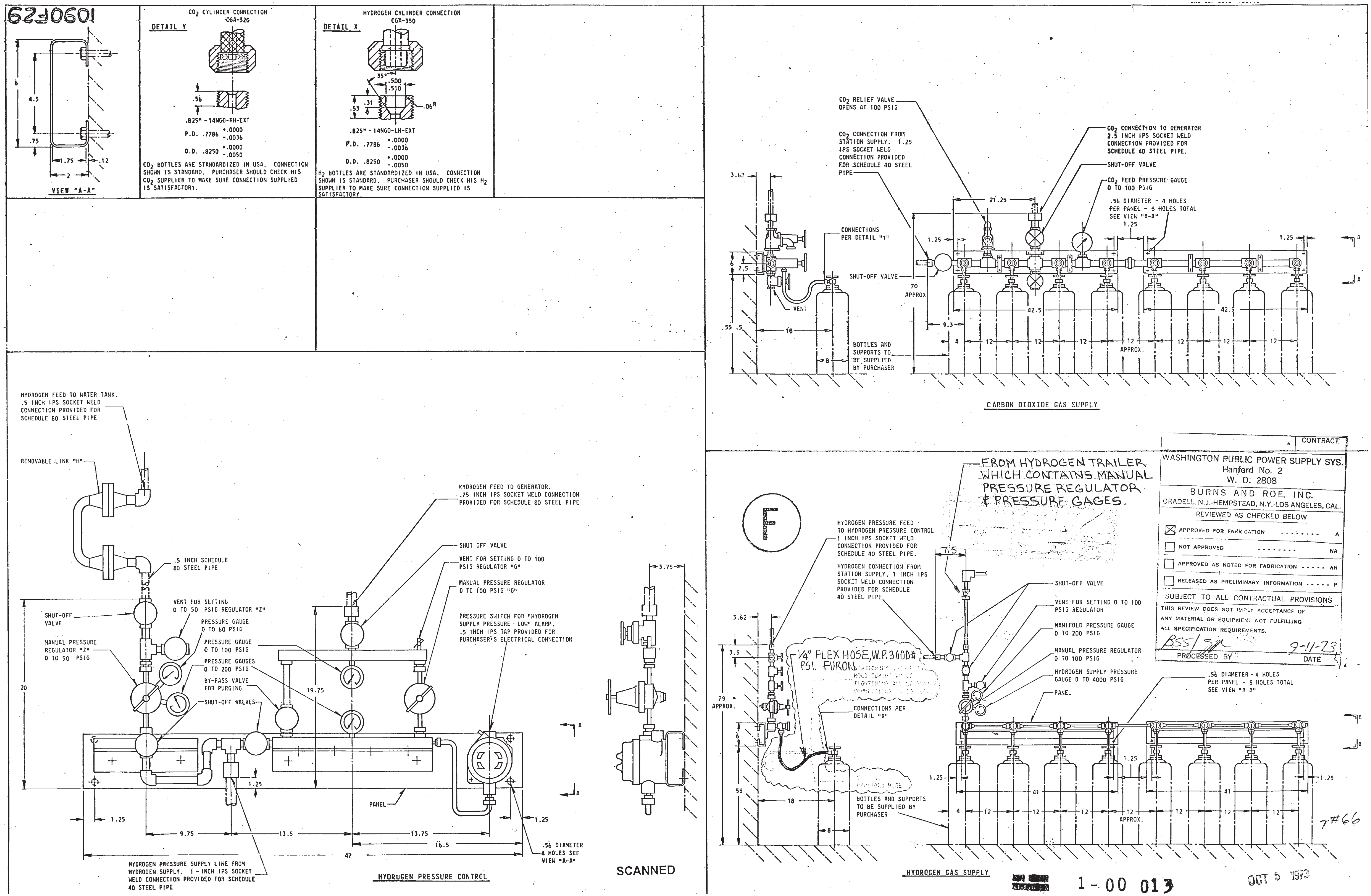
AC Turbine Generator Gas Diagram

Draw. No. M957	Rev. 29	Figure 10.2-4

Form No. 960690ai

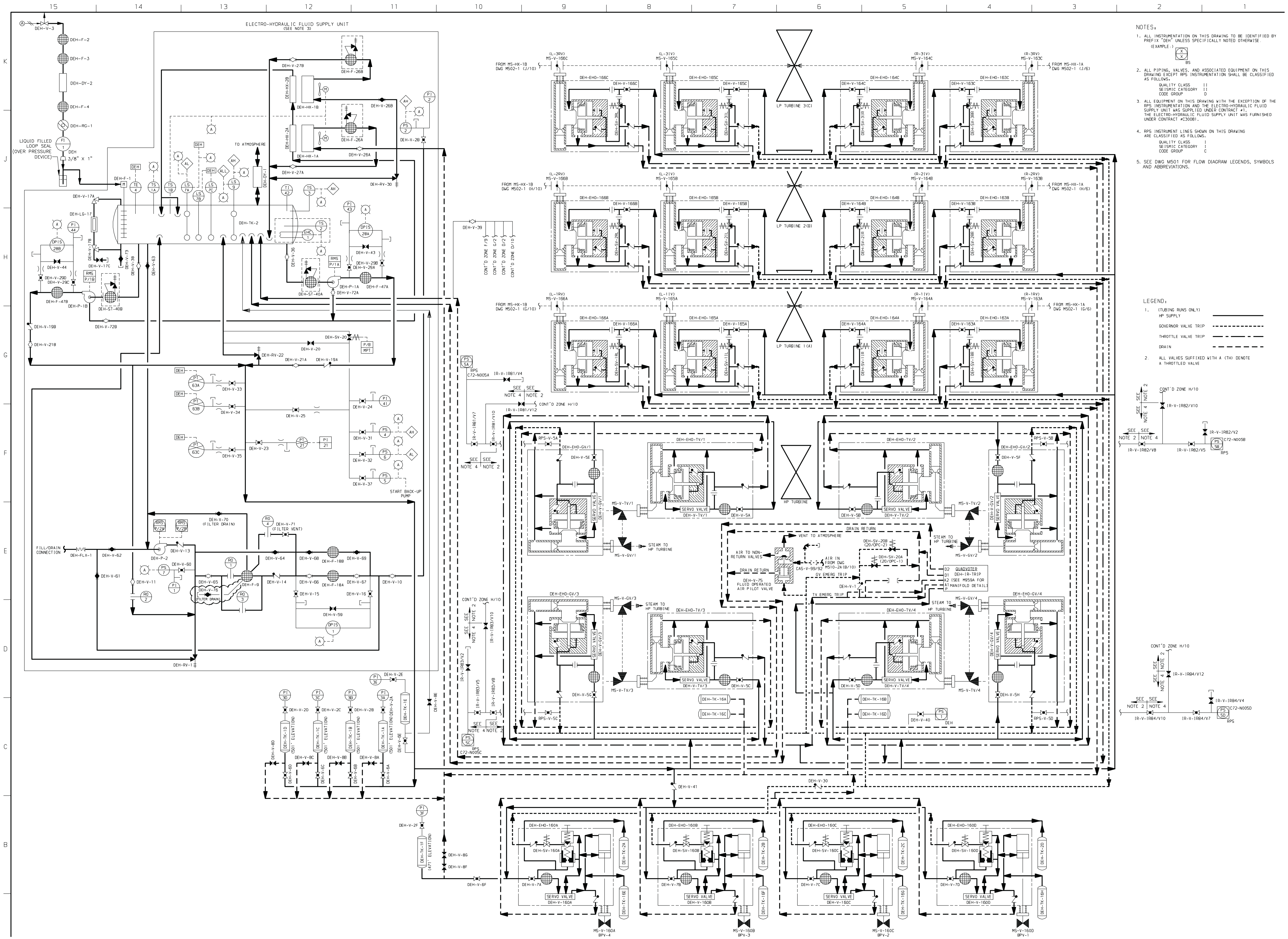


Columbia Generating Station
Final Safety Analysis Report

AC Turbine Generator Gas Supply Outline

Draw. No. 01-00,60 | Rev. 4 | Figure 10.2-5

Amendment 64
December 2017



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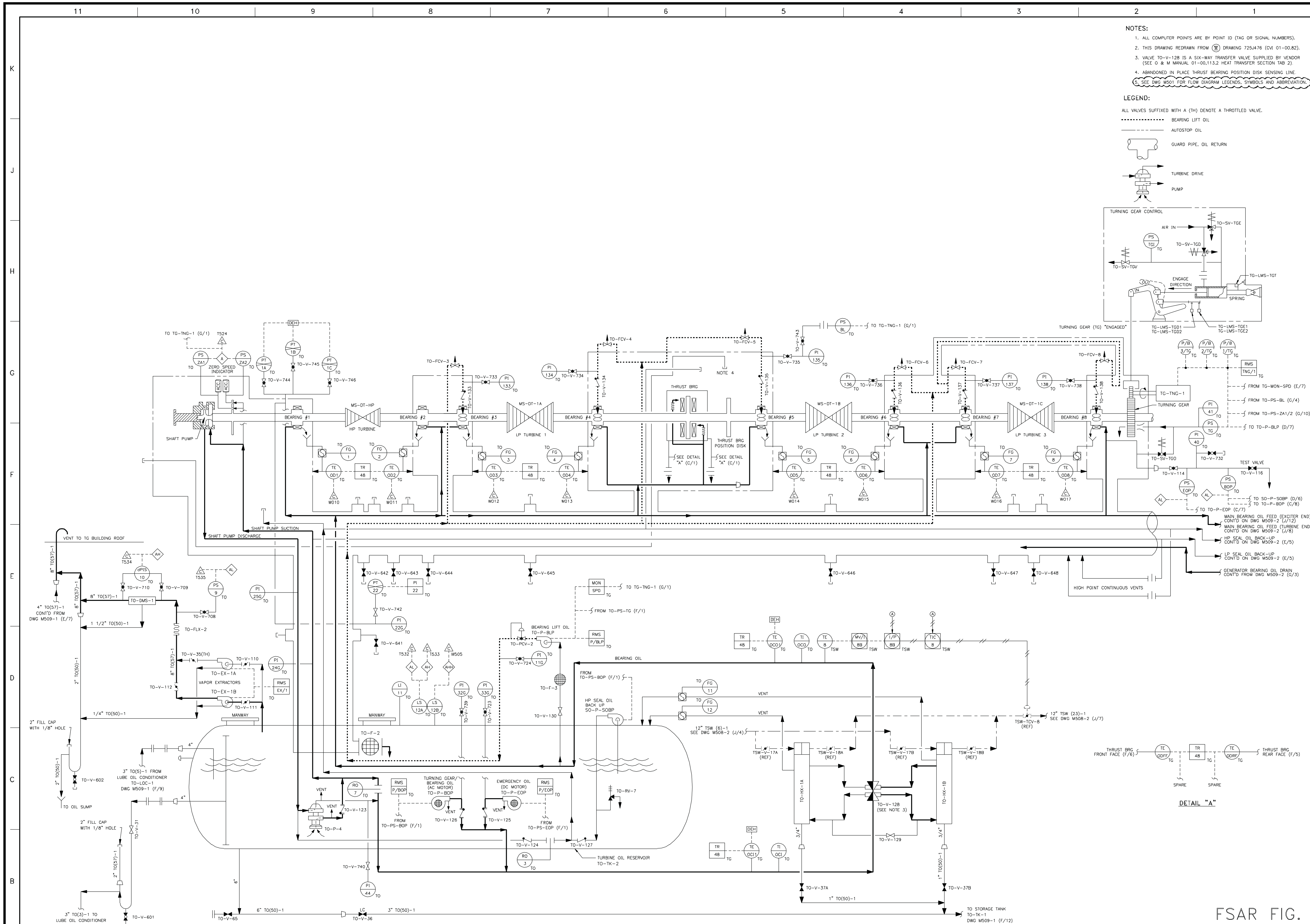
Electrohydraulic HP Fluid and
Lube Oil Diagram

Draw. No. M959 | Rev. 19 | Figure 10.2–6.1

Form No. 960690ai



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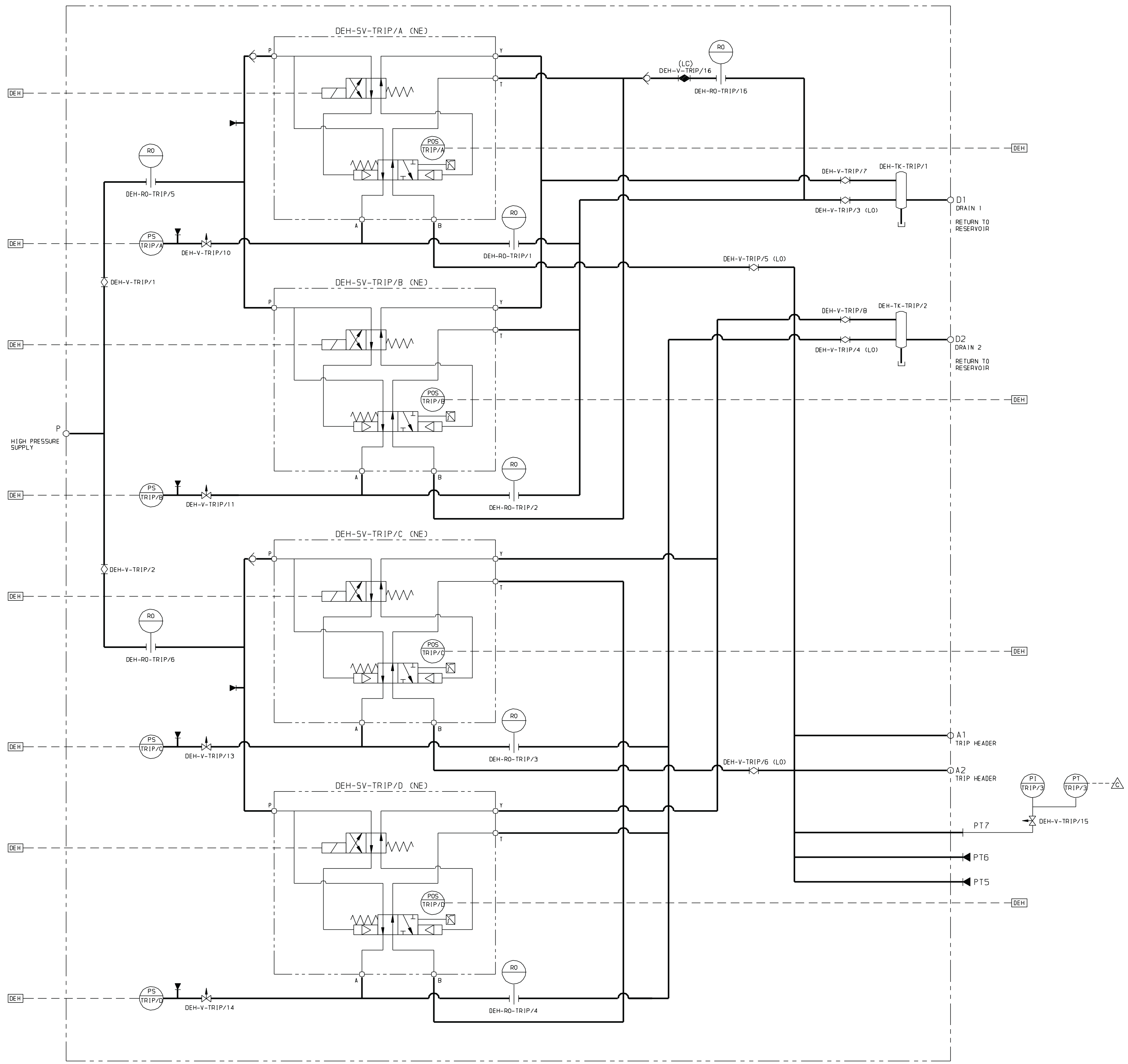
Electrohydraulic HP Fluid and Lube Oil Diagram

Draw. No. M960 | Rev. 11 | Figure 10.2-6.2

Form No. 960690ai

Amendment 63
December 2015



NOTES:

1. ALL INSTRUMENTATION ON THIS DRAWING TO BE IDENTIFIED BY PREFIX "DEH" UNLESS SPECIFICALLY NOTED OTHERWISE.
   (EXAMPLE:) BS
2. ALL PIPING, VALVES, AND ASSOCIATED EQUIPMENT ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS:

   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP D
3. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.

REFERENCE DRAWINGS

CVI 1095-00,18,1 - QUADVOTER TRIP SYSTEM OUTLINE HYDRAULIC OUTLINE

DEH-IR-TRIP
QUADVOTER MANIFOLD DETAIL
SEE M959 (ZONE E/4)

FSAR FIG

Columbia Generating Station Final Safety Analysis Report	Electrohydraulic HP Fluid and Lube Oil Diagram		
	Draw. No. M959A	Rev. 3	Figure 10.2-6.3

Form No. 960690ai

NOTES:
1. FOR SYMBOL LIST SEE DW

SYMBOL LIST

PRESSURE SWITCHES

63/BL	-BEARING LIFT OIL-OPENS AT 850 PSIG DECR. PRESS.-LOCATED IN TERM. BOX "TB"
DEH-PS-5	-EH FLUID BACKUP PUMP START-CLOSES ON DECREASING PRESSURE 1800 PSIG
63/TG-1	-TURNING GEAR-OPENS AT INCREASE OF "ENGAGE AIR" PRESS. TO 20 PSIG
63/BOP	-BEARING OIL-CLOSES ON DECREASING PRESS AT 11-12 PSIG.-LOCATED IN TERM. BOX "L"
63/EOP	-BEARING OIL-CLOSES ON DECREASING PRESS. AT 10-11 PSIG.-LOCATED IN TERM. BOX "L"
63/XO	-EXHAUST HOOD SPRAY-OPENS ON PRESS. INCREASE AT 9 PSIG (EQUAL TO 15% LOAD)
DEH-PS-2	-EH FLUID RETURN-CLOSES ON INCREASING PRESS. AT 30 PSIG
DEH-DPIS-28A	-EH FLUID PUMP #1 FILTER DIFF. PRESS. SW. CLOSES -50 PSID
DEH-DPIS-28B	-EH FLUID PUMP #2 FILTER DIFF. PRESS. SW. CLOSES -50 PSID
SW-10	-DIFF. PRESS. SW. CLOSES ON LOW AIR SIDE SEAL OIL PUMP PRESSURE
SCW-DPS-15(SW-15)	-DIFF. PRESS. SW. OPERATES WHEN DIFF. PRESS. <60 PSI ON PUMP SCW-P-1
SCW-DPS-16(SW-16)	-DIFF. PRESS. SW. OPERATES WHEN DIFF. PRESS. <60 PSI ON PUMP SCW-P-2
63/EHS-1	-EXHAUST HOOD SPRAY-CLOSES ON DECREASING PRESSURE
63/EHS-2	-EXHAUST HOOD SPRAY-CLOSES ON DECREASING PRESSURE
63/EHS-3	-EXHAUST HOOD SPRAY-CLOSES ON DECREASING PRESSURE
DEH-PS-6	-EH FLUID PRESS. LOW-OPENS ON DECREASE PRESS. 1900 PSIG
DEH-PS-4	-EH FLUID PRESS HI-CLOSES ON INCREASE PRESS. 2420 PSIG
SW-9	-DIFF. PRESS. SW. CLOSES WHEN AIR SIDE SEAL OIL PUMP IS OFF
SW-12	-PRESS SW. CLOSES WHEN SEAL OIL TURBINE BACK-UP PRESS LO.
SW-13	-DIFF. PRESS. SW. CLOSES WHEN H_2 SIDE SEAL OIL PUMP OFF
SW-14	-DIFF. PRESS. SW. CLOSES WHEN AIR SIDE SEAL OIL BACK-UP PUMP RUNNING
SW-17	-DIFF. PRESS. SW. CLOSES WHEN COOLING WATER FILTER CLOGGED
SW-18	-PRESS. SW. CLOSES WHEN MAKE-UP WATER FLOW IS ON
SW-23	-PRESS. SW. CLOSES WHEN STATOR COIL COOLING WATER TANK PRESS. IS HI
SW-24	-DIFF. PRESS. SW. CLOSES WHEN STATOR COIL COOLING WATER FLOW IS LO
SW-25	-DIFF. PRESS SW. CLOSES WHEN STATOR COIL COOLING WATER FLOW IS LO-LO
SW-30	-DIFF. PRESS. SW. CLOSES WHEN GEN. H_2/H_2O DIFF PRESS. LO
63/RPS	-RPS PRESS. SWS. OPENS ON DECREASE OF PRESS.

LIMIT SWITCHES (33)

(FOR DEVELOPMENT SEE E520-2A, EXCEPT AS NOTED)

33/TGD-1	-TURNING GEAR LEVER POSITION SWITCH WIRED TO TERMINAL BOX "K"
33/TGD-2	-TURNING GEAR LEVER POSITION SWITCH WIRED TO TERMINAL BOX "K"
33/TGE-1	-TURNING GEAR LEVER POSITION SWITCH WIRED TO TERMINAL BOX "K"
33/TGE-2	-TURNING GEAR LEVER POSITION SWITCH WIRED TO TERMINAL BOX "K"
33/TGT-1	-TURNING GEAR LEVER POSITION SWITCH WIRED TO TERMINAL BOX "K"
33/RO	-RELATCH OPERATOR-WIRED TO TERM. BOX "A"
33/VTL	-VACUUM TRIP LATCH-WIRED TO TERM. BOX "A"
33/DV	-DRAIN VALVE LIMIT SW.
33/RR	-REHEAT STOP V RIGHT LIMIT SW.
33/RL	-REHEAT STOP V LEFT LIMIT SW.
33/IR	-INTERCEPTOR V RIGHT LIMIT SW.
33/IL	-INTERCEPTOR V LEFT LIMIT SW.
33/RCV	-REHEAT CONTROL VALVE
33/EHS	-EXHAUST HOOD SPRAY
33/a & 33/b	-DISCONNECT SWITCH CONTACTS - 500kV
33/EHB-1	-SEE 33/COND-V-609A DWG. E519-5 -EXHAUST HOOD BY-PASS SPRAY VALVE LIMIT SW.
33/EHB-2	-SEE 33/COND-V-609B DWG. E519-5 -EXHAUST HOOD BY-PASS SPRAY VALVE LIMIT SW.
33/EHB-3	-SEE 33/COND-V-609C DWG. E519-5 -EXHAUST HOOD BY-PASS SPRAY VALVE LIMIT SW.
33/AR-V-3A	- W #33/VB1 -VACUUM BKR LIMIT SWITCH SEE DWG. E523 FOR SW. DEVELOPMENT
33/AR-V-3B	- W #33/VB2 -VACUUM BKR LIMIT SWITCH SEE DWG. E523 FOR SW. DEVELOPMENT
33/AR-V-3C	- W #33/VB3 -VACUUM BKR LIMIT SWITCH SEE DWG. E523 FOR SW. DEVELOPMENT
33/GS1	-ROTOR GLAND STEAM SPILL-OVER S.O. VALVE LIMIT SW. -SEE DWG. E523 FOR SW. DEVELOPMENT
33/GS2	-ROTOR GLAND STEAM SPILL-OVER B.P. VALVE LIMIT SW. -SEE DWG. E523 FOR SW. DEVELOPMENT
DEH-POS-TRIP/A,B,C,D	-TRIP SOLENOID POSITION

ELECTRICALLY OPERATED VALVES (20)

20-1/OPC	-SOL. V-OVERSPEED PROTECTION CONTROLLER WIRED TO TERM. BOX "B"
20-2/OPC	-SOL. V-OVERSPEED PROTECTION CONTROLLER WIRED TO TERM. BOX "B"
20/TGE	-SOL. V-TURNING GEAR ENGAGE -WIRED TO TERM. BOX "K"
20/TGV	-SOL. V-TURNING GEAR ENGAGE AIR VENT -WIRED TO TERM. BOX "K"
20/TGD	-SOL. V-TURNING GEAR DISENGAGE -WIRED TO TERM. BOX "K"
20/DVI	-SOL. VALVE ENERGIZED TO ADMIT AIR TO DRAIN VALVE OPERATOR
20/DVII	-SOL. VALVE ENERGIZED TO ADMIT AIR TO DRAIN VALVE OPERATOR
20/TGO	-TURNING GEAR LOW OIL
20/MPT	-SOL. V-ELECTRO HYDRAULIC PUMP NO'S 1 & 2 AUTO START TEST
20-EHS-1,2,3	-SOL. V TO OPEN EXHAUST HOOD SPRAY WHEN TURB. SPEED EXCEEDS 600RPM
20/RL	-SOL. V -REHEAT STOP V TEST, LEFT
20/RR	-SOL. V -REHEAT STOP V TEST, RIGHT
20/IL	-SOL. V -INTERCEPTOR V TEST, LEFT
20/IR	-SOL. V -INTERCEPTOR V TEST, RIGHT
20/BV1,2,3,4	-SOL. V -STEAM BY-PASS
20/RVB	-SOL. V -FOR TEST OF 63/RVB PRESS SWS.
20/RPS	-SOL. V -FOR TEST OF 63/RPS PRESS SWS.
DEH-SV-TRIP-A,B,C,D	-TRIP BLOCK

MISCELLANEOUS

63-1/ZA	-TURNING GEAR ZERO SPEED ALARM ACTUATING CONTACTS (APPROX. 10 SEC. TIME DELAY)-LOCATED IN TERM. BOX "A"
63-2/ZA	-TURNING GEAR ZERO SPEED ALARM ACTUATING CONTACTS (APPROX. 10 SEC. TIME DELAY)-LOCATED IN TERM. BOX "A"
52a & 52b	-500kV CIRCUIT BREAKER CONTACTS. (IN MICRO WAVE CAB.)
DEH-LS-7A,B,C	-LOW EH FLUID LEVEL LOCKOUT LEVEL SWITCH CLOSES ON LOW EH FLUID
TG-RLY-SPD ARM1/NC1	-TURNING GEAR ACTUATING CONTACT-CLOSED FROM ZERO TO 600RPM TURBINE SPEED-LOCATED IN TG-MON-SPD-BOARD "B" (RELAY NORMALLY DE-ENERGIZED)
TG-RLY-SPD ARM3/NC3	-CONTACT OPENS ON DECREASING TURBINE SPEED 600RPM-LOCATED IN TG-MON-SPD-BOARD "B" (RELAY NORMALLY ENERGIZED)
(N)	-LOAD REJECTION CONTACT IN DEH CONTROLLER CABINET
ARC	-AUX. RELAY CABINET
(GAP)	-ALARM CONTACTS LOCATED IN GEN. AUX. CONTROL PANEL
60/GX	-GENERATOR P.T. FAILURE RELAY-LOCATED IN BOARD "F" IN CONTROL ROOM (DWG. E512-2)
✱	-FURNISHED AND INSTALLED BY W ON TURNING GEAR CONSOLET-WIRED TO TERM. BOX "K"
⬡	-FAN FAILURE ALARM RELAYS LOCATED IN FAN CONTROL PANEL
◇	-LOCATED IN GEN. AUX. CONTROL STATION "GACS."
ø	-LOCATED IN RPS/REVAB1 TEST PANEL
(ø)	-LOCATED IN RPS/REVAB2 TEST PANEL
‡	-CLOSES WHEN E.H. CONTROL POWER SUPPLY FAILS-LOCATED IN DEH GOVERNOR CONTROLLER CABINET
41	-VOLT. REGULATOR SUPPLY BREAKER
PB/MPT	-PUSH BUTTON ELECTRO HYDRAULIC PUMP TEST
71/OL	-HIGH & LOW LUBE OIL LEVEL SW
M/EHB-1	-SEE M/COND-V-609A DWG. E519-5 -EXHAUST HOOD BY-PASS SPRAY VALVE
M/EHB-2	-SEE M/COND-V-609B DWG. E519-5 -EXHAUST HOOD BY-PASS SPRAY VALVE
M/EHB-3	-SEE M/COND-V-609C DWG. E519-5 -EXHAUST HOOD BY-PASS SPRAY VALVE
AFS	-AIR FLOW SWITCH-OPENS ON NO. ISO. PHASE BUS DUCT AIR FLOW (DWG. E519-11)
SW-7	-THERMOSTAT CLOSES ON H_2 HIGH TEMPERATURE
SW-8	-TURBINE END -CONTACT CLOSES ON DEFOAMING TANK LEVEL HI
SW-8A	-EXCITER END -CONTACT CLOSES ON DEFOAMING TANK LEVEL HI
SW-11	-CLOSES ON H_2 SIDE SEAL OIL LEVEL LO
TG-P/B-CR/TT1	-EMERGENCY TRIP PUSHBUTTON-MOUNTED IN BD "B"
TG-P/B-CR/TT2	-EMERGENCY TRIP PUSHBUTTON-MOUNTED IN BD "B"
K-7A,B,C	-REACTOR HIGH WATER LEVEL TRIP
DEH-TS-3	-TEMPERATURE SWITCH EH FLUID TEMPERATURE HI
TG-P/B-TG/TT1,TT2	-TURB. TRIP PUSHBUTTON TG 501
TG-DET-TP/A,B,C,	- THRUST BEARING
HD-LS-HL-20A,B, 21A,B,23A,B	-MSR HIGH LEVEL A,B,C
RFW-LS-624A,B,C	-REACTOR HIGH WATER LEVEL
TG-SE-OS/A,B,C	-SPEED PICKUP
TG-SE-SC/A,B,C	-SPEED PICKUP

RELAYS

86XU	-UNIT TRIP LOCKOUT RELAY (DWG. E512-2)
86XUOA	-UNIT TRIP OVERALL TRIP LOCKOUT RELAY (DWG. E512-2)
14/ZSX	-ZERO SPEED RELAY-LOCATED IN TERM BOX "K"
ACR/BOP	-EMERGENCY OIL PP-AUX. RELAY DEENERGIZED FAILURE. FOR BOP CIRCUIT POWER.
VHTX	-VOLTS/HERTZ AUX. RELAY
VHT	-VOLTS/HERTZ REGULATOR TRIP RELAY
49X-EOP	-EMERGENCY OIL PUMP OVERLOAD
49X-ASOBP	-AIR SIDE SEAL OIL BACK-UP PUMP OVERLOAD
59/81	-EXCESSIVE VOLTS/HERTZ RELAY
59/81T	-EXCESSIVE VOLTS/HERTZ TIMING RELAY
41TD	-REGULATOR SUPPLY BREAKER AUX. RELAY
41/aX1,2	-REGULATOR SUPPLY BREAKER AUX. RELAY
DCF	-DC FAILURE IN GENERATOR AUX. CONTROL PANEL
62/TGS	-TURN GEAR FAILURE T.D. ALARM RELAY 0-30 SEC.
QSV	-QUICK SOL. VALVE OPERATING RELAY
MX	-EMERG. OIL PUMP RUNNING
MX-X	-AUX. RELAY FOR EMERGENCY OIL PUMP RUNNING-ALARM
TG-X/ZSX	-AUX. RELAY FOR 14/ZSX/ZERO SPEED ALARM
86X1U	-UNIT PRIMARY LOCKOUT RELAY (DWG. E512-2)
86X1UOA	-UNIT SECONDARY LOCKOUT RELAY (DWG. E512-2)
30/64F	-FIELD GROUND DETECTION PWR. SUPPLY FAILURE RELAY
90C	-VOLTAGE REGULATOR ON RELAY
94RB	-VOLTAGE REGULATOR TRIP RELAY
K4	-VOLTAGE REGULATOR FORCING ALARM RELAY
V/HZ	-MAXIMUM VOLTZ/HERTZ RELAY
PC1, PC2	-VOLTAGE REGULATOR LOSS OF FIRING PULSE RELAY
41-A	-REGULATOR BKR. AUTO TRIP ALARM RELAY
94-A	-VOLTAGE REGULATOR TRIP ALARM RELAY
FFA	-REG. COOLING FAN FAILURE ALARM RELAY
FA	-FIELD FORCING ALARM RELAY
VH/A	-VOLTZ/HERTZ LIMIT EXCEEDED ALARM RELAY
FP/A	-REGULATOR BLOWN FUSE OR LOSS OF POWER ALARM RELAY
41TD/A	-REGULATOR BKR. CLOSED ALARM RELAY
PSF/A	-REGULATOR POWER SUPPLY FAILURE ALARM RELAY
R1	-D.C. POWER FAILURE RELAY FOR GEN. AUX. CONTROL PANEL
R3	-STATOR COIL WATER INLET CONDUCTIVITY HI RELAY
R4	-STATOR COIL WATER INLET CONDUCTIVITY HI-HI RELAY
R5	-DEMINERALIZER OUTLET CONDUCTIVITY HI RELAY
64X	-GENERATOR GROUND DETECTION ALARM RELAY
64E/X	-EXCITER GROUND DETECTION ALARM RELAY
63Y/AM1	-AUX. RELAY

TRANSMITTERS

TO-PT-1A/1B/1C	-BRG. OIL PRESS TRANSMITTERS
MS-PT-8A,B,C	-CONDENSER 1 PRESSURE TRANSMITTER
DEH-PT-63A,B,C	-EH HEADER PRESS TRANSMITTERS
DEH-PS-TRIP/A,B,C,D	-TRIP BLOCK PRESS
MS-DPT-63A,B,C	-TURB. 1ST STAGE DIFF PRESS.

) SYSTEM
ALARM RELAYS LOCATED IN ARC

EHL/1X - LEVEL HI
EHL/2X - LEVEL LO-LO
EHL/3X - LEVEL LO
EHPR/X - PRESSURE RETURN HI
EHPF/X - PRESSURE DIFF. ON FILTER
EHPH/X - PRESSURE HI
EHPL/X - PRESSURE LO
EHTH/X - TEMPERATURE HI

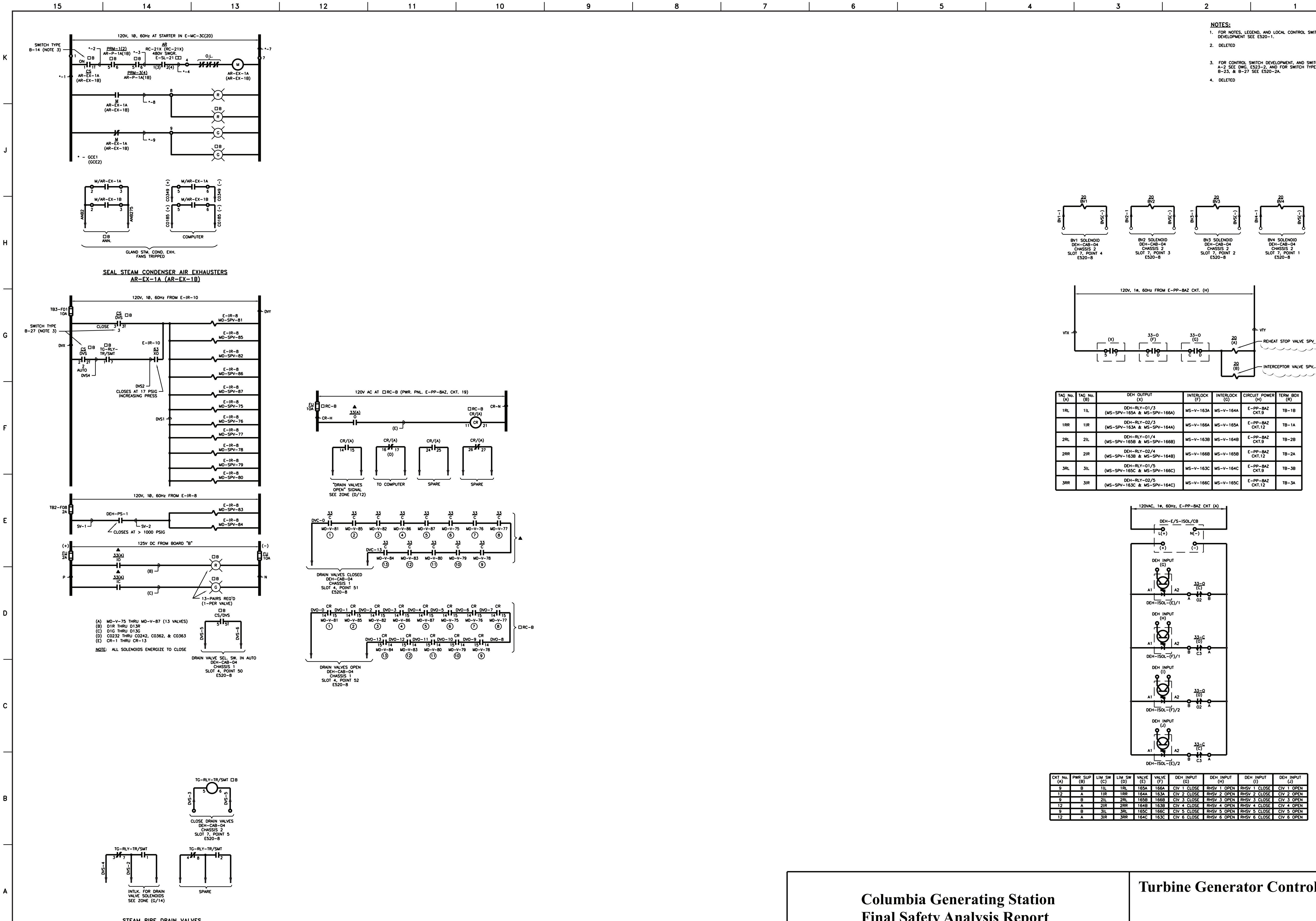
VALVE	EPN	LOCATION
RCV1	MS-TCV-115A	
RCV2	MS-TCV-115B	
RCV3	MS-TCV-115C	
RCV4	MS-TCV-115D	
BV1	MS-V-160D	
BV2	MS-V-160C	
BV3	MS-V-160B	
BV4	MS-V-160A	
1RR	MS-V-163A	NORTH
1IR	MS-V-164A	NORTH
1IL	MS-V-165A	SOUTH
1RL	MS-V-166A	SOUTH
2RR	MS-V-163B	NORTH
2IR	MS-V-164B	NORTH
2IL	MS-V-165B	SOUTH
2RL	MS-V-166B	SOUTH
3RR	MS-V-163C	NORTH
3IR	MS-V-164C	NORTH
3IL	MS-V-165C	SOUTH
3RL	MS-V-166C	SOUTH

REFERENCE DWG. LIST

TITLE	DWG.
TURNING GEAR	E520-3
TURNING GEAR SOLENOID VALVES	E520-3
ELECTRO-HYDRAULIC PUMPS	E520-3
BEARING OIL PUMP	E520-3
SEAL OIL BACK UP PUMP	E520-3
BEARING LIFT PUMP	EWD-48E-004
EMERGENCY OIL PUMP	E520-3
OIL VAPOR EXTRACTOR, RESERVOIR	E520-4
OIL VAPOR EXTRACTOR, GENERATOR	E520-4
GLAND STEAM COND AIR EXHAUSTERS	E520-4
EXHAUST HOOD SPRAY VALVES	E520-3
AUTO STOP RESET & VAC TRIP LATCH	E520-4
INTERCEPT & REHEAT STOP VALVES TEST & LIGHTS	E520-4
LOW EH FLUID LEVEL LOCK OUT RELAY	E520-5
ANTIMONITORING PROTECTION	E520-5
TURBINE TRIP CIRCUIT	E520-5
REVERSE CURRENT VALVES	E520-5
TURBINE ALARMS	E520-5
ROTOR GLAND STEAM SPILLOVER SHUT OFF VALVE	E520-4
ROTOR GLAND STEAM SPILLOVER BY PASS VALVE	E520-4
STEAM INLET VALVE TEST INDICATION	E520-4
STEAM BY PASS VALVES INDICATION	E520-4
STEAM PIPE DRAIN VALVES	E520-4
REHEATER CONTROL VALVES	E520-2
VOLTAGE REGULATOR CONTROL	E520-6
VOLTAGE REGULATOR BASE ADJUSTER	E520-6
VOLTAGE REGULATOR VOLTAGE ADJUSTER	E520-6
VOLTAGE REGULATOR ALARMS	E520-6
VOLTAGE REGULATOR SUPPLY BKR	E520-6
FIELD GROUND PROTECTION	E520-6
CONTROL SW DEVELOPMENTS	E520-2A
LIMIT SW DEVELOPMENTS	E520-2
HYDROGEN SIDE SEAL OIL PUMP	E520-7
AIR SIDE SEAL OIL PUMP	E520-7
AIR SIDE SEAL OIL BACK-UP PUMP	E520-7
GENERATOR AUX. PNL. REMOTE ALARMS	E520-7
VACUUM BREAKERS	E520-4
ANTI MOTORING LOCKOUT CIRCUIT	E520-5
QUICK SOL. VALVES	E520-5
RPS/REVAB - GOV. VALVE FAST CLOSURE EH PRESS SW. TEST	E520-2
EXCITER FIELD GROUND DETECTION	E520-6
EWD-51E-017 STATOR COIL WATER PUMP SCW-P-1	-
EWD-51E-018 STATOR COIL WATER PUMP SCW-P-2	-
CVI 218-03,7864 GENERATOR AUX. CONTROL STATION "GACS"	-
TRIP TRICON I/O CARD POINTS	E520-8
MONITORING TRICON I/O CARD POINT-9	E520-9
CONTROL TRICON #1 I/O CARD POINT-10	E520-10
CONTROL TRICON #2 I/O CARD POINT-11	E520-11

NOTES:

1. FOR NOTES, LEGEND, AND LOCAL CONTROL SWITCH DEVELOPMENT SEE E520-1.
2. DELETED
3. FOR CONTROL SWITCH DEVELOPMENT, AND SWITCH TYPE A-2 SEE DWG. E523-2, AND FOR SWITCH TYPE B-14, B-23, & B-27 SEE E520-2A.
4. DELETED



TAG No. (A)	TAG No. (B)	DEH OUTPUT (X)	INTERLOCK (F)	INTERLOCK (G)	CIRCUIT POWER (H)	TERM BOX (R)
1RL	1IL	DEH-RLY-01/3 (MS-SPV-165A & MS-SPV-166A)	MS-V-163A	MS-V-164A	E-PP-8AZ CKT.9	TB-1B
1RR	1IR	DEH-RLY-02/3 (MS-SPV-163A & MS-SPV-164A)	MS-V-166A	MS-V-165A	E-PP-8AZ CKT.12	TB-1A
2RL	2IL	DEH-RLY-01/4 (MS-SPV-165B & MS-SPV-166B)	MS-V-163B	MS-V-164B	E-PP-8AZ CKT.9	TB-2B
2RR	2IR	DEH-RLY-02/4 (MS-SPV-163B & MS-SPV-164B)	MS-V-166B	MS-V-165B	E-PP-8AZ CKT.12	TB-2A
3RL	3IL	DEH-RLY-01/5 (MS-SPV-165C & MS-SPV-166C)	MS-V-163C	MS-V-164C	E-PP-8AZ CKT.9	TB-3B
3RR	3IR	DEH-RLY-02/5 (MS-SPV-163C & MS-SPV-164C)	MS-V-166C	MS-V-165C	E-PP-8AZ CKT.12	TB-3A

CKT No. (A)	PWR SUP (B)	LIM SW (C)	LIM SW (D)	VALVE (E)	VALVE (F)	DEH INPUT (G)	DEH INPUT (H)	DEH INPUT (I)	DEH INPUT (J)
9	B	1IL	1RL	165A	166A	CIV 1 CLOSE	RHSV 1 OPEN	RHSV 1 CLOSE	CIV 1 OPEN
12	A	1IR	1RR	164A	163A	CIV 2 CLOSE	RHSV 2 OPEN	RHSV 2 CLOSE	CIV 2 OPEN
9	B	2IL	2RL	165B	166B	CIV 3 CLOSE	RHSV 3 OPEN	RHSV 3 CLOSE	CIV 3 OPEN
12	A	2IR	2RR	164B	163B	CIV 4 CLOSE	RHSV 4 OPEN	RHSV 4 CLOSE	CIV 4 OPEN
9	B	3IL	3RL	165C	166C	CIV 5 CLOSE	RHSV 5 OPEN	RHSV 5 CLOSE	CIV 5 OPEN
12	A	3IR	3RR	164C	163C	CIV 6 CLOSE	RHSV 6 OPEN	RHSV 6 CLOSE	CIV 6 OPEN

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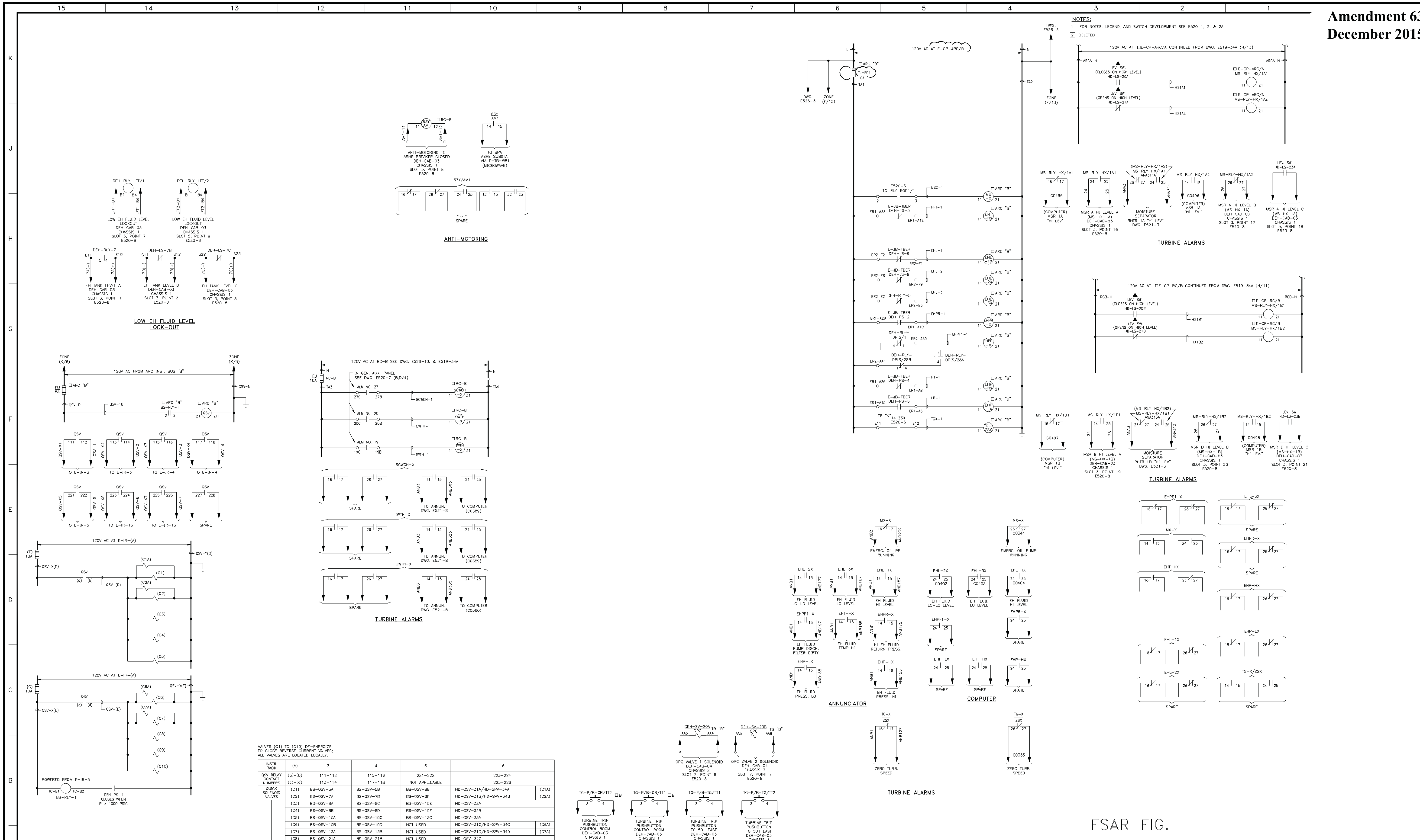
Turbine Generator Control Elementary Diagram

Draw. No. E520-4 | Rev. 32 | Figure 10.2-7.2



VALVES (C1) TO (C10) DE-ENERGIZE TO CLOSE REVERSE CURRENT VALVES; ALL VALVES ARE LOCATED LOCALLY.

INSTR. RACK	(A)	3	4	5	16	
QSV RELAY CONTACT NUMBERS	(a)-(b)	111-112	115-116	221-222	223-224	
	(c)-(d)	113-114	117-118	NOT APPLICABLE	225-226	
QUICK SOLENOID VALVES	(C1)	BS-QSV-5A	BS-QSV-5B	BS-QSV-8E	HD-QSV-31A/HD-SPV-34A	(C1A)
	(C2)	BS-QSV-7A	BS-QSV-7B	BS-QSV-8F	HD-QSV-31B/HD-SPV-34B	(C2A)
	(C3)	BS-QSV-8A	BS-QSV-8C	BS-QSV-10E	HD-QSV-32A	
	(C4)	BS-QSV-8B	BS-QSV-8D	BS-QSV-10F	HD-QSV-32B	
	(C5)	BS-QSV-10A	BS-QSV-10C	BS-QSV-13C	HD-QSV-33A	
	(C6)	BS-QSV-10B	BS-QSV-10D	NOT USED	HD-QSV-31C/HD-SPV-34C	(C6A)
	(C7)	BS-QSV-13A	BS-QSV-13B	NOT USED	HD-QSV-31D/HD-SPV-34D	(C7A)
	(C8)	BS-QSV-21A	BS-QSV-21B	NOT USED	HD-QSV-32C	
	(C9)	NOT USED	NOT USED	NOT USED	HD-QSV-32D	
	(C10)	NOT USED	NOT USED	NOT USED	HD-QSV-33B	
WIRE NUMBER SUFFIX	(D)	1	3	5	6	
	(E)	2	4	NOT USED	7	
FUSE NUMBER	(F)	TB1A-F21	TB1A-F21	TB6B-F14	TB1A-F07	
	(G)	TB1A-F22	TB1A-F22	NOT USED	TB1B-F07	

QUICK ACTING SOL. VALVES FOR REVERSE CURRENT VALVES

Columbia Generating Station
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Turbine Generator Control Elementary Diagram

Draw. No. E520-5 | Rev. 40 | Figure 10.2-7.3

10.3 MAIN STEAM SUPPLY SYSTEM

10.3.1 DESIGN BASES

The main steam supply system is designed for the following conditions:

a. Deliver steam from the reactor to the turbine generator from warmup to 105% of rated load,

b. Provide steam for the second-stage reheaters and steam-jet air ejectors,

c. Bypass steam to the main condenser during startup and in the event steam requirements of the turbine generator are less than that produced by the reactor,

d. Provide steam to the gland seal steam evaporator during startup, low load operation, and shutdown,

e. Provide steam to drive reactor feedwater pumps during startup and low load operation, and

f. Provide steam to the offgas preheaters.

The design pressure and temperature of the main steam piping is 1250 psig and 575°F.

The main steam lines are designed to include accesses to permit inservice inspection and testing (refer to Sections 5.2.4 and 6.6).

Design codes are given in Table 3.2-1, item 2, Nuclear Boiler System, and item 43, Power Conversion System. The environmental design bases for the main steam supply system are contained in Section 3.11.

10.3.2 SYSTEM DESCRIPTION

The main steam supply system is shown in Figures 10.3-1 and piping drawings are shown in Figures 3.6-32, 3.6-33, 3.6-34, 3.6-35, 3.6-50, 3.6-51, 3.6-53, 3.6-58, 3.6-60, and 10.3-2. The main steam line piping consists of four 30-in. (26-in. in reactor building) lines extending from the reactor pressure vessel to the main steam header located upstream of the turbine stop and control valves. This header placement ensures a positive means of bypassing steam via the turbine bypass system during transient conditions and startup. Branch lines from the main steam line provide the steam requirements for the reactor feed pumps, second stage reheaters, gland seal steam evaporator, offgas preheaters, and steam jet air ejectors.

The MSIVs are a wye-pattern-type globe valve utilizing pneumatic air to open and spring load with pneumatic air assist to close. Energizing control valves provide pilot and actuator air to open the valve. Deenergizing the control valves removes pilot air which vents actuator opening air and directs air to assist the spring force in closing the valve.

Loss of compressed air failure mode results are

a. Loss of compressed air due to loss of nonseismic air lines* results in loss of pilot air and closure of the MSIV by both spring force and pneumatic air cylinder force.

b. Loss of compressed air due to loss of Seismic Category I air lines† results in loss of both pilot air and actuator air with the MSIVs closing by spring force only.

Under normal operation the air supply maintains the required air for holding the valve open and charging the air storage tank. The check valve at the air storage tank inlet ensures a pneumatic supply for assist in closing the valve. No safety-related makeup supply is required for closure of the MSIVs for safe plant shutdown.

The removal of electrical power or failure of both the solenoids on the control valve automatically initiates closure of the MSIVs. Safety-related components ensuring removal of power to the solenoids when required are the only electrical power requirement. Section 9.3.1 describes the compressed air systems.

Equalizing lines connecting steam lines outside of the containment are used to equalize pressure across the main steam line isolation valves prior to restart following a steam line isolation. Assuming all steam line isolation valves have closed, the outer containment isolation valves are opened first and the drain lines are used to warm up and pressurize the outside steam lines. Following warmup the inboard main steam line isolation valves are opened.

10.3.3 SAFETY EVALUATION

Table 3.2-1 lists the applicable seismic category, quality group classification, and safety class for the main steam supply system. The effects of main steam line breaks and other accident conditions outside the containment are evaluated in Chapter 15. Protection against dynamic effects associated with the postulated rupture of piping inside or outside of containment is discussed in Section 3.6.

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* Nonseismic lines are those lines supplying air to the isolation valve upstream of the check valve and to the pilot side of the air pilot valves (Figure 9.3-1, detail B).

† Seismic Category I lines include an air storage tank, check valve, and lines from the check valve to the actuator pilot valve (Figure 9.3-1, detail B).

10.3.4 INSPECTION AND TESTING REQUIREMENTS

The main steam lines were hydrostatically tested prior to initial operation. Nondestructive testing is performed in accordance with the applicable code requirements.

Preoperational and inservice inspection of the main steam lines and the main steam line isolation valves are presented in Sections 5.2.4 and 6.6.

The use of four main steam lines permits inspection and testing of the turbine stop, control, reheat stop, and intercept valves and main steam line isolation valves during plant operation with a minimum of load reduction.

10.3.5 WATER CHEMISTRY

This section is not applicable to a BWR. See Section 10.4.6 for reactor coolant water chemistry considerations.

10.3.6 STEAM AND FEEDWATER SYSTEM MATERIALS

10.3.6.1 Fracture Toughness

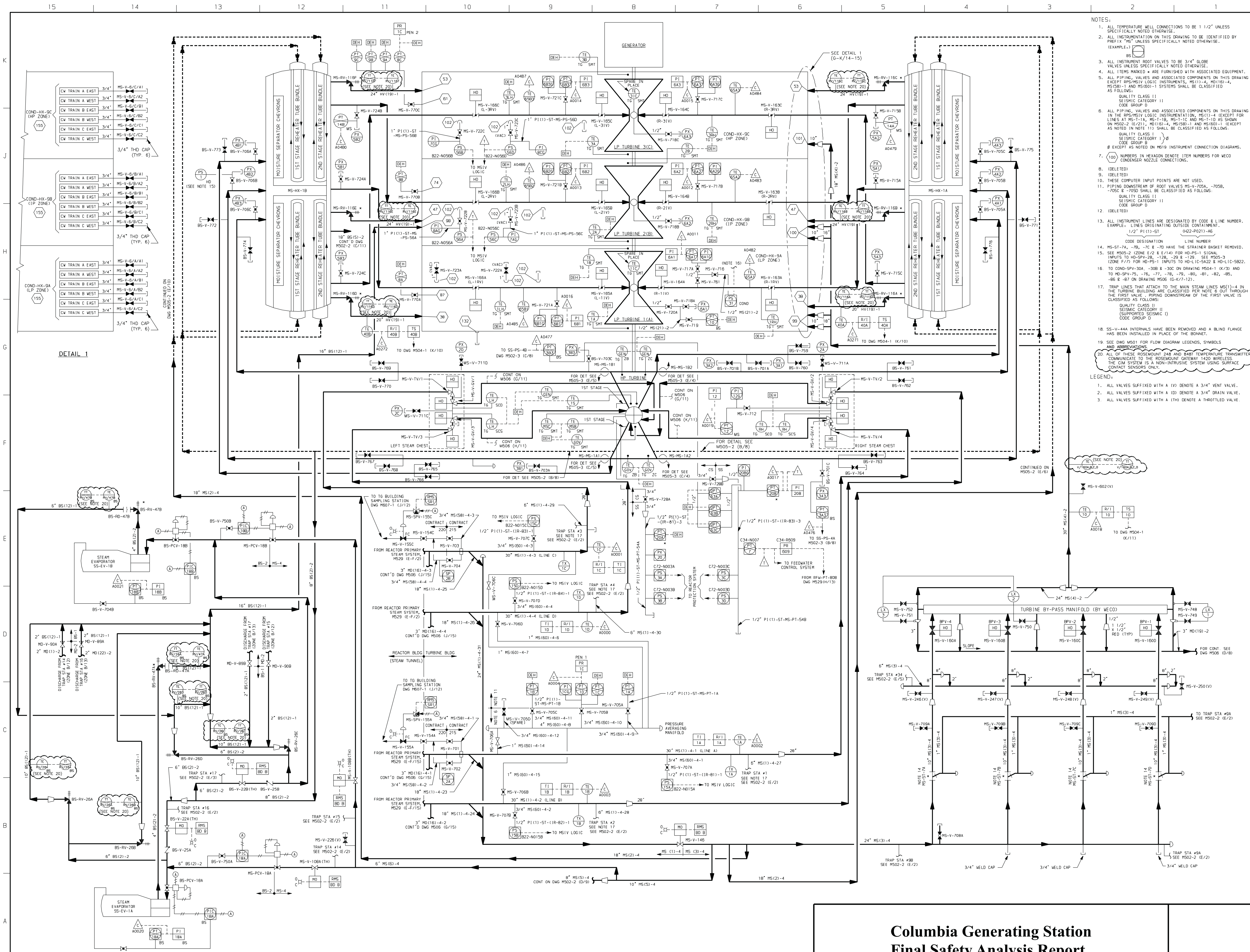
Impact tests in accordance with the size limitations specified in ASME Code Section III, Class 1, are performed on all ASME Code Section III, Class 1, main steam and feedwater materials, as well as Class 2 main steam system materials for all pressure retaining ferritic steel parts. The tests are conducted at a temperature of 45°F or lower in accordance with NB or NC-2310 of the Summer 1972 or Winter 1973 Addendum of ASME Code Section III, as applicable.

10.3.6.2 Materials Selection and Fabrication

All materials used for portions of the main steam system described in this section are included in Appendix I to Section III of the ASME Boiler and Pressure Vessel (B&PV) Code. The requirements for welding the main steam piping from the reactor to the turbine generator are in accordance with ASME Section III, 1971 Edition through the Winter 1973 Addenda. The welding requirements for other steam and feedwater piping are in accordance with ANSI B31.1, October 1973 (see Section 3.2).

Cleaning of components in the main steam system is in accordance with ANSI N45.2.1 (October 1973) or ASTM A380-57 (October 1973) for stainless steel surfaces and Regulatory Guide 1.37.

Degree of conformance to the following Regulatory Guides is addressed in Section 1.8: 1.31, Control of Stainless Steel Welding; 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel; 1.44, Control of the Use of Sensitized Stainless Steel; 1.50, Control of Preheat Temperature for Welding of Low-Alloy Steel; and 1.71, Welder Qualification for Areas of Limited Accessibility.



NOTES:

1. ALL TEMPERATURE WELL CONNECTIONS TO BE 1 1/2" UNLESS SPECIFICALLY NOTED OTHERWISE.
2. ALL INSTRUMENTATION ON THIS DRAWING TO BE IDENTIFIED BY PREFIX "MS" UNLESS SPECIFICALLY NOTED OTHERWISE.
   (EXAMPLE:) BS
3. ALL INSTRUMENT ROOT VALVES TO BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
4. ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
5. ALL PIPING, VALVES AND ASSOCIATED COMPONENTS ON THIS DRAWING EXCEPT RPS/MSIV LOGIC INSTRUMENTS, MS(1)-4, MD(16)-4, MS(58)-1 AND MS(60)-1 SYSTEMS SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP D
6. ALL PIPING, VALVES AND ASSOCIATED COMPONENTS ON THIS DRAWING IN THE RPS/MSIV LOGIC INSTRUMENTATION, MS(1)-4 (EXCEPT FOR LINES AT MS-T-1A, MS-T-1B, MS-T-1C AND MS-T-1D AS SHOWN ON M502-2 (E/2)), MD(16)-4, MS(58)-1 AND MS(60)-1 (EXCEPT AS NOTED IN NOTE 11) SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS I
   SEISMIC CATEGORY I
   CODE GROUP B
   } Ø
   Ø EXCEPT AS NOTED ON M619 INSTRUMENT CONNECTION DIAGRAMS.
7. (100) NUMBERS IN HEXAGON DENOTE ITEM NUMBERS FOR WECO CONDENSER NOZZLE CONNECTIONS.
8. (DELETED)
9. (DELETED)
10. THESE COMPUTER INPUT POINTS ARE NOT USED.
11. PIPING DOWNSTREAM OF ROOT VALVES MS-V-705A, -705B, -705C & -705D SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP D
12. (DELETED)
13. ALL INSTRUMENT LINES ARE DESIGNATED BY CODE & LINE NUMBER. EXAMPLE: LINES ORIGINATING OUTSIDE CONTAINMENT.
   1/2" PI(1)-ST (CODE DESIGNATION) (H22-P021)-A6 (LINE NUMBER)
14. MS-ST-7A, -7B, -7C & -7D HAVE THE STRAINER BASKET REMOVED.
15. SEE M505-2 (ZONE E/2 & E/14) FOR HD-PS-1 SIGNAL INPUTS TO HD-SPV-28, -128, -29 & -129. SEE M505-3 (ZONE F/7) FOR HD-PS-1 INPUTS TO HD-LIC-5A22 & HD-LIC-5B22.
16. TO COND-SPV-30A, -30B & -30C ON DRAWING M504-1 (K/3) AND TO MD-SPV-75, -76, -77, -78, -79, -80, -81, -82, -85, -86 & -87 ON DRAWING M506 (G-K/7-12).
17. TRAP LINES THAT ATTACH TO THE MAIN STEAM LINES MS(1)-4 IN THE TURBINE BUILDING ARE CLASSIFIED PER NOTE 6 OUT THROUGH THE FIRST VALVE . PIPING DOWNSTREAM OF THE FIRST VALVE IS CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATEGORY II
   (SUPPORTED SEISMIC I)
   CODE GROUP D
18. SS-V-44A INTERNALS HAVE BEEN REMOVED AND A BLIND FLANGE HAS BEEN INSTALLED IN PLACE OF THE BONNET.
19. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.
20. ALL OF THESE ROSEMOUNT 248 AND 848T TEMPERATURE TRANSMITTERS COMMUNICATE TO THE ROSEMOUNT GATEWAY 1420 WIRELESS. THE CIM SYSTEM IS A NON-INTRUSIVE SYSTEM USING SURFACE CONTACT SENSORS ONLY.

LEGEND:

1. ALL VALVES SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

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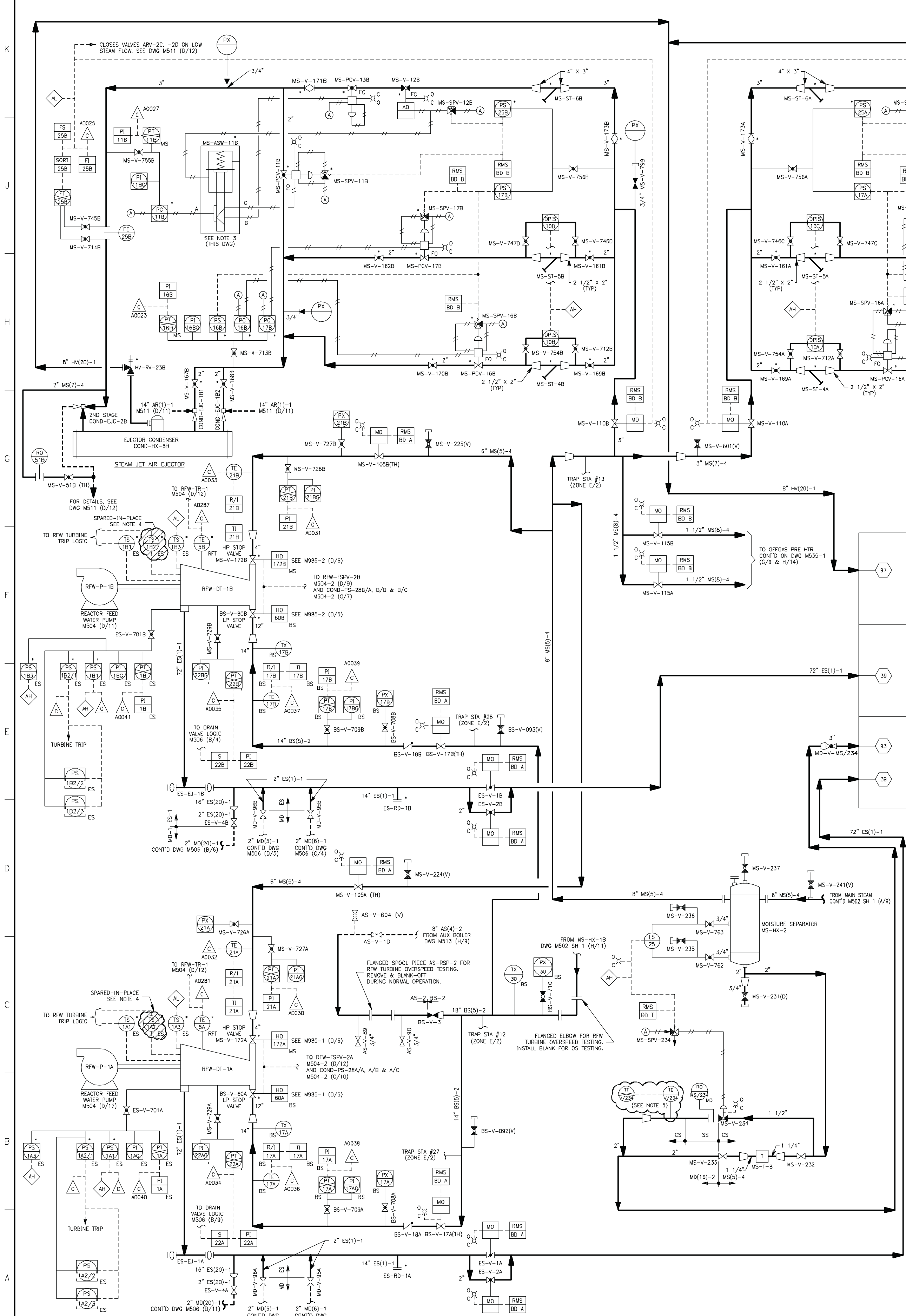
Main Steam Supply System

Draw. No. M502-1 | Rev. 43 | Figure 10.3-1.1

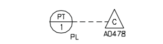


NOTES: (CONT'D)

4. ES-TS-1A2 AND ES-TS-1B2 ARE SPARED-IN-PLACE BUT HAVE BEEN DESIGNATED AS SPARES FOR ES-TS-1A1 AND ES-TS-1B1 RESPECTIVELY, AND AS SUCH ARE NOT TO BE REMOVED FOR USE IN OTHER AREAS OF THE PLANT.
5. ALL OF THESE ROSEMOUNT 248 AND 848T TEMPERATURE TRANSMITTERS COMMUNICATE TO THE ROSEMOUNT GATEWAY 1420 WIRELESS. THE CIM SYSTEM IS A NON-INTRUSIVE SYSTEM USING SURFACE CONTACT SENSORS ONLY.

NOTE:

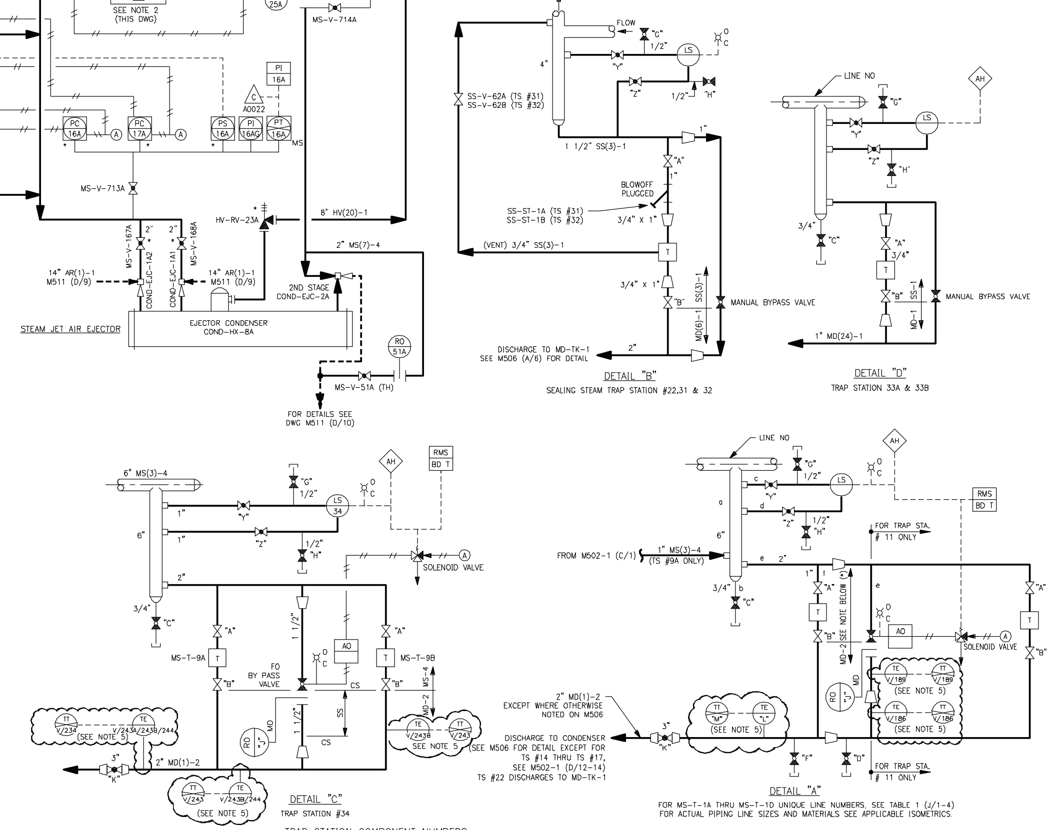
1. FOR NOTES AND LEGEND, SEE M502 SH 1.
2. a) WHEN MS-SPV-12A IS ENERGIZED, MS-ASW-11A PORT A TO B IS CONNECTED.
   b) WHEN MS-SPV-12A IS DE-ENERGIZED, MS-ASW-11A PORT A TO C IS CONNECTED.
3. a) WHEN MS-SPV-12B IS ENERGIZED, MS-ASW-11B PORT A TO B IS CONNECTED.
   b) WHEN MS-SPV-12B IS DE-ENERGIZED, MS-ASW-11B PORT A TO C IS CONNECTED.



PLANT BAROMETRIC PRESSURE

TABLE I

TRAP STATION NO	UNIQUE LINE NUMBER (SEE DETAIL A)						
	a	b	c	d	e		f
MS-T-1A	6" MS(1)-4-27	3/4" MS(55)-4-10	1" MS(60)-4-19	1" MS(60)-4-23	2" MS(50)-4-12	1" MS(50)-4-12	1" MS(50)-4-16
MS-T-1B	6" MS(1)-4-28	2" MS(55)-4-11	1" MS(60)-4-20	1" MS(60)-4-24	2" MS(50)-4-13	1" MS(50)-4-13	1" MS(50)-4-17
MS-T-1C	6" MS(1)-4-29	3/4" MS(55)-4-12	1" MS(60)-4-21	1" MS(60)-4-25	2" MS(50)-4-14	1" MS(50)-4-14	1" MS(50)-4-18
MS-T-1D	6" MS(1)-4-30	3/4" MS(55)-4-13	1" MS(60)-4-22	1" MS(60)-4-26	2" MS(50)-4-15	1" MS(50)-4-15	1" MS(50)-4-19



DETAIL "A"

FOR MS-T-1A THRU MS-T-1D UNIQUE LINE NUMBERS, SEE TABLE 1 (J/1-4)
FOR ACTUAL PIPING LINE SIZES AND MATERIALS SEE APPLICABLE ISOMETRICS.

TRAP STATION COMPONENT NUMBERS

TRAP STA NO	TRAP LOCATION ON DWG	DISCHARGE LOCATION ON DWG	TRAP NO	DETAIL	LEVEL SWITCH NO	SOLENOID VALVE NO	BY PASS VALVE NO	LINE NO *	ROOT VALVE NO "Y"	ROOT VALVE NO "Z"	"A" VALVE NO	"B" VALVE NO	"C" VALVE NO	"D" VALVE NO	"F" VALVE NO	"G" VALVE NO	"H" VALVE NO	"J" RESTRICTING ORIFICE NO	"K" VALVE NO	"L" TE TEMP ELEMENT	"M" TT TEMP TRANSMITTER
1	M502-1 (C/7)	M506 (F/9)	MS-T-1A	A	MS-LS-24A	MS-SPV-120A	MS-V-120A	30" MS(1)-4	MS-V-730A	MS-V-731A	MS-V-117A	MS-V-118A		MD-V-621(D)	——	MS-V-174A	MS-V-174B	——	——	V/120A	V/120A
2	M502-1 (B/8)	M506 (F/9)	MS-T-1B	A	MS-LS-24B	MS-SPV-120B	MS-V-120B	30" MS(1)-4	MS-V-730B	MS-V-731B	MS-V-117B	MS-V-118B		MD-V-627(D)	——	MS-V-175A	MS-V-175B	——	——	V/120B	V/120B
3	M502-1 (E/9)	M506 (F/10)	MS-T-1C	A	MS-LS-24C	MS-SPV-120C	MS-V-120C	30" MS(1)-4	MS-V-730C	MS-V-731C	MS-V-117C	MS-V-118C		MD-V-642(D)	——	MS-V-176A	MS-V-176B	——	——	V/120C	V/120C
4	M502-1 (D/9)	M506 (F/10)	MS-T-1D	A	MS-LS-24D	MS-SPV-120D	MS-V-120D	30" MS(1)-4	MS-V-730D	MS-V-731D	MS-V-117D	MS-V-118D		MD-V-643(D)	——	MS-V-177A	MS-V-177B	——	——	V/120D	V/120D
5	M505-2 (B/12)	M506 (F/7)	BS-T-1	A	BS-LS-30	BS-SPV-48	BS-V-48	12" BS(1)-2	BS-V-711	BS-V-712	BS-V-49	BS-V-50	BS-V-51	——	——	BS-V-72A	BS-V-72B	——	——		
6	M505-2 (E/10)	M506 (E/7)	MS-T-2	A	MS-LS-24	MS-SPV-121	MS-V-121	12" MS(2)-4	MS-V-732	MS-V-733	MS-V-122	MS-V-123	MS-V-124	——	——	MS-V-178A	MS-V-178B	——	——		
7	M505-2 (K/6)	M506 (E/7)	MS-T-3A	A	MS-LS-24G	MS-SPV-125A	MS-V-125A	12" MS(2)-4	MS-V-734A	MS-V-735A	MS-V-126A	MS-V-127A	MS-V-128A	——	——	MS-V-179A	MS-V-179B	——	——		
8	M505-2 (K/10)	M506 (E/4)	MS-T-3B	A	MS-LS-24H	MS-SPV-125B	MS-V-125B	12" MS(2)-4	MS-V-734B	MS-V-735B	MS-V-126B	MS-V-127B	MS-V-128B	——	——	MS-V-180A	MS-V-180B	——	——		
9A	M502-1 (B/2)	M506 (E/11)	MS-T-4A	A	MS-LS-24J	MS-SPV-186	MS-V-186	24" MS(3)-4	MS-V-736	MS-V-737	MS-V-187	MS-V-188	MS-V-132	——	——	MS-V-181A	MS-V-181B	MD-RO-MS/186	MD-V-MS/186		
9B	M502-1 (B/7)	M506 (E/11)	MS-T-4B	A	MS-LS-24M	MS-SPV-189	MS-V-189	24" MS(3)-4	MS-V-759	MS-V-760	MS-V-190	MS-V-191	MS-V-192	——	——	MS-V-181C	MS-V-181D	MD-RO-MS/189	MD-V-MS/189		
10	M505-2 (B/8)	M506 (F/11)	BS-T-3	A	BS-LS-29A	BS-SPV-27	BS-V-27	16" BS(1)-2	BS-V-713	BS-V-714	BS-V-28	BS-V-29	BS-V-30	MD-V-641(D)	——	BS-V-73A	BS-V-73B	——	——		
11	M505-2 (B/8)	M506 (H/12)	BS-T-4 BS-T-4A	A	BS-LS-29B	BS-SPV-31	BS-V-31	16" BS(1)-2	BS-V-715	BS-V-716	BS-V-32 BS-V-32A	BS-V-33 BS-V-33A	BS-V-34	—— ——	MD-V-631(D)	BS-V-74A	BS-V-74B	——	——		
12	M502-2 (C/11)	M506 (E/5)	BS-T-5	A	BS-LS-29C	BS-SPV-35A	BS-V-35A	18" BS(5)-2	BS-V-717A	BS-V-718A	BS-V-36A	BS-V-37A	BS-V-38A	——	MD-V-097(D)	BS-V-75A	BS-V-75B	——	——		
13	M502-2 (G/11)	M506 (E/5)	MS-T-5	A	MS-LS-24L	MS-SPV-133B	MS-V-133B	4" MS(7)-4	MS-V-738B	MS-V-739B	MS-V-134B	MS-V-135B	MS-V-136B	——	——	MS-V-182A	MS-V-182B	——	——		
14	M502-1 (B/11)	M502-1 (D/14)	MS-T-6A	A	MS-LS-137A	MS-SPV-137A	MS-V-137A	6" MS(6)-4	MS-V-740A	MS-V-741A	MS-V-138A	MS-V-139A	MS-V-140A	MD-V-632(D)	——	MS-V-183A	MS-V-183B	——	——		
15	M502-1 (A/11)	M502-1 (D/13)	MS-T-6B	A	MS-LS-137B	MS-SPV-137B	MS-V-137B	6" MS(6)-4	MS-V-740B	MS-V-741B	MS-V-138B	MS-V-139B	MS-V-140B	MD-V-634(D)	——	MS-V-184A	MS-V-184B	——	——		
16	M502-1 (D/12)	M502-1 (D/13)	BS-T-6A	A	BS-LS-39A	BS-SPV-39A	BS-V-39A	6" BS(2)-2	BS-V-719A	BS-V-720A	BS-V-40A	BS-V-41A	BS-V-42A	MD-V-625(D)	——	BS-V-76A	BS-V-76B	——	——		
17	M502-1 (B/12)	M502-1 (D/13)	BS-T-6B	A	BS-LS-39B	BS-SPV-39B	BS-V-39B	6" BS(2)-2	BS-V-719B	BS-V-720B	BS-V-40B	BS-V-41B	BS-V-42B	MD-V-629(D)	——	BS-V-77A	BS-V-77B	——	——		
18	M505-2 (K/12)	M506 (F/4)	BS-T-7A	A	BS-LS-52A	BS-SPV-52A	BS-V-52A	12" BS(1)-2	BS-V-721A	BS-V-722A	BS-V-53A	BS-V-54A	BS-V-55A	——	——	BS-V-78A	BS-V-78B	——	——		
19	M505-2 (K/14)	M506 (E/5)	BS-T-7B	A	BS-LS-52B	BS-SPV-52B	BS-V-52B	12" BS(1)-2	BS-V-721B	BS-V-722B	BS-V-53B	BS-V-54B	BS-V-55B	——	——	BS-V-79A	BS-V-79B	——	——		
20	M502-3 (D/12)	M506 (E/5)	SS-T-1A	A	SS-LS-12A	SS-SPV-12A	SS-V-12A	3" SS(2)-2	SS-V-709A	SS-V-709B	SS-V-13A	SS-V-14A	SS-V-15A	——	MD-V-099(D)	SS-V-55A	SS-V-55B	——	——		
21	M502-3 (G/12)	M506 (E/5)	SS-T-1B	A	SS-LS-12B	SS-SPV-12B	SS-V-12B	3" SS(2)-2	SS-V-710A	SS-V-710B	SS-V-13B	SS-V-14B	SS-V-15B	——	MD-V-100(D)	SS-V-56A	SS-V-56B	——	——		
22	M502-3 (K/14)	M506 (A/6)	SS-T-2	A	SS-LS-16	SS-SPV-16	SS-V-16	6" SS(3)-1	SS-V-707	SS-V-708	SS-V-17	SS-V-18	SS-V-19	——	——	SS-V-57A	SS-V-57B	——	——		
23	M502-3 (B/9)	M506 (G/10)	SS-T-3	A	SS-LS-20	SS-SPV-20	SS-V-20	6" SS(1)-2	SS-V-711	SS-V-712	SS-V-21	SS-V-22	SS-V-35	——	——	SS-V-58A	SS-V-58B	——	——		
24	M505-2 (E/11)	M506 (F/10)	MS-T-7	A	MS-LS-147	MS-SPV-147	MS-V-147	18" MS(2)-4	MS-V-742	MS-V-743	MS-V-148	MS-V-149	MS-V-150	MD-V-638(D)	——	MS-V-185A	MS-V-185B	——	——	V/147/149	V/147/149
25	M505-2 (B/8)	M506 (F/15)	BS-T-8	A	BS-LS-56	BS-SPV-56	BS-V-56	16" BS(1)-2	BS-V-723	BS-V-724	BS-V-57	BS-V-58	BS-V-59	MD-V-088(D)	——	BS-V-80A	BS-V-80B	——	——		
26	M502-3 (C/3)	M506 (H/13)	SS-T-4	A	SS-LS-1	SS-SPV-30	SS-V-30	8" SS(10)-1	SS-V-705	SS-V-706	SS-V-28	SS-V-29	SS-V-36	——	——	SS-V-59A	SS-V-59B	——	——		
27	M502-2 (B/12)	M506 (E/5)	BS-T-9	A	BS-LS-64	BS-SPV-64	BS-V-64	14" BS(5)-2	BS-V-755	BS-V-756	BS-V-65	BS-V-66	BS-V-67	——	MD-V-093(D)	BS-V-81A	BS-V-81B	——	——		
28	M502-2 (E/12)	M506 (E/5)	BS-T-10	A	BS-LS-68	BS-SPV-68	BS-V-68	14" BS(5)-2	BS-V-757	BS-V-758	BS-V-69	BS-V-70	BS-V-71	——	MD-V-094(D)	BS-V-82A	BS-V-82B	——	——		
29	DELETED																				
30	DELETED																				
31	M502-3 (G/15)	M506 (A/6)	SS-T-5A	B	SS-LS-24A	——	SS-V-51A	4" SS(3)-1	SS-V-715A	SS-V-716A	SS-V-52A	SS-V-53A	——	——	——	SS-V-60A	SS-V-60B	——	——		
32	M502-3 (K/15)	M506 (A/6)	SS-T-5B	B	SS-LS-24B	——	SS-V-51B	4" SS(3)-1	SS-V-715B	SS-V-716B	SS-V-52B	SS-V-53B	——	——	——	SS-V-61A	SS-V-61B	——	——		
33A	M502-3 (B/1)	M506 (F/10)	SS-T-7A	D	SS-LS-33A	——	SS-V-76	1 1/2"SS(9)-1	SS-V-80	SS-V-81	SS-V-77	SS-V-78	SS-V-79	——	——	SS-V-74	SS-V-75	——	——		
33B	M502-3 (B/1)	M506 (F/10)	SS-T-7B	D	SS-LS-33B	——	SS-V-68	2 1/2"SS(8)-1	SS-V-72	SS-V-73	SS-V-69	SS-V-70	SS-V-71	——	——	SS-V-66	SS-V-67	——	——		
34	M502-1 (C/5)	M506 (E/12)	MS-T-9A MS-T-9B	C	MS-LS-34	MS-SPV-34	MS-V-244	6" MS(3)-4	MS-V-251	MS-V-252	MS-V-242A MS-V-242B	MS-V-243A MS-V-243B	MS-V-245	—— ——	—— ——	MS-V-254	MS-V-253	MD-RO-MS/244	MD-V-MS/244		

* ALL PIPING UPSTREAM OF THE BY PASS VALVE & VALVE "B" HAVE THE SAME DESIGNATION & PRESSURE RATING OF THE SOURCE LINE.
FOR LINE SIZES SMALLER THAN 6" DRIP LEG SHALL HAVE SAME DIAMETER AS MAIN LINE

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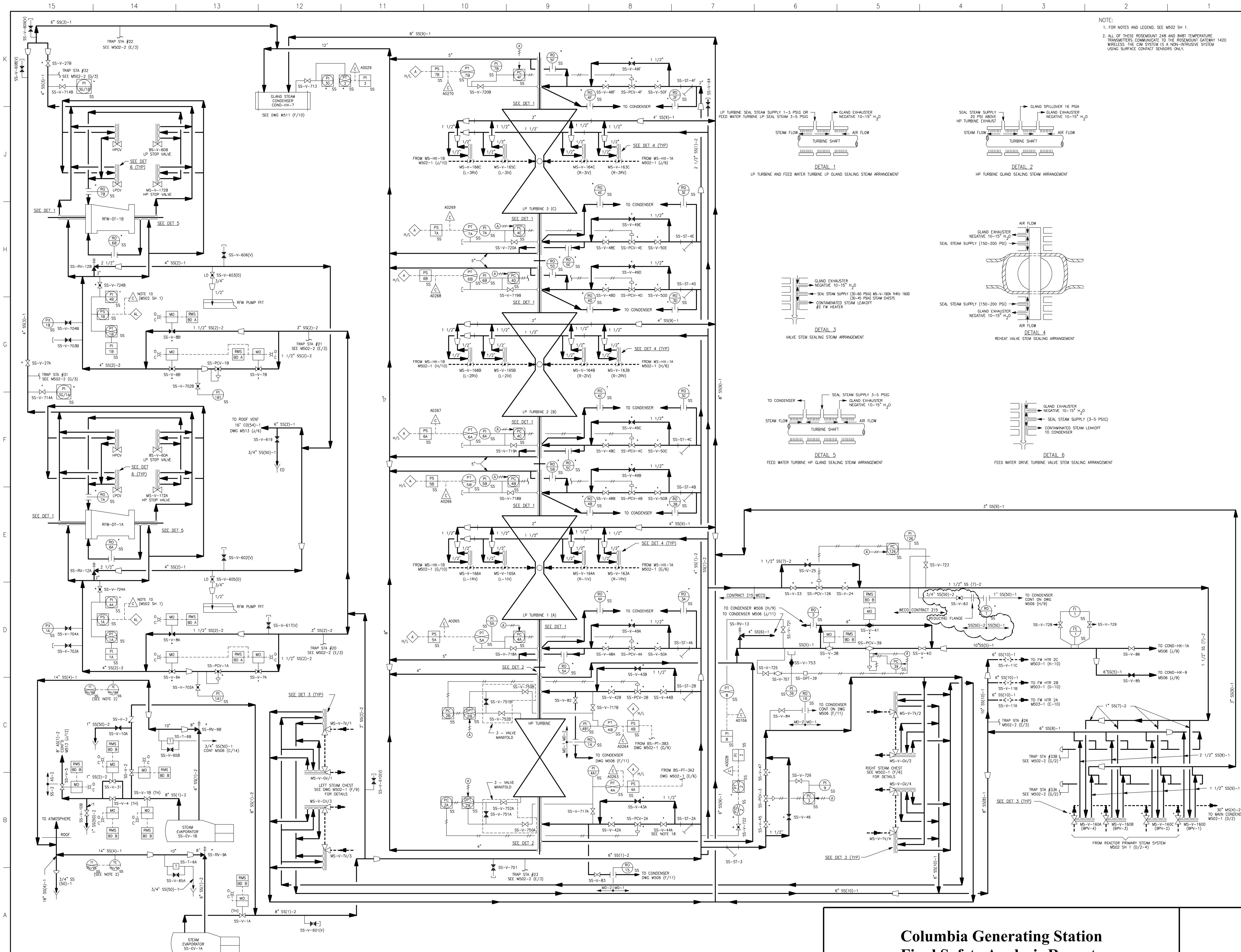
Draw. No. M502-2 | Rev. 34 | Figure 10.3-1.2



Columbia Generating Station
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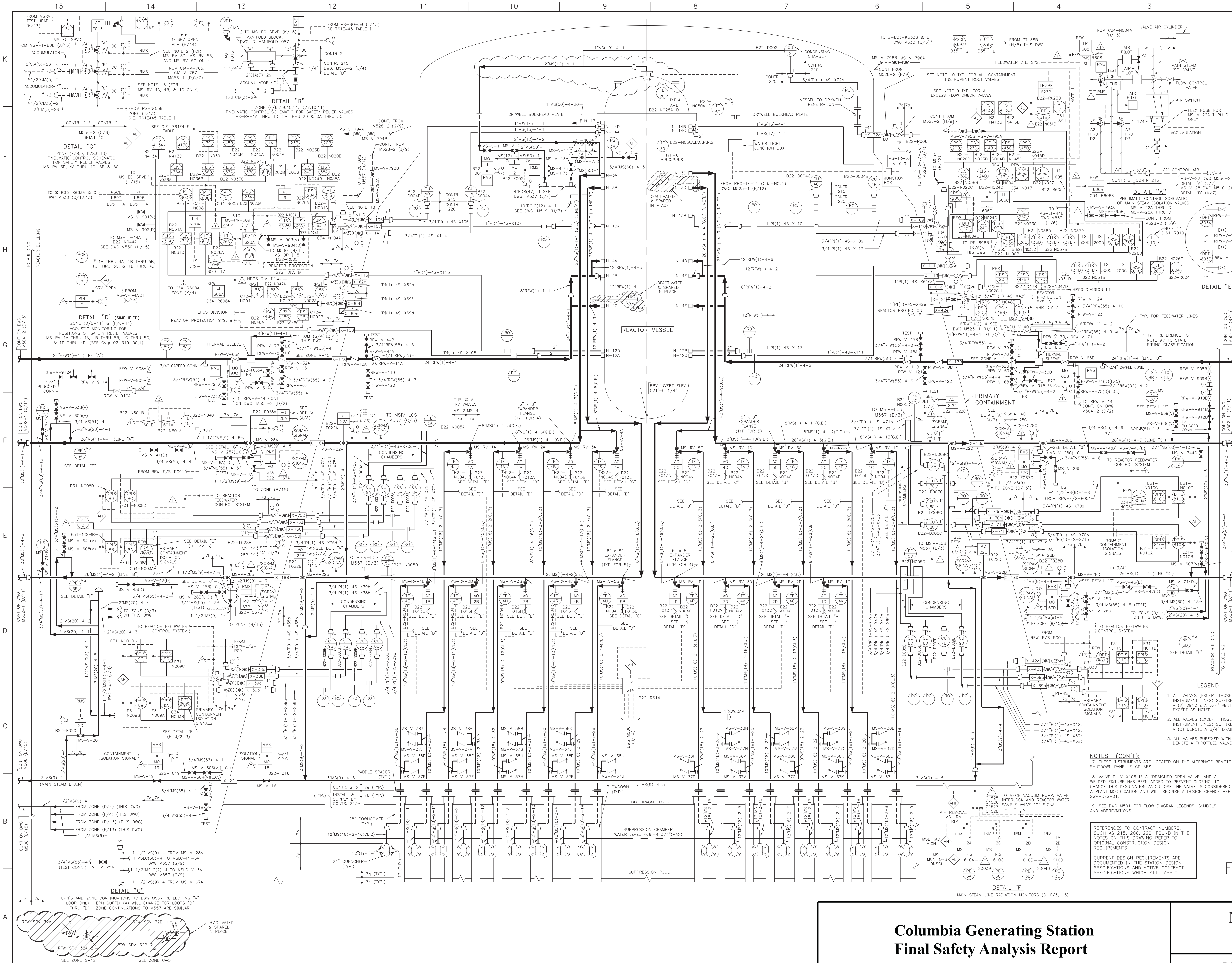
Main Steam Supply System

Draw. No. M502-3	Rev. 28	Figure 10.3-1.3



LEGEND

1. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE EXCEPT AS NOTED.
2. ALL VALVES (EXCEPT THOSE ON INSTRUMENT LINES) SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

NOTES:

1. FOR ALL INITIATION SIGNALS SEE G.E. NUCLEAR BOILER SYSTEM B22-1040, DOCUMENT #761E445, TABLE IV & V.
2. THIS EQUIPMENT IS NORMALLY CONTROLLED FROM THE MAIN CONTROL ROOM. IF THE MAIN CONTROL ROOM MUST BE VACATED, THIS CONTROL POINT MAY BE ISOLATED AND CONTROL TRANSFERRED TO THE ALTERNATE REMOTE SHUTDOWN PANEL E-CD-ARS.
3. (DELETED)
4. ALL PIPING SYSTEMS IDENTIFIED BY THE PREFIX "PI" SHALL BE SUPPLIED AND INSTALLED BY CONTRACT #220.
5. (DELETED)
6. (DELETED)
7. PIPING, VALVES, AND ASSOCIATED COMPONENTS ON THIS DWG SHALL BE CLASSIFIED AS FOLLOWS (BREAK POINTS ARE INDICATED ON THE FLOW DIAGRAM):

   a. PROCESS (FEEDWATER OR STEAM) AND INSTRUMENT PIPING WITHIN THE PRIMARY CONTAINMENT AND OUT THROUGH THE OUTERMOST ISOLATION VALVES, EXCEPT SAFETY/RELIEF VALVE DISCHARGE.

      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP A

   b. MAIN STEAM PROCESS PIPING BEYOND THE OUTERMOST CONTAINMENT ISOLATION VALVES, EXCEPT BRANCH LINES BEYOND THE FIRST VALVE THAT IS NORMALLY CLOSED OR CAPABLE OF TIMELY ACTUATION. SAFETY/RELIEF DISCHARGE PIPING AS NOTED. INSTRUMENTATION TO THE INSTRUMENT OR AS NOTED TO THE FIRST VALVE THAT IS CAPABLE OF TIMELY ACTUATION.

      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP B

   c. FEEDWATER PIPING BEYOND THE OUTERMOST CONTAINMENT ISOLATION VALVES. PIPING DOWNSTREAM OF THE FIRST VALVE CAPABLE OF TIMELY ACTUATION ON BRANCH LINES THAT CONNECT TO THE MAIN STEAM LINE BETWEEN PRIMARY CONTAINMENT AND TURBINE STOP VALVE.

      SEISMIC CATEGORY II ⊕
      QUALITY CLASS II
      CODE GROUP D

   ⊕ THE FEEDWATER PIPING, STEAM PIPING, AND THEIR PIPE SUPPORTS IN THE STEAM TUNNEL, ARE TO BE DESIGNED TO SEISMIC CATEGORY I LOADS.

   d. INSTRUMENTATION CONNECTED TO PRIMARY CONTAINMENT. INSTRUMENT PIPING OUTBOARD OF THE SECOND ISOLATION VALVES ON LINES THAT PENETRATE PRIMARY CONTAINMENT.

      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP B

   e. DOWNCOMERS AND SAFETY/RELEIF VALVE DISCHARGE PIPING AS NOTED.

      SEISMIC CATEGORY I
      QUALITY CLASS I
      CODE GROUP C

      CODE STAMP NOT REQUIRED FOR DOWNCOMERS

      ⊕ SEE SECTION 15Q, WNP-2 SPEC

   f. LINE 1 1/2"MS(9)-4 DOWNSTREAM OF MS-V-67A, B, C & D THROUGH THE REACTOR BUILDING WALL IS,

      SEISMIC CATEGORY II ⊕
      QUALITY CLASS II
      CODE GROUP D
      ⊕ SUPPORTED SEISMIC CATEGORY I

      IN THE TURBINE BUILDING THIS LINE IS,

      SEISMIC CATEGORY II
      QUALITY CLASS II
      CODE GROUP D

   g. MSRV QUENCHERS AND DOWNCOMERS AS NOTED SHALL BE CLASSIFIED AS FOLLOWS:

      SEISMIC CATEGORY I
      QUALITY CLASS I

      CODE GROUP: B FOR MATERIAL FABRICATION AND INSTALLATION ONLY. CODE STAMP NOT REQUIRED.

8. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
9. EXCESS FLOW CHECK VALVES SHALL BE TAGGED AS FOLLOWS: PI-EFC-X (PENETRATION NO).
10. CONTAINMENT INSTRUMENTATION ROOT VALVES SHALL BE TAGGED AS FOLLOWS: PI-V-X (PENETRATION NO).
11. THESE INSTRUMENTS ARE LOCATED ON REMOTE SHUTDOWN PANEL.
12. ALL INSTRUMENTATION ON THIS DRAWING TO BE IDENTIFIED BY A PREFIX "MS" UNLESS OTHERWISE NOTED.
13. ALL INSTRUMENT LINES ARE DESIGNED BY CODE & LINE NUMBER.

    A.) EXAMPLE: LINES ORIGINATING OUTSIDE CONTAINMENT,

    $\underbrace{\text{1/2"PI(1)-ST}}_{\text{CODE DESIGNATION}}$ – $\underbrace{\text{(H22-P021)-A6}}_{\text{LINE NUMBER}}$

    B.) EXAMPLE: LINES ORIGINATING INSIDE CONTAINMENT,

    $\underbrace{\text{1/2"PI(1)-4S}}_{\text{CODE DESIGNATION}}$ – $\underbrace{\text{X106}}_{\text{LINE NUMBER}}$

    C.) EXAMPLE: CONTINUATION OF THE LINE OUTSIDE CONTAINMENT IS DESIGNATED AS FOLLOWS, WITH LINE NUMBERS NOT SHOWN.

    $\underbrace{\text{1/2"PI(1)-4S}}_{\text{CODE DESIGNATION}}$ – $\underbrace{\text{X106}}_{\text{LINE NUMBER}}$

14. ASME CODE CASE N-241 MAY BE APPLIED TO 10" & 12"MS(18)-2 BELOW ELEVATION 466'-0 3/4".
15. CODE BREAK DEFINITIONS FOR THERMOWELLS ARE SHOWN ON M610, "INSTALLATION OF THERMOWELLS & SAMPLE PROBES."
16. THIS EQUIPMENT IS NORMALLY CONTROLLED FROM THE MAIN CONTROL ROOM. IF THE MAIN CONTROL ROOM MUST BE VACATED, THIS CONTROL POINT MAY BE ISOLATED AND CONTROL TRANSFERRED TO THE REMOTE SHUTDOWN PANEL C61-P001.

(NOTES CONTINUED ON LEFT)

NOTES (CON'T):

17. THESE INSTRUMENTS ARE LOCATED ON THE ALTERNATE REMOTE SHUTDOWN PANEL E-CP-ARS.
18. VALVE PI-V-X106 IS A "DESIGNED OPEN VALVE" AND A WELDED FIXTURE HAS BEEN ADDED TO PREVENT CLOSING. TO CHANGE THIS DESIGNATION AND CLOSE THE VALVE IS CONSIDERED A PLANT MODIFICATION AND WILL REQUIRE A DESIGN CHANGE PER SWP-DES-01.
19. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.

REFERENCES TO CONTRACT NUMBERS, SUCH AS 215, 206, 220, FOUND IN THE NOTES ON THIS DRAWING REFER TO ORIGINAL CONSTRUCTION DESIGN REQUIREMENTS.

CURRENT DESIGN REQUIREMENTS ARE DOCUMENTED IN THE STATION DESIGN SPECIFICATIONS AND ACTIVE CONTRACT SPECIFICATIONS WHICH STILL APPLY.

FSAR FIG.

Columbia Generating Station
Final Safety Analysis Report

Main Steam Supply System Piping

Draw. No. M529 | Rev. 106 | Figure 10.3-2

10.4 OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEM

10.4.1 MAIN CONDENSER

10.4.1.1 Design Bases

The purpose of the main condenser is to provide a heat sink for condensing the turbine exhaust steam, turbine bypass steam, reactor feed pump drive turbine exhaust steam, and to provide a receiver for miscellaneous drains. It also provides deaeration, noncondensable gas removal, and storage of condensate, which is returned to the condensate system after a period of radioactive decay.

The main condenser is designed for the following conditions (approximate values):

a.	Duty	7.7×10^9 Btu/hr
b.	Circulating water flow	555,600 gpm
c.	Circulating water inlet temperatures	78°F
d.	Circulating water outlet temperatures	106°F
e.	Total steam condensed	8,128,700 lb/hr
f.	Total condensate outflow	15,017,000 lb/hr
g.	Outlet temperature of condensate	105.0°F
h.	Condenser pressure (triple)	2.4 in Hg abs (average)
i.	Cleanliness factor	85%
j.	Number of passes	1
k.	Air inleakage flow rate limit	50 scfm
l.	Hotwell storage capacity	163,000 gal

The main condenser is designed to accept a maximum of 25% of the rated reactor steam flow from the turbine bypass system (described in Section 10.4.4) plus 75% of the rated reactor steam flow through the turbine. This steam flow is accommodated without increasing the condenser backpressure to the turbine trip setpoint or exceeding the allowable turbine exhaust temperature.

The main condenser is designed to deaerate the condensate and provide an oxygen content in the hotwell condensate between 30-100 ppb per liter over the entire load range.

Feedwater quality is maintained by the condensate filter demineralizer system described in Section 10.4.6.

The condenser hotwell is designed to contain the condensate that is required during 5 minutes of full power operation of the turbine. Baffling in the hotwell provides a minimum of 3 minutes condensate hold-up time which permits the decay of short-lived radioactive isotopes.

Condenser construction is designed in accordance with requirements of the Heat Exchange Institute, Standards for Steam Surface Condensers (October 1971). Construction of condenser module bundle replacement is designed in accordance with the Tenth Edition.

The piping associated with the condenser is designed, fabricated, inspected, and erected in accordance with ANSI B31.1 (October 1971). Seismic category, safety class, and design codes are discussed in Section 3.2.

10.4.1.2 System Description

Steam from the low-pressure turbine is exhausted directly downward into the condenser shells through exhaust openings in the bottom of the turbine casings and is condensed. The condenser serves as a heat sink for several other flows, such as exhaust steam from the reactor feedwater pump turbines, cascading feedwater heater drains, air ejector condenser drains, gland seal steam condenser drain, feedwater heater shell operating vents, turbine gland seals, and the offgas preheater drains.

Other flows to the condenser originate from the startup vents of the condensate pumps, the reactor feedwater pumps, condensate booster pumps, the condensate pumps, feedwater line startup flushing, reactor feedwater pump turbine drains, low-point drains, condensate makeup, reactor water cleanup (RWCU), and feedwater heater dumps or drains. All high temperature drains into the condenser shell have impingement baffles or spray pipes to prevent the steam and entrained water particles from impinging on the surface of the tubes. Stainless steel lagging is provided where required to protect other condenser components. The bypass valves are described in Section 10.4.4.2.

During transient conditions, the condenser is designed to receive turbine bypass steam and feedwater heater and drain tank high-level discharges. The condenser is also designed to receive relief valve discharges from moisture separators, feedwater heater shells, steam seal regulators, and various steam supply lines.

The condenser is cooled by the circulating water system described in Section 10.4.5. Air inleakage and noncondensable gases are removed by the main condenser evacuation system described in Section 10.4.2.

Before leaving the condenser, the condensate is deaerated to reduce the level of dissolved oxygen.

10.4.1.3 Safety Evaluation

During operation, radioactive steam, gases, and condensate are present in the shell of the main condenser. The inventory of radioactive contaminants during operation is discussed in Section 12.2.1.2.2.7. Shielding for and controlled access to the main condenser is provided in

Chapter 12. The means of controlling and detecting the leakage of this radioactive inventory in and out of the main condenser is discussed in Sections 11.3 and 11.5.2.2.

Hydrogen is generated by radiolysis in the reactor and injected by the Hydrogen Water Chemistry system.

Hydrogen generation buildup during operation is prevented by continuous evacuation of the main condenser by the air removal system (see Section 10.4.2) and the offgas system (see Section 11.3.2). The radiolytic decomposition rate at rated power is 102 scfm of hydrogen and 51 scfm of oxygen. Hydrogen is introduced into the condensate/feedwater system to mitigate intergranular stress corrosion cracking (IGSCC). The addition of hydrogen into a feedwater system results in reduced radiolysis. The net hydrogen in the steam during Hydrogen Water Chemistry (HWC) is less than during normal water chemistry (NWC) (without hydrogen injection). During plant shutdown, there are no hydrogen sources to the condenser. Hydrogen injection is shut down whenever the reactor is shut down. The inadvertent introduction of hydrogen to condensate/feedwater, from HWC, during extended shutdowns is prevented by isolation of the hydrogen supply, and purging the hydrogen injection system with nitrogen.

The main condenser is not required for safe shutdown of the reactor and does not perform safety functions. However, degradation of the condenser in the form of a leak, loss of circulating water, or air ejector malfunction could lead to a loss of condenser vacuum which removes the effective ability of the condenser as a heat sink. As a consequence, loss of vacuum provides a main steam isolation valve closure signal. See Section 7.3.1.1.2 for a further description. Due to the distance of the main condenser from safety-related equipment areas, there will be no damage to necessary safe shutdown equipment from flooding caused by failure of the condenser.

Exhaust hood overheating protection is provided by sprays located downstream of the last-stage blades of the turbine.

Loss of main condenser vacuum causes the turbine to trip. Should the turbine stop valves, control valves, or bypass valves fail to close on loss of condenser vacuum, rupture diaphragms on each turbine exhaust connection to the condenser protect the condenser and turbine exhaust hoods against overpressurization. In this event, steam would exhaust to the turbine building.

The main condenser is constructed with titanium tubes and the tubesheet is titanium clad carbon steel. Corrosion protection of the wetted carbon steel water boxes, inlet and outlet valves, and circulating water piping is being performed by a combination of a high quality coating, the use of stainless steels, and sacrifical anodes. The sacrifical anodes attach to the wall of the water boxes and will provide protection in case of any coating breaches or coating failure. All small bore nozzles on the circulating water side penetrating the water box 4 inches and less are of stainless steel. The water box coating will wrap into these connections to

prevent any carbon steel exposure. Isolation kits will electrically isolate any attached drain piping to help minimize any stray current corrosion.

10.4.1.4 Tests and Inspections

The condenser shell received a field hydrostatic test prior to initial operation. This test consisted of filling the condenser shell with water, and inspecting the entire tube sheet and shell welds and surfaces for visible leakage and/or excessive deflection.

The condenser module bundle replacement final test for tube or joint leakage was performed using a vacuum-bubble leak test.

During normal plant operation, the following parameters are routinely monitored:

a. Condenser vacuum,
b. Conductivity, and
c. Condensate temperature.

Any divergence from established limits for these parameters requires an investigation and testing as necessary to determine the extent of the divergence and correct the problem.

10.4.1.5 Instrumentation

The condenser shell is provided with local and remote hotwell level and pressure indication. The remote indication is by means of indicators and alarms in the main control room. The condensate level in the condenser hotwell is maintained within proper limits by automatic controls that provide for transfer of condensate to and from the condensate storage tanks as needed to satisfy the requirements of the steam system. Condensate temperature is measured in the outlet line of the condensate pumps.

Turbine exhaust hood temperature is monitored and controlled with water sprays to provide protection from exhaust hood overheating.

A main condenser low vacuum alarm is provided. Automatic turbine trip is activated on continued loss of main condenser vacuum followed by main steam isolation valve closure on further degradation of condenser vacuum. See Section 7.3.1.1.2 for a further description of main steam isolation.

Water box pressure and temperature measurements are provided.

Circulating water inleakage to the main condenser is monitored by conductivity elements located in the tube sheet troughs, in the condenser outlet line, and at the condensate pump discharge line (alarm in the main control room). Conductivity of the condensate demineralizer

influent is monitored and alarmed in the radwaste control room. A conductivity/chemical species sampling and measurement system is available for characterizing inleakage. Leakage is controlled (prevented) to the extent possible by maintaining chemistry control in the circulating water to provide optimization between scale formation and corrosion. Tube leakage can be corrected by isolating and draining the tube sections containing leaking tubes and then locating and plugging the leaking tube.

10.4.2 MAIN CONDENSER EVACUATION SYSTEM

10.4.2.1 Design Bases

The main condenser evacuation system removes gases from the turbine generator, the reactor feedwater pump turbines, and the main condenser during plant startup and maintains the condenser essentially free of noncondensable gases during operation. This system handles all noncondensable gases which may enter the main turbine and reactor feed pump turbines through their seals, the condensate piping, or which is generated by dissociation of water in the reactor. The main condenser evacuation system discharges to the offgas system through the steam jet air ejectors during normal operation (see Section 11.3).

The piping system associated with the main condenser evacuation system is designed, fabricated, and erected in accordance with ANSI B31.1 (October 1971). The air removal equipment is designed in accordance with the standards for Steam Surface Condensers, published by the Heat Exchange Institute (October 1971).

10.4.2.2 System Description

The main condenser evacuation system includes, for normal operation, two 100%-capacity steam jet air ejector units. Each unit consists of a twin-element first-stage steam jet air ejector and a single-element second-stage, steam jet air ejector which discharges to the offgas system. The capacity of each steam jet air ejector unit is 663 scfm at 70°F total equivalent of mixed gases and vapor at 1-in. Hg absolute. The main condenser design air inleakage flow rate is 50 scfm.

Two mechanical vacuum pumps are provided for hogging operation during startup (see Figure 10.4-1). During startup, both mechanical vacuum pumps can be used to rapidly remove air and noncondensable gases from the main condenser.

The discharge from the vacuum pumps is routed with the gland seal steam exhauster discharge to the reactor building elevated release duct. Because the reactor power is low, a minimal amount of activity is discharged to the environment. A radiation detector monitors the discharge and isolates the vacuum pumps when the radiation level exceeds established limits. The vacuum pumps operate until sufficient steam pressure is available to start the steam jet air ejector.

The source of steam to operate the steam jet air ejector is taken from the main steam header branch line, with steam pressure being regulated by the steam jet air ejector control valves. Air inleakage, noncondensable gases, as well as entrained water vapor, are removed from the main condenser by the first stage of the steam jet air ejector. The gas-vapor mixture is then discharged into the ejector condenser where the vapor is condensed. The resulting condensate is drained back to the main condenser via a loop seal. The ejector condenser is cooled by the condensate discharge from the condensate pumps. The noncondensing second stage of the steam jet air ejector removes the noncondensable gases and some entrained vapor from the ejector condenser and exhausts them to the offgas system (see Section 11.3). The offgas system processes the noncondensable gases and limits the release of radioactive gases to the environment.

10.4.2.3 Safety Evaluation

The main condenser evacuation system is not safety related. Consequently, the system is not designed to Seismic Category I requirements. Safety class and design codes are presented in Section 3.2.

The radionuclides in the effluent from the steam jet air ejector unit have been evaluated in Section 11.3.

The offgas from the main condenser contains hydrogen gas from dissociation of water in the reactor and from hydrogen injection. In the second-stage steam jet air ejector, sufficient steam is provided to dilute the hydrogen content to less than 4% by volume to keep the mixture below flammability limits.

10.4.2.4 Tests and Inspections

The mechanical vacuum pumps and the steam jet air ejectors were cleaned, inspected, and tested at the vendors' plant. System preoperational tests as described in Chapter 14 were successfully performed after installation. Main condenser evacuation system monitoring during normal operation along with routine maintenance and inspection ensures proper functioning and performance in accordance with its design bases. Instrumentation permits the operators to monitor system performance during operation.

10.4.2.5 Instrumentation

A radiation monitor is installed in the air removal piping discharge to the reactor building elevated release duct (a common exhaust line to the mechanical vacuum pumps and the gland seal steam condenser exhaust). A high-radiation signal from the monitor will trip both mechanical vacuum pump motors: The trip causes the suction and discharge valves to close and trips the mechanical vacuum pump seal water pumps. The vacuum pump is equipped with instrumentation to ensure proper operation (Figure 10.4-1).

A main steam line radiation monitor (MSLRM) high-radiation signal also trips both mechanical vacuum pump motors. The signal will also trip both gland seal steam condenser exhauster motors (see Section 11.5.2.1.1).

Low steam flow to the second-stage air ejector causes a signal to close the inlet gas valves to the first-stage air ejector. Steam pressure indicators for the first- and second-stage ejectors and a steam flow indicator for the second-stage ejector are provided in the main control room.

10.4.3 TURBINE GLAND SEALING SYSTEM

10.4.3.1 Design Bases

a. The turbine gland sealing system prevents air leakage into, or radioactive steam leakage out of, the main turbine and reactor feedwater pump turbines; and

b. The turbine gland sealing system is designed to provide nonradioactive (clean) sealing steam, at all loads, to the turbine shaft glands and valve stems (main stop, control reheat stop, intercept, and bypass valves). The condensate from the gland seal steam condenser is returned to the main condenser, and the noncondensable gases (inleaking air) are exhausted to the reactor building elevated release duct.

The turbine gland sealing system is in strict conformance with the latest edition in effect at the time of fabrication of the applicable ANSI, ASME, and IEEE standards. The major portion of manufacture was performed during 1975. The gland seal steam evaporators are designed, fabricated, inspected, tested, and stamped in accordance with Section VIII of the ASME Boiler and Pressure Vessel (B&PV) Code and the Standards of the Tubular Exchanger Manufacturers Association, Class R (May 1972). Seismic category, safety class, and design codes are provided in Section 3.2.

10.4.3.2 System Description

The turbine gland sealing system consists of two 100%-capacity gland seal steam evaporators, seal steam pressure regulators, seal steam header, gland seal steam condenser, exhauster blowers, and the associated piping, valves, and instrumentation (see Figures 10.3-1 and 10.4-2). Sealing steam for turbine shaft seal glands and valve stem seal glands (stop, control, reheat stop, intercept, and bypass valves) is supplied from the seal steam header at 200 psig. The source of sealing steam is from the gland seal steam evaporators or the auxiliary steam boiler. The sealing steam is produced in an evaporator which is heated by extraction steam taken from the high pressure turbine. The condensate fed to the evaporator is taken from the suction header of the reactor feedwater pumps in the feedwater system. During startup and low load operations, a branch line taken off the main steam header supplies the necessary heating steam for the evaporator.

Separate seal steam regulators are provided to regulate the pressure of sealing steam for the high pressure turbine, each low pressure turbine, each reactor feed pump turbine shaft seal, the bypass valve assembly, and the main stop and control valve assembly stems.

Since the low pressure (LP) turbine and reactor feedwater pump turbine exhaust pressures are at a vacuum, sufficient sealing steam is supplied to maintain positive pressure in the glands to prevent air inleakage along the shaft. The high pressure (HP) turbine exhaust pressure varies with load and is approximately 177 psia at its maximum. The system is designed to maintain the seal steam supply to the HP turbine glands at a pressure of 16 to 20 psi above HP turbine exhaust to prevent HP turbine exhaust steam leakage through the shaft gland seal.

The main stop, control, and bypass valve stems are provided with an intermediate zone to which sealing steam is supplied. This nonradioactive steam leaks in both directions, towards the HP stem leakoff and towards the LP stem leakoff. The HP stem leakoff contains radioactive steam and is directed to an LP feedwater heater. The LP steam leakoff is nonradioactive and is sent to the gland seal steam condenser. The reheat stop and intercept valve stems are supplied with sealing steam at a pressure greater than the crossover pressure so that any leakage that occurs is into the crossover pipes.

The reactor feedwater pump turbines at the high pressure end are provided with sealing steam at an intermediate point in the turbine gland seal. This nonradioactive steam leaks in both directions, towards the HP end leakoff and towards the LP end leakoff. The HP leakoff contains radioactive steam and is directed to the sixth stage of the turbine. The LP steam leakoff is nonradioactive and is sent to the gland seal steam condenser. Sealing steam for the reactor feed pump turbine LP stop valve, HP stop valve, and control valve is provided in a similar manner.

The outer leakoff of all glands is routed to the gland seal steam condenser which is maintained at a slight vacuum by the exhauster blower. During plant operation, the gland seal steam condenser and one motor-driven blower is in operation. The exhauster blower discharges gland air inleakage to the atmosphere via the reactor building elevated release duct. The gland seal steam condenser is cooled by the main condensate flow.

The steam evaporator is a shell-and-tube heat exchanger designed to provide a continuous supply of clean sealing steam to the seal steam header.

10.4.3.3 Safety Evaluation

The turbine gland sealing system is not safety related.

A supply of clean steam is always available from either of two 100%-capacity steam evaporators or the auxiliary steam boiler. Should the steam packing exhauster fail to function,

the sealing steam would continue to flow into the turbine and would be the only steam that could flow out of the glands and into the turbine building. Therefore, no reactor steam would be released to the environment.

A radiation monitor in the discharge of the blower alerts the operator to tube ruptures in the gland seal steam evaporator or other system malfunctions. Sealing system radioactive releases are discussed in Section 11.3.

Relief valves in the seal steam system prevent excessive steam pressure. The valves vent to the condenser and atmosphere.

10.4.3.4 Tests and Inspection

Prior to installation at the site, the gland seal steam evaporator and gland seal steam condensers were cleaned, inspected, and tested at the vendor's plant. Preoperational testing of this equipment included a hydrostatic test for visual inspection of welded joints to confirm leaktightness. The turbine gland sealing system is regularly inspected and monitored during operation to ensure proper functioning and performance in accordance with its design bases.

10.4.3.5 Instrumentation

The level in both the shell and tube sides of the steam evaporator are controlled by level-control valves: the condensate (shell) side by maintaining the water level surrounding the tubes and the steam side by maintaining the water level in the steam evaporator drain tank. The flow of heating steam is regulated by the steam pressure control valve.

Liquid level in the gland seal steam condenser is maintained by a trap connected to the main condenser. A local pressure indicator and high-level alarm switch are provided on the gland seal steam condenser. Temperature and pressure gauges and test points are provided to monitor operation and testing of the system. Instruments for monitoring system operation are provided in the main control room. Low and high level alarms are provided on the gland seal steam evaporator.

10.4.4 TURBINE BYPASS SYSTEM

10.4.4.1 Design Bases

a. The turbine bypass system controls reactor steam pressure by sending excess steam flow directly to the main condenser. This permits independent control of reactor pressure and power during reactor vessel heatup to rated pressure prior to and while the turbine is brought up to speed and synchronized under turbine speed-load control and when cooling down the reactor. Following main turbine generator trips and during power operation when the reactor steam generation

exceeds the transient turbine steam requirements, the turbine bypass controls reactor overpressure within its capacity and in accordance with the steam generation rate;

b. The turbine bypass system capacity is 25% of rated reactor steam flow. The bypass system can accommodate a 25% turbine load rejection without causing a significant change in reactor steam flow; and

c. The turbine bypass valves are capable of remote manual operation.

10.4.4.2 System Description

The turbine bypass system consists of four hydraulically operated control valves which are mounted on a valve manifold (see Figure 10.4-3). They are connected to the main steam line header upstream of the turbine main stop valves by four 10 in. lines. The four individual valves lower the pressure of the steam by reducing its flow velocity before it enters the condenser system.

Each valve outlet discharges into the manifold which is piped directly to pressure-reducing perforated pipes located in the condenser shell (see Figure 10.3-1).

The turbine DEH control system is designed to prevent spurious or unnecessary opening of the bypass valves, due to control signal noise or minor transients. The four valves in the manifold are operated automatically by the control system. The amount of steam flow allowed to pass through the turbine is limited by the DEH control system demand signal, which limits the amount that the governor valves can open. The DEH control system controls the governor valve position to maintain reactor pressure. When the governor valve opening position, required to maintain reactor pressure, exceeds the load demand limit, a signal is sent to the bypass valves to open to maintain reactor pressure. The bypass valves automatically trip closed whenever the vacuum in the main condenser is greater than approximately 23 in. Hg absolute. They have regulation capability and a fast-opening response approximately equivalent to the fast closure of the turbine stop and control valves.

The turbine bypass system piping and valves are designed to the class and seismic category presented in Table 3.2-1. The valve body is forged carbon steel while the internals are stainless steel. The environmental design bases for this system are contained in Section 3.11.

Each valve is sized for 8% of the total rated flow; however, all four valves are designed for 25% of the total flow. If the bypass system capacity is exceeded, the main steam relief valves open on high reactor pressure and excess steam is vented to the suppression pool.

10.4.4.3 Safety Evaluation

The effects of a malfunction of the turbine bypass system valves and the effects of such failures on other systems and components are evaluated in Section 15.2.2.

All safety-related components and the turbine speed control system are located remote from the turbine bypass piping and valves. The bypass system is located on the second floor of the turbine building (see Figure 1.2-3) and the speed control and safety-related components are located on the floor above, thus being separated by a concrete floor and wall making any adverse affects from a high-energy line failure in the turbine bypass system extremely unlikely. The turbine overspeed protection system is a fail-safe design, as described in Section 10.2.2.

The effects of a steam line break on the safety-related components in the turbine building are discussed in Section 3.6.1.

10.4.4.4 Tests and Inspections

The opening and closing of the turbine bypass system valves were checked during initial startup and shutdown for performance and timing. The bypass steam lines upstream of the bypass valves to MS-V-146 were hydrostatically tested to confirm leaktightness. Radiography and visual inspection of all pipe weld joints were performed on this piping. The branch connections and branch lines of this piping were examined in accordance with ANSI B31.1 rules.

Each turbine bypass valve can be tested independently and remotely during plant operation. The testing is conducted as required by the Technical Specifications.

10.4.4.5 Instrumentation

The controls and valves are designed so that the bypass valves shut on loss of control system electric power or hydraulic pressure. For testing the bypass valves during operation, the stroke time of the individual valves is increased during testing to limit the rate of bypass flow increase and decrease to approximately 1% per sec of reactor rated flow. Upon turbine trip or generator load rejection, the start of bypass steam flow is not delayed more than 0.1 sec after the start of the stop valve or the control valve fast closure motion. A minimum of 80% of the rated bypass capacity is established within 0.3 sec after the start of the stop valve or the control valve closure motion. For more detail refer to Section 7.7.1.5.

10.4.5 CIRCULATING WATER SYSTEM

10.4.5.1 Design Bases

The circulating water system is designed to provide cooling water for the condenser using the atmosphere as a heat sink via six circular mechanical-induced draft cooling towers designed to remove 7.962 x 10^9 Btu/hr from the circulating water. The design heat gain in the condenser is approximately 7.7 x 10^9 Btu/hr. In addition, the cooling towers have the capacity to cool the plant service water during normal operation and the standby service water during shutdown operation. The operation of the towers is not essential to the safety of the plant.

Makeup for tower evaporation, wind loss, and blowdown is obtained from the Columbia River by makeup pumps. Cooled blowdown from the cooling towers is discharged to the river.

Chemical treatment is provided for the circulating water system to preclude scale, biological growth, and consequent fouling of heat transfer surfaces.

The piping system associated with the circulating water system is designed, fabricated, inspected, and erected in accordance with ANSI B31.1 (October 1973) and AWWA C201. Major piping components are fabricated from carbon steel. Seismic category, safety classification and design codes are given in Section 3.2.

10.4.5.2 System Description

The circulating water system is shown schematically in Figure 10.4-4. The circulating water system is a closed cycle cooling system using six mechanical induced draft, cross-flow cooling towers. Three circulating water pumps, each having a total head of 95 ft at 186,000 gpm, are provided. These pumps, located in the circulating water pump house, take suction from a common intake plenum and discharge through a common 12-ft-diameter pipe to the three waterboxes of the single-pass triple-pressure zone condenser. The water from the condenser is returned to the cooling towers, cooled, and collected in the cooling tower basins which supply the circulating water pumps intake plenum.

In addition, as part of the cooling tower piping, a cooling tower bypass is provided for plant startup during the winter to prevent icing conditions at the towers. Inlet motor-operated valves are provided at each tower to isolate a tower for maintenance.

The towers are designed such that the buildup of ice will not restrict air flow through the louvers. Temperature range and low water temperature limits are maintained during reduced heat loads by shutting down individual towers (fans and flow) as required. In extreme cold weather, desired cold water temperatures can be achieved with all fan motors shut down but free to rotate with natural draft through the towers.

The six mechanical draft cooling towers are located such that there can be no physical interaction between them and plant structures important to safety in the unlikely event of a tower collapse.

The quantity of makeup water to the system is dependent upon cooling tower evaporation, drift losses, and system blowdown requirements. The system blowdown quantity is dependent on the concentration of dissolved solids allowed in the circulating water. The concentration of dissolved solids varies with operating status and the cycles of concentration which are controlled by operation of the blowdown valve. Makeup to the circulating water system is provided via the cooling tower makeup pumps located in a pump house adjacent to the Columbia River. The makeup pumps are designed to pump 12,500 gpm each with a TDH of 204 ft; the makeup flow can be directed into the circulating water bay or to one or both of the plant service water pump suctions by means of a weir box and sluice gate arrangement located in the circulating water inlet bay.

The evaporative-type cooling towers have the potential for creating visible plumes of water vapor under certain atmospheric conditions. The cooling tower system is designed to keep this environmental impact minimal. The cooling tower plumes rarely produce ground level fog or ice in the basin area where the plant is located and do not restrict traffic at the local airports. Since fogging occurs naturally in the area, the estimated incremental occurrences of fog attributable to cooling tower operations are small compared to the natural occurrences.

Cooling tower drift has been identified as a cause of arcing in switchyard equipment. Switchyard equipment is monitored and cleaned (as necessary) to preclude this phenomenon. Offsite environmental effects are monitored per Section 4.2.1 of the Environmental Protection Plan.

The radiological impact and the impact of thermal discharges on the environment are insignificant. The effect of cooling tower blowdown has no significant effect on the Columbia River temperature, and the environmental effect of chemical discharges is considered negligible. The system has no measurable effect on area groundwater.

The environmental considerations mentioned above are discussed in detail in Chapter 5 of the Environmental Report - Operating License Stage.

10.4.5.3 Safety Evaluation

The circulating water system is a non-safety-related system. Consequently, the circulating water system is not designed to Seismic Category I requirements. See Section 9.2.5 for a description of the ultimate heat sink which is designed to perform safety-related functions.

The condenser design ensures that the pressure on the tube side is always maintained higher than the pressure on the shell side, thus eliminating leakage into the circulating water system

should tube failure occur. Consequently, the design of the circulating water system precludes radioactive leakage into the system. Chemicals used to treat the CW system to preclude scale and biological growth and to control pH are evaluated in accordance with administrative controls to ensure compatibility with systems and components.

Two evaluations were performed to determine the effects of a postulated failure in the circulating water system inside the turbine building: a "realistic evaluation" and a bounding evaluation. For the "realistic evaluation," a moderate energy crack was postulated to occur in the circulating water system barriers (e.g., the rubber expansion joints) at the inlet to the main condenser. The inlet side was selected because it yields the severest results. For the bounding evaluation, a complete circumferential expansion joint break was assumed.

The entire condenser area is drained by means of sumps (see Figure 9.3-9), each equipped with duplex pumps. Sumps T-2 and T-3, servicing the inlet and outlet of the condenser, each have 50 gpm pumps. Each of these sumps is equipped with a level alarm and is therefore capable of detecting a circulating water system barrier failure. The level alarm will annunciate in the main control room upon reaching high level, providing a means of detecting the postulated failure within 5 minutes.

<u>"Realistic" Break</u>

The crack area for this postulated failure was assumed to be equal to one-half the pipe diameter times one-half the pipe wall thickness.

$$A = \frac{d}{2} x \frac{t}{2}$$ (see Section 3.6.2.1.4.2)

The flow exiting from such a crack would be an orifice flow. The head at expansion joint for normal three-pump operation at 186,000 gpm each was determined (from system energy gradients) to be 90 ft. The flow for these conditions was calculated to be

$$Q = 1737 \text{ gpm}$$

The system has different operating pressures for the various modes of pump operation. The piping was designed for an internal pressure of 60 psig, which is well above the design energy gradient.

The motor-operated inlet and outlet valves at the condenser are designed and manufactured to close in 60 sec to avoid excessive pressures caused by fast valve closure. Therefore, rapid valve closure is not a consideration. After closure of the inlet and outlet valves, however, the system will be operating with two-thirds of the condenser capacity. With three circulating water pumps in operation and two sections of the condenser in operation, the system flow as determined from the pump operating point diagram will be approximately 450,000 gpm.

Comparing the system energy gradients for this mode of operation to that when all three condenser units are in operation, the resultant difference in pressures will be

a. At the inlet side, an increase of approximately 4.3 ft of head (2 psi) occurs,
b. At the outlet side, a decrease of approximately 5.2 ft of head (2 psi) occurs.

Detection of the postulated failure will occur within 5 minutes, as described above, by the annunciation in the control room of the sump high level alarm. It is assumed that there will be a 15-minute time allowance for an operator in the control room to check the circulating water system barriers and close both the inlet and outlet valves of one unit of the condenser as may be required. This closure is accomplished by the activation of a remote manual switch in the control room, and therefore no control circuitry time delays nor coastdown times are involved. Flow will continue, however, after valve closure for about 106 minutes at a decreasing rate, until the remaining water from the condenser is completely discharged.

In the first 5 minutes after a crack, 8435 gal of water will spill into the inlet basin. The capacity of each basin and its capability to store excess flow were calculated to be as follows:

a. Inlet basin: 22,500 gal from el. 436 to el. 441,
b. Outlet basin: 27,500 gal from el. 436 to el. 441, and
c. Net volume under condenser: 180,500 gal from el. 433 to el. 441.

The time required to fill the inlet basin, after a postulated crack occurs, is computed to be 13.3 minutes. This includes the 50-gpm outflow from the sump pump. The circulating water leakage flow will continue for 6.7 minutes after filling the inlet basin, until reaching the total estimated shutoff time of 20 minutes. It can be assumed that 10% of this water will flow out over the floor at el. 441, and the remainder, about 10,170 gal, will flow into the condenser basin area. During this same time period, four sump pumps in the condenser basin area will have alternately pumped out 670 gal, leaving 9500 gal or 0.42 ft of water in the condenser basin. The rate of rise of water, therefore, is 0.021 ft/minute during the first 20 minutes after the postulated crack occurs. Note that on the high sump level, both pumps run simultaneously rather than alternately, thus doubling the calculated outflow capacity.

After the valves are closed, the water contained in the condenser unit water box will continue to discharge to the area. The quantity of water remaining is estimated to be 87,000 gal. The flow will vary with a diminishing head, the head going from about 25 ft to 0 ft. Using a 20-ft head and the same orifice flow criteria, the rate of flow will be approximately 819 gpm, discharging the remaining water in about 106 minutes. There will be an outflow from all the sump pumps of 150 gpm, with 10% of the flow from the crack again assumed to flow out over the floor. The water will accumulate in the condenser basin at about 590 gpm. After 106 minutes, the water level in this basin will rise an additional 2.77 ft, at 0.0261 ft/minute. The total height of water when the discharge has stopped is therefore 3.19 ft to el. 436.19.

This elevation is 5 ft below the floor level of the turbine building (el. 441), thus there will be no impacts on safety-related equipment from this event.

Boundary Evaluation

A complete circumferential expansion joint break in the circulating water system would result in the release of large amounts of water into the turbine generator building. The water would fill the net volume under the condenser, tripping the sump high level alarms that annunciate in the main control room. Remote-manual operation of the circulating water pumps and butterfly valves is provided in the main control room to mitigate the accident.

Disregarding operator action, however, the following evaluation is provided. Water would spill across the grade level floor of the turbine generator building at el. 441 ft, exiting through the railroad bay and access doors. Water could flow into the reactor building stairwells and elevator shafts from 441-ft el. down to the 422-ft 3-in. el., eventually filling the stairwells and elevator shafts with water. There is no safe shutdown equipment located in the stairwells or elevator shafts. The access doors to the emergency core cooling system (ECCS) and RCIC/CRD pump rooms at el. 422 ft 3 in. are designed to withstand a static head of approximately 44 ft (measured from centerline of door) of water. All penetrations into the reactor building below the 466 ft el. are designed to minimize flooding effects. Flooding will not affect any required safe shutdown equipment in the reactor building.

Water could also spill across the grade level floor into the radwaste/control building. The basement level of this building is 437 ft. It is thus possible to flood this level with 4 ft of water before the water would exit at grade level (441 ft) through access doors. No safety-related components will be affected by this flooding. The railroad bay and access doors of the turbine generator building are not watertight and are not designed to withstand any static head of water; therefore, no significant depth of water could accumulate in the turbine generator building. All safety-related equipment in the turbine generator building is located above the 471-ft el. and would not be affected.

In conclusion, a complete circumferential expansion joint break in the circulating water system inside the turbine generator building would have no effect on safety-related equipment in the turbine generator building and no effect on safe shutdown equipment in the reactor building.

Discharge operation of water accumulated under the condenser shall be performed in accordance with radioactivity checking requirements for sump discharges.

10.4.5.4 Tests and Inspections

All system components, except the condenser, are accessible during operation and may be inspected visually. The circulating water pumps were tested during preoperational testing.

The condenser was field hydrostatically tested in accordance with the Steam Surface Condenser Standards published by the Heat Exchange Institute.

All major components were inspected and cleaned prior to installation in the system, and preoperational tests were performed after system installation.

Sampling stations and test connections are provided to allow inservice testing during operation of the system.

10.4.5.5 Instrumentation

The circulating water pumps are individually equipped with shutoff valves that are interlocked with their respective pump motors to prevent startup unless the valve is closed and to prevent shutdown unless the valve is less than 15% open. Isolation valves are provided at the inlets of each condenser shell, which enable any water box to be isolated. The isolation valves are equipped with limit switches and are operated by manual switches located in the main control room. The system is monitored for temperature, pressure, level, and pH.

10.4.6 CONDENSATE FILTER DEMINERALIZER SYSTEM

10.4.6.1 Design Bases

The condensate filter demineralizer rated system capacity is 32,000 gpm, which is in excess of the 100% system capacity of 31,000 gpm.

As a design basis for this system, the effluent water quality is as follows:

NORMAL OPERATION FEEDWATER QUALITY TO THE REACTOR [a]

Parameter Frequency	*Limit*	*Sample*
Conductivity	*0.1 μmho/cm at 25°C* [b]	*Continuous*
pH	*6.5 to 7.5 at 25°C*	*As Required*
Total Metallic Impurity Filter Sample	*15 parts per billion (ppb)*	*Weekly; collected continuously*
Total Copper (Cu)	*2 ppb*	*Weekly*
Total Iron	*5 ppb*	*Weekly*

[a] *Measure after the last feedwater heater unless noted.*
[b] *Measured at demineralizer outlet.*

Nickel (Ni)	*2 ppb*	*Weekly*
Total Silica (SiO_2)	*5 ppb*	*As Required*
Chloride (Cl)	*10 ppb* [c]	*Daily*
Oxygen	*20 to 200 ppb*	*Continuous*

The design basis effluent water quality and sampling frequency was originally specified based on vendor fuel warranty and regulatory guidance. Since that time, enhanced analytical and testing techniques have determined that considerably lower concentrations of impurities can cause damage to system components. Through industry sponsored research, guidelines that are much more restrictive than the original design specifications have been developed and adopted. The implementation and documentation of these more restrictive specifications is controlled by chemistry administrative procedures.

This results in the system being designed to maintain feedwater quality such that the reactor water limits are not exceeded. This is achieved by the following:

a. Operation of the system to a less than 0.065 µS/cm conductivity end-point. After reaching this limit during normal operating conditions, the filter-demineralizer(s) are taken off line, backwashed, and precoated;

b. Establishing metallic impurity limits to preserved fuel performance by controlling the amount available for deposition on heat transfer and fluid transport surfaces. In addition, controlling corrosion product input minimizes the radiological impact from corrosion product activation, transport, and deposition;

c. Controlling undesirable anionic (chloride and sulfate) impurity input to maintain the reactor coolant concentrations below the levels where stress corrosion cracking is induced; and

d. Control of feedwater dissolved oxygen levels between 20 and 200 ppb falls in the minimal portion of the combined generalized and pitting corrosion curve for carbon steel piping. Piping is preserved and corrosion product activation, transport, and deposition are restricted.

This ensures that in conjunction with the RWCU system, reactor water quality will be maintained.

[c] *Or 25% of influent level, whichever is lower, to maintain reactor water quality of 200 ppb at rated operating pressure.*

The condensate filter-demineralizers are designed, fabricated, tested, and stamped in accordance with the ASME B&PV Code Section VIII, Division 1 (November 1971). Seismic category, safety class, and design codes are presented in Section 3.2.

10.4.6.2 System Description

The condensate filter demineralizer system consists of the necessary piping, valves, appurtenances, and instrumentation to control the condensate impurity concentration during plant operation (see Figure 10.4-5).

Six filter demineralizers are provided to polish 100% of the condensate flow: five or six are normally in operation at full power. The six filter demineralizers and associated piping, valves, and instrumentation are similar and piped in parallel.

Each filter demineralizer has an associated hold pump which is brought into service during low flow conditions to recirculate condensate through the filter demineralizer and hold the precoat material on the filter elements.

The individual effluent lines from the filter demineralizer vessels are provided with resin traps to prevent passage of ion exchange resins to the feedwater system.

The system design incorporates the following service systems which are common to all filter demineralizers:

a. Chemical mixing and supply system to circulate a chemical cleaning solution (e.g., inhibited citric acid) for the purpose of cleaning the filter demineralizer units and directing the waste to the chemical waste system (this system is not normally used);

b. A backwash system to remove the spent resin from the filter demineralizers and direct the radioactive waste to the backwash receiving tank (for further discussion of the backwash system discharge to the liquid waste management system, see Section 11.2); and

c. A precoat system wherein fresh precoat material is prepared and then circulated through the filter demineralizers to coat the filter elements.

The system control panels are located outside the equipment areas to permit remote operation of the condensate filter demineralizers without requiring the operator to enter high radiation areas.

The control panel has a graphic display. All major isolation valves, position indicators, and instrumentation are displayed at their respective locations.

10.4.6.3 Safety Evaluation

The condensate filter demineralizer system removes corrosion products, condenser inleakage impurities, and impurities present in the condensed steam.

Purified condensate and feedwater limits ensure sustained, safe plant operation by preserving the integrity of nuclear steam supply system components, vessel internals, fuel, and transport piping.

Due to improved water quality limits, any appreciable circulating water inleakage would result in water chemistry conditions outside acceptable limits and require action(s) to return the water quality to within applicable limits for continued plant operation.

Compliance with Regulatory Guide 1.56 is discussed in Section 1.8.

10.4.6.4 Tests and Inspections

The original condensate filter demineralizers, precoat and chemical mixing tanks, holding pumps, and system valves were hydrostatically tested prior to shipment by the manufacturer.

Field tests were performed after equipment installation to check satisfactory operation and functioning of control equipment, as well as to demonstrate guarantee performance. The guarantee performance test was governed by the ASTM Testing Method Procedures for High Purity Industrial Water.

10.4.6.5 Instrumentation

Instrumentation is provided for the condensate filter demineralizer system for proper operation, control, and protection against malfunction of the equipment.

The system design includes automatic flow balancing control for each filter demineralizer to maintain equal flow through each of the operating vessels by regulating the effluent discharge valve. The filter demineralizer flows are normally balanced manually and routinely monitored to maintain adequate flow balance. The cumulative flow through each filter demineralizer is recorded. Conductivity elements downstream of the flow control valves measure and record demineralizer performance. Differential pressure and conductivity alarms for each filter demineralizer annunciate when the pressure differential across a unit reaches a predetermined value or when the effluent conductivity indicates a significant reduction in ion exchange capacity. System influent and effluent conductivity are monitored and recorded. Alarms are provided for individual demineralizer differential pressures and outlet conductivities, and for the system inlet and outlet conductivities. These alarms annunciate at predetermined levels and

corrective action is initiated in accordance with plant procedures and licensee controlled specifications.

Conductivity instrumentation is calibrated in accordance with applicable ASTM Procedures.

An automatic bypass maintains the condensate system flow in the event the number of filter demineralizers in operation or the flow capacity of the units (due to clogging) is inadequate to handle the required flow.

The resin replacement equipment is designed for semiautomatic operation. A remote manual override is included as an alternate mode of operation.

Conductivity recorders, a grab sample rack with the necessary instrumentation and appurtenances to test influent and effluent condensate, differential pressure monitors, pressure indicators, and local alarms are provided for each unit in addition to the main graphic display control panel.

10.4.6.6 Demineralizer Resins

Compliance with Regulatory Guide 1.56 is discussed in Section 1.8.

Pressure precoat filter/demineralizer media on individual vessels is replaced on a cyclic basis when the pressure drop exceeds 25 psid or the effluent conductivity exceeds 0.065 µS/cm during normal operating conditions. The conductivity limitation does not apply when condenser vacuum is broken and during the period when condenser vacuum is being restored.

10.4.6.7 Water Chemistry Analyses

The filter-demineralizer condition during normal power operation is assessed by the effluent conductivity and ionic content. The influent conductivity is related to impurity concentration through the equivalent conductance of the constituents of the process fluid.

Chemical analysis methods used for determination of conductivity and ionic content are as follows:

Conductivity	Measured in accordance to ASTM-D-1125
Chloride	Determined by ion chromatography in accordance with the vendor's operating manual.

10.4.7 CONDENSATE AND FEEDWATER SYSTEMS

10.4.7.1 Design Bases

The condensate and feedwater system provides a reliable source of high purity feedwater during both normal operation and anticipated transient conditions. The system is designed with sufficient capacity to provide for 110% of the feedwater flow at rated load. This provides sufficient margin to provide flow under anticipated transient conditions. The feedwater heaters are designed to provide the required temperature of feedwater to the reactor. The final feedwater temperature is 423°F at rated load.

The condensate and feedwater system is designed and fabricated in accordance with ANSI B31.1, (October 1973) and ASME B&PV Code, Section VIII, Pressure Vessels (November 1971) and 2004 ASME B&PV Code, Section VIII, including 2005 Addenda for RFW-HX-6A and RFW-HX-6B. Seismic category and safety class are discussed in Section 3.2. The environmental design bases for this system are in Section 3.11.

10.4.7.2 System Description

The condensate and feedwater system shown in Figure 10.4-6 is a six-heater regenerative feedwater heating cycle. The extraction steam system supplying heating steam to each feedwater heater is shown in Figure 10.4-7. A discussion of the condensate supply system is presented in Section 9.2.6.

Feedwater heaters 1, 2, 3, and 4 are divided into three one-third capacity parallel trains; heaters nmber 5 and 6 are split into two one-half capacity parallel strings. The final feedwater temperature is approximately 423°F at design output. Tube material for RFW-HX-6A and RFW-HX-6B is type 316 stainless steel. For the rest of the heaters the tubes are type 304 stainless steel. The first-stage heaters are located in the condenser exhaust neck.

Figure 10.4-8 shows the heater drain system. All feedwater heater drains are cascaded back to the condenser (6-5-4-3-2-1 condenser). Reheater drains are carried to the number 6 heaters whereas the moisture separators drain to the number 5 heaters.

Condensate from the condenser hotwell is pumped by three motor-driven pumps of one-third capacity each. The condensate is pumped through the gland seal steam condenser, the steam jet air ejector condensers, the offgas condenser, the condensate demineralization system, and then to the suction of the condensate booster pumps. The condensate pumps are designed to pump approximately 11,000 gpm each with a TDH of 375 ft.

Three motor-driven condensate booster pumps are provided in the system. The capacity of the booster pumps matches that of the condensate pumps, one-third capacity for each pump at design rated feedwater flow. The booster pumps provide the required head to pump the

condensate through the five low pressure heaters and provide sufficient excess head to ensure sufficient net positive suction head (NPSH) at the reactor feedwater pumps suction.

The condensate booster pumps are designed to pump approximately 11,000 gpm each with a TDH of 925 ft. Using the condensate pumps, the condensate booster pumps and a series of heat exchangers, the system delivers 14,981,600 lb/hr of condensate at 467 psig and 366°F to the reactor feedwater pumps. Minimum flow through the gland seal steam condenser and steam jet air ejector condenser is controlled by using a recirculation control valve located in the condensate pump discharge lines to permit recirculation of condensate to the condenser.

Two one-half nominal capacity turbine-driven reactor feedwater pumps are provided. Each pump is capable of providing two-thirds of the rated feedwater flow during one pump operation. Minimum flow through the reactor feedwater pumps is controlled by using recirculation control valves located in the pump discharge lines to permit recirculation of feedwater to the condenser.

To minimize the corrosion product input to the reactor, a startup recirculation line is provided from the reactor feedwater supply lines, downstream of the high pressure feedwater heaters, to the main condenser.

The feedwater control system automatically controls the flow of feedwater into the reactor pressure vessel to maintain the water level in the vessel within predetermined levels during all modes of plant operation.

A hydrogen injection system is installed across the condensate booster pumps. The system uses discharge pressure to the pumps to feed dissolved hydrogen into the suction of the booster pumps.

A depleted zinc oxide (DZO) passive injection system (zinc) is installed across the feedwater pumps. The discharge pressure of the pumps can be used to inject a soluble zinc solution into the suction header of the feedwater pumps. Injection of zinc reduces the radioactive contamination on primary piping and components by reducing the levels of cobalt-60.

An iron injection system, installed on the suction line of the condensate booster pumps, can be used to inject an iron oxalate solution into the reactor feedwater. This iron injection system can be used to increase the reactor feedwater iron concentration to build a thin iron film on the inside of the piping to the vessel, vessel internals, and on the fuel.

Table 10.4-1 presents some of the major characteristics of equipment in this system.

10.4.7.3 Safety Evaluation

During operation, radioactive steam and condensate are present in the feedwater heating portion of the system which includes the extraction steam piping, feedwater heater shells, heater drain piping, and heater vent piping. Shielding and controlled access are discussed in Chapter 12. The condensate and feedwater system is designed to minimize leakage with welded construction used throughout the piping system. Feedwater heater shell-side relief valve discharges and operating vents are routed to the condenser.

The condensate and feedwater system is not required to effect or support the safe shutdown of the reactor or perform safety functions.

If it is necessary to remove a component such as a feedwater heater, pump, or control valve from service, continued operation of the system is possible by use of the multistream arrangement and the provisions for isolating and bypassing equipment and sections of the system.

The analysis of both the condensate and feedwater individual component failures is bounded by the feedwater component system failure analysis. These analyses are provided in Sections 15.1.1, 15.1.2, and 15.2.7. Included also in Section 15.6.6, are the isolation provisions that minimize release of radioactivity to the environment.

Criteria for feedwater isolation of the reactor coolant system is presented in Section 6.2.4.

10.4.7.4 Tests and Inspections

Each feedwater heater, heater drain tank, condensate pump, condensate booster pump, reactor feedwater pump, and system valves were shop hydrostatically tested at 1.5 times their design pressure. All pumps were shop performance tested. All tube joints of feedwater heaters were shop leak tested. Prior to initial operation portions of the completed ANSI B31.1 feedwater system welds were 100% X-rayed. The remainder of the completed condensate and feedwater system received a field hydrostatic test.

Pressure, temperature, conductivity, and flow instrumentation are provided to monitor system performance during operation. A separate, additional wireless monitoring system comprised of pressure, temperature and flow measuring equipment, is installed to monitor Feedwater Heaters 6A and 6B, as well as the Main Condensate Heaters 5A and 5B. The EMI/RFI characteristics were evaluated and found to be acceptable. Inservice inspection of applicable reactor feedwater piping is presented in Section 5.2.4.

10.4.7.5 Instrumentation

Feedwater flow-control instrumentation measures the feedwater flow rate from the condensate and feedwater system. This measurement is used by the feedwater control system that regulates the feedwater flow to the reactor to meet system demands. The feedwater control system is described in Section 7.7.1.4.

The isolation criteria for the feedwater system is loss of feedwater flow. Isolation valves are remotely operated from the main control room using signals which indicate loss of feedwater flow.

Instrumentation and controls regulate pump recirculation flow rate for the condensate pumps, condensate booster pumps, and reactor feed pumps. Measurements of pump suction and discharge pressures are provided for all pumps in the system. Sampling means are provided for monitoring the quality of the final feedwater (see Section 9.3.2). Temperature measurements are provided for each stage of feedwater heating and these include measurements at the inlet and outlet on both the steam and water sides of the heaters. Steam-pressure measurements are provided at each feedwater heater. Instrumentation and controls are provided for regulating the heater drain flow rate to maintain the proper condensate level in each feedwater heater shell and heater drain tank. High-level alarm and automatic dump-to-condenser on high level are provided.

Pressure, temperature, conductivity, and flow instrumentation are provided to monitor system performance. The operation of the hotwell makeup and high level dump valves is controlled by the hotwell level controller (Figure 10.4-6).

10.4.8 STEAM GENERATOR BLOWDOWN SYSTEMS

This section is not applicable to a BWR.

10.4.9 AUXILIARY FEEDWATER SYSTEM

This section is not applicable to a BWR.

10.4.10 HYDROGEN WATER CHEMISTRY SYSTEM

10.4.10.1 Design Bases

The Hydrogen Water Chemistry System (HWC) is designed to lower the electrochemical corrosion potential (ECP) of reactor coolant. Studies have shown that the lowering of ECP in the core below $-230\ mV_{SHE}$ will mitigate any existing intergranular stress corrosion cracking (IGSCC) and prevents future development of IGSCC in stainless steel components in the reactor coolant recirculation piping and lower reactor internals.

The HWC system injects hydrogen into the feedwater stream, increasing the concentration of dissolved hydrogen in reactor coolant. The presence of dissolved hydrogen suppresses the radiolytic generation of oxygen in the core and acts as a catalyst for the recombination of hydrogen and oxidants on the surface of piping and reactor internals. As a result, oxygen concentrations are reduced. The lower oxygen concentration reduces the ECP of the coolant.

The HWC system can inject up to 30 SCFM of hydrogen into the condensate/feedwater stream, resulting in a hydrogen concentration of up to 0.52 ppm in feedwater. Since the radiolytic generation of oxygen is suppressed, HWC also injects Service Air into the condenser Offgas stream to ensure that a stoichiometic ratio of hydrogen and oxygen are maintained for recombination.

10.4.10.2 System Description

The HWC system is shown schematically in Figures 10.4-9.1, 10.4-9.2, and 10.4-9.3. The system consists of a Hydrogen Storage and Supply Facility (HSSF), a hydrogen injection module, an air injection module, and a main control panel. The HSSF is located approximately 0.6 miles south-southeast of the Plant. A buried 2-inch pipe supplies hydrogen gas from the HSSF to the Turbine Generator Building (TGB) at approximately 200 psig. The hydrogen injection module, located on TGB 441′, regulates the gas flow to a sparger in a bypass line across the condensate booster pumps.

10.4.10.2.1 Hydrogen Storage and Supply Facility

The HSSF stores up to 14,000 gallons of liquid hydrogen at approximately 80 psig. The liquid H_2 is pumped, vaporized, and stored in six ASME storage tubes at approximately 2450 psig. The ASME tubes have a 40,000 SCF capacity, and serve as the primary source of gaseous H_2 for the HWC system. Gaseous hydrogen flows to a pressure control manifold that reduces the pressure to the 200 psig for supply to the TGB.

Two 100% capacity parallel pump trains, each with its own vaporizer, are provided for system reliability. One pump operates while the other acts as a backup. The ASME storage tubes are pressurized in a batch process, with pumping initiated when pressure in the tubes decays to approximately 650 psig.

The HSSF has a backup tube trailer that stores approximately 120,000 SCF of gaseous H_2. The HSSF has space for a second tube trailer that is used in the event of the functional loss of the liquid H_2 supply. Upon depletion of the inventory in the ASME tubes, the plant supply of hydrogen automatically switches to the tube trailer. As the inventory of the tube trailer is depleted, a replacement tube trailer is brought in. The depleted tube trailer is removed, replenished, and returned. In this way, the HSSF ensures a continuous, reliable supply of gaseous H_2 to the HWC system.

The HSSF also has a liquid nitrogen storage tank and vaporizer. Gaseous nitrogen is used for HSSF control functions and purging operations.

The operation of the hydrogen supply system at the HSSF is normally automatic, using programmable logic controllers (PLCs) located in a local pump control panel and hydrogen control panel.

The piping at the HSSF is designed to ASME B31.3, Chemical Plant and Petroleum Refinery Piping. The underground yard piping is designed to the requirements of ASME B31.1, Power Piping. All liquid and gas storage vessels are designed, fabricated, and stamped as ASME Boiler and Pressure Vessel Code, Section VIII, Division I, Unfired Pressure Vessels. The applicable fire protection codes are found in Appendix F Table F.3-1.

10.4.10.2.2 Hydrogen and Air Injection

The rate of hydrogen injection is regulated by a PLC in the HWC main control panel, located on the 471′ elevation of the TGB. Hydrogen injection is manually initiated above 5% reactor power, and is then automatically maintained at a rate proportional to reactor power when above 20% power. The injection rate is modulated based on reactor power.

HWC's suppression of radiolysis in the core results in an imbalance of hydrogen and oxygen in the condenser offgas stream. This imbalance is corrected by the injection of air into the Offgas system, upstream of the catalytic hydrogen recombiners. The Service Air system supplies the air for injection into offgas. The injection system is designed for a maximum flow rate of 93 SCFM.

The air injection rate is modulated based upon the rate of hydrogen injection. Since air leakage into the condenser contributes to the oxygen available for recombination, the required rate of air injection is reduced to account for the rate of air in-leakage.

The Mitigation Monitoring System (MMS) provides an indication of ECP in the reactor water. The MMS system contains an iron oxide element and a platinum element that may be used for measurement of the ECP of reactor water. ECP may also be monitored using an ECP LPRM probe in the core (see Section 7.6.1.4.2.2). The MMS and ECP LPRM provide an initial correlation of ECP to hydrogen injection to establish a baseline for operation of HWC.

10.4.10.3 Safety Evaluation

The HWC system does not fall within the definitions of any of the safety classifications identified in FSAR Section 3.2.3. However, the storage and handling of a combustible gas entails numerous safety issues that were addressed in the system's design.

The system was designed, procured, and installed to Quality Class II and Seismic Category II requirements. Exceptions are identified below.

- HWC cable terminations, electrical relays, and switches in the main control room panels are Quality Class I. Indicating lights in control room panels are Seismic Category 1M.

- HWC piping in the interfacing portion of the offgas system is Quality Class II+.

- The HSSF liquid hydrogen storage tank, its support foundation and soil were analyzed and installed to Quality Class I and Seismic Category I requirements.

All tanks and pipelines are provided with relief valves for overpressure protection. The liquid H_2 storage tank has redundant rupture discs for added protection.

The effects of the catastrophic failures of HSSF tanks at normal or elevated pressure were analyzed, and it was determined that the energy from such failures would not directly affect safety related or important to safety structures, systems, or components. Missiles generated from vessel failures at normal operating pressure would have insufficient energy to reach safety related or important to safety structures, systems, or components. Analysis has shown that missiles generated from an over pressurization event would have a total annual probability of impact less than 10^{-7} and therefore are not considered credible.

The hydrogen storage tanks at the HSSF are designed to stay in place for all natural phenomena (i.e., earthquakes, tornado winds, floods). No event will cause the tanks to be transported closer to the Plant. Local flooding from a Probable Maximum Precipitation (PMP) event would submerge the tanks (see Section 2.4.2.3), but not dislodge them. Similarly, the vent stack of the liquid H_2 storage tank is designed for flood conditions, with its outlet above PMP flood level. The vent stack design ensures that the tank will continue to off-gas vaporized hydrogen during the flood.

The HSSF is not designed to withstand tornado missiles. A tornado missile could cause the gross failure of a storage tank at the HSSF. However, as noted above, vessel failures have no effect on safety related or important to safety structures, systems, or components.

Similarly, an atmospheric release of all hydrogen stored at the HSSF will have no adverse impact on control room habitability. The HSSF has a maximum storage capacity of approximately 9800 pounds of liquid and gaseous hydrogen. The storage of this amount of hydrogen at the HSSF is not considered a hazard for control room habitability due to the distance from the plant air and remote air intakes to the HSSF.

Malfunctions at the HSSF will not have any adverse impact on Plant safety. The major potential effect of failures is the loss of supply of gaseous hydrogen for injection into condensate. The loss of hydrogen injection will cause a release of metallic radionuclides in reactor coolant, a transient N^{16} spike in the Main Steam lines and increase ECP in the reactor core. The Plant will continue to operate safely on loss of H_2 injection.

The buried supply line between the HSSF and the Plant is welded 2 in. schedule 80 pipe. Since the buried pipe passes under, and is routed next to a railroad track, the design was demonstrated to be in compliance with the American Railway Engineering Association (AREA) Manual for Railway Engineering.

In the area of the Plant, the buried H_2 supply line is encased in a guard pipe. The guard pipe provides mechanical protection, and a means to monitor the pipe for leakage. The vent of the buried line's guard pipe is directed to a hydrogen detector at the hydrogen supply valve station immediately outside the TGB. Hydrogen detectors are also located in the hydrogen injection module and at the Condensate system injection point. The HWC system is automatically shut down upon receipt of a high-high hydrogen signal from any of these detectors.

The hydrogen does not add to the combustible material in the TGB, since it is contained within welded pipe, and appropriate flow-limiting devices are included in the system. Excess flow valves are located in the supply line at both the HSSF and outside the TGB. The automatic closure of either excess flow valve would mitigate the effects of ruptures of the hydrogen supply line in or around the TGB.

The inadvertent introduction of hydrogen to condensate/feedwater during extended shutdowns is prevented by isolation of the hydrogen supply, and purging the hydrogen injection system with nitrogen.

The injection of hydrogen results in a transient increase in N^{16} activity in steam exiting the reactor, and an increase in the transport of metallic radionuclides to the recirculation system's piping and components. These effects are similar to those resulting from NobleChem injection (Section 5.2.3.2.2). The increased dose rates have no effect on equipment qualification.

The injection of hydrogen into feedwater reduces radiolysis in the reactor core, and results in a net reduction of hydrogen in Main Steam. The injection of hydrogen decreases secondary-side concentrations of dissolved oxygen. Secondary side chemistry is maintained in accordance with EPRI NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations". Dissolved oxygen is maintained in a range where Flow-Accelerated Corrosion (FAC) will not be exacerbated.

Hydrogen injection terminates automatically when feedwater flow drops below 25%. Hydrogen injection can also be terminated manually from the control room.

The injection of air into offgas increases the transport rate of radioactive species through the system with no adverse effect on system function. Offsite releases are maintained within the limits of 10 CFR 50 Appendix I and 10 CFR 20 Appendix B, Table II.

With the lowered net flow of hydrogen through offgas, the condenser's steam jet air ejectors provide sufficient dilution steam to maintain the hydrogen concentration below the maximum allowable concentration of 4% by volume (FSAR Section 10.4.2.3).

10.4.10.4 Tests and Inspections

The Hydrogen Water Chemistry system underwent a series of factory acceptance tests and site tests. The tests verified the operability of components, the logic of PLC programming, and the functionality of the integrated system. Site testing included the tuning of the HWC system to achieve the required ECP in the core, and benchmarking the Columbia Station's response to hydrogen and air injection.

10.4.10.5 Instrumentation

The Hydrogen Water Chemistry system is controlled by an integrated system of instrumentation and programmable logic controllers.

The HSSF is automatically operated by two PLCs, one located in the pump control panel and the other in the hydrogen control panel. The PLCs:

- Monitor pressures, temperatures and levels of the hydrogen and nitrogen systems,
- Control and monitor the hydrogen pumps and associated interlocks,
- Contain industrial safety interlocks for the HSSF facility, and
- Provide system status, process and alarm data to a remote annunciator panel.

The remote annunciator panel is located in the chemistry laboratory of the Radwaste Building, el 487′. The panel includes a human-machine interface PC that displays mimics, process information, and all HSSF alarms. The panel has one control interface with the HSSF, allowing the remote isolation of the HSSF from the Plant. The panel also receives HWC process information from a PLC in the HWC control panel, located on TGB el 441′.

The HWC control panel's PLC controls the injection of hydrogen and air into the condensate and offgas systems, respectively. The PLC control logic is based upon input flow signals from the hydrogen control module, oxygen control module, and feedwater system. In addition, the PLC receives signals from TGB hydrogen leakage detectors, offgas hydrogen analyzers, and a

summed alarm from the HSSF PLCs. Finally, the PLC receives an enable signal from a switch located on a panel in the main control room. The switch is a permissive for the operation of the hydrogen water chemistry system. The switch can also be used to shut the system down.

Hydrogen and air injection is manually initiated from the HWC Control Panel. Injection is automatically terminated by the PLC when the reactor is shut down. Injection can also be manually terminated at the HWC control panel.

Alarms from the HSSF and the HWC injection system are annunciated at the HWC control panel and in the main control room. HSSF process alarms display as a summed "HSSF Trouble Alarm" at the HWC control panel and as "HWC Trouble" in the main control room. The HWC control panel has annunciators for the display of specific, local process alarms. The HWC process alarms also actuate the summed "HWC Trouble" alarm in the main control room.

Any manual or automatic shutdown of HWC is annunciated as "HWC Shutdown" in the main control room. Automatic shutdown occurs if any of the following signals is received:

Reactor Scram
Offgas Isolation
High Hydrogen Flow
PLC Fault
High Hydrogen Pressure
Loss of Feedwater or H_2 Flow Signal
Low Process Air Pressure
Offgas Analyzers O.O.S.
High-High Area Hydrogen
Low Condensate flow at Injector
Offgas % Hydrogen High Shutdown
Purge
Local Shutdown Demand
Control Room Shutdown Demand

These signals simultaneously annunciate on the HWC control panel.

Table 10.4-1

Feedwater System Equipment Characteristics[a]

Condensate pumps	
Quantity	3
Capacity[a]	11,000 gpm/pump
Total discharge head	375 ft
Minimum flow	5600 gpm/pump
Driver	1250 hp ac motor
Condensate booster pumps	
Quantity	3
Capacity[a]	11,030 gpm/pump
Total discharge head	925 ft
Minimum flow	2500 gpm/pump
Driver	3000 hp ac motor
Reactor feedwater pumps	
Quantity	2
Capacity[a]	18,520 gpm/pump
Total discharge head	2585 ft
Minimum flow	Designed for 4600 gpm/pump at 5100 ft breakdown
Driver	Steam turbine
Steam jet air ejectors condenser	
Quantity	2-100%
Minimum cooling flow	5000 gpm
Gland seal steam condenser	
Quantity	2-100%
Design flow	6500 gpm
Pressure drop	6 psi

Table 10.4-1

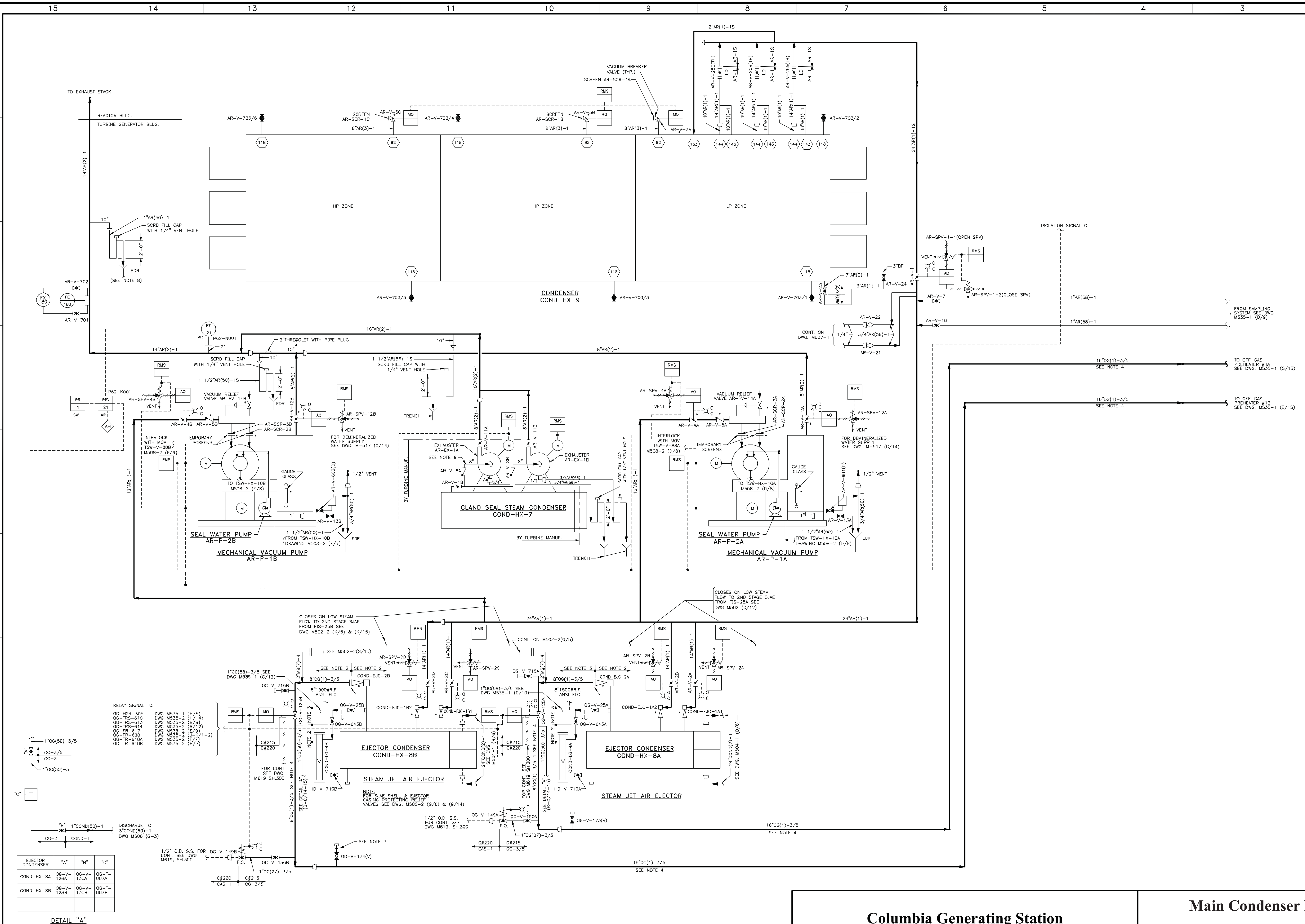
Feedwater System Equipment Characteristics (Continued)

Feedwater heaters	
Quantity	16 (one-third capacity up to and including heater 4 and one-half capacity for heaters 5 and 6)

Condensate (tube-side original design conditions)

Heaters	Flow/Chain (lb/hr)	Pressure Drop at Design Flow (psi)	Inlet Temp (°F)	Outlet Temp (°F)
1A, 1B, 1C	4,752,000	5.8	109.4	168.8
2A, 2B, 2C	4,752,000	5.4	168.8	208.5
3A, 3B, 3C	4,752,000	3.5	208.5	262.1
4A, 4B, 4C	4,752,000	6.0	262.1	291.3
5A, 5B	7,128,000	4.0	291.3	358.4
6A, 6B	7,128,000	9.9	360.1	419.8

[a] Capacity is based on 115% of the original rated condensate/feedwater flow.



NOTES:

1. ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
2. ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS ON THIS DRAWING IN THE AIR REMOVAL SYSTEM SHALL BE AS FOLLOWS EXCEPT FOR THE AIR EJECTOR DISCHARGES
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP D
3. ALL PIPING, VALVES, & ASSOCIATED COMPONENTS ON THIS DRAWING IN THE OFF GAS SYSTEM OR AIR EJECTOR DISCHARGES SHALL BE AS FOLLOWS:
   QUALITY CLASS II+
   SEISMIC CATEGORY II
   CODE GROUP D+
   +SEE NOTE 12, WNP-2 SPEC, SECTION 15B.1, TABLE 2 NOTES.
4. IF L/D RATIO OF THIS LINE IS < 7.0 DECREASED DESIGN PRESSURE PERMITS USE OF SCHEDULE 40 PIPING AND 600LB ANSI-RATED FLANGES.
5. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
6. GLAND STEAM CONDENSER EXHAUSTER (AR-EX-1A & 1B) SUCTION PIPING INCLUDING CHECK VALVES AR-V-8A & 8B ARE FURNISHED WITH THE GLAND STEAM CONDENSER (CONTRACT #1) WITH THE EXCEPTION THAT THROTTLE VALVE AR-V-18 AND ITS MATING SET OF FLANGES ARE TO BE FURNISHED AND INSTALLED BY THE 215 CONTRACTOR IN THE LOCATION SHOWN.
7. INDICATES VENT OR DRAIN VALVE PIPE CAP WAS SEAL WELDED TO ENABLE SYSTEM TO PASS HELIUM LEAK TESTING.
8. THIS EQUIPMENT DRAIN ROUTES TO FDR SUMP R4 (SEE M537 & M539).
9. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.

LEGEND

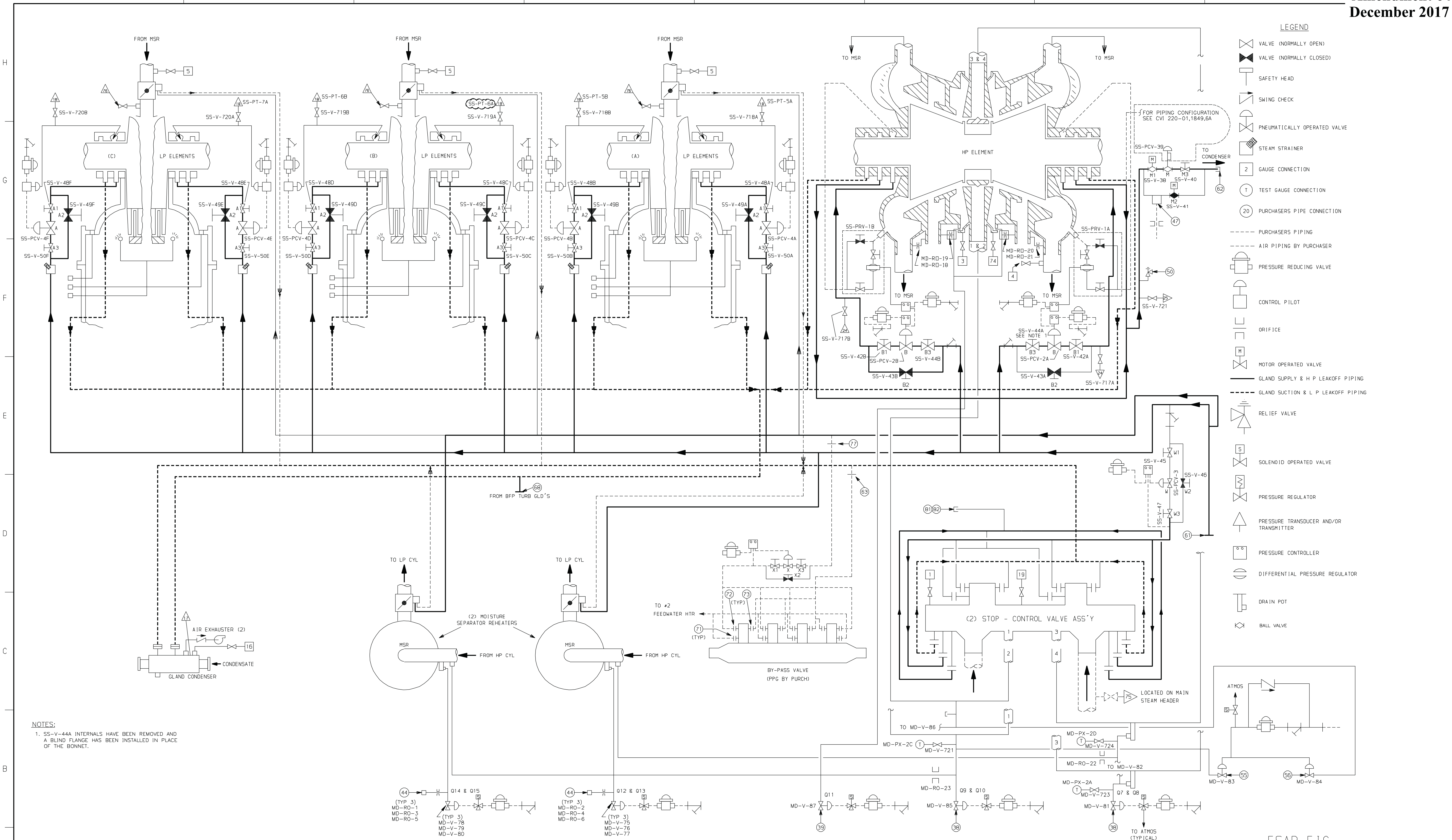
1. ALL VALVES SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

FSAR FIG.

Columbia Generating Station
Final Safety Analysis Report

Main Condenser Evacuation System

Draw. No. M511	Rev. 57	Figure 10.4-1

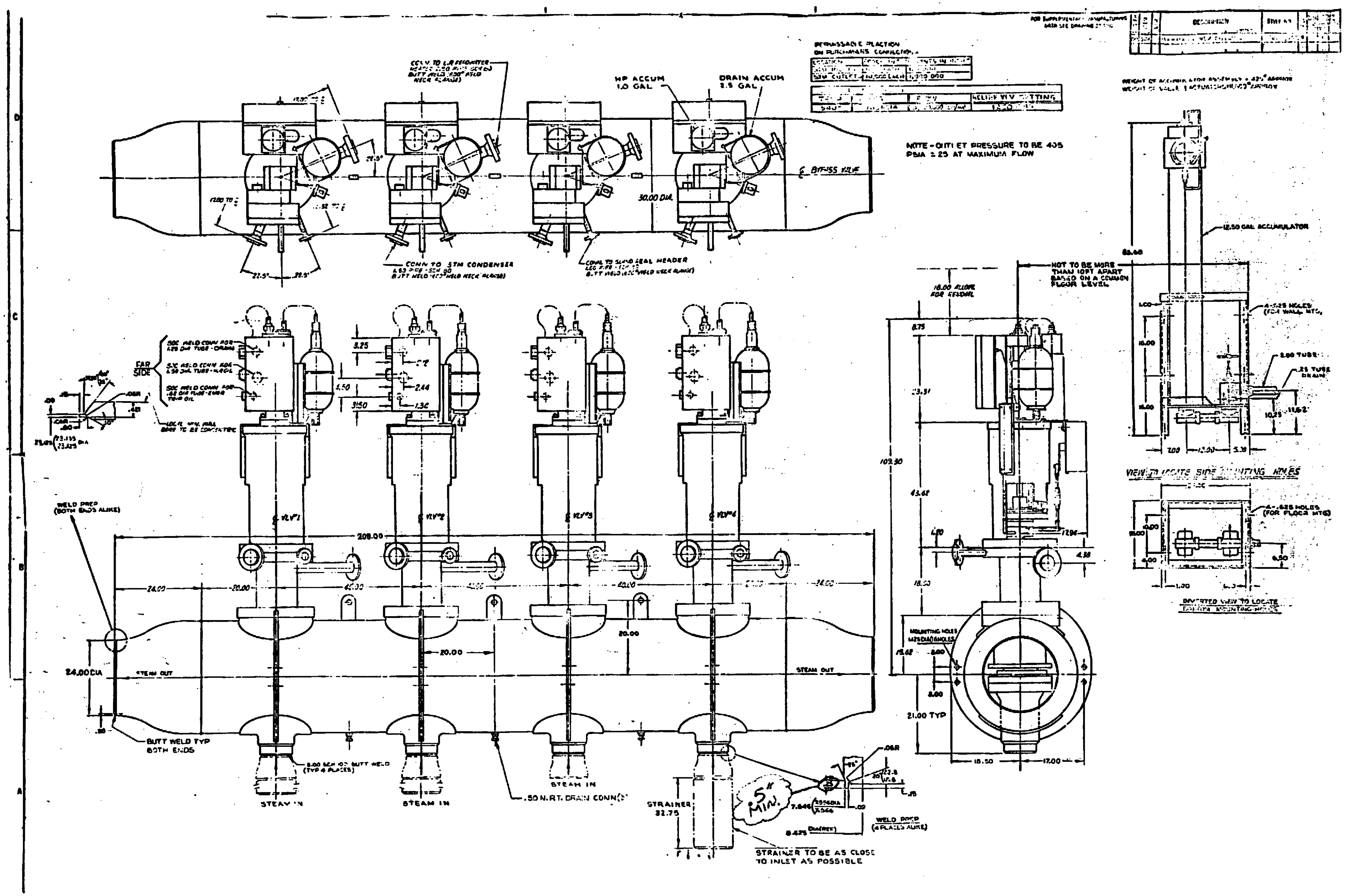


NOTES:
1. SS-V-44A INTERNALS HAVE BEEN REMOVED AND A BLIND FLANGE HAS BEEN INSTALLED IN PLACE OF THE BONNET.

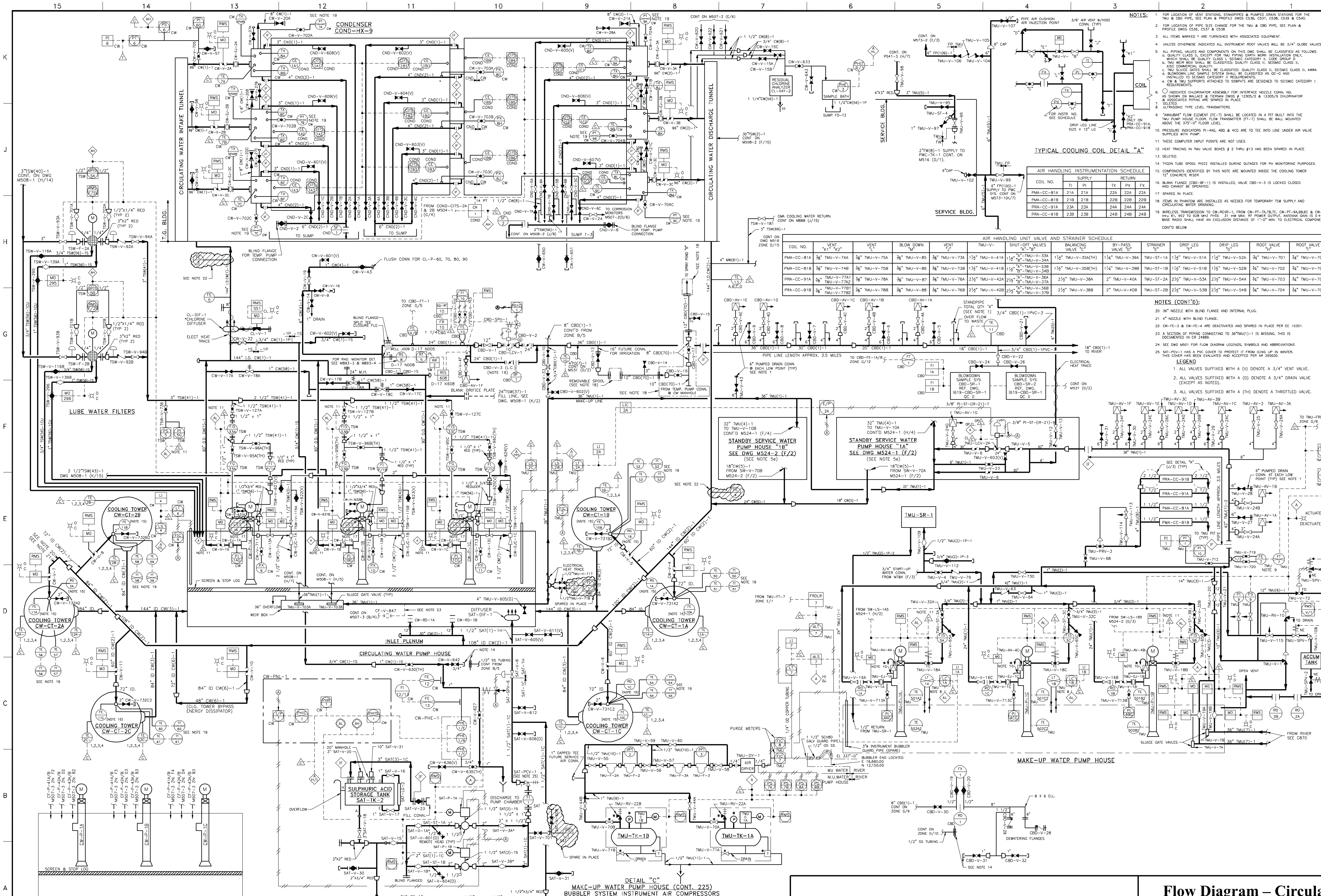
Columbia Generating Station
Final Safety Analysis Report

Turbine Gland Sealing System

Draw. No. 01-00,143,1 | Rev. 16 | Figure 10.4-2



Columbia Generating Station Final Safety Analysis Report	Turbine Bypass Valve Outline
	Draw. No. 01-00,110 · Rev. 4 · Figure 10.4-3



NOTES:

1. FOR LOCATION OF VENT STATIONS, STANDPIPES & PUMPED DRAIN STATIONS FOR THE TMU & CBD PIPE, SEE PLAN & PROFILE DWGS C536, C537, C538, C539 & C540.
2. FOR LOCATION OF PIPE SIZE CHANGE FOR THE TMU & CBD PIPE, SEE PLAN & PROFILE DWGS C536, C537 & C538.
3. ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
4. UNLESS OTHERWISE INDICATED ALL INSTRUMENT ROOT VALVES WILL BE 3/4" GLOBE VALVES.
5. ALL PIPING, VALVES AND COMPONENTS ON THIS DWG SHALL BE CLASSIFIED AS FOLLOWS:
   a. QUALITY CLASS II, EXCEPT FOR TMU PIPING EARTH WORK INSTALLATION ONLY, WHICH SHALL BE QUALITY CLASS I, SEISMIC CATEGORY II, CODE GROUP D.
   b. TMU WEIR BOX SHALL BE CLASSIFIED: QUALITY CLASS II, SEISMIC CLASS II, AISC COMMERCIAL QUALITY.
   c. TMU SLUICE GATES SHALL BE CLASSIFIED: QUALITY CLASS II, SEISMIC CLASS II, AWWA.
   d. BLOWDOWN LINE SAMPLE SYSTEM SHALL BE CLASSIFIED AS QC-G AND INSTALLED TO SEISMIC CATEGORY II REQUIREMENTS.
   e. CW & TMU SUPPORTS ATTACHED TO SSWPH'S ARE DESIGNED TO SEISMIC CATEGORY I REQUIREMENTS.
6. ◯ INDICATES CHLORINATOR ASSEMBLY FOR INTERFACE NOZZLE CONN. NO. AS SHOWN ON WALLACE & TIERNAN DWGS # 12305/2 & 12305/9 CHLORINATOR & ASSOCIATED PIPING ARE SPARED IN PLACE.
7. DELETED.
8. ULTRASONIC TYPE LEVEL TRANSMITTERS.
9. "ANNUBAR" FLOW ELEMENT (FE-7) SHALL BE LOCATED IN A PIT BUILT INTO THE TMU PUMP HOUSE FLOOR. FLOW TRANSMITTER (FT-7) SHALL BE WALL MOUNTED ABOVE THE 375'-0" FLOOR LEVEL.
10. PRESSURE INDICATORS PI-4AG, 4BG & 4CG ARE TO TEE INTO LINE UNDER AIR VALVE SUPPLIED WITH PUMP.
11. THESE COMPUTER INPUT POINTS ARE NOT USED.
12. HEAT TRACING IN TMU VALVE BOXES # 2 THRU #13 HAS BEEN SPARED IN PLACE.
13. DELETED.
14. TYGON TUBE SPOOL PIECE INSTALLED DURING OUTAGES FOR PH MONITORING PURPOSES.
15. COMPONENTS IDENTIFIED BY THIS NOTE ARE MOUNTED INSIDE THE COOLING TOWER 72" CONCRETE RISER.
16. BLANK FLANGE (CBD-BF-1) IS INSTALLED, VALVE CBD-V-3 IS LOCKED CLOSED AND CANNOT BE OPERATED.
17. SPARED IN PLACE.
18. ITEMS IN PHANTOM ARE INSTALLED AS NEEDED FOR TEMPORARY TSW SUPPLY AND CIRCULATING WATER DRAINING.
19. WIRELESS TRANSMISSION TO CW-RCVR-1, FROM CW-PT-7A,7B,7C, CW-PT-9A,9B,9C & CW-TT-50 thru 61, 902 TO 928 MHZ FHSS. 31 mW MAX. RF POWER OUTPUT. ANTENNA GAIN IS 0.44 DBI. BASE RADIO SHALL HAVE AN EXCLUSION DISTANCE OF 1'-0" MIN. TO ELECTRICAL COMPONENTS

CONT'D BELOW

AIR HANDLING INSTRUMENTATION SCHEDULE

COIL NO.	SUPPLY			RETURN		
	TI	PI		TX	PX	FX
PMA-CC-81A	21A	21A		22A	22A	22A
PMA-CC-81B	21B	21B		22B	22B	22B
PRA-CC-91A	23A	23A		24A	24A	24A
PRA-CC-91B	23B	23B		24B	24B	24B

AIR HANDLING UNIT VALVE AND STRAINER SCHEDULE

COIL NO.	VENT "K1" "K2"	VENT "L"	BLOW DOWN "M"	VENT "J"	TMU-V-	SHUT-OFF VALVES "A"-"B"	BALANCING VALVE "C"	BY-PASS VALVE "D"	STRAINER "E"	DRIP LEG "F"	DRIP LEG "G"	ROOT VALVE "H"	ROOT VALVE "I"
PMA-CC-81A	3/8" TMU-V-74A	3/8" TMU-V-75A	3/8" TMU-V-85	3/8" TMU-V-73A	1 1/2" TMU-V-41A	1 1/2" "A"-TMU-V-33A "B"-TMU-V-34A	1 1/2" TMU-V-35A(TH)	1 1/4" TMU-V-39A	TMU-ST-1A	1 1/2" TMU-V-51A	1 1/2" TMU-V-52A	3/4" TMU-V-701	3/4" TMU-V-705
PMA-CC-81B	3/8" TMU-V-74B	3/8" TMU-V-75B	3/8" TMU-V-86	3/8" TMU-V-73B	1 1/2" TMU-V-41B	1 1/2" "A"-TMU-V-33B "B"-TMU-V-34B	1 1/2" TMU-V-35B(TH)	1 1/4" TMU-V-39B	TMU-ST-1B	1 1/2" TMU-V-51B	1 1/2" TMU-V-52B	3/4" TMU-V-702	3/4" TMU-V-706
PRA-CC-91A	3/8" TMU-V-77A1 TMU-V-77A2	3/8" TMU-V-78A	3/8" TMU-V-87	3/8" TMU-V-76A	2 1/2" TMU-V-42A	2 1/2" "A"-TMU-V-36A "B"-TMU-V-37A	2 1/2" TMU-V-38A	2" TMU-V-40A	TMU-ST-2A	2 1/2" TMU-V-53A	2 1/2" TMU-V-54A	3/4" TMU-V-703	3/4" TMU-V-707
PRA-CC-91B	3/8" TMU-V-77B1 TMU-V-77B2	3/8" TMU-V-78B	3/8" TMU-V-88	3/8" TMU-V-76B	2 1/2" TMU-V-42B	2 1/2" "A"-TMU-V-36B "B"-TMU-V-37B	2 1/2" TMU-V-38B	2" TMU-V-40B	TMU-ST-2B	2 1/2" TMU-V-53B	2 1/2" TMU-V-54B	3/4" TMU-V-704	3/4" TMU-V-708

NOTES (CONT'D):

20. 36" NOZZLE WITH BLIND FLANGE AND INTERNAL PLUG.
21. 4" NOZZLE WITH BLIND FLANGE.
22. CW-FE-3 & CW-FE-4 ARE DEACTIVATED AND SPARED IN PLACE PER EC 10301.
23. A SECTION OF PIPING CONNECTING TO 36"TMU(1)-1 IS MISSING. THIS IS DOCUMENTED IN CR 24889.
24. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.
25. SAT-PCV-1 HAS A PVC COVER TO PROTECT IT FROM ICING UP IN WINTER. THIS COVER HAS BEEN EVALUATED AND ACCEPTED PER AR 295530.

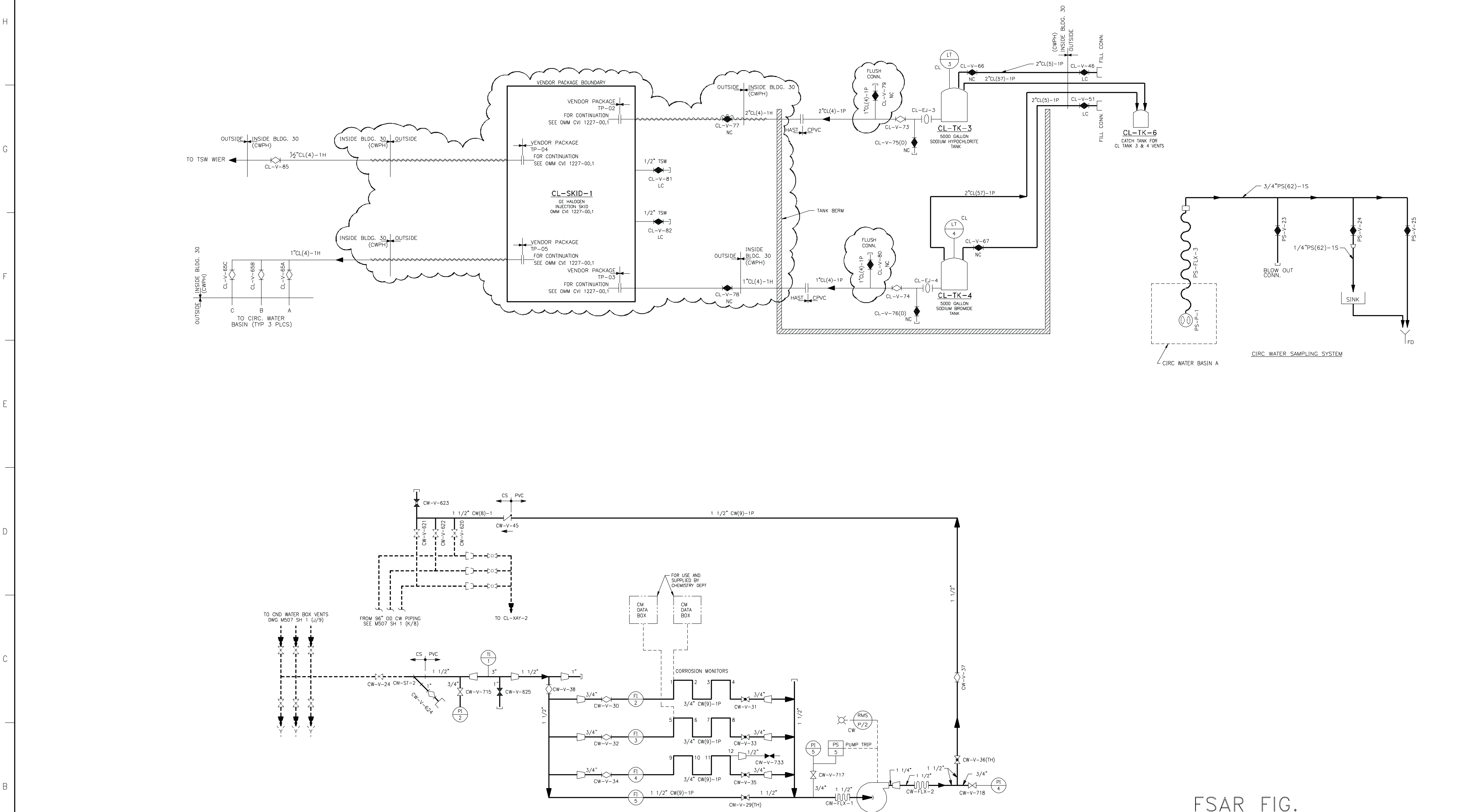
LEGEND

1. ALL VALVES SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE (EXCEPT AS NOTED).
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

Columbia Generating Station Final Safety Analysis Report	Flow Diagram – Circulating Water System – Turbine Generator Building and Yard
	Draw. No. M507-1 / Rev. 169 / Figure 10.4-4.1

NOTES:

1. FOR NOTES AND LEGEND SEE DRAWING M507-1.

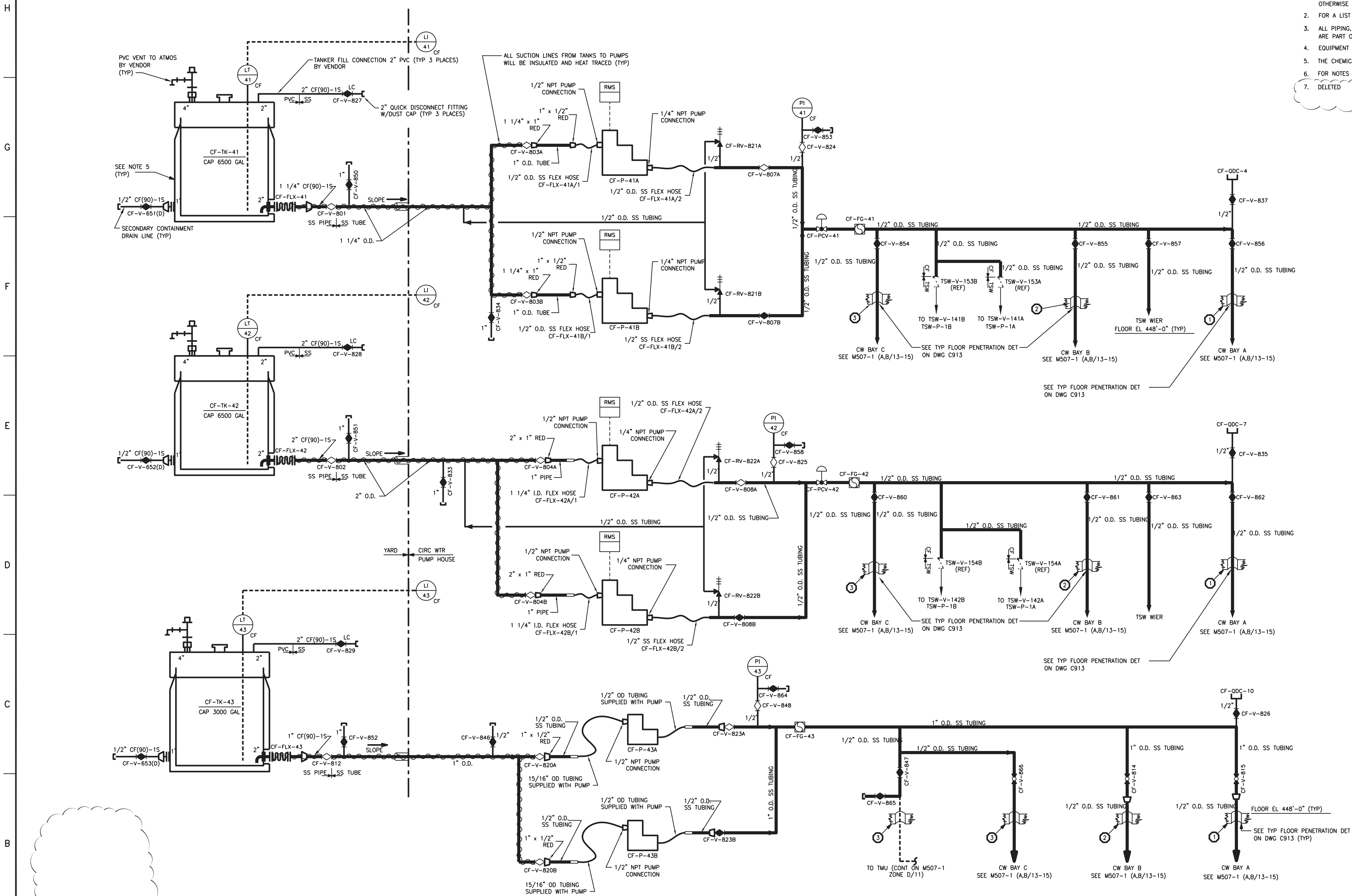


Columbia Generating Station
Final Safety Analysis Report

Flow Diagram - Circulating Water System - Turbine Generator Building and Yard

Draw. No. M507-2	Rev. 8	Figure 10.4-4.2

Form No. 960690ai



NOTES:

1. ALL WORK ON THIS DRAWING SHALL BE QUALITY CLASS G UNLESS OTHERWISE NOTED.
2. FOR A LIST OF REFERENCE DWGS REFER TO DWG C912.
3. ALL PIPING, VALVES, TANKS, PUMPS, INSTRUMENTS, ETC. ON THIS DWG ARE PART OF THE CHEMICAL FEED SYSTEM (CF).
4. EQUIPMENT IS LOCATED IN SOUTHEAST PORTION OF CIRC WATER PUMPHOUSE.
5. THE CHEMICAL STORAGE TANKS ARE HEAT TRACED AND INSULATED.
6. FOR NOTES AND LEGEND SEE DRAWING M507-1.
7. DELETED

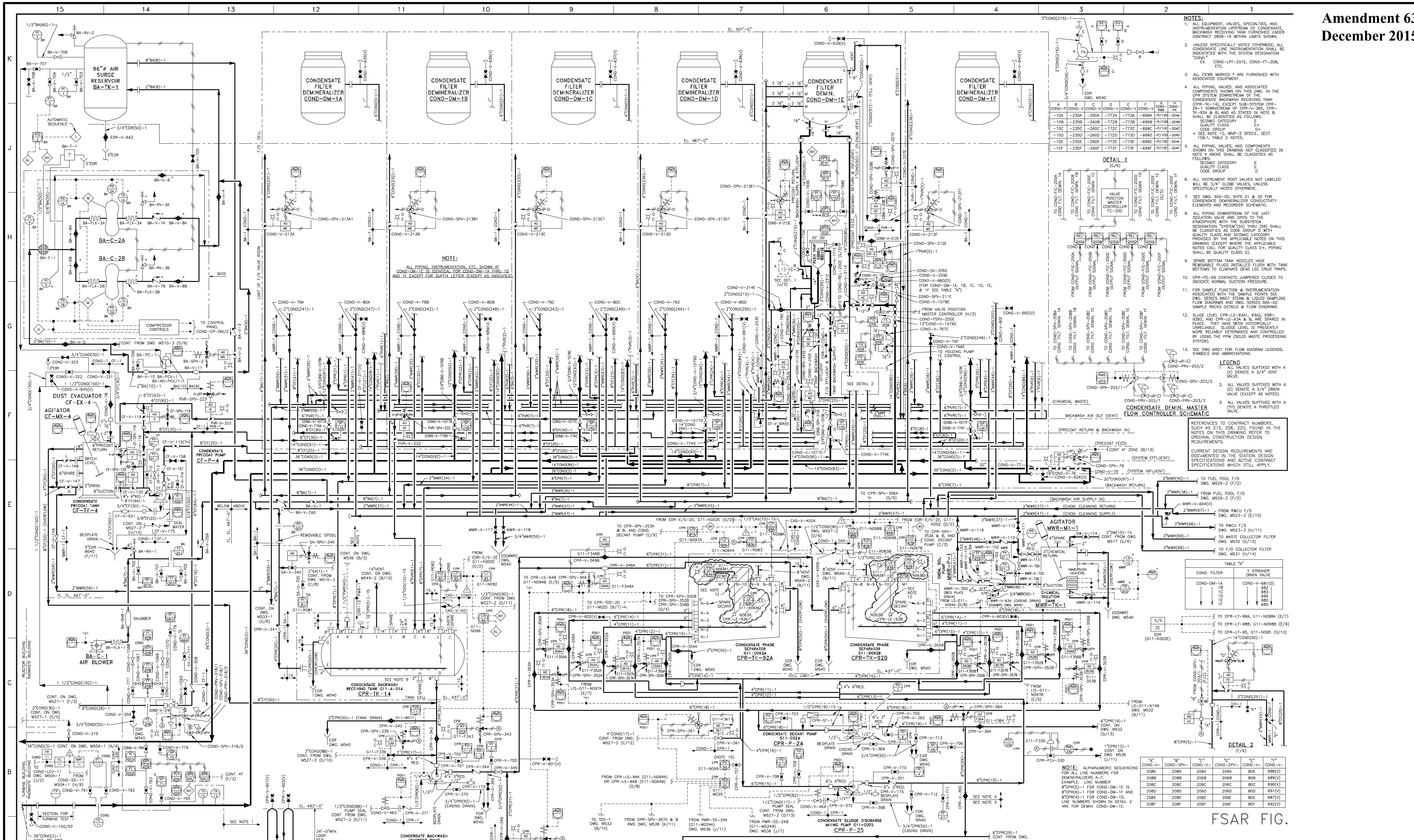
FLOOR PENETRATION LOCATION

PEN NO	SUCTION BAY	APPROXIMATE COLUMN GRID LOCATION	
1	CW-P-1A	4.8	W.55
2	CW-P-1B	4.8	X.55
3	CW-P-1C	4.8	Z.55

Columbia Generating Station
Final Safety Analysis Report

Flow Diagram - Circulating Water System - Turbine Generator Building and Yard

Draw. No. M507-3 | Rev. 13 | Figure 10.4-4.3

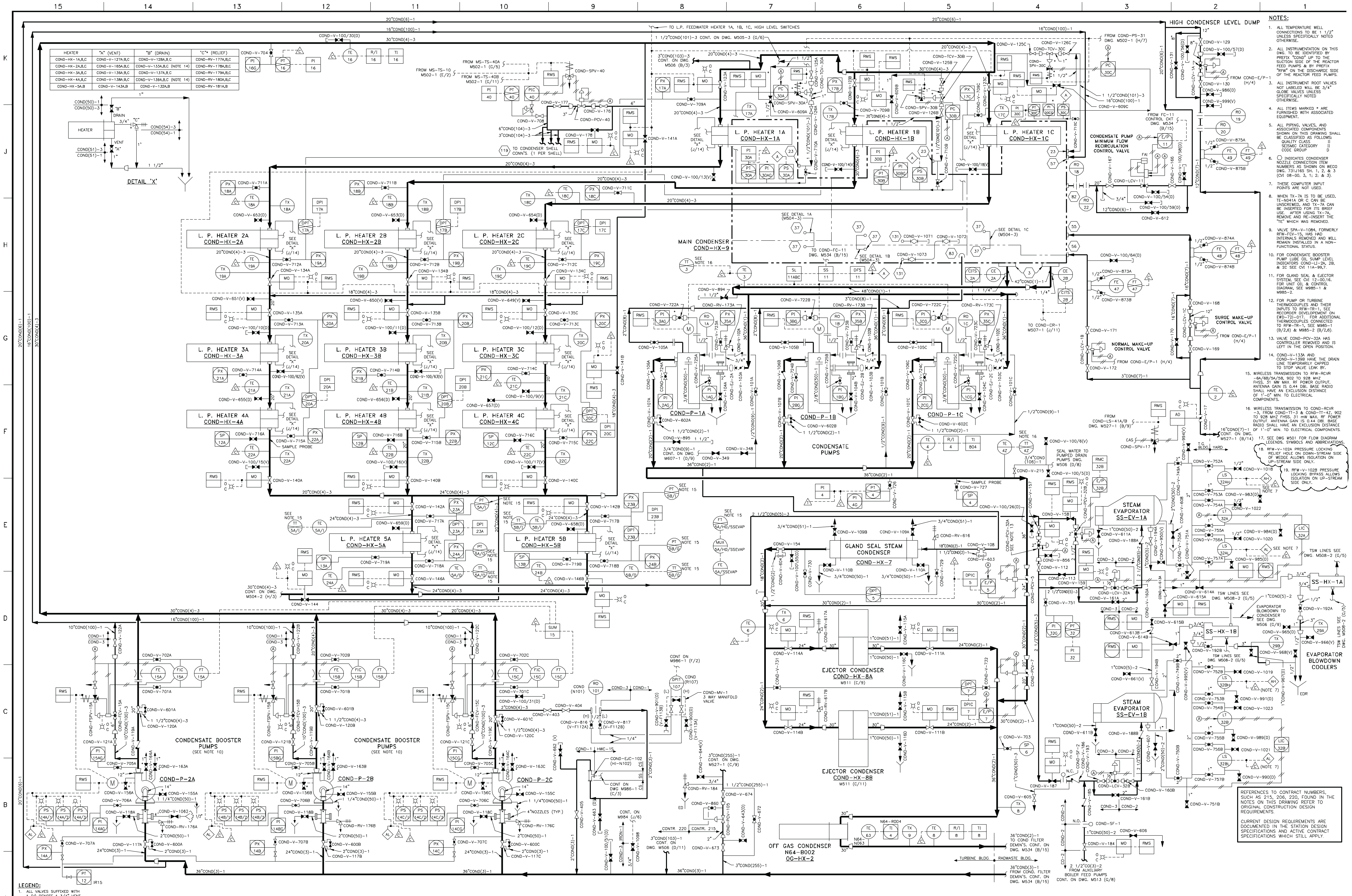


Columbia Generating Station
Final Safety Analysis Report

Flow Diagram - Radioactive Waste System
Condensate Demineralization

Draw. No. M534 | Rev. 71 | Figure 10.4-5

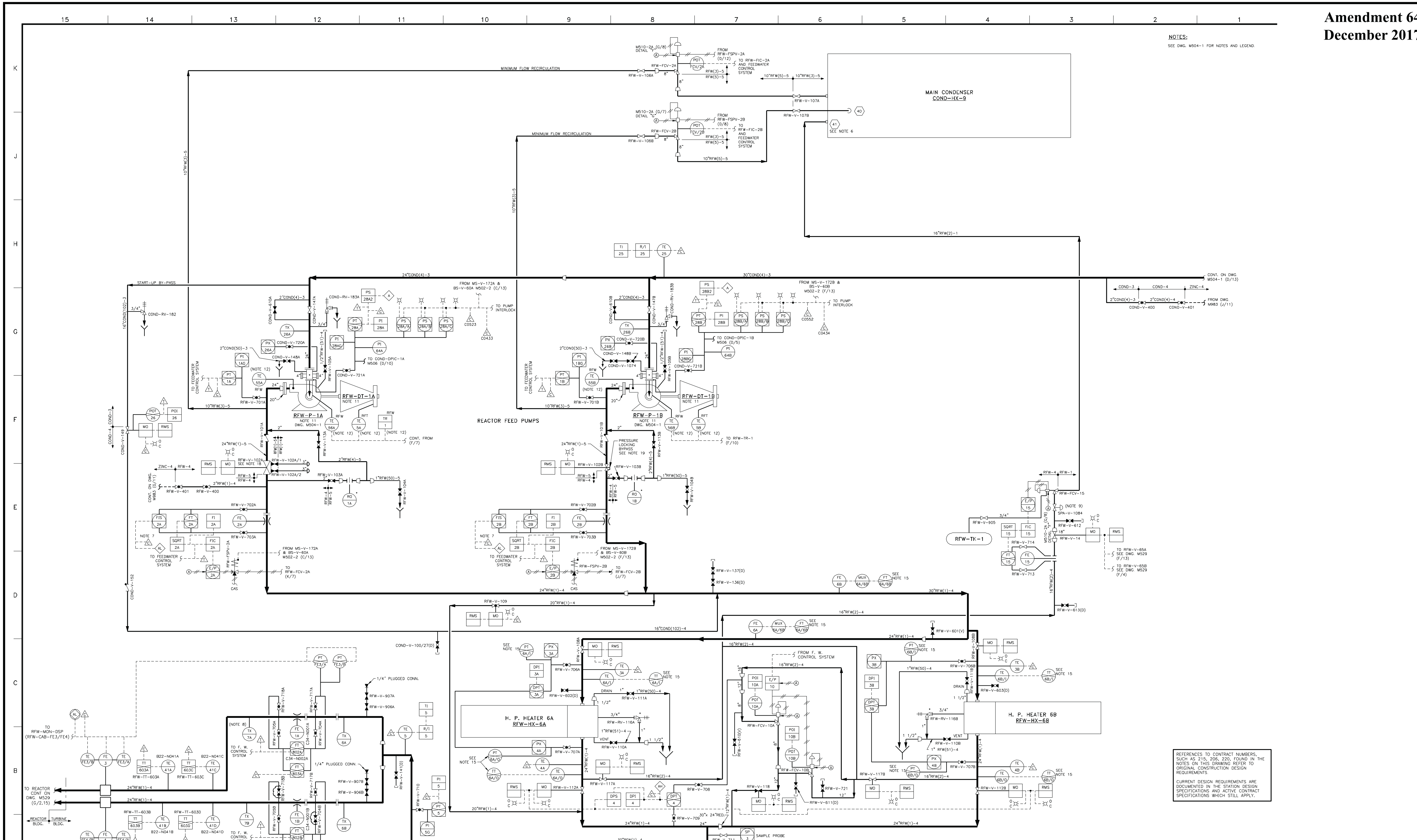
Form No. 960690ai



Columbia Generating Station
Final Safety Analysis Report

Flow Diagram - Condensate and Feedwater Systems

Draw. No. M504-1	Rev. 110	Figure 10.4-6.1



Columbia Generating Station
Final Safety Analysis Report

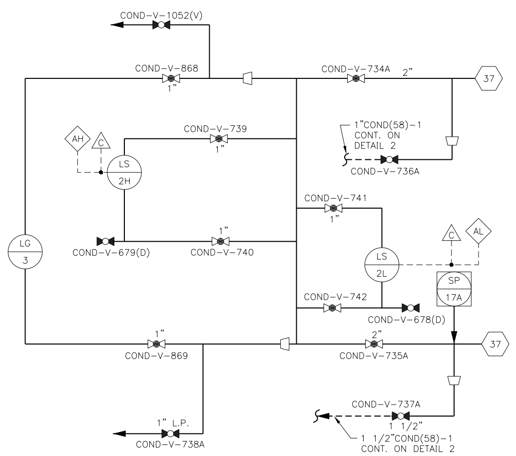
Flow Diagram – Condensate and Feedwater Systems

Draw. No. M504-2	Rev. 15	Figure 10.4-6.2

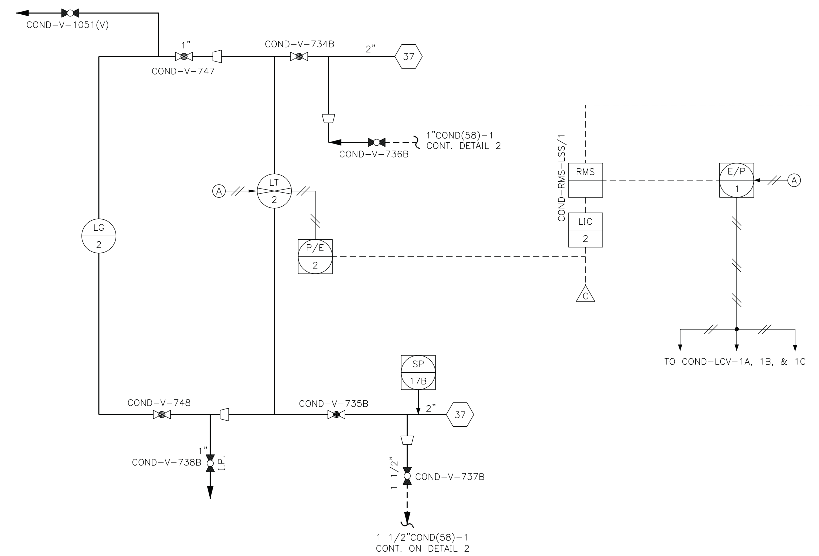
Amendment 61
December 2011

NOTES:

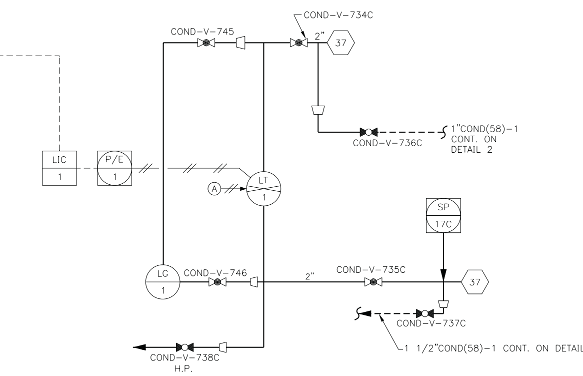
1. SEE DWG. M504-1 FOR NOTES AND LEGEN



DETAIL 1A

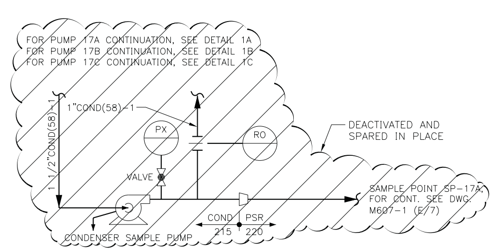


DETAIL 1B



DETAIL 1C

LEVEL CONTROL ARRGT. AT CONDENSER HOTWELL



CONDENSER SAMPLE PUMP TAG NUMBERS					
PUMP No.	RO	SP	PX	VALVE	REMARKS
COND-P-17A	17A	17A	10A	COND-V-855A	PUMP IS NOT INSTALLED. PIPING IS SPARED IN PLACE.
COND-P-17B	17B	17B	10B	COND-V-855B	PUMP IS NOT INSTALLED. PIPING IS SPARED IN PLACE.
COND-P-17C	17C	17C	10C	COND-V-855C	PUMP IS NOT INSTALLED. PIPING IS SPARED IN PLACE.

DETAIL 2

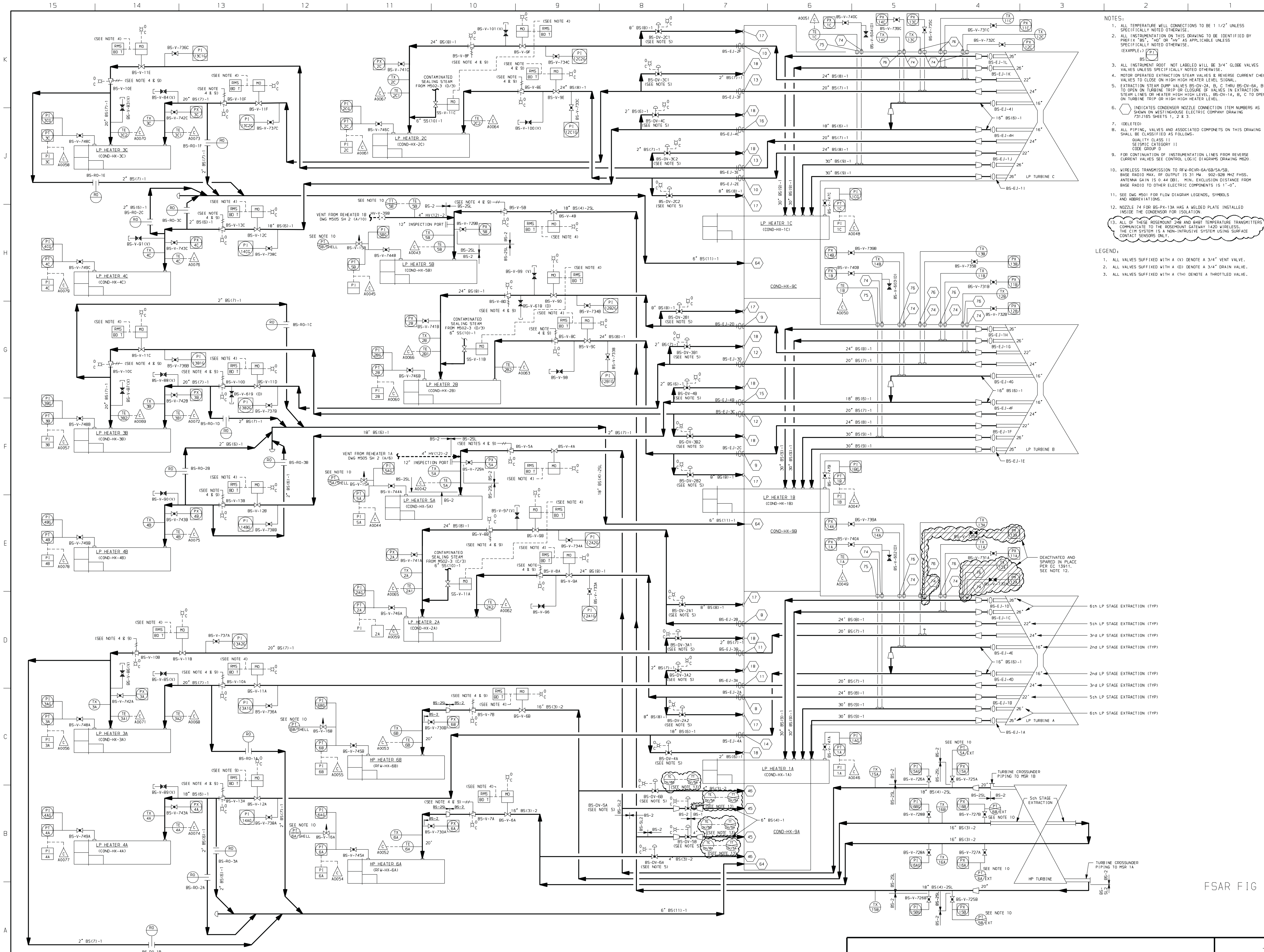
Columbia Generating Station
Final Safety Analysis Report

Flow Diagram - Condensate and Feedwater Systems

Draw. No. M504-3 | Rev. 1 | Figure 10.4-6.3

Form No. 960690ai

Amendment 64
December 2017

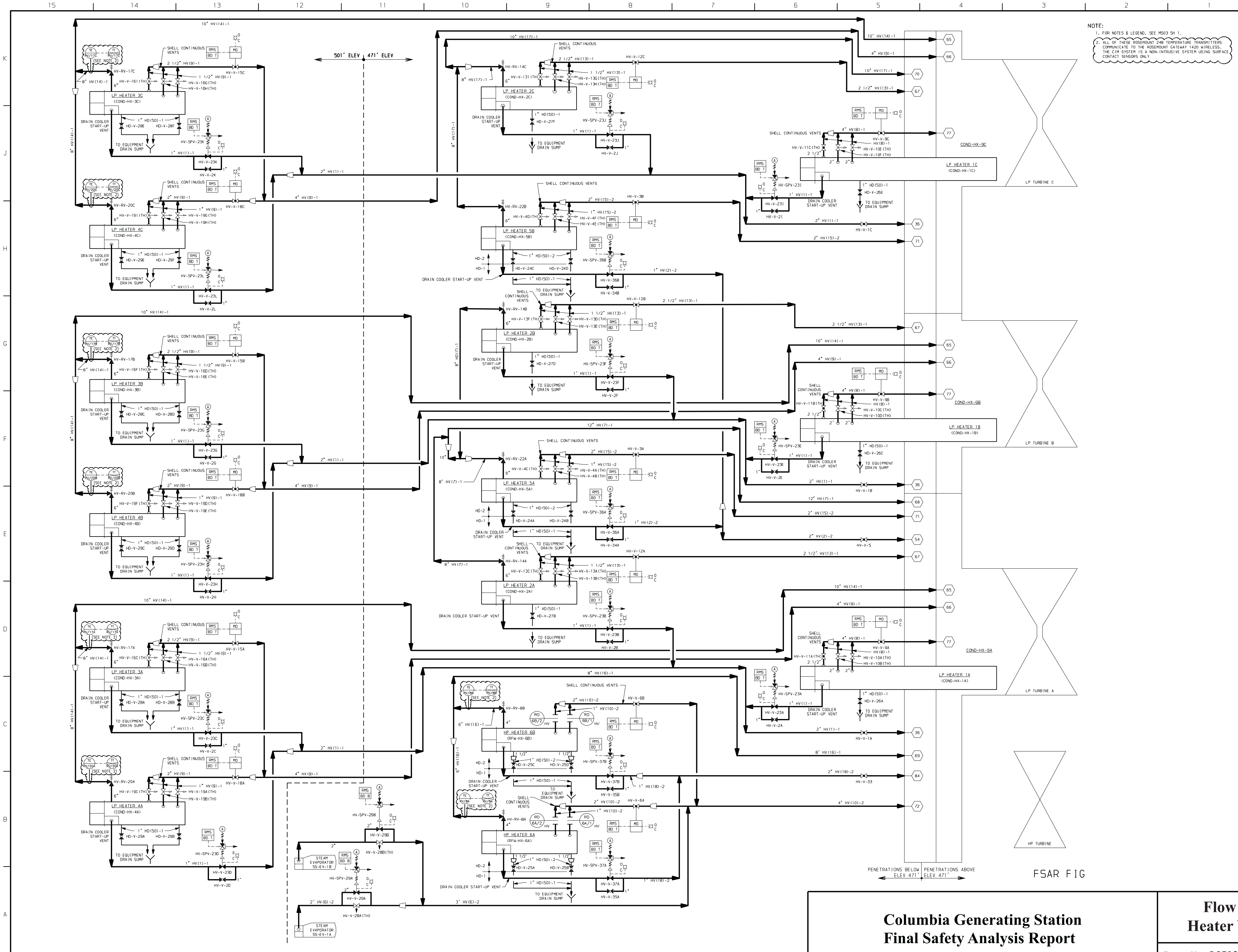


Columbia Generating Station
Final Safety Analysis Report

Flow Diagram - Extraction Steam and Heater Vents - Turbine Generator Building

Draw. No. M503-1	Rev. 16	Figure 10.4-7.1

Form No. 960690ai

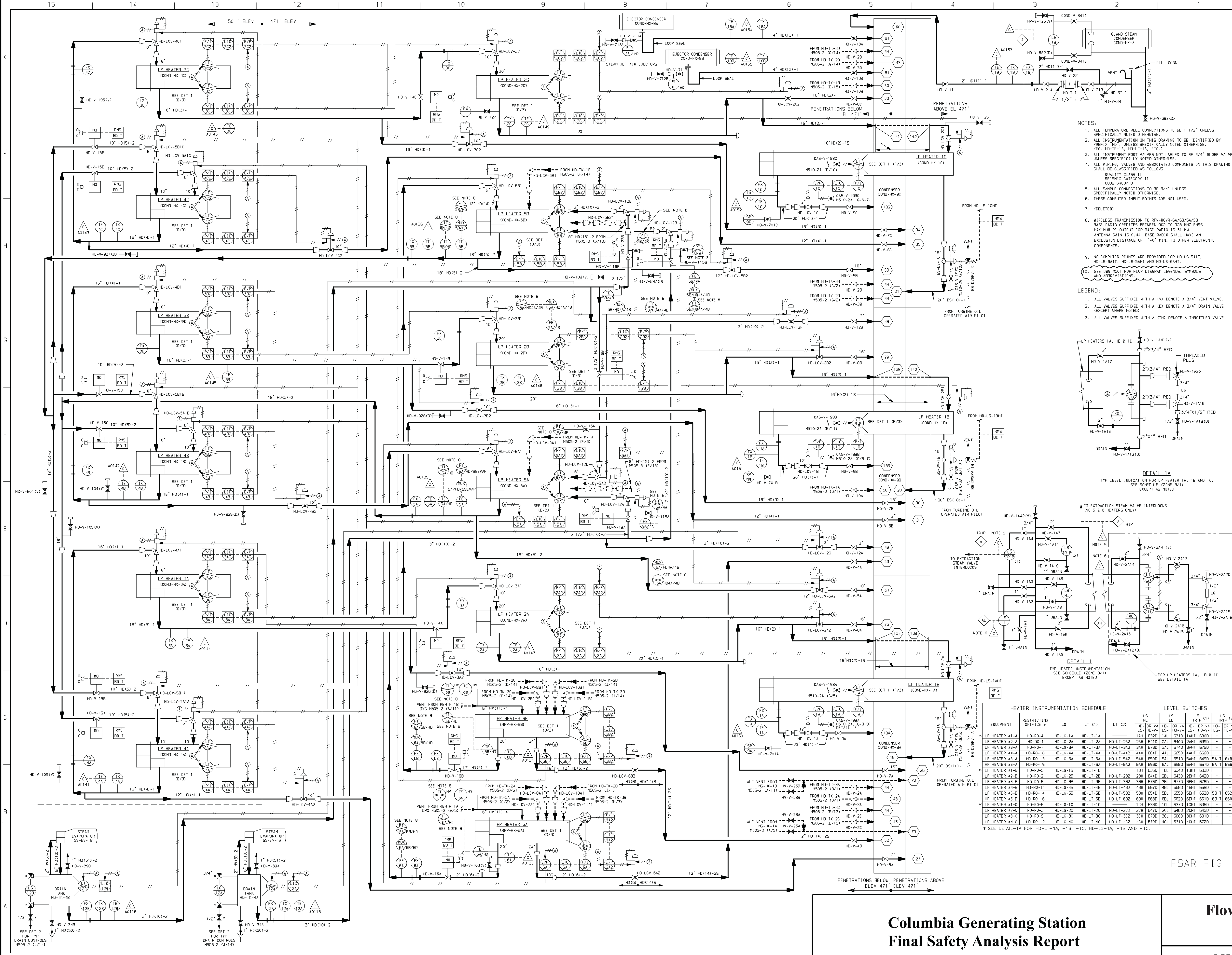


Columbia Generating Station
Final Safety Analysis Report

Flow Diagram - Extraction Steam and Heater Vents - Turbine Generator Building

Draw. No. M503-2	Rev. 5	Figure 10.4-7.2

Amendment 63
December 2015



NOTES:

1. ALL TEMPERATURE WELL CONNECTIONS TO BE 1 1/2" UNLESS SPECIFICALLY NOTED OTHERWISE.
2. ALL INSTRUMENTATION ON THIS DRAWING TO BE IDENTIFIED BY PREFIX "HD", UNLESS SPECIFICALLY NOTED OTHERWISE. (EG. HD-TE-1A, HD-LT-1A, ETC.)
3. ALL INSTRUMENT ROOT VALVES NOT LABLED TO BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
4. ALL PIPING, VALVES AND ASSOCIATED COMPONETS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP D
5. ALL SAMPLE CONNECTIONS TO BE 3/4" UNLESS SPECIFICALLY NOTED OTHERWISE.
6. THESE COMPUTER INPUT POINTS ARE NOT USED.
7. (DELETED)
8. WIRELESS TRANSMISSION TO RFW-RCVR-6A/6B/5A/5B BASE RADIO OPERATES BETWEEN 902 TO 928 MHZ FHSS. MAXIMUM RF OUTPUT FOR BASE RADIO IS 31 MW. ANTENNA GAIN IS 0.44 BASE RADIO SHALL HAVE AN EXCLUSION DISTANCE OF 1'-0" MIN. TO OTHER ELECTRONIC COMPONENTS.
9. NO COMPUTER POINTS ARE PROVIDED FOR HD-LS-5AIT, HD-LS-6AIT, HD-LS-5AHT AND HD-LS-6AHT.
10. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.

LEGEND:

1. ALL VALVES SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE. (EXCEPT WHERE NOTED)
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

DETAIL 1A

TYP LEVEL INDICATION FOR LP HEATER 1A, 1B AND 1C. SEE SCHEDULE (ZONE B/1) EXCEPT AS NOTED

DETAIL 1

TYP HEATER INSTRUMENTATION SEE SCHEDULE (ZONE B/1) EXCEPT AS NOTED

HEATER INSTRUMENTATION SCHEDULE					LEVEL SWITCHES							
EQUIPMENT	RESTRICTING ORIFICE #	LG	LT (1)	LT (2)	LS HL		LS LL		LS TRIP (1)		LS TRIP (2)	
					HD-LS-	DR VA HD-V-	HD-LS-	DR VA HD-V-	HD-LS-	DR VA HD-V-	HD-LS-	DR VA HD-V-
* LP HEATER #1-A	HD-RO-4	HD-LG-1A	HD-LT-1A	———	1AH	632D	1AL	631D	1AHT	630D	-	-
LP HEATER #2-A	HD-RO-1	HD-LG-2A	HD-LT-2A	HD-LT-2A2	2AH	641D	2AL	640D	2AHT	639D	-	-
LP HEATER #3-A	HD-RO-7	HD-LG-3A	HD-LT-3A	HD-LT-3A2	3AH	673D	3AL	674D	3AHT	675D	-	-
LP HEATER #4-A	HD-RO-10	HD-LG-4A	HD-LT-4A	HD-LT-4A2	4AH	664D	4AL	665D	4AHT	666D	-	-
LP HEATER #5-A	HD-RO-13	HD-LG-5A	HD-LT-5A	HD-LT-5A2	5AH	650D	5AL	651D	5AHT	649D	5AIT	648D
HP HEATER #6-A	HD-RO-15		HD-LT-6A	HD-LT-6A2	6AH	659D	6AL	658D	6AHT	657D	6AIT	656D
* LP HEATER #1-B	HD-RO-5	HD-LG-1B	HD-LT-1B	———	1BH	635D	1BL	634D	1BHT	633D	-	-
LP HEATER #2-B	HD-RO-2	HD-LG-2B	HD-LT-2B	HD-LT-2B2	2BH	644D	2BL	643D	2BHT	642D	-	-
LP HEATER #3-B	HD-RO-8	HD-LG-3B	HD-LT-3B	HD-LT-3B2	3BH	676D	3BL	677D	3BHT	678D	-	-
LP HEATER #4-B	HD-RO-11	HD-LG-4B	HD-LT-4B	HD-LT-4B2	4BH	667D	4BL	668D	4BHT	669D	-	-
LP HEATER #5-B	HD-RO-14	HD-LG-5B	HD-LT-5B	HD-LT-5B2	5BH	654D	5BL	655D	5BHT	653D	5BIT	652D
HP HEATER #6-B	HD-RO-16		HD-LT-6B	HD-LT-6B2	6BH	663D	6BL	662D	6BHT	661D	6BIT	660D
* LP HEATER #1-C	HD-RO-6	HD-LG-1C	HD-LT-1C	———	1CH	638D	1CL	637D	1CHT	636D	-	-
LP HEATER #2-C	HD-RO-3	HD-LG-2C	HD-LT-2C	HD-LT-2C2	2CH	647D	2CL	646D	2CHT	645D	-	-
LP HEATER #3-C	HD-RO-9	HD-LG-3C	HD-LT-3C	HD-LT-3C2	3CH	679D	3CL	680D	3CHT	681D	-	-
LP HEATER #4-C	HD-RO-12	HD-LG-4C	HD-LT-4C	HD-LT-4C2	4CH	670D	4CL	671D	4CHT	672D	-	-

* SEE DETAIL-1A FOR HD-LT-1A, -1B, -1C, HD-LG-1A, -1B AND -1C.

FSAR FIG

Columbia Generating Station
Final Safety Analysis Report

Flow Diagram - Heater Drain System - Turbine Generator Building

Draw. No. M505-1	Rev. 20	Figure 10.4-8.1

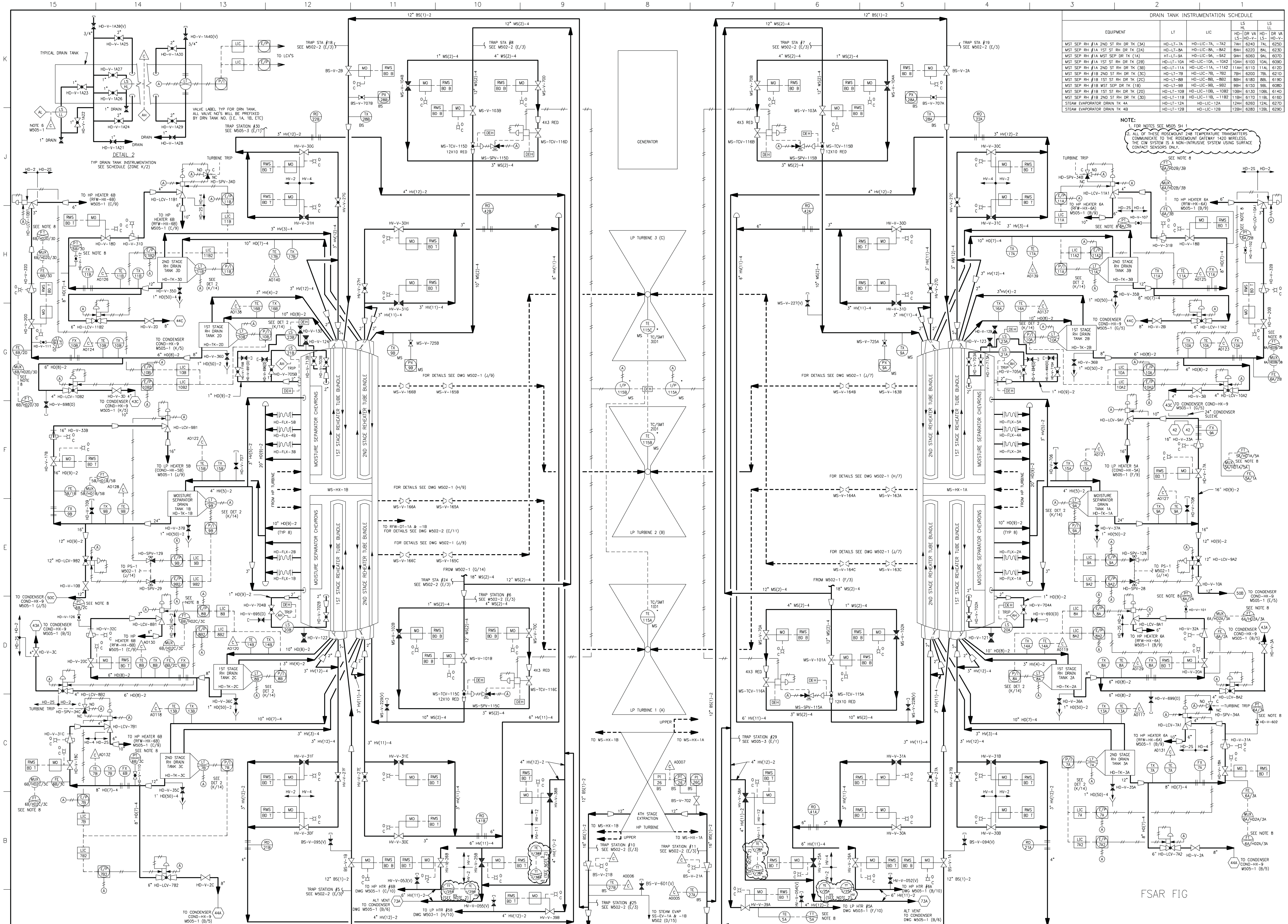
Form No. 960690ai

Amendment 64
December 2017

DRAIN TANK INSTRUMENTATION SCHEDULE

EQUIPMENT	LT	LIC	LS HL HD-LS-	LS HL DR VA HD-V-	LS LL HD-LS-	LS LL DR VA HD-V-
MST SEP RH #1A 2ND ST RH DR TK (3A)	HD-LT-7A	HD-LIC-7A, -7A2	7AH	624D	7AL	625D
MST SEP RH #1A 1ST ST RH DR TK (2A)	HD-LT-8A	HD-LIC-8A, -8A2	8AH	622D	8AL	623D
MST SEP RH #1A MST SEP DR TK (1A)	HT-LT-9A	HD-LIC-9A, -9A2	9AH	606D	9AL	607D
MST SEP RH #1A 1ST ST RH DR TK (2B)	HD-LT-10A	HD-LIC-10A, -10A2	10AH	610D	10AL	609D
MST SEP RH #1A 2ND ST RH DR TK (3B)	HD-LT-11A	HD-LIC-11A, -11A2	11AH	611D	11AL	612D
MST SEP RH #1B 2ND ST RH DR TK (3C)	HD-LT-7B	HD-LIC-7B, -7B2	7BH	620D	7BL	621D
MST SEP RH #1B 1ST ST RH DR TK (2C)	HD-LT-8B	HD-LIC-8B, -8B2	8BH	618D	8BL	619D
MST SEP RH #1B MST SEP DR TK (1B)	HD-LT-9B	HD-LIC-9B, -9B2	9BH	615D	9BL	608D
MST SEP RH #1B 1ST ST RH DR TK (2D)	HD-LT-10B	HD-LIC-10B, -10B2	10BH	613D	10BL	614D
MST SEP RH #1B 2ND ST RH DR TK (3D)	HD-LT-11B	HD-LIC-11B, -11B2	11BH	617D	11BL	616D
STEAM EVAPORATOR DRAIN TK 4A	HD-LT-12A	HD-LIC-12A	12AH	626D	12AL	627D
STEAM EVAPORATOR DRAIN TK 4B	HD-LT-12B	HD-LIC-12B	12BH	628D	12BL	629D

NOTE:
1. FOR NOTES SEE M505 SH 1
2. ALL OF THESE ROSEMOUNT 248 TEMPERATURE TRANSMITTERS COMMUNICATE TO THE ROSEMOUNT GATEWAY 1420 WIRELESS. THE CIM SYSTEM IS A NON-INTRUSIVE SYSTEM USING SURFACE CONTACT SENSORS ONLY.



Columbia Generating Station
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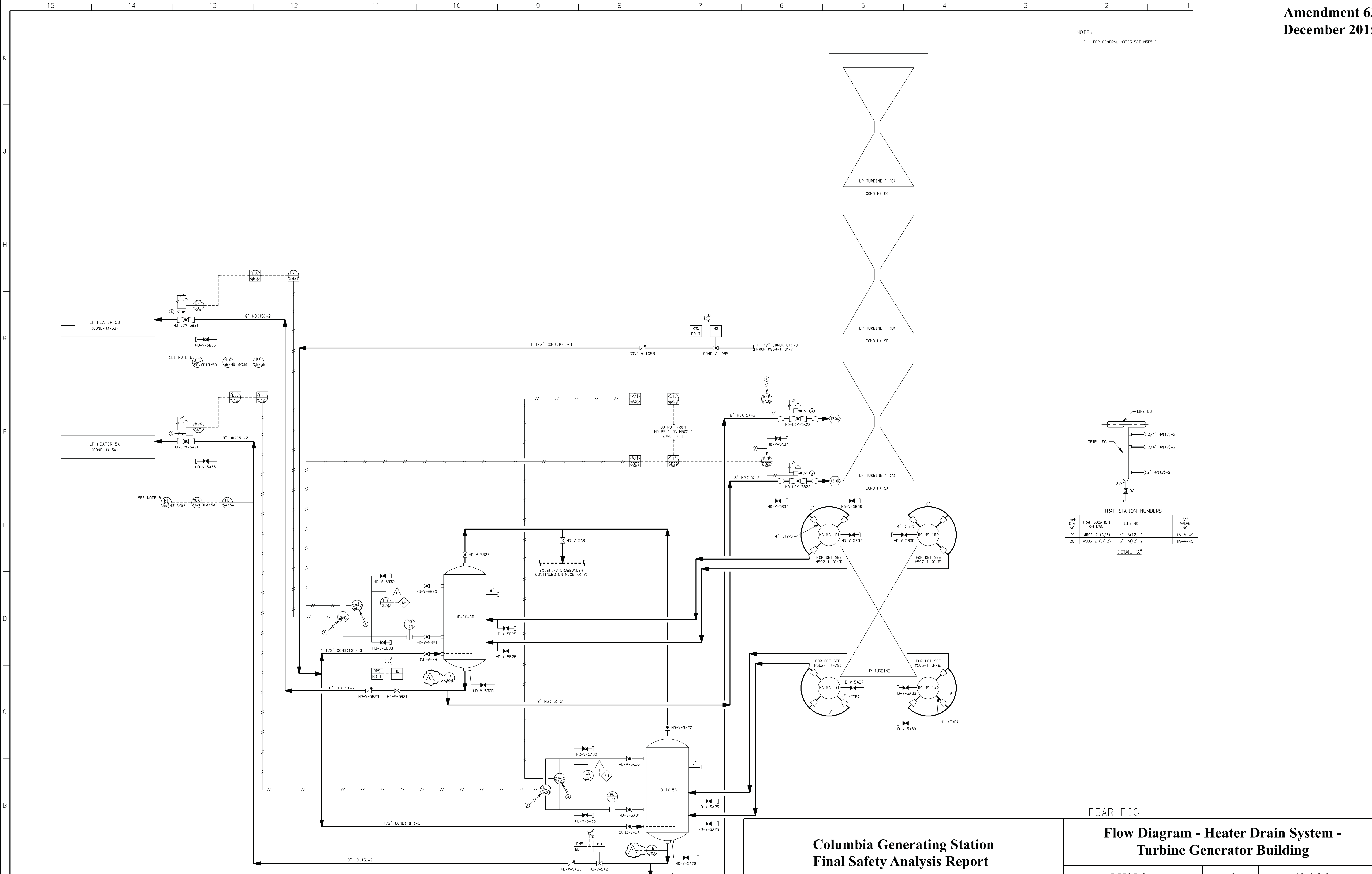
Flow Diagram – Heater Drain System – Turbine Generator Building

Draw. No. M505-2	Rev. 15	Figure 10.4-8.2

Form No. 960690ai

Amendment 63
December 2015

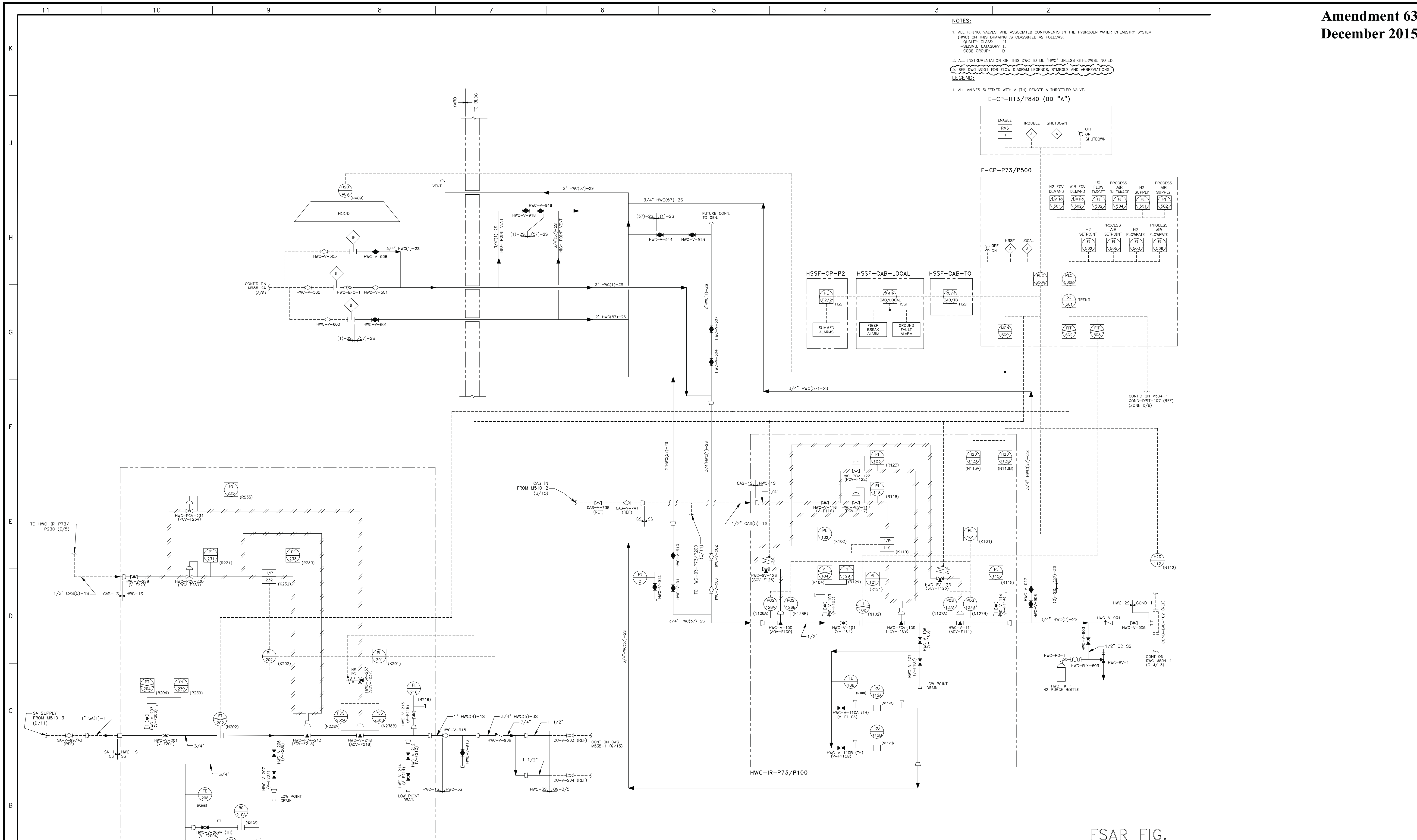


TRAP STA NO	TRAP LOCATION ON DWG	LINE NO	"A" VALVE NO
29	M505-2 (C/7)	4" HV(12)-2	HV-V-49
30	M505-2 (J/13)	3" HV(12)-2	HV-V-45

Columbia Generating Station
Final Safety Analysis Report

Flow Diagram - Heater Drain System - Turbine Generator Building

Draw. No. M505-3 | Rev. 9 | Figure 10.4-8.3



Columbia Generating Station
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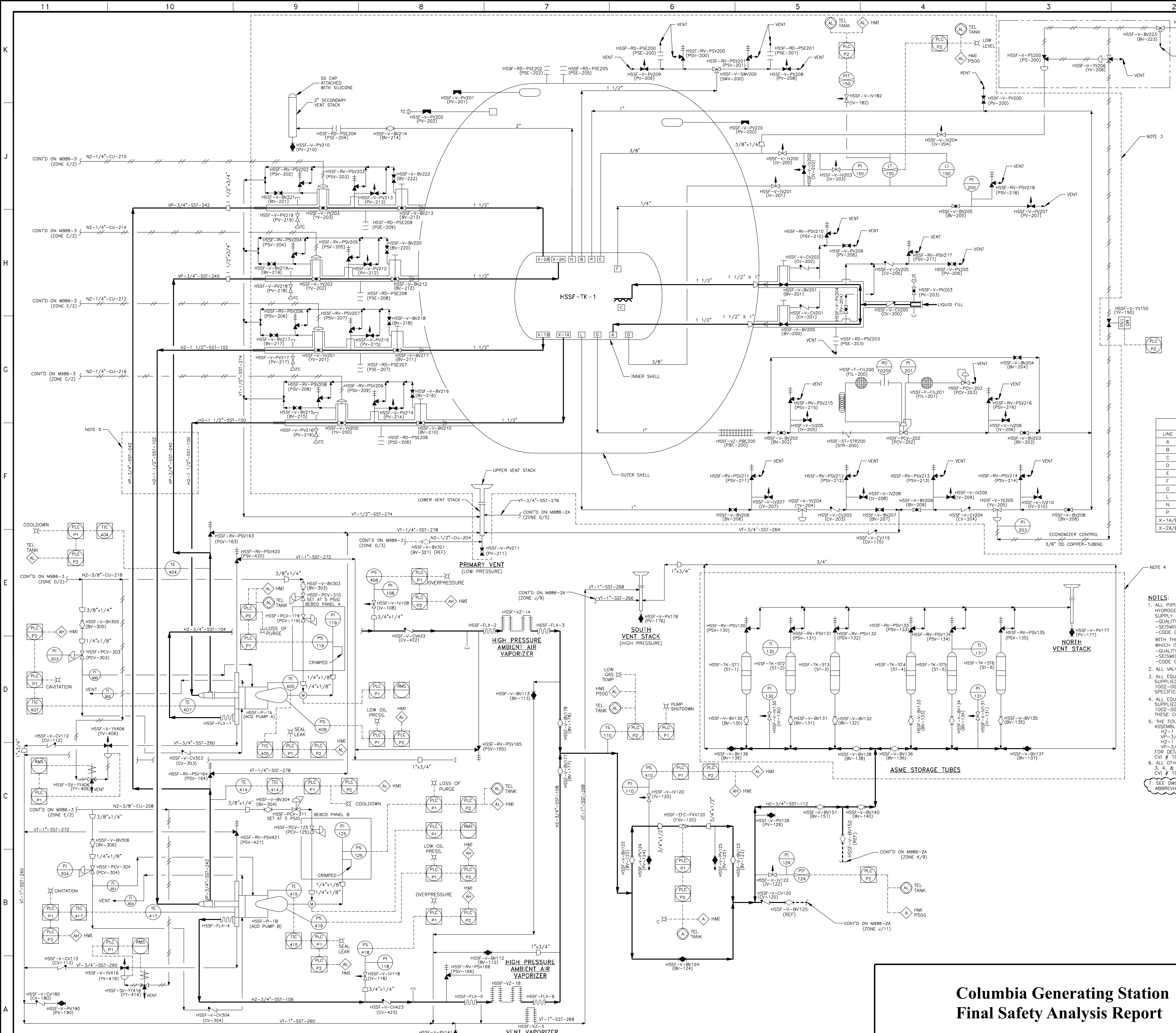
Flow Diagram - Hydrogen Water Chemistry Turbine Generator Building and Hydrogen Storage and Supply Facility

Draw. No. M986-1	Rev. 5	Figure 10.4-9.1

Form No. 960690ai



INTERNAL LINE SIZES		
LINE	SIZE	FUNCTION
A	1 1 /2" NPS	BOTTOM FILL
B	1 1 /2" NPS	VENT
C	1 1 /2" NPS	TOP FILL
D	3/8" ODT	BOTTOM GAGE LINE
E	3/8" ODT	TOP GAGE LINE
F	1/4" NPS	FULL TRYCOCK
G	1" NPS	PRESSURE BUILD LIQUID
L	1" NPS	LIQUID DRAW
N	2" NPS	AUXILIARY VENT LINE
P	1" NPS	PRESSURE BUILD VAPORS
X-1A/B	1 1 /2" NPS	PUMP LIQUID LINE (2 LINES)
X-2A/B	1 1 /2" NPS	PUMP VAPOR LINE (2 LINES)

NOTES:

1. ALL PIPING, VALVES AND ASSOCIATED COMPONENTS IN THE HYDROGEN WATER CHEMISTRY SYSTEM (HWC) AND THE HYDROGEN SUPPLY STORAGE FACILITY (HSSF) ON THIS DRAWING ARE
   - -QUALITY CLASS: II
   - -SEISMIC CATEGORY: II
   - -CODE GROUP: D

   WITH THE EXCEPTION OF THE HYDROGEN TANK (HSSF-TK-1) WHICH IS
   - -QUALITY CLASS: II+
   - -SEISMIC CATEGORY: I
   - -CODE GROUP: D
2. ALL VALVES AND INSTRUMENTS HSSF UNLESS OTHERWISE NOTED.
3. ALL EQUIPMENT AND INTERCONNECTING PIPING WITHIN THIS BOX SUPPLIED AND DESIGNED BY CHART MANUFACTURING. SEE CVI # 1002-00,102 AND CVI # 1002-00,76 FOR BILL OF MATERIALS AND SPECIFICATIONS FOR THESE COMPONENTS AND PIPING.
4. ALL EQUIPMENT AND INTERCONNECTING PIPING WITHIN THIS BOX SUPPLIED AND DESIGNED BY ELEET CRYOGENICS INC. SEE CVI # 1002-00,86 FOR BILL OF MATERIALS AND SPECIFICATIONS FOR THESE COMPONENTS AND PIPING.
5. THE FOLLOWING LINE NUMBERS ARE ACTUALLY DUAL FLEX HOSE ASSEMBLIES BUILT BY CPC/CRYOLABS.
   H2-1 1/2"-SST-100
   VP-3/4"-SST-240
   H2-1 1/2"-SST-102
   VP-3/4"-SST-242
   FOR DETAILS ON THESE FLEX HOSE ASSEMBLIES SEE CVI # 1002-00,109.
6. ALL OTHER EQUIPMENT ON THIS DRAWING, NOT SPECIFIED BY NOTES 3, 4, & 5 MEETS THE REQUIREMENTS AS SPECIFIED IN CVI # 1002-00,7 AND CVI # 1002-00,8.
7. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.

Columbia Generating Station
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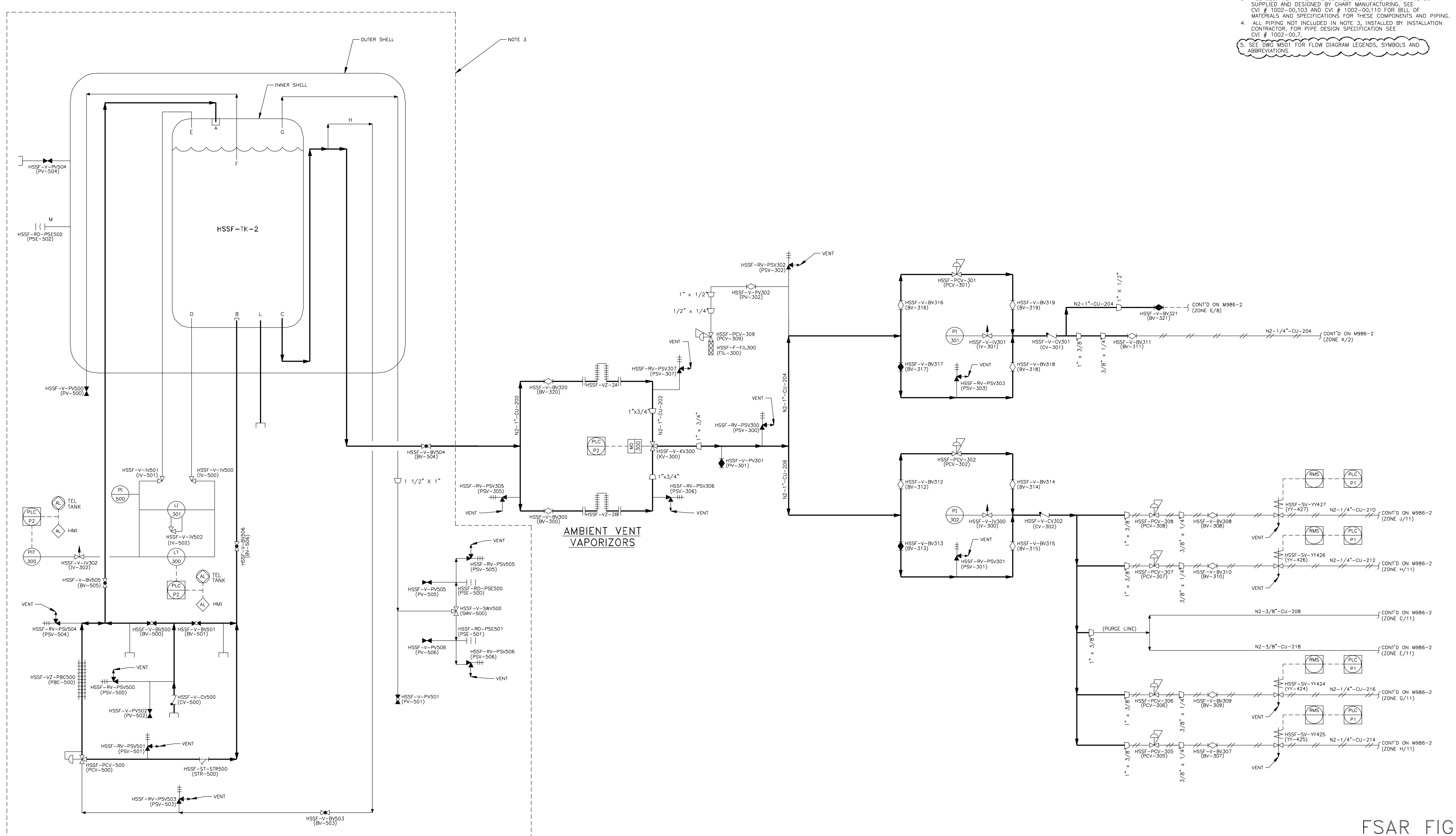
Flow Diagram - Hydrogen Water Chemistry Turbine Generator Building and Hydrogen Storage and Supply Facility

Draw. No. M986-2	Rev. 3	Figure 10.4-9.2

Form No. 960690ai

NOTES:

1. ALL PIPING, VALVES AND ASSOCIATED COMPONENTS IN THE HYDROGEN WATER CHEMISTRY SYSTEM (HWC) AND THE HYDROGEN SUPPLY STORAGE FACILITY (HSSF) ON THIS DRAWING ARE
   - QUALITY CLASS: II
   - SEISMIC CATEGORY: II
   - CODE GROUP: D
2. ALL VALVES AND INSTRUMENTS HSSF UNLESS OTHERWISE NOTED.
3. ALL EQUIPMENT AND INTERCONNECTING PIPING WITHIN THIS BOX SUPPLIED AND DESIGNED BY CHART MANUFACTURING. SEE CVI # 1002-00,103 AND CVI # 1002-00,110 FOR BILL OF MATERIALS AND SPECIFICATIONS FOR THESE COMPONENTS AND PIPING.
4. ALL PIPING NOT INCLUDED IN NOTE 3, INSTALLED BY INSTALLATION CONTRACTOR, FOR PIPE DESIGN SPECIFICATION SEE CVI # 1002-00.7.
5. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.



INNER VESSEL CONNECTIONS	
A	TOP FILL
B	BOTTOM FILL
C	LIQUID DRAW DIP TUBE
D	BOTTOM GAUGE LINE
E	TOP GAUGE LINE
F	FULL TRYCOCK
G	VENT
H	ECONOMIZER LINE
L	BOTTOM WITHDRAWAL LIQUID LINE
M	EVACUATION LINE

FSAR FIG.

Columbia Generating Station
Final Safety Analysis Report

Flow Diagram - Hydrogen Water Chemistry Turbine Generator Building and Hydrogen Storage and Supply Facility

Draw. No. M986-3	Rev. 3	Figure 10.4-9.3

Chapter 11

RADIOACTIVE WASTE MANAGEMENT

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RADIOACTIVE WASTE MANAGEMENT

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Chapter 11

RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

The italicized information is historical and was provided to support the application for an operating license.

Radioactive material sources (activation products and fission product release from fuel) have been evaluated in operating boiling water reactors (BWRs) over the past decade. These source terms are reviewed and periodically revised to incorporate up-to-date information. Release of radioactive material from operating BWRs has resulted in doses to offsite persons which have been only a small fraction of 10 CFR 20 permissible or natural background doses.

The information provided in this section defines the design-basis radioactive material levels in the reactor water, steam, and offgas. The various radioisotopes listed have been grouped as coolant activation products, noncoolant activation products, and fission products. The fission product levels are based on measurements of BWR reactor water and offgas at several stations through mid-1971. Emphasis was placed on observations made at Kernkraftwerk RWE-Bayernwerk GmbH (KRB) and Dresden-2. The design-basis radioactive material levels do not necessarily include all the radioisotopes observed or predicted theoretically to be present. The radioisotopes included are considered significant to one or more of the following criteria:

- *a. Plant equipment design,*
- *b. Shielding design,*
- *c. Understanding system operation and performance,*
- *d. Measurement practicability, and*
- *e. Evaluating radioactive material releases to the environment.*

The inventory of radionuclides used to determine shielding requirements of system components are discussed in Chapter 12.

11.1.1 FISSION PRODUCTS

11.1.1.1 Noble Radiogas Fission Products

The noble gas radionuclide fission product source terms observed in operating BWRs are generally complex mixtures whose sources vary from miniscule defects in cladding to "tramp" uranium on external cladding surfaces. The relative concentrations or amounts of noble radiogas can be described as follows.

Equilibrium: $R_{g,i} = K_1 Y_1$ *(11.1-1)*

Recoil: $R_{g,i} = K_2 Y_1 \lambda_i$ *(11.1-2)*

The nomenclature in Section 11.1.1.4 defines the terms in these and succeeding equations. The constants K_1 and K_2 describe the fractions of the total fissions that are involved in each of the releases. The equilibrium and recoil mixtures are the two extremes of the mixture spectrum that are physically possible. When a sufficient time delay occurs between the fission event and the time of release of the radiogases from the fuel to the coolant, the radiogases approach equilibrium levels in the fuel and the equilibrium mixture results. Where there is no delay between the fission event and the release of the radiogases, the recoil mixture is observed. Prior to Vallecitos Boiling Water Reactor (VBWR) and Dresden-1 experience, it was assumed that noble radiogas leakage from the fuel would be the equilibrium mixture of the noble radiogases present in the fuel.

The VBWR and early Dresden-1 experience (Reference 11.1-1) indicated that the actual mixture most often observed approached a distribution which was intermediate in character to the two extremes. This intermediate decay mixture was termed the "diffusion" mixture. It must be emphasized that this "diffusion" mixture is merely one possible point on the mixture spectrum ranging from the equilibrium to the recoil mixture and does not have the absolute mathematical and mechanistic basis for the calculational methods possible for equilibrium and recoil mixtures. However, the "diffusion" distribution pattern which has been described is as follows:

Diffusion: $R_{g,i} = K_3 Y_i \lambda_i^{0.5}$ *(11.1-3)*

The constant K_3 describes the fraction of total fissions that are involved in the release. The value of the exponent of the decay constant, λ_i, is midway between the values for the equilibrium case, 0, and recoil case 1. The "diffusion" pattern value of 0.5 was originally derived from diffusion theory.

Although the previously described "diffusion" mixture was used by GE as a basis for design since 1963, the design-basis release magnitude used has varied from 0.5 Ci/sec to 0.1 Ci/sec as measured after 30-minute decay (t=30 minutes). Since about 1967, the design-basis release magnitude used (including the 1971 source terms) was established at an annual average of 0.1 Ci/sec (t = 30 minutes). This design basis is considered as an annual average with some time above and some time below this value. This design value was selected on the basis of operating experience rather than predictive assumptions. Several judgment factors,*

**The noble radiogas source term rate after 30-minute decay has been used as a conventional measure of the design-basis fuel leakage rate since it is conveniently measurable and was consistent with the nominal design-basis 30-minute offgas holdup system used on a number of previous plants.*

including the significance of environmental release, reactor water radioisotope concentrations, liquid waste handling and effluent disposal criteria, building air contamination, shielding design, and turbine and other component contamination affecting maintenance, have been considered in establishing this level.

Noble radiogas source terms from fuel above 0.1 Ci/sec (t = 30 minutes) can be tolerated for reasonable periods of time. Continual assessment of these values is made on the basis of actual operating experience in BWRs (References 11.1-2 and 11.1-3).

While the noble radiogas source-term magnitude was established at 0.1 Ci/sec (t = 30 minutes), it was recognized that there may be a more statistically applicable distribution for the noble radiogas mixture. Sufficient data were available from KRB operations from 1967 to mid-1971 along with Dresden-2 data from operation in 1970 and several months in 1971 to more accurately characterize the noble radiogas mixture pattern for an operating BWR.

The basic equation for each radioisotope used to analyze the collected data is:

$$R_{g,i} = K_g Y_i \lambda_i^{m}\left(1 - e^{-\lambda_i T}\right)\left(e^{-\lambda_i t}\right) \qquad (11.1\text{-}4)$$

With the exception of Kr-85 with a half-life of 10.74 years, the noble radiogas fission products in the fuel are essentially at an equilibrium condition after an irradiation period of several months (rate of formation is equal to the rate of decay). So for practical purposes the term $(1 - e^{-\lambda_i T})$ approaches 1 and can be neglected when the reactor has been operating steady state for long periods of time. The term $(1 - e^{-\lambda_i T})$ is used to adjust the releases from the fuel (t = 0) to the decay time for which values are needed. Historically, t = 30 minutes has been used. When discussing long steady-state operation and leakage from the fuel (t = 0), the following simplified form of Equation 11.1-4 can be used to describe the leakage of each noble radiogas:

$$R_{g,i} = K_g Y_i \lambda_i^{m} \qquad (11.1\text{-}5)$$

The constant, K_g, describes the magnitude of leakage from the fuel. The relative rates of leakage of the different noble radiogas isotopes is accounted for by the variable, m, the exponent of the decay constant, λ_i.

Dividing both sides of Equation 11.1-5 by y_i, the fission yield, and taking the logarithm of both sides results in the following equation:

$$\log\left(R_{g,i} / Y_i\right) = m \log\left(\lambda_i\right) + \log\left(K_g\right) \qquad (11.1\text{-}6)$$

Equation 11.1-6 represents a straight line when log ($R_{g,i}/y_i$) is plotted vs. log (λ_i); m is the slope of the line. This straight line is obtained by plotting ($R_{g,i}/y_i$) vs. (λ_i) on logarithmic graph paper. By fitting actual data from KRB and Dresden-2 (using least squares techniques) to the equation, the slope, m, can be obtained. This can be estimated on the plotted graph. With radiogas leakage at KRB over the nearly 5-year period varying from 0.001 to 0.056 Ci/sec (t = 30 minutes) and with radiogas leakage at Dresden-2 varying from 0.001 to 0.169 Ci/sec (t = 30 minutes), the average value of m was determined. The value for m is 0.4 with a standard deviation of ±0.07. This is illustrated in Figure 11.1-1 as a frequency histogram. As can be seen from this figure, variations in m were observed in the range of m = 0.1 to m = 0.6. After establishing the value of m = 0.4, the value of K_g can be calculated by selecting a value for R_g, or as has been done historically, the design basis is set by the total design-basis source-term magnitude at t = 30 minutes. With SR_g at 30 minutes = 100,000 mCi/sec, K_g can be calculated as being 2.6×10^7 and Equation 11.1-4 becomes:

$$R_{g,i} = 2.6x10^7 \ Y_i \lambda_i^{\ 0.4}\left(1 - e^{-\lambda_i t}\right)\left(e^{-\lambda_i t}\right) \qquad (11.1\text{-}7)$$

This updated noble radiogas source-term mixture has been termed the "1971 mixture" to differentiate it from the "diffusion" mixture. The noble gas source term for each radioisotope can be calculated from Equation 11.1-7. The resultant source terms are presented in Table 11.1-1 as leakage from fuel (t = 0) and after 30-minute decay. While Kr-85 can be calculated using Equation 11.1-7, the number of confirming experimental observations was limited by the difficulty of measuring very low release rates of this isotope. Therefore, the table provides an estimated range for ^{85}Kr based on a few actual measurements.

11.1.1.2 Radiohalogen Fission Products

Historically, the radiohalogen design-basis source term was established by the same equation as that used for noble radiogases. In a fashion similar to that used with gases, a simplified equation can be shown to describe the release of each radiohalogen:

$$R_{h,i} = K_h Y_i \lambda_i^{\,n} \qquad (11.1\text{-}8)$$

The constant, K_h, describes the magnitude of leakage from fuel. The relative rates of radiohalogen leakage is expressed in terms of n, the exponent of the decay constant, λ_i. As was done with the noble radiogases, the average value was determined for n. The value for n is 0.5 with a standard deviation of ±0.19. This is illustrated in Figure 11.1-2 as a frequency histogram. As can be seen from this figure, variations in n were observed in the range of n = 0.1 to n = 0.9.

It appeared that the use of the previous method of calculating radiohalogen leakage from fuel was overly conservative. Figure 11.1-3 relates KRB and Dresden-2 noble radiogas versus I-131 leakage. While it can be seen from Dresden-2 data, during the period August 1970 to January 1971, that there is a relationship between noble radiogas and I-131 leakage under one fuel condition, there was no simple relationship for all fuel conditions experienced. Also, it can be seen that during this period, high radiogas leakages were not accompanied by high radioiodine leakage from the fuel. Except for one KRB datum point, all steady-state I-131 leakages observed at KRB or Dresden-2 were equal to or less than 505 μCi/sec. Even at Dresden-1 in March 1965, when severe defects were experienced in stainless steel clad fuel, I-131 leakages greater than 500 μCi/sec were not experienced. Figure 11.1-3 shows that these higher radioiodine leakages from the fuel were related to noble radiogas source terms of less than the design-basis value of 0.1 Ci/sec (t = 30 minutes). This may be partially explained by inherent limitations due to internal plant operational problems that caused plant derating.

In general, it would not be anticipated that operation at full power would continue for any significant time period with fuel cladding defects, which would be indicated by ^{131}I leakage from the fuel in excess of 700 μCi/sec. When high radiohalogen leakages are observed, other fission products will be present in greater amounts.

Using these judgment factors and experience to date, the design-basis radiohalogen source terms from fuel were established based on ^{131}I leakage of 700 μCi/sec. This value, as seen in Figure 11.1-3, accommodates the experience data and the design-basis noble radiogas source term of 0.1 Ci/sec (t = 30 minutes). With the I-131 design-basis source-term established, K_h can be calculated as being 2.4×10^7 and radiohalogen release can be expressed by the following equation:

$$R_{h,i} = 2.4x10^7 \; Y_i \lambda_i^{0.5} \left(1 - e^{-\lambda_i T}\right)\left(e^{-\lambda_i t}\right) \qquad (11.1\text{-}9)$$

Concentrations of radiohalogens in reactor water can be calculated using the following equation:

$$C_{h,i} = \frac{R_{h,i}}{(\lambda_i + \beta + \gamma)M} \qquad (11.1\text{-}10)$$

The observed "carryover" for radiohalogens has varied from 0.1% to about 2% on newer plants. The average of observed radiohalogen carryover measurements has been 1.2% by weight of reactor water in steam with a standard deviation of ±0.9%. In the present source term definition, a radiohalogen carryover of 2% (0.02 fraction) was used.

The radiohalogen release rate from the fuel was calculated from equation 11.1-9.

Concentrations in reactor water were calculated from equation 11.1-10.

The resultant concentrations are presented in Table 11.1-2. Radiohalogens with half-lives less than 3 minutes were omitted.

11.1.1.3 Other Fission Products

The observations of other fission products (and transuranic nuclides, including ^{239}Np) in operating BWRs are not adequately correlated by simple equations. For these radioisotopes, design-basis concentrations in reactor water have been estimated conservatively from experience data (Reference 11.1-8) and are presented in Table 11.1-3. Radioisotopes with half-lives less than 10 minutes were not considered. Carryover of these radioisotopes from the reactor water to the steam is estimated to be ≤0.1% (≤ 0.001 fraction) (Reference 11.1-8). In addition to carryover, however, decay of noble radiogases in the steam leaving the reactor will result in production of noble gas daughter radioisotopes in the steam and condensate systems.

Some daughter radioisotopes (e.g., yttrium and lanthanum) were not listed as being in reactor water. Their independent leakage to the coolant is negligible; however, these radioisotopes may be observed in some samples in equilibrium or approaching equilibrium with the parent radioisotope.

Except for ^{239}Np, trace concentrations of transuranic isotopes have been observed in only a few samples where extensive and complex analyses were carried out. The predominant alpha emitter present in reactor water is ^{242}Cm at an estimated concentration of 10^{-6} µCi/g or less, which is below the maximum permissible concentration in drinking water applicable to continuous use by the general public. The concentration of alpha-emitting plutonium radioisotopes is more than one order of magnitude lower than that of ^{242}Cm.

Plutonium-241 (a beta emitter) may also be present in concentrations comparable to the ^{242}Cm level.

11.1.1.4 Nomenclature

The following list defines the terms used in equations for source term calculations:

$R_{g,i}$ = *leakage rate of noble gas radioisotope i (µCi/sec)*

$R_{h,i}$ = *leakage rate of a halogen radioisotope i (µCi/sec)*

Y_i = *fission yield of a radioisotope i (atoms/fission)*

λ_I = *decay constant of a radioisotope i (sec^{-1})*

T = *fuel irradiation time (sec)*

t = *decay time following leakage from fuel (sec)*

m = *noble radiogas decay constant exponent (dimensionless)*

n = *radiohalogen decay constant exponent (dimensionless)*

K_g = *a constant establishing the level of noble radiogas leakage from fuel*

K_h = *a constant establishing the level of radiohalogen leakage from fuel*

$C_{h,i}$ = *concentration of a radiohalogen i in reactor water (μCi/g)*

M = *mass of water in the operating reactor (g)*

β = *cleanup system removal constant (sec^{-1})*

$$\beta = \frac{\text{cleanup system flow rate (g/sec)}}{M(g)}$$

g = *grams mass*

γ = *halogen steam carryover removal constant (sec^{-1})*

$$\gamma = \frac{\text{concentration of radiohalogen isotope in steam } (\mu Ci/g)}{C_{h,i}\,(\mu Ci/g)} \times \frac{\text{steam flow } (g/sec)}{M\,(g)}$$

11.1.2 ACTIVATION PRODUCTS

11.1.2.1 Coolant Activation Products

The coolant activation products are not adequately correlated by simple equations. Design basis concentrations in reactor water and steam have been estimated conservatively from experience data. The resultant concentrations are presented in Table 11.1-4.

11.1.2.2 Noncoolant Activation Products

The activation products formed by activation of impurities in the coolant or by corrosion of irradiated system materials are not adequately correlated by simple equations. The design-basis source terms of noncoolant activation products have been estimated conservatively

from experience data (Reference 11.1-8). The resultant concentrations are presented in Table 11.1-5. Carryover of these isotopes from the reactor water to the steam is estimated to be <0.1% (<0.001 fraction) (Reference 11.1-8).

11.1.2.3 Steam and Power Conversion System N-16 Inventory

The main steam and reactor feedwater systems sources are discussed in Section 12.2.1.2.2.7. This section discusses the N-16 source strength in the moisture separators and reheaters, main condenser and hotwell, feedwater heaters, and associated piping.

11.1.3 TRITIUM

In a BWR, tritium is produced by three principal methods:

a. *Activation of naturally occurring deuterium in the primary coolant,*
b. *Nuclear fission of UO_2 fuel, and*
c. *Neutron reactions with boron used in reactivity control rods.*

The tritium formed in control rods, which may be released from a BWR in liquid or gaseous effluents, is believed to be negligible. The prime source of tritium available for release from a BWR is that produced from activation of deuterium in the primary coolant. Some fission product tritium may also transfer from fuel to primary coolant. This discussion is limited to the uncertainties associated with estimating the amounts of tritium generated in a BWR which are available for release. All of the tritium produced by activation of deuterium in the primary coolant is available for release in liquid or gaseous effluents. The tritium formed in a BWR from deuterium activation is calculated using the equation

$$R_{act} = \frac{\Sigma \phi V \lambda}{3.7 x 10^4 P} \qquad (11.1\text{-}11)$$

where:

R_{act} = *tritium formation rate by deuterium activation act (μCi/sec/MWt)*

Σ = *macroscopic thermal neutron cross section (cm^{-1})*

ϕ = *thermal neutron flux (neutrons/cm^2-sec)*

V = *coolant volume in core (cm^3)*

λ = *tritium radioactive decay constant* (1.78×10^{-9} sec^{-1})

P = *reactor power level (MWt)*

For recent BWR designs, R_{act} *is calculated to be* $(1.3 \pm 0.4) \times 10^{-4}$ *µCi/sec/MWt. The uncertainty indicated is derived from the estimated errors in selecting values for the coolant volume in the core, coolant density in the core, abundance of deuterium in light water [some additional deuterium will be present because of the* $H(n,\gamma)$ *D reaction], thermal neutron flux, and microscopic cross section for deuterium.*

The fraction of tritium produced by fission which may transfer from the fuel to the coolant (which will then be available for release in liquid and gaseous effluents) is more difficult to estimate. However, since zircaloy-clad fuel rods are used in BWRs, essentially all fission product tritium will remain in the fuel rods unless defects are present in the cladding material (Reference 11.1-4).

The study made at Dresden-1 in 1968 by the U.S. Public Health Service (USPHS) suggests that essentially all of the tritium released from the plant could be accounted for by the deuterium activation source (Reference 11.1-3). For purposes of estimating the leakage of tritium from defective fuel, it is assumed that it leaks in a manner similar to the leakage of noble radiogases. Thus, use is made of the empirical relationship described as the "diffusion" mixture used for predicting the source term of individual noble radiogas isotopes as a function of the total noble gas source term. The equation which describes this relationship is:

$$R_{dif} = Ky\lambda^{0.5} \qquad (11.1\text{-}12)$$

where:

R_{dif} = *leakage rate of tritium from fuel (µCi/sec)*

y = *fission yield fraction (atoms/fission)*

λ = *radioactive decay constant* (sec^{-1})

K = *a constant related to total tritium leakage rate.*

When the total noble radiogas source term is 100,000 µCi/sec after 30-minutes decay, leakage from fuel is calculated to be about 0.24 µCi/sec of tritium. To place this value in perspective, in the USPHS study (Reference 11.1-3), the observed rate of ^{85}Kr *(which has a half-life similar to that of tritium) was 0.06 to 0.4 times that calculated using the "diffusion" mixture relationship. This would suggest that the actual tritium leakage rate might range from 0.015 to 0.10 µCi/sec. Since the annual average noble radiogas leakage from a BWR is expected to*

be less than 100,000 μCi/sec (t = 30-minutes), the annual average tritium release rate from the fission source is conservatively estimated at 0.12 ± 0.12 μCi/sec, or 0.0 to 0.24 μCi/sec. Based on this approach, the estimated total tritium appearance rate in reactor coolant and release rate in the effluent is about 19 Ci/yr.

Tritium formed in the reactor is present as tritiated oxide (HTO) and to a lesser degree as tritiated gas (HT). Tritium concentration on a weight basis in the steam formed in the reactor is the same as in the reactor water at any given time. This tritium concentration is also present in condensate and feedwater. Since radioactive effluents generally originate from the reactor and power cycle equipment, radioactive effluents also have this tritium concentration. The condensate storage tanks receive treated water from the liquid waste management systems and rejected water from the condensate system. Thus, all plant process water approaches a common tritium concentration.

Offgases released from the plant contain tritium, which is present as tritiated gas (HT) resulting from reactor water radiolysis as well as tritiated water vapor (HTO). In addition, water vapor from the turbine gland seal steam packing exhauster and some water vapor present in ventilation air due to process steam leaks and evaporation from sumps and tanks also contain tritium. The remainder of the tritium leaves the plant in liquid effluents or with solid wastes.

Recombination of radiolysis gases in the offgas system forms water, which is condensed and returned to the main condenser. This tends to reduce the amount of tritium leaving in gaseous effluents. Reducing the gaseous tritium release results in a slightly higher tritium concentration in the plant process water. Reducing the amount of liquid effluent discharged will also result in a higher process coolant tritium concentration.

Essentially all tritium in the primary coolant is eventually released to the environs, either as water vapor and gas to the atmosphere or as liquid effluent to the plant discharge or as solid waste. Reduction due to radioactive decay is negligible due to the 12.3-year half-life of tritium.

The USPHS study at Dresden-1 estimated that approximately 90% of the tritium release was observed in liquid effluent, with the remaining 10% leaving as gaseous effluent (Reference 11.1-5).

Efforts to reduce the volume of liquid effluent discharges may change this distribution so that a greater amount of tritium will leave as gaseous effluent. From a practical standpoint, the fraction of tritium leaving as liquid effluent may vary between 60% and 90% with the remainder leaving in gaseous effluent.

The amount of tritium released to the environment in liquid and gaseous effluents is based on the draft Regulatory Guide 1.CC, BWR-GALE code analyses. This is discussed in Sections 11.2 and 11.3 respectively.

11.1.4 FUEL FISSION PRODUCT INVENTORY AND FUEL EXPERIENCE

11.1.4.1 Fuel Fission Product Inventory

Fuel fission product inventory information is used in establishing fission product source terms for accident analysis and is, therefore, discussed in Chapter 15.

11.1.4.2 Fuel Experience

A discussion of BWR fuel experienced including fuel failure, burn-up and thermal conditions under which the experience was gained, is presented in References 11.1-2, 11.1-3, and 11.1-6.

11.1.5 RADIOACTIVITY LEAKAGE AND EFFLUENT SOURCES

Process leakage results in potential release paths for noble gases and other volatile fission products via ventilation systems. Liquids from process leaks are collected and routed to radioactive equipment and floor drain systems. Radioisotope releases via ventilation paths are at extremely low levels and have been insignificant compared to process offgas from operating BWR plants. However, because the implementation of improved process offgas treatment systems makes the ventilation release comparatively significant, measurements have been conducted to identify and quantify these low-level release paths. In addition an awareness of measurements by the Electric Power Research Institute, other organizations, and routine measurements by other utilities with operating BWRs has been maintained. Design basis estimates of the various liquid, gaseous, and solids effluents are discussed in Sections 11.2, 11.3, and 11.4 which follow.

Concurrently, analytical and mathematical model studies are being performed to provide a description of the transport, residence, and release of various radionuclides in and from an operating BWR.

Process leakage measurement, detection, and control methods are further discussed in Sections 5.2.5, 11.2.2, 11.3.2, and 12.1.2.

The effect of process leakage sources on the in-plant airborne radionuclide concentrations and the adequacy of plant ventilation systems is discussed in Section 12.2.2. Liquid radioactive sources are discussed in Section 11.2 and gaseous radioactive sources are discussed in Section 11.3.

11.1.6 REFERENCES

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11.1-2 Williamson, H. E., Ditmore, D. C., "Experience with BWR Fuel Through September 1971," NEDO-10505, May 1972. (Update)

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11.1-4 Ray, J. W., "Tritium in Power Reactors," <u>*Reactor and Fuel Processing Technology,*</u> *12 (1), pp. 19-26, Winter 1968-1969.*

11.1-5 Kahn, B., et al, "Radiological Surveillance Studies at a Boiling Water Nuclear Power Reactor," BRH/DER 70-1, March 1970.

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11.1-7 Marrero, T. R., "Airborne Releases From BWRs for Environmental Impact Evaluation," NEDO-21159, March 1976.

11.1-8 Gilbert, R. S., Skarpelos, J. M., "Technical Derivation of 1971 BWR Design Basis Radioactive Material Source Terms," NEDO-10871, March 1973.

Table 11.1-1

Noble Radiogas Source Terms

Isotope	*Half-Life*	*Source Term t = 0 (μCi/sec)*	*Source Term t = 30 minutes (μCi/sec)*
^{83m}Kr	1.86 hr	3.4×10^{3}	2.9×10^{3}
^{85m}Kr	4.4 hr	6.1×10^{3}	5.6×10^{3}
^{85}Kr	10.74 years	10 to 20[a]	10 to 20[a]
^{87}Kr	76 minutes	2.0×10^{4}	1.5×10^{4}
^{88}Kr	2.79 hr	2.0×10^{4}	1.8×10^{4}
^{89}Kr	3.18 minutes	1.3×10^{5}	1.8×10^{2}
^{90}Kr	32.3 sec	2.8×10^{5}	---
^{91}Kr	8.6 sec	3.3×10^{5}	---
^{92}Kr	1.84 sec	3.3×10^{5}	---
^{93}Kr	1.29 sec	9.9×10^{4}	---
^{94}Kr	1.0 sec	2.3×10^{4}	---
^{95}Kr	0.5 sec	2.1×10^{3}	---
^{97}Kr	1.0 sec	1.4×10^{1}	---
^{131m}Xe	11.96 days	1.5×10^{1}	1.5×10^{1}
^{133m}Xe	2.26 days	2.9×10^{2}	2.8×10^{2}
^{133}Xe	5.27 days	8.2×10^{3}	8.2×10^{3}
^{135m}Xe	15.7 minutes	2.6×10^{4}	6.9×10^{3}
^{135}Xe	9.16 hr	2.2×10^{4}	2.2×10^{4}
^{137}Xe	3.82 minutes	1.5×10^{5}	6.7×10^{2}

Table 11.1-1

Noble Radiogas Source Terms (Continued)

Isotope	*Half-Life*	*Source Term t = 0 (μCi/sec)*	*Source Term t = 30 minutes (μCi/sec)*
^{138}Xe	14.2 minutes	8.9×10^{4}	2.1×10^{4}
^{139}Xe	40 sec	2.8×10^{5}	---
^{140}Xe	13.6 sec	3.0×10^{5}	---
^{141}Xe	1.72 sec	2.4×10^{5}	---
^{142}Xe	1.22 sec	7.3×10^{4}	---
^{143}Xe	0.96 sec	1.2×10^{4}	---
^{144}Xe	9.0 sec	5.6×10^{2}	---
	Totals	$\sim 2.5 \times 10^{6}$	$\sim 1.0 \times 10^{5}$

[a] *Estimated from experimental observations.*

Table 11.1-2

Halogen Radioisotopes in Reactor Water

Isotope	*Half-Life*	*Concentration (μCi/g)*
^{83}Br	2.40 hr	1.5×10^{-2}
^{84}Br	31.8 minutes	2.7×10^{-2}
^{85}Br	3.0 minutes	1.7×10^{-2}
^{131}I	8.07 days	1.3×10^{-2}
^{132}I	2.28 hr	1.2×10^{-1}
^{133}I	20.8 hr	8.8×10^{-2}
^{134}I	52.3 minutes	2.4×10^{-1}
^{135}I	6.7 hr	1.3×10^{-1}

Table 11.1-3

Other Fission Product Radioisotopes in Reactor Water

Isotope	*Half-Life*	*Concentration (μCi/g)*
^{89}Sr	50.8 days	3.1×10^{-3}
^{90}Sr	28.9 years	2.3×10^{-4}
^{91}Sr	9.67 hr	6.9×10^{-2}
^{92}Sr	2.69 hr	1.1×10^{-1}
^{95}Zr	65.5 days	4.0×10^{-5}
^{97}Zr	16.8 hr	3.2×10^{-5}
^{95}Nb	35.1 days	4.2×10^{-5}
^{99}Mo	66.6 hr	2.2×10^{-2}
^{99m}Tc	6.007 hr	2.8×10^{-1}
^{101}Tc	14.2 minutes	1.4×10^{-1}
^{103}Ru	39.8 days	1.9×10^{-5}
^{106}Ru	368 days	2.6×10^{-6}
^{129m}Te	34.1 days	4.0×10^{-5}
^{132}Te	78.0 hr	4.9×10^{-2}
^{134}Cs	2.06 years	1.6 x- 10^{-4}
^{136}Cs	13.0 days	1.1×10^{-4}
^{137}Cs	30.2 years	2.4×10^{-4}
^{138}Cs	32.3 minutes	1.9×10^{-1}
^{139}Ba	83.2 minutes	1.6×10^{-1}

Table 11.1-3

Other Fission Product Radioisotopes in Reactor Water (Continued)

Isotope	Half-Life	Concentration (μCi/g)
^{140}Ba	12.8 days	9.0×10^{-3}
^{141}Ba	18.3 minutes	1.7×10^{-1}
^{142}Ba	10.7 minutes	1.7×10^{-1}
^{141}Ce	32.53 days	3.9×10^{-5}
^{143}Ce	33.0 hr	3.5×10^{-5}
^{144}Ce	384.4 days	3.5×10^{-5}
^{143}Pr	13.58 days	3.8×10^{-5}
^{147}Nd	11.06 days	1.4×10^{-5}
^{239}Np	2.35 days	2.4×10^{-1}

Table 11.1-4

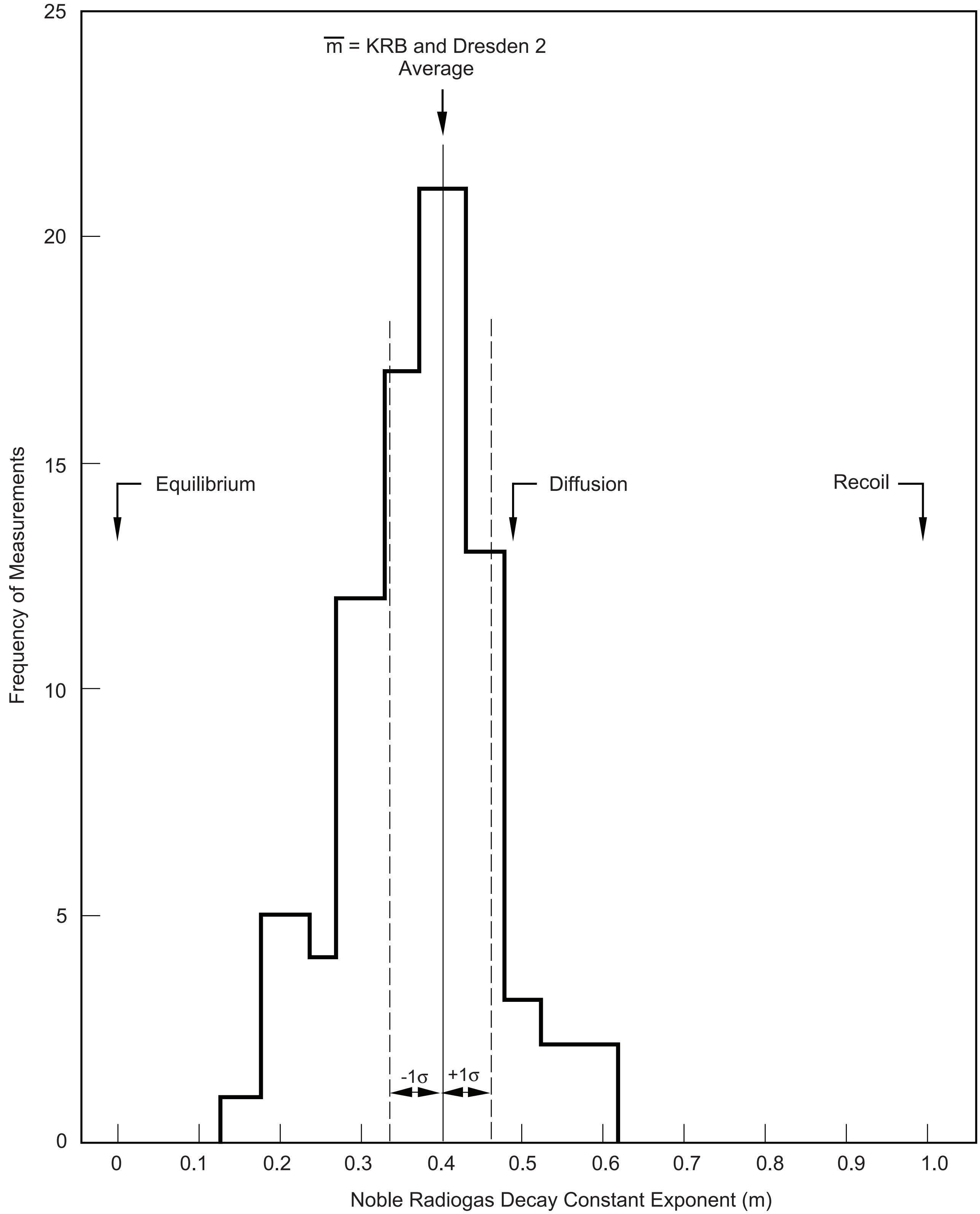
Coolant Activation Products in Reactor Water and Steam

Isotope	Half-Life	Steam Concentration (μCi/g)	Reactor Water Concentration (μCi/g)
^{13}N	9.99 minutes	7×10^{-3}	4×10^{-2}
^{16}N	7.13 sec	5×10^{1}	4×10^{1}
^{17}N	4.14 sec	2×10^{-2}	6×10^{-3}
^{19}O	26.8 sec	8×10^{-1}	7×10^{-1}
^{18}F	109.8 minutes	4×10^{-3}	4×10^{-3}

Table 11.1-5

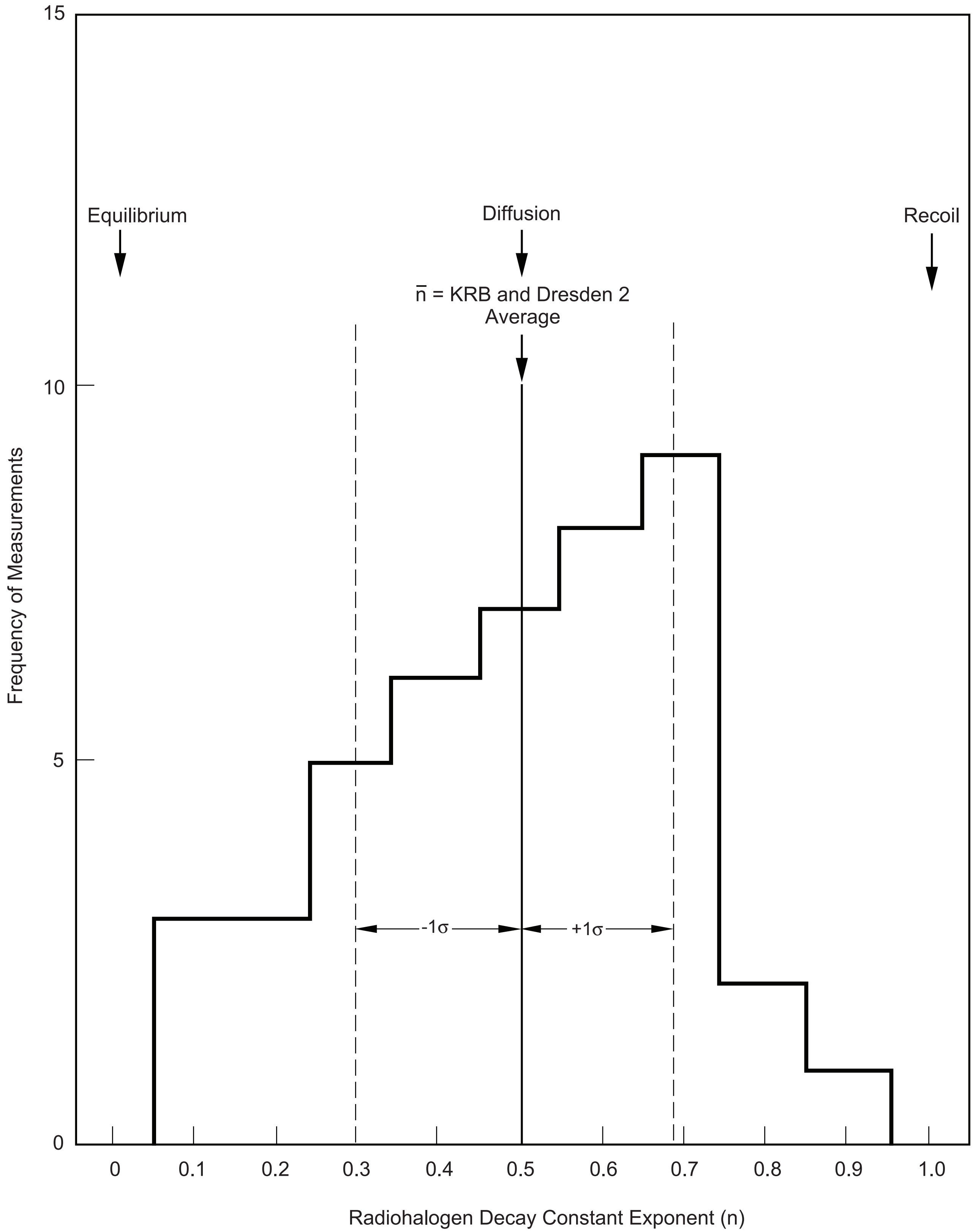
Noncoolant Activation Products in Reactor Water

Isotope	Half-Life	Concentration (μCi/g)
^{24}Na	15 hr	2×10^{-3}
^{32}P	14.31 days	2×10^{-5}
^{51}Cr	27.8 days	5×10^{-4}
^{54}Mn	313 days	4×10^{-5}
^{56}Mn	2.58 hr	5×10^{-2}
^{58}Co	71.4 days	5×10^{-3}
^{60}Co	5.25 years	5×10^{-4}
^{59}Fe	45 days	8×10^{-5}
^{65}Ni	2.55 hr	3×10^{-4}
^{65}Zn	243.7 days	2×10^{-6}
^{69m}Zn	13.7 hr	3×10^{-5}
^{110m}Ag	253 days	6×10^{-5}
^{187}W	23.9 hr	3×10^{-3}



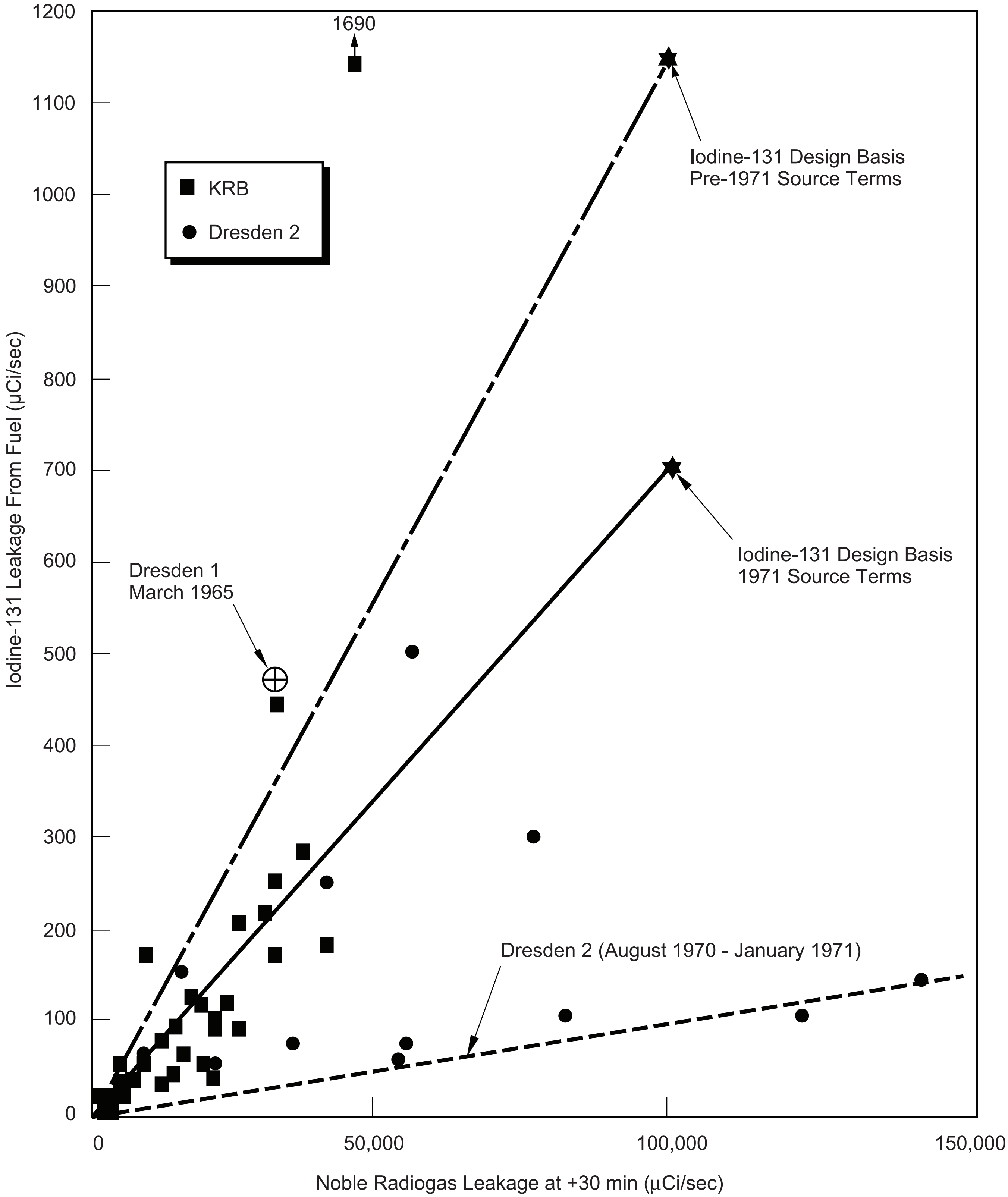
Columbia Generating Station Final Safety Analysis Report	**Noble Radiogas Decay Constant Exponent Frequency Histogram**

Draw. No. 900547.45 | Rev. | Figure 11.1-1



Radiohalogen Decay Constant Exponent Frequency Histogram

Draw. No. 900547.46 | Rev. | Figure 11.1-2



Noble Radiogas Leakage Versus
Iodine-131 Leakage

11.2 LIQUID WASTE MANAGEMENT SYSTEM

11.2.1 DESIGN BASIS

The liquid waste management system is designed to collect, segregate, store, and process potentially radioactive liquids generated during normal plant operation and anticipated operational occurrences. The design objective is to keep the radiation dose in unrestricted areas as low as is reasonably achievable (ALARA) within the guidelines of Appendix I to 10 CFR 50. The design incorporates the objectives of maximum recycle and minimum release of radioactive liquids without limiting plant operations or availability.

The criteria considered in the design of this system include volume, radioactivity, operational exposure, and required quality for recycle of the processed liquid. Radioisotopic inventories of the components used for the design are listed in Tables 11.2-1 through 11.2-5. These values are based on the reactor water source term associated with the design-basis fuel leakage rate. Allowance is made for concentration, decay and daughter product buildup in filters, demineralizers, and tanks. Equipment locations and arrangements are shown in Section 1.2.

The system is designed to treat process liquids with radioisotope concentrations associated with the design-basis fuel leakage and produce a quality of water which allows its recycle for plant reuse. Water inventory will occasionally require the discharge of processed liquids to the environs, in which case concentrations of radioisotopes in the effluent (Table 11.2-6) will be significantly less than the values specified in 10 CFR 20 and within the release limits established in the Technical Specifications. Radiation exposure to persons in unrestricted areas resulting from liquid waste discharged during normal operation and anticipated operational occurrences is less than the guidelines specified in 10 CFR 50, Appendix I.

Tanks that hold radioactive liquid, including the condensate storage tanks, are monitored for level and alarm primarily in the radwaste control room. Table 11.2-7 lists the design features of the tanks in the radwaste building used to prevent uncontrolled releases due to spillage and shows overflow alarms and drainage paths. The radwaste systems are operated from the radwaste control room; hence, additional local alarms are not required for radwaste tank levels.

All liquid waste management system tank overflows and drains are routed to the radwaste building equipment and floor drain sumps (see Figure 9.3-11). Radioactive liquid samples are primarily routed to sampling sinks. Those samples, which are local, drain into floor drain trenches, equipment or floor drain funnels, and pump beds. The above sample receivers are routed to various radioactive sumps, all of which are processed by the liquid waste management system.

Indoor tanks that hold radioactive liquid are not enclosed by individual curbs or elevated thresholds. The portion of the radwaste building that houses radioactive liquid tanks is Seismic

Category I, as discussed in Section 3.8.4.1.2. The radwaste building can retain the normal operating capacity of all liquid radwaste tanks, which are nonseismic. Analysis of a postulated radioactive release due to a liquid radwaste tank failure is presented in Section 15.7.3.

The radioactive and nonradioactive equipment and floor drains within the plant are segregated. Equipment and floor drains within the reactor building and radwaste building are routed to the liquid waste management system. The turbine building equipment and floor drains are segregated for radioactivity by component source and area (see Section 9.3.3). Although three of the turbine building sumps are designated as collectors of nonradioactive waste water, there is a possibility of low level contamination of the effluent water. One contamination source is steam leaks inside the building which condense on interior surfaces and are routed to floor drains. Because of this possibility, the discharge of these three sumps is routed to the radwaste system for processing.

The design of the system was accomplished prior to the issuance of Regulatory Guide 8.8. However, the system does incorporate substantially the guidance provided in this regulatory guide.

Like the nonradioactive sumps, the storm water drainage system (see Section 9.3.3.2.3.1) is not intended to collect radioactive materials. Nonetheless, radionuclides have been detected in the pond water and sediments. These concentrations are attributed to unanticipated and unavoidable occurrences. For example, special system draindowns during maintenance activities may have contributed minor amounts of tritium and corrosion products. Tritium that leaves the plant as a vapor can condense on building roofs and exterior walls and be carried to the ponds in storm drainage. Also, water treatment filter backwashes that are routed to the ponds can contribute radionuclides which were withdrawn from the river.

The degree of compliance with Regulatory Guide 1.143 is described in Section 1.8.

The liquid waste management system is designed to the requirements of General Design Criteria, Appendix A to 10 CFR 50, as follows:

General Design Criterion 60

The system capacity as required by General Design Criterion 60 is sufficient for the volume of liquid waste expected from normal operation and anticipated operational occurrences such as condenser leakage, maintenance activities, and process equipment down time. Flow rates are listed in Tables 11.2-8 through 11.2-10.

General Design Criterion 64

Radioactivity monitoring in the sample tanks and in the effluent discharge path ensures that excess liquid discharged to the environs does not exceed the limits of 10 CFR 20. Sampled

fluids exceeding these limits are returned to an appropriate collector tank for reprocessing. The radwaste effluent radiation monitoring system is described in Section 11.5.2.2.2.3.

11.2.2 SYSTEM DESCRIPTION

11.2.2.1 Process Description

Radioactive liquid wastes are collected and segregated into three categories: high purity waste, low purity waste, and chemical waste. Wastes thus classified are treated in subsystems designated as equipment drain, floor drain, and chemical waste, respectively.

High purity wastes, treated in the equipment drain subsystem, have low conductivity and relatively high radioactivity concentrations. Radioactive material is removed from these wastes by filtration and ion exchange. Following treatment and batch sampling, the processed waste is normally returned to condensate storage for reuse in plant.

Low purity wastes, treated in the floor drain subsystem, have moderate conductivity and generally low radioactivity. As with high purity wastes, radioactive material is removed by filtration and ion exchange. Following treatment and batch sampling, the processed waste is normally returned to condensate storage for reuse in the plant.

The high conductivity and organic content of chemical wastes preclude normal treatment in the system demineralizers by ion exchange. These wastes can be neutralized by adding appropriate neutralizing agent and thorough mixing. If necessary, following neutralization, these wastes are routed to a backwash tank or phase separator where unexpended ion-exchange capacity of the resins is used to remove contaminants from the chemical waste. After thorough mixing and a period of quiescence to allow the ion-exchange process to proceed, the excess liquid is decanted to the floor drain system for further processing. The spent resins in the separators are then processed as described below.

The installed chemical waste concentrators are currently not used. Their preoperational testing and use has been deferred until a need is identified. There are currently no plans to activate the concentrators.

The installed detergent drain tanks are not normally used. Any wastes requiring disposal are handled on a case-by-case basis and routed for processing and disposal in accordance with plant procedures and regulatory requirements and guidelines.

All liquid radwaste process streams terminate in one of the sample or distillate tanks. Since the liquid waste management system is operated on a batch basis, this arrangement allows each treated batch to be sampled to ensure that the treatment was effective. If the sample indicates that the processed liquid is still above acceptable radioactivity limits or substandard in purity, equipment is provided to either recycle the batch through the same treatment or through a

subsystem providing a higher degree of treatment. If the sample indicates that the level of activity is within limits required to discharge and the water is in excess of inventory capacity or is substandard in purity, the processed liquid may be discharged. The actual release of effluents from any processed and sampled batch tank requires the opening of a key-locked valve in accordance with written operating procedures. All required information regarding the batch release must be documented. These procedures are established to prevent inadvertent release of liquids that have not been suitably processed and analyzed.

Expended ion exchange resins are removed by backwashing to the spent resin tank and phase separators. Excess backwash water is removed from the phase separators by decantation and routed to either the floor drain or equipment drain collector tank for treatment. The powdered resin sludge is accumulated for radioactive decay. Following accumulation of successive layers in the phase separator, the sludge is then transferred to the radwaste processing system for dewatering or solidification. The deep bed resins in the spent resin tank, after a decay period, are also transferred to the solid radwaste processing system. Water separated from the wastes is returned via the waste sludge phase separator to the floor drain collector tank for treatment. The solid waste management system is described in Section 11.4.

Noncondensable gases from the liquid waste processing vessels are vented through the radwaste building exhaust system. This system is described in Section 9.4.3.2.

A process flow diagram, Figure 11.2-1, together with process flow diagram data, Table 11.2-11, show the volumes, flow rates, and radioactivity concentration used in the design of the liquid and solid waste management system.

The radionuclide distribution of liquid waste management system influents is based on the reactor coolant concentrations, as discussed in Section 11.1, taking into consideration mixing and dilution sources.

Decontamination factors which were used for evaluations of the system are those values specified in draft Regulatory Guide 1.CC. Table 11.2-12 shows the process equipment design basis decontamination factors which were used to generate the radioactivity values in Table 11.2-11.

11.2.2.2 Subsystems Description

11.2.2.2.1 Equipment Drain Subsystem

The equipment drain subsystem consists of a waste collector tank, waste surge tank, pressure precoat filter, deep bed demineralizer, two waste sample tanks, and auxiliary equipment necessary to operate the subsystem. Sizes and capacities of the equipment are listed in Table 11.2-13. The waste surge tank principally serves as a receptacle for reactor hydrotest and thermal expansion water and residual heat removal (RHR) system flush water during

startup and testing of these systems. The waste surge tank also serves as backup during equipment downtime. The piping and instrumentation drawing for this subsystem is shown in Figure 11.2-2.

High purity (low solids content) liquid wastes are collected in the waste collector tank from the following sources:

a. Drywell equipment drain sump,
b. Reactor building equipment drain sump,
c. Radwaste building equipment drain sump,
d. Turbine building equipment drain sump,
e. Reactor water cleanup system,
f. RHR system,
g. Cleanup phase separators (decant water),
h. Condensate phase separators (decant water), and
i. Fuel pool seal rupture drains (to waste surge tank).

The quantities of these wastes are summarized in Table 11.2-8. Since these wastes can contain a high percentage of primary reactor water, the radioactive concentration could be relatively high (on the order of 2.4 µCi/ml).

In the event of a component malfunction within the equipment drain subsystem, sufficient cross ties are provided to the floor drain subsystem to permit continued processing of the wastes. Sufficient capacity and treatment capability are provided to handle such conditions.

Normally, the equipment drain subsystem treated effluents are recycled to the condensate storage tanks for reuse within the plant. When condensate storage capacity is not available or the water is substandard in purity, the purified liquid from this subsystem may be sampled, analyzed and, if acceptable for release as described previously, routed to the blowdown line for discharge. Liquid waste that is unacceptable for discharge is reprocessed.

11.2.2.2.2 Floor Drain Subsystem

The floor drain subsystem consists of a floor drain collector tank, pressure precoat filter, deep bed demineralizer, sample tank, and auxiliary equipment necessary to operate the subsystem. Sizes and capacities of the equipment are listed in Table 11.2-13. The flow diagram for this subsystem is shown in Figure 11.2-3.

Intermediate purity liquid wastes are collected in the floor drain collector tank from the following sources:

a. Drywell floor drain sump,
b. Reactor building floor drain sumps,

c. Radwaste building floor drain sumps,
d. Turbine building floor drain sump, and
e. Waste sludge phase separator (decant water).

The quantities of these wastes are summarized in Table 11.2-9. These wastes are normally of intermediate purity (50 mho/cm and higher) and have radioactive concentrations on the order of 0.1 µCi/ml.

Similar to the equipment drain subsystem, the floor drain subsystem normally functions as an independent process stream. Equipment redundancy and intersystem cross ties have been provided to allow substitution for any failed component.

High purity effluent is routed to condensate storage for reuse in the plant. When condensate storage capacity is exceeded or the processed liquid is substandard in purity, it may be discharged via the blowdown line, if it meets acceptable limits.

11.2.2.2.3 Chemical Waste Subsystem

The chemical waste subsystem consists of two of each of the following: detergent drain tanks, chemical waste tanks, decontamination solution concentrators, decontamination solution concentrated waste tanks, and distillate tanks. It contains also a polishing (deep bed) demineralizer and auxiliary equipment necessary to operate the subsystem. Sizes and capacities of the equipment are listed in Table 11.2-13. The flow diagram for this subsystem is shown in Figure 11.2-4. The decontamination solution concentrators, concentrated waste tanks, distillate tanks, and polishing demineralizer have been installed but will not be used until plant operating experience indicates a need and system testing is accomplished. Chemical wastes are currently processed by routing to a backwash tank or phase separator for use of unexpended ion-exchange capacity of the resins to clean the water prior to decanting to the floor drain subsystem for further processing.

Chemical wastes collected in the chemical waste tank are from the following sources:

a. Detergent drains,
b. Shop decontamination solutions,
c. Reactor and turbine building decontamination drains,
d. Low purity wastes from either the equipment or floor drain subsystems,
e. Filter demineralizer element chemical cleaning solutions,
f. Battery room drains,
g. Chemical system overflows and tank drains, and
h. Laboratory drains.

The quantities of these wastes are summarized in Table 11.2-10. These chemical wastes are of such high conductivity and organic content as to preclude normal treatment by ion exchange,

and the radioactivity concentrations are variable. These wastes are processed by routing to a backwash tank or phase separator and to the floor drain system for further processing.

If the concentrators were ever activated, the evaporator concentrates would be processed by the solid waste management system and the distillate would be routed to the distillate tank. After analysis, the distillate could be routed through a polishing demineralizer to further reduce impurities, recycled through the evaporator, or sent directly to condensate storage for plant reuse. As with the other subsystems, when high purity water storage capacity is exceeded or the processed liquid is of substandard purity, liquid within 10 CFR 20 release limits could be discharged via the blowdown line.

11.2.2.2.4 Shared Equipment

Other than serving as mutual backup, main process equipment normally is not shared between subsystems. Auxiliary equipment not in the direct process stream is shared between subsystems. Shared equipment includes the following:

a. The waste precoat tank and waste precoat pump are shared between the waste collector filter and the floor drain filter,

b. The waste filter aid tank is shared between the waste collector filter and floor drain filter,

c. The resin addition tank is shared between the waste demineralizer, floor drain demineralizer, and polishing demineralizer, and

d. The chemical addition tanks, caustic and acid, and associated pumps are shared between the waste collector tank, floor drain collector tank, detergent tanks, and chemical waste tanks.

11.2.2.2.5 Surge Capacities

The radwaste system process data is the basis for sizing of the equipment. Tables 11.2-8, 11.2-9, and 11.2-10 list startup flows, daily flows, and maximum flows for the equipment drain subsystem, floor drain subsystem and chemical waste subsystem, respectively. Anticipated operational occurrences such as startup operations, equipment malfunction, and shutdown operations are accounted for in these tabulations. The bases for these types of values are presented in NEDO-10951, "Releases from BWR Radwaste Management Systems," July 1973.

The surge storage and process capacities can be envisioned by comparing the normal and maximum daily volumes, listed in Table 11.2-11, with the design flow rates of pumps and tank volumes listed in Table 11.2-13. Alternate processing rather than bypass operations are used

during equipment downtime. The equipment and floor drain subsystems are sized such that with either subsystem inoperative, the remaining subsystem is capable of processing the maximum expected volume of both subsystems. Additionally, the waste surge tank provides reserve storage capacity. The chemical waste subsystem incorporates two parallel processing paths. Cross connections allow individual components from either process path to serve as a substitute in the other process path. The parallel path processing and adequate storage capacity ensures that inoperability of any component in this subsystem will not limit plant operation.

11.2.2.2.6 Design Features

The design pressures and temperatures for individual components are listed in Table 11.2-13. Collection and storage tanks are designed for atmospheric pressure. The mixed-bed demineralizer units, precoat filter units, and concentrators are pressure vessels. The quality classification for the system is Quality Group D+ as defined in Section 3.2.

Chapter 12 discusses the design features incorporated in the system to maintain occupational exposure ALARA. As shown in the general arrangement drawings in Section 1.2, the radwaste processing equipment is located in shielded rooms and cells. Process lines that penetrate shield walls are routed to prevent a direct radiation path from the tanks and equipment to normally occupied areas.

Control of the liquid waste management system is from a shielded radwaste control room and from shielded local operating galleries. Flanges are provided where required for maintenance in an otherwise all-welded system. The cells and concrete rooms provide secondary enclosures which facilitate collection of spills or leaks from the system for processing. Vessels that may have their contents mixed with air or that may have batches transferred into them by means of pneumatic transfer are closed and vented to the radwaste building ventilation system. Piping and tubing 2 in. and under is field routed but required to be in specified space envelopes for shielding and in-plant exposure consideration.

The liquid waste management system is designed to minimize the effects of equipment malfunction and operator error. After initiation by the operator, valve positioning, equipment startup, and system operation can be automatically performed by a process controller. Failure of any valve to properly align stops the sequence and prevents a pump start. Once on line, processing continues until the feed volume is processed or the capacity of a piece of process equipment is reached. In either case, automatic shutdown occurs with valves returning to the shutdown position. During initial system startup or in the event of controller failure, the processes are manually controlled. System variables, such as tank levels, flow rates, pressures, and conductivity are indicated and alarmed in the radwaste control room.

The discharge from the liquid waste subsystems to the blowdown line is normally isolated by a closed manual valve and a key-locked closed manual valve separated by a telltale drain. This path is further protected by air-operated isolation valves, flow control valves, flow indication

and radiation monitoring instrumentation. These design features are complimented by administrative procedures which prevent inadvertent radioactive liquid releases and releases of liquids that exceed the Technical Specifications release limit.

11.2.3 RADIOACTIVE RELEASES

The decision to recycle or to release a particular batch of processed liquid is based on plant water inventory and the type and concentration of chemical impurities being processed. It is not expected that tritium buildup in the plant will determine the frequency of releases. Concentration of radionuclides released to the environment will be significantly less than those values specified in 10 CFR 20. The quantities of radioactive materials released during normal operation will not result in dose rates to persons in unrestricted areas in excess of 10 CFR 50, Appendix I, values as implemented by the Technical Specifications release limits.

11.2.3.1 Release Point and Dilution

Excess liquid effluent is discharged to the circulating water blowdown line at a variable rate up to about 190 gpm. Dilution is furnished by circulating water system blowdown when required or when normal blowdown is in progress. The circulating water blowdown line terminates in the Columbia River. Applicable concentration limits specified in 10 CFR 20 apply at the point of discharge to the river.

The storm water drainage ponds described in Section 9.3.3.2.3.1 are a point of release of detectable radionuclides. For the reasons discussed in Section 11.2.1, most of the activity is believed to be either condensed gaseous effluent or material of external origin. The ponds are within a plant restricted area (see Section 2.1.1.3), and public access is restricted by a fence which surrounds the ponds.

11.2.3.2 Calculation of Releases of Radioactive Materials

Quantities of radioactive materials released with liquids were calculated for initial plant licensing using the GALE code presented in draft Regulatory Guide 1.CC to show compliance with Appendix I to 10 CFR 50 for normal operation plus anticipated operational occurrences. The plant operational parameters, including source terms, were those presented in Appendix B of the guide with waste stream flow rates adjusted to the Columbia Generating Station plant design. The calculated quantities of radioactive materials released with liquids used for initial plant licensing are presented in Table 11.2-14 in terms of curies per year. The radionuclide concentrations in the effluent are presented in Table 11.2-6 and are compared with the values of 10 CFR 20, Appendix B, Table II, Column 2.

Since becoming operational, releases of radioactive materials in liquid effluents have been determined using actual flow volumes and quantitative and qualitative laboratory analyses. Doses due to radioactive materials released in liquid effluents are determined to be in

compliance with Technical Specifications at specified intervals. Compliance is reported in the Annual Radioactive Effluent Release Report using the NRC LADTAP II computer code along with parameters outlined in the Offsite Dose Calculation Manual.

11.2.3.3 Exposure of Persons at or Beyond the Site Boundary

Estimated annual exposure of persons in unrestricted areas resulting from liquid effluents (Table 11.2-6) is discussed in Section 5.2 of the Environmental Report, Operating License stage. The estimated total body dose of 2.3 mrem per year and largest calculated single organ dose of 1.6 mrem per year to the bone are well within the guidelines of 10 CFR 50, Appendix I.

11.2.3.4 Cost-Benefit Analysis

It has been determined that a cost-benefit analysis, as described in Appendix I to 10 CFR 50 Section II.D is not required for Columbia Generating Station.

The question of eligibility of Columbia Generating Station to dispense with the Appendix I cost-benefit analysis as to "per site" limitations was reviewed by the NRC in connection with their review of WPPSS Nuclear Project No. 1 (WNP-1) and WPPSS Nuclear Project No. 4 (WNP-4), Docket Nos. 50-460 and 50-513. The Staff concluded in its testimony at the Atomic Safety and Licensing Board (ASLB) hearing for WNP-1 and WNP-4 held on November 11, 1975, that

> The aggregate doses associated with WPPSS Nuclear Project No. 1, WPPSS Project No. 2, and WPPSS Nuclear Project No. 4 operation meet the RM-50-2 (i.e., Annex) design objectives [Tr. 724-727].

These conclusions were ratified by the ASLB on its decision pertaining to WNP-1 and WNP-4 of December 22, 1975, RAI-75/12 922, 934.

Table 11.2-1

Liquid Waste Management System Radioisotope Inventory
Equipment Drain Subsystem

Radioisotope	Radioisotope Inventory (μCi)			
	Waste Collector Tank[a]	Waste Filter[a]	Waste Demineralizer[a]	Waste Sample Tank[a]
^{83}Br	4.53E4		3.15E4	3.18E2
^{84}Br	4.67E4		9.19E3	9.28E1
^{85}Br	1.17E4		4.17E-4	4.21E-6
^{89}Sr	4.35E4		1.88E6	4.35E2
^{90}Sr	3.25E3		2.12E5	3.25E1
^{91}Sr	7.12E5		1.10E6	9.13E3
^{92}Sr	3.40E5		2.48E5	2.50E3
^{90}Y	3.53E2		2.00E5	3.92E0
^{91}Y	4.21E3		5.10E5	4.69E1
^{91M}Y	4.12E5		7.31E5	5.82E3
^{92}Y	2.15E5		2.37E5	2.33E3
^{95}Zr	5.63E2	2.81E2	1.31E4	5.62E0
^{97}Zr	6.94E1	1.73E1	1.70E1	3.46E-1
^{95}Nb	5.93E2	2.97E2	1.69E4	5.93E0
^{99}Mo	2.79E5	1.38E5	6.13E5	2.75E3
^{99M}Tc	1.70E6	8.18E3	2.02E6	1.51E4
^{101}Tc	2.05E5		5.15E3	5.20E1
^{103}Ru	2.66E2	1.33E2	5.17E3	2.66E0
^{106}Ru	3.67E1	1.84E1	1.12E3	3.67E-1
^{129M}Te	5.60E2		2.00E4	5.59E0
^{129}Te	3.03E2		1.27E4	3.31E0
^{132}Te	6.30E5		3.21E6	6.23E3
^{129}I	2.83E-8		9.76E-5	3.16E-10
^{131}I	1.77E5		2.12E6	1.76E3
^{132}I	8.53E5		3.45E6	7.83E3
^{133}I	9.03E5		1.57E6	8.67E3
^{134}I	4.66E5		1.75E5	1.77E3
^{135}I	7.73E5		7.35E5	6.80E3
^{134}Cs	2.26E3		1.44E5	2.26E1
^{135}Cs	4.19E-5		1.97E-2	5.95E-7
^{136}Cs	1.52E3		2.80E4	1.51E1
^{137}Cs	3.39E3		2.21E5	3.39E1
^{138}Cs	3.29E5		6.54E4	6.60E2
^{137M}Ba	2.99E3		2.07E5	3.17E1

Table 11.2-1

Liquid Waste Management System Radioisotope Inventory
Equipment Drain Subsystem (Continued)

	Radioisotope Inventory (μCi)			
Radioisotope	Waste Collector Tank[a]	Waste Filter[a]	Waste Demineralizer[a]	Waste Sample Tank[a]
^{139}Ba	3.67E5		1.98E5	2.00E3
^{140}Ba	1.24E5		2.29E6	1.24E3
^{141}Ba	2.64E5		1.50E4	1.52E2
^{142}Ba	2.32E5		2.14E3	2.16E1
^{141}Ce	1.18E3	5.99E2	3.04E4	1.22E1
^{143}Ce	3.98E2	1.94E2	4.80E2	3.88E0
^{144}Ce	4.94E2	2.47E2	1.48E4	4.94E0
^{140}La	2.07E4		2.28E6	2.29E2
^{141}La	3.66E4		4.61E4	4.60E2
^{142}La	2.34E4		3.08E4	3.11E2
^{143}Pr	5.37E2	2.69E2	1.71E4	5.37E0
^{144}Pr	4.40E2	2.46E2	1.48E4	4.91E0
^{147}Nd	1.92E2	9.59E1	1.52E3	1.92E0
^{239}NP	2.98E6		1.13E7	2.93E4
^{24}Na	1.80E4		2.52E4	1.70E2
^{32}P	2.77E2		5.54E3	2.76E0
^{51}Cr	6.99E3	3.49E3	1.10E5	6.98E1
^{54}Mn	5.65E2	2.82E2	1.70E4	5.65E0
^{56}Mn	1.58E5	5.67E4	5.57E4	1.13E3
^{58}Co	7.04E4	3.52E4	1.68E6	7.03E2
^{60}Co	7.07E3	3.53E4	2.26E5	7.07E1
^{59}Fe	1.12E3	5.61E2	2.30E4	1.12E1
^{65}Ni	9.40E2	3.36E2	3.30E2	6.73E0
^{65}Zn	2.82E1		1.67E3	2.82E-1
^{69M}Zn	2.63E2		3.52E2	2.47E0
^{110M}Ag	8.47E2	4.24E2	2.52E4	8.47E0
^{187}W	3.16E4	1.53E4	2.99E4	3.05E2

[a] The radioisotope inventory listed above for the waste collector tank, waste filter, waste demineralizer, and waste sample tank is assumed to be applicable to the floor drain collector tank, floor drain filter, floor drain demineralizer, and floor drain sample tank, respectively. This is done for the purpose of shielding analysis. In actuality, the radionuclide inventory in any component of the floor drain system is discussed in Sections 11.2.2.2.1 and 11.2.2.2.2. The values from the equipment drain subsystem are maximum due to cross-ties for alternate processing.

Table 11.2-2

Liquid Waste Management System Radioisotope Inventory
Chemical Waste Subsystem

	Radioisotope Inventory (μCi)				
Radioisotope	Chemical Waste Tank	DSC Waste Tank[a]	DSC Waste Measuring Tank[b]	Distillate Tank	Decon. Solution Concentrator
^{83}Br					
^{84}Br					
^{85}Br					
^{89}Sr	1.11E3	2.82E3	1.74E3	1.11E0	1.48E3
^{90}Sr	7.87E1	2.00E2	1.23E2	7.87E-2	1.05E2
^{91}Sr					
^{92}Sr					
^{90}Y					
^{91}Y					
^{91M}Y					
^{92}Y					
^{95}Zr	1.43E1	3.65E1	2.25E1	1.43E-2	1.92E1
^{97}Zr					
^{95}Bv					
^{99}Mo	7.87E3	2.00E4	1.23E4	7.87E0	1.05E4
^{99M}Tc					
^{101}Tc					
^{103}Ru					
^{106}Ru					
^{129M}Te	1.43E1	3.65E1	2.25E1	1.43E-2	1.92E1
^{129}Te					
^{132}Te	1.75E4	4.45E4	2.75E4	1.75E1	2.34E4
^{129}I					
^{131}I	4.65E3	1.13E4	6.95E3	4.65E0	5.92E3
^{132}I					
^{133}I					
^{134}I					
^{135}I					
^{134}Cs	5.75E1	1.46E2	9.02E1	5.75E-2	7.68E1
^{135}Cs					
^{136}Cs	3.94E1	1.00E2	6.19E1	3.94E-2	5.27E1
^{137}Cs	8.60E1	2.19E2	1.35E2	8.60E-2	1.15E2

Table 11.2-2

Liquid Waste Management System Radioisotope Inventory
Chemical Waste Subsystem (Continued)

	Radioisotope Inventory (μCi)				
Radioisotope	Chemical Waste Tank	DSC Waste Tank[a]	DSC Waste Measuring Tank[b]	Distillate Tank	Decon. Solution Concentrator
^{138}Cs					
^{137M}Ba					
^{139}Ba					
^{140}Ba	3.22E3	8.18E3	5.05E3	3.22E0	4.30E3
^{141}Ba					
^{142}Ba					
^{141}Ce	1.40E1	3.56E1	2.20E1	1.40E-2	1.87E1
^{143}Ce	1.25E1	3.17E1	1.96E1	1.25E-2	1.67E1
^{144}Ce	1.25E1	3.17E1	1.96E1	1.25E-2	1.67E1
^{140}La					
^{141}La					
^{142}La					
^{143}Pr	1.36E1	3.44E1	2.12E1	1.36E-2	1.81E1
^{144}Pr					
^{147}Nd	5.01E0	1.27E1	7.85E0	5.01E-3	6.69E0
^{239}Np	8.60E4	2.19E5	1.35E5	8.60E1	1.15E5
^{24}Na					
^{32}P	7.18E0	1.82E1	1.13E1	7.18E-3	9.58E0
^{51}Cr	1.79E2	4.54E2	2.80E2	1.79E-1	2.38E2
^{54}Mn	1.43E1	3.65E1	2.25E1	1.43E-2	1.92E1
^{56}Mn					
^{58}Co	1.79E3	4.54E3	2.80E3	1.79E0	2.38E3
^{60}Co	1.79E2	4.54E2	2.80E2	1.79E-1	2.38E2
^{59}Fe	2.87E1	7.26E1	4.48E1	2.87E-2	3.82E1
^{65}Ni					
^{65}Zn	7.18E-1	1.82E0	1.13E0	7.18E-4	9.58E-1
^{69M}Zn					
^{110M}Ag	2.15E1	5.46E1	3.37E1	2.15E-2	2.87E1
^{187}W					

[a] Decontamination solution concentrator waste tank.
[b] Decontamination solution concentrator waste measuring tank.

Table 11.2-3

Liquid Waste Management System Radioisotope Inventory
RWCU and Condensate Filter Demineralizers

	Radioisotope Inventory (μCi)	
Radioisotope	RWCU Filter Demineralizer	Condensate Filter Demineralizer
^{83}Br	2.50E6	2.30E5
^{84}Br	9.00E5	9.12E4
^{85}Br	3.00E4	5.42E3
^{89}Sr	1.40E7	3.00E5
^{90}Sr	1.10E6	2.56E4
^{91}Sr	3.20E7	2.13E5
^{92}Sr	1.60E7	9.12E4
^{90}Y	1.10E6	2.09E4
^{91}Y		4.54E4
^{91M}Y	2.20E7	1.28E5
^{92}Y		9.12E4
^{95}Zr	1.80E5	2.59E4
^{97}Zr	2.40E4	8.19E1
^{95}Nb	5.60E5	2.97E4
^{99}Mo	5.50E7	3.04E6
^{99M}Tc	2.40E7	3.18E6
^{101}Tc	1.90E6	1.04E4
^{103}Ru	9.00E4	1.16E4
^{106}Ru	1.20E4	1.85E3
^{129M}Te	1.50E6	3.60E3
^{129}Te		2.29E3
^{132}Te	3.70E7	1.20E6
^{129}I		4.89E-6
^{131}I	8.50E7	1.34E7
^{132}I	5.80E7	2.96E6
^{133}I	1.40E8	1.19E7
^{134}I	1.40E7	1.35E6
^{135}I	6.50E7	5.55E6
^{134}Cs	7.50E5	1.77E4
^{135}Cs		2.48E-2
^{136}Cs	4.20E5	7.37E3
^{137}Cs	1.20E6	2.67E4
^{138}Cs	5.50E6	3.23E4
^{127M}Ba	1.20E6	2.50E4

Table 11.2-3

Liquid Waste Management System Radioisotope Inventory
RWCU and Condensate Filter Demineralizers (Continued)

Radioisotope	Radioisotope Inventory (µCi) RWCU Filter Demineralizer	Condensate Filter Demineralizer
^{139}Ba	1.20E7	7.23E4
^{140}Ba	3.60E7	6.03E5
^{141}Ba	3.00E6	1.63E4
^{142}Ba	1.60E6	9.94E3
^{141}Ce	1.80E5	2.86E4
^{143}Ce	5.00E4	2.40E3
^{144}Ce	1.60E5	2.47E4
^{140}La	4.10E7	5.60E5
^{141}La		1.63E4
^{142}La		9.94E3
^{143}Pr	1.60E5	2.74E4
^{144}Pr		2.47E4
^{147}Nd	5.50E4	5.65E3
^{239}Np	5.00E8	4.27E6
^{24}Na	1.50E6	9.56E3
^{32}P	9.50E4	1.40E3
^{51}Cr	2.60E6	2.80E5
^{54}Mn	2.20E5	2.83E4
^{56}Mn	6.00E6	2.69E5
^{58}Co	2.60E7	3.27E6
^{60}Co	2.80E6	3.61E5
^{59}Fe	4.30E5	4.96E4
^{65}Ni	3.80E4	1.59E3
^{65}Zn	1.10E4	2.16E2
^{69M}Zn	2.00E4	1.34E2
^{110M}Ag	3.40E5	4.24E4
^{187}W	3.60E6	1.49E5

Table 11.2-4

Liquid Waste Management System Radioisotope Inventory
Spent Resin and Condensate Backwash Receiving Tanks

Radioisotope	Radioisotope Inventory (μCi)	
	Spent Resin Tank	Condensate Backwash Receiving Tank
^{83}Br	3.15E4	2.30E5
^{84}Br	9.19E3	9.12E4
^{85}Br	4.17E-4	5.42E3
^{89}Sr	1.88E6	3.00E5
^{90}Sr	2.12E5	2.56E4
91Sr	1.10E6	2.13E5
^{92}Sr	2.48E5	9.12E4
^{90}Y	2.00E5	2.09E4
^{91}Y	5.10E5	4.54E4
^{91M}Y	7.31E5	1.28E5
^{92}Y	2.37E5	9.12E4
^{95}Zr	1.31E4	2.59E4
^{97}Zr	1.70E1	8.19E1
^{95}Nb	1.69E4	2.97E4
^{99}Mo	6.13E5	3.04E6
^{99M}Tc	2.02E6	3.18E6
^{101}Tc	5.15E3	1.04E4
^{103}Ru	5.17E3	1.16E4
^{106}Ru	1.12E3	1.85E3
^{129M}Te	2.00E4	3.60E3
^{129}Te	1.27E4	2.29E3
^{132}Te	3.21E6	1.20E6
^{129}I	9.76E-5	4.89E-6
^{131}I	2.12E6	1.34E7
^{132}I	3.45E6	2.96E6
^{133}I	1.57E6	1.19E7
^{134}I	1.75E4	1.35E6
^{135}I	7.35E4	5.55E6
^{134}Cs	1.44E4	1.77E4
^{135}Cs	1.97E-2	2.48E-2
^{136}Cs	2.80E4	7.37E3
^{137}Cs	2.21E5	2.67E4
^{138}Cs	6.54E4	3.23E4
^{137M}Ba	2.07E5	2.50E4

Table 11.2-4

Liquid Waste Management System Radioisotope Inventory
Spent Resin and Condensate Backwash Receiving Tanks (Continued)

	Radioisotope Inventory (μCi)	
Radioisotope	Spent Resin Tank	Condensate Backwash Receiving Tank
^{139}Ba	1.98E5	7.23E4
^{140}Ba	2.29E6	6.03E5
^{141}Ba	1.50E4	1.63E4
^{142}Ba	2.14E3	9.94E3
^{141}Ce	3.04E4	2.86E4
^{143}Ce	4.80E2	2.40E3
^{140}Ce	1.48E4	2.47E4
^{141}La	2.28E6	5.60E5
^{141}La	4.61E4	1.63E4
^{142}La	3.08E4	9.94E3
^{143}Pr	1.71E4	2.74E4
^{144}Pr	1.48E4	2.47E4
^{147}Nd	1.52E3	5.65E3
^{239}Np	1.13E7	4.27E6
^{24}Na	2.52E4	9.56E3
^{32}P	5.54E3	1.40E3
^{51}Cr	1.10E5	2.80E5
^{54}Mn	1.70E4	2.83E4
^{56}Mn	5.57E4	2.69E5
^{58}Co	1.68E6	3.27E6
^{60}Co	2.26E5	3.61E5
^{59}Fe	2.30E4	4.96E4
^{65}Ni	3.30E2	1.59E3
^{65}Zn	1.67E3	2.16E2
^{69M}Zn	3.52E2	1.34E2
^{110M}Ag	2.52E4	4.24E4
^{187}W	2.99E4	1.49E5

Table 11.2-5

Liquid Waste Management System Radioisotope Inventory
Phase Separators

Radioisotope	Radioisotope Inventory (μCi)		
	RWCU Phase Separator	Condensate Phase Separator	Waste Sludge Phase Separator
^{83}Br	4.95E6	2.26E5	
^{84}Br	1.71E6	7.04E4	
^{85}Br	3.73E4	8.21E2	
^{89}Sr	3.48E8	2.94E6	
^{90}Sr	3.95E7	3.23E5	
^{91}Sr	6.40E7	2.21E5	
^{92}Sr	3.17E7	8.98E4	
^{90}Y	3.95E7	3.15E5	
^{91}Y	5.66E6	4.76E5	
^{91M}Y	4.40E7	1.35E5	
^{92}Y	2.33E5	9.57E4	
^{95}Zr	4.83E6	2.55E5	3.70E1
^{97}Zr	4.70E4	7.14E1	4.53E-1
^{95}Nb	1.41E7	3.29E5	3.96E1
^{99}Mo	1.93E8	5.23E6	1.35E4
^{99M}Tc	1.27E8	2.17E6	8.76E3
^{101}Tc	3.41E6	7.67E-2	
^{103}Ru	2.06E6	1.36E5	1.74E1
^{106}Ru	4.09E5	2.14E4	2.45E0
^{129M}Te	3.15E7	3.48E-4	
^{129}Te	1.82E7	2.07E3	
^{132}Te	1.43E8	2.40E6	
^{129}I	1.29E-1	6.20E-5	
^{131}I	6.70E8	5.31E7	
^{132}I	1.87E8	4.16E6	
^{133}I	3.00E8	1.33E7	
^{134}I	2.72E7	1.18E6	
^{135}I	1.30E8	5.69E6	
^{134}Cs	2.63E7	2.19E5	
^{135}Cs	1.98E-1	3.48E-1	
^{136}Cs	4.00E6	4.07E4	
^{137}Cs	4.31E7	3.37E5	
^{138}Cs	1.05E7	2.50E4	

Table 11.2-5

Liquid Waste Management System Radioisotope Inventory
Phase Separators (Continued)

Radioisotope	Radioisotope Inventory (μCi) RWCU Phase Separator	Condensate Phase Separator	Waste Sludge Phase Separator
^{137M}Ba	4.00E7	3.14E5	
^{139}Ba	2.36E7	6.74E4	
^{140}Ba	4.18E8	3.33E6	
^{141}Ba	5.51E6	1.01E4	
^{142}Ba	2.79E6	4.69E3	
^{141}Ce	3.80E6	2.37E5	7.78E1
^{143}Ce	1.22E5	2.88E3	1.46E1
^{144}Ce	5.37E6	2.83E5	3.29E1
^{140}La	4.78E8	3.64E6	
^{141}La	3.86E4	5.32E2	
^{142}La	4.89E4	6.33E2	
^{143}Pr	5.68E6	3.26E5	3.59E1
^{144}Pr	5.08E6	2.83E5	3.29E1
^{147}Nd	5.61E5	2.66E4	1.18E1
^{239}Np	1.57E9	6.92E6	
^{24}Na	3.07E6	1.02E4	
^{32}P	1.19E6	8.19E3	
^{51}Cr	4.93E7	2.15E6	4.50E2
^{54}Mn	7.43E6	3.26E5	3.76E1
^{56}Mn	1.19E7	2.52E5	1.52E3
^{58}Co	7.14E8	3.27E7	4.64E3
^{60}Co	9.98E7	4.30E6	4.72E2
^{59}Fe	1.03E7	4.50E5	7.35E1
^{65}Ni	7.52E4	1.49E3	9.01E0
^{65}Zn	3.63E5	2.57E3	
^{69M}Zn	4.06E4	1.41E2	
^{110M}Ag	1.14E7	4.84E5	5.64E1
^{187}W	7.95E6	1.63E5	9.81E2

Table 11.2-6

Annual Average Concentration of Radionuclides in Liquid Effluent

Nuclide	Effluent Concentration (μCi/ml)	10 CFR 20 Table II (μCi/ml)	Fraction of 10 CFR 20 Effluent/10 CFR 20
^{24}Na	1.3E-9	5E-5	0.000026
^{32}P	5.2E-11	9E-6	0.00000575
^{51}Cr	1.3E-9	5E-4	0.0000026
^{54}Mn	1.6E-11	3E-5	0.000000533
^{56}Mn	1.4E-9	7E-5	0.00002
^{55}Fe	2.8E-10	1E-4	0.0000028
^{59}Fe	8E-12	1E-5	0.00000008
^{58}Co	5.4E-11	2E-5	0.0000027
^{60}Co	1.1E-10	3E-6	0.0000367
^{65}Ni	8E-12	1E-4	0.00000008
^{64}Cu	4E-9	2E-4	0.00002
^{65}Zn	5.4E-11	5E-6	0.000011
^{69m}Zn	2.8E-10	6E-5	0.00000467
^{69}Zn	3E-10	8E-4	0.000000375
^{187}W	5.4E-11	3E-5	0.0000018
^{239}Np	1.6E-9	2E-5	0.00008
^{83}Br	7.2E-11	9E-4	0.00000008
^{84}Br	6E-12	4E-4	0.000000015
^{89}Rb	4.2E-11	9E-4	0.0000000467
^{89}Sr	2.8E-11	8E-6	0.0000035
^{91}Sr	4.4E-10	2E-5	0.000022
^{91m}Y	2.8E-10	2E-3	0.00000014
^{91}Y	1.4E-11	8E-6	0.00000175
^{92}Sr	3E-10	4E-5	0.0000075
^{92}Y	6.2E-10	4E-5	0.0000155
^{93}Y	4.6E-10	2E-5	0.000023
^{98}Nb	1.6E-11	2E-4	0.00000008
^{99}Mo	4.6E-10	2E-5	0.000023
^{99m}Tc	1.8E-9	1E-3	0.0000018
^{101}Tc	4E-12	2E-3	0.000000002
^{103}Ru	6E-12	3E-5	0.0000002

Table 11.2-6

Annual Average Concentration of Radionuclides in Liquid Effluent (Continued)

Nuclide	Effluent Concentration (μCi/ml)	10 CFR 20 Table II (μCi/ml)	Fraction of 10 CFR 20 Effluent/10 CFR 20
^{103m}Rh	6E-12	6E-3	0.000000001
^{104}Tc	1.2E-11	4E-4	0.00000003
^{105}Ru	1.1E-10	7E-5	0.00000165
^{105m}Rh	1.2E-10	--	--
^{105}Rh	3.8E-11	5E-5	0.00000076
^{129m}Te	1E-11	7E-6	0.00000150
^{129}Te	6E-12	4E-4	0.00000016
^{131m}Te	2E-11	8E-6	0.0000026
^{131}Te	4E-12	8E-5	0.00000005
^{131}I	1.4E-9	1E-6	0.0014
^{132}Te	2E-12	9E-6	0.000000233
^{132}I	7E-10	1E-4	0.000007
^{133}I	3.6E-9	7E-6	0.000510
^{134}I	2.9E-10	4E-4	0.0000007
^{134}Cs	1.6E-9	9E-7	0.00175
^{135}I	1.7E-9	3E-5	0.0000580
^{136}Cs	9.9E-10	6E-6	0.000165
^{137}Cs	3.6E-9	1E-6	0.0036
^{137m}Ba	3.4E-9	--	--
^{138}Cs	1.5E-9	4E-4	0.0000037
^{139}Ba	1E-10	2E-4	0.0000005
^{140}Ba	1E-10	8E-6	0.0000131
^{140}La	2.3E-11	9E-6	0.00000256
^{141}La	3.6E-11	5E-5	0.00000071
^{141}Ce	8E-12	3E-5	0.000000280
^{142}La	7.6E-11	1E-4	0.00000076
^{143}Ce	6E-12	2E-5	0.0000003
^{143}Pr	1E-11	2E-5	0.0000005
All others	1.5E-11	1E-8	0.0015
		Total	0.009020094

Table 11.2-7

Tank Design Features

Component	High Level Alarm	Overflows and Drains
Floor drain collector tank	Yes (RCR)	Floor drain sump, W-2
Waste sludge phase separator	Yes (RCR)	Floor drain sump, W-2
Floor drain sample tank	Yes (RCR)	Floor drain sump, W-2
Waste collector tank	Yes (RCR)	Floor drain sump, W-2
Spent resin tank	Yes (RCR)	Floor drain sump, W-2
Waste surge tank	Yes (RCR)	Equipment drain sump, W-3
Waste sample tanks	Yes (RCR)	Equipment drain sump, W-3
Detergent drain tanks[b]	No[a]	Chemical waste sump, W-4
Chemical waste tanks	Yes (RCR)	Chemical waste sump, W-4
Distillate tanks[b]	No[a]	Chemical waste sump, W-4
Nonoperational decontamination solution concentrator waste tanks[b]	No	Chemical waste sump, W-4
Condensate backwash receiving tank	Yes (RCR)	Equipment drain sump, W-3
Condensate phase separators	Yes (RCR)	Equipment drain sump, W-3
Decontamination solution concentrator waste measuring tank[c]	No[a]	Decontamination solution concentrator waste tanks
RWCU phase separators	Yes (RCR)	Equipment drain sump, W-3
Condensate storage tanks	Yes (MCR)	Floor drain sump, T4

[a] Alarms not installed.
[b] Nonoperational.
[c] Not used.

LEGEND
RCR - radwaste control room
MCR - main control room

Table 11.2-8

Equipment Drain Subsystem Sources

Source	Startup Flows (gpd)	Regular Daily Flows (gpd)	Irregular Flows (gpd)	Maximum Daily Flows (gpd)
Equipment drains				
Drywell	3,860	3,860		28,800
Reactor building	3,755	3,755		14,400
Turbine building	5,726	5,726		5,726
Radwaste building	1,000	1,000		1,000
Reactor hydrotest and water level reduction to operating state	56,720	0		0
RHR system flush water			4,000[a]	0
Condensate demineralizer backwash	27,000		13,500[b]	40,500[c]
Cleanup demineralizer backwash	2,430		1,215[d]	
Water inleakage to condenser	0	0		14,400
Total	100,491	14,341		104,826

[a] Occurs every shutdown prior to placing the RHR system in operation for shutdown cooling.

[b] Under normal operating conditions, one condensate filter demineralizer would be precoated every 4 days.

[c] The maximum daily flow is based on a main condenser inleakage of 10 gpm, which corresponds to 3 condensate demineralizer precoatings daily and maximum leak and drain inflows. Up to 36 gpm of condenser inleakage can be accommodated. This requires precoating of one condensate demineralizer every 3 hr. This inleakage rate would result in overloading the equipment drain subsystem but could be tolerated for short periods of time during location and repair of the leak.

[d] Under normal operating conditions, each cleanup demineralizer may be precoated every 6.8 days.

Table 11.2-9

Floor Drain Subsystem Sources

Source	Regular Daily Flows (gpd)	Irregular Flows (gpd)	Maximum Daily Flows (gpd)
Floor drains			
Drywell	700		28,800
Reactor building	2,000		15,000
Radwaste building	1,000		1,000
Turbine building	2,000		2,000
Waste sludge phase separator decant	0	8,489[a]	8,489
Total	5,700	8,489	55,289

[a] Under normal operating conditions, the waste sludge phase separator tank is decanted every 3-4 days.

Table 11.2-10

Chemical Waste Subsystem Sources

Source	Regular Daily Flow (gpd)	Irregular Flow (gpd)	Maximum Daily Flow (gpd)
Detergents drains/shop decontamination solutions	1,000		2,000
Laboratory drains	500		500
Decontamination drains reactor and turbine buildings		1,000	1,000
From floor drain or equipment drain subsystem		15,000	15,000
Filter demineralizer chemical cleaning solutions		Infrequent 2,000	2,000
Battery room drains		Infrequent 100	100
Total	1,500		20,600

Table 11.2-11

Radwaste System Process Flow Diagram Data

	Equipment Drain Subsystem						
Flow path	1	2	3	4	5	6	8
Batches/day (normal)	8.5	4.1	1.1	6.3	1.0/6.8	4.0/7.4	1.0/30.0
Batches/day (maximum)	63.4	15.8	1.1	6.3	1.0	4.0/2.0	-
Volume/batch (gal)	455	909	909	909	2430	13,500	4000
Normal daily volume (gal)	3860	3755	1000	5726	-	-	-
Normal activity (μCi/cm^3)	5.16E-2	1.30E-2	7.18E-5	1.91E-5	1.59E-2	7.18E-6	9.22E-7
Maximum daily volume (gal)	28,800	14,400	1000	5726	2430	27,000	4000
Maximum activity (μCi/cm^3)	1.72E0	4.32E-1	2.39E-3	1.26E-2	5.32E-1	3.59E-4	3.08E-5
Flow rate (gal/minute)	50	50	50	50	53	450	Batch
Daily activity (μCi/day)							
Normal	7.19E5	1.76E5	2.59E2	3.94E2	-	-	-
Maximum	2.51E7	6.14E6	9.05E3	2.73E3	4.89E6	3.67E4	4.66E2

Table 11.2-11

Radwaste System Process Flow Diagram Data (Continued)

	Equipment Drain Subsystem (Continued)						
Flow path	9	33	12	14	17	18	19
Batches/day (normal)	1.0	1.0	1.0	1.0	1.5	2.2	1.1
Batches/day (maximum)	7.0	7.0	7.0	7.0	63.4	16.5	1.1
Volume/batch (gal)	15,000	15,000	15,000	15,000	455	909	909
Normal daily volume (gal)	15,000	15,000	15,000	15,000	700	2000	1000
Normal activity ($\mu Ci/cm^3$)	1.16E-2	5.81E-3	1.16E-4	1.16E-4	7.18E-6	1.05E-6	1.44E-5
Maximum daily volume (gal)	104,826	104,826	104,826	104,826	28,800	15,000	1000
Maximum activity ($\mu Ci/cm^3$)	3.59E-1	3.53E-1	3.53E-3	3.53E-3	6.84E-2	1.00E-3	4.78E-4
Flow rate (gal/minute)	190	190	190	190	50	100	50
Daily activity (μCi/day)							
Normal	6.30E5	3.14E5	6.30E3	6.30E3	1.81E1	7.57E0	5.18E1
Maximum	2.04E7	2.00E7	2.03E5	2.04E5	1.81E5	7.57E3	1.81E3

Table 11.2-11

Radwaste System Process Flow Diagram Data (Continued)

	Equipment Drain Subsystem (Continued)					
Flow path	20	117	21	23	107	108
Batches/day (normal)	2.2	10./3.4	1.0/2.6	1.0/2.6	1.0/2.6	1.0/2.6
Batches/day (maximum)	2.2	1.0	3.7	3.74	3.7	3.7
Volume/batch (gal)	909	8489	15,000	15,000	15,000	15,000
Normal daily volume (gal)	2000		5700	5700	5700	5700
Normal activity (μCi/cm^3)	7.18E-7	7.18E-6	1.00E-6	4.99E-5	1.00E-6	1.00E-6
Maximum daily volume (gal)	2000	8489	55,489	55,489	55,489	55,489
Maximum activity (μCi/cm^3)	2.39E-5	3.59E-4	1.08E-2	1.08E-2	1.08E-4	1.08E-4
Flow rate (gal/minute)	50	50	190	190	190	190
Daily activity (μCi/day)						
Normal	5.18E0	-	2.56E3	1.03E3	2.06E1	2.06E1
Maximum	1.81E2	-	2.33E5	2.33E5	2.33E3	2.33E3

Table 11.2-11

Radwaste System Process Flow Diagram Data (Continued)

	Waste Surge Subsystem	
Flow path	104	37
Batches/day (normal)	1.0/yr	1.0/yr
Batches/day (maximum)	1.0	1.0
Volume/batch (gal)	56,720	56,720
Normal daily volume (gal)		
Normal activity ($\mu Ci/cm^3$)	7.18E-6	7.18E-6
Maximum daily volume (gal)	56,720	56,720
Maximum activity ($\mu Ci/cm^3$)	3.59E-4	3.59E-4
Flow rate (gal/minute)	Batch	190

Table 11.2-11

Radwaste System Process Flow Diagram Data (Continued)

	Chemical Waste Subsystem				
Flow path	27	109	120	121	122
Batches/day (normal)	1.0				
Batches/day (maximum)	2.0	2.0	2.0	1.0	3.7
Volume/batch solids(lb)			581	1162	314
liquids (gal)	1000	12,153	12,153	615	168
Normal daily volume (gal)	1000				
Normal activity solids (μCi/batch)					
liquids (μCi/cm^3)	1.05E-5	2.76E-3	2.71E-3	1.07E-1	1.07E-1
Maximum daily volume (gal)	2000	24,306	24,306		
Maximum activity solids (μCi/batch)					
liquids (μCi/cm^3)	1.05E-5	-	2.71E-3	1.07E-1	1.07E-1
Flow rate (gal/minute)	25	190	10	31	20

Table 11.2-11

Radwaste System Process Flow Diagram Data (Continued)

	Waste Sludge Subsystem (Condensate Backwash)			
Flow path	56	58	60	6
Batches/day (normal)	4.0/7.4	4.0/7.4	1/18.5	4/7.4
Batches/day (maximum)	4.0	4.0	1	4.0/2.0
Volume/batch solids (lb)	330	330	3300	-
liquids (gal)	13,500	13,500	7527	13,500
Normal daily volume (gal)	7300	7300	-	-
Normal activity solids (μCi/batch)	2.20E6	2.20E6	1.26E7	-
liquids ($\mu Ci/cm^3$)	7.18E-6	7.18E-6	7.18E-6	7.18E-6
Maximum daily volume (gal)	54,000	54,000	7527	27,000
Maximum activity solids (μCi/batch)	5.26E7	5.26E7	2.67E7	-
liquids ($\mu Ci/cm^3$)	3.59E-4	3.59E-4	3.59E-4	3.59E-4
Flow rate (gal/minute)	2500	450	20	450

Table 11.2-11

Radwaste System Process Flow Diagram Data (Continued)

	Waste Sludge Subsystem (Cleanup Backwash)		
Flow path	54	59	5
Batches/day (normal)	2.0/6.8	1.0/60	1.0/6.8
Batches/day (maximum)	2.0	1.0/60	1.0
Volume/batch solids (lb)	29.7	524	-
liquids (gal)	1215	1196	2430
Normal daily volume (gal)	360	-	-
Normal activity solids(μCi/batch)	2.55E7	1.19E8	-
liquids (μCi/cm^3)	1.59E-2	1.59E-2	1.59E-2
Maximum daily volume (gal)	2430	1196	2430
Maximum activity solids (μCi/batch)	7.68E8	1.80E8	-
liquids (μCi/cm^3)	5.32E-1	5.32E-1	5.32E-1
Flow rate (gal/minute)	270	20	53

Table 11.2-11

Radwaste System Process Flow Diagram Data (Continued)

	Waste Sludge Subsystem (Spent Resin)			
Flow path	64	119	69	71
Batches/day (normal)	1.0/66	1.0/67	1.0/165	-
Batches/day (maximum)	1.0/29	1.0/49	1.0/100	-
Volume/batch solids (lb)	1539	1539	1539	1539
liquids (gal)	746	746	746	3510
Normal activity solids (μCi/batch)	2.39E6	6.15E3	1.16E2	-
liquids (μCi/cm^3)	7.18E-6	7.18E-6	7.18E-6	7.18E-6
Maximum activity solids (μCi/batch)	2.74E7	2.34E5	1.26E2	-
liquids (μCi/cm^3)	3.59E-4	3.59E-4	3.59E-4	3.59E-4
Flow rate (gal/minute)	37	37	37	20

Table 11.2-11

Radwaste System Process Flow Diagram Data (Continued)

	Equipment Drain Subsystem				
Flow path	61	63	62	-	65
Batches/day (normal)	1.0	1.0/3.4	1.0/5.2	-	1.0/3.4
Batches/day (maximum)	2.9	1.1/1.0	1.0/5.2	-	1.0
Volume/batch solids (lb)	41.36	41.36	59.4	-	220
liquids (gal)	1692	1692	2430	-	501
Normal daily volume (gal)	1692	-	-	-	-
Normal activity solids (μCi/batch)	3.30E4	8.42E1	1.47E4	-	4.87E4
liquids (μCi/cm^3)	7.18E-6	7.18E-6	7.18E-6	-	7.18E-6
Maximum daily volume (gal)	4906	1861	-	-	501
Maximum activity solids (μCi/batch)	3.30E4	8.42E1	1.47E6	-	7.33E5
liquids (μCi/cm^3)	3.59E-4	3.59E-4	3.59E-4	-	9.88E-4
Flow rate (gal/minute)	376	376	540	-	20

Table 11.2-11

Radwaste System Process Flow Diagram Data (Continued)

NOTES:

Process Diagram, Figure 11.2-1, forms part of this data.

The following definitions are used for this data:

Normal volume -	Expected flow during steady state normal operation
Maximum volume -	Maximum expected flow during unsteady state operation such as startup, shutdown, etc.
Normal activity -	Activity level expected during operation with no fuel leaks and corrosion product reactor water activity concentration of 0.1 $\mu Ci/cm^3$
Maximum activity -	Activity level expected during operation with fuel leak rate equivalent to reactor water activity concentration of 2.3 $\mu Ci/cm^3$ and design basis noble radiogas release rate of 100,000 µCi/sec (corrosion and fission products present)

Maximum volume and maximum activity are not concurrent.

For activity values: E-1 = number x 10^{-1}; E1 = number x 10^{1}
E-4 = number x 10^{-4}; E4 = number x 10^{4}

Fractional values on tables denote the number of items per occurrence divided by the number of days between each occurrence (i.e., 1/30 batches/day means one batch processed every 30 days).

Waste system input activities are based on a reactor water-to-steam partition coefficient of 1.0E-3.

Values for design purposes only. Actual flows and activities may differ.

Table 11.2-12

Radwaste Process Equipment Design Basis

Equipment	Radioactivity Decontamination Factor	
Deep bed demineralizers	Soluble	100
	Insoluble	50
Precoat filters	Soluble	1
	Insoluble	2
Evaporators (influent/distillate)[a]	1000	

[a] Concentration factor (influent/bottoms) = 0.6/32.

Table 11.2-13

Liquid Radwaste Equipment

	Material of Construction	Quantity	Size or Capacity	Remarks
EQUIPMENT DRAIN SUBSYSTEM				
Waste collector tank	Carbon steel	1	20,000 gal	Note 11
Waste collector pump	Stainless steel	1	190 gal/minute @ 155 ft TDH	
Waste sample tanks	Carbon steel	2	20,000 gal	Note 11
Waste sample pumps	Stainless steel	2	190 gal/minute @ 100 ft TDH	
Waste collector filter	Carbon steel	1	188 ft^2 of filter area	Notes 3, 4
Waste filter hold pump	Stainless steel	1	75 gal/minute @ 50 ft TDH	Note 3
Waste demineralizer	Carbon steel	1	65 ft^3 resin bed	Note 2
Waste surge tank	Carbon steel	1	75,000 gal	Note 11
Waste surge pump	Stainless steel	1	190 gal/minute @ 155 ft TDH	
FLOOR DRAIN SUBSYSTEM				
Floor drain collector tank	Carbon steel	1	20,000 gal	Note 11
Floor drain collector pump	Stainless steel	1	190 gal/minute @ 155 ft TDH	
Floor drain sample tank	Carbon steel	1	20,000 gal	Note 11
Floor drain sample pump	Stainless steel	1	190 gal/minute @ 100 ft TDH	
Floor drain filter	Carbon steel	1	188 ft^2 of filter area	Notes 3, 4
Floor drain filter hold pump	Stainless steel	1	75 gal/minute @ 50 ft TDH	Note 3
Floor drain demineralizer	Carbon steel	1	65 ft^3 resin bed	Note 2

Table 11.2-13

Liquid Radwaste Equipment (Continued)

	Material of Construction	Quantity	Size or Capacity	Remarks
CHEMICAL WASTE SUBSYSTEM				
Detergent drain tank	Carbon steel	2	1600 gal	Note 12
Detergent drain pumps	Stainless steel	2	27 gal/minute @ 90 ft TDH	
Detergent drain filter	Carbon steel	1	50 gal/minute	
Chemical waste tanks	Stainless steel	2	15,000 gal	Note 12
Chemical waste pumps	Stainless steel	2	285 gal/minute @ 200 ft TDH	
Concentrator feed pumps	Stainless steel	2	30 gal/minute @ 95 ft TDH	
Concentrator (evaporator)	Stainless steel	2	10 gal/minute	Note 6
Heating element	Stainless steel/carbon steel	2		Note 10
Concentrator recycle pump	Stainless steel	2	2300 gal/minute @ 40 ft TDH	Note 5
Concentrator condenser	Stainless steel/carbon steel	2		Note 7
Distillate tanks	Stainless steel	2	15,000 gal	Note 12
Distillate pumps	Stainless steel	2	142 gal/minute @ 230 ft TDH	
Polishing demineralizer	Carbon steel	1	65 ft^3 resin bed	Note 2
AUXILIARY EQUIPMENT				
Waste precoat tank	Carbon steel	1	210 gal	Notes 4, 8
Waste precoat pump	Ductile iron	1	325 gal/minute @ 50 ft TDH	Note 3
Waste filter aid tank	Carbon steel	1	630 gal	Notes 4, 8
Waste filter aid metering pump	Stainless steel	2	0 to 154 gal/hr @ 145 psi	
Resin addition tank	Stainless steel	1	1200 gal	Note 12
Chemical addition tanks	Carbon steel	2	200 gal	Note 9
Chemical addition pumps	Carpenter 20 steel	2	79 gal/hr	

Table 11.2-13

Liquid Radwaste Equipment (Continued)

NOTES:

1.	Unless otherwise noted, the design pressure of the equipment is 150 psig, design temperature is 150°F.
2.	Vessel is rubber lined, internals are stainless steel.
3.	Design temperature is 220°F.
4.	Vessel is lined with phenolic/epoxy cross linked, alkaline polymerized coating, internals are stainless steel.
5.	Design temperature is 300°F.
6.	Design pressure is 30 psig, design temperature is 274°F.
7.	Design flow is: coolant inlet-750 gpm @ 105°F, outlet -120°F; design pressure: shell side - 30 psig, tube side - 150 psig; design temperature - 274°F.
8.	Design pressure is atmospheric.
9.	Vessel is lined with acid-resistant high-bake phenolic coating.
10.	Design flows: shell side - 6300 lb/hr @ 274°F, tube side - 2300 gpm @ 218°F; design pressures: shell side - 60 psig, tube side - 75 psig; design temperature - 307°F.
11.	Tank design pressure is atmospheric plus static head pressure, design temperature is 140°F.
12.	Tank design pressure is atmospheric plus static head pressure, design temperature is 150°F.

Table 11.2-14

Annual Releases of Radioactive Material as Liquid

Nuclide	Half-Life (days)	Concentration in Primary Coolant (μCi/ml)	High Purity (Ci)	Low Purity (Ci)	Total Laws (Ci)	Adjusted Total (Ci/yr)[a]	Total (Ci/yr)
CORROSION AND ACTIVATION PRODUCTS							
^{24}Na	6.25E-1	8.79E-3	0.00036	0.00043	0.00079	0.00688	0.00692
^{32}P	1.43E1	2.06E-4	0.00001	0.00002	0.00003	0.00027	0.00027
^{51}Cr	2.78E1	5.14E-3	0.00027	0.00052	0.00081	0.00704	0.00703
^{54}Mn	3.03E2	6.18E-5	0.00000	0.00001	0.00001	0.00008	0.00008
^{56}Mn	1.07E-1	4.28E-2	0.00052	0.00034	0.00085	0.00747	0.00745
^{55}Fe	9.50E2	1.03E-3	0.00005	0.00010	0.00017	0.00143	0.00147
^{59}Fe	4.50E1	3.08E-5	0.00000	0.00000	0.00000	0.00004	0.00004
^{58}Co	7.13E1	2.06E-4	0.00001	0.00002	0.00003	0.00028	0.00028
^{60}Co	1.92E2	4.12E-0	0.00002	0.00004	0.00006	0.00058	0.00058
^{65}Ni	1.07E-1	2.57E-4	0.00000	0.00000	0.00000	0.00004	0.00004
^{64}Cu	5.33E-1	2.91E-2	0.00111	0.00128	0.00239	0.02098	0.02098
^{65}Zn	2.45E2	2.06E-4	0.00001	0.00002	0.00003	0.00028	0.00028
^{69m}Zn	5.57E-1	1.94E-3	0.00007	0.00009	0.00017	0.00146	0.00147
^{69}Zn	3.96E-2	0.0	0.00008	0.00009	0.00018	0.00153	0.00157
^{187}W	9.96E-1	2.98E-4	0.00001	0.00002	0.00003	0.00028	0.00028
^{239}Np	2.35E0	7.10E-3	0.00036	0.00060	0.00094	0.00831	0.00790
FISSION PRODUCTS							
^{83}Br	1.00E-1	2.42E-3	0.00003	0.00002	0.00004	0.00039	0.00039
^{84}Br	2.21E-2	3.67E-3	0.00000	0.00000	0.00000	0.00003	0.00003
^{89}Rb	1.07E-2	3.60E-3	0.00001	0.00002	0.00002	0.00022	0.00022
^{89}Sr	5.20E1	1.03E-5	0.00001	0.00001	0.00002	0.00015	0.00015
^{91}Sr	4.03E-1	3.82E-3	0.00013	0.00014	0.00026	0.00233	0.00231

Table 11.2-14

Annual Releases of Radioactive Material as Liquid (Continued)

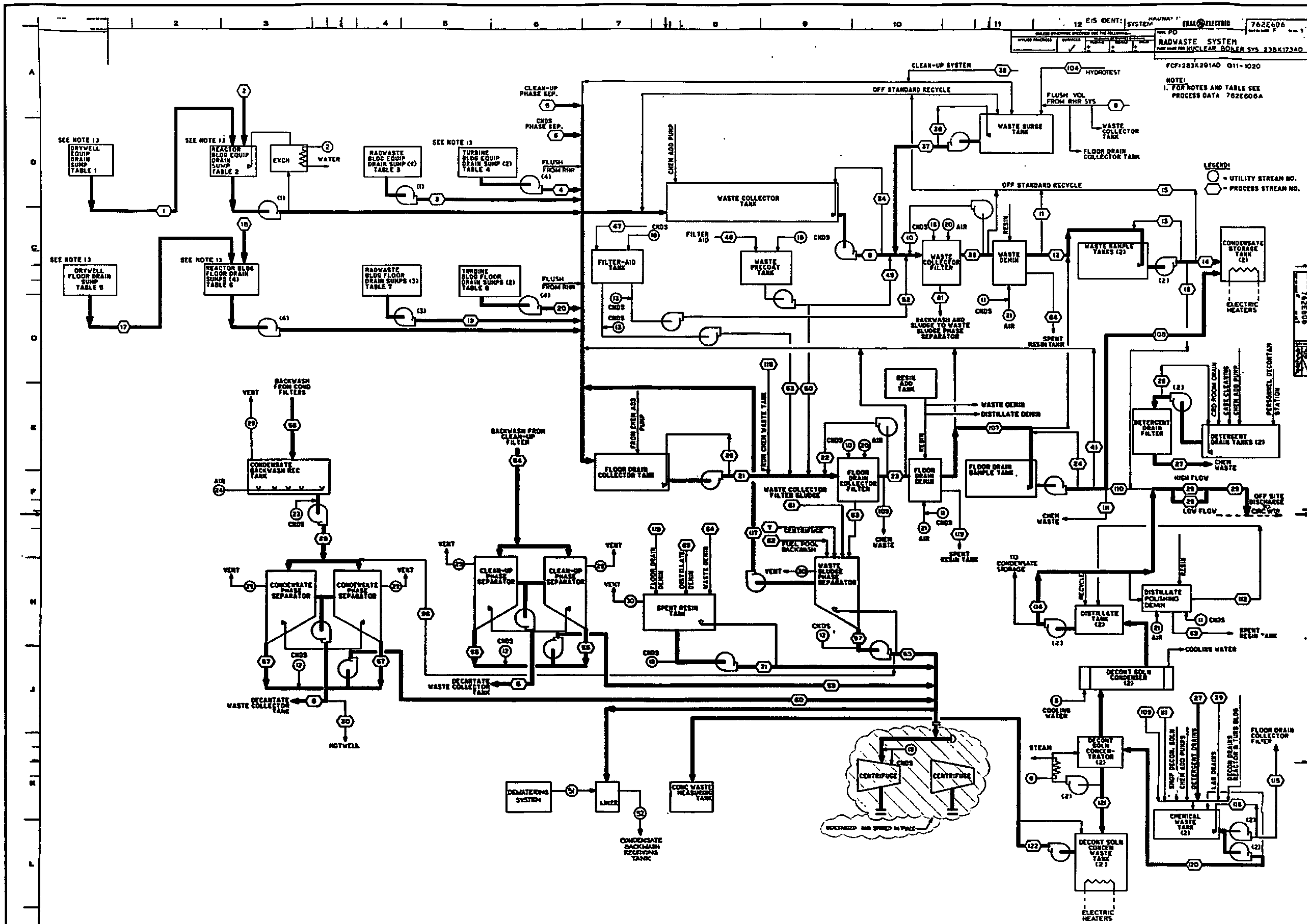
Nuclide	Half-Life (days)	Concentration in Primary Coolant (µCi/ml)	High Purity (Ci)	Low Purity (Ci)	Total Laws (Ci)	Adjusted Total (Ci/yr)[a]	Total (Ci/yr)
^{91m}Y	3.47E-2	0.0	0.00008	0.00008	0.00017	0.00146	0.00147
^{91}Y	5.88E1	4.12E-5	0.00000	0.00001	0.00001	0.00007	0.00007
^{92}Sr	1.13E-1	8.60E-3	0.00012	0.00007	0.00018	0.00159	0.00157
^{92}Y	1.47E-1	5.29E-3	0.00021	0.00016	0.00038	0.00328	0.00325
^{93}Y	4.25E-1	3.83E-3	0.00014	0.00014	0.00027	0.00241	0.00241
^{98}Nb	3.54E-2	3.11E-3	0.00001	0.00000	0.00001	0.00008	0.00008
^{99}Mo	2.798E0	2.04E-3	0.00010	0.00018	0.00028	0.00244	0.00241
^{99m}Tc	2.5E-1	1.85E-2	0.00054	0.00054	0.00107	0.00893	0.00937
^{101}Tc	9.72E-3	6.56E-2	0.00000	0.00000	0.00000	0.00002	0.00002
^{103}Ru	3.96E1	2.06E-5	0.00000	0.00000	0.00000	0.00003	0.00003
^{103m}Rh	3.96E-2	0.0	0.00000	0.00000	0.00000	0.00003	0.00003
^{104}Tc	1.25E-2	5.87E-2	0.00000	0.00000	0.00001	0.00006	0.00006
^{105}Ru	1.85E-1	1.80E-3	0.00004	0.00003	0.00006	0.00055	0.00058
^{105m}Rh	5.21E-4	0.0	0.00004	0.00003	0.00006	0.00056	0.00059
^{105}Rh	1.50E0	0.0	0.00001	0.00001	0.00002	0.00018	0.00019
^{129m}Te	3.40E1	4.11E-5	0.00000	0.00000	0.00001	0.00005	0.00005
^{129}Te	4.79E-2	0.0	0.00000	0.00000	0.00000	0.00003	0.00003
^{131m}Te	1.25E0	1.00E-4	0.00000	0.00001	0.00001	0.00010	0.00010
^{131}Te	1.74E-2	0.0	0.00000	0.00000	0.00000	0.00002	0.00002
^{131}I	8.05E0	5.11E-3	0.00027	0.00049	0.00077	0.00643	0.00671
^{132}Te	3.25E0	1.02E-5	0.00000	0.00000	0.00000	0.00001	0.00001
^{132}I	9.58E-2	2.41E-2	0.00026	0.00016	0.00041	0.00345	0.00367
^{133}I	8.75E-1	1.93E-2	0.00084	0.00115	0.00199	0.01667	0.01783
^{134}I	3.67E-2	5.26E-2	0.00010	0.00006	0.00017	0.00144	0.00147

Table 11.2-14

Annual Releases of Radioactive Material as Liquid (Continued)

Nuclide	Half-Life (days)	Concentration in Primary Coolant (μCi/ml)	High Purity (Ci)	Low Purity (Ci)	Total Laws (Ci)	Adjusted Total (Ci/yr)[a]	Total (Ci/yr)
^{134}Cs	7.49E2	3.09E-5	0.00008	0.00081	0.00089	0.00741	0.00776
^{135}I	2.79E-1	1.77E-2	0.00050	0.00044	0.00094	0.00788	0.00829
^{136}Cs	1.30E1	2.05E-5	0.00005	0.00051	0.00057	0.00473	0.00493
^{137}Cs	1.10E4	7.21E-5	0.00020	0.00188	0.00206	0.01730	0.01783
^{137M} Ba	1.77E-3	0.0	0.00018	0.00175	0.00193	0.01618	0.01678
^{138}Cs	2.24E-2	7.34E-3	0.00022	0.00065	0.00087	0.00724	0.00755
^{139}Ba	5.76E-2	8.09E-3	0.00004	0.00002	0.00006	0.00053	0.00056
^{140}Ba	1.28E1	4.11E-4	0.00002	0.00004	0.00006	0.00053	0.00056
^{140}La	1.67E0	0.0	0.00000	0.00001	0.00001	0.00011	0.00011
^{141}La	1.62E-1	0.0	0.00001	0.00001	0.00002	0.00017	0.00018
^{141}Ce	3.24E1	3.08E-5	0.00000	0.00000	0.00000	0.00004	0.00004
^{142}La	6.39E-2	4.08E-3	0.00003	0.00002	0.00004	0.00036	0.00038
^{143}Ce	1.38E0	3.01E-5	0.00000	0.00000	0.00000	0.00003	0.00003
^{143}Pr	1.37E1	4.11E-5	0.00000	0.00000	0.00001	0.00005	0.00005
All others	1.32E-2	1.38E-2	0.00000	0.00001	0.00007	0.00007	
Total (except tritium)		4.31E-01	0.00719	0.01307	0.02026	0.17962	0.17834
Total release						13.0	13.0

[a] Adjusted total includes an additional 0.15 ci/yr with the same isotopic distribution as the calculated source term to account for anticipated occurrences such as operator errors resulting in unplanned releases.



Columbia Generating Station
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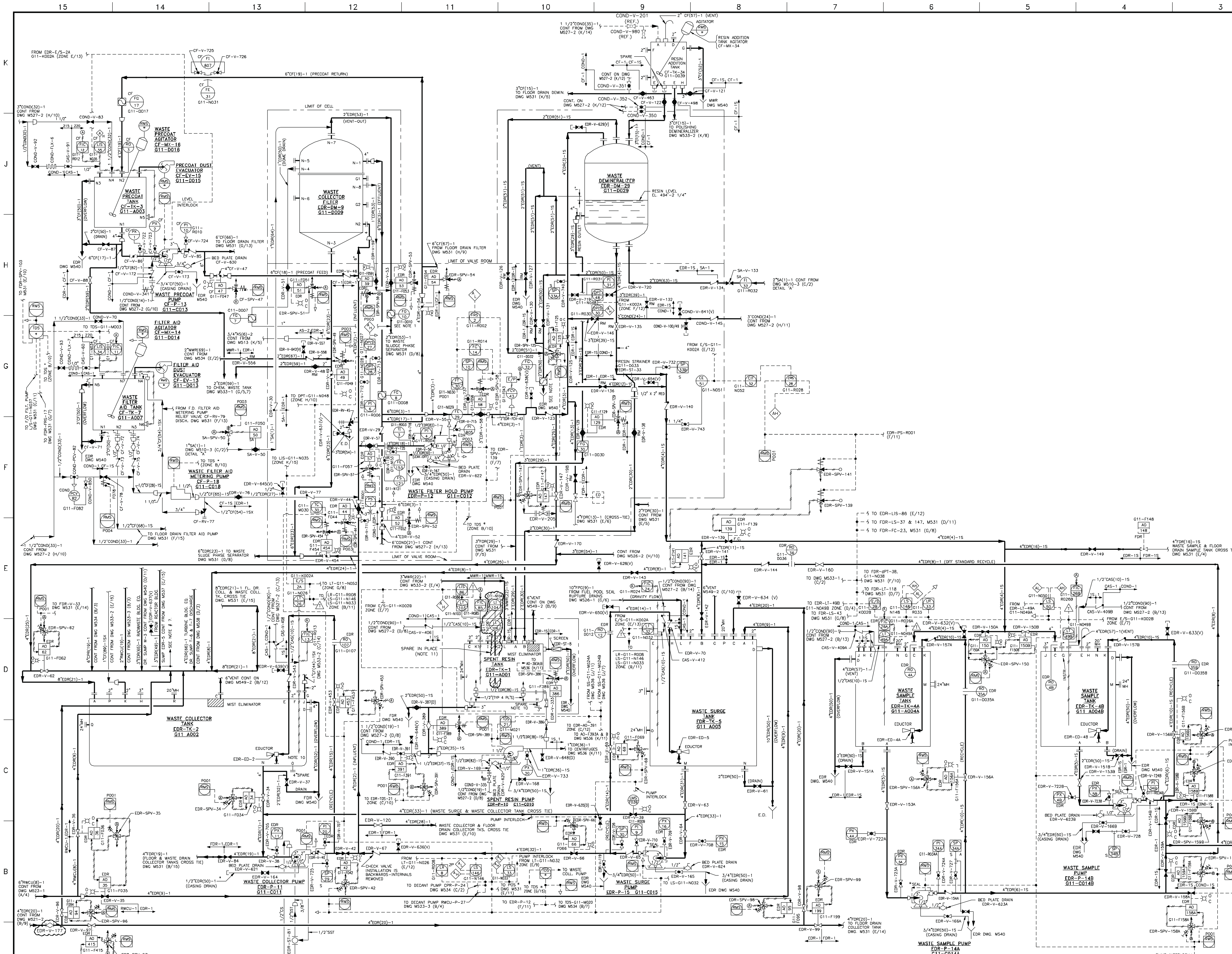
Radwaste System Process Diagram

Draw. No.	2G11-04,6	Rev.	8	Figure	11.2-1



NOTES:

1. LOCATE SIGHT GLASSES OUTSIDE OF SHIELDED AREA. RETURN PIPING DOWNSTREAM OF SIGHT GLASSES TO BE IN SHIELDED AREA.
2. ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
3. EXCEPT AS NOTED IN NOTE 7 & 8 ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS IN THE EDR SYSTEM ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS: II+
   SEISMIC CATEGORY: II
   CODE GROUP: D+
   (+ SEE NOTE 12, WNP-2 SPEC., SECTION 15B.1, TABLE 2 NOTES)
4. ALL PIPING, VALVES, & ASSOCIATED COMPONENTS, NOT IN THE EDR SYSTEM ON THIS DWG SHALL BE CLASSIFIED IN ACCORDANCE WITH THE CLASSIFICATIONS SHOWN ON THEIR RESPECTIVE SYSTEM FLOW DIAGRAMS.
5. ALL INSTRUMENT ROOT VALVES NOT LABELED TO BE 3/4" GLOBE VALVES UNLESS OTHERWISE NOTED.
6. ALL INSTRUMENT PIPING AND TUBING DOWNSTREAM OF THE INSTRUMENT ROOT VALVES SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS: II
   SEISMIC CATEGORY: II
   CODE GROUP: D
7. PIPING AND ASSOCIATED COMPONENTS IN SUBSYSTEM EDR (49) SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS: II
   SEISMIC CATEGORY: II
   CODE GROUP: D
8. ALL PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE WITH THE SUBSYSTEM DESIGNATION "SYSTEM" (50) THRU (59) SHALL BE CLASSIFIED AS CODE GROUP D WITH QUALITY CLASS AND SEISMIC CATEGORY PROVIDED BY THE APPLICABLE NOTES ON THIS DRAWING, EXCEPT WHERE THE APPLICABLE NOTES CALL FOR QUALITY CLASS II+, PIPING SHALL BE QUALITY CLASS II.
9. ALL INSTRUMENTATION ON THIS DRAWING TO BE IDENTIFIED BY THE PREFIX "EDR" UNLESS OTHERWISE NOTED.
10. SPARE BOTTOM TANK NOZZLES HAVE REMOVABLE PLUGS INSTALLED FLUSH WITH TANK BOTTOMS TO ELIMINATE DEAD LEG CRUD TRAPS.
11. SLUDGE LEVEL EDR-LE-106 AND EDR-LI-104 ARE SPARED IN PLACE. THEY HAVE BEEN HISTORICALLY UNRELIABLE. SLUDGE LEVEL IS PRESENTLY MORE RELIABLY DETERMINED AND CONTROLLED BY USING THE PPM (SOLID WASTE PROCESSING SYSTEM).
12. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.

LEGEND:

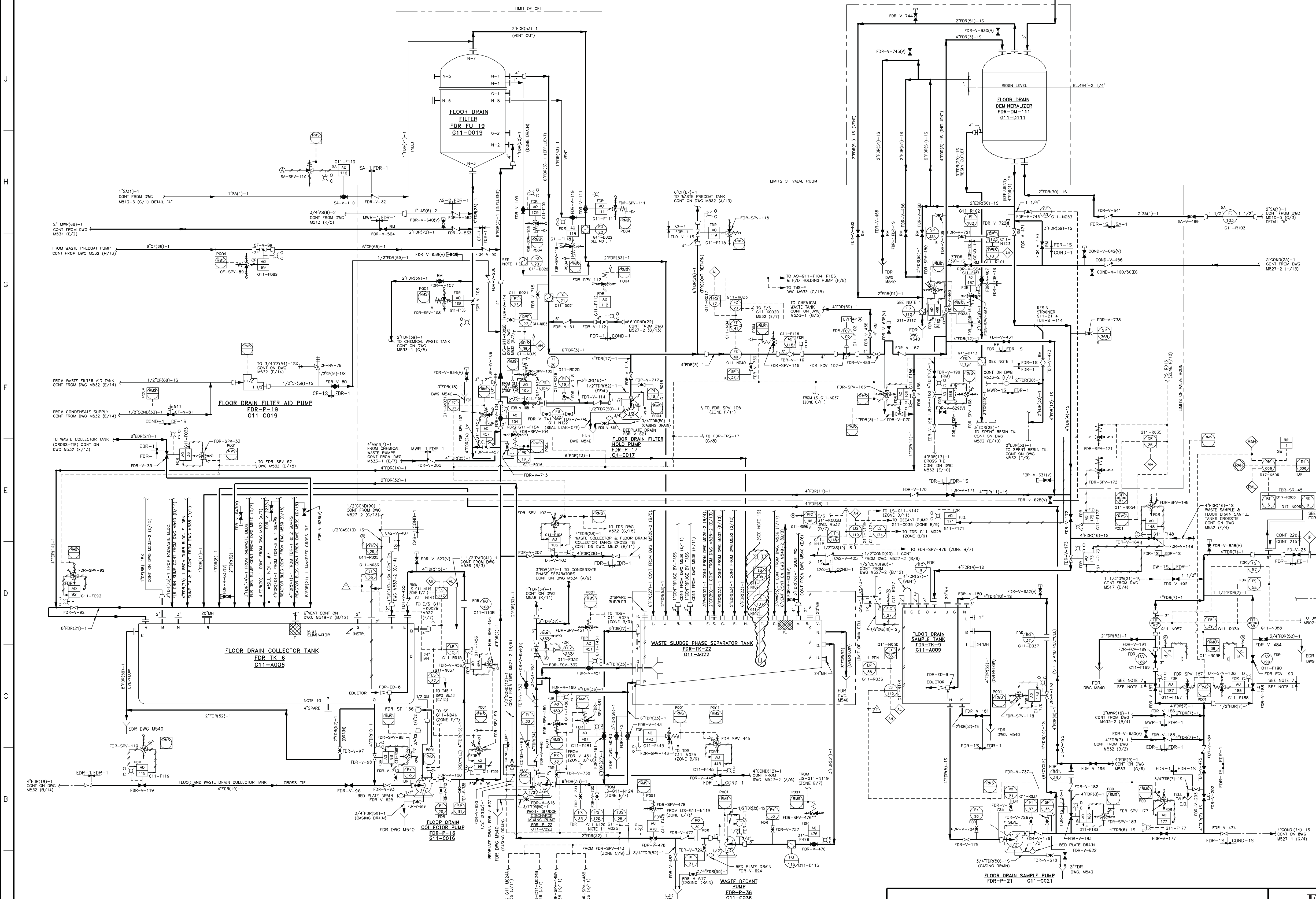
1. ALL VALVES SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE. (EXCEPT WHERE NOTED).
2. ALL VALVES SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE. (EXCEPT WHERE NOTED).
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

REFERENCES TO CONTRACT NUMBERS, SUCH AS 215, 206, 220, FOUND IN THE NOTES ON THIS DRAWING REFER TO ORIGINAL CONSTRUCTION DESIGN REQUIREMENTS.

CURRENT DESIGN REQUIREMENTS ARE DOCUMENTED IN THE STATION DESIGN SPECIFICATIONS AND ACTIVE CONTRACT SPECIFICATIONS WHICH STILL APPLY.

FSAR FIG.

Columbia Generating Station Final Safety Analysis Report	Flow Diagram Radioactive Waste System Equipment Drain Processing		
	Draw. No. M532	Rev. 79	Figure 11.2-2



NOTES:

1. LOCATE SIGHT GLASSES OUTSIDE OF SHIELDED AREA. RETURN PIPING DOWN STREAM OF SIGHT GLASSES TO BE IN SHIELDED AREA.
2. ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
3. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
4. EXCEPT AS NOTED IN NOTE 7 & 8 ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS IN THE FDR SYSTEM ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II+
   SEISMIC CATEGORY II
   CODE GROUP D+
5. ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS NOT IN THE FDR SYSTEM ON THIS DWG. SHALL BE CLASSIFIED IN ACCORDANCE WITH THE CLASSIFICATIONS SHOWN ON THEIR RESPECTIVE SYSTEM FLOW DIAGRAMS.
6. ALL INSTRUMENT PIPING AND TUBING DOWNSTREAM OF THE INSTRUMENT ROOT VALVES SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP D
7. PIPING AND ASSOCIATED COMPONENTS IN SUBSYSTEMS FDR(40) & FDR(41) AND FDR(7) DOWNSTREAM OF FDR-V-187 AND FDR-V-188 SHALL BE CLASSIFIED AS FOLLOWS:
   QUALITY CLASS II
   SEISMIC CATEGORY II
   CODE GROUP D
8. ALL PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE AND OPEN TO THE ATMOSPHERE WITH THE SUBSYSTEM DESIGNATION "SYSTEM"(50) THRU (59) SHALL BE CLASSIFIED AS CODE GROUP D WITH QUALITY CLASS AND SEISMIC CATEGORY PROVIDED BY THE APPLICABLE NOTES ON THIS DRAWING, EXCEPT WHERE THE APPLICABLE NOTES CALL FOR QUALITY CLASS II+, PIPING SHALL BE QUALITY CLASS II.
9. ALL INSTRUMENTATION ON THIS DRAWING TO BE IDENTIFIED BY THE PREFIX "FDR" UNLESS OTHERWISE NOTED.
   (+SEE NOTE 12, WNP-2 SPEC. SECTION 15B.1, TABLE 2 NOTES.)
10. SPARE BOTTOM TANK NOZZLES HAVE REMOVABLE PLUGS INSTALLED FLUSH WITH TANK BOTTOMS TO ELIMINATE DEAD LEG CRUD TRAPS.
11. FDR-PS-120 CONTACTS JUMPERED CLOSED TO INDICATE NORMAL SUCTION PRESSURE.
12. SLUDGE LEVEL FDR-LE-107 AND FDR-LS-109 ARE SPARED IN PLACE. THEY HAVE BEEN HISTORICALLY UNRELIABLE. SLUDGE LEVEL IS PRESENTLY MORE RELIABLY DETERMINED AND CONTROLLED BY USING THE PPM (SOLID WASTE PROCESSING SYSTEM).
13. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.

LEGEND

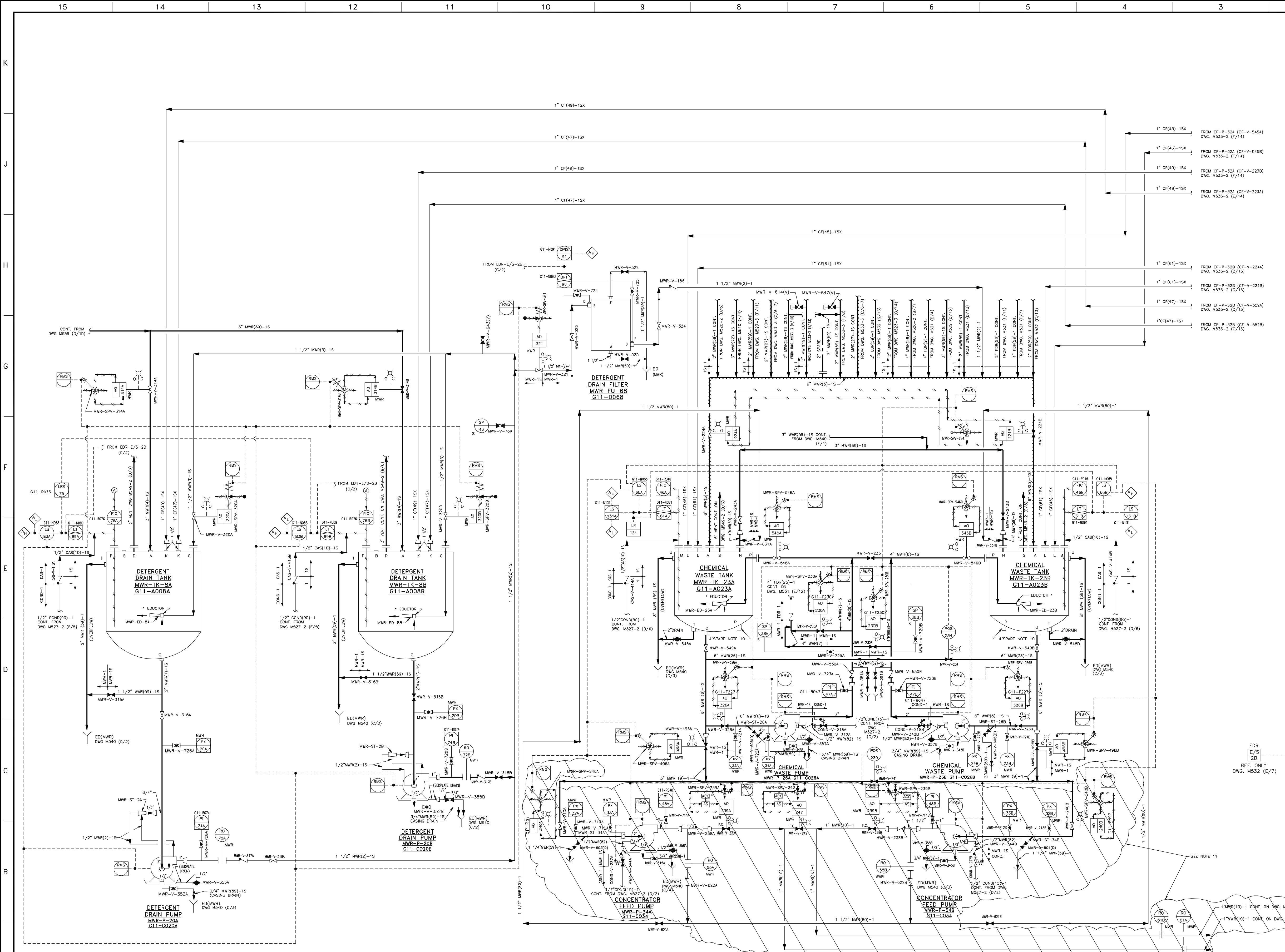
1. ALL VALVES SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

REFERENCES TO CONTRACT NUMBERS, SUCH AS 215, 206, 220, FOUND IN THE NOTES ON THIS DRAWING REFER TO ORIGINAL CONSTRUCTION DESIGN REQUIREMENTS.

CURRENT DESIGN REQUIREMENTS ARE DOCUMENTED IN THE STATION DESIGN SPECIFICATIONS AND ACTIVE CONTRACT SPECIFICATIONS WHICH STILL APPLY.

FSAR FIG.

Columbia Generating Station Final Safety Analysis Report	Flow Diagram Radioactive Waste System Floor Drain Processing		
	Draw. No. M531	Rev. 76	Figure 11.2-3



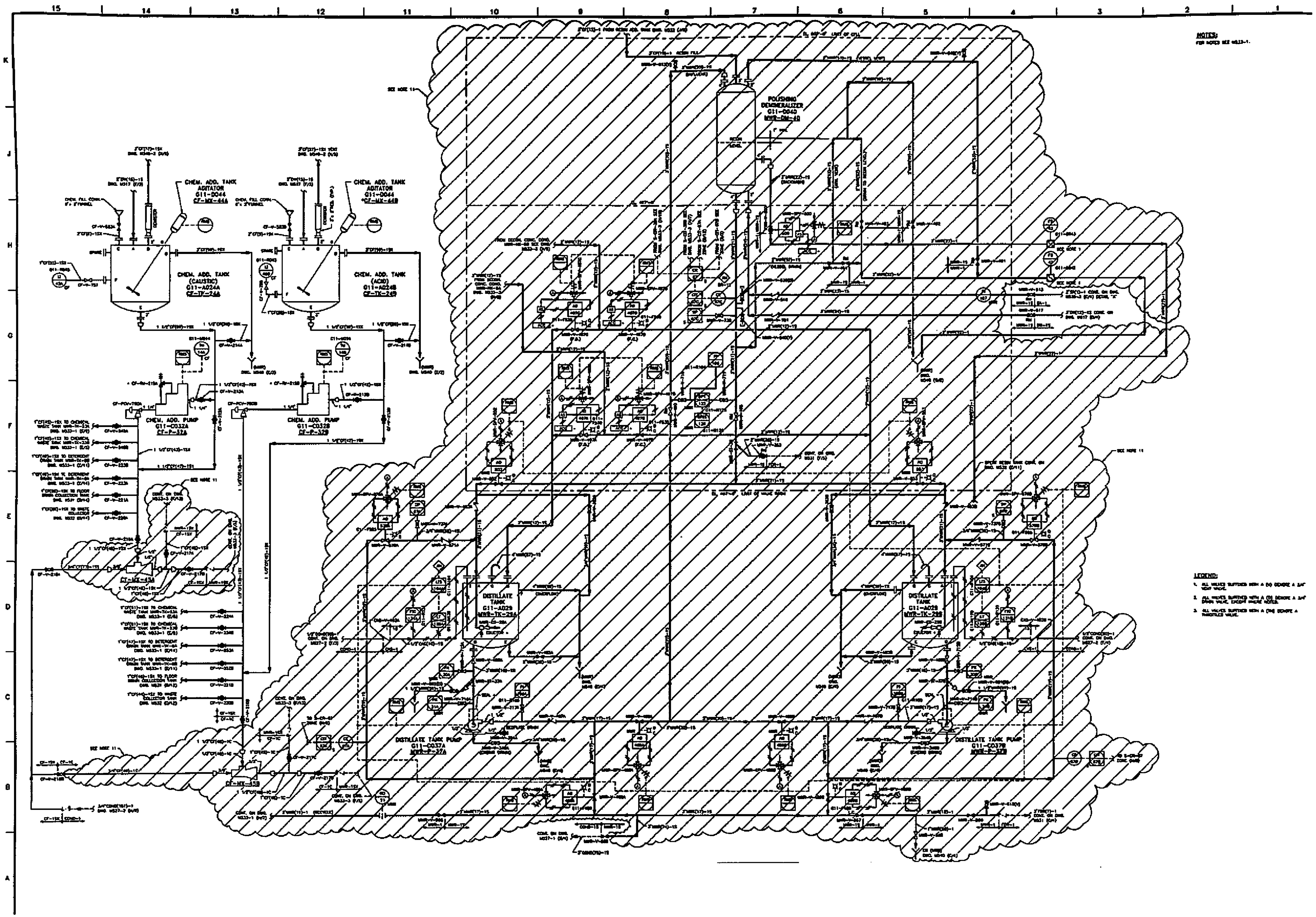
NOTES:

1. LOCATE SIGHT GLASSES OUTSIDE OF SHIELDED AREA. RETURN PIPING DOWNSTREAM OF SIGHT GLASSES TO BE IN SHIELDED AREA.
2. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
3. ALL PIPING VALVES AND ASSOCIATED COMPONENTS IN THE MWR AND PVR SYSTEMS SHOWN ON THIS DWG. ARE CLASSIFIED AS FOLLOWS UNLESS OTHERWISE NOTED:
   SEISMIC CATEGORY II
   QUALITY CLASS II+
   CODE GROUP D+
   [CODE GROUP D FOR INSTRUMENT PIPING AND SUBSYSTEM MWR-(57)-1S AND MWR-(59)-1SX PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE CONNECTED TO VESSELS, TANKS, PUMPS, HEAT EXCHANGERS, ETC., AND MWR(30)-1S].
4. ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS NOT IN THE MWR AND PVR SYSTEMS ON THIS DWG. SHALL BE CLASSIFIED IN ACCORDANCE WITH THE CLASSIFICATION SHOWN ON THEIR RESPECTIVE FLOW DIAGRAMS.
5. LOCAL CONDUCTIVITY ELEMENTS, ESPECIALLY IN LINES WITH INTERMITTENT FLOWS, ARE TO BE KEPT WET AT ALL TIMES BY A LOOP SEAL.
6. LINES SHOWN AS ARE ELECTRICAL HEAT TRACED.
7. ALL ITEMS MARKED WITH AN ASTERISK(*) ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
8. ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS IN THE CF SYSTEM AND/OR REFERRING TO THIS NOTE SHALL BE CLASSIFIED AS FOLLOWS:
   SEISMIC CATEGORY II
   QUALITY CALSS II
   CODE GROUP D
9. ALL PIPING DOWNSTREAM OF THE LAST ISOLATION VALVE AND OPEN TO THE ATMOSPHERE WITH THE SUBSYSTEM DESIGNATION "SYSTEM" (50) THRU (59) SHALL BE CLASSIFIED AS CODE GROUP D WITH QUALITY CLASS AND SEISMIC CATEGORY PROVIDED BY THE APPLICABLE NOTES ON THIS DRAWING, EXCEPT WHERE THE APPLICABLE NOTES CALL FOR QUALITY CLASS II+, PIPING SHALL BE QUALITY CLASS II, (+ SEE NOTE 12, WNP-2 SPEC., SECTION 15B.1 TABLE 2 NOTES.)
10. SPARE BOTTOM TANK NOZZLES HAVE REMOVABLE PLUGS INSTALLED FLUSH WITH TANK BOTTOMS TO ELIMINATE DEAD LEG CRUD TRAPS.
11. SCOPED AND HASHED AREAS ARE DEFERRED FOR LATER PRE-OPERATIONAL TESTING AND ARE NOT PRESENTLY OPERATIONAL.
12. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.

LEGEND

1. ALL VALVES SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE. (EXCEPT WHERE NOTED)
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

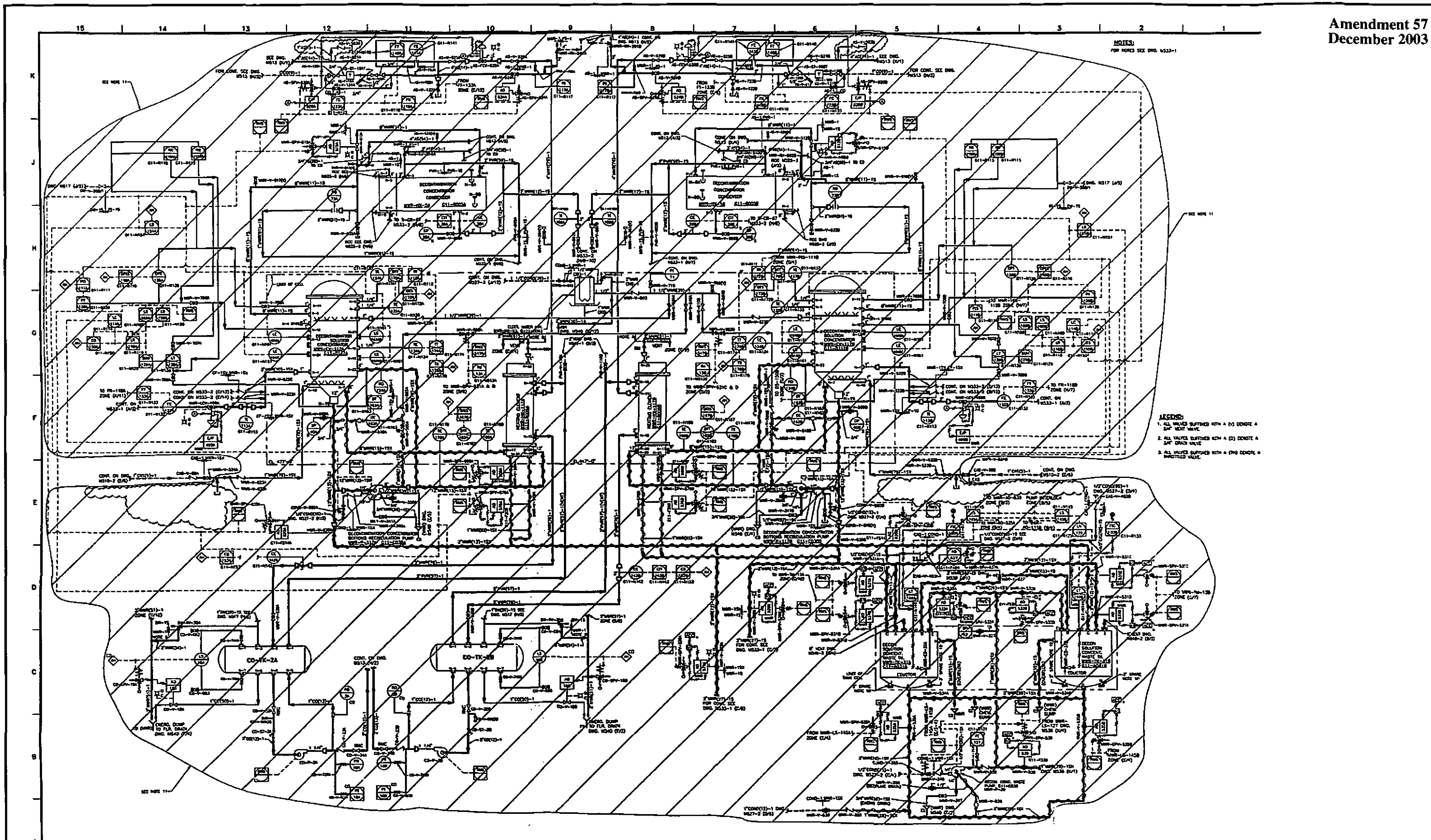
Columbia Generating Station Final Safety Analysis Report	Flow Diagram Chemical Waste Processing
	Draw. No. M533-1 | Rev. 24 | Figure 11.2-4.1



Columbia Generating Station
Final Safety Analysis Report

Flow Diagram Chemical Waste Processing

Draw. No.	M533-2	Rev.	21	Figure	11.2-4.2



Columbia Generating Station
Final Safety Analysis Report

Flow Diagram Chemical Waste Processing

Draw. No.	M533-3	Rev.	29	Figure	11.2-4.3

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

11.3.1 DESIGN BASES

The objective of the gaseous waste management system is to process and control the release of gaseous radioactive effluents to the site environs so as to maintain as low as reasonably achievable the exposure of persons in unrestricted areas to radioactive gaseous effluents, during normal and anticipated operational occurrences. This is to be accomplished while maintaining occupational exposure as low as is reasonably achievable (ALARA) and without limiting plant operation or availability.

The gaseous waste management systems are designed to limit the dose to offsite persons from routine station releases to significantly less than the limits specified in 10 CFR 20 and to operate within the emission rate limits established in the Technical Specifications.

To evaluate the offgas system design compliance with the limits of 10 CFR 20, an annual average noble radiogas source term (based on 30-minute decay) of 100,000 µCi/sec of the "1971 Mixture" as discussed in Section 11.1 is used. The noble radiogas effluent release rate from the charcoal adsorbers is about 49-59 µCi/sec based upon 30 scfm air inleakage and injection. The isotopic composition is given in Table 11.3-1 in units of µCi/sec and Ci/yr.

To evaluate that the annual average exposure at the site boundary during normal operation and anticipated operational occurrences from gaseous effluents does not exceed the dose objectives of Appendix I to 10 CFR 50, an average source term release rate derived from field measurements (1993-2003) is used.

Equipment and components used to collect, process, or store gaseous radioactive waste are not designed as Seismic Category I. To evaluate that equipment failure will not result in offsite whole body doses exceeding 0.5 rem in two hours, the Technical Specification release rate limit was used, consistent with the guidance presented in Reference 11.3-7.

The gaseous radwaste equipment is selected, arranged, and shielded to maintain occupational exposure ALARA. The system was designed prior to the issuance of Regulatory Guide 8.8. However, the system incorporates substantially the guidance provided in this regulatory guide. The gaseous effluent treatment system conforms to the requirements of General Design Criteria 60 and 64 as specified in Section 3.1.

A list of the major offgas system components and design features is provided in Table 11.3-2. Equipment and piping is designed and constructed in accordance with the requirements of the applicable codes as given in Tables 3.2-1 and 3.2-2.

The quality group classifications of the various systems are shown in Table 3.2-1. Seismic category, safety class, quality assurance requirements, and principal construction codes

information are contained in Section 3.2. The system is designed to Quality Group Classification D+.

11.3.2 SYSTEM DESCRIPTION

11.3.2.1 Main Condenser Steam Jet Air Ejector RECHAR System

The offgas from the main condenser steam jet air ejector (SJAE) is treated by means of a system utilizing catalytic recombination and charcoal adsorption (RECHAR system) (see Figure 11.3-1). Descriptions of the major process components including design temperature and pressure are given in Table 11.3-2 and in the following.

Noncondensable radioactive offgas is continuously removed from the main condenser by the air ejector during plant operation. The air ejector offgas will normally contain activation gases, principally ^{16}N, ^{19}O, and ^{13}N. The ^{16}N and ^{19}O have short half-lives and are readily decayed. ^{13}N with a 10-minute half-life is present in small amounts that are further reduced by decay. Activation gas source terms are presented in Table 11.1-4.

The air ejector offgas will also contain radioactive noble gases including parents of biologically significant ^{89}Sr, ^{90}Sr, ^{140}Ba, and ^{137}Cs. The concentration of these noble gases depends on the amount of tramp uranium in the coolant and on the cladding surfaces (usually extremely small) and the number and size of fuel cladding defects.

A main condenser offgas treatment system has been incorporated in the plant design to reduce the radioactive gaseous effluents from the station. The offgas system uses a catalytic recombiner to recombine hydrogen and oxygen. After cooling (to approximately 130°F) to strip the condensables and reduce the volume, the remaining noncondensables (principally air with traces of krypton and xenon) are delayed in a holdup line. The gas is cooled to 45°F and processed through a HEPA filter. The gas is then passed through a desiccant dryer that reduces the dewpoint to approximately -90°F. Charcoal adsorption beds selectively adsorb and delay the xenon and krypton from the bulk carrier gas (principally dry air). With an air inleakage of 30 scfm, this treatment system results in a delay of 15 hr for krypton and 9.5 days for xenon. After the delay, the gas is again passed through a HEPA filter and discharged to the environment through the reactor building elevated release duct.

Figure 11.3-1 is the process flow diagram for the system. The process data for startup and normal operating conditions are submitted as proprietary data under separate cover as Table 11.3-3. The information supporting the process data is presented in Reference 11.3-2. The system is mechanically capable of processing three times the source terms of Table 11.3-1 without affecting delay time of the noble gases.

Table 11.3-1 also lists isotopic activities at the discharge of the system, from which the decontamination factor for each noble gas isotope can be determined.

The flow diagram is shown in Figure 11.3-2. The main process routing is indicated by a heavy line.

The basis for sizing the recombiner is to maintain the hydrogen concentration below 4% (including steam) at the inlet and below 1% at the outlet on a dry basis. The exit hydrogen concentration is normally well below the 1% maximum allowed. The hydrogen generation rate of the reactor is based on data from nine boiling water reactors (BWR). The hydrogen generation rate is given in Table 11.3-3.

The krypton and xenon holdup time is closely approximated by the following equation:

$$T = \frac{K_D M}{V}$$

where:

T = holdup time of a given gas, seconds.

K_D = dynamic adsorption coefficient for the given gas, $\frac{cm^3}{gram}$

M = weight of charcoal, grams

V = flow rate of the carrier gas, $\frac{cm^3}{sec}$

Dynamic adsorption coefficient values for xenon and krypton used to determine gaseous effluent releases are discussed in Reference 11.3-1. Moisture has a detrimental effect on adsorption coefficients. To prevent moisture from reaching the charcoal, fully redundant, adsorbent (-90°F dewpoint) air dryers are supplied. There are redundant moisture analyzers that will alarm on breakthrough of the dryer beds; however, breakthrough is not expected since the dryer beds will be regenerated on a time basis. The system is slightly pressurized, which together with very stringent leak rate requirements, prevents leakage of moist air into the charcoal.

After the hydrogen and oxygen are removed by the recombiner, the remaining carrier gas is primarily nitrogen with a small percentage of oxygen due to the air inleakage from the main condenser and air injection from the HWC system. Analyses have shown that the system can maintain condenser vacuum at combined air injection and air inleakage rates as high as 93 scfm. Reference 11.3-3, Par. S1 (c) (2) indicates that with certain conditions of stable operation and suitable construction, noncondensables (not including radiological decomposition products) should not exceed 6 scfm per condenser shell for large condensers. Dilution air is not added to the system unless the air inleakage is less than 6 scfm. In that event, 6 scfm is

added to provide for dilution of residual hydrogen from the recombiner. An initial bleed of oil-free air is added on startup until the recombiner reaches operating temperature.

The charcoal adsorbers design process flow sheet is for an ambient temperature. Operation at ambient temperature is sufficient to reduce gaseous radioactivity levels to a fraction of that allowed by 10 CFR 50 Appendix I. The decay heat is sufficiently small that, even in the no-flow condition, there is no significant loss of adsorbed noble gases due to temperature rise in the adsorbers. The adsorbers are located in a shielded room and maintained at ambient room temperature. A radiation monitor is provided to monitor the radiation level in the air handling room of the charcoal bed vault for high gas activity from a system leak. High radiation will cause an alarm in the control room.

Channeling in the charcoal adsorbers is prevented by supplying an effective flow distributor on the inlet, having long columns and having a high bed-to-particle diameter ratio of approximately 500. Underhill has stated that channeling or wall effects may reduce efficiency of the holdup bed if this ratio is not greater than 12 (Reference 11.3-4). During transfer of the charcoal into the charcoal adsorber vessels, radial sizing of the charcoal will be minimized by pouring the charcoal (by gravity or pneumatically) over a cone or other instrument to spread the granules over the surface.

A valve is provided to bypass the charcoal adsorbers. The main purpose of this bypass is to protect the charcoal during preoperational and startup testing when gas activity is zero or very low.

It may be desirable to use the bypass for short periods during startup or normal operations. This bypass mode would not be used for normal operation unless some unforeseen system malfunction would necessitate shutting down the power plant or operating in the bypass mode and remaining within the technical specification radioactivity release limits. The radioactivity released is controlled by a process monitor upstream of the vent isolation valve that will cause the bypass valve to close on a high radiation alarm. This interlock can be defeated only by a key lock switch. In addition, there is a high-high-high alarm setpoint on the same monitor that will cause isolation of the offgas system if established release rate limits are reached.

Leakage of radioactive gases from the system is limited by welding piping connections where possible and using bellows stem seals or equivalent leakage control valving. The system operates at a maximum of 7 psig during startup and less than 2 psig during normal operation so that the differential pressure to cause leakage is small.

Hydrogen concentration of gases from the air ejector is kept below the flammable limit by maintaining adequate process steam flow for dilution at all times. This steam flow rate is monitored and alarmed in the main control room.

Two parallel independent hydrogen analyzers are used to measure the hydrogen content of the offgas process flow downstream of the offgas condenser. The hydrogen concentration percentage output from the analyzers is indicated and recorded in the main control room along with independent alarm annunciation for a high hydrogen concentration. Each hydrogen analyzer continuously withdraws a sample of the process offgas, conditions the gas to a constant pressure, analyzes the hydrogen content, and returns the sample gas to the main condenser. The main condenser vacuum provides the pumping force to withdraw the sample gas from the offgas process line and through the hydrogen analyzer system. The analyzer element is a thermal conductivity cell type unit and does not serve as an ignition source to a detonable hydrogen-oxygen mixture.

Piping and tubing 2 in. and under is field routed, but required to be in specified space envelopes for shielding and in-plant exposure consideration. In the offgas system this includes drain lines, steam lines, and sample lines which are shown in Figure 11.3-2.

There are several liquid seals to prevent gas escape through drains shown in Figure 11.3-2. These seals are protected against permanent loss of liquid by an enlarged section downstream of the seal that can hold the seal volume and will drain by gravity back into the loop after a momentary pressure surge has passed. Each seal has a manual valve that is used to fill the loop with condensate after receiving a loop seal low level alarm.

Iodine input into the offgas system is small by virtue of its retention in reactor water and condensate. The iodine remaining is essentially removed by adsorption in the charcoal. This is supported by the fact that 2-in. charcoal filters remove more than 90% of the influent iodine, whereas this system has approximately 76 ft of charcoal in the flow path.

Particulates are removed with a 99.95% efficiency by a HEPA filter prior to the gas entering the charcoal adsorbers. The noble gas decays within the interstices of the activated charcoal and daughters are entrapped there. The charcoal serves as an excellent filter for particulates and essentially no particulates exit from the charcoal. The charcoal is followed with a HEPA filter which is a safeguard against escape of charcoal dust. Particulate activity discharged from this system is essentially zero.

With an airflow of about 30 scfm and the charcoal adsorbers bypassed, the delay time of the system decreases to approximately 10 minutes. This bypass line is only intended to be used during preoperational testing, and initial system startup operation until proper functioning of upstream equipment is established. This prevents possible degradation of the charcoal due to the introduction of excessive moisture or other contaminants.

In the unlikely event it is necessary to bypass the charcoal adsorbers to continue operation, the bypassing operation shall be allowed only if the radioactive effluents are within the release limits.

The isotopic inventory of each equipment piece, based on the source term discussed in Section 11.1, is given in Reference 11.3-5.

Performance of a similar system operating at ambient temperatures and the results of related experimental testing are discussed in Reference 11.3-2. The Tsuruga and Fukushima 1 plants in Japan have similar recombiners in service. Similar systems (ambient temperature charcoal) are in service at Dresden 2 and 3, Pilgrim, Quad Cities 1 and 2, Nuclenor, Hatch, Browns Ferry 1, 2, and 3, and Duane Arnold.

Design provisions are incorporated which preclude the uncontrolled release of radioactivity to the environment as a result of any single operator error or of any single failure. A comprehensive discussion of single failures is provided in Table 11.3-5.

Design precautions taken to prevent uncontrolled releases of radioactivity include the following:

a. The system design eliminates ignition sources so that a hydrogen detonation is highly unlikely even in the event of a recombiner failure;

b. The system pressure boundary is detonation resistant;

c. All discharge paths to the environment are monitored--the normal effluent path by the process radiation monitoring system and equipment areas by the area radiation monitoring system; and

d. Dilution steam flow to the SJAE is monitored and alarmed. Valve control logic causes the air ejector suction valves to close on low steam flow.

11.3.2.2 Other Radioactive Gas Sources

There are three buildings that contain radioactive gas sources. They are the reactor building, the turbine generator building, and the radwaste building. The ventilation systems for these three buildings are described in Section 9.4. Building volumes and ventilation flow rates are shown in Table 11.3-8. The sources of gaseous radioactivity in these buildings are discussed below. In-plant airborne radioactivity concentrations are discussed in Section 12.2.2.

The primary containment is divided into two sections which are designated as the drywell and suppression chamber. These are separated by the drywell floor which serves as a pressure barrier between the drywell and suppression chamber.

Radioactive halogens and noble gases can be introduced into the drywell atmosphere from two sources. One source is the leakage that occurs from the valves, especially the main steam isolation valves (MSIVs), the inner refueling bellows seal support drains and the main

recirculation pumps. This leakage is collected by means of leak-off lines which are directed to the drywell equipment drain sump.

The other source of activity results when a pressure transient causes the main steam relief valves to open with the resulting flow of steam into the suppression pool. When this occurs, there is a pressure buildup in the suppression chamber atmosphere. A tabulation of the expected relative frequency of pressure relief valve venting to the suppression pool is provided in Table 11.3-10. If the pressure differential across the drywell floor is greater than 0.5 psi, the vacuum breaker valves relieve this pressure into the drywell atmosphere. Thus, radioactivity in the suppression chamber atmosphere is introduced into the drywell. The drywell atmosphere is purged to the environment via the reactor building elevated release duct when access is required, either directly or through the standby gas treatment system, considering airborne radiation levels and release limits.

The reactor building ventilation system supplies fresh air to the secondary containment and exhausts air through the reactor building elevated release duct. There is a small amount of activity in the secondary containment atmosphere that emanates from the various reactor support systems. The reactor building ventilation exhaust is monitored and a radiation level, set by administrative control to ensure Technical Specifications compliance, will cause automatic heating, ventilating, and air-conditioning (HVAC) system isolation, startup of the standby gas treatment system (SGTS), and an alarm in the control room.

There are three sources of nontreated radioactive gas sources in the turbine generator building which are as follows:

a. Gland seal steam leakage and condenser exhaust,
b. Equipment leakage, and
c. Main condenser offgas during startup.

The activity from these sources is presented in Table 11.3-7.

Gland seal leakage and condenser exhaust contribute essentially no airborne radioactivity releases due to use of "clean" steam as discussed in Section 10.4. During startup, the mechanical vacuum pumps remove the gas present inside the main condenser. The exhaust from these pumps is discharged through the reactor building elevated release duct. Due to radioactive decay during shutdown, only a small amount of activity is exhausted by the vacuum pumps during startup.

Sources of gaseous radioactivity in the radwaste building include,

a. Offgas system leakage,
b. Liquid leakage to the radwaste building,

c. Liquid waste management system tank vents, and
d. Hydropneumatic transfer of resins.

Measures taken to minimize leakage from the offgas system are described in Section 11.3.2.1. Liquid leakage from equipment and floor drains in the radwaste building is collected in sumps. The sumps and tanks containing liquid radwaste are vented to the radwaste building exhaust system described in Section 9.4.3.2. The air blow during the backwash of the spent powdered resins to the phase separators generates airborne radioactivity which is vented to the radwaste building filtered exhaust system.

11.3.2.3 Cost-Benefit Analysis

The cost-benefit analysis is discussed in Section 11.2.3.4.

11.3.2.4 Design Features of the Offgas System

Design features of other gaseous waste management systems may be found in Section 9.4.

11.3.2.4.1 Maintainability

Design features which reduce or ease required maintenance include the following:

a. Redundant components for all active, in-process equipment pieces, and

b. No rotating equipment in the process stream and elsewhere in the system only where maintenance can be performed while the system is in operation.

11.3.2.4.2 Pressure Boundaries

Design features and requirements which reduce leakage and releases of radioactive material include the following:

a. Extremely stringent leak rate requirements placed on all equipment, piping, and instruments, and enforced by requiring as-installed leak tests of the entire process system,

b. Use of welded joints except where servicing access to equipment and instrumentation access dictates use of flanged joints,

c. Valve types with extremely low leak rate characteristics, i.e., bellows seals double stem seal, or diaphragm,

d. Use of loop seals with enlarged discharge section to avoid siphoning and to be self-refilling by gravity following a pressure surge, and

e. Stringent seat-leak characteristics for valves in lines discharging to the environment via other systems.

11.3.2.4.3 Building Seismic Design

The offgas system is located in the turbine generator and radwaste buildings. The portion of the turbine generator building housing the offgas system is a modified Seismic Category II structure designed to withstand the effects of a safe shutdown earthquake (SSE) and maintain its structural integrity. The portion of the radwaste building housing the offgas system is a Seismic Category I structure. The seismic classification of the turbine generator and radwaste buildings are discussed in Sections 3.8.4.1.3 and 3.8.4.1.2, respectively.

11.3.2.4.4 Construction of Process Systems

Pressure retaining components of process systems utilize welded construction to the maximum practicable extent. Process piping systems include the first root valve on sample and instrument lines. Process lines are not less than 0.75 in. nominal pipe size. Sample and instrument lines are not considered as portions of the process systems. Flanged joints or suitable rapid disconnect fittings are not used except where maintenance requirements clearly indicate that such construction is preferable. Screwed connections in which threads provide the only seal are not used. Screwed connections backed up by seal welding or mechanical joints are used only on lines of 0.75 in. nominal pipe size or less. In lines 0.75 in. or greater, but less than 2.5 in. nominal pipe size, socket type welds are used. In lines 2.5 in. nominal pipe size and larger, pipe welds are of the butt joint type.

11.3.2.4.5 Instrumentation and Control

The offgas system is monitored by flow, temperature, pressure, and humidity instrumentation, and by hydrogen analyzers to ensure correct operation and control. Table 11.3-4 lists the process parameters that are instrumented to alarm in the control room and indicates whether the parameters are recorded or just indicated.

The radioactivity of the gas entering and leaving the offgas system is continuously monitored. Thus, system performance is known to the operator at all times. A radiation monitor after the offgas condenser continuously monitors radioactivity release from the reactor and input to the charcoal adsorbers. This radiation monitor is used to provide an alarm on high radiation in the offgas. A sample rack with two radiation monitors is also provided at the outlet of the charcoal adsorbers to continuously monitor the radioactivity from the adsorber beds. These radiation monitors are used to isolate the offgas system on high radioactivity to prevent gas of

unacceptably high activity from entering the reactor building elevated release duct. Only one monitor is required to be operable.

The offgas system at the SJAE is sampled periodically in accordance with the Technical Specifications. Provision is made for sampling and periodic analysis of the influent and effluent gases for purposes of determining their compositions. This information is used in calibrating the monitors and in relating the release to calculated offsite doses. Process radiation instrumentation is described in detail in Sections 11.5 and 7.6.1.1.

11.3.2.4.6 Detonation Resistance

The pressure boundary of the system is designed to be detonation resistant. The pressure vessels are designed to withstand 350 psig static pressure, and piping and valves are designed to resist dynamic pressures encountered in long runs of piping at the design temperature. This analysis is covered in a proprietary report submitted to the NRC (Reference 11.3-6).

By this procedure a designer obtains the required wall thickness for specific equipment to sustain a hydrogen and oxygen detonation. The wall thickness is then translated by using the appropriate code calculation to the corresponding equipment that must contain the detonation static pressure. The method assumes the absence of simultaneous secondary events such as earthquakes.

11.3.2.4.7 Operator Exposure Criteria and Controls

This system is normally operated from the main control room. Equipment and process valves containing radioactive fluid are placed in shielded cells maintained at a pressure negative to normally occupied areas. Ventilation air flows from areas of low airborne contamination to areas of higher airborne contamination. Operating offgas process equipment does not require personnel access. Redundant equipment is located in separate cells, minimizing exposure of maintenance personnel. No process fluid is passing through instrumentation panels. Signals from the process streams are transmitted to the instrumentation panels by means of electrical signal converters. Design features minimizing occupational exposure are discussed in Sections 11.3.2.4.1 and 12.3.1.3.

11.3.2.4.8 Equipment Malfunction

Malfunction analyses indicating consequences and design precautions taken to accommodate failure of various components of the system are given in Table 11.3-5.

11.3.2.5 Offgas System Operating Procedure

11.3.2.5.1 Prestartup Preparations

Prior to starting the main condenser SJAEs, the glycol coolant is chilled to near 35°F and is circulated through the cooler condenser, a desiccant dryer is regenerated and valved in, the offgas condenser cooling water is valved in, and the recombiner heaters are turned on.

11.3.2.5.2 Startup

As the reactor is pressurized, preheater steam is supplied and air is bled through the preheater and recombiner. The recombiner is preheated to at least 300°F with this air bleed and/or by admitting steam to the final SJAE. With the recombiners preheated and the desiccant dryer and charcoal adsorbers valved in, the SJAE string is started. The bleed air is terminated. As the condenser is pumped down and the reactor power increases, the recombiner inlet stream is diluted to less than 4% hydrogen by volume by a fixed steam supply, and the offgas condenser outlet is maintained at less than 1% hydrogen by volume.

11.3.2.5.3 Normal Operation

After startup, the noncondensables pumped by the SJAE will stabilize. Recombiner performance is closely followed by the recorded temperature profile in the recombiner catalyst bed. The hydrogen effluent concentration is measured by a hydrogen analyzer.

Normal operation is terminated following a normal reactor shutdown or a scram by terminating steam to the SJAEs and the preheater.

11.3.2.6 Offgas System Performance Tests

This system is used on a routine basis and does not require specific testing to ensure operability. Monitoring equipment will be calibrated and maintained on a specific schedule and on indication of malfunction.

11.3.2.6.1 Recombiner

Recombiner performance is continuously monitored and recorded by catalyst bed thermocouples that monitor the bed temperature profile and by a hydrogen analyzer that measures the hydrogen concentration of the effluent.

11.3.2.6.2 Prefilter

These particulate filters were tested at the time of filter installation using dioctylphthalate (DOP) aerosol to determine whether an installed filter meets the minimum in-place efficiency 99.95% particle retention.

The DOP from filter testing is not allowed to enter into the desiccant or the activated charcoal. This equipment is isolated during filter DOP testing and is bypassed until the process lines have been purged clear of test material. Because the DOP may have a detrimental effect on the desiccant and charcoal, the prefilter will not be periodically tested. This is justified because the main function of this prefilter is to prevent the long-lived daughters of the radioactive xenons generated in the holdup pipe from depositing in the downstream equipment. Leakage through the filter would be unimportant to environmental release.

11.3.2.6.3 Desiccant Gas Dryer

Desiccant gas dryer performance is continuously monitored by an on-stream humidity analyzer.

11.3.2.6.4 Charcoal Performance

The ability of the charcoal to delay the noble gases can be continuously evaluated by comparing radioactivity measured and recorded by the process activity monitors at the exit of the offgas condenser and at the exit of the charcoal adsorbers.

Grab sample points are located upstream and downstream of the first charcoal bed and downstream of the last charcoal bed and can be used for periodic sampling if the monitoring equipment indicates degradation of system delay performance.

11.3.2.6.5 Post Filter

On installation these particulate filters were tested using a DOP aerosol test or equivalent, as described in Section 11.3.2.6.2.

11.3.3 RADIOACTIVE RELEASES

11.3.3.1 <u>Release Points</u>

The reactor building elevated release duct serves the following systems:

Offgas system,
Mechanical vacuum pump and gland seal condenser exhaust,
Reactor building ventilation exhaust,

Containment purge, and
Standby gas treatment system.

The reactor building elevated release duct location is shown in reactor building general arrangement drawings in Section 1.2.

The radwaste building ventilation exhausts through three louver houses 67 ft above plant grade. The location is shown on Figure 1.2-17.

The exhaust from the turbine generator building ventilation system is through four exhaust fans located on the radwaste building, 119 ft above plant grade. Their location is shown on Figure 1.2-16 (part plan of roof at el. 542 ft-0 in.) and in Figure 1.2-17.

The height, flow rate, heat content, and dimensions of the three release points are shown on Table 11.3-6.

11.3.3.2 Dilution Factors

The dispersion and dilution of gaseous radioactive effluents released from the plant depends on the meteorology of the site and its environs. To determine these parameters, onsite meteorological data has been obtained and analyzed as described in Section 2.3. Annual atmospheric dilution factors have been calculated to determine resultant annual doses and concentrations of radionuclides from normal operation.

The building ventilation exhaust ducts do not rise above the buildings; therefore, atmospheric releases for dose analysis purposes were considered ground level.

11.3.3.3 Estimated Releases

Releases of radioactive material in gaseous effluents for initial plant licensing were calculated using the GALE code presented in NUREG-0016 to show compliance with 10 CFR 20, Appendix B, and 10 CFR 50, Appendix I, for normal operation plus anticipated operational occurrences. The operational parameters including source terms were those presented in Appendix B. The values obtained are presented in Table 11.3-7. These values were used with the meteorology data from Section 2.3.5 to calculate maximum concentrations at the restricted area boundary and maximum individual dose offsite. These data provide maximum annual average (χ/Q) values and does not include a building wake factor.

Restricted area boundary concentrations used for this initial licensing analysis are tabulated by radionuclide and compared with 10 CFR 20 limits in Table 11.3-9. The estimated annual dose to persons offsite is presented in Section 5.2 of the Environmental Report and is well within the numerical guidelines of 10 CFR 50, Appendix I.

Since becoming operational, releases of radioactive materials in gaseous effluents have been determined using actual flow rates and quantitative and qualitative analyses. Doses due to radioactive materials in gaseous effluents are determined to show Technical Specifications compliance at specified intervals. Compliance is reported in the Annual Radioactive Effluent Release Report. Doses are ascertained using the NRC GASPAR II computer code, current meteorology (or historical meteorology if current data are unavailable), and parameters outlined in the Offsite Dose Calculation Manual.

Tritium contamination of the auxiliary boiler and associated components has resulted in release paths which were not intended (see Section 9.4.16.2). The released radioactivity attributable to the worst case tritium concentration (2E + 06 ρCi/liter) in the boiler water is an insignificant portion of the total activity released in liquid and gaseous effluents and has been analyzed to result in a correspondingly insignificant radiation dose. The tritium concentration levels and makeup water volume are monitored and evaluated to ensure that tritium effluent releases from the plant are adequately quantified.

11.3.4 REFERENCES

11.3-1 NUREG-0016 (BWR-GALE Code), January 1979

11.3-2 Miller, C. W., Experimental and Operational Confirmation of Off-Gas System Design Parameters, NEDO-10751, January 1973 (Proprietary).

11.3-3 Standards for Steam Surface Condensers, Sixth Edition, Heat Exchange Institute, New York, NY, 1970.

11.3-4 Underhill, Dwight, et al., "Design of Fission Gas Holdup Systems," Processing of the Eleventh AEC Air Cleaning Conference, 1979, p. 217.

11.3-5 Miller, C. W., et al., A General Justification for Classification of Effluent Treatment System Equipment as Group D, NEDO-10734, February 1973.

11.3-6 Nesbitt, L. B., Design Basis for New Gas Systems, NEDE-11146, July 1971 (Proprietary).

11.3-7 NUREG-0800 USNRC Standard Review Plan, Revision 2, July 1981, Branch Technical Position ETSB 11-5.

Table 11.3-1

Design Air Ejector Offgas Release Rates (30 cfm inleakage) [a]

		T=0	T=30 Minutes	Normal Discharge from Charcoal Adsorbers		Additional Discharge from Charcoal Adsorbers During Startup	
Isotope	Half-life	μCi/sec	μCi/sec	μCi/sec	Ci/yr[b]	μCi/sec	Ci/startup
^{83m}Kr	1.86 hr	3.4×10^{3}	2.9×10^{3}	-	-	-	-
^{85m}Kr	4.4 hr	6.1×10^{3}	5.6×10^{3}	4.3	1.2×10^{2}	1.1×10^{1}	1.4
^{85}Kr[c]	10.74 yr	10-20	10-20	10-20	280-560	0	0
^{87}Kr	76 minutes	2.0×10^{4}	1.5×10^{4}	-	-	-	-
^{88}Kr	2.79 hr	2.0×10^{4}	1.8×10^{4}	2.1×10^{-1}	6.0	1.4	1.7×10^{-1}
^{89}Kr	3.18 minutes	1.3×10^{5}	1.8×10^{2}	-	-	-	-
^{90}Kr	32.3 sec	2.8×10^{5}	-	-	-	-	-
^{91}Kr	8.6 sec	3.3×10^{5}	-	-	-	-	-
^{92}Kr	1.84 sec	3.3×10^{5}	-	-	-	-	-
^{93}Kr	1.29 sec	9.9×10^{4}	-	-	-	-	-
^{94}Kr	1.0 sec	2.3×10^{4}	-	-	-	-	-
^{95}Kr	0.5 sec	2.1×10^{3}	-	-	-	-	-
^{97}Kr	1 sec	1.4×10^{1}	-	-	-	-	-
^{131m}Xe	11.96 days	1.5×10^{1}	1.5×10^{1}	1.3	3.7×10^{1}	3.0×10^{-2}	1.07×10^{-1}
^{133m}Xe	2.26 days	2.9×10^{2}	2.8×10^{2}	-	-	-	-
^{133}Xe	5.27 days	8.2×10^{3}	8.2×10^{3}	3.3×10^{1}	9.4×10^{2}	1.9	6.8
^{135m}Xe	15.7 minutes	2.6×10^{4}	6.9×10^{3}	-	-	-	-
^{135}Xe	9.16 hr	2.2×10^{4}	2.2×10^{4}	-	-	-	-
^{137}Xe	3.82 minutes	1.5×10^{5}	6.7×10^{2}	-	-	-	-
^{138}Xe	14.2 minutes	8.9×10^{4}	2.1×10^{4}	-	-	-	-
^{139}Xe	40 sec	2.8×10^{5}	-	-	-	-	-
^{140}Xe	13.6 sec	3.0×10^{5}	-	-	-	-	-
^{141}Xe	1.72 sec	2.4×10^{5}	-	-	-	-	-
^{142}Xe	1.22 sec	7.3×10^{4}	-	-	-	-	-
^{143}Xe	0.96 sec	1.2×10^{4}	-	-	-	-	-
^{144}Xe	9 sec	5.6×10^{2}	-	-	-	-	-
	Totals	2.5×10^{6}	1.0×10^{5}	49-59	1383-1663	14.3	8.5

[a] Based on the 1971 mixture.
[b] This is based on curies present at time of release. No decay in environment is included.
[c] Estimated from experimental observations.

Table 11.3-2

Offgas System Major Equipment Items

Offgas Preheaters - two required

Construction: Stainless steel tubes and carbon steel shell. 350 psig shell design pressure, 1000 psig tube design pressure. 400°F shell design temperature, 575°F tube design temperature.

Catalytic Recombiners - two required

Construction: Stainless steel cartridge, carbon steel shell. Catalyst cartridge containing a precious metal catalyst on metal base. Catalyst cartridge to be replaceable without removing vessel. 350 psig design pressure, 900°F design temperature.

Offgas Condenser - one required

Construction: Low alloy steel shell. Stainless steel tubes. 350 psig shell design pressure. 250 psig tube design pressure, 900°F shell design temperature, 150°F tube design temperature.

Water Separator - one required

Construction: Carbon steel shell, stainless steel wire mesh. 350 psig design pressure, 250°F design temperature.

Cooler-Condenser - two required

Construction: Stainless steel shell. Stainless steel tubes. 100 psig tube design pressure, 350 psig shell design pressure. 32/150°F tube design temperature, 32/150°F shell design temperature.

Moisture Separators (downstream of cooler-condenser) - two required

Construction: Carbon steel shell, stainless steel wire mesh. 350 psig design pressure, 32/150°F design temperature.

Desiccant Dryer - four required

Construction: Carbon steel shell packed with Type 3A molecular sieve or equivalent. 350 psig design pressure, 32°F/ 500°F design temperature.

Desiccant Regeneration Skid - two required

Table 11.3-2

Offgas System Major Equipment Items (Continued)

Dryer Chiller - two required

Construction: Carbon steel shell, stainless steel tubes, design temperature 32°F/500°F, design pressure 50 psig.

Regenerator Blower - two required

Construction: Electrical, design pressure 50 psig design temperature 32°F/150°F.

Dryer Heater - two required

Construction: Electrical, design temperature 32°F/500°F, design pressure 50 psig.

Glycol Cooler Skid - one required

Glycol Storage Tank - one required

Construction: Carbon steel 3000 gal. Water-filled hydrostatic design pressure. 32°F design temperature. API-650.

Glycol Solution Refrigerators and Motor Drives - three required

Construction: Conventional refrigeration units. Glycol solution exit temperature 35°F.

Glycol Pumps and Motor Drives - three required

Construction: Cast iron, 0°F design temperature.

Prefilters and After Filters - two required of each type

Construction: Carbon steel shell. High-efficiency, moisture-resistant filter element. Flanged shell. 350 psig design pressure. -50/150°F design temperature.

Charcoal Adsorbers - eight beds

Construction: Carbon steel. Approximately 4-ft. O.D. x 21-ft vessels each containing approximately 3 tons of activated carbon. Design pressure 350 psig, design temperature -50/250°F.

Table 11.3-3

Process Data For The Offgas (RECHAR) System

The Information On This Page Is
Proprietary And Was Submitted
Under Separate Cover

(CVI DWG 02N64-04,11,1, Rev. 4 or 02N64-04,11,2, Rev. 2)

Table 11.3-4

Offgas System Alarmed Process Parameters

Parameters	Control Room Indicated	Control Room Recorded
Air ejector discharge pressure - high	X	
Preheater discharge temperature - low	X	
Recombiner catalyst temperature - high/low		X
Offgas condenser water level (dual) - high/low	X	
Offgas condenser gas discharge temperature - high (local)	X	
H_2 analysis (offgas condenser discharge) - dual - high		X
Offgas condenser discharge radiation - high		X
Gas flow - high/low		X
Cooler - condenser discharge temperature - high/low		X
Glycol solution temperature - high/low		X
Glycol solution level - low		
Gas drier discharge humidity - high (local)	X	
Prefilter dP - high	X	
Charcoal adsorber temperature - high		X
Charcoal train flow - high/low		X
After filter dP - high	X	
Offgas (charcoal bed discharge) radiation - high		X
Steam flow - low	X	
Desiccant dryer outlet temperature - high/low		X
Dryer chiller outlet temperature - high (local)	X	
Dryer heater temperature - high		
Dryer heater outlet temperature - high (local)	X	
Loop seals water level - low		

Table 11.3-5

Equipment Malfunction Analysis

Equipment Item	Malfunction	Consequences	Design Precautions
Steam jet air ejectors	Low flow of motive high-pressure steam	When the hydrogen and oxygen concentrations exceed 4 and 5 vol %, respectively, the process gas becomes flammable.	Alarm provided on steam for low steam flow. Recombiner temperature alarm.
		Inadequate steam flow will cause overheating and deterioration of the catalyst.	Steam flow to be held at constant maximum flow regardless of plant level during operation.
	Wear of steam supply nozzle of ejector	Increased steam flow to recombiner. This could reduce degree of recombination at low power levels.	Low temperature alarms on preheater exit (recombiner inlet). Recombiner outlet H_2 analyzers.
Preheaters	Steam leak	Would further dilute process offgas. Steam consumption would increase.	Spare preheater.
	Low pressure steam supply	Recombiner performance would fall off at low power level and hydrogen content of recombiner gas discharge may increase, eventually to a combustible mixture.	Low-temperature alarms on preheater exit (recombiner inlet). Recombiner outlet H_2 analyzers.
Recombiners	Catalyst gradually deactivates	Temperature profile changes through catalyst. Eventually excess H_2 would be detected by H_2 analyzer or by gas flowmeter. Eventually the stripped gas could become combustible.	Temperature probes in recombiner and H_2 analyzer provided. Spare recombiner.
	Catalyst gets wet at start	H_2 conversion falls off and H_2 is detected by downstream analyzers. Eventually the gas could become combustible.	Condensate drains, temperature probes in recombiner. Air bleed system at startup. Recombiner thermal blanket, spare recombiner, and heater. Hydrogen analyzer.

Table 11.3-5

Equipment Malfunction Analysis (Continued)

Equipment Item	Malfunction	Consequences	Design Precautions
Offgas condenser	Cooling water leak	The coolant (reactor condensate) would leak to the process gas (shell) side. This would be detected if drain well liquid level increases. Moderate leakage would be of no concern from a process standpoint. (The process condensate drains to the hotwell).	None
	Liquid level instruments fail	If both drain valves fail to open, water will build up in the condenser and pressure drop will increase. The high ΔP, if not detected by instrumentation, could cause pressure buildup in the main condenser and eventually initiate a reactor scram. If a drain valve fails to close, gas will recycle to the main condenser, increase the load on the SJAE, and increase operating pressure of the main condenser.	Two independent drain systems, each provided with high and low-level alarms.
Water separator	Corrosion of wire mesh element	Higher quantity of water collected in holdup line and routed to radwaste.	Stainless steel mesh specified.
Holdup line	Corrosion of line	Leakage to soil of gaseous and liquid fission products.	Outside of pipe dipped and wrapped. 0.25-in. corrosion allowance.
Cooler-condenser	Corrosion of tubes	Glycol-water solution would leak into process (shell) side and be discharged to clean radwaste. If not detected at radwaste, the glycol solution would discharge to the reactor condensate system.	Stainless-steel tubes specified. Low level alarm glycol tank level. Spare cooler condenser provided.

Table 11.3-5

Equipment Malfunction Analysis (Continued)

Equipment Item	Malfunction	Consequences	Design Precautions
Cooler-condenser (continued)	Icing up of the tubes	Shell side of cooler could plug up with ice, gradually building up pressure drop. If this happens, the spare unit could be activated. Complete blockage of both units would increase ΔP and lead to a reactor scram.	Design glycol-H_2O solution temperature well above freezing point. Spare unit provided. Temperature indication and low alarms on glycol temperature and process gas temperature.
Glycol refrigeration machines	Mechanical failure	If both spare units fail to operate, the glycol solution temperature will rise and the dehumidification system performance will deteriorate. This will require rapid regeneration cycles for the desiccant beds and may raise the gas dewpoint as it is discharged from the drier.	Two spare refrigerators during normal operation are provided. Glycol solution temperature alarms provided. Gas moisture detectors provided downstream of gas driers.
Moisture separators	Corrosion of wire mesh	Increased moisture would be retained in process gas routed to gas driers element. Over a long period, the desiccant drier cycle period would deteriorate as a result of moisture pickup. Pressure drop across prefilter may increase if filter media is wetted.	Stainless steel mesh specified. Spare unit provided. High ΔP alarm on prefilter.
Prefilters	Loss of integrity of filter media	More radioactivity would deposit on the drier desiccant. This would increase the radiation level in the drier vault and make maintenance more difficult, but would not affect releases to the environment.	Spare unit provided in separate vault. ΔP instrumentation provided.

Table 11.3-5

Equipment Malfunction Analysis (Continued)

Equipment Item	Malfunction	Consequences	Design Precautions
Desiccant drier	Moisture breakthrough	Increased moisture in air entering charcoal adsorbers would decrease adsorption effectiveness, thus reducing radioisotope retention time.	Drier cycles on timer. Redundant gas humidity analyzers and alarms supplied. Redundant gas drier system supplied. Gas drier and first charcoal bed can be bypassed through alternate drier to second charcoal bed.
Desiccant regeneration equipment	Mechanical failure	Inability to regenerate desiccant.	Redundant, shielded desiccant beds and drier equipment is supplied.
Charcoal adsorbers	Charcoal accumulates moisture	Charcoal performance will deteriorate gradually as moisture deposits. Holdup times for krypton and xenon would decrease, and plant emissions would increase. Provisions made for drying charcoal as required during annual outage.	Highly instrumented, mechanically simple gas dehumidification system with redundant equipment.
After filters	Loss of integrity of filter media	Probably of no real consequence. The charcoal media itself should be a good filter at the low air velocity.	ΔP instrumentation provided. Spare unit provided.
System	Internal detonation	Release of radioactivity if pressure boundary fails.	Main process equipment and piping are designed to contain a detonation.
	Earthquake damage	Release of radioactivity.	Dose consequences are within 10 CFR 20 limits. Analysis is included in Reference 11.3-5.

Table 11.3-6

Release Point Data

Note: Refer to ODCM Table 3-13.

Table 11.3-7

Gaseous Waste System Release

Nuclide	Coolant Conc. (μCi/g)	Gaseous Release Rate[a] (Ci/yr) Containment Building	Turbine Building	Reactor Building	Radwaste Building	Gland Seal	Air Ejector	Mechanical Vac Pump	Total
^{83m}Kr	1.200E-03	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
^{85m}Kr	2.000E-03	3.1E 00	7.1E 01	3.1E 00	0.0	0.0	2.1E 00	0.0	8.0E 01
^{85}Kr	6.300E-06	0.0	0.0	0.0	0.0	0.0	2.8E 02	0.0	2.8E 02
^{87}Kr	6.900E-03	3.1E 00	2.0E 02	3.1E 00	0.0	0.0	0.0	0.0	2.1E 02
^{88}Kr	6.900E-03	3.1E 00	2.4E 02	3.1E 00	0.0	0.0	0.0	0.0	2.4E 02
^{89}Kr	4.300E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
^{131m}Xe	4.900E-06	0.0	0.0	0.0	0.0	0.0	5.2E 00	0.0	5.2E 00
^{133m}Xe	9.400E-05	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
^{133}Xe	2.700E-03	6.9E 01	2.9E 02	6.9E 01	1.0E 01	0.0	2.3E 01	2.3E 03	2.8E 03
^{135m}Xe	8.800E-04	4.8E 01	6.8E 02	4.8E 01	0.0	0.0	0.0	0.0	7.8E 02
^{135}Xe	7.600E-03	3.6E 01	6.6E 02	3.6E 01	4.5E 01	0.0	0.0	3.5E 02	1.1E 03
^{137}Xe	4.900E-02	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
^{138}Xe	2.900E-02	7.3E 00	1.5E 03	7.3E 00	0.0	0.0	0.0	0.0	1.5E 03
Total noble gases									7.0E 03
^{131}I	3.618E-03	1.8E-02	2.0E-01	1.8E-01	5.0E-02	0.0	0.0	3.0E-02	4.8E-01
^{133}I	1.549E-02	7.1E-02	8.0E-01	7.1E-01	1.8E-01	0.0	0.0	0.0	1.8E-00
Tritium gaseous release									7.1E 01

NOTE: 0.0 appearing in the table indicates release is less than 1.0 Ci/yr for noble gas, 0.0001 Ci/yr for iodine.

Table 11.3-7

Gaseous Waste System Release (Continued)

Airborne Particulate Release Rate (Ci/yr)

Nuclide	Containment Building	Turbine Building	Reactor Building	Radwaste Building	Mechanical Vac Building	Total
^{51}Cr	3.1E-06	1.4E-02	3.1E-04	9.4E-05	0.0	1.4E-02
^{54}Mn	3.1E-05	5.2E-04	3.1E-03	4.7E-04	0.0	4.3E-03
^{59}Fe	4.2E-06	5.2E-04	4.2E-04	1.6E-04	0.0	1.1E-03
^{58}Co	6.3E-06	6.3E-04	6.3E-04	4.7E-05	0.0	1.4E-03
^{60}Co	1.0E-04	2.1E-03	1.0E-02	9.4E-04	0.0	1.5E-02
^{65}Zn	2.1E-05	2.1E-04	2.1E-03	1.6E-05	0.0	2.3E-03
^{89}Sr	9.4E-07	6.3E-03	9.4E-05	4.7E-06	0.0	6.4E-03
^{90}Sr	5.2E-08	2.1E-05	5.2E-06	3.1E-06	0.0	2.9E-05
^{95}Zr	4.2E-06	1.0E-04	4.2E-04	5.2E-07	0.0	5.2E-04
^{124}Sb	2.1E-06	3.1E-04	2.1E-04	5.2E-07	0.0	5.2E-04
^{134}Cs	4.2E-05	3.1E-04	4.2E-03	4.7E-05	3.1E-06	4.6E-03
^{136}Cs	3.1E-06	5.2E-05	3.1E-04	4.7E-06	2.1E-06	3.8E-04
^{137}Cs	5.8E-05	6.3E-04	5.8E-03	9.4E-05	1.0E-05	6.6E-03
^{140}Ba	4.2E-06	1.1E-02	4.2E-04	1.0E-06	1.1E-05	1.1E-02
^{141}Ce	1.0E-06	6.3E-04	1.0E-04	6.3E-05	0.0	8.0E-04

[a] Estimated release based on GALE code evaluation.

Table 11.3-8

Building Volume and Ventilation Rates

Building	Free Air Volume (ft^3)	Ventilation Rate (cfm)
Secondary containment (reactor building)	3.5×10^6	80,000
Radwaste building	2.0×10^6	83,000
Turbine building	5.7×10^6	360,000
Primary containment (drywell)	2.0×10^5	10,500[a]
Primary containment (wetwell)	1.4×10^5	7,500[a]

[a] During primary containment purge only.

Table 11.3-9

Maximum Sector Annual Average Concentrations of Gaseous Radioactive Materials at the Original Restricted Area Boundary

Nuclide	Annual Average Release Rate (Ci/sec)	Boundary Concentration[a] ($\mu Ci/cm^3$)	Derived Air Concentration (DAC)[b] ($\mu Ci/cm^3$)	Concentration/ DAC
^{3}H	2.3 E-6	1.8 E-11	1 E-7	0.00018
^{85m}Kr	2.5 E-6	1.9 E-11	1 E-7	0.0002
^{85}Kr	9.0 E-6	6.9 E-11	7 E-7	0.000099
^{87}Kr	6.6 E-6	5.1 E-11	2 E-8	0.002
^{88}Kr	8.0 E-6	6.2 E-11	9 E-9	0.00069
^{131m}Xe	1.7 E-7	1.3 E-12	2 E-6	0.0000006
^{133}Xe	9.0 E-5	6.9 E-10	6 E-7	0.0011
^{135m}Xe	2.4 E-5	1.9 E-10	4 E-8	0.0047
^{135}Xe	3.7 E-5	2.8 E-10	7 E-8	0.0041
^{138}Xe	4.6 E-5	3.6 E-10	2 E-8	0.018
^{131}I	1.6 E-8	1.3 E-13	2 E-10	0.0006
^{133}I	5.7 E-8	4.4 E-13	1 E-9	0.00044
^{51}Cr	4.3 E-10	3.4 E-15	3 E-8	0.00000011
^{54}Mn	1.3 E-10	9.7 E-16	1 E-9	0.00000097
^{59}Fe	3.7 E-11	2.8 E-16	5 E-10	0.00000057
^{58}Co	4.3 E-11	3.4 E-16	1 E-9	0.00000034
^{60}Co	4.3 E-10	3.4 E-15	5 E-11	0.000067
^{65}Zn	7.3 E-11	5.7 E-16	4 E-10	0.0000015
^{89}Sr	2.0 E-10	1.6 E-15	1 E-9	0.0000016
^{90}Sr	9.3 E-13	7.2 E-18	6 E-12	0.0000013
^{95}Zr	1.7 E-11	1.3 E-16	4 E-10	0.0000003
^{124}Sb	1.7 E-11	1.3 E-16	3 E-10	0.0000004
^{134}Ca	1.5 E-10	1.1 E-15	2 E-10	0.0000058
^{136}Cs	1.1 E-11	8.9 E-17	9 E-10	0.000000099
^{137}Cs	2.1 E-10	1.6 E-15	2 E-10	0.0000079
^{140}Ba	3.7 E-10	2.8 E-15	2 E-9	0.0000015
^{141}Ce	2.5 E-11	1.9 E-16	1 E-9	0.00000019
			MPC	0.03

[a] χ/Q factor of 7.7 x 10^{-6} at 0.5 miles distance in SE sector used. No building wake factor included; meteorological data from 4/74 through 3/75.

[b] 10 CFR 20, Appendix B, to 20.1001-20.2401, Table II Column I.

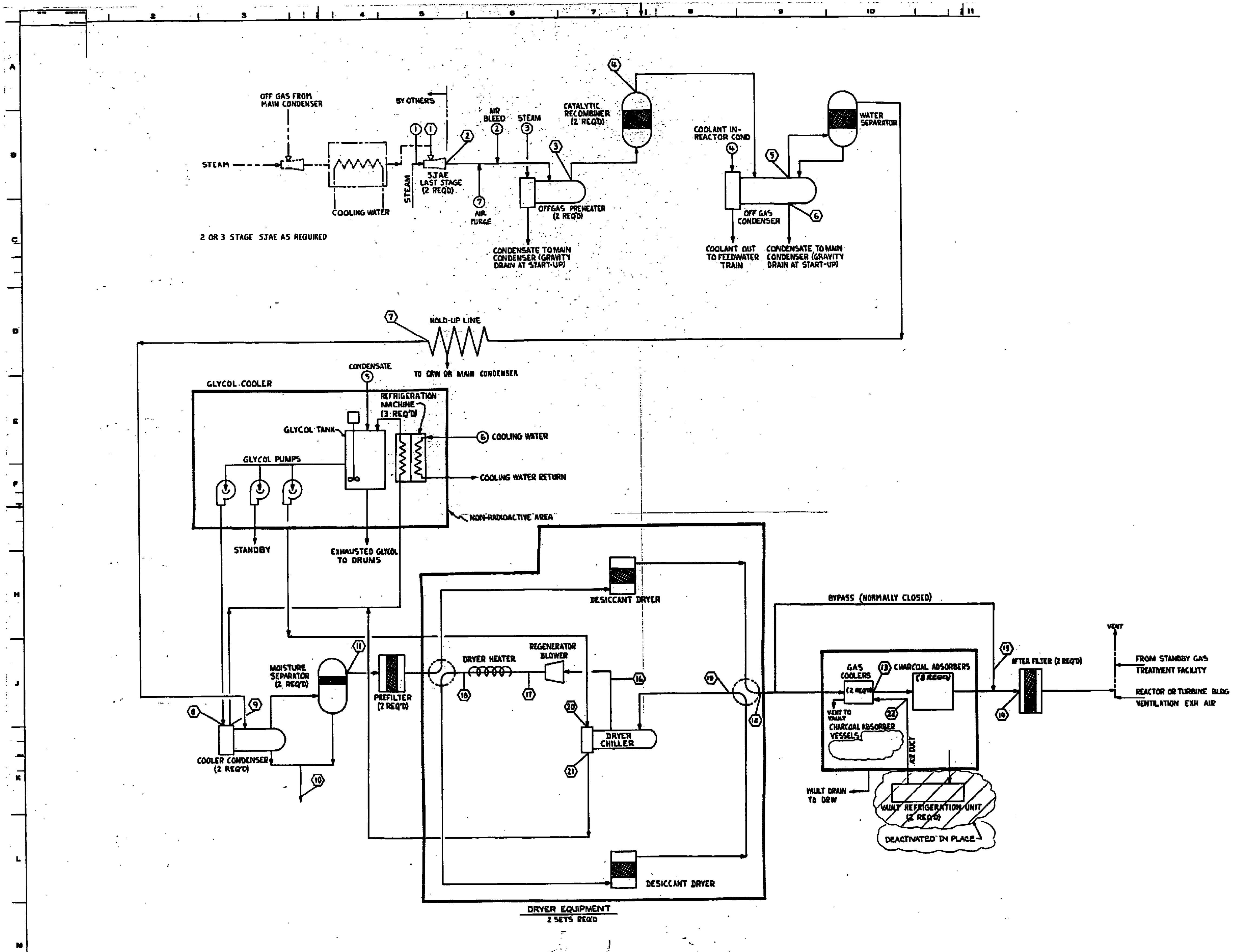
Table 11.3-10

Frequency and Quantity of Steam Discharged to Suppression Pool

Event	Frequency Category	Quantity of Steam (lb/event)
1. RCIC test (monthly)	Moderate	29,000
2. RCIC test (vessel injection at startup)	Moderate	116,000
3. Inadvertent RCIC injection	Moderate	5,000
4. SRV test	Moderate	118,000
5. Inadvertent SRV opening	Moderate	118,000
6. Trip of both recirculation pumps	Moderate	260,000[a]
7. Turbine trip	Moderate	18,000
8. Generator load rejection	Moderate	18,000
9. Pressure regulator failure - open	Moderate	256,000[a]
10. Recirculation flow control failure - decreasing	Moderate	260,000[a]
11. Loss of all feedwater flow	Moderate	267,000[a]
12. Inadvertent closure - all MSIV	Moderate	271,000[a]
13. Loss of condenser vacuum	Moderate	291,000[a]
14. Feedwater control failure - maximum demand	Moderate	270,000[a]
15. Loss of auxiliary transformer	Moderate	251,000[a]
16. Loss of all grid connections	Moderate	280,000[a]
17. Turbine trip w/o bypass	Moderate	126,000[a]
18. Generator load rejection w/o bypass	Moderate	127,000[a]
19. Stuck open SRV	Moderate	817,000

Notes: Events 1, 2, and 3 based on steam flow quantity during test mode per RCIC system process diagram.
Events 4 and 5 assuming test and inadvertent opening at 1000 psi reactor pressure for 10 minutes.
Events 6 through 18 based on event description from Chapter 15.
Event 19 based on results from 251 Standard Plant Suppression Pool Response Analyses.

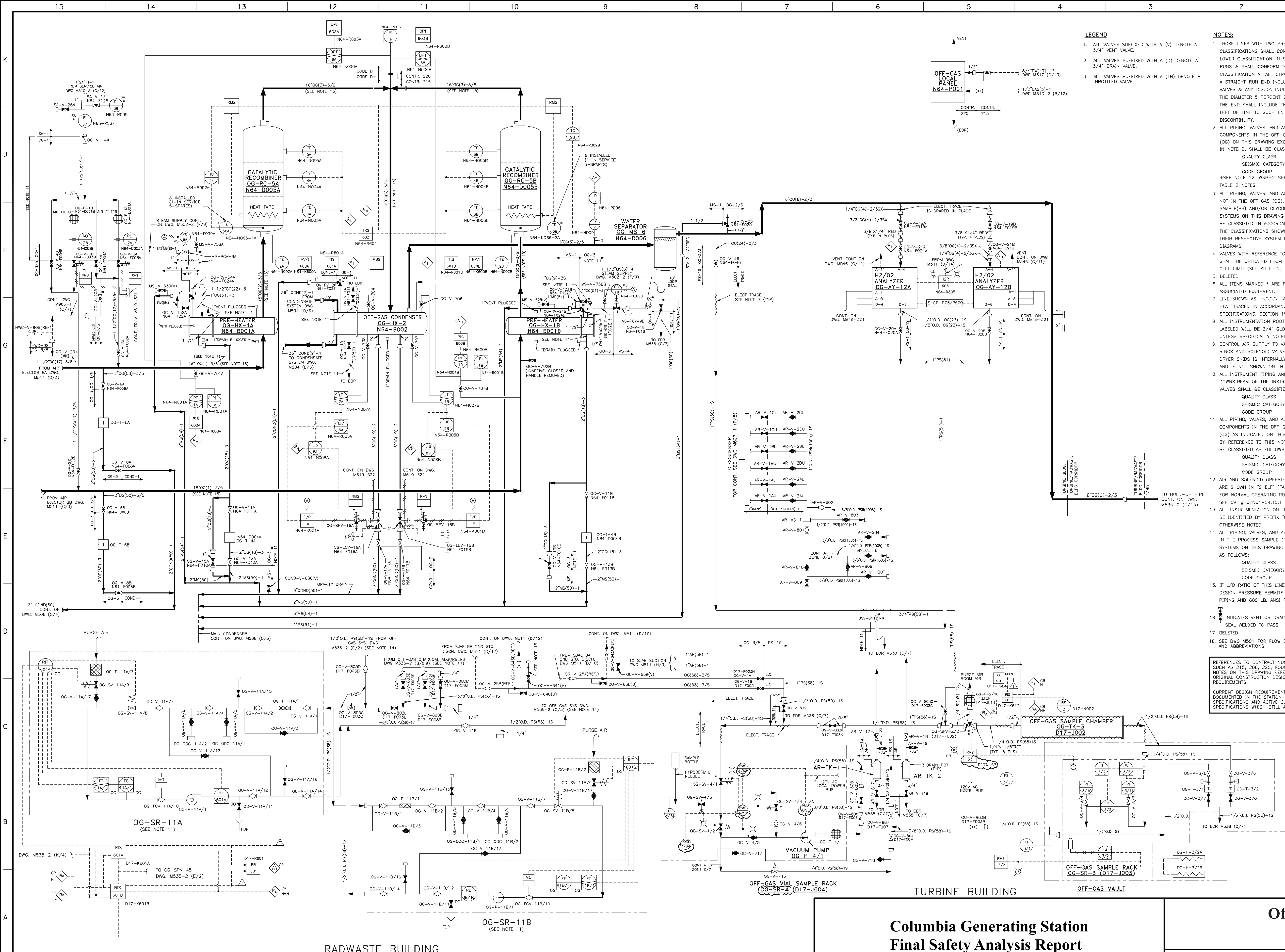
[a] Isolation event. It is assumed that SRV cycling is terminated at T = 30 minutes and reactor depressurization.



Columbia Generating Station
Final Safety Analysis Report

Offgas System - Low Temperature

Draw. No. 02N64-04,6,1 | Rev. 3 | Figure 11.3-1



LEGEND

1. ALL VALVES SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

NOTES:

1. THOSE LINES WITH TWO PRESSURE INTEGRITY CLASSIFICATIONS SHALL CONFORM TO THE LOWER CLASSIFICATION IN STRAIGHT PIPE RUNS & SHALL CONFORM TO THE HIGHER CLASSIFICATION AT ALL STRAIGHT RUN ENDS. A STRAIGHT RUN END INCLUDES BENDS, VALVES & ANY DISCONTINUITY REDUCING THE DIAMETER 5 PERCENT OR MORE. THE END SHALL INCLUDE THE LAST TEN FEET OF LINE TO SUCH END OR DISCONTINUITY.
2. ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS IN THE OFF-GAS SYSTEM (OG) ON THIS DRAWING EXCEPT AS NOTED IN NOTE II, SHALL BE CLASSIFIED AS FOLLOWS:
   - QUALITY CLASS II+
   - SEISMIC CATEGORY II
   - CODE GROUP D+

   +SEE NOTE 12, WNP-2 SPEC., SECT. 15B.1, TABLE 2 NOTES.
3. ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS NOT IN THE OFF GAS (OG), PROCESS SAMPLE(PS) AND/OR GLYCOL(GY) SYSTEMS ON THIS DRAWING SHALL BE CLASSIFIED IN ACCORDANCE WITH THE CLASSIFICATIONS SHOWN ON THEIR RESPECTIVE SYSTEM FLOW DIAGRAMS.
4. VALVES WITH REFERENCE TO THIS NOTE SHALL BE OPERATED FROM OUTSIDE OF CELL LIMIT (SEE SHEET 2)
5. DELETED
6. ALL ITEMS MARKED * ARE FURNISHED WITH ASSOCIATED EQUIPMENT.
7. LINE SHOWN AS ∿∿∿ ARE ELECTRICAL HEAT TRACED IN ACCORDANCE WITH WNP-2 SPECIFICATIONS, SECTION 15D.1.
8. ALL INSTRUMENTATION ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
9. CONTROL AIR SUPPLY TO VALVE LANTERN RINGS AND SOLENOID VALVES ON REGENERATOR DRYER SKIDS IS INTERNALLY MANIFOLDED AND IS NOT SHOWN ON THIS FLOW DIAGRAM.
10. ALL INSTRUMENT PIPING AND TUBING DOWNSTREAM OF THE INSTRUMENT ROOT VALVES SHALL BE CLASSIFIED AS FOLLOWS:
    - QUALITY CLASS II
    - SEISMIC CATEGORY II
    - CODE GROUP D
11. ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS IN THE OFF-GAS SYSTEM (OG) AS INDICATED ON THIS DRAWING BY REFERENCE TO THIS NOTE, SHALL BE CLASSIFIED AS FOLLOWS:
    - QUALITY CLASS II
    - SEISMIC CATEGORY II
    - CODE GROUP D
12. AIR AND SOLENOID OPERATED VALVES ARE SHOWN IN "SHELF" (FAILED) POSITIONS. FOR NORMAL OPERATING POSITIONS SEE CVI # 02N64-04,15,1 THRU 5 (G.E.'S DWG. 761E908AD).
13. ALL INSTRUMENTATION ON THIS DRAWING TO BE IDENTIFIED BY PREFIX "OG" UNLESS OTHERWISE NOTED.
14. ALL PIPING, VALVES, AND ASSOCIATED COMPONENTS IN THE PROCESS SAMPLE (PS) AND GLYCOL (GY) SYSTEMS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS:
    - QUALITY CLASS II
    - SEISMIC CATEGORY II
    - CODE GROUP D
15. IF L/D RATIO OF THIS LINE IS <7.0, DECREASED DESIGN PRESSURE PERMITS USE OF SCH. 40 PIPING AND 600 LB. ANSI RATED FLANGES.
16. INDICATES VENT OR DRAIN VALVE PIPE CAP WAS SEAL WELDED TO PASS HELIUM LEAK TESTING
17. DELETED
18. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.

REFERENCES TO CONTRACT NUMBERS, SUCH AS 215, 206, 220, FOUND IN THE NOTES ON THIS DRAWING REFER TO ORIGINAL CONSTRUCTION DESIGN REQUIREMENTS.

CURRENT DESIGN REQUIREMENTS ARE DOCUMENTED IN THE STATION DESIGN SPECIFICATIONS AND ACTIVE CONTRACT SPECIFICATIONS WHICH STILL APPLY.

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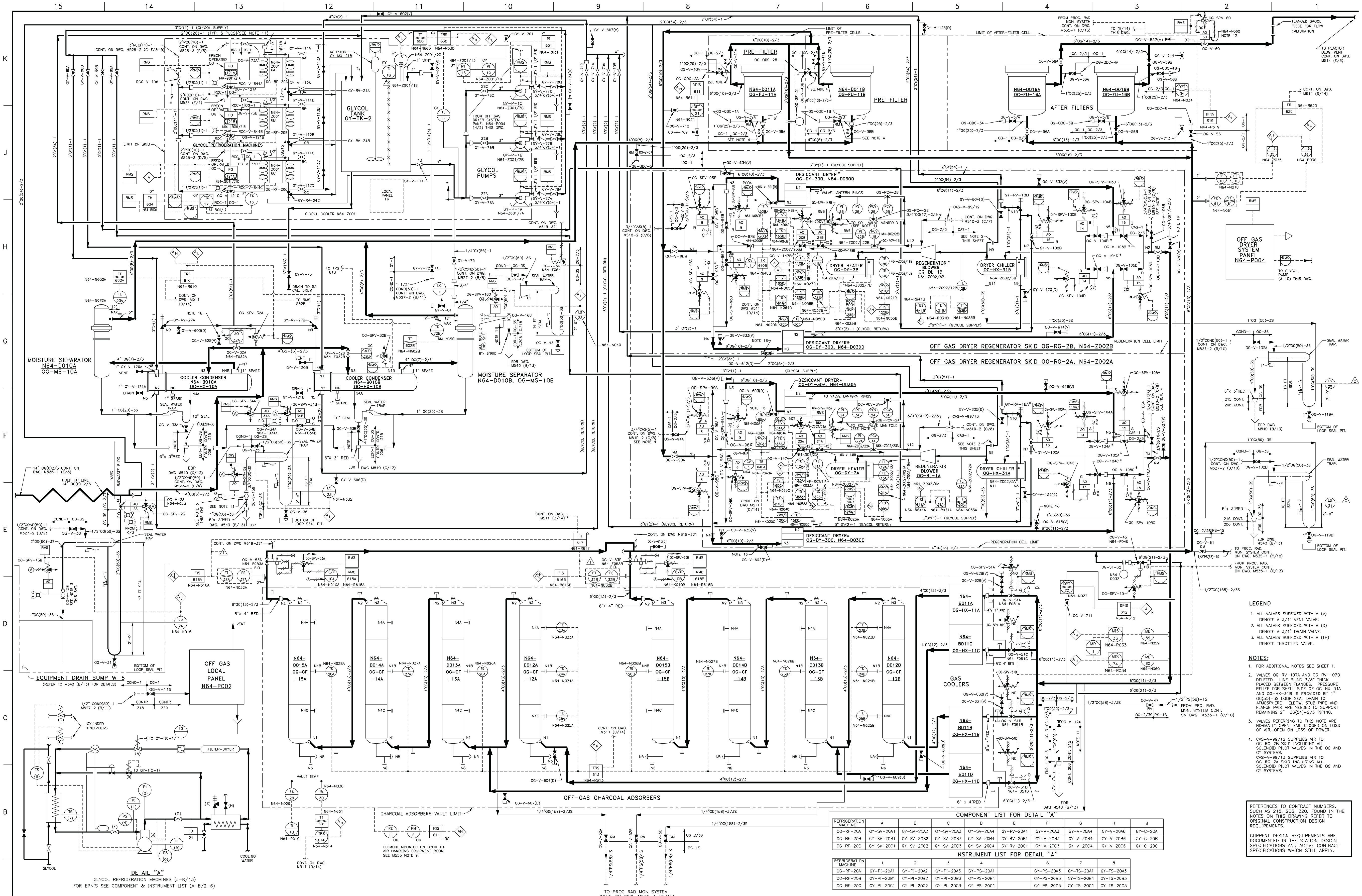
Offgas System P&ID

Draw. No. M535-1	Rev. 81	Figure 11.3-2.1



LEGEND

1. ALL VALVES SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE THROTTLED VALVE.

NOTES:

1. FOR ADDITIONAL NOTES SEE SHEET 1.
2. VALVES OG-RV-107A AND OG-RV-107B DELETED. LINE BLIND 3/8" THICK PLACED BETWEEN FLANGES. PRESSURE RELIEF FOR SHELL SIDE OF OG-HX-31A AND OG-HX-31B IS PROVIDED BY 1" OG(50)-3S LOOP SEAL DRAIN TO ATMOSPHERE. ELBOW, STUB PIPE AND FLANGE PAIR ARE NEEDED TO SUPPORT REMAINING 2" OG(54)-2/3 PIPING.
3. VALVES REFERRING TO THIS NOTE ARE NORMALLY OPEN, FAIL CLOSED ON LOSS OF AIR, OPEN ON LOSS OF POWER.
4. CAS-V-99/12 SUPPLIES AIR TO OG-RG-2B SKID INCLUDING ALL SOLENOID PILOT VALVES IN THE OG AND GY SYSTEMS.
   CAS-V-99/13 SUPPLIES AIR TO OG-RG-2A SKID INCLUDING ALL SOLENOID PILOT VALVES IN THE OG AND GY SYSTEMS.

REFERENCES TO CONTRACT NUMBERS, SUCH AS 215, 206, 220, FOUND IN THE NOTES ON THIS DRAWING REFER TO ORIGINAL CONSTRUCTION DESIGN REQUIREMENTS.

CURRENT DESIGN REQUIREMENTS ARE DOCUMENTED IN THE STATION DESIGN SPECIFICATIONS AND ACTIVE CONTRACT SPECIFICATIONS WHICH STILL APPLY.

COMPONENT LIST FOR DETAIL "A"

REFRIGERATION MACHINE	A	B	C	D	E	F	G	H	J
OG-RF-20A	GY-SV-20A1	GY-SV-20A2	GY-SV-20A3	GY-SV-20A4	GY-RV-20A1	GY-V-20A3	GY-V-20A4	GY-V-20A6	GY-C-20A
OG-RF-20B	GY-SV-20B1	GY-SV-20B2	GY-SV-20B3	GY-SV-20B4	GY-RV-20B1	GY-V-20B3	GY-V-20B4	GY-V-20B6	GY-C-20B
OG-RF-20C	GY-SV-20C1	GY-SV-20C2	GY-SV-20C3	GY-SV-20C4	GY-RV-20C1	GY-V-20C3	GY-V-20C4	GY-V-20C6	GY-C-20C

INSTRUMENT LIST FOR DETAIL "A"

REFRIGERATION MACHINE	1	2	3	4		6	7	8
OG-RF-20A	GY-PI-20A1	GY-PI-20A2	GY-PI-20A3	GY-PS-20A1		GY-PS-20A3	GY-TS-20A1	GY-TS-20A3
OG-RF-20B	GY-PI-20B1	GY-PI-20B2	GY-PI-20B3	GY-PS-20B1		GY-PS-20B3	GY-TS-20B1	GY-TS-20B3
OG-RF-20C	GY-PI-20C1	GY-PI-20C2	GY-PI-20C3	GY-PS-20C1		GY-PS-20C3	GY-TS-20C1	GY-TS-20C3

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Draw. No. M535-2 | Rev. 70 | Figure 11.3-2.2

11.4 SOLID WASTE MANAGEMENT SYSTEM

11.4.1 DESIGN BASIS

Power plant operation results in various solid radioactive wastes that require disposal. These wastes can be in the form of wet solids, such as powdered ion exchange resins from filter demineralizers, expended bead resins from deep bed demineralizers, small quantities of miscellaneous liquid, and miscellaneous dry materials such as paper, rags, plastic, and laboratory wastes.

The objective of the solid waste management system is to collect, monitor, process, and package these waste products in a suitable form for offsite shipment and burial. In designing the system to meet the stated objective, the following criteria were applied:

The system has the capacity to handle the volumes of waste from normal operations and anticipated operational occurrences. The expected annual volumes of wet solid wastes are shown in Table 11.4-1.

The system is designed to process the quantity of waste and concentration of radionuclides listed in Table 11.4-3 while maintaining occupational exposure as low as is reasonably achievable (ALARA). This is done by controlling the pipe run locations for shielding and exposure considerations, placing the process equipment in shielded areas, by providing remote operating stations, and by performing dewatering operations in radiation shields if required. The radwaste building shielding is designed for the highest radioactivity source, reactor water cleanup (RWCU) resins. The equipment layout is shown in general arrangement drawings in Section 1.2. Table 11.4-2 lists capacities, design pressure, and design temperature of the major equipment.

In keeping with the ALARA philosophy and Appendix I to 10 CFR 50, the solid waste management system's contribution to offsite doses is minimized by filtration and by directing the ventilation air flow from areas of low airborne contamination to areas of higher airborne contamination. The filtered ventilation radioactive releases are discussed in Section 11.3.

The solid waste management system operations and procedures are designed to limit the dose to offsite persons from station operations to significantly less than the limits specified in 10 CFR Part 20. Water separated in processing is returned to the liquid waste management system for treatment as described in Section 11.2.2.1 and shown in Figure 11.2-1.

The system can accommodate a variety of shipping container sizes and shapes with and without shields. Provisions are made for the detection and removal of loose surface contamination on the waste containers. The radiation levels of the waste containers are monitored so that

provisions can be made to ensure that shipping regulation radiation levels are not exceeded. Compliance with applicable regulations, e.g., 10 CFR Parts 61 and 71 and 49 CFR is discussed in Sections 11.4.2.9 and 11.4.3.

The safety class, quality group classification, quality class, and seismic category of radwaste systems are specified in Table 3.2-1. The solid waste management system is not designed to Seismic Category I. It is located in the Seismic Category I portion of the radwaste building. The seismic classification of the radwaste building is discussed in Section 3.8.4.1.2.

See Section 3.1.2.6.4 for a discussion of systems provided to meet General Design Criterion 63.

11.4.2 SYSTEM DESCRIPTION

A portable solid waste management system is used by Columbia Generating Station (CGS), as described in the following subsections.

It is required by 10 CFR 61 that, if certain activity criteria are exceeded, wet solid wastes be stabilized by solidification or processed to remove free standing liquids with containment in high integrity containers (HICs). The HICs may provide stability alone or in conjunction with an engineered barrier at the disposal site. The presently installed plant system is designed to interface with the portable dewatering/drying system which, when coupled with HICs, meet the requirements for stabilization in compliance with 10 CFR 61.

There are two resin dewatering systems used at CGS: the Self-Engaging Rapid Dewatering System (SERDS) and the Self Engaging Dewatering System (SEDS). The resin dewatering systems are vendor provided with NRC approved topical reports (RDS-25506-01-NP, Revision 1 for the SERDS and CNSI-DW-11118-01-P for the SEDS), and the systems are operated according to the technical requirements of this report. The types and quantities of waste to be processed are described below. The NRC licensed shipping casks and associated liners are used for processing, transporting, and disposing of wet solid wastes when required.

System operation is closely monitored by Energy Northwest personnel. If a vendor operates the dewatering system, the vendor will be required to submit their operating procedures for Energy Northwest review and approval. The vendor procedures will either be incorporated into a CGS procedure or will be approved "as is" by CGS prior to use.

Dewatering of resins to meet applicable dewatering criteria is conducted in accordance with approved procedures inside the radwaste building in the liner storage area or in shipping casks on trucks, where any spills are routed to existing floor drain sumps and the building ventilation filtration system ensures no unfiltered airborne releases occur from dewatering activities.

11.4.2.1 General

The sources of the various radioactive wet resin waste inputs to the system are shown in Figure 11.2-1. Table 11.2-11 shows the design basis expected frequency of input, the quantities of solids generated, the radioactivity level of the solids after accumulation, and the volume of liquid used in sluicing accumulated solids to the processing equipment. The excess liquid is subsequently returned to the liquid waste management system.

These values are based on experience from operational BWR nuclear power stations. Figure 11.4-1 shows the solid waste management system up to and including the portable portion of the system described in the vendor's topical reports. The phase separation and concentration portions of the system are shown on Figures 5.4-22, 10.4-5, 11.2-2, and 11.2-3. Tanks containing radioactive waste are provided with overflow connections which direct any overflow to drain sumps.

The solid waste processing areas are located in the radwaste building, where wet and dry solid wastes may be processed. Wet solid wastes include backwash resin from the RWCU system, the condensate filter demineralizer system, the fuel pool filter demineralizers, the floor drain and equipment drain filter demineralizers, and spent resin from the floor drain demineralizer and the waste demineralizer. Dry solid wastes include items such as rags, paper, plastics, small equipment parts, and laboratory wastes.

11.4.2.2 Radwaste Disposal System For Reactor Water Cleanup Resin

The backwash discharge from the cleanup filter demineralizers is collected and concentrated in two 4500-gal cleanup phase separators which are located below the cleanup demineralizers in the radwaste building. After several backwashes are accumulated, the waste is transferred to the portable dewatering system.

The cleanup phase separators are designed to concentrate the resin from 0.5% by weight solids to approximately 5% by weight solids by sedimentation and decantation of the slurry. While the working separator is filling, the other previously filled tank is held isolated to the extent practicable to allow for additional decay of resin activity.

After each backwash batch is received by the working separator, the batch is allowed to settle for a period of time and the decantate is then transferred by pumping to the waste collector tank. When sufficient resin has accumulated, the working separator is isolated and allowed to stand for a period to permit radioactive decay. At the end of this decay period the sludge is fluidized to approximately 5% weight solids and transferred by pumping to the portable dewatering system.

11.4.2.3 Radwaste Disposal System For Condensate Demineralizer Resin

The backwash discharge from the condensate filter demineralizers is collected in the condensate backwash receiving tank which is located below the condensate filter demineralizers in the radwaste building. After collection, the waste is transferred by pumping to one of the two condensate phase separators for processing.

Operation of the condensate phase separators is similar to that for the cleanup phase separators. Backwash resin is received at 0.5% by weight solids and concentrated to approximately 5% by weight solids, allowed to stand for a period of radioactive decay and then decanted and transferred by pumping to the portable dewatering system.

11.4.2.4 Radwaste Disposal System For Fuel Pool, Floor Drain, and Waste Collector Filter Resin

Backwash resin wastes from the fuel pool filter demineralizers, floor drain, and waste collector filter demineralizers are backwashed to the waste sludge phase separator tank. The waste sludge phase separator is designed to concentrate the resin from 0.5% by weight solids to approximately 5% by weight solids by sedimentation and decantation.

After each backwash batch is received by the separator, it is allowed to settle for a period of time and the decantate is then transferred by pumping to the floor drain collector tank.

When an appropriate quantity of resin is accumulated, the resin is fluidized to approximately 5% by weight solids and transferred by pumping to the portable dewatering system.

11.4.2.5 Radwaste Disposal System For Spent Resin

Spent bead resins from the floor and equipment drain polishing demineralizers are hydropneumatically transferred to the spent resin tank. The tank is designed to retain one batch of resins plus resin transfer water plus freeboard.

The decay time of any single batch is governed by the need to make the spent resin tank available to receive a subsequent batch from an alternate demineralizer. The frequency of spent resin discharge from the floor and equipment drain polishing demineralizers is estimated to be about once every 2 months. Each batch of the spent resin is transferred at approximately 40% by weight solids to the portable dewatering system.

11.4.2.6 Resin Container Handling and Storage

Filled containers from the dewatering operation in the container storage area are lifted by a crane and placed on a track-riding dolly. The dolly moves the containers from the storage area to the loading area where another crane lifts the container onto the truck for offsite shipment.

Alternatively, the dewatering operation for high activity resin waste can be performed in a cask on or off the truck bed. General locations and arrangements are shown in Figure 1.2-13.

11.4.2.7 Miscellaneous Dry Solid Waste System

Dry active waste may consist of air filtration media, miscellaneous paper, plastic and rags from contaminated areas, contaminated clothing, tools and equipment parts which cannot be effectively decontaminated, solid laboratory wastes, and other similar materials. The radioactivity of much of this waste is low enough to permit handling by contact. Compressible wastes may be compacted into metal containers to reduce their volumes. Alternately, container vans (C-vans) or other containers suitable for shipment may be used for dry radioactive waste shipped to a vendor for volume reduction services.

A relatively small quantity of high activity compactable and noncompactable dry active radioactive waste (DAW) may be loaded into the open top or encapsulation liners. Other containers whose geometry is compatible with shipment in shielded shipping casks may be used.

A locally controlled compactor system, if used, will include the following features: hydraulic pump with motor, hydraulic oil storage, high-efficiency filters, fan, and accessories. Ventilation air will be pulled across the top of the containers and then through high-efficiency filters by a fan during the compression process and exhausted to the radwaste building exhaust system described in Section 9.4.

Solid wastes and other nonliquid radioactive material and C-vans may be stored temporarily near the truck loading area outside the radwaste building, on the outdoor curbed pad adjacent to the radwaste building, or other suitable location. Noncompressible solid wastes are packaged in containers suitable for the waste. Low activity waste can be stored until enough is accumulated to permit economical transportation to an offsite disposal site or to a vendor for volume reduction of the waste prior to disposal.

Irradiated reactor components consisting of spent control rod blades, fuel channels, in-core ion chambers, and other equipment are stored in the spent fuel storage pool to allow for radioactive decay prior to shipment.

The packaging of dry solid wastes will be administratively controlled to ensure the 10 CFR 61 and/or burial facility free standing liquid criteria are met.

11.4.2.8 Expected Volumes

The design basis expected frequency of solids input, the quantities of solids generated, and the radioactivity of the solids after accumulation are shown in Table 11.2-11. Table 11.4-3 shows the expected solids production rate and significant nuclides associated with each batch of

dewatered waste. The distribution of these nuclides (dependent on the concentration in the reactor water) corresponds to the design basis offgas noble gas release rate as described in Section 11.3.1.

Table 11.4-1, excluding dry and compacted waste, shows the expected annual container production of solid wastes based on the process diagram data inputs. No decay after container filling operations has been considered.

11.4.2.9 Packaging

Radioactive wastes are packaged and shipped from CGS in containers that meet the requirements established in 49 CFR 171-180 for the Department of Transportation and 10 CFR 71 for the NRC.

Packaging of wastes being dewatered is typically performed remotely behind shielding as described below. Empty containers may be brought into the processing area using the dolly and filled with waste for processing. The quantity of wastes packaged in the container is controlled by operating procedures.

Dewatered liners are capped, surveyed for surface contamination, and decontaminated as necessary prior to shipment.

11.4.2.10 Storage Facilities

The general arrangement drawings in Section 1.2 show the layout of the radwaste handling areas in the radwaste building.

For the liners presently in use (of up to 210 ft^3 capacity), the storage area can accommodate about 15 filled liners. High activity containers can be stored to allow for 6 months decay prior to shipment if necessary. It is expected, however, that most radwaste containers will be shipped within 1 to 3 months of generation.

11.4.2.11 Shipment

The following describes a typical loading sequence for liners: A truck containing the cask is moved into the truck loading area. The dolly is moved to the storage area loading station and a capped container of waste is placed on the dolly by the storage area crane. The dolly is moved back to the truck loading area where the liner is lifted into the cask, and the cask lid is placed on the cask. The cask is decontaminated if necessary for shipment. Similar operations are performed when loading unshielded containers onto the truck. Dewatering of high activity wastes can also be performed in liners “in cask” in place on the shipping truck.

Any unshielded containers found to have external contamination exceeding 49 CFR limits are decontaminated. Further smear tests and cleaning are carried out as required until the activity on the container is within acceptable limits.

Radwaste tank failure and spent fuel cask drop incidents are discussed in Sections 15.7.3 and 15.7.5, respectively.

11.4.2.12 Process Monitoring

Process monitoring is performed by the dewatering system operator and the operator in the radwaste control room. They are in communication during waste transfer to the dewatering system. The dewatering system processing is monitored by remote closed-circuit TV cameras and other instrumentation as described in the topical reports for the process.

Each RWCU phase separator is equipped with one level indicating device for total liquid level. The total liquid level indicator utilizes an air bubbler and a pressure sensing level transmitter which drives a 0-100% level gauge and a high-level alarm in the radwaste control room. The level transmitter also drives a level indicator on the local control panel and provides control functions for the decant pump, the resin discharge pump, and the phase separator inlet selector valve.

The waste sludge phase separator has total liquid level indication. It uses an air bubbler and a pressure-sensing level transmitter. In addition to the level gauge and high-level alarm in the radwaste control room, the level transmitter provides control inputs to the decant pump, the stop and flush circuit on the sludge discharge pump, and the discharge valves from the waste collector and floor drain collector tanks to the waste sludge phase separator.

The condensate phase separators level instrumentation is the same as that described for the RWCU phase separators. Level indication for the spent resin tank is essentially the same as that described for the RWCU phase separators. The concentrated waste measuring and waste mixing tanks are not in service.

11.4.3 PROCESS CONTROL PROGRAM

11.4.3.1 Objective

The objectives of the process control program are to characterize and classify radwaste and ensure the complete solidification of all wet wastes being solidified and to ensure that dewatered wet wastes and disposal packages meet the free standing liquid and stability requirements of 10 CFR 61. To meet these objectives, the process control program has incorporated the recommendations set forth in NUREG-0800 and Branch Technical Position - ETSB 11-3.

11.4.3.2 Process Control Program

To ensure that acceptable waste forms are produced for disposal in compliance with the requirements in 10 CFR 61, the process control program provides for characterization of individual waste streams, classification of final waste products, proper disposal packaging, and verification that waste dewatering and/or solidification has been successful. In addition, nonstable waste form processing, waste storage, handling, and transportation activities take place under the process control program to ensure compliance with all applicable regulations in 10 CFR Parts 20, 61, and 71.

Stability of waste form at Columbia Generating Station is normally achieved through the combined properties of dewatered media and HICs or HICs plus engineered barriers. If solidification for stability is performed at Columbia Generating Station, it will be done in accordance with approved procedures to meet applicable disposal site license conditions and other applicable requirements.

Other processing methods such as portable deionization, solidification for nonstable waste form, evaporation, filtration, etc., may be used to process nonhazardous radioactively contaminated liquids. If these activities will result in the disposal of waste at a licensed burial site, they will be conducted in accordance with approved procedures to ensure that applicable disposal site license conditions and other requirements are met. Solidification agents listed in the burial facility license may be used to ensure free-standing liquid criteria are achieved.

The process control program contains the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to ensure compliance with 10 CFR Parts 20, 61, and 71, state regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive wastes.

Changes to the process control program are documented, and records of reviews performed are retained for the duration of the operating license. This documentation contains sufficient information to support the change together with the appropriate analyses or evaluations justifying the change and a determination that the change will maintain the overall conformance of the solidified waste product to the existing requirements of federal, state, and other applicable regulations. Changes become effective after review and acceptance in accordance with the Operational Quality Assurance Program Description (OQAPD).

The radioactive waste process control program incorporates the following elements:

a. Waste stream descriptions,
b. Process controls,
c. Characterization,

d. Processing and stabilization,
e. Sampling,
f. Scaling factors,
g. Classification,
h. Computer code usage,
i. Analytical methods,
j. Packaging,
k. ALARA,
l. Shipping,
m. Documentation,
n. Equipment maintenance and calibration,
o. Minimization and segregation,
p. Storage,
q. Trending, and
r. Reporting.

Plant procedures implement the process control program.

11.4.3.3 Process Control Systems

The transfer of wet solid wastes to the processing system is monitored and controlled from the process control panel in the radwaste control room. The resin transfer system interfaces with the portable dewatering system at the point of connection with the dewatering equipment.

At the interface point, the applicable dewatering system is operated per the technical requirements of the NRC-approved topical report and is designed to ensure that processed waste in conjunction with the dewatering HIC or liner is prepared for burial and meets the 10 CFR Part 61 criteria. The process control program is described and controlled by procedure. The dewatering systems control panels are in a remote location shielded from the waste processing resin containing components. It is designed for automatic operation and provides indication and alarm for liner level, temperature, pressure, and other operating parameters.

If a waste processing vendor is used, they are monitored by Energy Northwest to ensure that applicable procedures, Energy Northwest or vendor, are being followed and that an acceptable end product is formed as described in the procedure used.

11.4.3.4 Waste Characterization

The wet wastes at CGS to be processed are characterized in individual streams for RWCU resins, equipment drain radioactive (EDR) and floor drain radioactive (FDR) powdered resins, EDR and FDR bead resins, and condensate resins. The four major systems producing wastes are described below.

a. Floor drain system - Wastes from the turbine building, reactor building, and radwaste building floor drain sumps are routinely monitored and collected for processing in the floor drain collector tank. The floor drain filter and demineralizer sludges are combined with equipment drain filters and sludges to form a mixture which is sampled prior to processing. Similarly, the EDR/FDR bead (polishing) resins are also sampled prior to processing.

b. Equipment drain system - Wastes from the turbine building, reactor building, and radwaste building equipment drain sumps are routinely monitored and collected in the waste collector tank. Sludges and resins from the high purity filter demineralizers are sampled prior to processing as described above in the FDR system description.

c. Condensate filter demineralizer system - The condensate polishing filter demineralizers use pressure precoated ion exchange media filters. Resins are sampled prior to processing.

d. Reactor water cleanup filter demineralizer system - The RWCU filter demineralizers use pressure precoated ion exchange media filters. Resins are sampled prior to processing.

Accumulations in collection tanks and phase separators will be tracked to aid in determining processing schedules and potential problems in system operation by tracking volumes. Selected coolant isotopic concentrations are trended to provide early indication of changes important to waste classification.

The waste streams in the foregoing systems are characterized by a periodic sampling and analysis program that establishes plant-specific isotopic correlation factors and relationships for inferring concentrations (i.e., scaling factors) of all 10 CFR 61 nuclides from easy to measure gamma-emitting species.

Individual waste stream activities and concentrations are determined for each batch prior to shipment for disposal. The evaluation of historical evidence is used as a screen to determine the appropriate disposal container followed by a formal evaluation based on actual samples or dose to curie determinations as appropriate.

11.4.3.5 Processing Methods (Wet Wastes)

The dewatering units are portable systems containing all necessary equipment and controls for removing the free water from ion-exchange resins and filter media.

Containers used for dewatering are furnished to CGS with factory installed “internals” functionally identical to those used during qualification testing. The determination of “functionally identical” may be performed by the vendor in accordance with their NRC approved Quality Assurance program or by CGS in accordance with the Quality Assurance program. The internals are free-standing and self-supporting, without protuberances which might damage the container. A fill head interfaces with the container and liner dewatering internals. The dewatering equipment is operated using the labeled controls on the control panel.

The resins that are below the Class A limits of 10 CFR 61 are normally dewatered in carbon steel containers. Resins exceeding Class A limit are dewatered in HICs to provide waste form stability.

The RDS process includes use of moisture indicators and the SEDS process uses dewatering verification to verify that free liquid criteria are met.

11.4.3.6 Control Instrumentation and Sampling Program

Processing of radwaste at CGS is conducted using instrumentation and controls for each batch to ensure that (a) suitable, well characterized waste is delivered to the various waste processing subsystems, (b) adequate process control information is provided to system operators to ensure adherence to proper operating parameter limits, such as tank levels, flow rates, release concentrations, etc., and (c) sufficient information is available to limit personnel exposures in conformance with the ALARA program.

The control instrumentation used at CGS includes in-process instruments, as well as portable radiation monitoring instruments.

The sampling program at CGS is a twofold program. Individual waste streams are characterized and classified by CGS personnel with the responsibility for shipping radioactive waste. Additionally, samples are sent to an offsite laboratory on a periodic basis to validate scaling factors.

11.4.3.7 Maintenance and Calibration

The control of the waste processing system is ensured by a thorough preventative and corrective maintenance program described in plant procedures for scheduled maintenance systems and maintenance work requests.

The control instrumentation is an integral part of the process system. The instruments providing the controlling functions are calibrated on a predetermined schedule and after each maintenance activity as applicable.

The periodic verification of calibration or recalibration assures that the process control program associated instruments are maintained and that conditions within the system are known.

The maintenance and calibration activities are performed in accordance with written procedures.

11.4.3.8 Waste Processing System Capacity

Wastes can be dewatered in up to 210 ft^3 liners in approximately 8 to 12 hr. This gives two shift operation a processing capacity of at least 100,000 ft^3 of waste per year. Similar processing capacity is available from portable solidification equipment which could be moved onsite if solidification to provide stabilization became necessary.

Dry wastes are segregated and monitored to reduce volumes where practicable. The expected design basis volume of radwaste is approximately 20,000 ft^3/year, however, actual volumes have been reduced to less than 10,000 ft^3/year. Forced outages and refueling outages increase volume but can be limited by preplanning material usage.

The radwaste processing capacity at CGS is sized to provide the needed capacity for anticipated occurrences and normal operation. This includes wet wastes, liquids, and solid wastes. Table 11.4-1 lists some of the major flowrates and capacities for several of the waste processes. Table 11.4-2 tabulates the major equipment items in the permanently installed waste processes.

11.4.3.9 Waste Storage Capacity

The storage capacity for DAW is adequate due to containerization allowing outside storage prior to shipping.

For the liners currently in use (up to 210 ft^3 capacity), approximately 15 liners can be readily accommodated. This storage space is available for the portable waste processing system and to store processed liners. Sufficient casks and transport packages have been factored into the planning to allow shipment of processed radwaste at a rate to preclude storage problems.

11.4.3.10 Compliance With ALARA Principles

a. Facility layout for the portable dewatering equipment - The portable dewatering systems are designed to meet Regulatory Guide 8.8. The waste container filling, capping, and decontamination operations are either performed remotely from a control area or as quickly as possible to minimize exposures. The system operator can view container filling and processing operations using a television monitoring system to prevent overfilling the waste container.

The portable system is located to limit the exposure of personnel to high dose rate piping and is designed to minimize the accumulation of crud deposits in the system. Piping and pumps are designed to allow complete flushing and, when possible, piping is flushed prior to maintenance. The control area is located in a relatively low radiation area away from the dewatering operation. Equipment with clean components is segregated from the areas containing waste except in the processing area. This allows maintenance on non-contaminated equipment to be performed in low radiation areas.

The placement of the portable processing system was determined based on criteria from Regulatory Guide 8.8 on ALARA and quality assurance provisions from Regulatory Guide 1.143. Processing radiation shields, in-cask processing and/or careful planning of processing activities contribute to efforts to minimize exposures from radwaste system operations.

b. Plant layout for ALARA - The installed radwaste processing system (excluding the portable dewatering system) was designed for remote operation from the radwaste control room in the 467-ft elevation of the radwaste building. The control room is designed with visual aids mimicking the processes being controlled. This allows for visual indication of processing status.

c. Exposure control - The radwaste operators control the processing of radioactive waste from the radwaste control room that is located in a low radiation area.

Most radwaste processing system components and systems can be remotely aligned from the radwaste control room without going into the radiation areas. This allows flexibility of operation and ensures processing capacity is maintained with reduced exposure to personnel. The systems and components can be flushed remotely from the control room reducing radiation levels for maintenance activities, also helping meet the ALARA concept.

The installed systems at CGS are sized with the capacity to process peak volumes of waste to reduce the effluents offsite, meeting criteria of 10 CFR 50, Appendix I.

The processing control from a remote control room and the ability to process all waste streams meets the intent of ALARA for both onsite and offsite.

11.4.3.11 Unanticipated Wastes

The waste streams at CGS have been characterized for normal expected waste components. Periodically, there may be waste produced from operations such as decontamination or cleaning that have not been characterized.

These wastes are classified and processed for disposal based on the practical experience of the CGS staff. It is unlikely that any wastes will be produced at CGS that cannot be prepared for disposal in accordance with 10 CFR 61. When changes in the system or process do occur that are out of the ordinary, processing requirements will be determined on a case-by-case basis.

11.4.3.12 Waste Classification

Waste is classified at CGS to determine if wastes meet 10 CFR 61 Class A, B, or C. The individual waste stream will be sampled and analyzed for nuclides in Tables 1 and 2 of 10 CFR 61 to establish the waste classification. This analysis may in many instances use scaling factors to tie difficult to measure nuclides, to more easily identified gamma emitting nuclides. The requirements for the determination of these scaling factors is administratively controlled by plant procedure.

11.4.3.13 Waste Packaging and Shipping

The radioactive waste shipped from CGS for disposal or further processing is containerized, prepared, and shipped in accordance with applicable state, DOT, and NRC regulations.

Table 11.4-1

Waste Processing Systems Capacities

Waste Processing Systems	Flow Rates and Capacities
Liquid Waste	
EDR subsystem	14,341 to 104,826 gpd
EDR storage	135,000 gal
FDR subsystem	5700 to 55,289 gpd
FDR storage	40,000 gal
Chemical waste subsystem	1500 to 20,600 gpd
Chemical waste storage	65,000 gal
Filter sludge and chemical waste concentrates	
Normal	8000 to 16,000 ft^3/year
Processed volume for disposal	8000 to 16,000 ft^3/year
Portable solidification system	
Normal	200 ft^3/day
Maximum	100,000 ft^3/year[a]
Storage	72 50-ft^3 liners
Dry compactible waste	
Compactible radwaste	20,000 ft^3/year
Dewatering system	
Normal	200 ft^3/12-hr shift
Maximum	100,000 ft^3/year[a]

[a] Figured on two shifts with increase in personnel.

Table 11.4-2

Solid Waste Management System Major Equipment Items

Equipment	Number Required	Construction	Design Pressure	Design Temperature	Capacity
Cleanup phase separators	2	Stainless steel shell and internals	Atmospheric	250°F	4500 gal each
Cleanup sludge discharge mixing pump	1	Stainless steel	150 psig	150°F	210 gpm at 170 ft TDH
Cleanup decant pump	1	Stainless steel	150 psig	150°F	53 gpm at 50 ft TDH
Condensate backwash receiving tank	1	Stainless steel shell and internals	Atmospheric	150°F	19,000 gal
Condensate backwash transfer pump	1	Stainless steel		150°F	450 gpm at 50 ft TDH
Condensate phase separator	2	Epoxy-coated carbon steel shell, stainless steel internals	Atmospheric	250°F	23,500 gal each
Condensate sludge discharge mixing pump	1	Stainless steel	150 psig	150°F	420 gpm at 160 ft TDH
Condensate decant pump	1	Stainless steel	150 psig	150°F	450 gpm at 50 ft TDH
Waste sludge phase separator tank	1	Epoxy-coated carbon steel, stainless steel internals	Atmospheric	150°F	13,000 gal
Waste decant pump	1	Stainless steel	150 psig	150°F	53 gpm at 50 ft TDH
Waste sludge discharge mixing pump	1	Stainless steel	150 psig	150°F	210 gpm at 105 ft TDH

Table 11.4-2

Solid Waste Management System Major Equipment Items (Continued)

Equipment	Number Required	Construction	Design Pressure	Design Temperature	Capacity
Spent resin tank	1	Stainless steel shell and internals	Atmospheric	150°F	1200 gal
Spent resin pump	1	Stainless steel	150 psig	150°F	21 gpm at 105 ft TDH
Decontamination solution concentrated waste tank[a]	2	Stainless steel shell and internals	Atmospheric	150°F	700 gal each
Decontamination solution concentrated waste pump[a]	1	Stainless steel	150 psig	150 °F	30 gpm at 70 ft TDH
Concentrated waste measuring tank[a]	1	Stainless steel	Atmospheric	150°F	400 gal
Transfer dolly[b]	1				

[a] Not in service.

[b] Track riding dolly for transfer of waste containers between the storage area and truck shipping area.

Table 11.4-3

Significant Isotope Activity in Dewatered Waste

Stream	Clean up Sludge	Waste Sludge	Equipment Drain Resin	Floor Drain Resin	Condensate Sludge
Batch Solid Production	524 lb/60 days	220 lb/3.4 days	1539 lb/66 days	1539 lb/67 days	3300 lb/18.5 days
Isotopes			Ci/ft^3		
^{89}Sr	0.63	--	9.0×10^{-3}	6.5×10^{-5}	4.8×10^{-3}
^{90}Sr	0.16	--	1.0×10^{-3}	7.2×10^{-6}	8.4×10^{-4}
^{91}Sr	--	--	5.2×10^{-3}	3.7×10^{-5}	--
^{92}Sr	0.46	--	1.0×10^{-3}	3.4×10^{-5}	7.1×10^{-6}
^{90}Y	0.16	--	1.0×10^{-3}	3.4×10^{-5}	8.4×10^{-4}
^{91}Y	0.023	--	2.3×10^{-3}	--	1.7×10^{-3}
^{91m}Y	--	--	3.6×10^{-3}	--	--
^{92}Y	--	--	1.0×10^{-3}	--	--
^{95}Zr	0.29	8.1×10^{-4}	--	--	4.2×10^{-4}
^{95}Nb	0.44	1.1×10^{-3}	8.4×10^{-4}	--	6.3×10^{-4}
^{99}Mo	--	2.0×10^{-3}	2.9×10^{-3}	2.4×10^{-5}	1.3×10^{-4}
^{99m}Tc	0.0015	1.3×10^{-3}	9.7×10^{-3}	1.1×10^{-4}	1.3×10^{-4}
^{103}Ru	--	--	--	--	2.1×10^{-4}
^{122}Sb	0.020	2.0×10^{-4}	1.1×10^{-5}	--	--
^{129m}Te	0.038	--	1.3×10^{-4}	6.9×10^{-7}	--
^{129}Te	0.024	--	--	--	--
^{132}Te	--	--	1.6×10^{-2}	1.3×10^{-4}	1.3×10^{-4}
^{83}Br	--	--	1.3×10^{-4}	4.5×10^{-6}	--

Table 11.4-3

Significant Isotope Activity in Dewatered Waste (Continued)

Stream	Clean up Sludge	Waste Sludge	Equipment Drain Resin	Floor Drain Resin	Condensate Sludge
Batch Solid Production	524 lb/60 days	220 lb/3.4 days	1539 lb/66 days	1539 lb/67 days	3300 lb/18.5 days
Isotopes			Ci/ft^3		
^{131}I	0.077	7.8×10^{-4}	1.01×10^{-2}	7.4×10^{-5}	2.3×10^{-2}
^{132}I	--	--	1.7×10^{-3}	3.5×10^{-5}	1.3×10^{-4}
^{133}I	--	--	7.3×10^{-3}	7.7×10^{-5}	--
^{134}I	--	--	8.4×10^{-4}	4.7×10^{-5}	--
^{135}I	--	--	3.6×10^{-3}	5.7×10^{-5}	--
^{134}Cs	0.17	3.0×10^{-3}	6.0×10^{-4}	4.9×10^{-6}	4.2×10^{-4}
^{136}Cs	0.0078	--	1.3×10^{-4}	--	--
^{137}Cs	0.18	4.4×10^{-3}	1.0×10^{-3}	7.6×10^{-6}	8.4×10^{-4}
^{138}Cs	--	--	2.1×10^{-4}	--	--
^{137m}Ba	0.17	--	1.0×10^{-3}	--	6.3×10^{-4}
^{139}Ba	--	--	8.4×10^{-4}	3.9×10^{-5}	--
^{140}Ba	0.076	--	1.0×10^{-2}	2.6×10^{-5}	2.7×10^{-3}
^{141}Ce	--	--	1.3×10^{-4}	--	3.8×10^{-3}
^{144}Ce	0.019	--	1.3×10^{-4}	4.7×10^{-6}	6.3×10^{-4}
^{140}La	0.087	3.9×10^{-5}	1.0×10^{-2}	--	2.9×10^{-3}
^{141}La	--	--	2.1×10^{-4}	--	--
^{142}La	--	--	1.3×10^{-4}	--	--
^{143}Pr	0.022	--	1.3×10^{-4}	--	8.4×10^{-4}

Table 11.4-3

Significant Isotope Activity in Dewatered Waste (Continued)

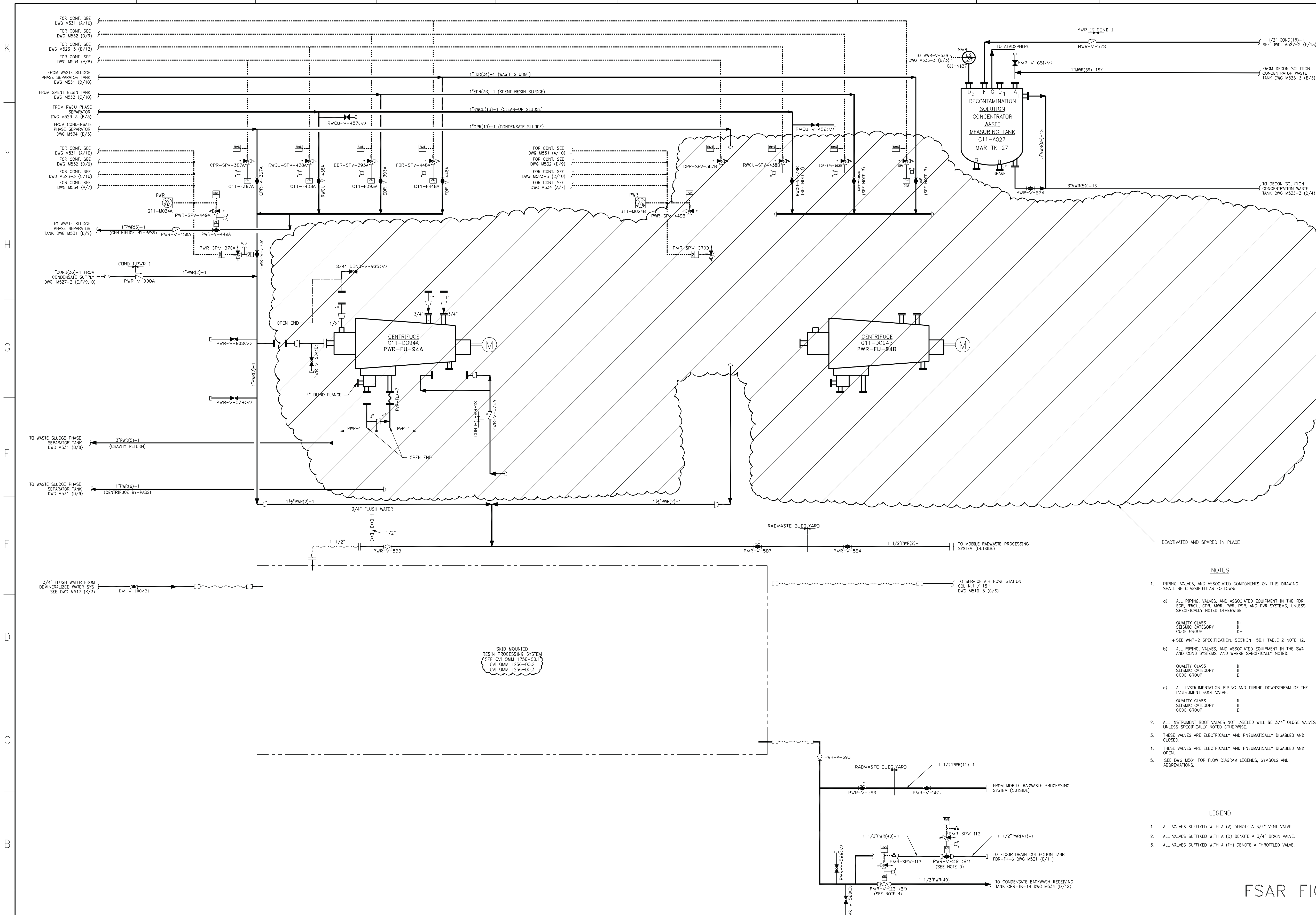
Stream	Clean up Sludge	Waste Sludge	Equipment Drain Resin	Floor Drain Resin	Condensate Sludge
Batch Solid Production	524 lb/60 days	220 lb/3.4 days	1539 lb/66 days	1539 lb/67 days	3300 lb/18.5 days
Isotopes			Ci/ft^3		
^{144}Pr	0.019	--	1.3×10^{-4}	--	6.3×10^{-4}
^{239}Np	--	--	5.4×10^{-2}	4.5×10^{-4}	--
^{51}Cr	1.8	1.5×10^{-2}	4.4×10^{-4}	4.1×10^{-6}	3.1×10^{-3}
^{54}Mn	0.089	2.1×10^{-3}	1.3×10^{-4}	7.0×10^{-5}	6.6×10^{-4}
^{56}Mn	--	2.3×10^{-4}	2.2×10^{-4}	7.8×10^{-6}	--
^{58}Co	1.8	3.3×10^{-3}	8.4×10^{-3}	6.1×10^{-5}	6.2×10^{-2}
^{60}Co	0.65	2.4×10^{-2}	1.1×10^{-3}	9.2×10^{-6}	9.7×10^{-3}
^{59}Fe	0.018	2.6×10^{-4}	1.3×10^{-4}	8.3×10^{-7}	8.8×10^{-4}
^{65}Zn	2.1	7.4×10^{-2}	3.5×10^{-4}	2.3×10^{-4}	4.4×10^{-4}
^{110m}Ag	0.14	8.8×10^{-6}	1.3×10^{-4}	9.0×10^{-6}	1.1×10^{-3}
^{187}W	--	1.5×10^{-4}	1.3×10^{-4}	1.4×10^{-6}	--
Total	9.67	1.4×10^{-1}	1.7×10^{-1}	1.7×10^{-3}	1.2×10^{-1}

Table 11.4-4

Expected Annual Production of Solids

	ft^3/year	Normal Activity Ci/Container[a]	Maximum Activity Ci/Container[a]
Cleanup filter demineralizer sludge	720	1196	1820
Condensate filter demineralizer sludge	7460	19	40
Waste floor drain and fuel pool filter demineralizer sludge	1000	0.2	3.4
Waste demineralizer resin	540	29	330
Total volume (ft^3)	9720		

[a] Based on 164 ft^3 of dewatered resin per container



NOTES

1. PIPING, VALVES, AND ASSOCIATED COMPONENTS ON THIS DRAWING SHALL BE CLASSIFIED AS FOLLOWS:

   a) ALL PIPING, VALVES, AND ASSOCIATED EQUIPMENT IN THE FDR, EDR, RWCU, CPR, MWR, PWR, PSR, AND PVR SYSTEMS, UNLESS SPECIFICALLY NOTED OTHERWISE:

QUALITY CLASS	II+
SEISMIC CATEGORY	II
CODE GROUP	D+

   \+ SEE WNP-2 SPECIFICATION, SECTION 15B.1 TABLE 2 NOTE 12.

   b) ALL PIPING, VALVES, AND ASSOCIATED EQUIPMENT IN THE SWA AND COND SYSTEMS, AND WHERE SPECIFICALLY NOTED:

QUALITY CLASS	II
SEISMIC CATEGORY	II
CODE GROUP	D

   c) ALL INSTRUMENTATION PIPING AND TUBING DOWNSTREAM OF THE INSTRUMENT ROOT VALVE:

QUALITY CLASS	II
SEISMIC CATEGORY	II
CODE GROUP	D

2. ALL INSTRUMENT ROOT VALVES NOT LABELED WILL BE 3/4" GLOBE VALVES UNLESS SPECIFICALLY NOTED OTHERWISE.
3. THESE VALVES ARE ELECTRICALLY AND PNEUMATICALLY DISABLED AND CLOSED.
4. THESE VALVES ARE ELECTRICALLY AND PNEUMATICALLY DISABLED AND OPEN.
5. SEE DWG M501 FOR FLOW DIAGRAM LEGENDS, SYMBOLS AND ABBREVIATIONS.

LEGEND

1. ALL VALVES SUFFIXED WITH A (V) DENOTE A 3/4" VENT VALVE.
2. ALL VALVES SUFFIXED WITH A (D) DENOTE A 3/4" DRAIN VALVE.
3. ALL VALVES SUFFIXED WITH A (TH) DENOTE A THROTTLED VALVE.

FSAR FIG.

Columbia Generating Station
Final Safety Analysis Report

Flow Diagram Radioactive Waste Disposal Solids Handling System

Draw. No. M536	Rev. 32	Figure 11.4-1

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The process and effluent radiological monitoring and sampling systems are provided to allow determination of the content of radioactive material in various gaseous and liquid process and effluent streams. The design objective and criteria are primarily determined by the system designation of either

a. Instrumentation systems required for safety, or
b. Instrumentation systems required for plant operation.

11.5.1 DESIGN BASIS

11.5.1.1 Design Objectives

The process and effluent radiological monitoring and sampling system is designed to provide for compliance with the requirements of 10 CFR Part 50 including the General Design Criteria (GDC) of Appendix A and provides the monitoring and sampling required to make measurements, evaluations, and reports recommended by Regulatory Guide 1.21, Revision l.

11.5.1.1.1 Systems Required for Safety

The main objective of radiation monitoring systems (RMS) required for safety is to initiate appropriate protective action to limit the release of radioactive materials from the reactor vessel and reactor building if predetermined radiation levels are exceeded in major process/effluent streams and to limit inflow of airborne radioactivity to the control room following an accidental release. Additional objectives are to have these systems available under all operating conditions, including accidents and postaccidents, and to provide control room personnel with an indication of the radiation levels in the major process/effluent streams plus alarm annunciation if high radiation levels are detected.

The RMS provided to meet these objectives are

a. Main steam line RMS,
b. Reactor building ventilation exhaust plenum RMS, and
c. Standby service water RMS.

11.5.1.1.2 Systems Required for Plant Operation

The main objective of the RMS is to provide operating personnel with measurement of the content of radioactive materials in all effluent and important process streams. This allows demonstration of compliance with Technical Specifications by providing gross radiation level

monitoring and collection of halogens and particulates on cartridges and filters (gaseous effluents). Additional objectives are to initiate discharge valve isolation on the offgas, liquid radwaste, or drain systems if predetermined release rates are exceeded and to provide for sampling at certain radiation monitor locations to allow determination of specific radionuclide content.

The RMS provided to meet these objectives are

a. For gaseous process streams

   1. Offgas pretreatment RMS,
   2. Offgas post-treatment RMS,
   3. Charcoal bed vault RMS, and
   4. Mechanical vacuum pump exhaust RMS.

b. For gaseous effluent streams

   1. Reactor building elevated release duct RMS,
   2. Turbine generator building ventilation release duct RMS, and
   3. Radwaste building ventilation release duct RMS.

c. For liquid process streams

   1. Standby service water RMS and
   2. Reactor building closed cooling water RMS.

d. For liquid effluent streams

   1. Radwaste effluent RMS,
   2. Circulating water (blowdown) RMS,
   3. Standby service water RMS, and
   4. Plant service water (TSW) RMS.

e. For primary containment monitoring

   1. Leak detection RMS and
   2. Loss-of-coolant accident (LOCA) tracking RMS.

11.5.1.2 Design Criteria

11.5.1.2.1 Systems Required for Safety

The design criteria for the RMS required for safety are that the systems shall:

a. Withstand the effect of natural phenomena (e.g., earthquakes) without loss of capability to perform their functions,

b. Perform its intended safety function in the environment resulting from normal and postulated accident conditions,

c. Meet the reliability, testability, independence, and failure mode requirements of engineered safety features,

d. Provide continuous output on control room panels,

e. Permit checking of the operational availability of each channel during reactor operation with provision for calibration function and instrument checks,

f. Ensure an extremely high probability of accomplishing its safety function in the event of anticipated operational occurrences,

g. Initiate prompt protective action prior to exceeding Technical Specifications limits,

h. Provide warning of increasing radiation levels indicative of abnormal conditions by alarm annunciation,

i. Insofar as practical, provide self-monitoring of components to the extent that power failure or component malfunction causes annunciation and channel trip,

j. Register full scale output if radiation detection exceeds full scale, and

k. Have sensitivities and ranges compatible with anticipated radiation levels.

The applicable GDC are 1, 2, 3, 4, 13, 19, 20, 21, 22, 23, 24, 29, 60, and 64 (see Section 3.1). The systems shall meet the design requirements for Safety Class 2, Seismic Category I systems, along with the quality assurance requirements of 10 CFR Part 50, Appendix B.

11.5.1.2.2 Systems Required for Plant Operation

The design criteria for operational RMS are that the systems shall

a. Provide continuous indication of radiation levels in the main control room,

b. Provide warning of increasing radiation levels indicative of abnormal conditions by alarm annunciation,

c. Insofar as practical, provide self-monitoring of components to the extent that power failure or component malfunction causes annunciation and, for systems initiating protective action, channel trip,

d. Monitor a sample representative of the bulk stream or volume,

e. Have provisions for calibration, function and instrumentation checks,

f. Have sensitivities and ranges compatible with anticipated radiation levels and Technical Specifications limits, and

g. Register full scale output if radiation detection exceeds full scale.

The RMS monitors discharges from the gaseous and liquid radwaste treatment systems and nonradioactive sumps have provisions to alarm and to initiate automatic closure of the effluent discharge valves on the affected treatment systems prior to exceeding the normal operation limits specified in the Technical Specifications. Additionally, the primary containment monitoring system meets the criteria, except for item g.

The applicable GDC are 13, 60, and 64 (see Section 3.1).

11.5.2 SYSTEM DESCRIPTION

11.5.2.1 Systems Required for Safety

Information on these systems is presented in Tables 11.5-1 and 11.5-2 and the arrangements shown in Figures 11.5-1 through 11.5-4. The equipment is designed to Quality Class I and Seismic Category I requirements. High reliability is further achieved by the use of redundancy as noted below.

11.5.2.1.1 Main Steam Line Radiation Monitoring System

This system monitors the gamma radiation level exterior to the main steam lines. The normal radiation level is produced primarily by coolant activation gases plus smaller quantities of

fission gases being transported with the steam. In the event of a gross release of fission products from the core, this monitoring system provides signals to the following:

a. Reactor water sample valves,
b. Mechanical vacuum pumps,
c. Mechanical vacuum pumps isolation valves,
d. Gland seal exhausters, and
e. Control room annunciators.

The system consists of four redundant instrument channels. Each channel consists of a local detector (gamma-sensitive ion chamber) and a control room radiation recorder and a readout module with an auxiliary trip unit. Power for two channels (A and C) is supplied from the reactor protection system (RPS) bus A and for the other two channels (B and D) from RPS bus B. Channels A and C are physically and electrically independent of channels B and D.

The detectors are located near the main steam lines in the steam tunnel as it enters the turbine building. The detectors are geometrically arranged so that this system is capable of detecting significant increases in radiation level with any number of main steam lines in operation.

Each radiation monitor has four trip circuits: two upscale (high-high and high), one downscale (low), and one inoperative. Each trip is visually displayed on the affected radiation readout module. A high-high or inoperative trip results in a channel trip in the auxiliary unit which is an input to the reactor water sample valves, mechanical vacuum pump shutdown, and discharge valve closure. A high trip actuates a main steam line (MSL) high control room annunciator common to all channels. A downscale and inoperative trip actuates a MSL downscale/inoperative control room annunciator common to all channels. High and low trips do not result in a channel trip. Each radiation monitor displays the measured radiation level.

Arrangement details are shown in Figure 11.5-1.

11.5.2.1.2 Reactor Building Exhaust Plenum Radiation Monitoring System

This system monitors the radiation level of the reactor building ventilation system exhaust plenum prior to its discharge from the building into the elevated release duct. A high radioactivity level in the exhaust system could be due to fission gases from a leak or an accident.

The system consists of four redundant instrument channels. Each channel consists of a local detection assembly (a sensor and converter unit containing a Geiger-Mueller (GM) tube and electronics) and a control room radiation readout module. The 120-V ac power for channels (A and B) is provided from Division 1, and for channels (C and D) from Division 2 power panels; the multipoint strip chart recorder supplied from the Division 2 uninterruptible power supply (UPS) power panel records the output of all four channels. The detection assemblies

are located outside the exhaust air plenum upstream of the secondary containment discharge isolation valves. The distance upstream from the inboard discharge isolation valve is such that, at the maximum design flow, the transport time from the detector location to the inboard discharge valve is greater than the total time required to respond to trip level radiation and close the inboard discharge valve before exceeding 10 CFR 50.67 dose limits (see Section 9.4.2.3).

Each radiation monitor has two trip circuits: one upscale (high-high) and one downscale/inoperative (fail safe design). Two-out-of-two upscale/downscale trips (channels A and B) initiates closure of the reactor building ventilation outboard isolation valves and the primary containment outboard purge and vent valves, and initiates startup of standby gas treatment (SGT) system train B. The same condition for channels C and D initiates closure of the corresponding inboard valves and initiates startup of SGT train A.

An upscale trip is displayed on the affected radiation readout module and actuates a reactor building vent high-high radiation control room annunciator common to all channels.
A downscale trip is also displayed on the radiation readout module and actuates a reactor building vent downscale control room annunciator common to all channels.

An additional trip signal for high radiation alarm is provided by the recorder and actuates a reactor building vent high radiation control room annunciator. Each radiation monitor displays the measured radiation level.

Arrangement details are shown in Figure 11.5-2.

11.5.2.1.3 Deleted

11.5.2.1.4 Standby Service Water Radiation Monitoring System

This system monitors gamma radiation levels of the service water liquid process and effluent streams.

Each monitor system consists of a gamma scintillation detector inserted into an offline chamber to which a process stream sample is piped. The detector locations are selected to obtain a reasonable geometry and are positioned away from crud trap and associated high background regions. Lead shielding is provided to further reduce background levels.

At each liquid offline detector location, a continuous sample from the liquid process pipe passes through a shielded detection assembly for gross radiation monitoring and then is returned to the process pipe. The detection assembly consists of a detector mounted in a shielded sample chamber. The local radiation monitor and the meter in the control room display the measured gross radiation level. The sample chamber and lines can be drained to allow assessment of background buildup. The flow meter at the sample rack provides local sample line flow indication.

Sample flow for each detector is from the standby service piping downstream of each of the two residual heat removal (RHR) heat exchanger (loops A and B). These monitors are designed to detect any primary coolant leakage into the standby service water through the RHR heat exchanger, during operation of the RHR heat exchangers in the shutdown heat removal mode.

Additional details are shown in Figure 11.5-4.

11.5.2.2 Systems Required for Plant Operation

All systems associated with the plant process cycle provide for indication and recording of radiation levels in the main control room in conjunction with alarm annunciation features.

Information on these systems is presented in Tables 11.5-1 and 11.5-2 and the arrangements are shown in Figures 11.5-2 and 11.5-4 through 11.5-10.

11.5.2.2.1 Gaseous Process and Effluent Radiation Monitoring System

11.5.2.2.1.1 Offgas Pretreatment Radiation Monitoring System. This system monitors radioactivity at the outlet of the water separator downstream of the catalytic recombiners. The monitor detects the radiation level which is attributable to the fission gases produced in the reactor and transported with steam through the turbine to the condenser.

A continuous sample is extracted from the offgas pipe via a stainless steel sample line that is then passed through a sample chamber and a sample panel before being returned to the suction side of the steam jet air ejector (SJAE). The sample chamber is a steel pipe which is internally polished to minimize plateout. It can be purged with room air to check detector response by using a three-way solenoid-operated valve. The valve is controlled by a switch located in the main control room. The sample panel measures and indicates sample line flow.

The detector is a gamma-sensitive ionization chamber mounted external to the sample chamber. The channel has a logarithmic radiation readout module which provides a system alarm output and is provided with a recorder.

The 120-V ac UPS power for count ratemeter and trip auxiliary circuit is supplied from critical Division 2 panel; strip chart recorder receives reliable (Division B) instrument power, and local 120-V ac for the offgas sample and vial sampler control panel.

The radiation readout module has four trip circuits: two upscale (high-high and high), one downscale (low), and one inoperative. The trip outputs are used for alarm function only. Each trip is displayed on the radiation monitor and actuates a control room annunciator: offgas high-high, offgas high, and offgas downscale/inoperative. Sample line flow is displayed at the sample panel.

The radiation level output by the monitor can be directly correlated to the concentration of the noble gases by using the semiautomatic vial sampler panel to obtain a grab sample. To draw a sample, a serum bottle is inserted into a sample chamber, the sample lines are evacuated, and a solenoid-operated sample valve is opened to allow offgas to enter the bottle. After the sampling bottle is removed, the sample is analyzed in the counting room with a multichannel gamma analyzer to determine the concentration of the various noble gas radionuclides.
A correlation between the observed activity and the monitor reading permits calibration of the monitor.

For arrangement details see Figure 11.5-5.

11.5.2.2.1.2 Offgas Posttreatment Radiation Monitoring System. This system monitors radioactivity in the offgas piping downstream of the offgas system charcoal vessels and upstream of the offgas system discharge valve. A continuous sample is extracted from the

offgas system piping, passed through the offgas posttreatment sample panels for monitoring and sampling, and returned to the offgas system piping. Each sample panel has a cartridge and filter (one for particulate collection and one for halogen collection) in series with filter bypass capabilities (with respect to flow) with two identical continuous gross radiation detection assemblies. Each gross radiation detection assembly consists of a shielded chamber, a radiation detector, and a check source. Two radiation monitors in the main control room display the measured gross radiation level.

The sample panels and shielded chambers can be purged with room air to check detector response by using solenoid valves that can be operated from the control room or at the sample panels. The sample panels measure and indicate sample line flow. A solenoid operated check source for each detection assembly operated from the control room or locally can be used to check operability of the gross radiation channel.

Channel A monitor receives reliable (Division A) 120-V ac power and channel B is supplied with Division 2 power; the ±24-V dc auxiliary trip circuits are supplied internally from the monitor power supply. Control room chart recorder receives reliable (Division B) instrument power. Offgas posttreatment sample panel 11A is provided 120-V ac (Division A) power and offgas posttreatment sample panel 11B is provided 120-V ac (Division B) power.

Each radiation readout module has four trip circuits: two upscale (high-high-high, and high), one downscale (low), and one inoperative. Each trip is visually displayed on the radiation monitor. The first three trips actuate corresponding control room annunciators: offgas posttreatment high-high-high radiation, offgas posttreatment high radiation, and offgas posttreatment downscale. A high-high trip from the recorder actuates an offgas posttreatment high radiation annunciator in the main control room. High or low sample flow measured at the sample panel actuates a control room offgas posttreatment trouble annunciator.

A trip auxiliary unit in the control room takes the high-high-high and downscale trip outputs and, if its logic is satisfied, initiates closure of the offgas system discharge and drain valves. The logic is satisfied if two high-high-high, one high-high-high and one downscale, or two downscale trips occur. The high-high-high trip setpoints are determined such that valve closure is initiated prior to exceeding Technical Specifications limits. Any one high upscale trip initiates closure of the offgas system bypass line valve and initiates opening of the charcoal adsorber treatment line valve.

Grab sampling functions required for isotopic analysis and gross monitoring calibrations can be performed at each redundant sample panel.

For arrangement details see Figure 11.5-5.

11.5.2.2.1.3 Offgas Charcoal Bed Vault Radiation Monitoring System. The charcoal bed vault air handling room is monitored for an increase in the gross gamma radiation level from

leakage of radioactive noble gases out of the treatment system. The channel includes a sensor, indicator, trip unit, and a locally mounted auxiliary unit. The detector is mounted outside of the air handling room door. The insulated door only attenuates 80 keV gamma (^{133}Xe) by approximately 40%. An indicator and trip unit is located in the main control room. The channel provides for sensing and readout, both local and remote, of gamma radiation over a range of six logarithmic decades (10^{+0} to 10^{+6} mR/hr).

The indicator and trip unit has one adjustable upscale trip circuit for alarm and one downscale trip circuit for instrument inoperative which annunciates in the main control room. The trip circuits are capable of convenient operational verification by means of test signals or through the use of portable gamma sources. Power is supplied from the channel A power supply of the reactor handling ventilation exhaust plenum RMS.

For arrangement details see Figure 11.5-2.

11.5.2.2.1.4 Mechanical Vacuum Pump Exhaust Radiation Monitoring System. The radiation monitor on the mechanical vacuum pump exhaust is designed to alarm, stop, and isolate the mechanical vacuum pumps in the case of high level of radioactive gases in air being exhausted to the reactor building elevated release duct. The mechanical vacuum pump is operated during plant startups to remove bulk air from the condenser and is secured at the point where the steam jet air ejection suction is available and condenser offgases are routed through the recombiner charcoal process treatment system. In addition to monitoring discharges via the mechanical vacuum pumps, the turbine gland seal air exhauster system is continuously monitored via this process radiation monitoring system. Clean sealing steam is used on the turbine gland seals to maintain the releases of radionuclides to as low as is reasonably achievable (ALARA) limits. The monitor complies with GDC 64 and is Quality Class II and Seismic Category II.

The channel includes an energy-compensated GM detector, a readout module, and recorder in the main control room. The channel provides for sensing and readout, both local and remote, of gamma radiation over a range of four logarithmic decades (10^{-2} to 10^{2} mR/hr).

The indicator and trip unit has two adjustable trip circuits. The upscale trip circuit generates a high radiation alarm and stops the mechanical vacuum pumps. The adjustable downscale trip circuit generates an instrument alarm.

For arrangement details see Figure 11.5-6.

11.5.2.2.1.5 Reactor Building Elevated Release Duct Radiation Monitoring System. This monitoring system measures radioactivity in the reactor building elevated release duct from the gland seal and mechanical vacuum pumps, the treated offgas effluent, the SGTS exhaust, and the exhaust air from the reactor building ventilation. This system consists of two main subsystems, an offline sampling system and an inline gamma radiation monitoring system.

The sampling system is used to comply with Regulatory Guide 1.21, Revision 1, and as such satisfies GDC 64. A continuous representative sample is extracted from the elevated release duct through an isokinetic probe, passes down a 50-ft vertical tube, through a filter to collect particulates, and through an impregnated charcoal cartridge to collect iodine. The sample travels through a flow indicator and then a sample pump prior to being returned downstream to the sampling point. Samples are analyzed at least weekly to determine the quantities of the specific radionuclides released. The sample flow rate is indicated and totalized locally and the exhaust duct flow is also indicated locally. Both of these variables are recorded and alarmed in the control room and are input to the transient data acquisition system (TDAS). Arrangement details are shown in Figure 11.5-6. Table 11.5-3 lists sampling frequencies and required sensitivities.

The gamma radiation monitoring system uses three separate detectors. The low range detector is used to satisfy Regulatory Guide 1.21, Revision 1, and as such, complies with GDC 64. The intermediate and high range detectors, along with the LOCA detectors of Section 11.5.2.2.3.2, are used to satisfy NUREG-0737 and Regulatory Guide 1.97, Revision 2, requirements and, as such, comply with GDC 13 and 64. This system is an EG&G ORTEC Gamma Spectroscopy system. It uses inline detectors to provide an isotopic analysis of the reactor building elevated release duct effluents. Two cryogenically cooled, high purity germanium coaxial detectors provide a range of 1.8×10^{-5} to 7.8×10^{4} $\mu Ci/cm^3$ with overlap.

a. Intermediate range — Approximately 1.8×10^{-5} to 1.8×10^{0} $\mu Ci/cm^3$ (^{133}Xe)

b. High range — Approximately 7.8×10^{-1} to 7.8×10^{4} $\mu Ci/cm^3$ (^{133}Xe)

These detectors provide approximately 40% efficiency to a 1332.5 keV gamma [compared to a 3 in. x 3 in. NaI (T1) detector at 25 cm]. Both detectors monitor activity through the release duct and are mounted in the reactor building at elevation 618 ft 7 in. Lead enclosures provide shielding from postaccident background radiation. Collimator design and detector locations ensure representative sampling.

A third high efficiency detector is located inside the release duct to monitor low level normal operation activity. The usable range for this detector is approximately 2.71×10^{-8} to 2.71×10^{-3} $\mu Ci/cm^3$. The high efficiency detector provides approximately120% efficiency to 1332.5 keV gamma [compared to a 3 in. x 3 in. NaI (T1) detector at 25 cm].

The postaccident system consists of a computer controlled acquisition and analysis loop, located in radwaste building el. 525 ft, feeding an additional computer located in the control room. Either computer can control detector, signal processing, and spectral analysis functions. Commercial application software from ORTEC is used for both the control of the detection process and nuclide identification. The libraries which are used for nuclide identification are

based on Tables 11.3-8 and 15.6-10 and 15.6-11. The radwaste computer is used for system status monitoring and data output. System status is monitored and alarmed.

This system provides two types of information and each has its different response time. Gross gamma, in counts/sec, originates at the local acquisition and analysis panel from a log count-rate meter fed directly from the spectroscopic amplifier. This real time signal is then input directly to the trending recorder and TDAS system. The recorder updates its digital display and alarm information every 6 sec. This conforms to the requirement in Regulatory Guide 1.97 to monitor noble gas as a Type C and E, Category 2 variable.

Effluent isotopic information availability is controlled by the detector counting times, which are a function of stack activity. Counting times are increased during periods of low stack activity and will decrease for periods of high activity. System responses will be the sum of the detector counting and data transmission times. Accurate response times for both sets of data will be determined via field testing. This conforms to the requirement in Regulatory Guide 1.97 to monitor particulate and halogen release as a Type E, Category 3 variable.

This system will satisfy the requirements of Regulatory Guide 1.97, Revision 2, and NUREG-0737 for normal and postaccident monitoring of noble gases, particulates, and halogens. All equipment is qualified to operate in the required postaccident environment and is supplied by reliable battery-backed power. It is designed to meet the pertinent sections of ANSI N42.18-1980 (formerly ANSI N13.10), “Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents.” Additionally, guidelines from ANSI N42.14-1991, “Calibration and Use of Germanium Spectrometers for the Measurement of Gamma-Ray Emission Rates of Radionuclides,” are used to monitor system performance as part of the surveillance and calibration process. In situ calibration uses National Institute of Standards and Technology (NIST) traceable standards. Additionally, transfer calibrations may be performed on gas samples drawn from the reactor stack and analyzed on NIST referenced equipment when sufficient stack gas activity is present.

Table 11.5-1 lists the sensitivity and range of each detector. Arrangement details are shown in Figure 11.5-11.

These detectors provide continuous monitoring with overlap from normal plant operation (typically in the mid 10^{-8} decade with no failed fuel) to a worst case DBA LOCA with an expected containment activity of approximately 2.92 x 10^4 $\mu Ci/cm^3$.

The system has built-in electronic test circuits. Calibration curves have been developed from the calibration data equating activity observed on the monitor to the $\mu Ci/cm^3$ concentration in the effluent. Initial system calibration parameters were established by comparisons between analysis done via the inline system and an NIST traceable gamma spectroscopy system. Calibration procedures use transfer and linearity standards. Electrical power for this

monitoring system is from a reliable power supply. Also see Figure 11.5-11, and Table 11.5-1.

11.5.2.2.1.6 Turbine Generator Building Ventilation Release Duct Radiation Monitoring System. This monitoring system measures the radioactivity in the turbine building exhaust prior to its discharge to the environment and in doing so complies with Regulatory Guide 1.21, Revision 1, GDC 64, and NUREG-0737. This monitor detects the fission and activation products from the steam which may leak from the turbine or the other primary components in the building. The gaseous activity in the exhaust is expected to normally be below detectable levels. The particulate and iodine activity is typically accumulated on a filter and cartridge respectively for a week to obtain sufficient activity to be detectable. These filters are analyzed to determine the quantities of specific radionuclides present and the results, together with the gaseous activity strip chart recorder, provide a permanent record of radioactivity released to the environment.

A continuous representative sample is extracted from the exhaust vent through a multi-ported isokinetic probe, down a 30-ft tube to pass through a filter paper to collect particulates, and through an impregnated charcoal cartridge to collect iodine. The sample travels through two gas detectors, a mass flow transducer, a local flow meter with switch, and then a sample pump prior to being exhausted to the radwaste building roof area by the turbine building exhaust fans. The sample flow rate is automatically adjusted to compensate for effluent flow changes.

The gas monitoring system includes two local detectors mounted in one lead-shielded chamber, a local ratemeter with display, a remote control/display unit, and a recorder in the main control room. The normal sample flows through the high range detector followed by the low range detector. Arrangement details are shown on Figure 11.5-7.

The monitoring system includes four channels: channel 1, low-range (10^{-7} to 10^{-1} μCi/cc), channel 2, high-range (10^{-3} to 10^{+3} μCi/cc), channel 3, calculated composite range (10^{-7} to 10^{+3} μCi/cc), and channel 4, gas effluent release rate range (10^{+1} to 10^{+11} μCi/sec). A two-decade overlap is provided between the low and high range channels. The low and high range channels are equipped with a radioactive test source.

The gas monitoring system has no control functions. The monitor provides an alert radiation alarm and an instrument inoperative alarm, both of which annunciate in the main control room. The reliable power for low/high range radiation monitor(s) and recorder(s) is provided from Division A power panel and backed by standby power.

11.5.2.2.1.7 Radwaste Building Ventilation Release Ducts Radiation Monitoring System. This monitor system measures the radioactivity in the radwaste building ventilation air exhaust as it is being discharged to the environment and in doing so complies with Regulatory Guide 1.21, Revision 1, GDC 64, and NUREG-0737. Radioactivity originates from radwaste tank vents, from primary water processing equipment, and from laboratory sampling hoods, as well as various cubicles having liquid process treatment systems within the building. A continuous sample is drawn from each of the two out of three exhaust fans that are operating. The sampling uses isokinetic probes with a fixed flow rate and samples are isokinetic over the normal ventilation operating range. The representative sample is withdrawn through a multi-ported duct probe, through a tube passing through a particulate filter, and through a charcoal cartridge to collect particulate and iodine samples, which are removed at least weekly for laboratory radiochemical analyses. The filtered air sample streams from the operable exhaust fans are combined to pass through two gas detectors, a mass flow transducer, a local flow meter with switch and a sample pump prior to being exhausted to the room.

The gas monitoring system includes two local detectors mounted in one leaded-shielded chamber, a local ratemeter with display, a remote control/display unit, and a recorder in the main control room. The normal sample flows through the high range detector followed by the low range detector. Arrangement details are shown in Figure 11.5-7.

The monitoring system includes four channels: channel 1, low-range (10^{-6} to 10^{-1} μCi/cc), channel 2 high-range (10^{-3} to 10^{+3} μCi/cc), channel 3, calculated composite range (10^{-6} to 10^{+3} μCi/cc), and channel 4, gas effluent release rate range (10^{+1} to 10^{+10} μCi/sec). A two-decade overlap is provided between the low and high range channels.

The gas monitoring system has no control functions. The monitor provides an alert radiation alarm and an instrument inoperative alarm, both of which annunciate in the main control room. The reliable power for low/high range radiation monitor(s) and recorder(s) is provided from Division A, power panel and backed by standby power.

11.5.2.2.1.8 NRC Safety Evaluation Report, NUREG-0892 Acceptance. System comparisons to ANSI 13.10 (redesignated as ANSI N42.18) have been performed.

An inline gamma spectroscopy system provides an isotopic analysis of all effluents exiting the reactor building elevated release duct. See Section 11.5.2.2.1.5, Figure 11.5-11, and Table 11.5-1.

Particulate/iodine sampling of the other buildings (radwaste and turbine) exhausts will be handled by the normal effluent samplers where the postaccident release concentration is quite low.

If there were a reactor accident with a core fission product release, the reactor building (secondary containment) immediately isolates. The atmosphere is maintained at a 0.25 in. H_2O vacuum by the SGT system. The only potential airborne contamination that could reach the other buildings is from the SGT system bypass leakage pathways as listed in Table 6.2-16. Shielding needed for the radwaste and turbine building normal effluent sampling systems was evaluated using assumptions consistent with the discussion in Appendix J (see Sections J.2 and J.3). *For the purpose of defining the shielding requirements for the radwaste and turbine building, an iodine concentration of 3.7 x* 10^{+4} *µCi/cm*^{3} *is assumed in the inleakage air. This iodine concentration is based on a 50% core inventory release to the drywell atmosphere and a 50% plate-out factor. This plate-out factor is conservative for determining sample shielding requirements. With 0.35 scfh bypass leakage into the radwaste building, the building exhaust (83,000 cfm) concentration will be 2.6 x* 10^{-3} *µCi/cm*^{3}*, ignoring building volume dilution. The normal effluent sampler operates at 3 cfm; therefore, the charcoal cartridge 30 minutes accumulation would be 6.7 mCi. This would result in a dose rate of 21 mR/hr at 1 ft from the cartridge. Doubling the dose rate to account for particulates yields 42 mR/hr at 1 ft from the sample assembly. Because of the high exhaust flow rate (260,000 cfm) and less inleakage (0.24 scfh), the turbine building exhaust is less concentrated.* The radwaste and turbine building normal effluent sampling systems are adequate for postaccident sampling and no shielding is necessary (see Section 11.5.2.2.1.5). The evaluation for shielding was not repeated for the implementation of the alternative source term (AST).

11.5.2.2.2 Liquid Process and Effluent Radiation Monitoring System

These systems monitor gamma radiation levels of liquid process and effluent streams.

Each monitor system consists of a gamma scintillation detector inserted into either a well in the process piping or a sump or an offline chamber to which a process stream sample is piped. The detector locations are selected to obtain a reasonable geometry and are positioned away from crud traps and associated high background regions. Lead shielding is provided to further reduce background levels.

At each liquid offline detector location a continuous sample is extracted from the liquid process pipe, passed through a shielded detection assembly for gross radiation monitoring, and then returned to the process pipe. The detection assembly consists of a detector mounted in a shielded sample chamber equipped with a check source. The meter and recorder in the control room displays the measured gross radiation level. The sample chamber and lines can be drained to allow assessment of background buildup. The flow meter at the sample rack provides local sample line flow indication. A solenoid-operated check source operated from the control room is used to check operability of the channel response.

The critical ±24-V dc for monitor(s) is supplied from Division 1 (and 2) power panels, except for the plant service water monitor which receives 120-V ac from Division A panel, reliable

power is provided for control room strip chart recorders, sample control panel receives local 120-V ac.

The detector's local preamplifier unit is designed to remain fully operational in the expected environment. If exposed to radiation transients which exceed the channel range, the channel maintains full scale deflection and returns to normal functioning when the transient has subsided.

Each radiation monitor, except for the circulating water, has four trip circuits: two upscale (high-high and high), one downscale (low), and one inoperative. Each trip is visually displayed on the affected radiation monitor. Two of these trips actuate corresponding control room annunciators: one upscale (high radiation) and the downscale for the affected liquid monitoring channel. High or low sample flow measured at the sample panel actuates a control room high-low flow annunciator for the affected liquid channel.

All alarms are annunciated in a control room. Liquid monitoring system details are given in Table 11.5-2 and the monitor arrangements are shown in Figure 11.5-4.

11.5.2.2.2.1 Standby Service Water Radiation Monitoring System. See Section 11.5.2.1.4.

11.5.2.2.2.2 Reactor Building Closed Cooling Water Radiation Monitoring System. The radiation detector is located offline and samples the closed cooling water piping side of the reactor closed cooling water system (RCC) heat exchangers.

The monitor system is a diagnostic tool to verify that no inleakage of primary plant water has occurred from the reactor water cleanup system nonregenerative heat exchanger system which uses RCC as coolant. Since the RCC is a closed system, inleakage would be detected by the monitor system.

11.5.2.2.2.3 Radwaste Effluent Radiation Monitoring System. This monitor system measures the radioactivity in the radwaste effluent discharge prior to its entering the cooling tower blowdown line.

Liquid waste can be discharged from several radwaste processed water tanks such as the floor drain sample tank, waste sample tanks, or distillate drain tanks. These tanks contain water that has been processed through one or more treatment systems such as evaporation, filtration, and ion exchange. Prior to the discharge from any tank, the liquid in the appropriate tank is sampled and analyzed in the laboratory for radioactivity. Based on this analysis, discharge is permitted as specified in the Offsite Dose Calculation Manual (ODCM).

The radiation detector is located offline and samples the common discharge line from the liquid radwaste system through which all liquid radwaste is discharged to the blowdown line. The

piping arrangement is designed so that the sample well can be flushed to lower background levels.

The high-high upscale trip on the radwaste effluent radiation monitor is used to initiate closure of the radwaste system discharge valve. The trip point is set such that closure is initiated prior to exceeding ODCM limits for liquid effluents. The high upscale trip actuates an annunciator in the radwaste control room as well as the main control room.

11.5.2.2.2.4 Circulating Water and Plant Service Water Radiation Monitoring Systems. The circulating water monitor is located on the discharge side of the circulating water pumps for the main condenser in the coolant blowdown line to the Columbia River. The location of this monitor permits detection of radioactive material leaking to the circulating water from any source, including the TSW system. If an alarm condition exists, circulating water blowdown to the Columbia River is terminated by automatic closure of the circulating water blowdown valve in the circulating water pump house (see Section 9.2.1.2), and it annunciates in the control room.

During plant outages, water is discharged to the river via temporary pumps installed in the circulating water pump house and connected to the blowdown line. Automatic termination of blowdown is not provided for this temporary arrangement. The only possible source of radiation would be through the TSW system which is monitored for radiation. There is about a 2-hr holdup time in the circulating water pipe giving ample time for blowdown to be manually secured if an alarm is received.

The TSW radiation monitor is located in the TSW return header to the circulating water system. The monitor is an offline type that continuously measures the radioactivity level of the TSW returning flow to the circulating water system. The radiation detector is lead shielded to minimize background radiation effects. The signal from the detector is displayed on a readout module and recorded in the main control room.

11.5.2.2.3 Primary Containment Radiation Monitoring System

This monitoring system is composed of two parts, a sensitive two-channel leak detection system and a four-channel high activity LOCA tracking system.

11.5.2.2.3.1 Leak Detection Monitors. This monitoring system measures the radioactivity in the drywell and in doing so complies with Regulatory Guide 1.45, Revision 0, and GDC 30. The radioactivity in the drywell is from coolant and corrosion activation products plus fission products produced in the reactor and released through leaks.

This monitoring system has two redundant subsystems, each having two detectors, individually monitoring particulates, and noble gas activity. Additionally a charcoal sample cartridge is provided to trap halogen gases. The detectors are housed in divisionally separated sample

racks located in a reactor building sample room. The sample racks have incorporated blowers and flow controls to withdraw gas samples from the primary containment atmosphere via stainless steel sampling lines and vent back to the containment. The environment in which the local cabinets are located is maintained to limit upper temperature excursions that may occur in the reactor building during an accident. Associated radiation readout modules and recorders are mounted in the main control room along with alarm annunciators.

Required 120-V ac supply for Division 1 and 2 equipment in both the main control room and the reactor building sample room is provided on a divisional basis by the 120/240-V ac critical (Class 1E) instrumentation power system.

The two-channel detector assemblies are provided with lead shielding to minimize the effects of background radiation to ensure high sensitivity. The detectors are of the beta scintillation type and are provided with check sources to verify system operability. The particulate detector views a fixed filter collector on which airborne particulates are trapped. The noble gas detector views a fixed volume of gas.

Each radiation monitor has three trip circuits: one upscale Hi to alarm and close sample line valve, one upscale Alert, and one downscale for instrument inoperative. All alarms annunciate in the main control room. This monitoring subsystem provides no control function and is a diagnostic tool which enables the main control room operator to take appropriate action.

Arrangement details are shown in Figure 11.5-9.

11.5.2.2.3.2 Loss-of-Coolant Accident Tracking Radiation Monitoring Systems (Containment Drywell). The LOCA monitoring systems, CMS-RE-27E and CMS-RE-27F, monitor the drywell atmosphere from inside the drywell; while CMS-RE-27A and CMS-RE-27B monitor the drywell atmosphere through penetrations in the bioshield wall. These four monitors provide LOCA monitoring for abnormal radioactivity levels following an accident condition involving rupture of the reactor coolant boundary.

The in-containment LOCA monitors CMS-RE-27E and CMS-RE-27F, located at approximately 517 ft level Azimuth 291° and 515 ft level Azimuth 51.5° respectively, track long-term decreases in containment radioactivity that take place with decay and decontamination and comply with GDC 13 and 64. The in-containment LOCA monitors each contain a Victoreen Model 877-1 ionization chamber with a range of 1 to 10^7 R/hr. These two detectors are qualified and meet the criteria of Table II.F.1-3 of NUREG 0737, including calibration with a high level gamma source. The chambers will respond to low energy gamma radiation such as 81 keV from ^{133}Xe fulfilling the Regulatory Guide 1.97, Revision 2 criteria.

The LOCA monitors, CMS-RE-27A and CMS-RE-27B, are housed in the bioshield wall and are against the containment steel shell. These monitors will monitor the drywell radiation in conjunction with the in-containment LOCA monitors CMS-RE-27E and CMS-RE-27F. The

LOCA monitors, while not required for plant operation, are used to monitor the drywell in the post-LOCA situation.

The LOCA monitoring systems (CMS-RE-27A, CMS-RE-27B, CMS-RE-27E, and CMS-RE-27F) are redundant and separately supplied with power from Division 1 and 2 in both the main control room and remote locations.

Each ionization chamber is wired to a local monitor. Output from each local monitor is wired to a monitor located in the main control room. Radiation levels within the primary containment for each LOCA monitoring system are recorded in the main control room.

Each monitor has alarm circuitry and indication for high radiation and for instrument inoperative that annunciate in the main control room. This monitoring subsystem provides no control function and is a diagnostic tool which enables the main control room operator to take appropriate action.

Arrangement details are shown in Figure 11.5-10.

11.5.2.3 Sampling

The following sections present a detailed description of the radiological sampling procedures, frequencies and objectives for all plant process and effluent sampling. This sample program provides the means to show compliance with the ODCM for the process radiation monitoring system and radwaste system.

11.5.2.3.1 Process Sampling

Section 9.3.2 describes the design of sampling facilities provided for general sampling. The sample frequency, type of analyses, analytical sensitivity, and the purpose of the sample are summarized in Table 11.5-4 for each liquid process sample location and in Table 11.5-5 for each gas process sample location. The analytical procedures used in sample analysis are presented in Section 11.5.2.3.3. These samples serve to monitor radioactivity levels within various plant systems.

11.5.2.3.2 Effluent Sampling

Effluent sampling of all potentially radioactive liquid and gaseous effluent paths is conducted on a regular basis to verify the adequacy of effluent processing to meet the discharge limits to unrestricted areas. This effluent sampling program will be of such a comprehensive nature as to provide the information for the effluent measuring and reporting programs required by 10 CFR Part 50.36a in annual reports to the NRC. The frequency of the periodic sampling and analysis described herein is normal and will be increased if effluent levels approach

Technical Specifications limits. Tables 11.5-3 and 11.5-6 summarize the sample and analysis schedules which correspond to Regulatory Guide 1.21, Revision 1, guidance.

11.5.2.3.3 Analytical Procedures

Samples of process and effluent gases and liquids will be analyzed in the laboratory by the following techniques: gross beta counting, gross alpha counting, gamma spectrometry, liquid scintillation counting, and radiochemical separations.

Instrumentation which is available in the laboratory for the measurement of radioactivity includes a single channel analyzer, alpha counter, beta counter, liquid scintillation counter, and multichannel gamma spectrometer.

Samples for beta counting are evaporated to dryness on metal planchets prior to counting. Sample volume, counting geometry, and counting time are chosen to achieve the required measurement sensitivities. Correction factors are applied for sample-detector geometry, self-absorption and counter-resolving time.

Gross beta and gross alpha analyses of liquid effluent samples may be performed with an internal proportional counter. The samples are prepared for counting by evaporation onto metal planchets. Gross beta counting is also performed using a liquid scintillation counter.

Sample volume and counting times are chosen to achieve the required measurement sensitivity. When possible, sample volume is selected to maintain a sample residue thickness of less than 1 mg/cm^2. Correction factors are applied for self-absorption.

Gamma ray spectrometry is used for isotopic analyses of gaseous, airborne particulate and iodine, and liquid samples. High-resolution germanium solid-state detectors are available for this purpose. The detectors are calibrated against NIST traceable gamma ray standards for a variety of sample detector geometries. The gamma spectra are resolved using a software program developed for computerized spectrum resolution.

Gaseous tritium samples are collected by condensations or adsorption, and the resultant liquid is analyzed by liquid scintillation counting techniques.

Radiochemical separations are used for the routine analysis of ^{89}Sr and ^{90}Sr.

11.5.2.3.4 Inservice Inspection, Calibration, and Maintenance

During reactor operation, checks of system operability are made by observing channel behavior. At monthly intervals during reactor operation, the response of each detector supplied with a remotely activated check source or LED source simulator will be recorded together with the instrument background count rate to ensure proper functioning of the

monitor. Any detector whose response is observed to be inconsistent with that expected for power operation and is not supplied with a remotely activated check source or LED source simulator will be checked with a portable source or by comparison of detector response to system activity. An exception to the portable source check requirement will be allowed for those detectors mounted in areas that are deemed hazardous to personnel.

The system has electronic testing and calibrating equipment, which permits channel testing without relocating or dismantling channel components. An internal trip test circuit, adjustable over the full range of the readout meter, is used for testing. Each channel is tested prior to performing a calibration check. Verification of valve operation, ventilation diversion, or other trip functions is done at this time if it can be done without jeopardizing the plant safety. The tests are performed in conformance with the ODCM test frequencies.

The continuous radiation monitors are calibrated to commercial radionuclide standards traceable to the NIST. Each continuous monitor is calibrated during the refueling outage or every 18 months as required by the ODCM, using standard sources with NIST traceability.

A calibration can also be performed by using liquid or gaseous radionuclide standards or by comparison analysis of particulate, iodine, liquid, or gaseous grab samples with laboratory instruments.

11.5.3 EFFLUENT MONITORING AND SAMPLING

The implementation of the requirements of GDC 64 concerning monitoring of gaseous effluent discharge paths for radioactivity is discussed in Section 11.5.2.2.1. Section 11.5.2.2.2 provides applicable discussions for liquid effluent radiation monitors.

11.5.4 PROCESS MONITORING AND SAMPLING

The implementation of the requirements of GDC 60 concerning automatic closure of isolation valves in gaseous and liquid effluent discharge paths is discussed in Sections 11.5.2.1 and 11.5.2.2. Section 11.5.2.2.1 provides a discussion of gaseous process radiation monitors. Section 11.5.2.2.2.2 provides a discussion of liquid process radiation monitors.

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Table 11.5-1

Process and Effluent Radiation Monitoring System
(Gaseous and Airborne Monitors)

Monitor	Detector Location (Number of Channels)	Type	Efficiency/ Sensitivity	Range	Principal Radionuclides Measured	Expected Activity	Upscale Setpoints: Alarms	Upscale Setpoints: Trips
A. SAFETY-RELATED SYSTEMS								
Main steam line radiation monitors MS-RE-3A, 3B, 3C, and 3D	Adjacent to steam lines (4)	γ-ion chamber	3 x 10^{-10} amp/R/h	10^0 - 10^6 mR/h (6 dec. log)	Coolant activation gases	Steam line activity defined in Table 11.1-4	Above full power background, below trip	N/A
Reactor building exhaust plenum radiation monitors REA-RE-9A, 9B, 9C, and 9D	Inline (4)	GM[b]		10^{-2} - 10^2 mR/hr (4 dec. log)		Reactor bldg. activity defined in Table 11.3-7	Above background, below trip	Tech Specs
B. SYSTEMS REQUIRED FOR PLANT OPERATION								
Offgas pretreatment radiation monitor OG-RE-2	Offline, adjacent to sample chamber (1)	γ-ion chamber	3 x 10^{-10} amp/R/h	10^0 - 10^6 mR/h (6 dec. log)	Noble gas fission products	Offgas activity defined in Table 11.3-1	Above background	Tech Specs
Offgas posttreatment radiation monitors OG-RE-601A and OG-RE-601B	Offline (2)	β-scint, Part. Filter, Iodine Filter	4.08 x 10^5 cpm/μCi/ mL	10^0 - 10^7 cpm (2 x 10^{-5} to 24 μCi/cc) (7 dec. log)	^{133}Xe [a]	Offgas activity defined in Table 11.3-1	Above background	ODCM

Table 11.5-1

Process and Effluent Radiation Monitoring System
(Gaseous and Airborne Monitors) (Continued)

Monitor	Detector Location (Number of Channels)	Type	Efficiency/ Sensitivity	Range	Principal Radionuclides Measured	Expected Activity	Upscale Setpoints	
							Alarms	Trips
Charcoal bed vault radiation monitor OG-RE-11	Charcoal bed vault (1)	GM	N/A	10^0 - 10^6 mR/h (6 dec. log)	Noble gas	Charcoal bed inventory defined in Section 11.3	Above background	N/A
Mechanical vacuum pump discharge AR-RE-21	Inline (1)	GM		10^{-2} - 10^6 mR/hr (4 dec. log)	^{133}Xe	Within monitor range	Above background	ODCM
Reactor building elevated discharge radiation monitor								
PRM-RE-1A	Inline (1)	HP Ge	3.69 x 10^8 cps/μCi/cm^3	2.71 x 10^{-8} to 2.71 x 10^{-3} μCi/cm^3	^{133}Xe [a]	RB activity Table 11.3-7	ODCM	N/A
PRM-RE-1B	Inline (1)	HP Ge	≈5.5 x 10^5 cps/μCi/cm^3	≈1.8 x 10^{-5} to 1.8 x 10^0 μCi/cm^3	^{133}Xe [a]	LOCA Table 15.6-10 and 15.6-11	N/A	N/A
PRM-RE-1C	Inline (1)	HP Ge	≈1.28 x 10^1 cps/μCi/cm^3	≈7.8 x 10^{-1} to 7.8 x 10^4 μCi/cm^3	^{133}Xe [a]	LOCA Table 15.6-10 and 15.6-11	N/A	N/A
Particulate filter	Offline	Filter	N/A	from 10^{-12} μCi/cm^3	γ emitters[c]	ODCM Table 6.2.2.1-1	N/A	N/A
Iodine filter	Offline	Charc. cart.	N/A	from 10^{-12} μCi/cm^3	γ emitters	ODCM Table 6.2.2.1-1	N/A	N/A

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Table 11.5-1

Process and Effluent Radiation Monitoring System
(Gaseous and Airborne Monitors) (Continued)

Monitor	Detector Location (Number of Channels)	Type	Efficiency Sensitivity	Range	Principal Radionuclides Measured	Expected Activity	Upscale Setpoints: Alarms	Upscale Setpoints: Trips
Turbine bldg. vent. exhaust radiation	Offline (2)	Part. filter[c] iodine filter				Turbine bldg. activity defined in Table 11.3-7	ODCM	N/A
TEA-RE-13 Low Range		ß-scint	N/A	10^{-7} -10^{-1} μCi/cc (6 dec. log)	^{133}Xe[a]			
TEA-RE-13A High Range		ß-scint	N/A	10^{-3} - 10^{3} μCi/cc (6 dec. log)	^{133}Xe[a]	LOCA mixture of F.P. activity see Table 15.6-9		
Radwaste bldg. vent. exhaust radiation	Offline (2)					Radwaste bldg. activity defined in Table 11.3-7	ODCM	N/A
WEA-RE-14 Low Range		ß-scint	N/A	10^{-6} -10^{-1} μCi/cc (5 dec. log)	^{133}Xe[a]			
WEA-RE-14A High Range		ß-scint	N/A	10^{-3} - 10^{3} μCi/cc (6 dec. log)	^{133}Xe[a]	LOCA mixture of F.P. activity see Table 15.6-9		
Primary containment LOCA monitors	Adjacent to containment steel walls	γ-ion chambers	amp/R/h	R/h (6 dec. log)	Fission product gases	Within monitor range	Above background	N/A
CMS-RE-27A, 27B	Outside(2)		1 x 10^{-10} amp/R/h	10^{-2} x 10^{4} R/h (6 dec. log)				
[b] CMS-RE-27E, 27F	Inside (2)		7 x 10^{-11} amp/R/h	10^{0} - 10^{7} R/h (6 dec. log)				

Table 11.5-1

Process And Effluent Radiation Monitoring System
(Gaseous And Airborne Monitors) (Continued)

Monitor	Location (Number of Channels)	Type	Efficiency/ Sensitivity	Range	Principal Radionuclides Measured	Expected Activity	Upscale Setpoints	
							Alarms	Trips
Primary containment radiation monitor	Offline					Containment discussed in Section 12.2		
Particulate CMS-RE-12-1A CMS-RE-12-1B	(2)	Part. filter ß-scint	N/A	10^{-12} - 10^{-6} μCi/cc (6 dec. log)	Fission gas daughter and corrosion activation product [d]		Above background, below trip	Full scale
Gas CMS-RE-12-3A CMS-RE-12-3B	(2)	ß-scint	N/A	10^{-7} – 10^{-1} μCi/cc (6 dec. log)			Above background, below trip	Full scale

[a] Sensitivity based on this radionuclide.

[b] Not required for plant operation.

[c] Composite of particulate filters analyzed for ^{89}Sr and ^{90}Sr.

[d] For particulate, Cs-137 is the fission product radionuclide used for converting counts per minute to μCi/cc. For gas, Kr-85 is the fission product radionuclide used for converting counts per minute to μCi/cc. The monitors display units of μCi/cc.

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Table 11.5-2

Process and Effluent Radiation Monitoring System (Liquid Monitors)

						Upscale Setpoints	
Monitor	Detector Location (Number of Channels)	Type	Range	Principal Radionuclides Measured	Expected Activity	Alarms	Trips
Residual heat removal standby service water radiation monitor SW-RE-4, 5	Offline (2)	γ-scint	10^{-1} - 10^{6} cps (7 dec. log)	^{137}Cs [a] ^{60}Co	Less than minimum detector sensitivity	Above background	Not applicable
Reactor building closed cooling water radiation monitor RCC-RE-7	Offline (1)	γ-scint	10^{-1} - 10^{6} cps (7 dec. log)	^{137}Cs [a] ^{60}Co	Less than minimum detector sensitivity	Above background	Not applicable
Radwaste effluent radiation monitor FDR-RE-6	Offline (1)	γ-scint	10^{-1} - 10^{6} cps (7 dec. log)	^{137}Cs [a] ^{60}Co	Section 11.2	Above background	ODCM[b]
Circulating water effluent radiation monitor CBD-RE-8	Inline (1)	γ-scint	10^{-1} -10^{6} cps (7 dec. log)	^{137}Cs [a] ^{60}Co	Less than minimum detector sensitivity	Above background	Section 11.5.2.2.2.4
Plant service water radiation monitor TSW-RE-5	Offline (1)	γ-scint	1.0E-08 to 1.0E-02 μCi/cc (6 dec. log)	^{137}Cs [a] ^{60}Co	Less than minimum detector sensitivity	Above background	Not applicable

[a] Sensitivity based on this radionuclide.

[b] The alarm point will be set, based upon the activity, radionuclides, and dilution factor so that the concentration in the discharge line is less than 10 CFR 20 Appendix B, Table II, Column 2 limits.

Table 11.5-3

Radiological Analysis Summary of Gaseous Effluent Samples

Grab Sample Sample Description	Frequency	Sensitivity Analysis	μCi/ml	Purpose
Reactor building elevated release exhaust	Weekly	Gamma spectrum[a]	10^{-11}	Effluent record
		Radioiodines[b]	10^{-10}	
	Quarterly	^{89}Sr and ^{90}Sr[c]	10^{-11}	
	Monthly	Gross alpha	10^{-11}	
		Tritium	10^{-11}	
Radwaste building exhaust	As above	As above		Effluent record
Turbine building exhaust	As above	As above		Effluent record

[a] On particulate filter

[b] On charcoal cartridge

[c] On composite of particulate filters

Table 11.5-4

Radiological Analysis Summary of Liquid Process Samples

Grab Sample Sample Description	Frequency	Sensitivity: Analysis	Sensitivity: μCi/ml	Purpose
Reactor coolant	In accordance with Technical Specifications			
		^{131}I, ^{133}I	10^{-6}	Evaluate fuel cladding integrity
		Gamma spectrum	10^{-6}	Determine radionuclides present in system
Reactor water cleanup system	Periodically	Gamma spectrum	10^{-6}	Evaluate cleanup efficiency
Condensate storage tanks	Weekly	Gamma spectrum	10^{-6}	Tank inventory
Fuel pool filter - demineralizer inlet and outlet	Periodically	Gamma spectrum	10^{-6}	Evaluate system performance
Waste collector tank	Batch[a]	Gamma spectrum	10^{-6}	Evaluate system performance
Floor drain collector tank	Batch[a]	Gamma spectrum	10^{-6}	Evaluate system performance
Chemical waste tank	Batch[a]	Gamma spectrum	10^{-6}	Evaluate system performance
Detergent drain tank (2)	Batch[a]	Gamma spectrum	10^{-6}	Tank inventory

[a]Analysis performed on an infrequent basis as needed to evaluate equipment performance or unusual water chemistry.

Table 11.5-5

Radiological Analysis Summary of Gaseous Process Samples

Grab Sample Sample Description	Frequency	Sensitivity Analysis	Sensitivity μCi/ml	Purpose
Containment atmosphere (drywell)	Periodically	Gamma spectrum[a]	10^{-11}	Evaluate prior to discharge
	Prior to entry	Gamma spectrum	10^{-10}	Determine need for respiratory equipment
Offgas pretreatment monitor sample	Periodically	Gamma spectrum	10^{-6}	Determine offgas activity
Offgas posttreatment sample	Periodically	Gamma spectrum[a,b]	10^{-10}	Determine offgas system cleanup performance
		Gamma spectrum[c]	10^{-6}	

[a] On particulate filter

[b] On charcoal cartridge

[c] Noble gas

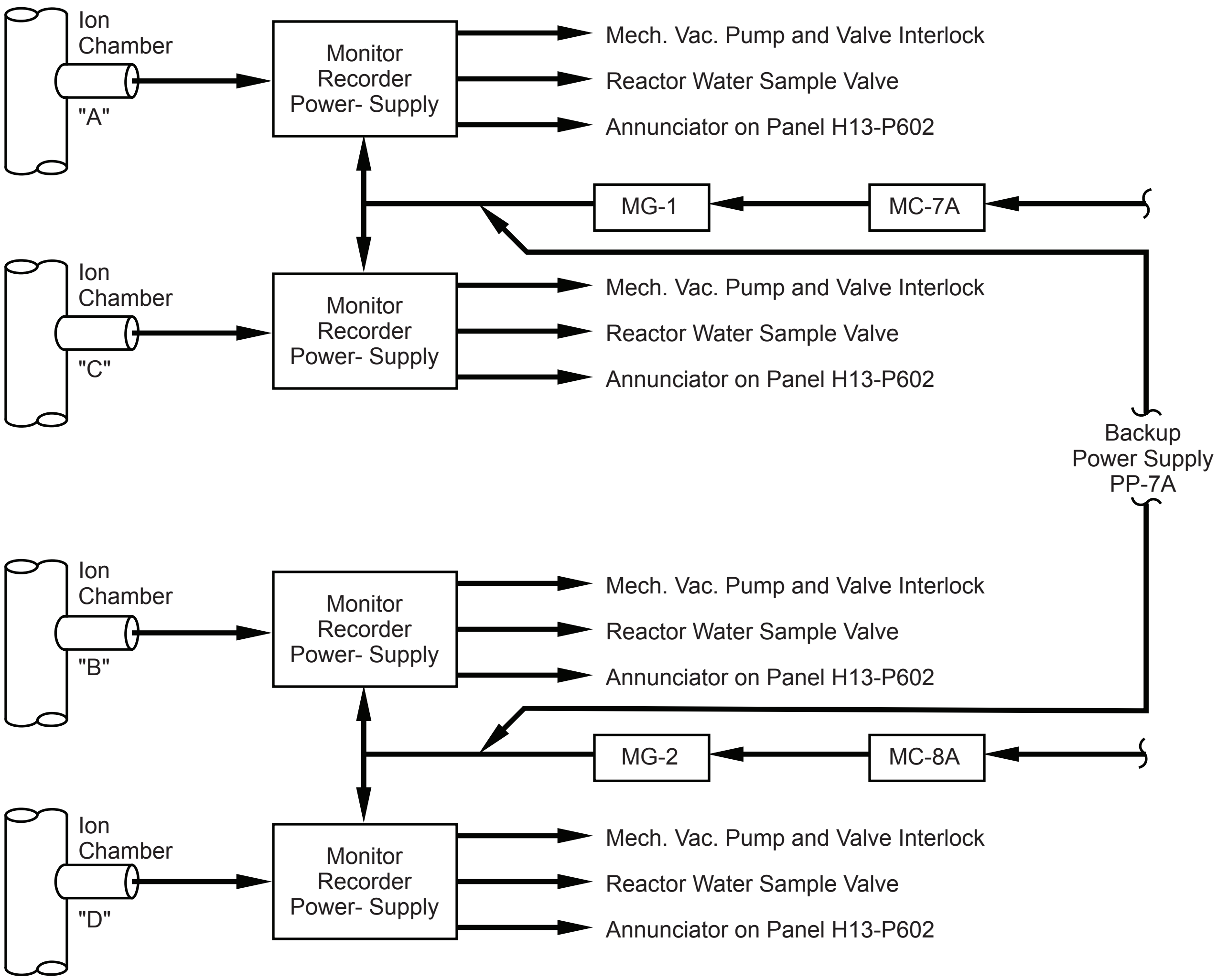
Table 11.5-6

Radiological Analysis Summary of Liquid Effluent Samples

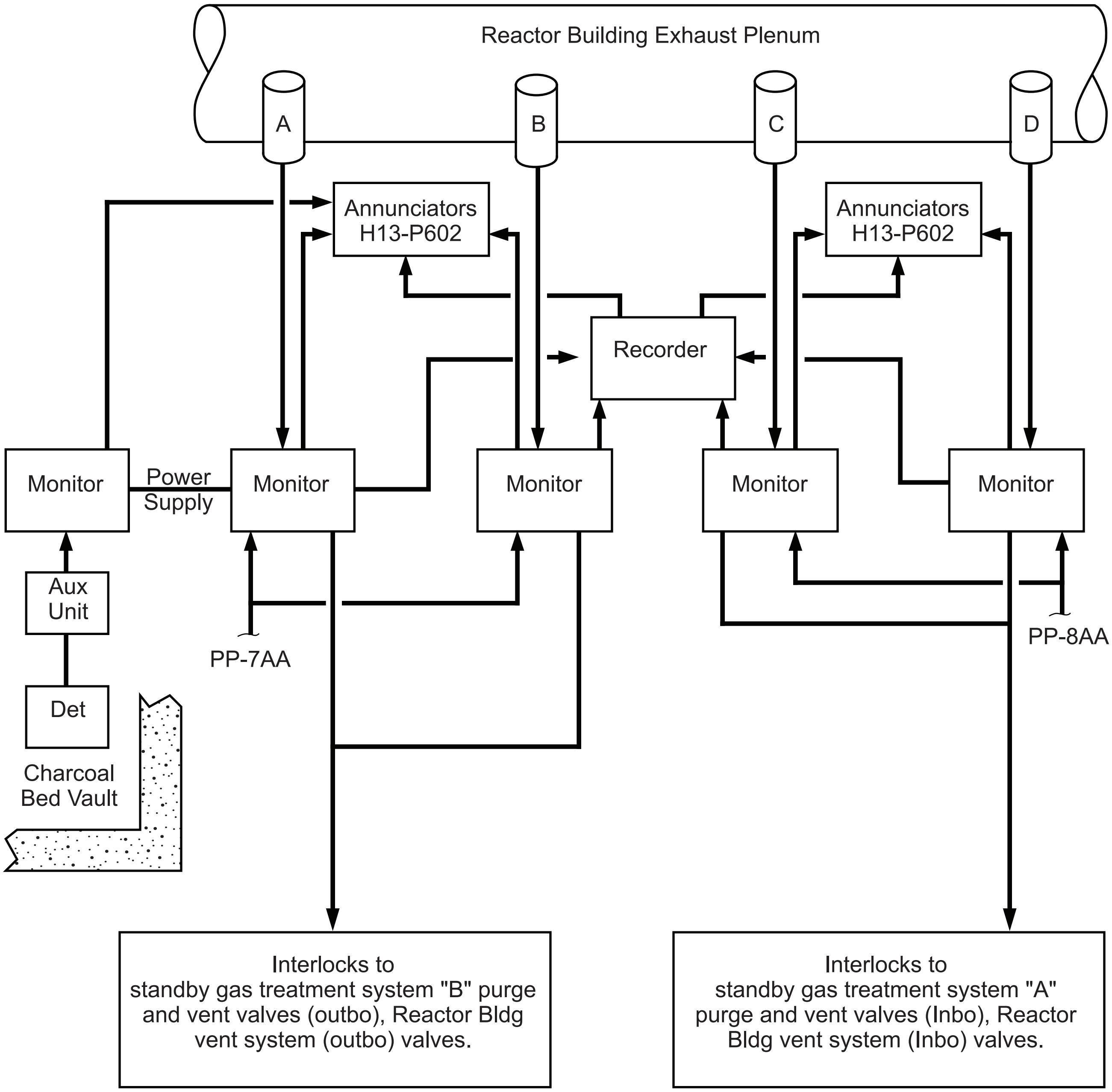
Grab Sample Sample Description	Frequency	Sensitivity Analysis	Sensitivity µCi/ml	Purpose
Floor drain sample tank	Batch[a]	Gamma spectrum	10^{-6}	Effluent discharge record
Waste sample tanks (2)	Batch[a]	Gamma spectrum	10^{-6}	Effluent discharge record
Liquid radwaste effluent (composite of all tanks discharged)	Monthly[b]	Gross alpha	10^{-7}	Effluent discharge record
		Tritium	10^{-5}	
	Quarterly[b]	$^{89}Sr/^{90}Sr$	10^{-8}	
Circulating water discharge line	Weekly grab of continuously collected proportional sample	Gross ß or Gamma Spectrum	10^{-6}	Effluent discharge record (backup sample)
		Tritium	10^{-5}	

[a] If tank is to be discharged, analyses will be performed on each batch. If tank is not to be discharged, analyses will be performed periodically to evaluate equipment performance.

[b] If liquids were discharged during the previous month or quarter, as applicable.



Columbia Generating Station Final Safety Analysis Report	Main Steam Line Monitors
Draw. No. 900547.32	Rev. Figure 11.5-1



Reactor Building Exhaust Plenum Monitors and Charcoal Bed Vault Monitor

Draw. No. 900547.48 | Rev. | Figure 11.5-2

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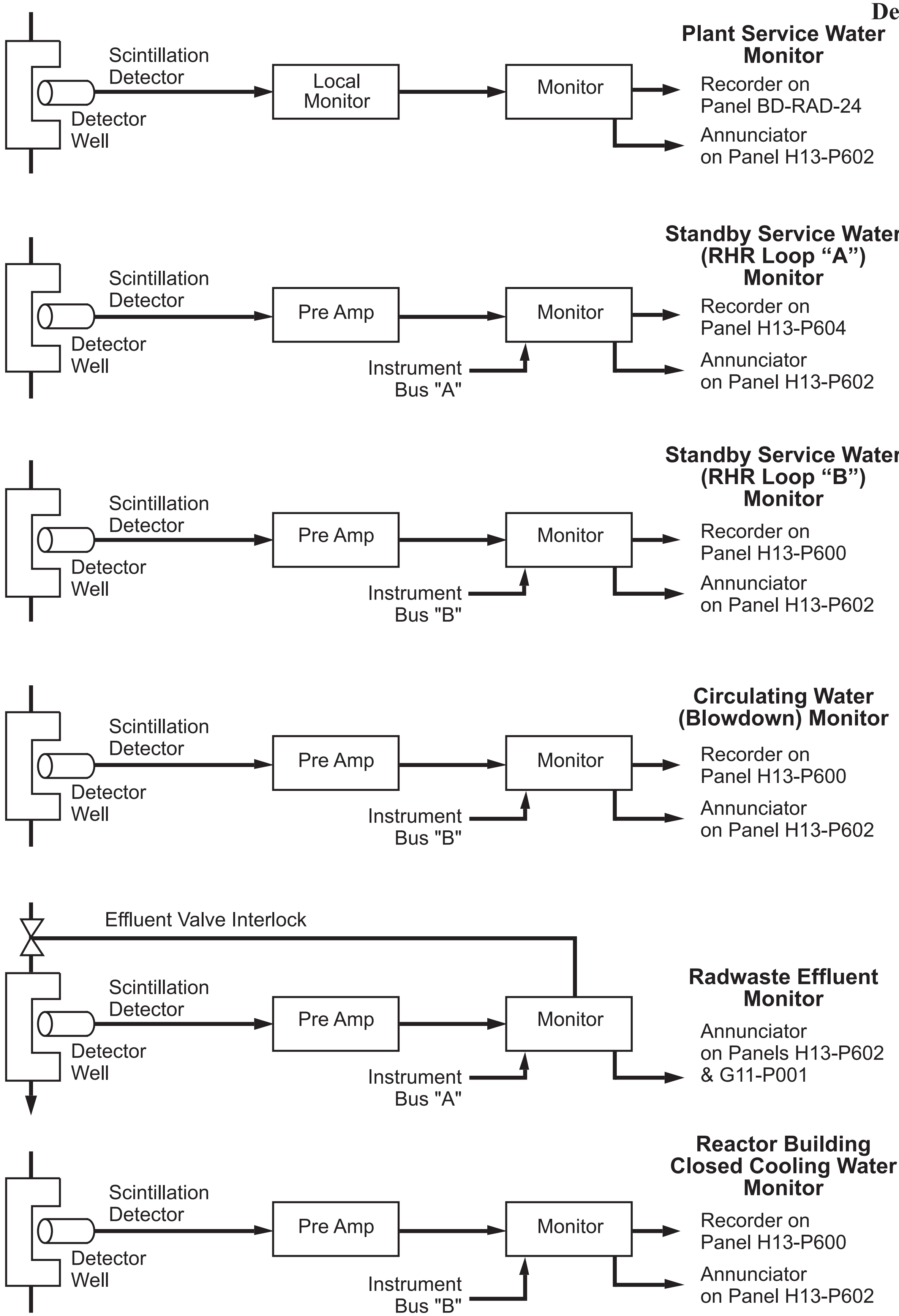
Control Room Fresh Air Intake Monitors

Columbia Generating Station
Final Safety Analysis Report

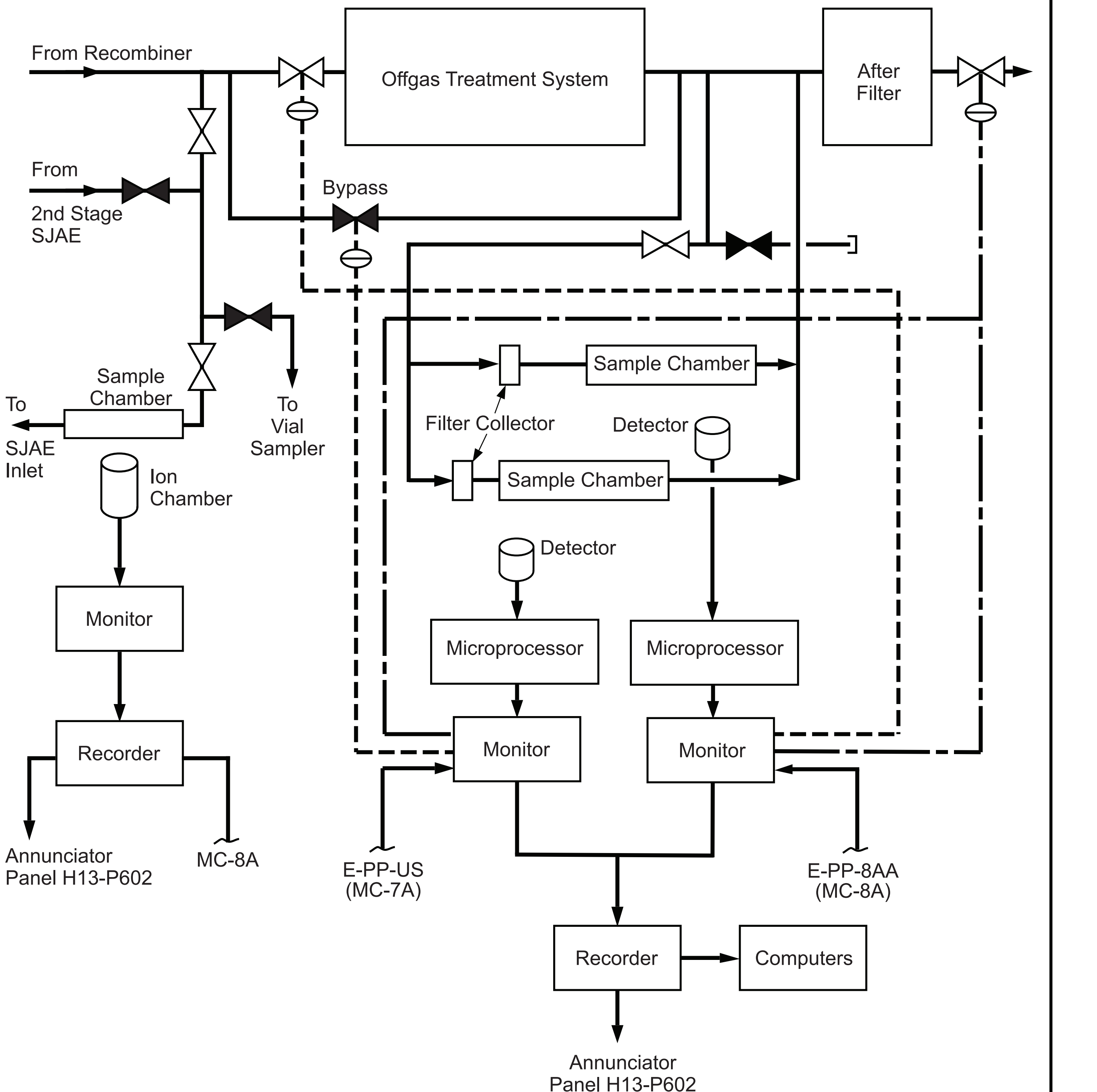
Draw. No. 900547.49 | Rev. 1 | Figure 11.5-3



Columbia Generating Station
Final Safety Analysis Report

Process and Effluent Liquid Radiation Monitors

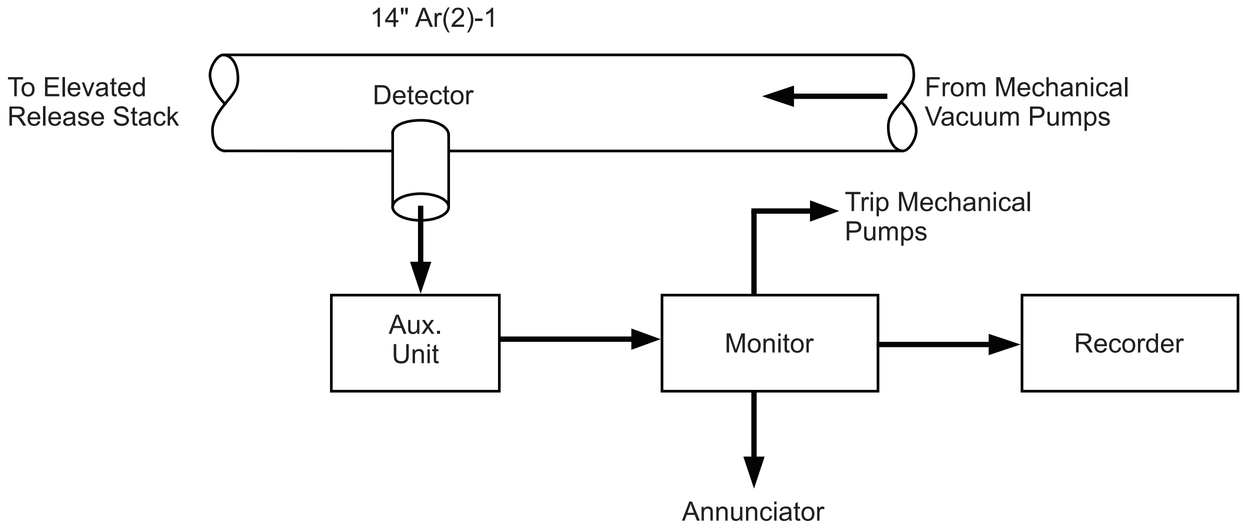
Draw. No. 900547.50	Rev.	Figure 11.5-4



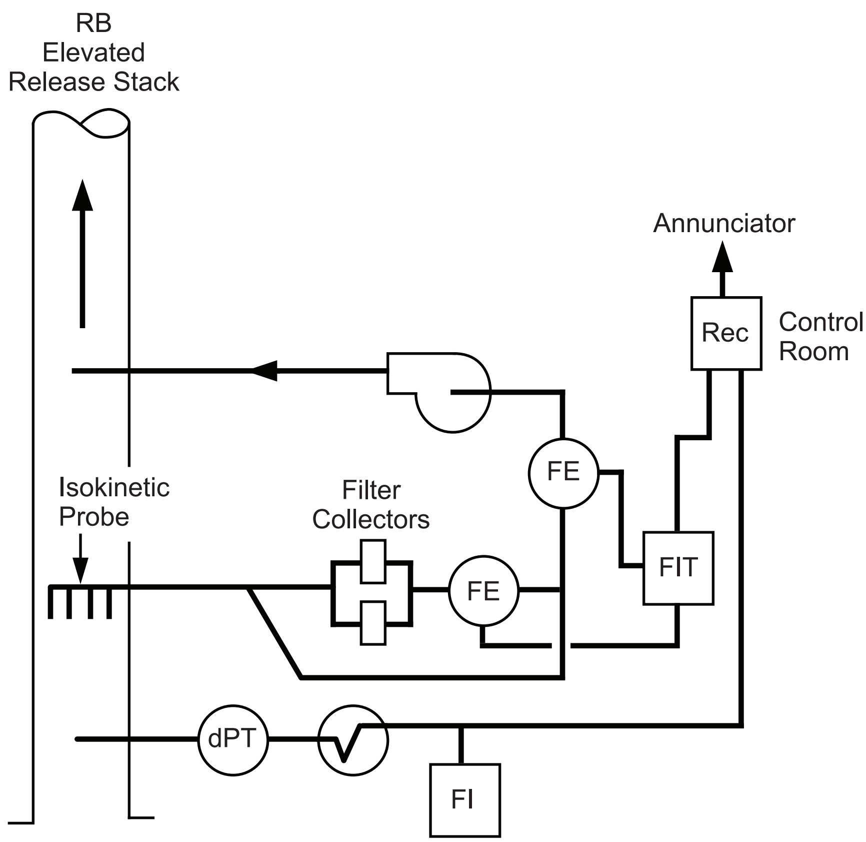
Offgas Pretreatment and Posttreatment Radiation Monitors

Draw. No. 900547.51 | Rev. | Figure 11.5-5

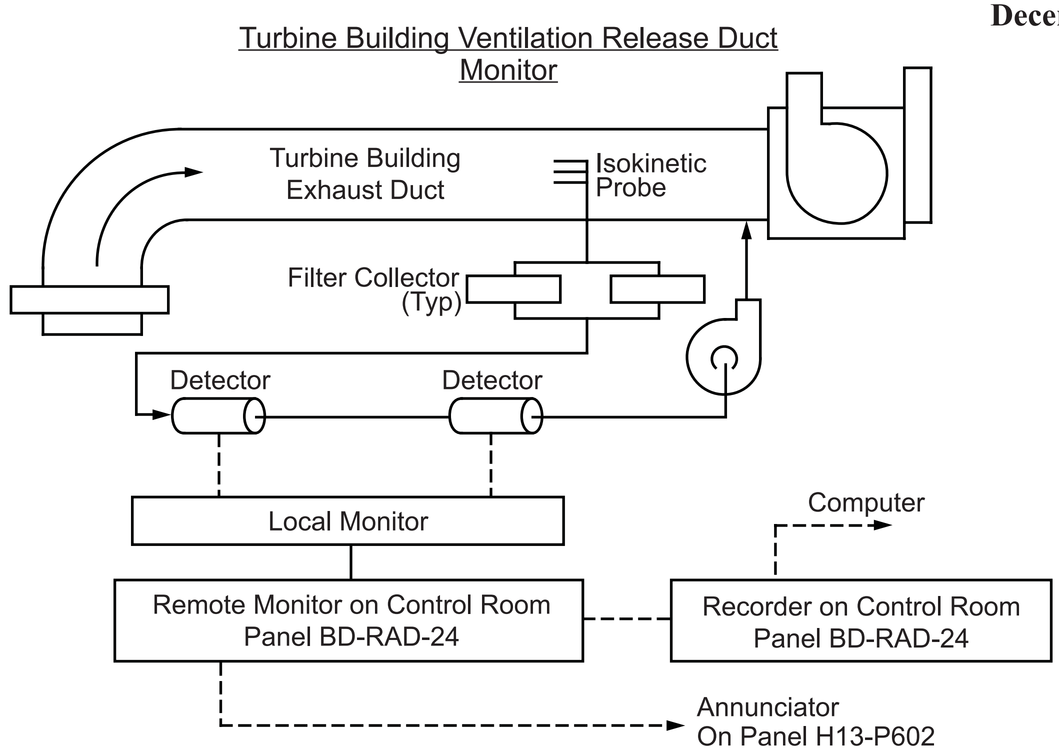
Mechanical Vacuum Pumps Exhaust Monitor

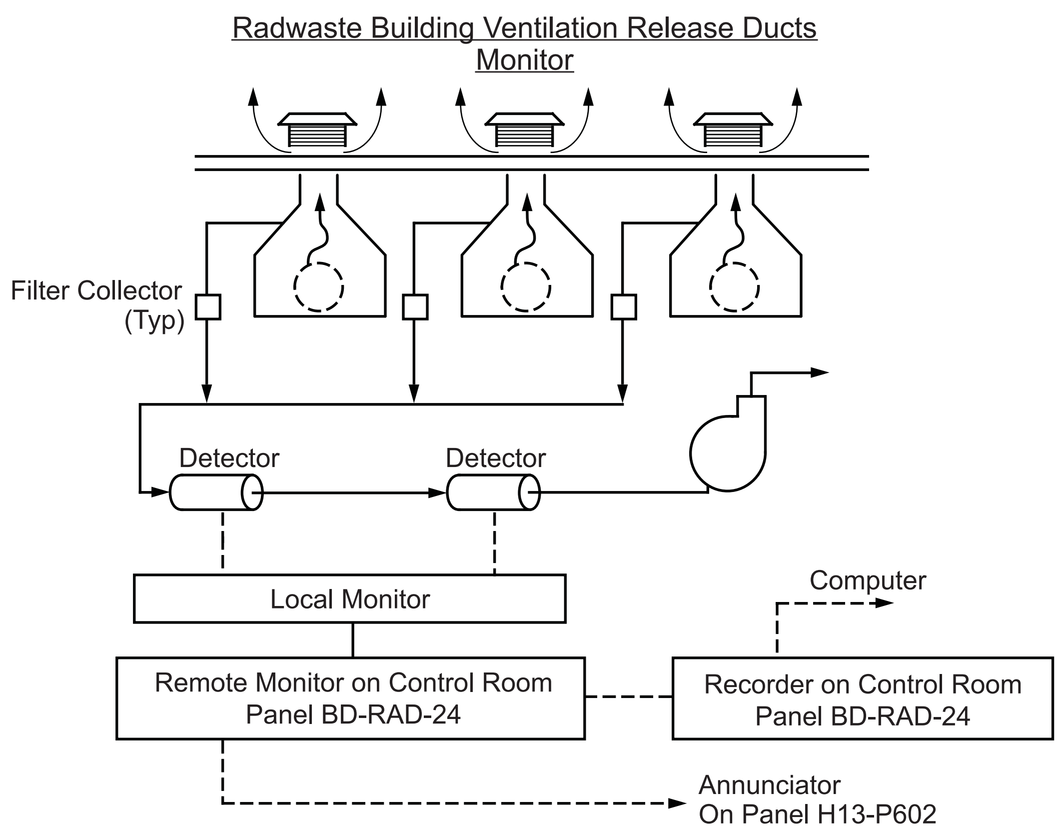


Reactor Building Ventilation Release Duct Monitor



Columbia Generating Station Final Safety Analysis Report	Mechanical Vacuum Pumps Exhaust Monitor and Reactor Building Elevated Release Stack Monitor		
	Draw. No. 950021.32	Rev.	Figure 11.5-6





Columbia Generating Station **Final Safety Analysis Report**	**Turbine and Radwaste Building Ventilation Release Duct Monitors**
	Draw. No. **960690.01** — Rev. **1** — Figure **11.5-7**

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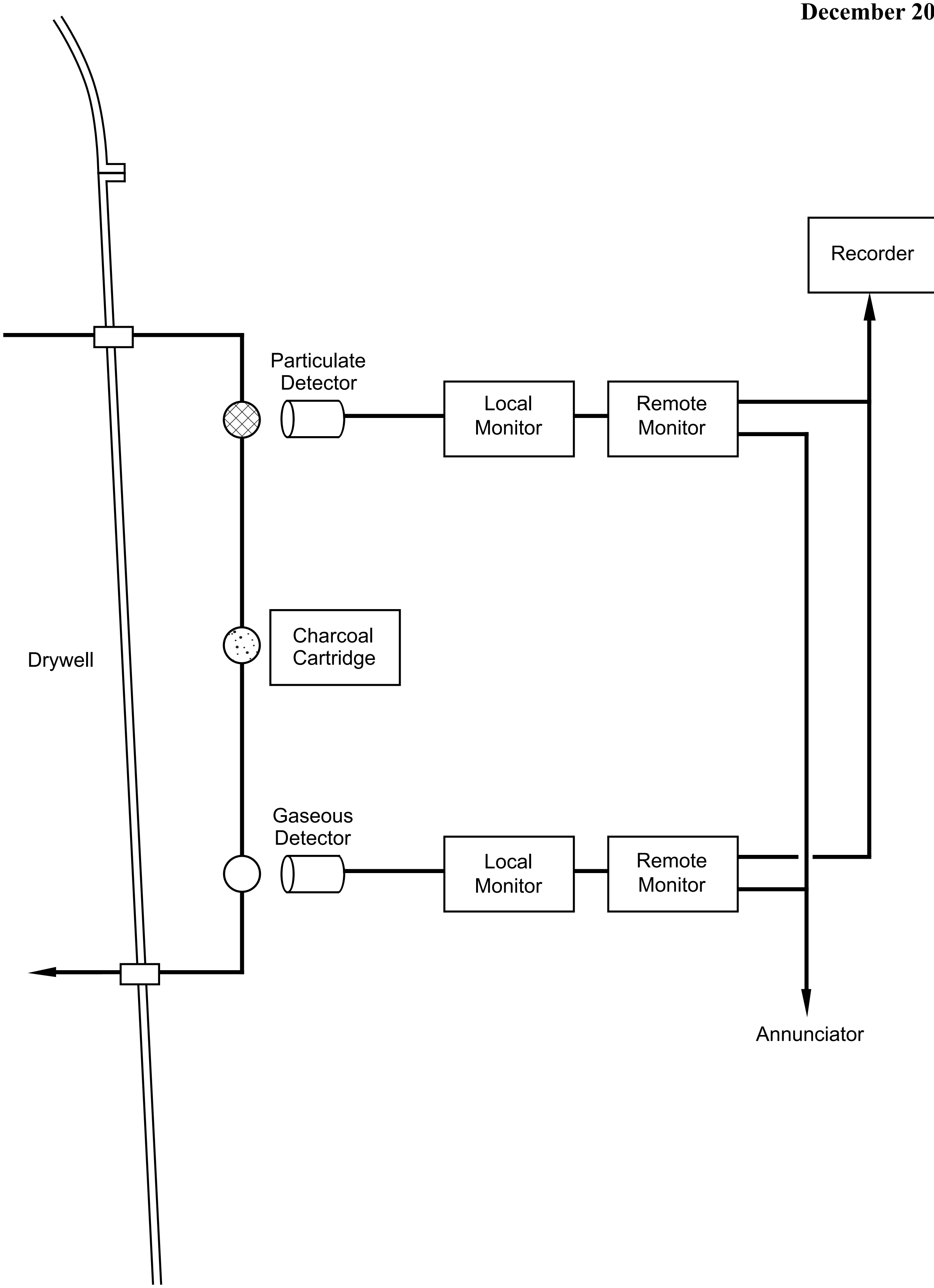
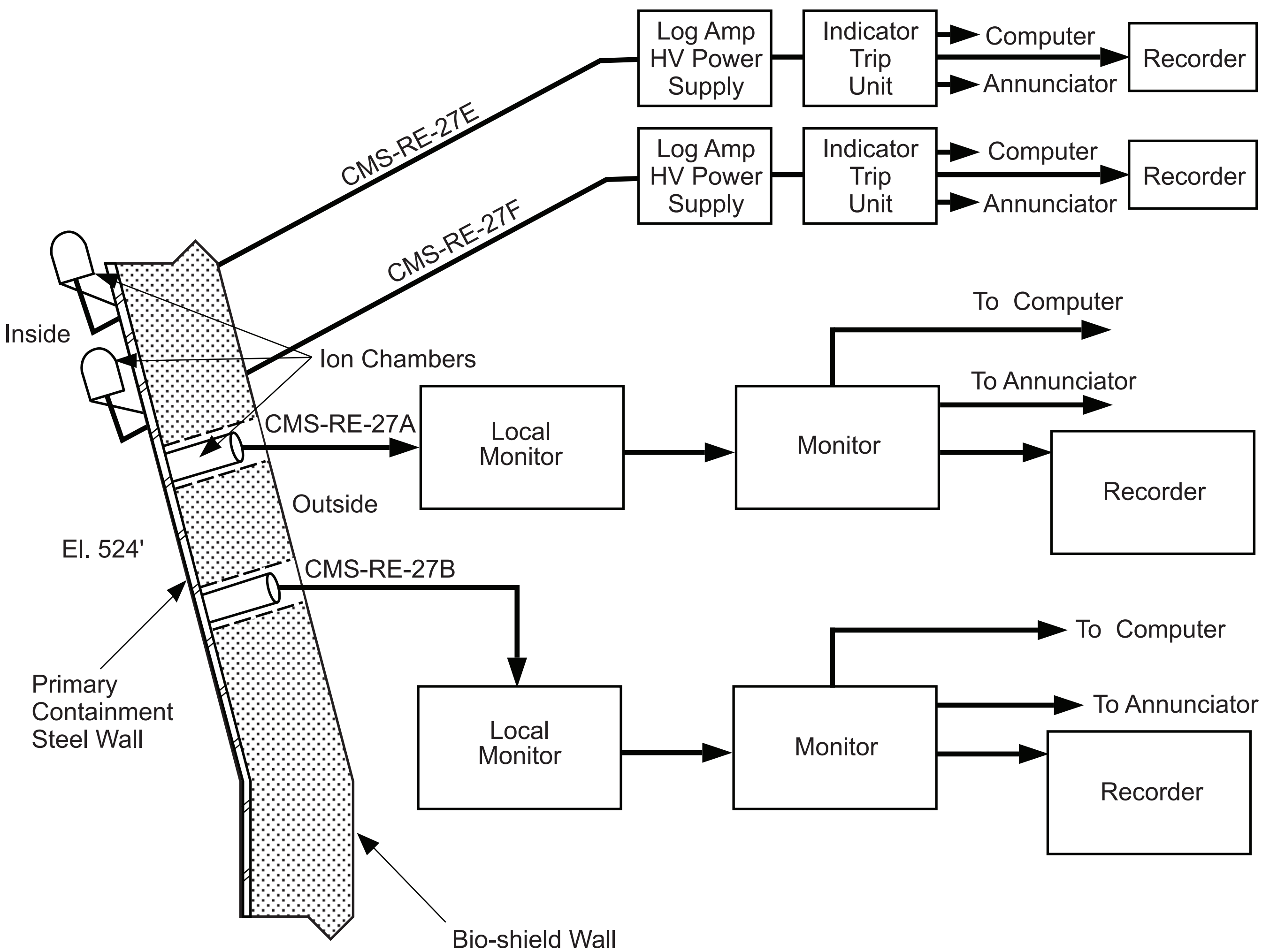


Primary Containment Leak Detection Monitor

Draw. No. 970187.21 | Rev. | Figure 11.5-9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Computer
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Computer
Annunciator
Recorder
CMS-RE-27E
CMS-RE-27F
To Computer
To Annunciator
Inside
Ion Chambers
CMS-RE-27A
Local
Monitor
Monitor
Recorder
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To Annunciator
Primary
Containment
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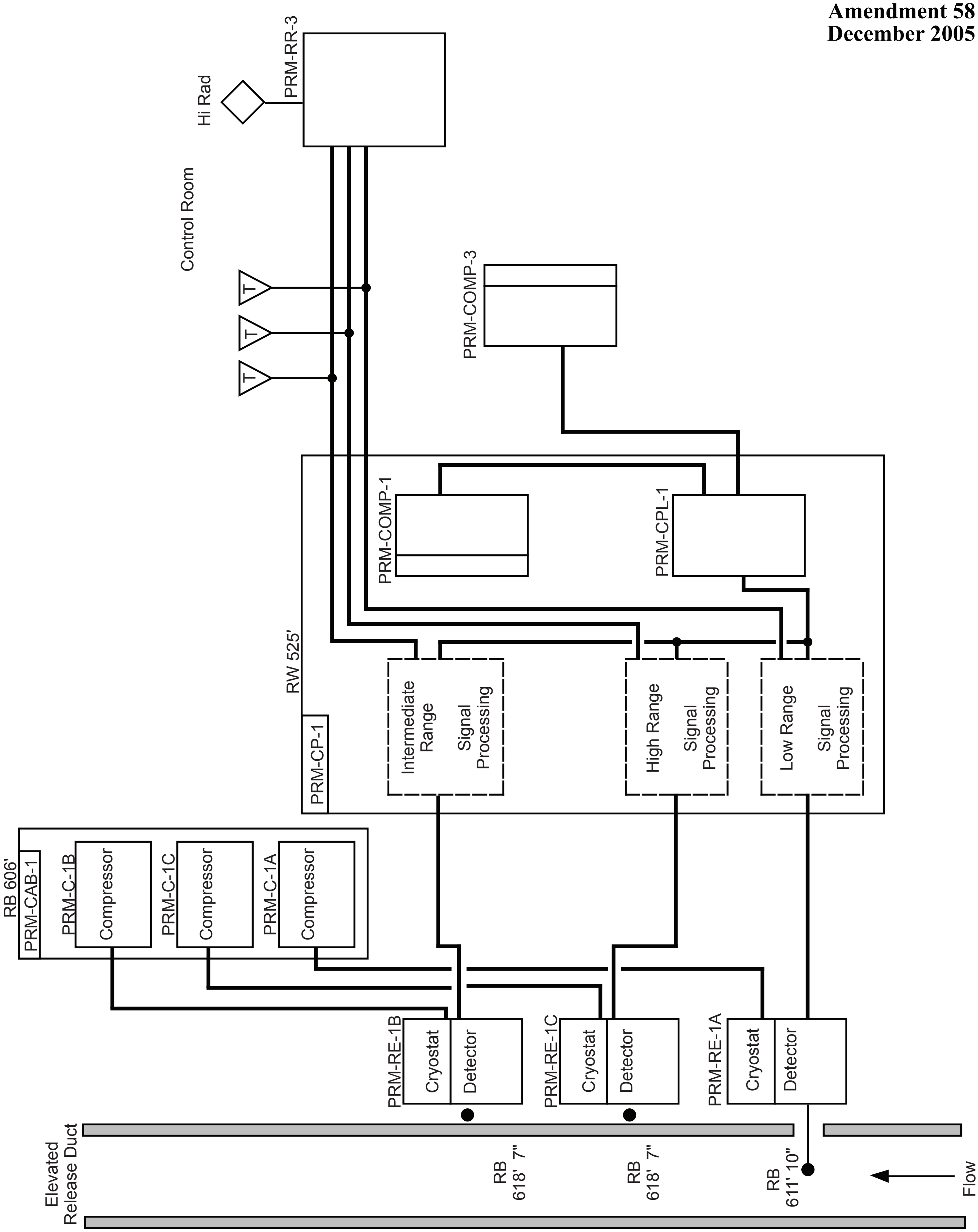
Monitor
Recorder
Bio-shield Wall



Columbia Generating Station
Final Safety Analysis Report

Elevated Release Stack LOCA Monitoring

Draw. No. 950021.29 | Rev. | Figure 11.5-11

11.6 POSTACCIDENT SAMPLING SYSTEM

11.6.1 DESIGN BASIS

Columbia Generating Station is using a General Electric postaccident sampling system (PASS) capable of sampling the primary containment and reactor building atmosphere and of obtaining liquid samples from the reactor, RHR loops and various reactor building sumps. This system is designed to obtain grab samples which may be analyzed onsite or transported to offsite facilities for more detailed analysis if necessary. The sample station is located in the radwaste building and is shielded to reduce radiation exposure rates to the operator. All remote-operated valves are controlled from this area. Lead pigs are provided for radiation protection when transporting samples either to onsite facilities or offsite.

All valves used are fully qualified for the environment in which they are located inside and outside reactor containment. Power for the postaccident sampling equipment is supplied from Division 1 and Division 2 critical power sources and will be available during accident conditions.

License Amendment No. 184 removed the requirement for PASS from the Technical Specifications effective January 27, 2003. While this amendment removes the Technical Specification requirements for PASS, it requires a commitment for contingency plans for obtaining and analyzing highly radioactive samples from the Reactor Coolant System, suppression pool, and containment atmosphere. Until further changes are made to the PASS hardware it will be operated as described in this FSAR Section. This operation meets the commitment for contingency plans for sampling.

11.6.2 SYSTEM DESCRIPTION

Gas samples will be obtained from locations in the drywell, the suppression pool atmosphere, and from the secondary containment atmosphere. The sample system is designed to operate at pressures ranging from subatmospheric to maximum design pressures of the primary and secondary containment. Heat-traced sample lines are used outside the primary containment to prevent precipitation of moisture and resultant loss of particulates and iodines in the sample lines. The gas samples may be passed through a particulate filter and silver zeolite cartridge for determination of particulate activity and iodine activity by subsequent analysis of the samples on a gamma spectrometer system. Alternatively, the sample flow bypasses the particulate/iodine sampler, is chilled to remove moisture, and a 15-ml grab sample can be taken for determination of gaseous radioactivity and for gas composition by gas chromatography. This size sample vial has been adopted for all gas samples to be consistent with present offgas sample vial counting factors.

Reactor coolant samples will be obtained from two points in the jet pump pressure instrument system when the reactor is at pressure. The jet pump pressure system has been determined to

be an optimum sample point for accident conditions. The pressure taps are well protected from damage and debris. If the recirculation pumps are secured, the water level will be raised about 18 in. above normal. This provides natural circulation of the bulk coolant past the taps. Also, the pressure taps are located sufficiently low to permit sampling at a reactor water level even below the lower core support plate.

A single sample line is also connected to both loops in the RHR system. This provides a means of obtaining a reactor coolant sample when the reactor is depressurized and at least one of the RHR loops is operated in the shutdown cooling mode. Similarly, a suppression pool liquid sample can be obtained from the RHR loop lined up in the suppression pool cooling mode. Samples from the five drain sumps in the reactor building are also available.

The sample system isolation valves are controlled from the local control panel. The sample system is designed for a purge flow of 1 gpm, which is sufficient to maintain turbulent flow in the sample line. Purge flow is returned to the suppression pool. The high flush flow also serves to alleviate cross-contamination of the samples when switching from one sample point to another.

All liquid samples are taken into septum bottles mounted on sampling needles. The sample station is basically a bypass loop on the sample purge line. In the normal lineup, the sample flows through a conductivity cell (readable range 0.1 to 1000 μS/cm) and then through a ball valve bored out to 0.10-ml volume. Flow through the sample panel is established, the valve is rotated 90°, and a syringe is used to flush the sample plus a measured volume of diluent (generally 10 ml) through the valve and into the sample bottle. This provides a dilution of 100:1 to the sample. Alternatively, the valve sampling sequence can be repeated 10 times to provide a 1-ml sample diluted 10:1. The sample is transported to the laboratory for further dilution and subsequent analysis. Alternatively, the sample flow can be diverted through a 70-ml bomb to obtain a large pressurized volume. This 70-ml volume can be circulated and depressurized into a known volume gas expansion chamber. The pressure change in this chamber will be used to calculate the total dissolved gases in the reactor coolant. A grab sample of these gases may be taken through a septum port for subsequent analysis.
Ten milliliter aliquots of this degassed liquid can also be taken for on or offsite chemical analyses requiring a relatively large sample. A radiation monitor in the liquid sample enclosure monitors liquid flow from the sample station to provide immediate assessment of the sample activity level. This monitor also provides information as to the effectiveness of the demineralized water flushing of the sample system following sample operation. The control instrumentation is installed in two 2 ft x 2 ft x 6 ft high standard cabinet control panels. One panel contains the conductivity and radiation level readouts. Another control panel contains the flow, pressure and temperature indicators, and the various control valves and switches.

A graphic display panel, installed directly below the main control panel, shows the status of the pumps and valves at all times. The panel also indicates the relative position of the pressure

gauges and other items of concern to the operator. The use of this panel will improve operator comprehension and assist in trouble-shooting operation.

Appropriate sample handling tools, a gas sampler vial positioner and gas vial cask are available to the operator at the sampling station. The gas vial is installed and removed by use of the vial positioner through the front of the gas sampler. The vial is then manually placed down in the cask with the positioner which allows the vial to be maintained about 3 ft from the individual performing the operation.

The small-volume (10 ml) liquid sample is remotely obtained through the bottom of the sample station by use of the small-volume cask and cask positioner. The cask positioner holds the cask and positions the cask directly under the liquid sampler. The sample vial is manually raised within the cask to engage the hypodermic needles. When the sample vial has been filled, the bottle is manually withdrawn into the cask. The sample vial is always contained within lead shielding during this operation. The cask is then lowered and sealed prior to transport to the laboratory.

A large-volume cask and cask positioner is available for transporting large liquid samples. A 21-ml bottle is contained within a lead shielded cask. This sample bottle is raised from its location in the cask to the sample station needles for bottle filling. The sample station will only deliver 10 ml to this sample bottle. When filled, the bottle is withdrawn into the cask. The sample bottle is always shielded by 5 to 6 in. of lead when in position under the sample station and during the fill and withdraw cycles, thus reducing operator exposure.

The cask is transported to the required position under the sample station by a dolly cask positioner. When in position this cask is hydraulically elevated approximately 1.5 in. by a small hand pump for contact with the sample station shielding under the liquid sample enclosure floor. The sample bottle is raised, held, and lowered by a simple push/pull cable. The cask is sealed by a threaded top plug that inserts above the sample bottle. The weight of this large-volume cask is approximately 700 lb.

The particulate filters and iodine cartridges are removed via a drawer arrangement. The quantity of activity which is accumulated on the cartridges is controlled by a combination of flow orificing and time sequence control of the flow valve opening. In addition, the deposition of iodine is monitored during sampling using a radiation detector installed adjacent to the cartridge. These samples will hence be limited to activity levels which will normally not require shielded sample carriers to transport the samples to the laboratory.

Based on information developed by General Electric, Energy Northwest has developed plant-specific procedures for the determination of the extent of core damage under accident conditions using inplant radiation and hydrogen monitors. This meets the commitment for ability to determine fuel damage, postaccident.