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SCHWENCER,A. Licensing Branch 2

revised 6/15/82 P.S.

SUBJECT: Forwards Amend 47 to OL,containing Revision 30 to FSAR.

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	NRR/DSI/AEB 26	1	1	NRR/DSI/ASB 27	1	1
	NRR/DSI/CPB 10	1	1	NRR/DSI/CSB 09	1	1
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	REG FILE 04	1	1	RGN1	2	2
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	NTIS	1	1			

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JUN 04 1982

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Project Management
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
AMENDMENT 47 TO OPERATING LICENSE APPLICATION
ER 100450 FILE 841-1 Docket Nos. 50-387
PLA-1116 50-388

Dear Mr. Schwencer:

Attached are sixty (60) copies of Amendment No. 47 to the operating license application. This ammendment contains Revision 30 to the Susquehanna SES Final Safety Analysis Report.

This amendment contains the following changes:

o	Section 1.2	Update of figures to show the location of the PASS system
o	Section 1.7	Update of electrical drawing table. Drawings are submitted under separate cover
o	Section 1.8	Update of P&ID legend and symbol figure
o	Section 2.4	Correction of typographical errors
		Correction of figure references
		Clarification of the design of the discharge channel from the spray pond
		Corrections of elevation and size scuppers
		Correction of the loadings due to wind-wave action and earthquakes on pipe supports
o	Section 3.1	Correction of typographical errors

B001
3/60

		Addition of references to other sections of the FSAR or other reports to add clarity
o	Section 3.2	Correction of typographical errors
		Correction of the principal construction codes and standards
o	Section 3.7b	Correction of typographical errors
		Clarification that rodofoam will be left in place in joints between structures and will have little effect on the structures
		Clarification of the use of dynamics analysis for equipment
		Clarification of the structure foundation interaction coefficients used in the analysis
		Revision of the floor response spectrum for the ESSW pumphouse
o	Section 3.10c	Correction of typographical errors
		Update section to reflect the status of seismic qualification of electrical equipment
o	Section 3.11	Update of Table 3.11-6 to state the normal and maximum plant environmental conditions
o	Section 3.12	Clarification of the separation criteria used for cables and raceways
		Clarification of the separation critiria used of confined spaces in the control structure
		Clarification of the separation criteria used in panels
o	Section 4.6	Correction of typographical errors
		Description of the modifications made to the scram discharge volume
		Update of P&ID figures
o	Section 5.1	Update of P&ID figures

o Section 5.2	Addition of piping categories for inspection of piping
	Deletion of the drywell floor drain sump level instrumentation accuracy
o Section 5.4	Correction of typographical errors
	Clarification that the RCIC pump suction automatically transfers to the suppression pool from the CST on a low water level signal and the interlocks associated with this function.
	Clarification of the bulk suppression pool temperature
	Update of P&ID figures
o Section 6.2	Correction of typographical errors
	Clarification that the instrument sensing lines are not an extension of the containment
	Update of the description of the hydrogen and oxygen monitoring system
	Clarification of the low pressure leak test frequency
	Update of the calculated peak drywell pressure
	Addition of the description of the capping of five downcomers
	Update of the containment penetration data tables
	Update of P&ID figures
o Section 6.3	Update of P&ID figures
o Section 6.4	Correction of typographical errors
o Section 6.5	Correction of typographical errors

Correction to state that the CSEOASS can be manually started

Addition of environmental design criteria on the ESF filter systems compliance with the recommendations of Regulatory Guide 1.52 table

o Section 6.6	Addition of bolting categories to reflect the preservice inspection upgrade for Class 1 and Class 2 pressure retaining bolting
o Section 6.7	Addition to describe to MSIF-LCS interlock with the inboard main steamline isolation valve
o Section 7.1	Correction of typographical errors
	Deletion of reference to IEEE 387-1982 since this applies to the diesel generators which are not part of the instrumentation discussed in this chapter
	Moved the discussion of the Rod Block Monitoring System to 7.7
	Moved to discussion of the refueling interlocks to Section 7.7
	Moved the discussion of the Reactor Manual Control System to Section 7.7
	Added reference to PP&L's Environmental Qualification of Class 1E Equipment program
o Section 7.2	Correction of typographical errors
	Correction of table, figure and section references
	Addition of discussion of the modifications to the SRV discharge volume
	Addition of a reference to PP&L's Environmental Qualification of Class 1E Equipment program

		Addition of reference to Technical Specifications for instrument setpoints
o	Section 7.3	Correction of typographical errors
		Correction of table, figure and section references
		Addition of reference to PP&L's Environmental Qualification of Class 1E Equipment program
		Clarification of divisionalization of HPCI & RCIC
		Addition of reference to Technical Specifications for instrument setpoints
		Addition of high main steamline temperature as an initiation signal for closure of the MSIV's
		Addition of a discussion on containment atmosphere control
o	Section 7.4	Correction of typographical errors
		Correction of table, figure and section references
		Revised the discussion of the RCIC system to discuss the automatic initiation
o	Section 7.5	Correction of typographical errors
		Correction of table, figure and section references
		Addition of reference to PP&L's Environmental Qualification of Class 1E Equipment Program
o	Section 7.6	Correction of typographical errors
		Correction of table, figure and section references
		Moved the discussion of Source Range Monitoring System to Section 7.7

Moved the discussion Rod Sequence Control System Instrumentation and Control to Section 7.7

Moved the discussion of Process Computer System Rod Worth Minimizer Instrumentation and Controls to Section 7.7

Addition of discussion of Primary Containment Radiation Monitoring System

o Section 7.7 — Correction of typographical errors

Correction of table, figure and section references

Revised the discussion on transient monitoring system to include GETARS

Revised the discussion of the MG set speed limiters

Addition of discussion of the ATWS-RPT

Addition of the following discussions previously in other sections

a) Rod Sequence Control System

b) Process Computer System - Rod Worth Monitoring Subsystem

c) Source Range Monitoring Subsystem

d) Rod Block Trip

o Section 8.1 — Revision to the method of isolation for circuit number 57 and 58 of Table 8.1-2

o Section 8.3 — Correction of typographical errors

o Section 9.1 — Update of P&ID figure

o Section 9.2 — Updated the description of the Turbine Building Closed Cooling Water System

Updated description of the Gaseous Radwaste Recombiner Closed Cooling Water System

Updated description of the Reactor Building Closed Cooling Water System

Updated description of the Makeup Demineralizer System

Updated description of the Ultimate Heat Sink

Updated description of the Service Water System

Updated description of the Emergency Service Water System

Updated description of the Raw Water Treatment System

Updated description of the Condensate Storage and Transfer System

Updated description of the Potable Water and sanitary Waste Systems

Update of P&ID figures

- Section 9.3 — Updated descriptions of the Compressed Air Systems

  Updated description of the Equipment and Floor Drainage System

  Updated description of the Standby Liquid Control System

  Updated P&ID figures

- Section 9.4 — Correction of typographical errors

  Correction to the temperature settings for the Diesel Generator Building Ventilation System

  Correction to the temperature settings for the Engineered Safeguard Service Water Pumphouse Ventilation System

Correction to the temperature settings for the Circulatory Water Pumphouse and Water Treatment Building HVAC

Updated description of the Reactor Building Ventilation System including revisions to the temperature settings and the addition of a description of the radiant heaters

Updated description of the Radwaste Building Ventilation System

Updated description of the Control Structure H&V System

Updated P&ID figures

o	Section 9.5	Updated P&ID figures
o	Section 10.4	Correction of typographical errors
		Correction to the maximum turbine bypass flows
		Correction to the turbine trip setting
		Correction to the pressure drop across the demineralizer unit
		Updated description to show a relief valve in the resin transfer line of the Condensate Cleanup System
		Updated description of the main condenser and auxiliary systems
		Updated P&ID figures
o	Section 11.2	Correction of typographical errors
		Correction to Liquid Radwaste System flows on Table 11.2-4
		Correction to the expected and design basis radionuclide activity inventories
		Updated P&ID figures
o	Section 11.3	Correction of typographical errors
		Correction to the annual gaseous releases

Updated description of the offgas System water removal/temperature reduction assembly temperatures

Updated offgas system flows

Updated frequency and quantity of steam discharged to the suppression pool

Updated P&ID figures

- Section 11.4 — Correction of typographical errors

  Updated description of wet radwaste solidification and packaging

  Updated description of radwaste crane

  Updated description of waste container storage and offsite disposal

  Updated description of the filter media time ventilation system

  Updated description of inputs to waste mixing tanks

  Updated P&ID figures

  Updated solid waste management system flow diagram

- Section 11.5 — Correction of typographical errors

  Addition of the description of the containment radiation monitoring systems

  Deletion of the Offgas Post Treatment Radiation Monitoring System

  Updated description of the RHR Service Water Radiation Monitoring System

  Updated description of the Reactor Building Vent Stack Exhaust Sampler

  Updated to the process and effluent radiation monitoring systems table

[illegible]

o Section 12.1	Correction of Health Physics personnel title
	Deletion of the reference to the ALARA Review Committee which has been disbanded
o Section 12.3	Correction of typographical errors
	Deletion of strainers on demineralizers
	Updated description of area radiation monitoring recording devices and readouts and alarms
	Updated description of the area radiation monitoring system
	Update of the mrem to personnel inside the control room for the 30-day period following the LOCA
	Updated P&ID figure
o Section 13.1	Updated technical support personnel resumes
	Updated qualifications of key plant supervisors
o Section 13.2	Updated description of the licensed operator requalification program
o Section 14.2	Correction of typographical errors
	Addition of prerequisite for test P55.1
	Addition of vacuum breaker testing to test P59.1
	Division of test P73.1 into three separate tests
	Updated description of the preoperational test program
	Updated organizational and staffing of the Integrated Startup Group
	Addition of the abstract for test A67.1, Loose Parts Monitoring System

Addition of the abstract for test P59.3, Primary Containment Isolation Valve Timing

Addition of the abstract for test P56.1B, Rod Sequence Control System

Addition of the abstract for test P56.1C, Rod Worth Minimizer System

Revised description of the abstract for test P83.1, Main Steam System

o Section 15.0	Correction of typographical error
o Section 18.1	Revised to reflect the changes previously submitted by letter
o Section 18.2	Revised to reflect the charges previously submitted by letter
o Question 32.27	Revised to reference PP&L's position on Regulatory Guide 1.97
o Question 32.28	Revised to reference PP&L's Environmental Qualification of Class 1E Equipment
o Question 32.44	Revised to reference PP&L's position on Regulatory Guide 1.97
o Question 32.84	Revised to reference the discussion of valve position indication monitors in Section 18.1
o Question 112.10	Revised to reference PP&L's position on IE Bulletin 80-07
o Question 211.8	Revised to reference the discussion of of SRV testing in Section 18.1
o Question 211.68	Revised to reference the submittal of the ODYN analysis
o Question 211.110	Revised to reflect that the figure has been revised
o Question 211.118	Correction of typographical errors
o Question 211.119	Correction of typographical errors
o Question 211.120	Correction of typographical errors

o	Question 211.139	Revised to reference past discussions of the Licensing Review Group
o	Question 211.140	Revised to correct table reference
o	Question 211.175	Revised to reference discussion of SRV tests in Section 18.1
o	Question 211.228	Revised to reference PP&L's Emergency Procedures
o	Question 211.272	Revised to reference ODYN submittal
o	Question 313.2	Revised to references setpoints in Technical Specifications
o	Question 321.7	Revised to reference submittal of the Process Control Program
o	Question 331.11	Revised to reference discussion on accident monitoring instrumentation in Section 18.1
o	Question 331.14	Revised to reference shielding study discussion in Section 18.1
o	Question 331.15	Revised to reference to PASS discussion Section 18.1
o	Question 331.16	Revised to reference Table 13.1-3 for the resume of the Health Physics Supervisor
o	Question 371.6	Revised to reference the submittal of the environmental report for Pond Hill
o	Question 422.6	Revised to correct table reference

Very truly yours,

NW Curtis

N. W. Curtis
Vice President, Engineering and Construction - Nuclear

NRC TRANS P:1

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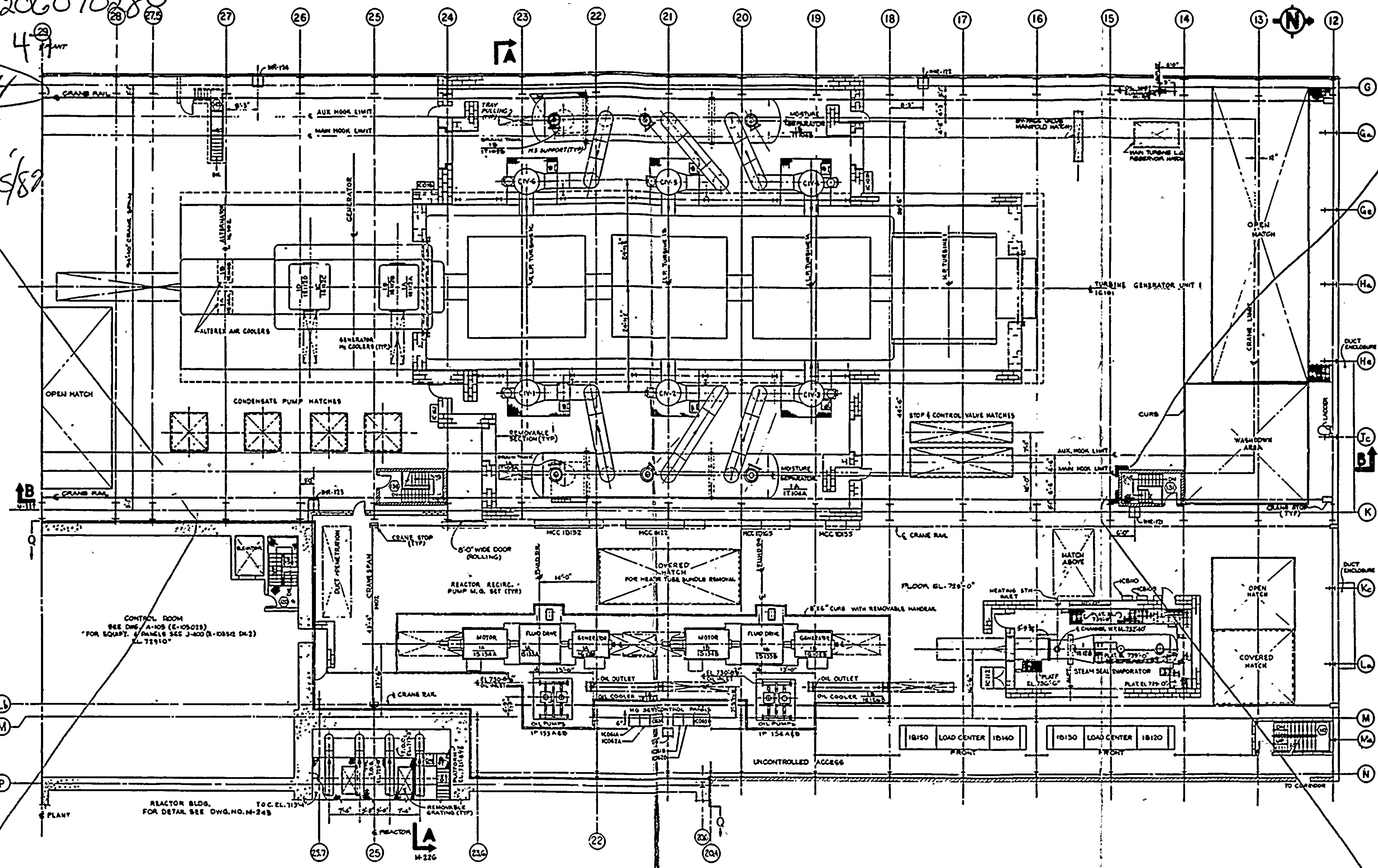
Superseded pages
50-387 8206070280
Amend 47
82/06/04
PS,
6/15/82



REV 8 NOTES:
RELOCATED: REMOVABLE HATCH @ COL. P & COL. 25.
REVISED: INCORP PCR-P1140 & PLATFORM FOR SEAL STEAM EVAP.

REV. 4 NOTES
RELOCATED: MCC 1D155, MCC 1D165, MCC 1B122, MCC 1B132, 1C112, 1C107B
REVISED: LOAD CENTER 1B120, 1B130, 1B140, 1B150
ADDED: 1CB109, 1CB110
DELETED: 1C114, 1C107C

REV. 5 NOTES:
ADDED: REFERENCE DWG FOR CONT. BLDG. OIL MIST ELIM. IK-108 SET COL. 14, 15 & COL. G, "X" LINED SYMBOL TO CONTR. & E.S.
REVISED: M.G. SET CONTR. PANELS RAILING @ COL. M 20 & 22, STAIR 131 & 136

REV 6 NOTES:
RELOCATED: M.G. SET CONTR. PANELS
ADDED: OUTLET EL. 730'-10" TO OIL COOLERS
REVISED: CONC. CURB @ COL 23 & Lb

REV. 7 NOTES:
ADDED: PLATFORM (C-6)

REV. 2 NOTES
RELOCATED: DOOR OPNG. @ COL. 14; MOISTURE SEPARATOR RELIEF VALVES; CORRIDOR ENTRANCE @ COL. 12
REVISED: 8'-0" WIDE ROLLING DR. WAS SLIDING, BLOCK WALLS @ COL. G & K WERE CONC.
ADDED: STAIRCASE (117) @ COL. M-12 & 13; STAIRCASE NOS. LADDER @ COL. Jc-12;
1C107A, B, C AND 1C114, PS. SEAL
DELETED. 1C109 & 1C110, CABLE TRAY SPACE

REV. 3 NOTES
ADDED: CRANE TAG NO. 1H102, VERT LADDER @ COL. K-Kc
FIRE HOSE REELS 1HR-121, 1HR-122, 1HR-123, 1HR-124
DELETED: STAIR @ COL. K-Kc, HATCH AT COL. 23 & Lb.
RELOCATED 1C107B, RAIL GUARD ALONG N COL. 19 & 23
REVISED M.G. SETS CONT. PANELS @ COL 21-M
SHIELDING PLATES @ COL. 16 & 24

REFERENCE DRAWINGS	
BECHTEL NO.	P P & L NO.
M-226	E-106331-SH7
M-227	E-106331-SH8
M-243	E-106333-SH4
M-224	E-106331-SH5
M-225	E-106331-SH6
J-400	E-103512-SH2
A-105	E-105023
J-400	E-103512 SH2

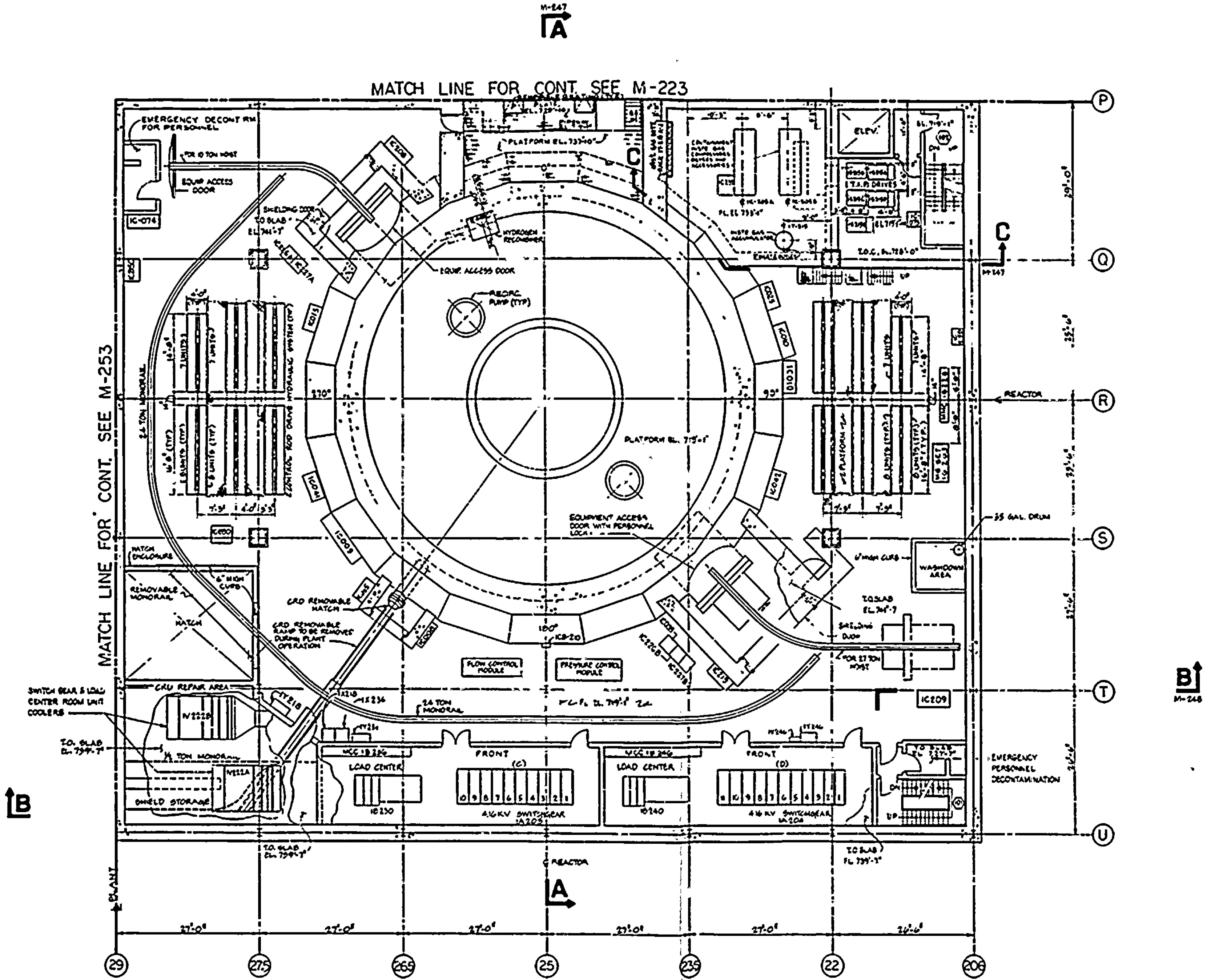
M-223

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

EQUIPMENT LOCATION
TURBINE BLDG. UNIT 1
PLAN OF OPERATING FLOOR
EL. 729'-0"

FIGURE 1.2-4

#8



REV. 2 NOTE

ADDED: 1C009, 1C010, 1C015, 1C025, 1C035, 1C036, 1C215C, 1C215D, 1B226A, 1B226B
CHANGED: 1B236 WAS 1B26; 1B246 WAS 1B226
RE-ARRANGED: CONTROL ROD DRIVE HYDRAULIC SYSTEM.
REVISED: SHIELDING DOORS (D-4 & F-6).
RELOCATED: SW. GEAR & LOAD CENTER ROOM UNIT COOLERS.

REV. 4 NOTE:

RELOCATED: 1C009; 1C010 & 1C015; 1C025; 1C215C; DOUBLE SWING DOORS AT SW GR RMS, MCC 1B236 & MCC 1B246, DOOR (F-7).
DELETED: 1C035; 1C036; 1B226A; 1B226B.
ADDED: 1C041, 1C042; 1C208; 1C219; 1C209, 1C008, 1C226 A & B, 1C035, 1C227 A & B, & 1C036. MAIN STEAM ISOLATION VALVES PANEL (G-4).
CHANGED: MONORAIL ROUTE FOR EQUIP ACCESS DOOR (F-5), INCREASED WASHDOWN AREA (D-3).

REV. 5 NOTE

CHANGED: 1C042 WAS 1C015 (E-4)
1C025 WAS 1C041 (F-4)
1C041 WAS 1C042 (E-6)
1C015 WAS 1C025 (F-6)
WASHDOWN AREA MOVED (D-3)
CRD REMOVABLE HATCH (1C008) SHORTENED (D-6)
MONORAIL ROUTE SHORTENED (D-4)
ADDED: NOS: 1C-073 & 1C-074
REFERENCE DWGS.

REV. 6 NOTE:

ADDED: DWG TO REF BLOCK, [illegible]
REVISED: CRD UNITS,
RELOCATED: CRD UNITS, 1E440, 1C041, 1C215D, 1C010, 1C-073, 1C-074

REV. 7 NOTE:

ADDED: PLATFORM (F-5), PANEL 1C-280 (D-7) MCC-1B229 & MG SET 1G203 (E-3) 1Y218 (C-6), 1CB-210 COL LINE 25
RELOCATED: 1C009 ON LINE S, 1C008 (D-6), 1C-074 AND EMERGENCY DECONT. ROOM FOR PERSONNEL ON COLUMN LINE 29 & P, DOORS ON SWITCHGEAR RM. ON LINE U.
CHANGED: DOOR & WALLS AT CRD REPAIR AREA AND ELEV. OF TO SLAB (SHIELDING DOOR) FR EL. 742'-7" TO EL. 741'-1"
REVISED: EMERGENCY PERSONNEL DECONTAMINATION RM. DOORS AND WALLS ON LINE U., DOORS AT (F-3), (C-3) & (C-2)
ADDED: DOORS AT (F-3), (D-4) & (C-3)

REV 8 NOTE:

ADDED: 1C-280 (D-7), INSTRUMENT GAS BOTTLE RACK 1T-215 AND AT (G-8)
RELOCATED: 1C-073, HATCH AT (G-5)
REVISED: 1E440D TO 1E440C

REFERENCE	DRAWINGS
BECHTEL NO.	P P & L NO.
M-235	E-106551-4
M-247	E-106553-6
M-248	E-106555-4
M-253	E-106934-4
M-25-4	E-106175-4
M-26-3	E-106176-3
M-26-7	E-106176-7
M-27-4	E-106177-4
M-28-4	E-106178-4
M-29-4	E-106179-4
V-25-4	E-106250-4
V-26-3	E-106251-3
V-26-4	E-106251-4
V-27-4	E-106252-4
V-28-4	E-106253-4
V-29-4	E-106254-4

M-243

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

EQUIPMENT LOCATION
REACTOR BUILDING UNIT 1
PLAN OF EL. 719'-1"

FIGURE 1.2-20

EQUIPMENT NUMBERING SYSTEM

SAMPLE: 1P - 101 A
(A)(B) (C) (D)

A. THIS NUMBER IDENTIFIES THE UNIT SERVED BY THE EQUIPMENT AS FOLLOWS:

0 - EQUIPMENT COMMON TO UNITS 1 AND 2
1 - UNIT 1 EQUIPMENT
2 - UNIT 2 EQUIPMENT

B. THIS LETTER CLASSIFIES EQUIPMENT ITEMS BY BASIC TYPES AS FOLLOWS:

A 13.8 & 4.16 KV SWITCHGEAR
B 480 L.C., MCC & DISTRIBUTION PANELS
C CONTROL BOARDS AND LOCAL PANELS
CB COMPONENT BOX
D DC EQUIPMENT
E HEAT EXCHANGERS
F FILTERS AND CLEANING EQUIPMENT (DEMINERALIZERS)
G GENERATORS (TURBINE, DIESEL) AND ASSOCIATED EQUIPMENT
H HOISTS, CRANES
HC HOSE RACK CABINETS
HR HOSE REELS
K AIR COMPRESSORS, CHILLER COMPRESSORS
P PUMPS
R 500 KV & 230 KV EQUIPMENT
S MISCELLANEOUS
T TANKS AND PRESSURE VESSELS
V VENTILATION AND AIR CONDITIONING FANS
X TRANSFORMERS
Y 120V AC POWER DISTRIBUTION PANELS
Z COMPUTER EQUIPMENT

C. THIS 3-DIGIT NUMBER DISTINGUISHES IDENTICAL EQUIPMENT ITEMS IN THE SAME SERVICE FROM OTHERS IN THE BASIC EQUIPMENT GROUP. FOR EXAMPLE, ALL CIRCULATING WATER PUMPS WILL HAVE THE SAME NUMBER TO IDENTIFY THEIR UNIQUE SERVICE AND DISTINGUISH THEM FROM ALL OTHER PUMPS.

NUMBERS ARE ASSIGNED ACCORDING TO PLANT AREAS OR NSSS CLASSIFICATION AS FOLLOWS:

001-099	NSSS LOCAL PANELS & RACKS
101-199	TURBINE BUILDING & CONTROL STRUCTURE
201-299	REACTOR BUILDING
301-399	RADWASTE BUILDING
401-499	PRIMARY CONTAINMENT VESSEL
501-599	MISCELLANEOUS LOCATIONS
601-699	CONTROL ROOM PANELS
801-899	UNCLASSIFIED-AS DEFINED IN EQUIPMENT LIST
901-999	MGM, SGH & CCC (SECURITY BLDGS)

UNIT 2 EQUIPMENT WILL HAVE THE SAME 3-DIGIT NUMBER AS UNIT 1 EQUIPMENT IN SIMILAR SERVICE. EQUIPMENT COMMON TO UNIT 1 AND UNIT 2 WILL BE IN THE UNIT 1 EQUIPMENT LIST AND WILL NOT BE DUPLICATED IN THE UNIT 2 EQUIPMENT LIST.

D. THIS LETTER DISTINGUISHES ONE ITEM FROM OTHER DUPLICATE ITEMS OF EQUIPMENT IN THE SAME SERVICE. FOR EXAMPLE, CIRCULATING WATER PUMPS WILL BE DESIGNATED AS A, B, C OR D TO DISTINGUISH THEM FROM EACH OTHER.

EXCEPT WHERE DESIGN CONSIDERATIONS TAKE PRECEDENCE, E.G., TO RETAIN THE CONTINUITY OF FLOW THROUGH AN "A", "B", OR "C" FEEDWATER HEATER TRAIN, EQUIPMENT ITEMS ARE LETTERED IN ALPHABETICAL ORDER AS FOLLOWS:

1. IN ELEVATION FROM TOP TO BOTTOM
2. FROM EAST TO WEST
3. FROM SOUTH TO NORTH
4. FROM NORTH TO SOUTH (UNIT 1 TURBINE BLDG ONLY)

PIPING NUMBERING SYSTEM

LINE CLASS

PIPE AND VALVE CLASSES ARE DESIGNATED BY A THREE-LETTER CODE. THE FIRST LETTER INDICATES THE PRIMARY VALVE AND FLANGE PRESSURE RATING; THE SECOND LETTER, THE TYPE OF MATERIAL; AND THIRD LETTER, THE CODE TO WHICH THE PIPING IS DESIGNED.

PIPING

FIRST LETTER

PRIMARY PRESSURE RATING (Unless otherwise noted, all ratings are in accordance with ANSI B16.5)

A = SPECIFIC PRESSURE AT SPECIFIC TEMPERATURE
B = 2500#
C = 1500#
D = 900#
E = 600#
F = 400#
G = 300#
H = 150#
J = 125# ANSI B16.1
K = 175# WOG, U.L.
L = 250# ANSI B16.1
X = GRAVITY RATING
Y = GENERAL USE AS DESIGNATED ON PIPING CLASS SHEETS.

SECOND LETTER

MATERIAL

A = ALLOY STEEL
B = CARBON STEEL
C = AUSTENITIC STAINLESS STEEL
D = COPPER, BRASS OR BRONZE
E = C.S.+NICKEL
F = CARBON STEEL-COPPER BEARING
G = CARBON STEEL-LINED
H = CAST IRON
J = CONCRETE
K = VITRIFIED CLAY
L = CARBON STEEL-IMPACT TESTED
M = CAST IRON-HIGH SILICON
N = CARBON STEEL-GALVANIZED
P = CAST IRON-CEMENT LINED
Q = ASBESTOS-CEMENT
R = CARBON STEEL - RHRSW, ESW & SW SYSTEMS

THIRD LETTER

APPLICABLE CODE

A = ASME B&PV CODE, SECT. III, CLASS 1*
B = ASME B&PV CODE, SECT. III, CLASS 2
C = ASME B&PV CODE, SECT. III, CLASS 3
D = POWER PIPING CODE, ANSI B31.1.0
F = NATIONAL FIRE PROTECTION ASSOCIATION CODE
G = UNIFORM PLUMBING CODE
H = ASME B&PV CODE, SECT. I
J = AMERICAN WATER WORKS ASSOCIATION STANDARDS

* PRIMARY PRESSURE RATING IN ACCORDANCE WITH ASME B&PV CODE, SECTION III

SERVICE NUMBER

PIPE LINE IDENTIFICATION NUMBERS (SERVICE NUMBERS) ARE AS FOLLOWS:

	SERVICE
PIPING COMMON TO BOTH UNITS	0-99 AND 300-3999
PIPING FOR UNIT 1	100-199 AND 1000-1999
PIPING FOR UNIT 2	200-299 AND 2000-2999

DUCT NUMBERING SYSTEM

FIRST LETTER GROUP

LV - LOW VELOCITY
HVM - HIGH VELOCITY, MEDIUM PRESSURE
HVH - HIGH VELOCITY - HIGH PRESSURE
HVNP - HIGH VELOCITY - NEGATIVE PRESSURE

NOTE: FIRST LETTER GROUP REPRESENTS DUCT CONSTRUCTION STANDARDS AS DEFINED BY "SHEET METAL AND AIR CONDITIONING CONTRACTORS NATIONAL ASSOCIATION (SMACNA)".

SECOND LETTER GROUP

MATERIAL

GS - GALVANIZED STEEL
SS - STAINLESS STEEL
AL - ALUMINUM

THIRD LETTER GROUP

G - GAS TIGHT

EXAMPLE:
HVM - GS - G
G - GAS TIGHT
GS - GALV STEEL
HVM - HIGH VELOCITY, MEDIUM PRESSURE

VALVE NUMBERING SYSTEMS

A. FOR MANUAL VALVES—BECHTEL SYSTEMS

SAMPLE: (INCLUDES BOX) SEE NOTE 27 & SEE M-101 SH1 G-8

1 - 51 - 019

019 - VALVE SEQUENCE NO. (800 SERIES VALVES ARE FIELD ASSIGNED NUMBERS)
51 - LAST TWO DIGITS OF P&ID NO.
1 - UNIT DESIGNATION:
0: COMMON
1: UNIT 1
2: UNIT 2

B. GE—NSSS SYSTEMS

MANUAL VALVES USE GE NUMBERING SYSTEM AND ARE SHOWN DIRECTLY WITH THE VALVE.

SAMPLE: F002 SEE NOTE 27

C. REMOTE OR SELF-ACTUATED VALVES

SELF-ACTUATED VALVES AND VALVES WITH OPERATORS WILL HAVE THE SAME NUMBER AS THEIR CORRESPONDING ACTUATOR OR OPERATOR, AND WILL FOLLOW THE INSTRUMENT NUMBERING SYSTEM. SEE H-3 THIS SHEET.

D. FOR VALVE NUMBERS OF MANUAL VALVES INCLUDED IN VENDOR SUPPLIED PACKAGES REFER TO APPROPRIATE VENDOR DRAWINGS.

REACTOR PRIMARY CONTAINMENT PENETRATION NUMBERING SYSTEM

(REFERENCE DRAWING 8856-C-279)
SAMPLE: (INCLUDES BOX)

X-040A-3

3 - I.D. NUMBER OF INSTRUMENT LINE PASSING THROUGH PENETRATION (FOR INSTRUMENT PENETRATIONS ONLY)
040A - PENETRATION I.D. NUMBER
X - DENOTES PRIMARY CONTAINMENT PENETRATION

INSTRUMENT NUMBERING SYSTEM

BOP SYSTEMS:

- SEE TABLE ON SHEET 1
- LETTER DIFFERENTIATING SIMILAR LOOPS AND/OR COMPONENTS
- INSTR. AND/OR LOOP NUMBER
- LAST 3 DIGITS OF P&ID NO. (I.E., "07" FOR P&ID M-107)
- UNIT DESIGNATION:
  - 0 = COMMON
  - 1 = UNIT 1
  - 2 = UNIT 2

GE-NSS SYSTEMS (COVERS ONLY INSTRUMENTS LISTED IN MPL. OTHER INSTRUMENTS ARE NUMBERED SAME WAY AS BOP SYSTEMS)

B-21

- SEE TABLE ON SHEET 1
- LETTER DIFFERENTIATING SIMILAR LOOPS AND/OR COMPONENTS
- G.E. NUMBER PER MPL (GE DOCUMENT NO. 22A3580AX)
- UNIT DESIGNATION:
  - 0 = COMMON
  - 1 = UNIT 1
  - 2 = UNIT 2
- G.E. SYSTEM IDENTIFICATION. THIS NUMBER APPEARS IN THE NOTES OF EACH P&ID AND APPLIES TO EACH INSTRUMENT ON GE MPL.

HEATING AND VENTILATING AIR FLOW AND CONTROL DIAGRAM DUCT & INSTRUMENTATION SYMBOLS

SEE SHEET 1 FOR MEASURED VARIABLE (TYPICAL)
AIR OPERATED DAMPER FAIL-OPEN ILLUSTRATED
FO
MOTOR OPERATED DAMPER FAIL-CLOSED ILLUSTRATED
FC
HAND CONTROLLED DAMPER
BACKDRAFT ISOLATION DAMPER

Q-LISTED DUCT
TERMINATION POINT OF Q LISTED DUCT.
LV-GS-G | LV-SS-G DUCTWORK CLASS AND/OR MATERIAL CHANGE
AIR FLOW - NOT DUCTED
SUPPLY AIR
RETURN OR EXHAUST AIR
AIR FLOW THRU LIGHT FIXTURE
CVR CONSTANT VOLUME REGULATORS
BPL BAFFLE PLATE
BALANCING DAMPER (BD)
BACK DRAFT DAMPER (BDD)
FPD FIRE PROTECTION DAMPER
MOD MOTOR OPERATED DAMPER
AOD AIR OPERATED DAMPER
ISOLATION DAMPER
PSD PRESSURE RELIEF DAMPER

CFM	CUBIC FEET PER MINUTE
OA	OUTSIDE AIR
SA	SUPPLY AIR
RA	RETURN AIR
EA	EXHAUST AIR

NOTES

1. ALL HAND SWITCHES SHALL HAVE PILOT LIGHT INDICATION WITH THE SWITCH. NO SYMBOL IS REQUIRED ON THE P&ID TO DENOTE THIS REQUIREMENT.
2. ALL MOTOR-OPERATED VALVES AND SOLENOID-OPERATED PNEUMATIC VALVES SHALL HAVE POSITION SWITCHES. NO SYMBOL IS REQUIRED ON THE P&ID TO DENOTE THIS REQUIREMENT. (SHEET 1, B-6) (ALL AIR OPERATED MODULATING VALVES SHALL BE EQUIPPED WITH POSITIONERS.)
3. RED AND AMBER VALVE POSITION INDICATING LIGHTS WILL BE MOUNTED ON THE SAME PANEL WITH THE HAND SWITCH CONTROLLING THE VALVE. (SHEET 1, B-6)
4. ALL POSITION SWITCHES WILL BE LISTED AND IDENTIFIED IN THE INSTRUMENT INDEX. (SHEET 1, B-6)
5. "X" IS USED TO REPRESENT ANY "SPECIAL" VARIABLES AND MAY BE DEFINED AS REQUIRED. EXAMPLES: XR = MASS FLOW RECORDER, XAH - OVERLOAD ALARM
6. WHEN "Q" IS USED AS A SECOND OR SUCCEEDING LETTER, IT DENOTES AN INTEGRATING MODIFIER. FOR EXAMPLE, "FQI" IS AN INDICATING FLOW INTEGRATOR (OR TOTALIZER). "J" IS FOR SCANNING.
7. SWITCH FUNCTIONS MAY BE FURTHER MODIFIED BY "L" FOR LOW AND "H" FOR HIGH, I.E., PSL, LSH. "HSS" MAY BE USED TO DESIGNATE: "HAND SWITCH SELECT."
8. "TW" DENOTES AN EMPTY THERMOWELL. "TE" DENOTES A THERMOWELL AND ITS ASSOCIATED TEMPERATURE ELEMENT (THERMOCOUPLE, RTD OR FILLED BULB SENSOR).
9. INSTRUMENT IDENTIFICATIONS ARE BASED UPON ISA STANDARD "S5.1".
10. HIGH-HIGH ALARMS WILL BE DESIGNATED "()AHH" AND LOW-LOW ALARMS = "()ALL." FOR EXAMPLE: LAHH DENOTES "HIGH-HIGH LEVEL ALARM."
11. MOTOR AND SOLENOID OPERATED VALVES ARE POWERED FROM THE NORMAL A.C. SUPPLY. WHEN SPECIFICALLY DESIGNATED AS A, B, C, D, I OR II STANDBY A.C. POWER SUPPLY IS REQUIRED. WHEN DESIGNATED A DC, I DC ETC. D.C. POWER SUPPLY IS REQUIRED.
12. "A()" IS USED FOR ALL ANALYTICAL VARIABLES, E.G., O_2, CO_2, pH, ETC.
13. FOR THREE-WAY VALVE NOTATION FOR SOLENOID VALVES, SEE DWG. NO. M-100, SHEET 1 - SOLENOID OPERATED VALVE.
14. FOR FIRE PROTECTION SYMBOLS, SEE DWG. NO. M-122, SHEET 1.
15. ALL PNEUMATIC VALVES INSIDE CONTAINMENT OPERATE FROM THE CONTAINMENT INSTRUMENT GAS SYSTEM (SEE DWG. NO. M-126).
16. ALL INSTRUMENT LINES CONNECTED TO Q-LISTED PROCESS LINES ARE ALSO Q-LISTED. INSTRUMENT LINES CONNECTING TO Q-LISTED DUCTING ARE AS DEFINED ON HVAC INSTRUMENT INSTALLATION DETAILS. Q DESIGNATION FOR INSTRUMENT LINES IS NOT SHOWN ON P&ID'S.
17. KEY SYMBOL IS ON DOCUMENTS THAT CLIENT WISHES TO REVIEW.
18. ROOT VALVES ARE NOT SHOWN ON P&ID - SEE PIPING DETAIL ON SHEET 1 OF P&ID M-123.
19. ELECTRONIC INSTRUMENTATION IS POWERED AS SHOWN ON INSTRUMENT LOOP DIAGRAMS, EXCEPT NSSS INSTRUMENTATION WHICH IS SHOWN ON ELECTRICAL SCHEMATIC DIAGRAMS.
20. "()Z" IS A FINAL CONTROL ELEMENT, I.E. YZ, HEATER ELEMENT, ETC.
21. "I()" IS USED FOR A CURRENT DEVICE, I.E. ISH - HIGH CURRENT SWITCH.
22.
23. FOR POWER SUPPLY ASSIGNMENTS FOR THE BALANCE OF PLANT ANALOG LOOPS REFER TO DRAWING 8856-J-649.
24. P&ID'S ARE NUMBERED AS FOLLOWS, M-1XX FOR UNIT 1 & COMMON SYSTEMS AND M-21XX FOR UNIT 2 SYSTEMS. HVAC CONTROL DIAGRAMS ARE NUMBERED VC-1XX OR VC 21XX. UNIT 2 DRAWINGS ARE LISTED IN THE DRAWING CONTROL.
25. REFER TO J-157 FOR TABULATION OF EXCESS FLOW CHECK VALVES AND ASSOCIATED COMPONENTS.
26. SWITCH HS-15700B AND INDICATOR XI-15700B ARE FOR TEST AND ALARM INDICATION OF DIV. I. (SH.1, B-8)
27. MANUAL VALVES NUMBERED BY THIS SYSTEM, WHICH ARE LOCATED IN "Q" LINES, ARE Q. FURTHER "Q" DESIGNATION OF THE NUMBER IS NOT NEEDED.
28. REFER TO J-600 FOR SPECIFICATION OF INSTRUMENT VALVES, FIVE-VALVE MANIFOLDS, TUBING AND TUBE FITTINGS.
29. DERIVED ALARMS ARE SHOWN ON LOGIC DIAGRAMS AND NOT ON THE P&ID'S UNLESS NOTED OTHERWISE ON THE P&ID.
30. FOR TEST POINTS (PP, FP, AP) ON Q LISTED LINES EXCEPT FOR THOSE INSTALLED REMOTE (SEE NOTE 7 J-0020), THE ADAPTER, NIPPLE, PIPE, PLUG CAP, DOWNSTREAM OF THE ROOT VALVE AND THE ADAPTER WELD TO THE ROOT VALVE ARE NOT Q LISTED.

HVAC CONTROL · DIAGRAMS (SEE NOTE 24)	
VC-173	CIRCULATING WATER PUMPHOUSE
VC-174	TURBINE BUILDING
VC-175	REACTOR BUILDING ZONE III, SHEETS 1,2 & 3
VC-176	REACTOR BUILDING ZONES I
VC-177	DRYWELL
VC-178	CONTROL STRUCTURE SHEET 1 & 2
VC-179	RADWASTE BUILDING
VC-180	CANCELLED
VC-181	MISCELLANEOUS BUILDINGS
VC-182	DIESEL GENERATOR BLDG., ESSW PUMPHOUSE

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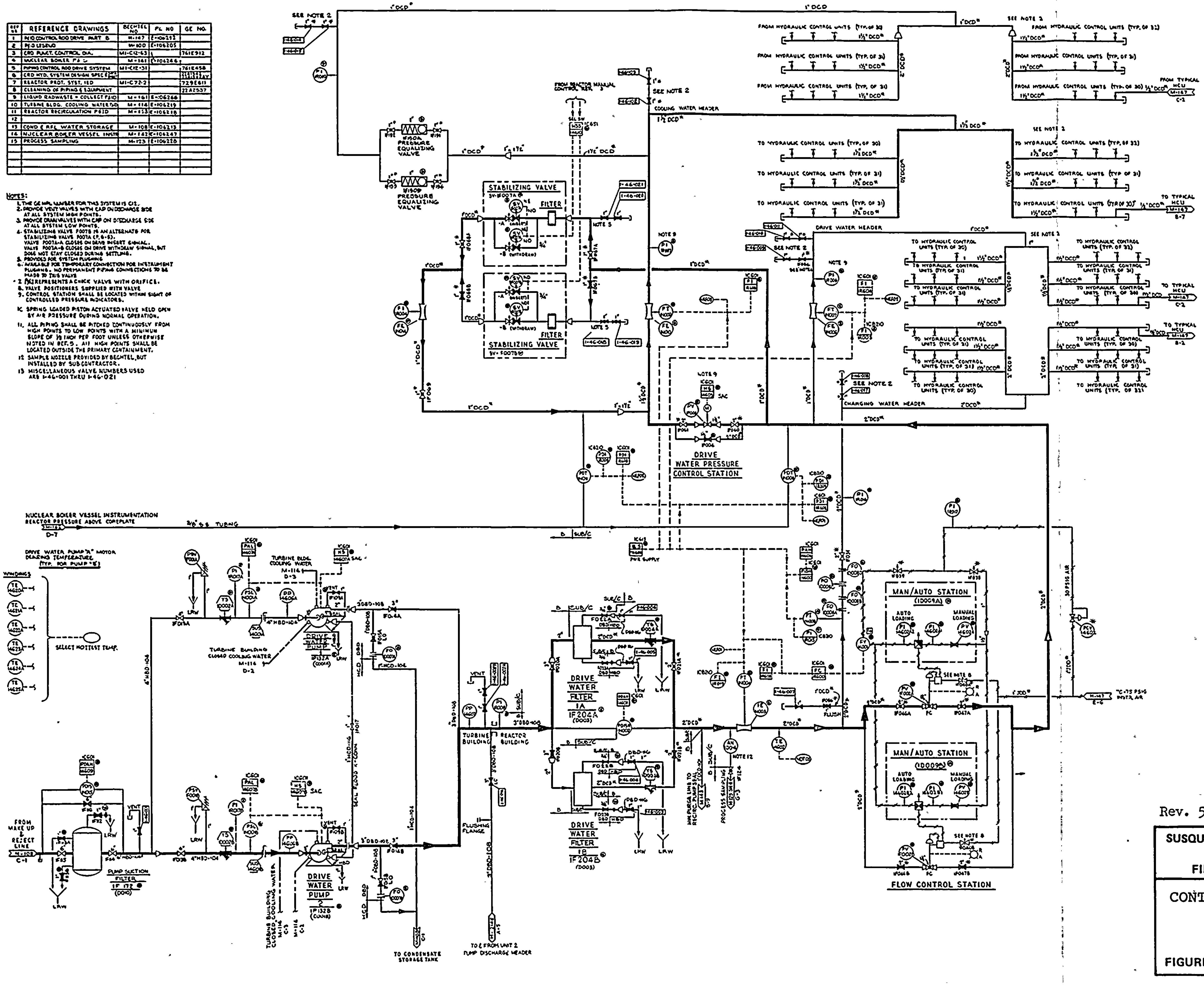
P&ID
LEGEND AND SYMBOLS

FIGURE 1.8-2b

REF NO	REFERENCE DRAWINGS	BECHTEL NO	PL NO	GE NO.
1	P&ID CONTROL ROD DRIVE PART B	M-147	E-106292	
2	P&ID LEGEND	M-100	E-106205	
3	CRD FUNCT. CONTROL DIA.	MI-C12-63		761E912
4	NUCLEAR BOILER P&ID	M-141	E-106246	
5	PIPING CONTROL ROD DRIVE SYSTEM	MI-C12-31		761E458
6	CRD HYD. SYSTEM DESIGN SPEC			[illegible]
7	REACTOR PROT. SYST. IED	MI-C72-2		729E611
8	CLEANING OF PIPING & EQUIPMENT			22A2537
9	LIQUID RADWASTE - COLLECT P&ID	M-161	E-106266	
10	TURBINE BLDG. COOLING WATER P&ID	M-114	E-106219	
11	REACTOR RECIRCULATION P&ID	M-143	E-106218	
12				
13	COND & RFL WATER STORAGE	M-108	E-106213	
14	NUCLEAR BOILER VESSEL INSTR	M-142	E-106247	
15	PROCESS SAMPLING	M-123	E-106228	

NOTES:

1. THE GE MPL NUMBER FOR THIS SYSTEM IS C12.
2. PROVIDE VENT VALVES WITH CAP ON DISCHARGE SIDE AT ALL SYSTEM HIGH POINTS.
3. PROVIDE DRAIN VALVES WITH CAP ON DISCHARGE SIDE AT ALL SYSTEM LOW POINTS.
4. STABILIZING VALVE F007B IS AN ALTERNATE FOR STABILIZING VALVE F007A (P, 6-6). VALVE F007A-A CLOSES ON DRIVE INSERT SIGNAL. VALVE F007A-B CLOSES ON DRIVE WITHDRAW SIGNAL, BUT DOES NOT STAY CLOSED DURING SETTLING.
5. PROVIDED FOR SYSTEM FLUSHING
6. AVAILABLE FOR TEMPORARY CONNECTION FOR INSTRUMENT FLUSHING. NO PERMANENT PIPING CONNECTIONS TO BE MADE TO THIS VALVE
7. REPRESENTS A CHECK VALVE WITH ORIFICE.
8. VALVE POSITIONERS SUPPLIED WITH VALVE
9. CONTROL STATION SHALL BE LOCATED WITHIN SIGHT OF CONTROLLED PRESSURE INDICATORS.
10. SPRING LOADED PISTON ACTUATED VALVE HELD OPEN BY AIR PRESSURE DURING NORMAL OPERATION.
11. ALL PIPING SHALL BE PITCHED CONTINUOUSLY FROM HIGH POINTS TO LOW POINTS WITH A MINIMUM SLOPE OF 1/8 INCH PER FOOT UNLESS OTHERWISE NOTED IN REF. 5. ALL HIGH POINTS SHALL BE LOCATED OUTSIDE THE PRIMARY CONTAINMENT.
12. SAMPLE NOZZLE PROVIDED BY BECHTEL, BUT INSTALLED BY SUBCONTRACTOR.
13. MISCELLANEOUS VALVE NUMBERS USED ARE 1-46-001 THRU 1-46-021



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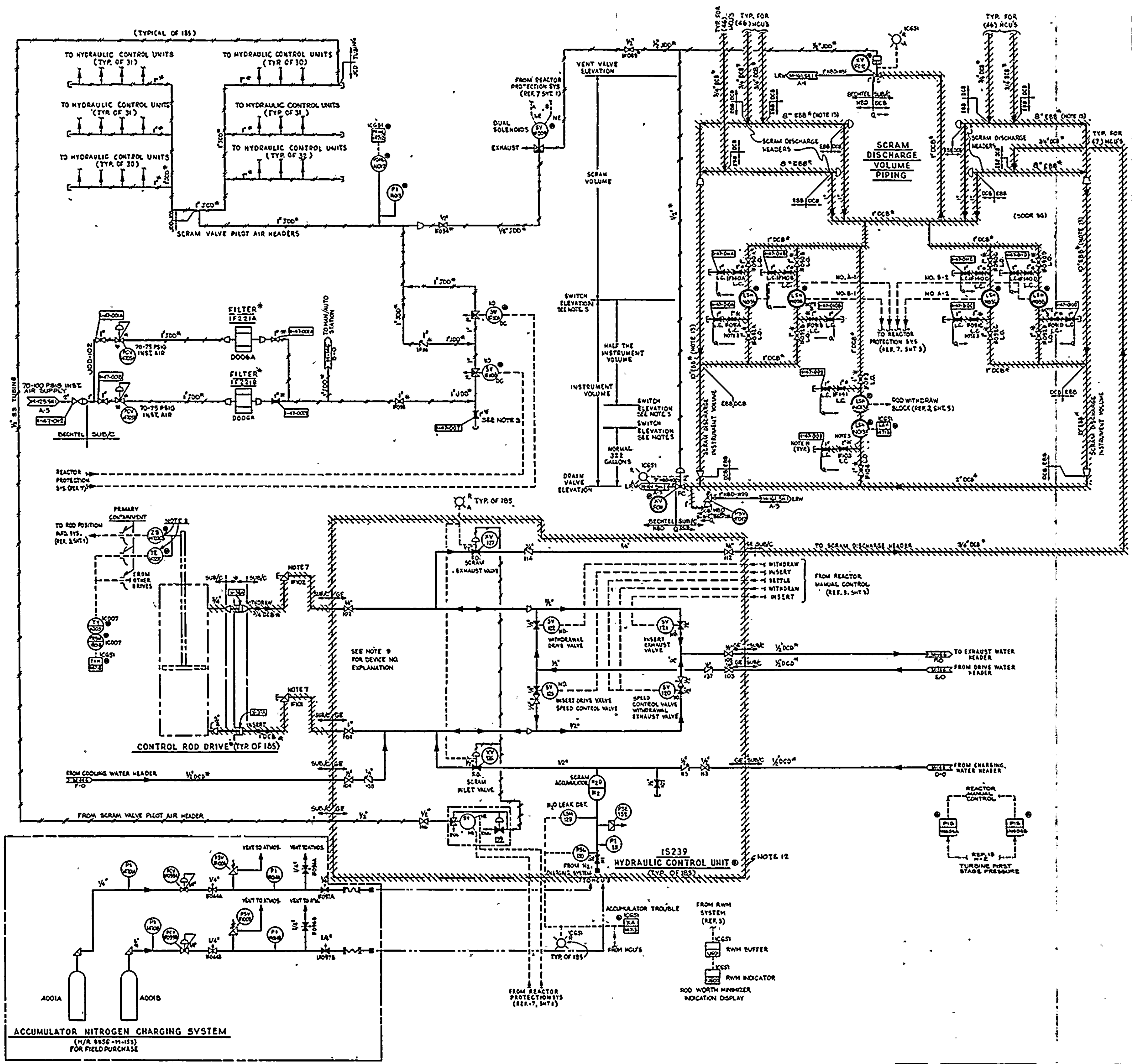
CONTROL ROD DRIVE-PART A

FIGURE 4.6-5a



REF.	REFERENCE DRAWINGS	BECHTEL NO	PL NO	GE NO
1	P&ID CONTROL ROD DRIVE HYDRAULIC PART A	M146	E-106231	
2	P&ID LEGEND	M100	E-106205	
3	CRD FUNCT. CONTROL DIAGRAM	M1-C12-63		761E912
4	NUCLEAR BOILER	M-141	E-106248	
5	CRD HYD. SYSTEM PROCESS DIAG.	M1-C12-4		921D896 921D896A
6	CRD HYD. SYS. DESIGN SPEC DATA SHT			22A1342 22A1342AY
7	REACTOR PROT. SYS. IED	M1-C72-2		729E611AC
8	FIELD CLEANING AND CLEANLINESS			22A2537
9	LIQUID RADWASTE-COLLECTION	M-161	E-106266	
10.	PIPING CONTROL ROD DRIVE SYSTEM	M1-C12-31		761E458
11	COMPRESSED AIR SHEET 1	M-125	E-106230	
12	MAKE-UP DEMINERALIZER	M-118	E-106223	
13	MSIV-LCS, P&ID	M-139	E-106244	

NOTES:

1. THE GE UPL NUMBER FOR THIS SYSTEM IS C12.
2. PROVIDE VENT VALVES WITH SCRD. CAP AT ALL SYSTEM HIGH POINTS.
3. PROVIDE DRAIN VALVES WITH SCRD CAP AT ALL SYSTEM LOW POINTS.
4. ALL PIPING SHALL BE PITCHED CONTINUOUSLY FROM HIGH POINTS TO LOW POINTS WITH A MINIMUM SLOPE OF 1/8" PER FOOT, UNLESS OTHERWISE NOTED IN REF. 10 ALL HIGH POINTS SHALL BE LOCATED OUTSIDE THE PRIMARY CONTAINMENT.
5. REFER TO LEVEL INSTRUMENT DIAGRAMS FOR EXACT SWITCH ELEVATION.
6. ACRS IMPLEMENTATION LATER.
7. BLOCK TYPE NEEDLE VENT VALVE (ANGLE PATTERN)
8. CONTROL ROD 2S & TE TYPICAL OF 185 EACH
9. DEVICE NUMBERS SHOWN INSIDE HCU ARE THOSE SHOWN ON GE P&D AND ARE LISTED IN THE REMARK COLUMN OF THE INSTRUMENT INDEX FOR CROSS REFERENCE TO BECHTEL TAG NO'S.
10. MISCELLANEOUS VALVE NUMBERS USED ARE 1-47-001 THRU 1-47-012.
11. VALVES 1-47-8, 9, 10A TO D, 11A TO D PURCHASED BY BECHTEL, INSTALLED BY NIBCO.
12. HYDRAULIC CONTROL UNIT VALVE NUMBERS SHALL HAVE A SUFFIX OF -1XXXX, THE SUFFIX INDICATING THE HCU LOCATION OF THE VALVE.
13. FOR INSERVICE INSPECTION IDENTIFICATION PURPOSES, LINE NO. VBB-402 IS ASSIGNED TO THIS VENDOR SUPPLIED PIPING (SEE DRAWINGS 8856-154-VBB-102-1 AND 8856-154-VBB-102-2)

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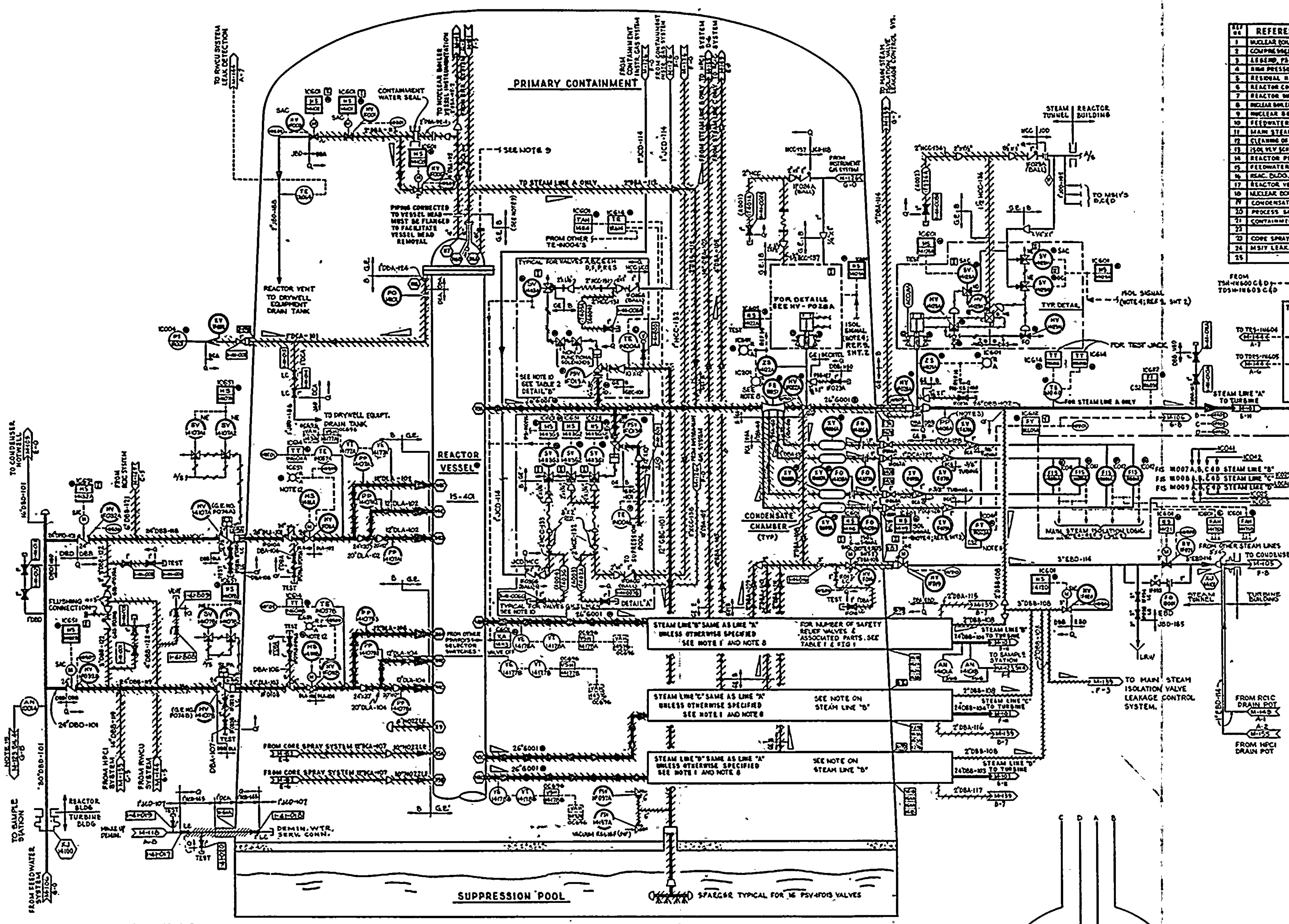
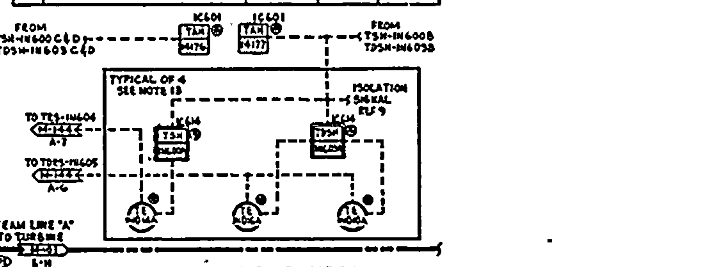
P&ID
CONTROL ROD DRIVE
PART B

FIGURE 4.6-5b





MAIN STEAM LINE LEAK DETECTION
REF. 16

REF NO.	REFERENCE DRAWINGS	BECHTEL NO.	PL. NO.	GE. NO.
1	NUCLEAR BOILER PROCESS DIAG. & DATA	M1-B21-21,22		[illegible]
2	COMPRESSED AIR P&ID	M-125	E-106230	
3	LEGEND, P&ID	M-100	E-106205	
4	HIGH PRESSURE COOLANT INJECT P&ID	M-155	E-106260	
5	RESIDUAL HEAT REMOVAL P&ID	M-151	E-106256	
6	REACTOR CORE ISOLATION COOL P&ID	M-149	E-106254	
7	REACTOR WATER CLEANUP P&ID	M-144	E-106249	
8	NUCLEAR BOILER VESSEL INSTRUMENT P&ID	M-142	E-106247	
9	NUCLEAR BOILER FCD			761E279AF
10	FEEDWATER P&ID	M-106	E-106211	
11	MAIN STEAM P&ID	M-101	E-106206	
12	CLEANING OF PIPING AND EQUIPMENT			22A2537
13	ISOL VLV SCHEMATIC CONTR DIAGRAM	M1-B21-4		732E150
14	REACTOR PROT SYSTEM	M1-C71-2		729E6WAE
15	FEEDWATER CONTROL SYSTEM IED	M1-C32-3		729E670W
16	REAC. BLDG. LINER PLATE PENET.	C-277 279		
17	REACTOR VESSEL	BH-131-B		919D900AE
18	NUCLEAR BOILER LEAK DET DESIGN SPEC	M1-SEE 22-127		22A4440
19	CONDENSATE P&ID	M-105	E-106215	
20	PROCESS SAMPLING P&ID	M-123	E-106228	
21	CONTAINMENT INSTR GAS P&ID	M-126	E-106231	
22				
23	CORE SPRAY P&ID	M-152	E-106257	
24	MSIV LEAKAGE CONTROL SYS P&ID	M-139	E-106244	
25				

NOTES:

1. STEAM LINES & ASSOCIATED EQUIPMENT ENCLOSED IN BOXES SHALL HAVE PART NRS. CORRESPONDING TO THEIR RESPECTIVE LINE LETTER UNLESS OTHERWISE NOTED.
   EXAMPLE: XXXXXB IS ON LINE B, YYYYYC IS ON LINE C.
2. THE G.E. MPL NUMBER FOR THIS SYSTEM IS B21.
3. LOCATE PRESSURE TEST POINTS PP14'38 A,B,C,D AS CLOSE AS POSSIBLE TO VALVES 1F028 A,B,C,D.
4. MAIN STEAM ISOLATION VALVE ISOLATION IS INITIATED BY ONE OF THE FOLLOWING CONDITIONS:
   A. REACTOR LOW, LOW WATER LEVEL.
   B. MAIN STEAM LINE HIGH RADIATION.
   C. MAIN STEAM LINE HIGH FLOW.
   D. LOW TURBINE INLET PRESSURE.
   E. HIGH STEAM LINE TUNNEL TEMPERATURE.
   F. LOW CONDENSER VACUUM.
5. AUTO DEPRESSURIZATION FUNCTION (ADF) IS INITIATED BY THE FOLLOWING SIMULTANEOUS SIGNALS:
   A. DRYWELL HIGH PRESSURE.
   B. REACTOR WATER LOW LEVEL.
   C. RHR OR CORE SPRAY PUMP MOTOR RUNNING.
6. MISCELLANEOUS VALVE NUMBERS USED ARE 1-41-001 THRU 1-41-020, AND 1-41-800 THRU 1-41-817.
7. DELETED.
8. FLOW ELEMENT FOR MAIN STEAM LINES B,C,D ARE 1N052, 1N053, AND 1N054 RESPECTIVELY.
9. G.E. SUPPLIES MATING FLANGE, EITHER NOZZLE 6A OR 6B MAY BE CONNECTED TO THE HEAD SPRAY LINE.
10. REFER TO SMALL PIPE SCOPING DRAWING FOR SRV AIR SUPPLY LINE DETAILS.
11. FT C32-1N003A&B ARE ON 1C041, FT C32-1N003D IS ON 1C015. FT C32-1N003C IS LOCALLY MOUNTED.
12. THE VALVE CONTROL SWITCH IS LOCATED ON THE VALVE OPERATOR.
13. THE SECOND ELEMENT IS ONLY USED ON STEAM LINE "A" AND IS A SPARE ON LINE B, C & D.
14. SEE DESIGN SPEC. FOR NUCLEAR PIPING 8856-M-175 SECTION 4.3, FOR ADDITIONAL DYNAMIC SEISMIC-SYSTEM ANALYSIS REQUIREMENTS ON LINES DBB-101, 102, 103, 104 & 105.
15. INSTALL ROOT VALVES (TWO 1" CCD-6B PER ANALYSIS NOZZLE) ON AN-B21-1D014 TO COMPLY WITH GE-NED SPEC.# 22A2708.

TABLE 2 CONTROL SWITCH ARRANGEMENT FOR DETAIL "D"

TYPICAL FOR 1F013A, B & C

PSV-1F013	1C201 CONTROL SWITCH	1C201 TRANSFER SWITCH	1C601 CONTROL SWITCH	SOLENOID VALVE
A	HS-141113A2	HSS-1F013A	HS-141113A1	SV-141113A
B	HS-141113B2	HSS-1F013B	HS-141113B1	SV-141113B
C	HS-141113C2	HSS-1F014A	HS-141113C1	SV-141113C

TYPICAL FOR 1F013 D,E,F,H,P,R & S

PSV-1F013	1C601 CONTROL SWITCH			SOLENOID VALVE
D	HS-141113D			SV-141113D
E	HS-141113E			SV-141113E
F	HS-141113F			SV-141113F
H	HS-141113H			SV-141113H
P	HS-141113P			SV-141113P
R	HS-141113R			SV-141113R
S	HS-141113S			SV-141113S

TABLE 1. SAFETY/RELIEF VALVE LOCATION, SUFFIX ASSIGNMENT & ASSOCIATED EQUIPMENT

		A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S
SAFETY / RELIEF VALVES 1F013		A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S
RELIEF VALVE SPRING SET PRESSURE (PSIG)		1175	1146	1175	1175	1146	1185	1205	1175	1195	1205	1195	1205	1195	1185	1185	1185
ACCUMULATORS	(ADS) 1T402(X) 150 PSIG							A		B	C	D	E	F			
	1T403(X) 90 PSIG	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S
TEMPERATURE ELEMENT 1N004		A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S
CHECK VALVES	1F036(X)	A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S
	1F040(X)(ADS)							G		J	K	L	M	N			
VACUUM BREAKER 1F037(X) & 1F137(X)		A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S
PRESSURE SWITCH 1N022 (REF. 8, F-2)		A	B	C	D	E	F	G	H	J	K	L	M	N	P	R	S
PRESSURE SET POINT OF 1N022 (PSIG)		1086	1076	1086	1086	1076	1096	1116	1086	1106	1116	1106	1116	1106	1096	1096	1096
SOLENOID VALVES 4113 ASSIGN. FUNCTION	ADS (A)							A		A	A	A	A	A			
	ADS (A)							A		A	A	A	A	A			
	141113X3 PRESS. RELIEF (Y)	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y	Y
VENT VALVE NO. ON S/R VALVE 1F013 DISCHARGE PIPE		1-41-800	1-41-810	1-41-803	1-41-813	1-41-802	1-41-814	1-41-801	1-41-815	1-41-807	1-41-816	1-41-811	1-41-806	1-41-817	1-41-805	1-41-812	1-41-804

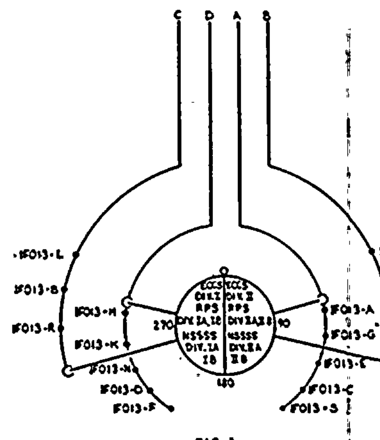


FIG. 1.
SAFETY/RELIEF VALVE LOCATION AND SEPARATION ASSIGNMENT OF VESSEL INSTRUMENTATION

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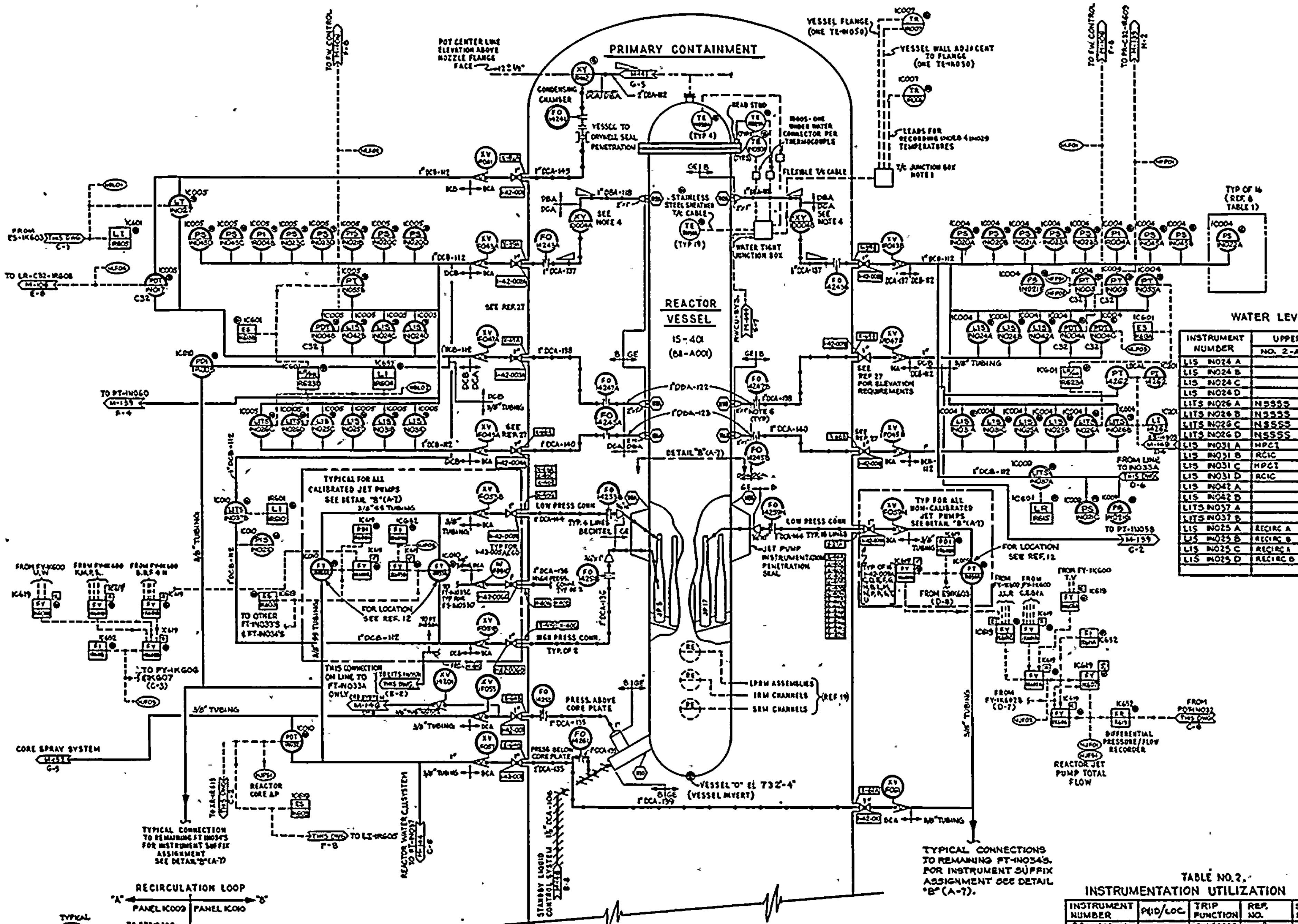
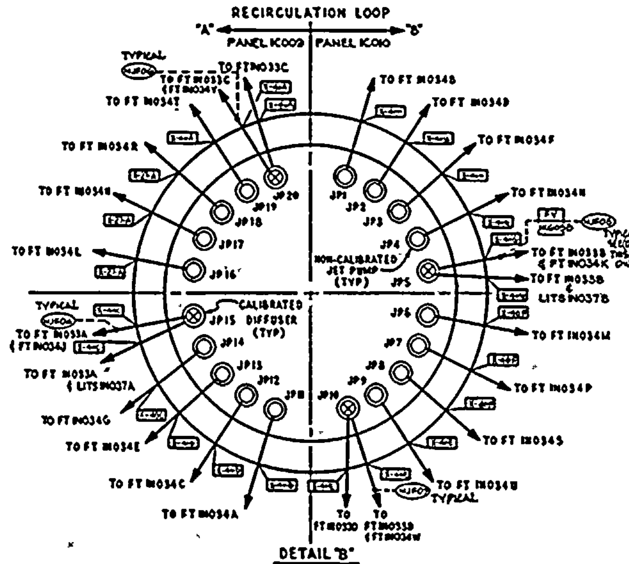
P&ID
NUCLEAR BOILER

FIGURE 5.1-3a





DETAIL "B"
JET PUMP & TRANSMITTER MATCHING

REF NO.	REFERENCE DRAWINGS	BECHTEL NO.	PL NO.	GE NO
1	CORE SPRAY	M-152	E-106257	
2	REACTOR RECIRCULATION	M-143	E-106248	
3	LEGEND	M-400	E-106203	
4	HIGH PRESSURE COOLANT INJECTION	M-155	E-106260	
5	RESIDUAL HEAT REMOVAL	M-151	E-106256	
6	REACTOR CORE ISOLATION COOLING	M-149	E-106254	-
7	REACTOR WATER CLEANUP	M-144	E-106230	
8	NUCLEAR BOILER	M-141	E-106246	
9	NUCLEAR BOILER FCD	MI-B21-92		761E279AE
10	FEEDWATER	M-106	E-106211	
11	MAIN STEAM	M-101	E-106206	
12	NSSS LOCAL PANELS SPECIFICATION	MI-H23		22A4016
13	ISOLATION VALVE SCHEMATIC CONTROL DIAGRAM	MI-B21-4		732E150
14	REACTOR PROTECTION SYSTEM IED	[illegible]		729E611AE
15				
16	CONTROL ROD DRIVE HYDRAULICS-PART B	M-147	E-106252	-
17	REACTOR RECIRC. SYSTEM FCD	[illegible]		729E625AE
18	FEEDWATER CONTROL SYSTEM IED	MI-C32-3		729E629AE
19	NEUTRON MONITORING SYSTEM IED	[illegible]		729E615A
20	STANDBY LIQUID CONTROL SYSTEM	M-148	E-106253	
21	CORE SPRAY SYSTEM FCD	MI-E21-3		729E613AE
22	HIGH PRESSURE COOLANT INJECTION FCD	[illegible]		729E627
23	RESIDUAL HEAT REMOVAL FCD	[illegible]		729E630
24	REACTOR CORE ISOLATION COOLING FCD	[illegible]		729E622AE
25	MSIV LEAKAGE CONTROL SYSTEM PC10	M-139	E106244	
26	NUCLEAR BOILER SYSTEM DESIGN SPEC. & DATA SHEETS			[illegible]
27	FEEDWATER DYNAMIC ANALYSIS DATA	MI-A50-2		158B8896D

TABLE NO.1
WATER LEVEL INSTRUMENT CONTACT UTILIZATION

INSTRUMENT NUMBER	UPPER CONTACTS NO. 2-A	UPPER CONTACTS NO. 2-B	TRIP NO.	LOWER CONTACTS NO. 1-A	LOWER CONTACTS NO 1-B	TRIP NO.	DIVISION	P&ID LOC	REF. NO.	SHEET NO. FOR REF.
LIS N024 A		RCIC	8	REACTOR PROTECTION SYSTEM AND NSSSS		3	I	E-3	9	1
LIS N024 B		HPCI	8			3	I	E-3	9	1
LIS N024 C		RCIC	8			3	II	E-6	9	1
LIS N024 D		HPCI	8			3	II	E-6	9	1
LITS N026 A	NSSSS		2			2	I	E-2	9	1,2
LITS N026 B	NSSSS		2			2	I	E-2	9	1
LITS N026 C	NSSSS		2			2	II	E-6	9	1
LITS N026 D	NSSSS		2			2	II	E-6	9	1
LIS N031 A	HPCI	RCIC	2	ADS (A)	RHR(A,C) CS(A)	1	I	E-3	9	3
LIS N031 B	RCIC	HPCI	2	ADS (B)	RHR(B,D) CS(B)	1	II	E-6	9	3
LIS N031 C	HPCI	RCIC	2	ADS (A)	RHR(A,C) CS(A)	1	I	E-3	9	3
LIS N031 D	RCIC	HPCI	2	ADS (B)	RHR(B,D) CS(B)	1	II	E-6	9	3
LIS N042 A				ADS (A)		3	I	E-2	9	3
LIS N042 B				ADS (B)		3	II	E-6	9	3
LITS N037 A					CONT. SPRAY	0	I	E-2		
LITS N037 B					CONT. SPRAY	0	II	E-6		
LIS N025 A	RECIRC A		2					E-3	17	1
LIS N025 B	RECIRC B		2					E-3	17	1
LIS N025 C	RECIRC A		2					E-6	17	1
LIS N025 D	RECIRC B		2					E-6	17	1

NOTES:

1. EACH T/C JUNCTION BOX IS TO HAVE ITS OWN SET OF TERMINALS (F-4).
2. AN EXPANSION LEG SHALL BE PROVIDED IN INSTRUMENT SENSING LINE BETWEEN XY 1002 AND THE WATER-TIGHT PENETRATION SEAL THROUGH BOTTOM OF REACTOR WELL. THE EXPANSION LEG & PIPING INSTALLATION SHALL BE DESIGNED TO ALLOW FOR MAXIMUM CHANGE OF VESSEL LENGTH WITH TEMPERATURE TO AVOID OVERSTRESSING THE PIPING OR THE SEAL OR DAMAGING THE INSULATION AROUND THE VESSEL. (G-5)
3. THE GE MPL NUMBER FOR THIS SYSTEM IS B21, UNLESS OTHERWISE NOTED.
4. CENTERLINE ELEVATION OF CONDENSING CHAMBER TO BE 1½" ABOVE CENTERLINE OF PRESSURE VESSEL NOZZLES N12 A & B.
5. THE FLOW ORIFICE IN EACH INSTRUMENT LINE SHALL BE LOCATED AS CLOSE AS POSSIBLE TO THE POINT AT WHICH THE INSTRUMENT LINE CONNECTS TO THE REACTOR COOLANT PRESSURE BOUNDRY. (E-4 TYP.)
6. MISCELLANEOUS VALVE NUMBERS USED ARE 1-42-001 THRU 1-42-010.

ALL INSTRUMENT LINES ON THIS P&I.D. ARE Q-LISTED, UNLESS OTHERWISED NOTED.

TABLE NO.2
INSTRUMENTATION UTILIZATION

INSTRUMENT NUMBER	P&ID/LOC.	TRIP FUNCTION	REF. NO.	SHEET NO. FOR REF.	DIVISION
PS N020 A,B	F-3	LOW PRESS	9	2	IA,IB
PS N020 C,D	F-6	INTERLOCK	9	2	IIA,IIB
PS N021 A,C	F-3 E-2	CS / RHR	21 / 23	1/2	I
PIS N021 B,D	F-6 D-6	CS/RHR/RECIRC	21 / 23	1/2	II
PS N023 A,B	F-2	RPS	14	3	IA,IB
PS N023 C,D	F-6	RPS	14	3	IIA,IIB
PS N045 A	F-2	Recir Pump A	17		
PS N045 B	F-2	Recir Pump B	17		
PS N045 C	F-6	Recir Pump A	17		
PS N045 D	F-6	Recir Pump B	17		
PS N015 A,B	[illegible]	NSSSS	9	2	IA,IB
PS N015 C,D	[illegible]	NSSSS	9	2	IIA,IIB
FIS N006 A,B	M-141/D-2	NSSSS	9	2	IA,IB
FIS N006 C,D	M-141/D-2	NSSSS	9	2	IIA,IIB
FIS N007 A,B	M-141/D-2	NSSSS	9	2	IA,IB
FIS N007 C,D	M-141/D-2	NSSSS	9	2	IIA,IIB
FIS N008 A,B	M-141/D-2	NSSSS	9	2	IA,IB
FIS N008 C,D	M-141/D-2	NSSSS	9	2	IIA,IIB
FIS N009 A,B	M-141/D-2	NSSSS	9	2	IA,IB
FIS N009 C,D	M-141/D-2	NSSSS	9	2	IIA,IIB
PSH N056 A,B	M-139/G-2	NSSSS	9	2	IA,IB
PSH N056 C,D	M-139/G-2	NSSSS	9	2	IIA,IIB
PT N055 A	F-2	VESSEL PRES	9	1	I
PT N055 B	E-6	VESSEL PRES	9	1	II
LT N027	F-7	SHUTDOWN LEVEL			
PDT N032	C-6	CORE PLATE			
PS N022 (ATMOS)	F-2	[illegible]			I
PS N021 E,G	F-3, E-2	RECIRC DISCHARGE			

TABLE NO.3
REACTOR PRESSURE VESSEL WATER LEVEL CORRELATION

REFERENCE	(COLD VESSEL) INCHES ABOVE VESSEL ZERO	DESCRIPTION OF TRIPS	INSTRUMENT(S) PROVIDING TRIP	REACTOR VESSEL LEVEL IDENTITY (REF 26)	FUEL ZONE LR-R615 LI-R610	WIDE RANGE LR/PR-R623A,B LI-R604	NARROW RANGE C32 LR-R608 C32 LI-R606A,B,C	UPSET C32 LR-R608	SHUTDOWN LI-R605
TOP OF HEAD FLANGE	1190 MAX								400"
STEAM LINE NOZZLE N3A,B,C,D	653.5							180"	
INSTRUMENT LINE NOZZLE N12A,B	590					+60"	+60"	+60"	
		TRIP RCIC & HPCI TURBINE, CLOSE MAIN TURBINE STOP VALVES, TRIP FEED PUMPS	TABLE I, REF 18	8			+54"	+54"	
		HIGH LEVEL ALARM	REF 18	7			+39"		
		NORMAL WATER LEVEL	REF 18	5A					
		LOW LEVEL ALARM	REF 18	4			+30"		
		SCRAM & CLOSE RHR SHUTDOWN COOLING ISOLATION VALVES; AUTO DEPRESSURIZATION PERMISSIVE	TABLE I	3			+12.5	+12.5	
FEEDWATER SYSTEM & REACTOR PROTECTION SYSTEM. INSTRUMENT ZERO EXCEPT FUEL ZONE. BOTTOM OF DRYER SKIRT.	527.5					0"	0"	0"	0"
INSTRUMENT LINE NOZZLE N11A,B	517.0	INITIATE RCIC & HPCI, TRIP RECIRCULATION PUMPS; CLOSE PRIMARY SYSTEM ISOLATION VALVES EXCEPT RHR SHUTDOWN COOLING.	TABLE I	2		-38"			
		INITIATE RHR & CORE SPRAY SYSTEMS, CONTRIBUTE TO AUTO DEPRESSURIZATION SYSTEM, START STANDBY DIESEL.	TABLE I	1	+50	-129"			
INSTRUMENT LINE NOZZLE N16A,B	366.0					-150			
TOP OF ACTIVE FUEL AND FUEL ZONE WATER LEVEL ZERO	366				0"				
JET PUMP INSTRUMENT NOZZLE N8A,B	192.0	CONTAINMENT SPRAY PERMISSIVE	TABLE I	0	-41"				
JET PUMP DIFFUSER TAIL	143.0				-150"				

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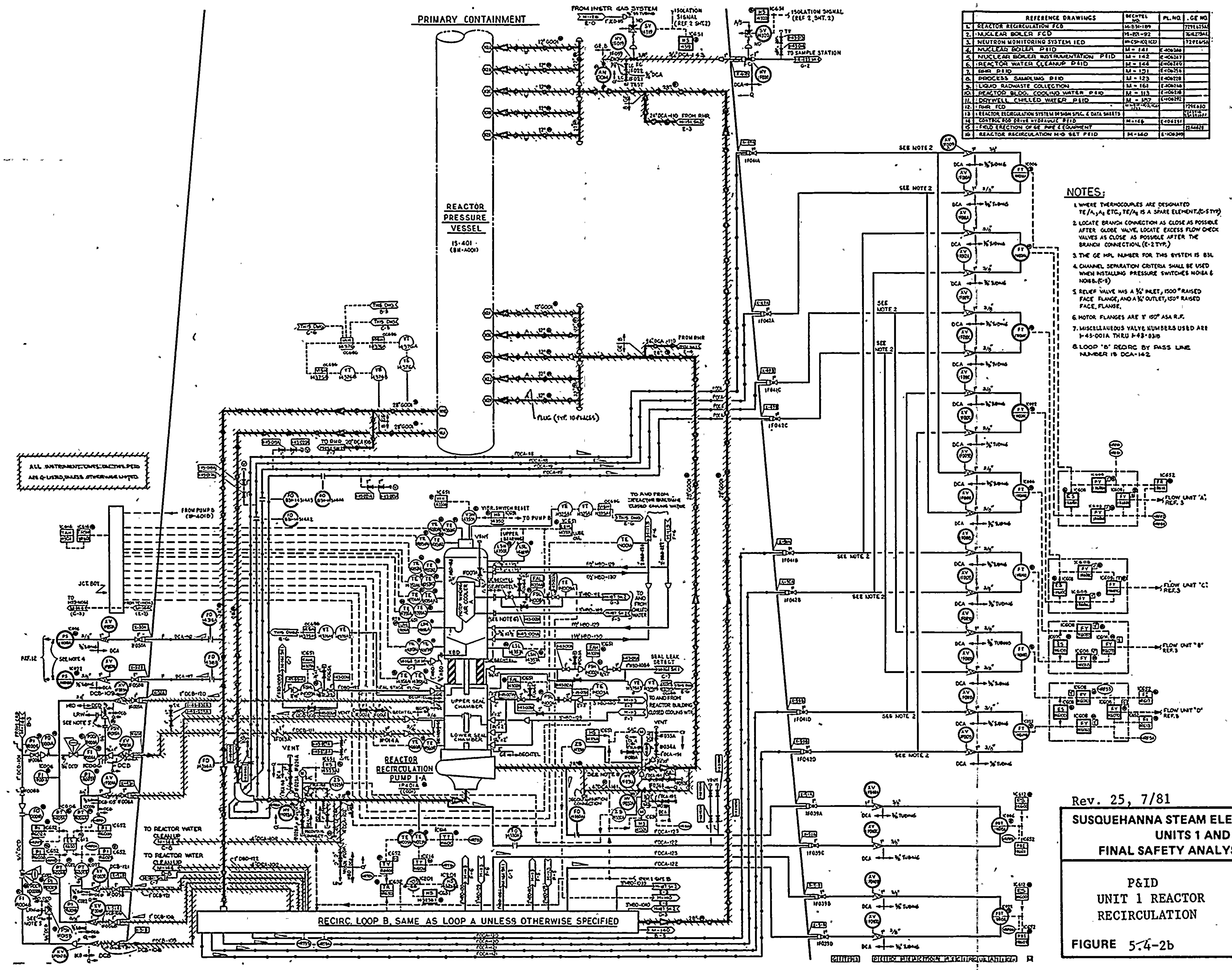
P&ID
NUCLEAR BOILER
VESSEL INSTRUMENTATION
FIGURE 5.1-3b



	REFERENCE DRAWINGS	BECHTEL NO.	PL. NO.	GE NO.
1.	REACTOR RECIRCULATION FCD	M-B31-109		729E625AE
2.	NUCLEAR BOILER FCD	M-B21-92		761E279AE
3.	NEUTRON MONITORING SYSTEM IED	M-C51-1(1)(2)		729E645A
4.	NUCLEAR BOILER P&ID	M-141	E-106246	
5.	NUCLEAR BOILER INSTRUMENTATION P&ID	M-142	E-106247	
6.	REACTOR WATER CLEANUP P&ID	M-144	E-106249	
7.	RHR P&ID	M-151	E-106256	
8.	PROCESS SAMPLING P&ID	M-123	E-106228	
9.	LIQUID RADWASTE COLLECTION	M-161	E-106266	
10.	REACTOR BLDG. COOLING WATER P&ID	M-113	E-106218	
11.	DRYWELL CHILLED WATER P&ID	M-187	E-106292	
12.	RHR FCD	[illegible]		729E610
13	REACTOR RECIRCULATION SYSTEM DESIGN SPEC. & DATA SHEETS			[illegible]
14	CONTROL ROD DRIVE HYDRAULIC P&ID	M-146	E-106251	
15	FIELD ERECTION OF GE PIPE & EQUIPMENT			[illegible]
16	REACTOR RECIRCULATION M-G SET P&ID	M-140	E-106245	

NOTES:

1. WHERE THERMOCOUPLES ARE DESIGNATED TE/A_1, A_2 ETC., TE/A_2 IS A SPARE ELEMENT. (C-5 TYP)
2. LOCATE BRANCH CONNECTION AS CLOSE AS POSSIBLE AFTER GLOBE VALVE. LOCATE EXCESS FLOW CHECK VALVES AS CLOSE AS POSSIBLE AFTER THE BRANCH CONNECTION. (E-2 TYP.)
3. THE GE MPL NUMBER FOR THIS SYSTEM IS B31.
4. CHANNEL SEPARATION CRITERIA SHALL BE USED WHEN INSTALLING PRESSURE SWITCHES N018A & N018B. (C-8)
5. RELIEF VALVE HAS A 3/4" INLET, 1500# RAISED FACE FLANGE, AND A 3/4" OUTLET, 150# RAISED FACE FLANGE.
6. MOTOR FLANGES ARE 1" 150# ASA R.F.
7. MISCELLANEOUS VALVE NUMBERS USED ARE 1-43-001A THRU 1-43-035
8. LOOP "B" RECIRC BY PASS LINE NUMBER IS DCA-142

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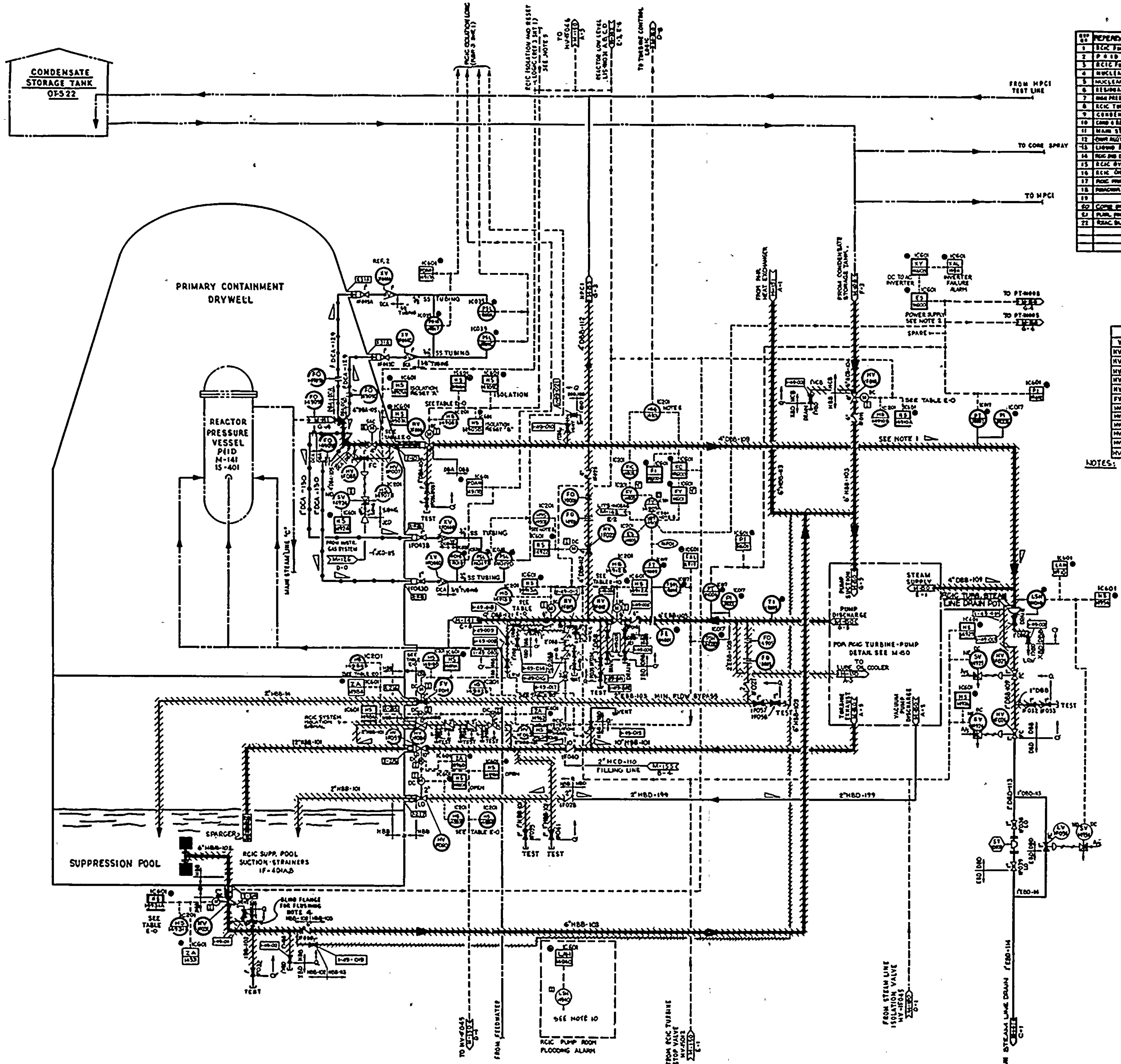
P&ID
UNIT 1 REACTOR
RECIRCULATION

FIGURE 5.4-2b



REF NO	REFERENCE DWGS. & SPECIFICATIONS	BECHTEL NO	PL NO	GE NO
1	RCIC PUMP TURBINE P&ID	M-150	E-106255	[illegible]
2	P & ID LEGEND	M-100	E-106205	[illegible]
3	RCIC FUNCTION CONTROL DIAGRAM	[illegible]		[illegible]
4	NUCLEAR BOILER P&ID	M-141	E-106246	[illegible]
5	NUCLEAR BOILER VESSEL INSTR	M-142	E-106247	
6	RESIDUAL HEAT REMOVAL P&ID	M-151	E-106256	[illegible]
7	HIGH PRESSURE COOLANT INJECTION P&ID	M-155	E-106260	[illegible]
8	RCIC TURBINE OUTLINE	[illegible]		[illegible]
9	CONDENSATE P&ID	M-105	M-106210	
10	COND & REF WATER STOR. & TRANSFER P&ID	M-108	E-106213	
11	MAIN STEAM P&ID	M-101	E-106206	
12	[illegible]			22A 2749
13	LIQUID RADWASTE-COLLECT P&ID	M-161	E-106266	
14	[illegible]			[illegible]
15	RCIC OVERSPEED TRIP	[illegible]		[illegible]
16	RCIC OIL PIPING DIAGRAM	[illegible]		[illegible]
17	[illegible]	[illegible]		[illegible]
18	REACTOR WATER CLEAN UP P&ID	M-144	E-106249	[illegible]
19				
20	CORE SPRAY P&ID	[illegible]	E-106257	[illegible]
21	FUEL POOL COOLING & CLEAN UP P&ID	M-153	E-106258	
22	REAC. BLDG. LINER PLATE PENE.	C272, 279		

TABLE E-0			
VALVE	PANEL 1C201	SELECTOR SWITCH	CONTROL ROOM
HV-F031	HS-14931B	HSS-14902	HS-14931A
HV-F013	HS-14913B	HSS-14904	HS-14913A
HV-F012	HS-14912B	HSS-14903	HS-14912A
HV-F008	HS-14908B	HSS-14904	HS-14908A
HV-F007	HS-14937B	HSS-14905	HS-14907A
FV-F019	HS-14919B	HSS-14903	HS-14919A
HV-F059	HS-14859B	HSS-14902	HS-14859A
HV-F060	HS-14860B	HSS-14905	HS-14860A
HV-F010	HS-14910B	HSS-14902	HS-14910A
HV-F084	HS-14984B	HSS-14909	HS-14984A
HV-F062	HS-14962B	HSS-14904	HS-14962A
HV-F022	—	HSS-14902	HS-14922

NOTES:

1. SLOPE STEAM LINE DOWN ALL THE WAY FROM MAIN STEAM LINE TO DRAIN POT JUST AHEAD OF TURBINE.
2. "AC" POWER FOR RCIC INSTRUMENTS SHALL BE DERIVED FROM "DC" SOURCES SEPARATE FROM THOSE WHICH SUPPLY THE HPCI SYSTEM.
3. PIPING HIGH POINT VENTS & LOW POINT DRAINS TO BE ADDED AS INDICATED.
4. PIPING SHOULD BE CLEANED AND FLUSHED IN ACCORDANCE WITH REF. 12. AFTER FLUSHING, DRY THE LINES WITH AIR. PURGE AND FILL WITH NITROGEN (REF. 12).
5. THE GE MPL NUMBER FOR THIS SYSTEM IS E51
6. FOR INTERLOCKING REQUIREMENTS AND AUTO VALVE ACTUATION SEE G.E. FUNCTIONAL CONTROL DIAGRAM REF 3.
7. RCIC ISOLATION IS INITIATED BY ANY OF THE FOLLOWING CONDITIONS:
   A. EQUIPMENT ROOM VENT. AIR INLET & OUTLET HIGH DIFF. TEMP.
   B. EMERGENCY AREA COOLER INLET HIGH TEMP.
   C. PIPE ROUTING AREA VENT. AIR INLET & OUTLET HIGH DIFF. TEMP. (TIME DELAY)
   D. PIPE ROUTING AREA HIGH TEMP. (TIME DELAY)
   E. STEAM LINE HIGH DIFF. PRESSURE (STEAM LINE BREAK)
   F. LOW REACTOR PRESSURE
   G. HIGH RCIC TURBINE EXHAUST DIAPHRAGM PRESSURE
   H. MANUAL INITIATION OF ISOLATION PUSH BUTTON.
8. HSS-14901 THRU HSS-14905 ARE MULTIPLE TRANSFER SELECTOR SWITCHES WHOSE DIFFERENT CONTACTS ARE USED SEVERAL TIMES ON THE DRAWING FACE AND TABLE E-0.
9. LAST VALVE NO. IS 1-49-021.
10. LSH-14940 IS SEISMICALLY QUALIFIED.

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P&ID
UNIT 1
REACTOR CORE
ISOLATION COOLING

FIGURE 5.4–9a,

GENERAL ELECTRIC 761E234AE

P&ID RCIC SYSTEM

NUCLEAR BOILER SYS 234X350AE

NOTES: CONT'D FROM SH I

10. "A-C" POWER FOR RCIC INSTRUMENTS SHALL BE DERIVED FROM A "D-C" SOURCE SEPARATE FROM THAT WHICH SUPPLIES THE HPCI SYSTEM.

11. FOR INTERLOCKING REQUIREMENTS AND AUTO VALVE ACTUATION SEE FUNCTIONAL CONTROL DIAGRAM, REF. 5.

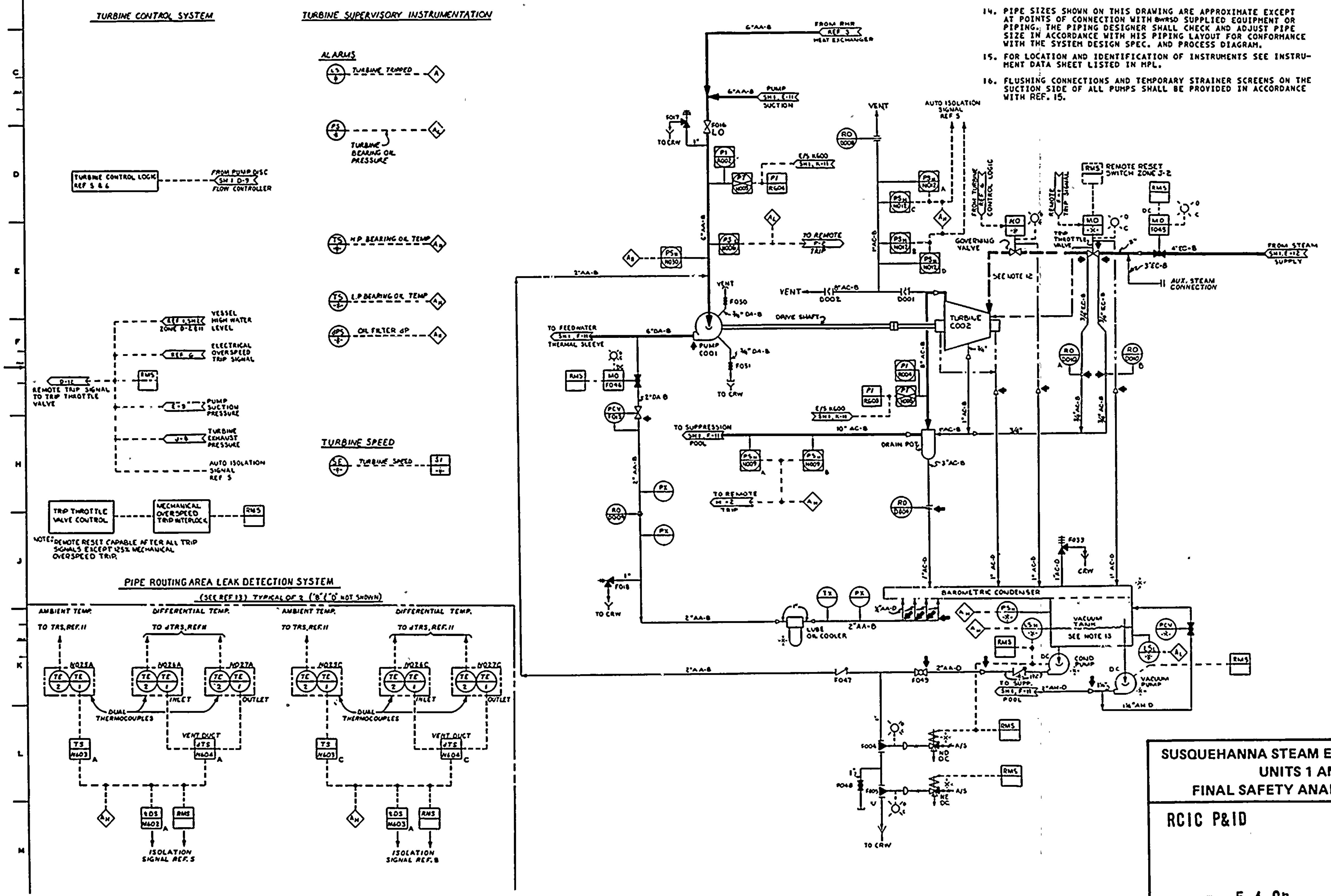
12. PIPING SHOWN IN PHANTOM LEADING TO TURBINE C002 TO BE ARRANGED & SUPPLIED BY TURBINE VENDOR.

13. THE BAROMETRIC CONDENSER AND VACUUM TANK SHALL BE LOCATED SUCH THAT ITS WATER LEVEL IS BELOW THE BOTTOM OF THE TURBINE EXHAUST FLANGE.

14. PIPE SIZES SHOWN ON THIS DRAWING ARE APPROXIMATE EXCEPT AT POINTS OF CONNECTION WITH BWRSD SUPPLIED EQUIPMENT OR PIPING. THE PIPING DESIGNER SHALL CHECK AND ADJUST PIPE SIZE IN ACCORDANCE WITH HIS PIPING LAYOUT FOR CONFORMANCE WITH THE SYSTEM DESIGN SPEC. AND PROCESS DIAGRAM.

15. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET LISTED IN MPL.

16. FLUSHING CONNECTIONS AND TEMPORARY STRAINER SCREENS ON THE SUCTION SIDE OF ALL PUMPS SHALL BE PROVIDED IN ACCORDANCE WITH REF. 15.



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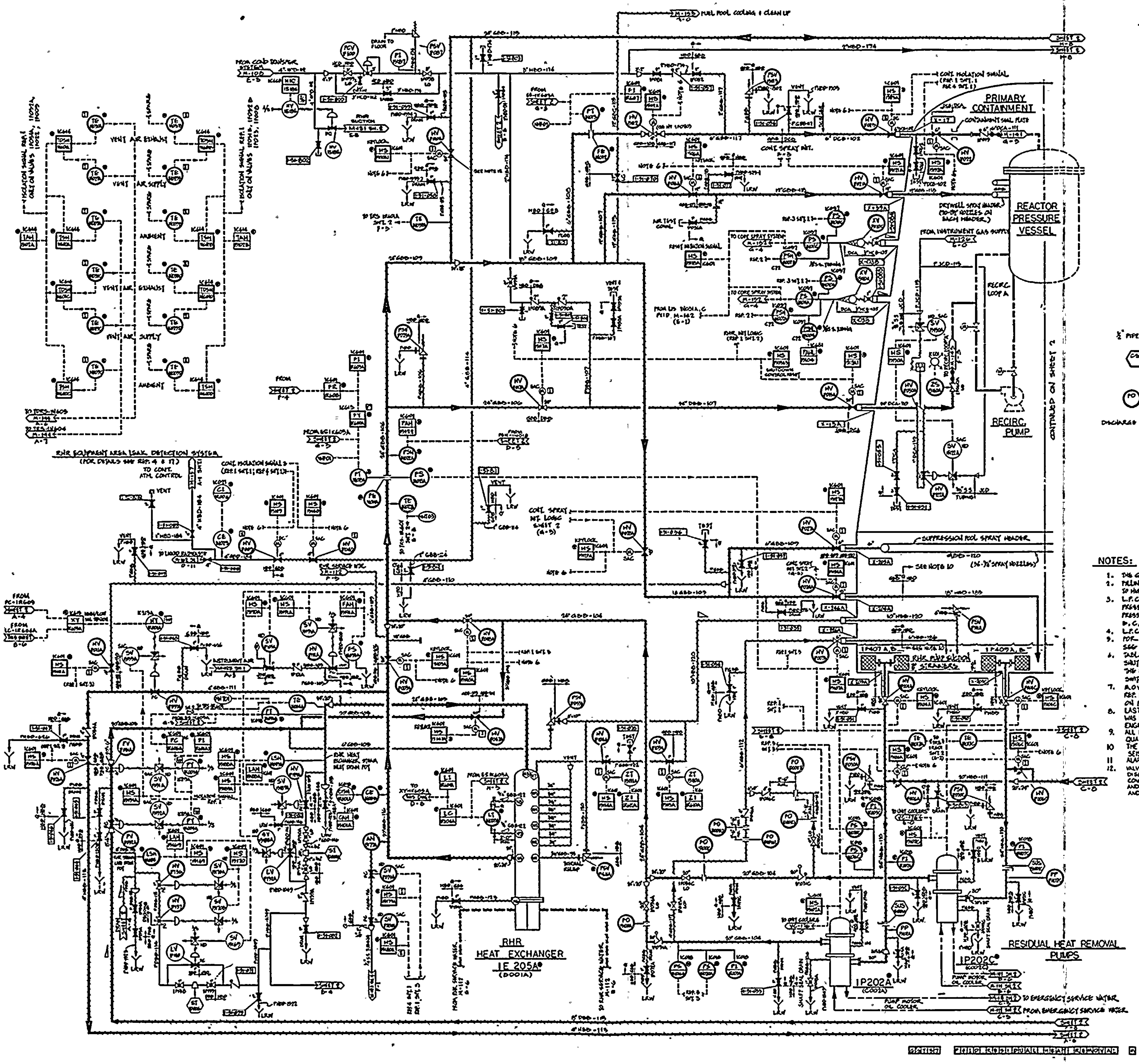
RCIC P&ID

FIGURE 5.4-9b

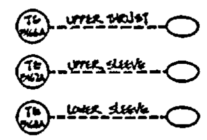


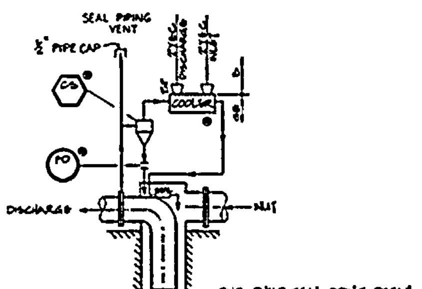
RHR PUMP MOTOR BEARINGS & WINDINGS TEMPERATURES:

WINDINGS

TE	TEMPERATURE ELEMENTS MOTORS A	B	C	D
15160	SPARE	TE15160B	SPARE	SPARE
15161	SPARE	SPARE	SPARE	SPARE
15162	SPARE	SPARE	SPARE	SPARE
15163	TE15163A	SPARE	SPARE	SPARE
15164	SPARE	SPARE	SPARE	TE15164D
15168	SPARE	SPARE	TE15168C	SPARE

BEARINGS (TYPICAL FOR PUMPS B,C,(D)





RHR PUMP SEAL PIPING DETAIL

PUMP	LINE NUMBER INLET	DISCHARGE	P&ID	LOCATION
1P202A	8"HRC-126	8"HRC-127	M-111 SH.2	D-6
1P202B	8"HRC-123	8"HRC-124	M-111 SH.2	C-3
1P202C	8"HRC-130	8"HRC-131	M-111 SH.2	D-6
1P202D	8"HRC-123	8"HRC-124	M-111 SH.2	B-3

NOTES:

1. THE GE MPL NUMBER FOR THIS SYSTEM IS E11.
2. FILLING LINE FROM CONDENSATE SYSTEM SHOULD BE CONNECTED TO HIGH POINT OF DISCHARGE PIPING.
3. L.P.C.I. MODE OF OPERATION IS INITIATED BY HIGH DRYWELL PRESSURE (PS-1N011A, B, C, D) CO-INCIDENT WITH LOW REACTOR PRESSURE (PS-1N021A, B, C, D) OR REACTOR LOW LEVEL (LS-1N081A, B, C, D).
4. L.P.C.I - LOW PRESSURE COOLANT INJECTION.
5. FOR INTERLOCKING REQUIREMENTS & AUTO. VALVE ACTUATION SEE GE FUNCTIONAL CONTROL DIAGRAM REF. 1.
6. TABLE "A" LISTS THE SWITCHES WHICH ARE INVOLVED IN REMOTE SHUTDOWN. ONLY THE CONTROL ROOM SWITCHES ARE SHOWN ON THE DRAWING. REFER TO TABLE "A" FOR REMOTE PANEL SWITCHES AND SELECTOR SWITCHES.
7. MOV TEST FUNCTION IS SHOWN ON ELEMENTARY DIAGRAM REF. 16 SHEET 1 AND FCD REF. 1 TEST SWITCHES MOUNTED ON PANEL 1C601.
8. LAST SEQUENTIAL VALVE NUMBER USED BY PROJECT ENGINEERING WAS 1-51-090 THE LAST VALVE NUMBER ASSIGNED BY FIELD ENGINEERING WAS 1-51-816.
9. ALL FLOODING DETECTOR LEVEL SWITCHES ARE SEISMICALLY QUALIFIED.
10. THE PORTION OF THE PIPE DESIGNATED WITH AN ARROW IS SEISMICALLY ANALYSED.
11. ALARMS FS101A,B ARE ENERGIZED BY TE-N004A,B OR TE-N005A,B ONLY.
12. VALVES 1F010A AND 1F010B SHALL BE PERMANENTLY CLOSED (POWER DISCONNECTED) UNLESS REACTOR IS IN THE COLD SHUTDOWN CONDITION. HOWEVER, ONE VALVE CAN BE TEMPORARILY ENERGIZED AND OPENED PROVIDED THE OTHER VALVE REMAINS DEENERGIZED AND CLOSED FOR PURPOSES OF SYSTEM FLUSHING.

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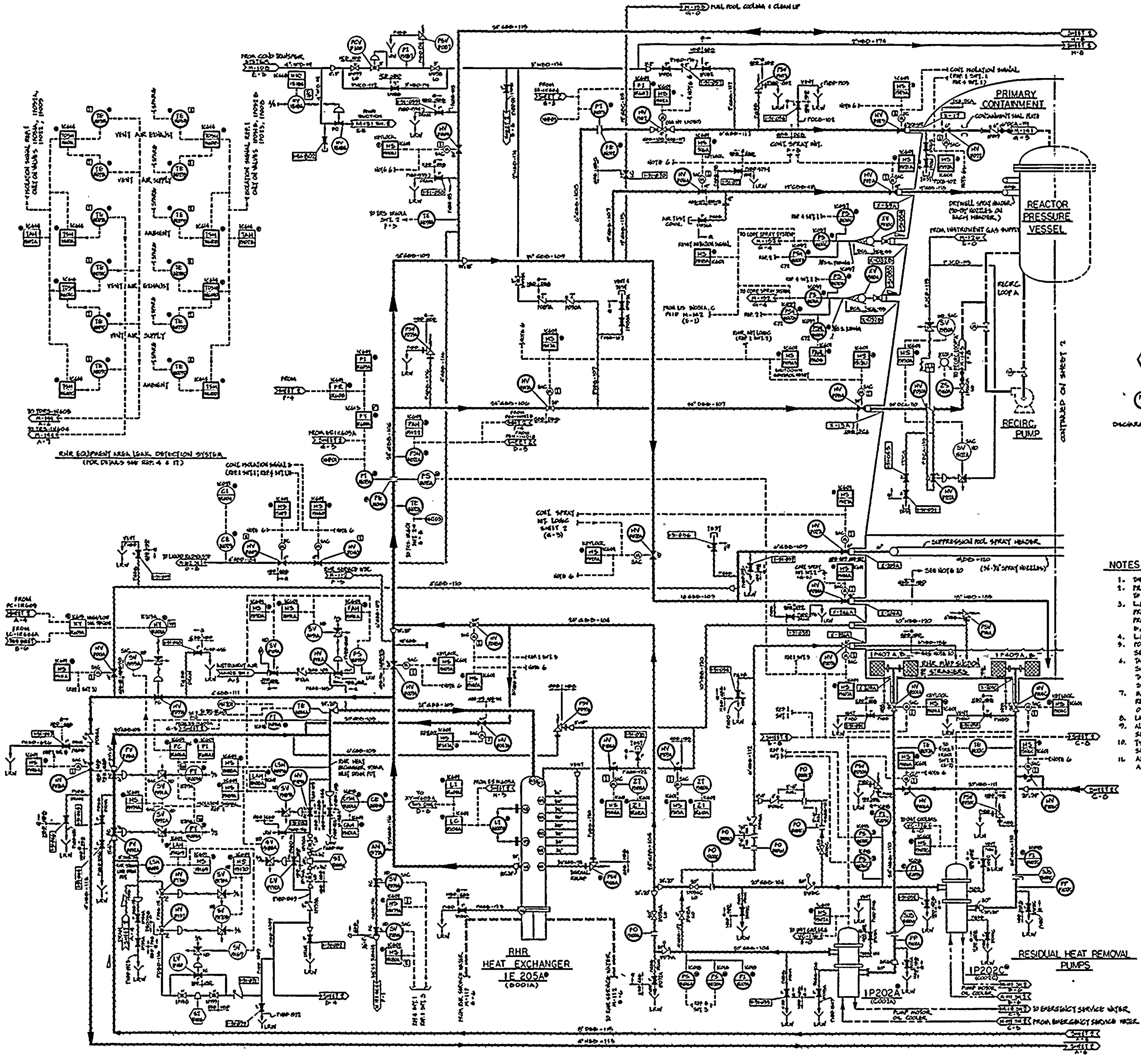
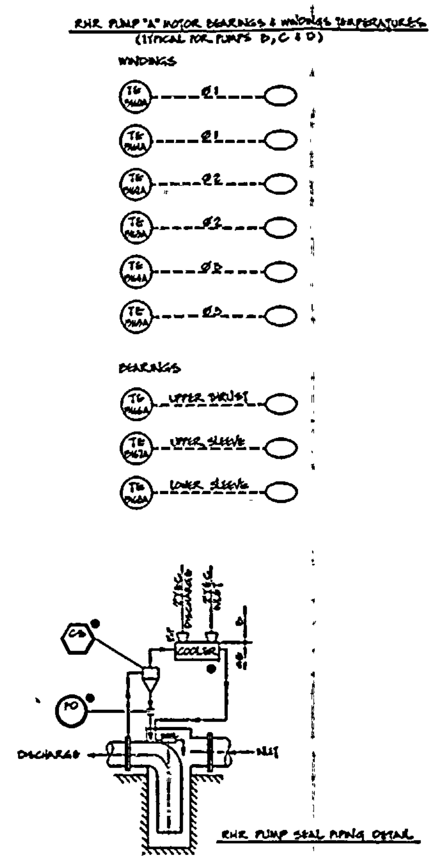
P&ID
RESIDUAL HEAT REMOVAL

FIGURE 5.4-13a





PUMP	LINE NUMBER		P&ID	LOCATION
	INLET	DISCHARGE		
1P202A	T-HRC-120	T-HRC-137	M-151 SH.2	D-6
1P202B	T-HRC-123	T-HRC-136	M-151 SH.2	C-3
1P202C	T-HRC-130	T-HRC-139	M-151 SH.2	D-6
1P202D	T-HRC-127	T-HRC-136	M-151 SH.2	D-3

NOTES:

1. THE GE MPL NUMBER FOR THIS SYSTEM IS E11.
2. FILLING LINE FROM CONDENSATE SYSTEM SHOULD BE CONNECTED TO HIGH POINT OF DISCHARGE PIPING.
3. L.P.C.I. MODE OF OPERATION IS INITIATED BY HIGH DRYWELL PRESSURE (PS-1N011A, B, C, D) CO-INCIDENT WITH LOW REACTOR PRESSURE (PS-1N021A, B, C, D) OR REACTOR LOW LEVEL (LS-1N031A, B, C, D).
4. L.P.C.I - LOW PRESSURE COOLANT INJECTION.
5. FOR INTERLOCKING REQUIREMENTS & AUTO. VALVE ACTUATION, SEE GE FUNCTIONAL CONTROL DIAGRAM REF. 1.
6. TABLE "A" LISTS THE SWITCHES WHICH ARE INVOLVED IN REMOTE SHUTDOWN. ONLY THE CONTROL ROOM SWITCHES ARE SHOWN ON THE DRAWING. REFER TO TABLE "A" FOR REMOTE PANEL SWITCHES AND SELECTOR SWITCHES.
7. M.O.V TEST FUNCTION IS SHOWN ON ELEMENTARY DIAGRAM REF. 16 SHEET 1 AND FCD REF. 1 TEST SWITCHES MOUNTED ON PANEL 1C601.
8. LAST SEQUENTIAL VALVE NUMBER USED WAS 1-51-087
9. ALL FLOODING DETECTOR LEVEL SWITCHES AND LIGHTS ARE SEISMICALLY QUALIFIED.
10. THE PORTION OF THE PIPE DESIGNATED WITH AN ARROW IS SEISMICALLY ANALYZED.
11. ALARMS 1SI01A, B ARE ENERGIZED BY TE-1N004 A, B OR TE-1N005 A, B ONLY.

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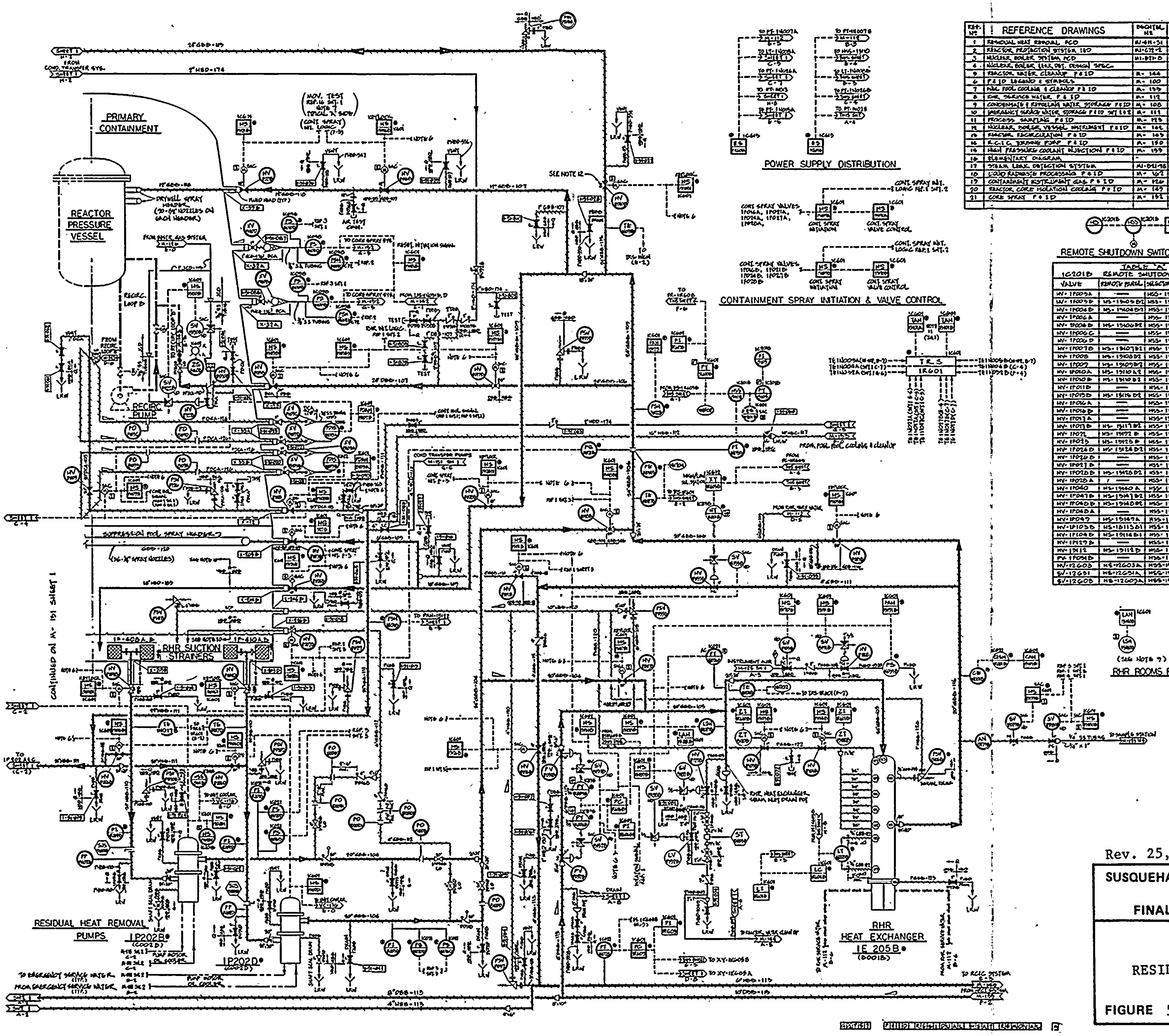
P&ID
RESIDUAL HEAT REMOVAL

FIGURE 5.4-13a



REF. NO.	REFERENCE DRAWINGS	BECHTEL NO.	FL. NO.	G.E. NO.
1	RESIDUAL HEAT REMOVAL FCD	M1-411-31		729E650
2	REACTOR PROTECTION SYSTEM IED	M1-C72-2		791E414AG
3	NUCLEAR BOILER SYSTEM FCD	M1-B21-D		761E771AG
4	NUCLEAR BOILER LEAK DET. DESIGN SPEC.			22A4440.
5	REACTOR WATER CLEANUP P&ID	M-144	E-106249	
6	P&ID LEGEND & SYMBOLS	M-100	E-106209	
7	FUEL POOL COOLING & CLEANUP P&ID	M-153	E-106250	
8	EMR. SERVICE WATER P&ID	M-112	E-106213	
9	CONDENSATE & REFUELING WATER STORAGE P&ID	M-108	E-106215	
10	EMERGENCY SERVICE WATER STORAGE P&ID SHT 1&2	M-111	E-106216	
11	PROCESS SAMPLING P&ID	M-123	E-106226	
12	NUCLEAR BOILER VESSEL INSTRUMENT P&ID	M-142	E-106247	
13	REACTOR RECIRCULATION P&ID	M-143	E-106248	
14	R.C.I.C. TURBINE PUMP P&ID	M-150	E-106255	
15	HIGH PRESSURE COOLANT INJECTION P&ID	M-155	E-106260	
16	ELEMENTARY DIAGRAM	"		791E410AF
17	STEAM LEAK DETECTION SYSTEM	M1-B21-98		
18	LIQUID RADWASTE PROCESSING P&ID	M-162	E-106267	
19	CONTAINMENT INSTRUMENT GAS P&ID	M-126	E-106231	
20	REACTOR CORE ISOLATION COOLING P&ID	M-149	E-106254	
21	CORE SPRAY P&ID	M-152	E-106257	

TABLE 'A'
1C201B REMOTE SHUTDOWN PANEL

VALVE	REMOTE PANEL	SELECTOR SWITCH	CONTROL ROOM
HV-1F003A	—	HSS-15112A	HS-15103A
HV-1F003B	HS-15103B2	HSS-15116A	HS-15103B1
HV-1F004B	HS-15104B2	HSS-15114A	HS-15104B1
HV-1F006A	—	HSS-15111A	HS-15106A
HV-1F006B	HS-15106B2	HSS-15113B	HS-15106B1
HV-1F006C	—	HSS-15111A	HS-15106C
HV-1F006D	—	HSS-15113B	HS-15106D1
HV-1F007B	HS-15107B2	HSS-15115A	HS-15107B1
HV-1F008	HS-15108B2	HSS-15113A	HS-15108B1
HV-1F009	HS-15109B2	HSS-15112B	HS-15109B1
HV-1F010A	HS-15110A2	HSS-15112B	HS-15110A1
HV-1F010B	HS-15110B2	HSS-15113B	HS-15110B1
HV-1F011B	—	HSS-15115A	HS-15111B
HV-1F015B	HS-15115B2	HSS-15116A	HS-15115B1
HV-1F016A	—	HSS-15111B	HS-15116A
HV-1F016B	—	HSS-15113B	HS-15116B
HV-1F017A	—	HSS-15112B	HS-15117A
HV-1F017B	HS-15117B2	HSS-15114A	HS-15117B1
HV-1F022	HS-15122B	HSS-15111B	HS-15122A
HV-1F023	HS-15123B	HSS-15113A	HS-15123A
HV-1F024B	HS-15124B2	HSS-15114B	HS-15124B1
HV-1F026B	—	HSS-15116A	HS-15126B
HV-1F027B	—	HSS-15115B	HS-15127B
HV-1F028B	HS-15128B2	HSS-15115A	HS-15128B1
HV-1F028A	—	HSS-15111A	HS-15128A
HV-1F040	HS-15140A	HSS-15111A	HS-15140
HV-1F047B	HS-15147B2	HSS-15115B	HS-15147B1
HV-1F048B	HS-15148B2	HSS-15115B	HS-15148B1
HV-1F048A	—	HSS-15112A	HS-15148A
HV-1F049	HS-15149A	HSS-15114A	HS-15149
HV-1F103B	HS-15113B1	HSS-15115B	HS-15113B
HV-1F104B	HS-15114B1	HSS-15116A	HS-15113B
HV-1F127B	—	HSS-15115B	HS-15107B
HV-15112	HS-15112B	HSS-15112A	HS-15112A
PV-1F051B	—	HSS-15116B	HS-15151B
HV-12603	HS-12603A	HSS-15112A	HS-12603
SV-12651	HS-12651A	HSS-15111B	HS-12651
SV-12605	HS-12605A	HSS-15114B	HS-12605

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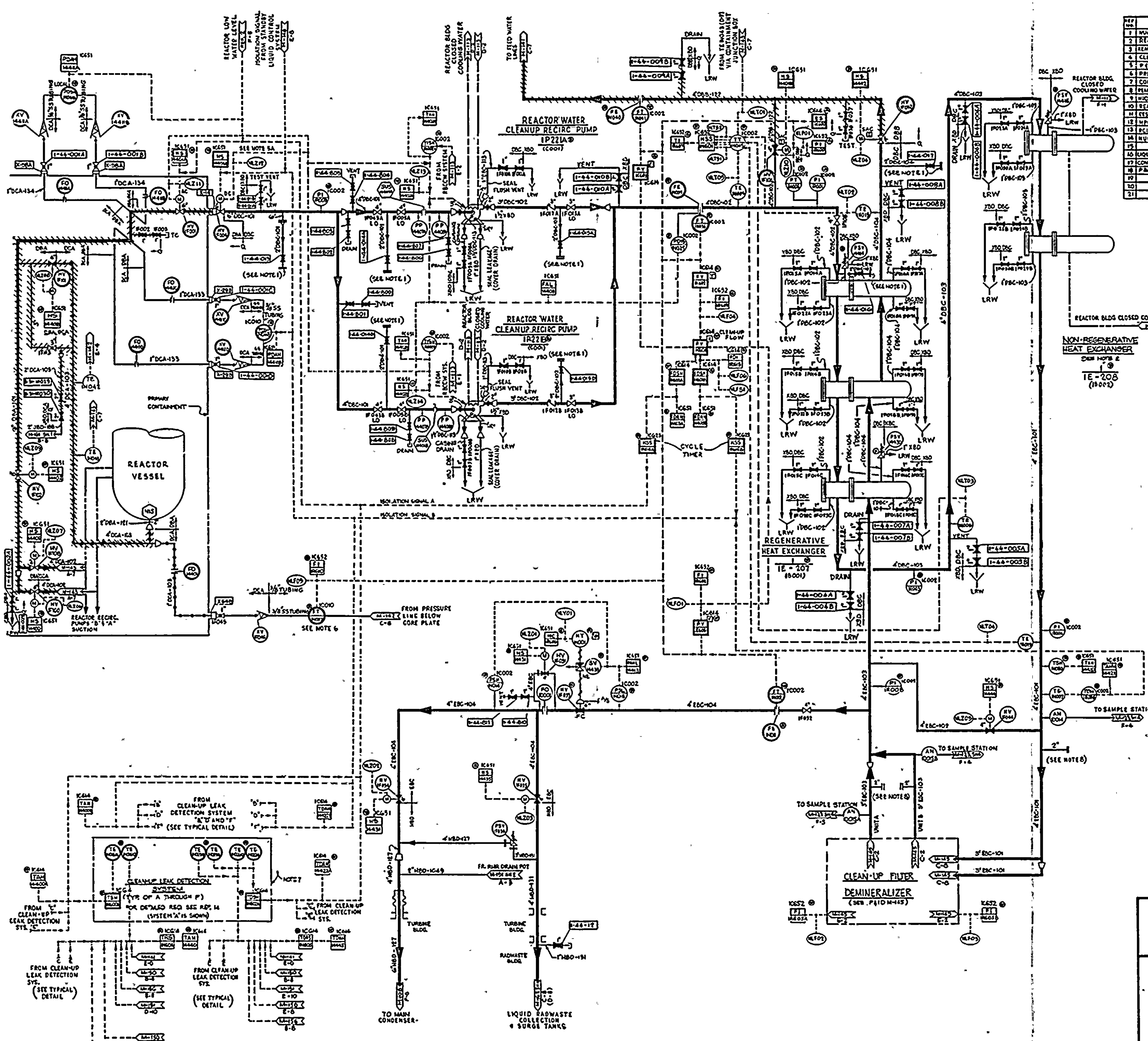
P&ID
RESIDUAL HEAT REMOVAL

FIGURE 5.4-13b



REF NO.	REFERENCE DRAWINGS	BECHTEL NO.	PL. NO.	G.E. NO.
1	NUCLEAR BOILER P&ID	M-141	E-106246	
2	REACTOR WATER CLEANUP FCD	M1-G33		729E608
3	REACTOR BLDG. CLOSED COOLING WATER P&ID	M-113	E-106218	
4	CLEANUP FILTER/DEMINERALIZER P&ID	M-145	E-106250	
5	P&ID LEGEND	M-100	E-106205	
6	PROCESS SAMPLING P&ID	M-123	E-106228	
7	CONDENSATE P&ID	M-105	E-106210	
8	REACTOR WATER CLEANUP SPEC & DATA SHEET			[illegible]
9	NUCLEAR BOILER VESSEL INSTR. P&ID	M-142	E-106247	
10	REACTOR WATER RECIRC P&ID	M-143	E-106248	
11	RESIDUAL HEAT REMOVAL P&ID	M-151	E-106256	
12	HPCI PUMP TURBINE P&ID	M-156	E-106261	
13	RCIC PUMP TURBINE P&ID	M-150	E-106255	
14	NUCLEAR BOILER LEAK DET. DES. SPEC.			22A4440
15				
16	LIQUID RADWASTE PROCESSING P&ID	M-162	E-106267	
17	CONTAINMENT ATMOS. CONTROL	M-157	E-106262	
18	P&ID REACTOR WATER CLEAN-UP SYS.			[illegible]
19				
20				
21				

NOTES:

1. CHEMICAL CLEANING AND DECONTAMINATION CONNECTIONS SHALL BE PROVIDED TO GIVE OPTIMUM DECONTAMINATION. CONNECTIONS SHALL BE ARRANGED TO PROVIDE DECONTAMINATION OF ONE PIECE OF EQUIPMENT SEPARATELY FROM ALL OTHER EQUIPMENT. THE CONNECTIONS SHALL BE ROUTED TO NEAR DOORWAY.
2. NON-REGENERATIVE HEAT EXCHANGER SHELLSIDE VALVES AND PIPING DESIGNED TO RBCCW SYSTEM CONDITIONS.
3. MISCELLANEOUS VALVE NUMBERS USED ARE 1-44-001A THRU 1-44-010.
4. THE GE MPL NUMBER FOR THIS SYSTEM IS G33.
5. CLEANUP SYSTEM ISOLATION IS INITIATED BY ANY OF THE FOLLOWING CONDITIONS:
   A. NON-REGENERATIVE HEAT EXCHANGER EFFLUENT HIGH TEMPERATURE. (ONLY F004)
   B. LEAK DETECTION SYSTEM HIGH DIFFERENTIAL FLOW.
   C. LEAK DETECTION SYSTEM HIGH DIFFERENTIAL TEMPERATURE
   D. LEAK DETECTION SYSTEM HIGH TEMPERATURE.
   E. REACTOR LOW WATER LEVEL.
   F. STANDBY LIQUID CONTROL SYSTEM ACTUATION (ONLY F004)
   G. HIGH ΔP, LEAK DETECTION SYSTEM.
6. LOCATE TAP FOR IN037 SUCH THAT:
   A) FULL RANGE IS NOT EXCEEDED PER INSTRUMENT DATA SHEET.
   B) ELEVATION SHALL BE WITHIN 20" BELOW VESSEL Q.
7. TWO SENSOR SYSTEMS PER ROOM. THE TEMPERATURE ELEMENTS ARE DUAL ELEMENT.
8. CHEMICAL CLEAN-OUT IN HOLDING PUMP ROOM.

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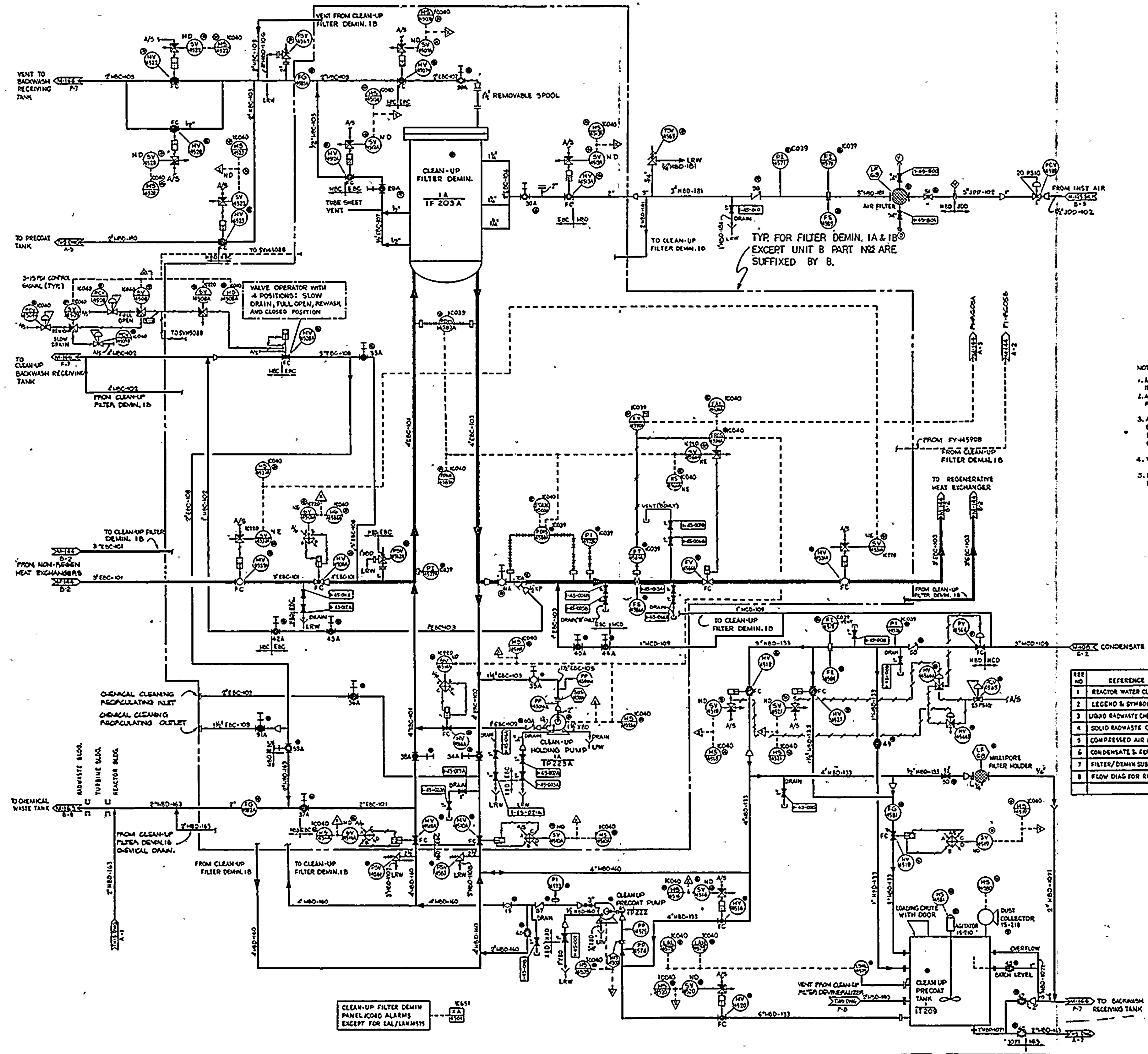
P&ID
REACTOR WATER
CLEAN-UP
FIGURE 5.4-16



NOTES:

1. ALL APED FURNISHED EQUIPMENT, VALVES AND INSTRUMENTS ARE UNDER MPL NO G 33-Z001.
2. ALL AIR OPERATED VALVES ARE SHOWN IN FILTERING PROCESS STATUS.
3. ALL REMOTE INSTRUMENTS AND CONTROLS WITH THE BOUNDARY SHALL BE ARRANGED ON VENDORS INSTRUMENT RACK 25 (BECHTEL NO. IC039) AND CONTROL PANEL 26 (BECHTEL NO. IC040)
4. VALVE STEM REACH RODS TO BE FURNISHED BY BECHTEL.
5. PIPING SYSTEM HIGH POINT VENTS AND LOW POINT DRAINS ON SMALL PIPING TO BE ADDED LATER

REF NO	REFERENCE DRAWINGS	BECHTEL NO.	PL NO.	GE NO.
1	REACTOR WATER CLEAN-UP P&ID	M-144	E-106249	
2	LEGEND & SYMBOLS	M-100	E-106205	
3	LIQUID RADWASTE CHEMICAL PROCESSING	M-163	E-106268	
4	SOLID RADWASTE COLLECTION P&ID	M-166	E-106271	
5	COMPRESSED AIR P&ID	M-125 SH.1	E-106230	
6	CONDENSATE & REFUELING WATER STORAGE P&ID	M-108	E-106213	
7	FILTER/DEMIN SUBSYS P&ID & DEVICE LIST	MI-G33-13 MI-G33-20		761E397
8	FLOW DIAG FOR REACTOR W. CLEAN-UP SYS (P&ID)	MI-G33-33		3249-164

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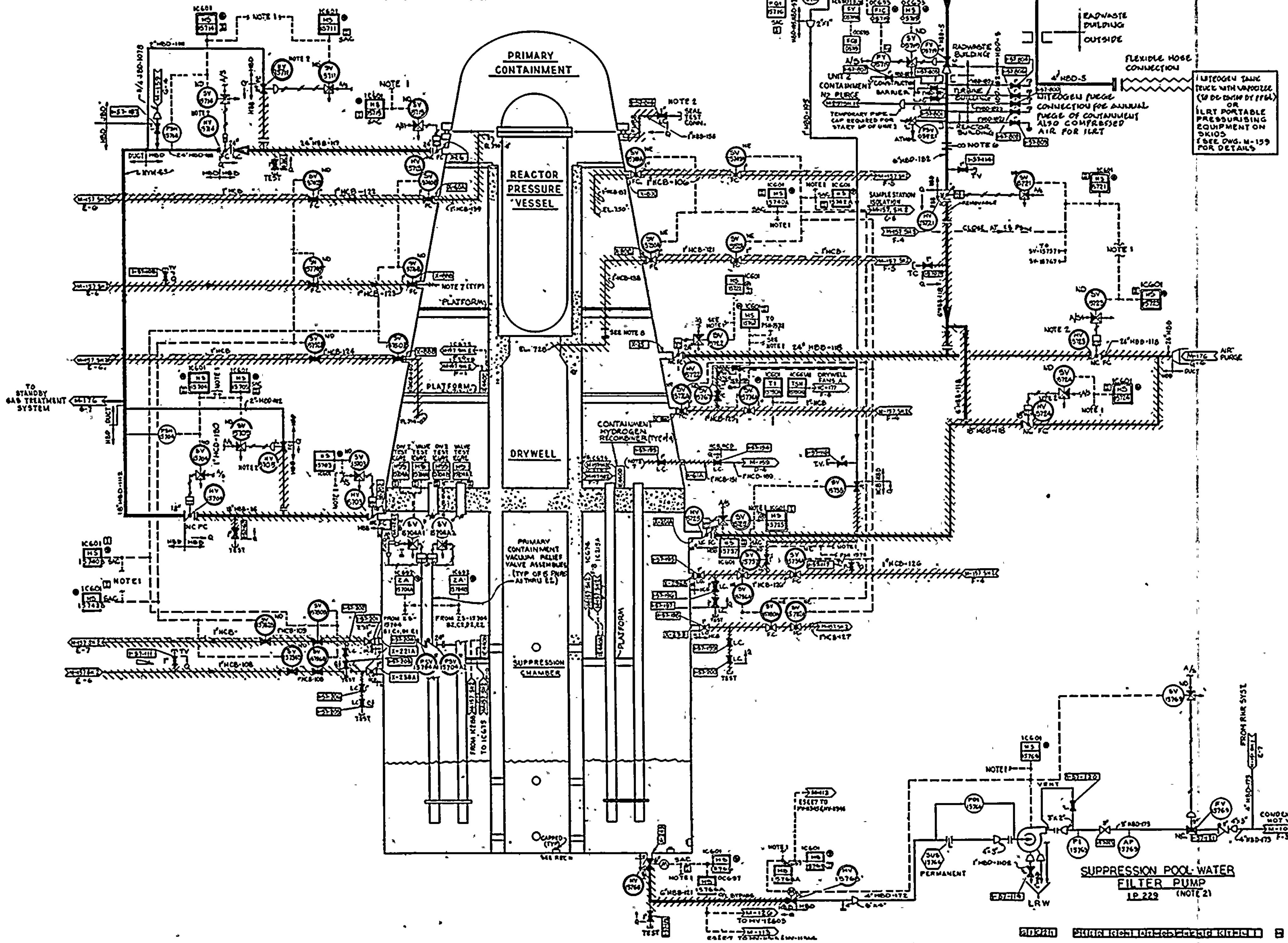
P&ID
CLEAN-UP
FILTER-DEMINERALIZER

FIGURE 5.4-18

REFERENCED DRAWING	BECHTEL NO	PL NO	GE NO
1. NUCLEAR BOILER - FUNCTIONAL CONTROL DIAGRAM	M-621-6(1), 6(2), 6(3) & 6(4)		761E275
2. LEGEND & SYMBOLS	M-100	E-106205	
3.			
4.			
5. UNIT 1 COMPRESSED AIR SYS.	M-125 SH2	E-106230	
6. CONTAINMENT INSTRUMENT GAS	M-126	E-106231	
7.			
8. GASEOUS RADWASTE - OXYGEN REMOVAL	M-169	E-106276	
9. RB AFD ZONES I & II	M-176	E-106281	
10. HVAC CONTROL DIAG. DRYWELL	VC-177	E-106482	
11. LINER PLATE LEAK CHASE SYSTEM	C-298	E-105305	

NOTES:

1. ALL VALVES ON PRIMARY CONTAINMENT ARE INTERLOCKED TO CLOSE UPON "CONTAINMENT ISOLATION" SIGNAL (REF 1 SH 1).
2. LOCATE AS NEAR AS POSSIBLE TO PRIMARY CONTAINMENT.
3. THE LOGIC FOR EXCESS-FLOW CHECK VALVES SHALL BE INCLUDED WITH THIS P&ID. SEE P&ID. M-100 FOR DETAIL OF EXCESS-FLOW CHECK VALVE ASSEMBLY.
4.
5. MISC. VALVE NO'S INCLUDED ON THIS P&ID ARE 1-57-001 THRU 1-57-210
6. SPECTACLE BLIND MUST BE INSTALLED DURING NORMAL OPERATIONS TO ELIMINATE SECONDARY CONTAINMENT BYPASS LEAKAGE.
7. THREAD THE END OF THE PIPING WHICH IS OPEN TO THE CONTAINMENT ATMOSPHERE FOR LINES 1" HCB-139, 1" HCB-123, 1" HCB-140, 1" HCB-137, 1" HCB-125, 1" HCB-109 & 1" HCB-151 (FOR USE IN LEAKAGE TESTING) OF ISOLATION VALVES AT PENETRATIONS X-60A, X-80C, X-88B & X-61A
8. FHCB-136 IS TO BE BROKEN AT SPOOL & A BLIND FLANGE USED IN LEAKAGE TESTING OF ISOLATION VALVES (PENETRATION X-80C)
9. SOLENOID VALVES SV-15776 A & B, SV-15776 A & B, SV-15736 A & B, SV-15734 A & B, SV-15767 & SV-15737 ARE TO BE INSTALLED OPPOSITE TO THE NORMAL DIRECTION OF FLOW. ALL OTHER PROCESS SOLENOID VALVES IN THE CONTAINMENT ATMOSPHERE SAMPLE LINES ARE TO BE INSTALLED IN THE SAME DIRECTION AS NORMAL FLOW.



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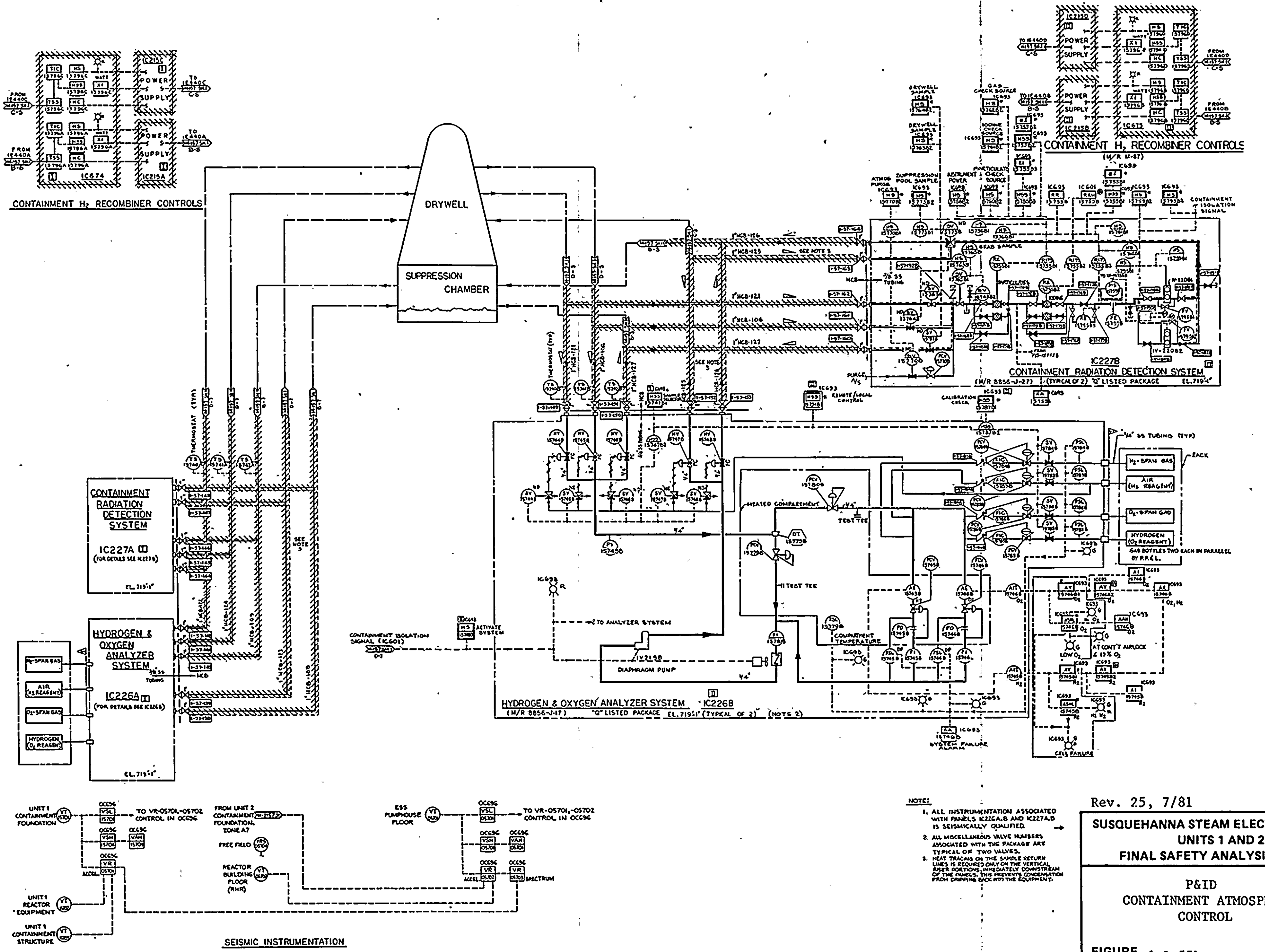
P&ID
CONTAINMENT ATMOSPHERE
CONTROL

FIGURE 6.2-55a



NOTE:

1. ALL INSTRUMENTATION ASSOCIATED WITH PANELS IC226A,B AND IC227A,B IS SEISMICALLY QUALIFIED.
2. ALL MISCELLANEOUS VALVE NUMBERS ASSOCIATED WITH THE PACKAGE ARE TYPICAL OF TWO VALVES.
3. HEAT TRACING ON THE SAMPLE RETURN LINES IS REQUIRED ONLY ON THE VERTICAL RISER PORTIONS, IMMEDIATELY DOWNSTREAM OF THE PANELS. THIS PREVENTS CONDENSATION FROM DRIPPING BACK INTO THE EQUIPMENT.

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P&ID
CONTAINMENT ATMOSPHERE
CONTROL

FIGURE 6.2-55b

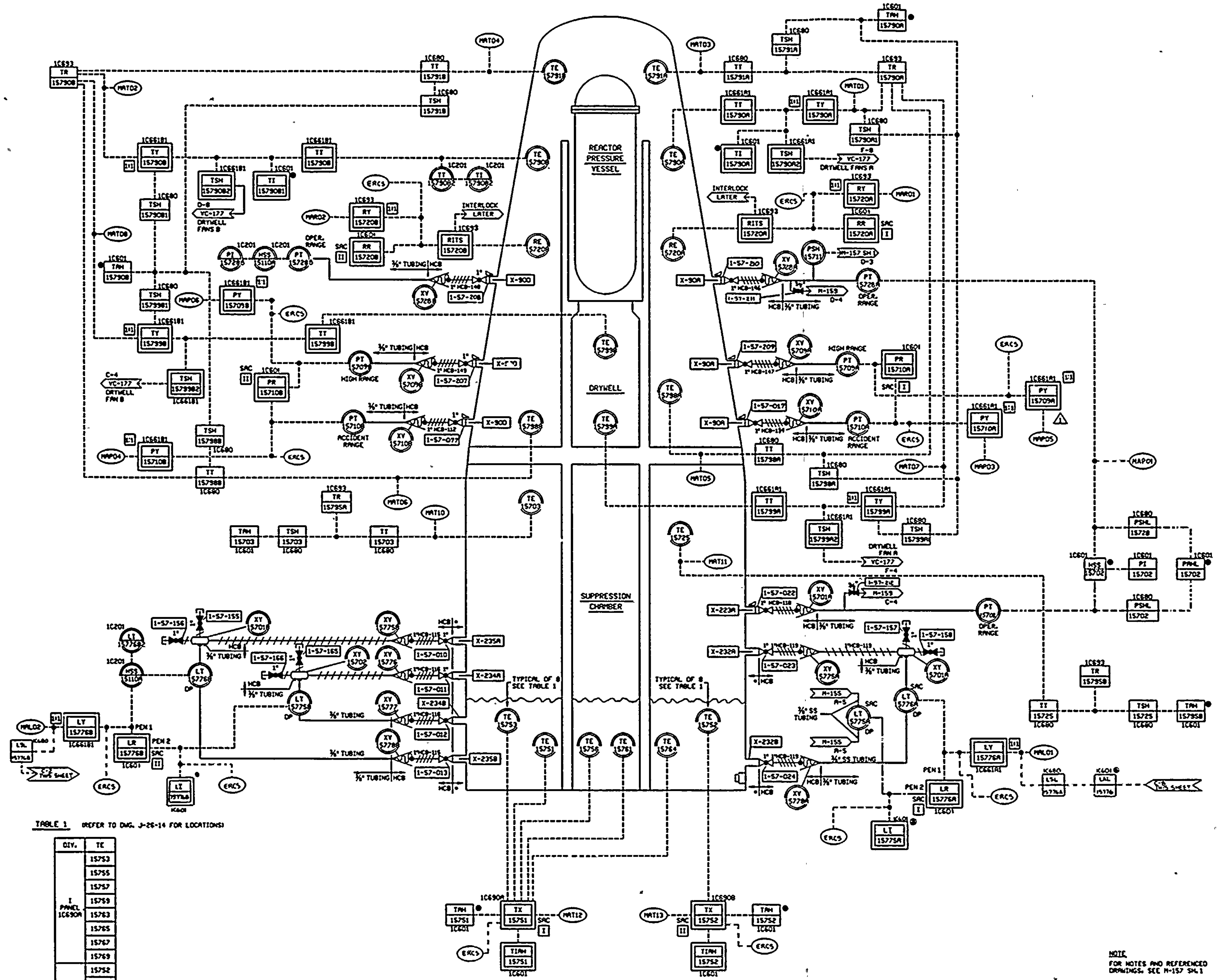

TABLE 1 (REFER TO DWG. J-26-14 FOR LOCATIONS)

DIV.	TE
I PANEL 1C690A	15753
	15755
	15757
	15759
	15763
	15765
	15767
	15769
II PANEL 1C690B	15752
	15754
	15758
	15760
	15762
	15766
	15768
	15770

NOTE
FOR NOTES AND REFERENCED DRAWINGS, SEE M-157 SH.1

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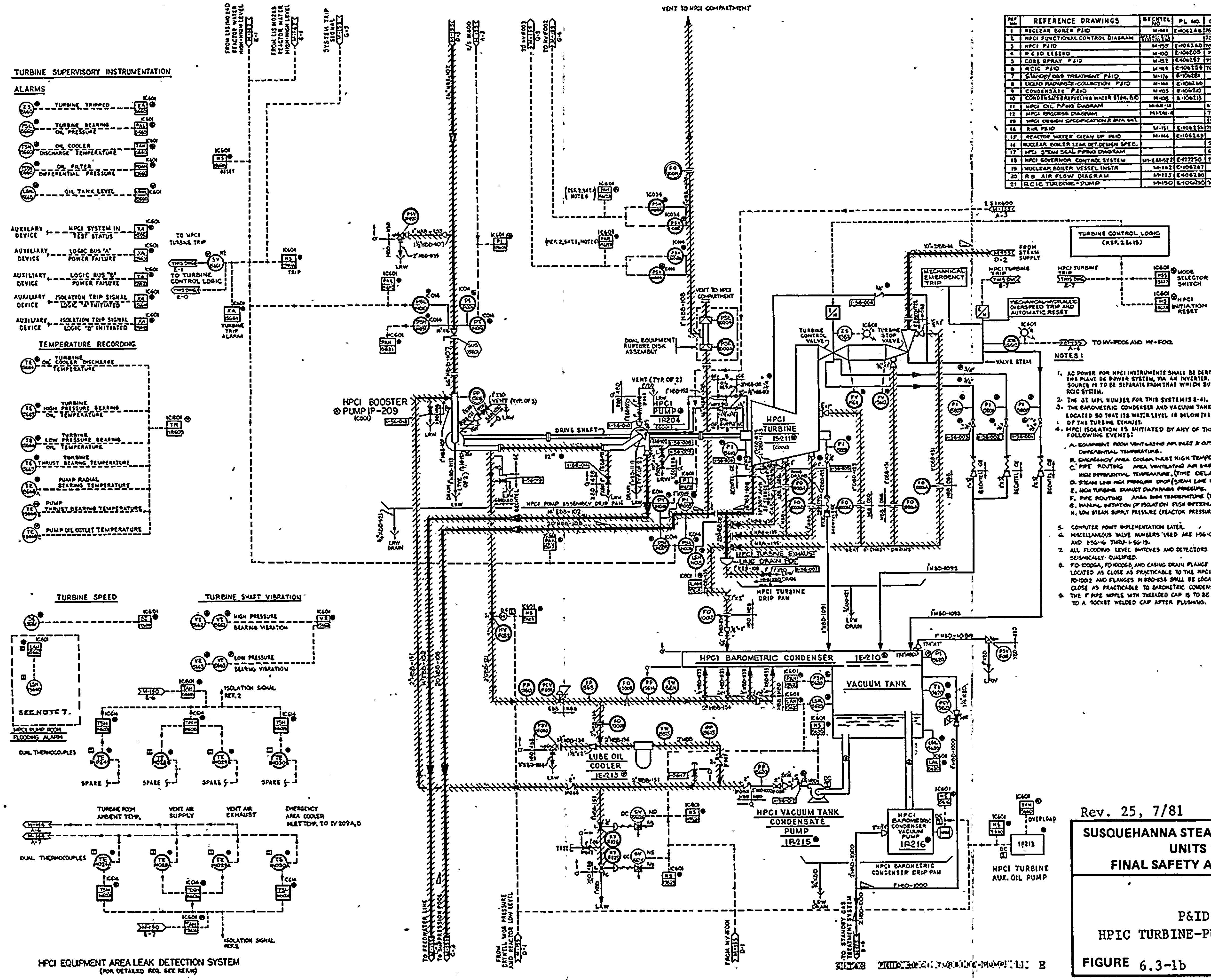
P&ID
UNIT 1
CONTAINMENT
ATMOSPHERE CONTROL

FIGURE 6.2–55c



REF. No.	REFERENCE DRAWINGS	BECHTEL NO.	PL NO.	GE. NO.
1	NUCLEAR BOILER P&ID	M-141	E-106246	761E250AE
2	HPCI FUNCTIONAL CONTROL DIAGRAM	[illegible]		729E627
3	HPCI P&ID	M-155	E-106260	761E250A
4	P&ID LEGEND	M-100	E-106205	197R567
5	CORE SPRAY P&ID	M-152	E-106257	731E764AE
6	RCIC P&ID	M-149	E-106254	761E234AE
7	STANDBY GAS TREATMENT P&ID	M-176	E-106281	
8	LIQUID RADWASTE-COLLECTION P&ID	M-161	E-106266	
9	CONDENSATE P&ID	M-105	E-106210	
10	CONDENSATE & REFUELING WATER STOR. P&ID	M-108	E-106213	
11	HPCI OIL PIPING DIAGRAM	M-E41-14		65998C
12	HPCI PROCESS DIAGRAM	M1-E41-4		761E270A
13	HPCI DESIGN SPECIFICATION & DATA SHT.			[illegible]
14	RHR P&ID	M-151	E-106256	761E232A
15	REACTOR WATER CLEAN UP P&ID	M-144	E-106249	
16	NUCLEAR BOILER LEAK DET. DESIGN SPEC.			22A1441
17	HPCI STEAM SEAL PIPING DIAGRAM			662620
18	HPCI GOVERNOR CONTROL SYSTEM	M1-E41-22	E-127250	2763-2261
19	NUCLEAR BOILER VESSEL INSTR	M-142	E-106247	
20	RB AIR FLOW DIAGRAM	M-175	E-106280	
21	RCIC TURBINE-PUMP	M-150	E-106255	761E234AE

NOTES:

1. AC POWER FOR HPCI INSTRUMENTS SHALL BE DERIVED FROM THE PLANT DC POWER SYSTEM, VIA AN INVERTER. THE DC SOURCE IS TO BE SEPARATE FROM THAT WHICH SUPPLIES THE RCIC SYSTEM.
2. THE GE MPL NUMBER FOR THIS SYSTEM IS E-41.
3. THE BAROMETRIC CONDENSER AND VACUUM TANK SHALL BE LOCATED SO THAT ITS WATER LEVEL IS BELOW THE BOTTOM OF THE TURBINE EXHAUST.
4. HPCI ISOLATION IS INITIATED BY ANY OF THE FOLLOWING EVENTS:
   A. EQUIPMENT ROOM VENTILATING AIR INLET & OUTLET HIGH DIFFERENTIAL TEMPERATURE.
   B. EMERGENCY AREA COOLER INLET HIGH TEMPERATURE.
   C. PIPE ROUTING AREA VENTILATING AIR INLET & OUTLET HIGH DIFFERENTIAL TEMPERATURE. (TIME DELAY)
   D. STEAM LINE HIGH PRESSURE DROP (STEAM LINE BREAK)
   E. HIGH TURBINE EXHAUST DIAPHRAGM PRESSURE.
   F. PIPE ROUTING AREA HIGH TEMPERATURE (TIME DELAY)
   G. MANUAL INITIATION OF ISOLATION PUSH BUTTON.
   H. LOW STEAM SUPPLY PRESSURE (REACTOR PRESSURE)
5. COMPUTER POINT IMPLEMENTATION LATER.
6. MISCELLANEOUS VALVE NUMBERS USED ARE 1-56-001 THRU 1-56-013 AND 1-56-16 THRU 1-56-19.
7. ALL FLOODING LEVEL SWITCHES AND DETECTORS ARE SEISMICALLY QUALIFIED.
8. FO-1000GA, FO-1000GB, AND CASING DRAIN FLANGE SHALL BE LOCATED AS CLOSE AS PRACTICABLE TO THE HPCI TURBINE. FO-1001? AND FLANGES IN HBD-134 SHALL BE LOCATED AS CLOSE AS PRACTICABLE TO BAROMETRIC CONDENSER.
9. THE 1" PIPE NIPPLE WITH THREADED CAP IS TO BE CHANGED TO A SOCKET WELDED CAP AFTER FLUSHING.

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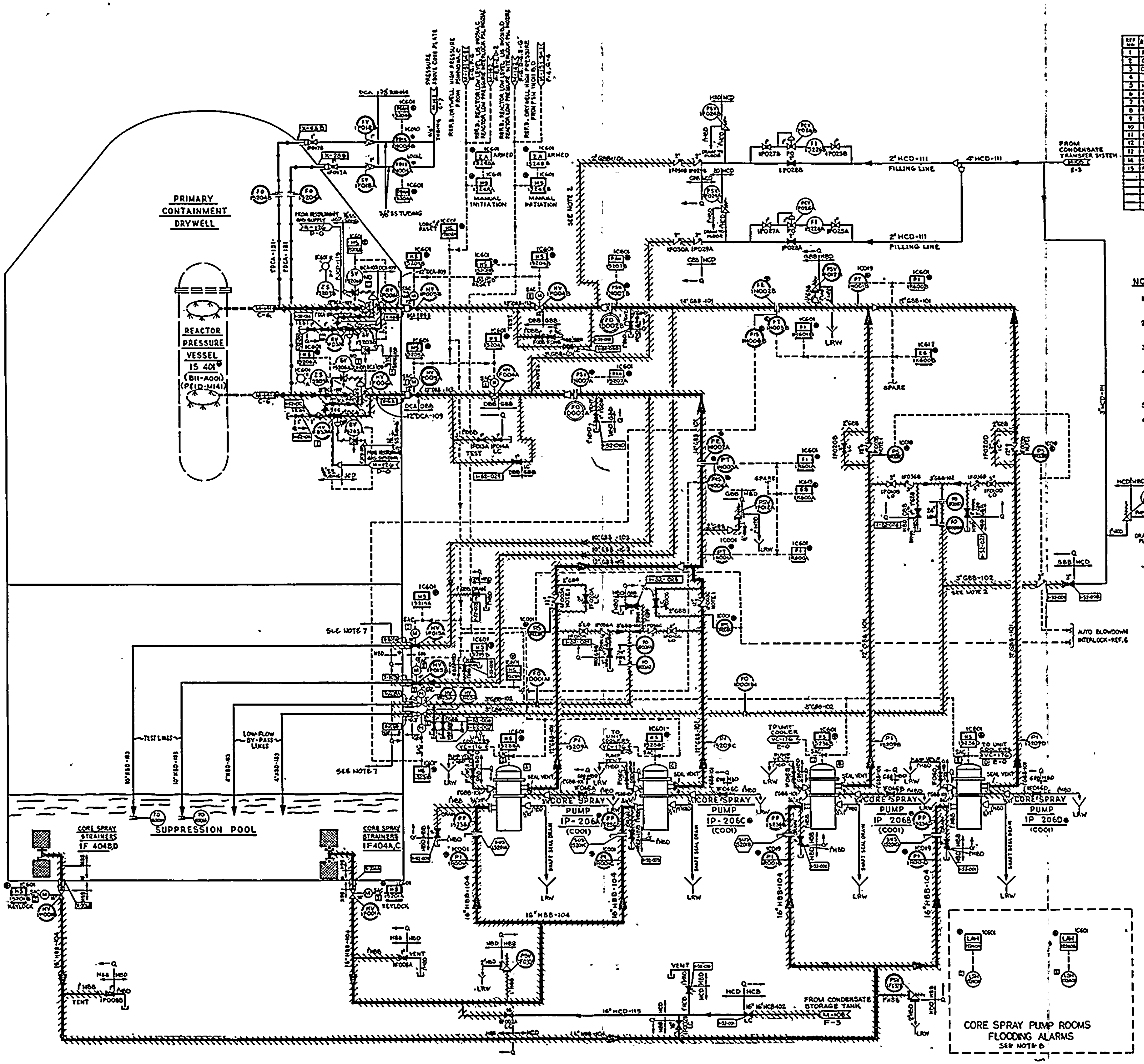
P&ID
HPIC TURBINE-PUMP

FIGURE 6.3-1b

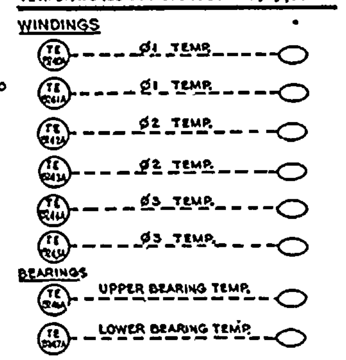
REF NO.	REFERENCE DRAWINGS & SPECIFICATIONS	BECHTEL NO.	PL. NO.	G.E. NO.
1	P&ID LEGEND	M-100	E-106205	117R367
2	CORE SPRAY SYST. FUNCT. CONTR. DIAG.	M-E21-3		729E611AE
3	COND. & RST WATER STORAGE & TRANS. P&ID	M-108		
4				
5	NUCLEAR BOILER P&ID	M-141	E-106246	761E250AE
6	NUCLEAR BOILER FUNCT. CONTR. DIAG.	[illegible]		791E279
7	NUCLEAR BOILER VESSEL INSTRUMENT P&ID	M-142	E-106247	
8	RESIDUAL HEAT REMOVAL P&ID	M-151	E-106256	761E232A
9	LIQUID RADWASTE - COLLECTION P&ID	M-161		
10	CLEANING OF PIPING & EQUIP.			22A2749
11	CORE SPRAY SYSTEM DESIGN SPEC.			22A3053
12	REACTOR BLDG LINER PLATE PENET.	C-272 C-279		
13	CONDENSATE P&ID	M-105	E-106210	
14	COND. & REFUELING WTR STORAGE P&ID	M-108	E-106213	
15	PROCESS SAMPLING P&ID	M-123	E-106228	



NOTES:

1. CHECK VALVES 1F003A THRU D SHALL BE LOCATED BELOW THE SUPPRESSION POOL WATER LEVEL (C-4, D-2).
2. FILLING LINES FROM CONDENSATE SYSTEM SHOULD BE CONNECTED TO HIGH POINT OF PIPING.
3. PIPING HIGH POINT VENTS AND LOW POINT DRAINS TO BE ADDED AS INDICATED
4. PIPING SHOULD BE CLEANED AND FLUSHED IN ACCORDANCE WITH REF. 10. AFTER FLUSHING, FILL THE LINES WITH DEMINERALIZED WATER.
5. THE G.E. MPL NUMBER FOR THIS SYSTEM IS E-21.
6. COMPUTER INPUTS TO BE ADDED LATER.
7. THE PORTION OF THE PIPE DESIGNATED WITH AN ARROW IS SEISMICALLY ANALYZED.
8. ALL FLOODING LEVEL SWITCHES AND DETECTORS ARE SEISMICALLY QUALIFIED.

CORE SPRAY PUMP "A" MOTOR BEARING & WINDING TEMPERATURES (TYP. FOR PUMPS B, C, & D).



NOTES CONT:

9. MISC. VALVE NO.'S INCLUDED ON THIS P&I.D. ARE 1-52-001 THRU 1-52-080.

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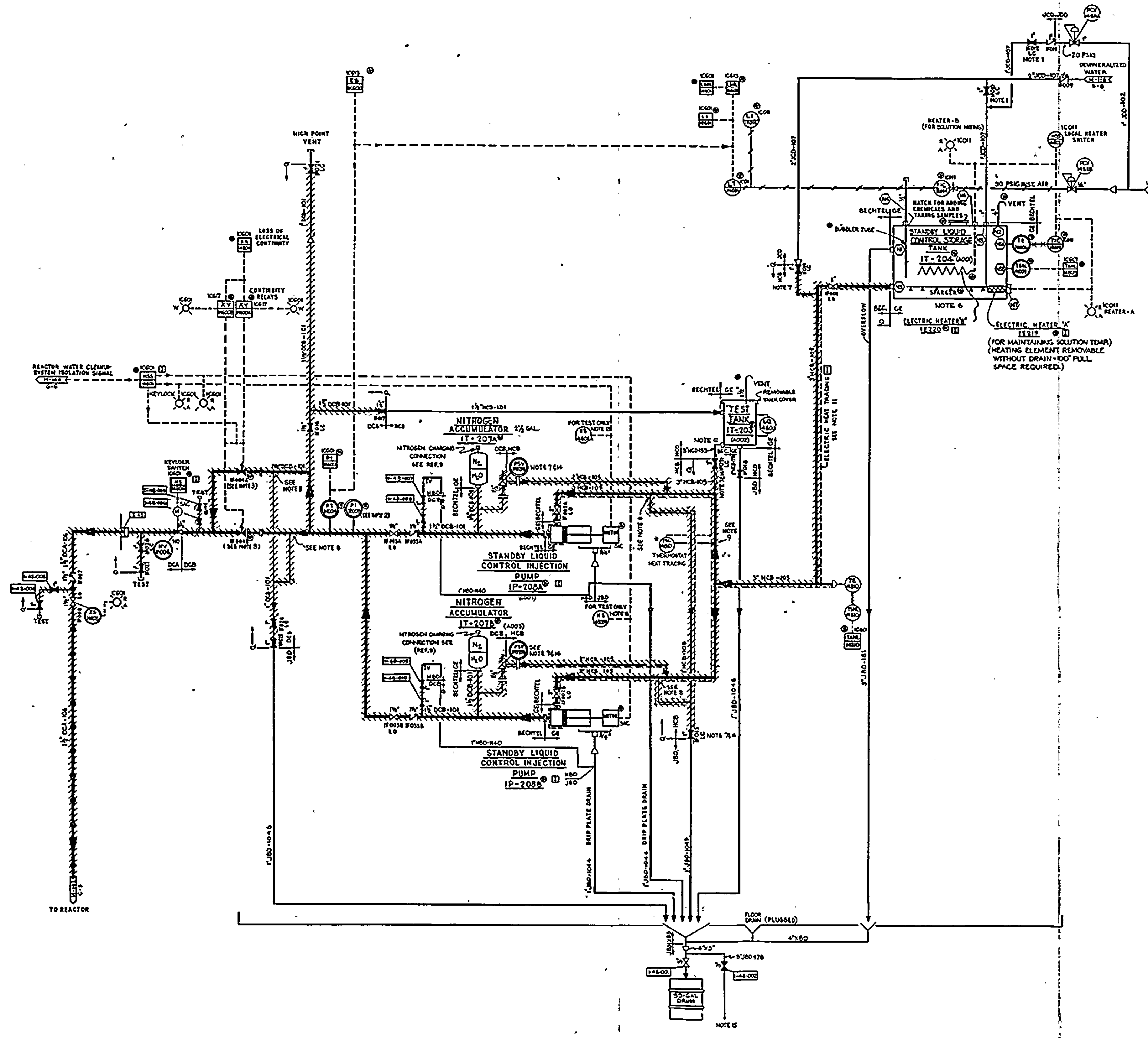
P&ID
CORE SPRAY

FIGURE 6.3-4



REF. NO.	REFERENCE DRAWINGS	BECHTEL NO.	PL. NO.	G.E.
1	NUCLEAR BOILER INSTR. P&ID	M-142	E-106247	
2	STANDBY LIQUID CONTROL FCD	M1-C41-14		851E886
3	COMPRESSED AIR P&ID	M-125	E-106230	
4	MAKE-UP DEMINERALIZER P&ID	M-118	E-106223	
5	P&ID LEGEND	M-100	E-106205	
6	STANDBY LIQUID CONTROL DESIGN SPEC.			[illegible]
7	REACTOR WATER CLEAN-UP P&ID	M-144	E-106249	
8	PROCESS INST. PIPING & TUBING SPEC.			22A1040AB
9	ACCUMULATOR INSTALLATION & ASS'Y	M1-C41-5		

NOTES:

1. ISOLATION VALVES F010 AND F012 SHALL BE LOCATED ABOVE THE TOP OF THE STORAGE TANK.
2. ORIENT PRESSURE INDICATOR 1R003 SO THAT IT CAN BE READ FROM VALVE 1F016 LOCATION (ZONE D-6).
3. IN ORDER TO SERVICE THESE VALVES AFTER FIRING, IT IS NECESSARY TO REMOVE A 6" SPOOL PIECE IMMEDIATELY UPSTREAM OF THE RESPECTIVE VALVE. EACH EXPLOSIVE VALVE IS FURNISHED WITH A MATING SOCKET WELDING TYPE FLANGE FOR SOCKET WELDING TO A 6" SPOOL PIECE.
4. THE GE MPL NUMBER FOR THIS SYSTEM IS C-41
5. PIPING HIGH POINT VENTS AND LOW POINT DRAINS ARE TO BE ADDED AT ALL SUCH HIGH OR LOW POINTS NOT SERVED BY EQUIPMENT VENTS AND DRAINS.
6. STORAGE AND TEST TANK ELEVATIONS SHALL BE SUCH AS TO ENSURE FLOODED SUCTION AT PUMP.
7. HAND VALVES F014, F015, F031 AND RELIEF VALVES PSV F029A AND B SHALL BE INSTALLED WITH AS SHORT PIPING RUNS TO HEADER HCB-105 AS POSSIBLE.
8. FLUSHING CONNECTIONS (SUPPLY AND DRAIN) SHALL BE LOCATED TO ALLOW FOR MAX. SYSTEM FLUSHING.
9. TEMPERATURE SENSED IN PUMP SUCTION PIPING AT LOCATION MOST REMOTE FROM STORAGE TANK DISCHARGE.
10. INSTRUMENT PIPING AND VALVING SHALL BE INSTALLED IN ACCORDANCE WITH REF. 8
11. STANDBY LIQUID CONTROL SYSTEM SHALL HAVE THERMOSTATICALLY CONTROLLED HEAT TRACING ON THE PUMP SUCTION LINE FROM STORAGE TANK.
12. PUMP SUCTION PIPING FROM THE STORAGE TANK AND THE TEST TANK SHALL CONFORM TO REQUIREMENTS IN REF. 6.
13. LOCATE HS-14508A&B LOCAL TO PUMPS & ORIENTED SO THAT THE TEST TANK LEVEL GLASS CAN BE READ FROM SWITCH LOCATION.
14. ALL PIPING CONNECTED TO THE SLC PUMP SUCTION PIPING SHALL BE MAINTAINED IN A FLOODED CONDITION TO THE FIRST NORMALLY SHUT VALVE TO PREVENT EVAPORATION AND PRECIPITATION. THE ELEVATION OF THESE LINES SHOULD BE BELOW THAT OF THE PUMP SUCTION LINE.
15. LOCATE 55 GALLON DRUM AND DRAIN LINE WITH-IN WASH DOWN AREA AT EL. 719'-1".
16. [illegible]
17. THE FOLLOWING MISCELLANEOUS VALVE NUMBERS HAVE BEEN ASSIGNED: 1-48-001 THROUGH 1-48-010.

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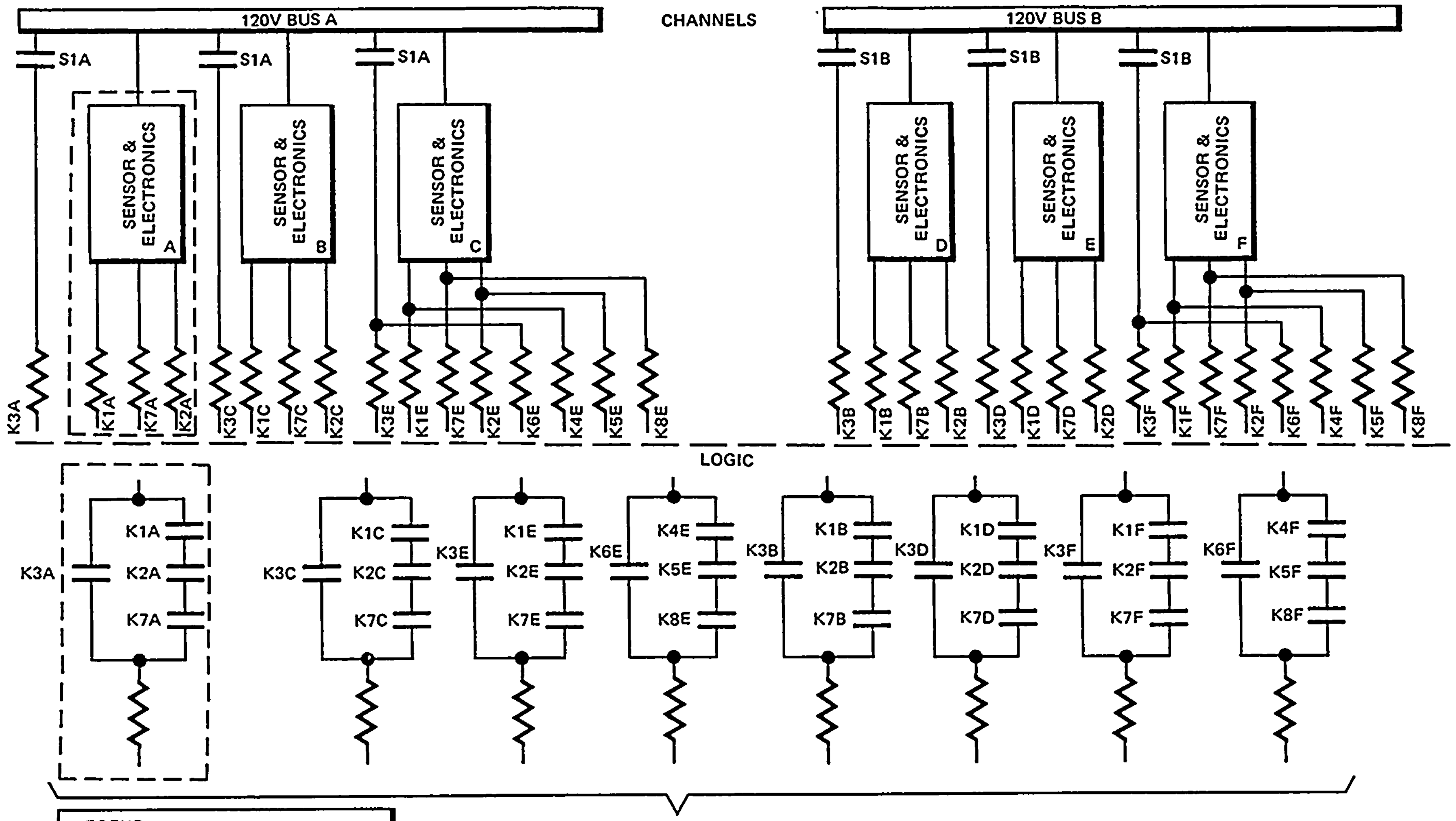
P&ID
STANDBY LIQUID
CONTROL SYSTEM
FIGURE 7.4-3



LEGEND:
K1 & K4 – INOPERATIVE TRIP
K2 & K5 – UPSCALE NEUTRON TRIP
K7 & K8 – UPSCALE THERMAL TRIP
S1A, B BYPASS SWITCH

NOTE: CONTACTS SHOWN IN TRIPPED, UNBYPASSED POSITION

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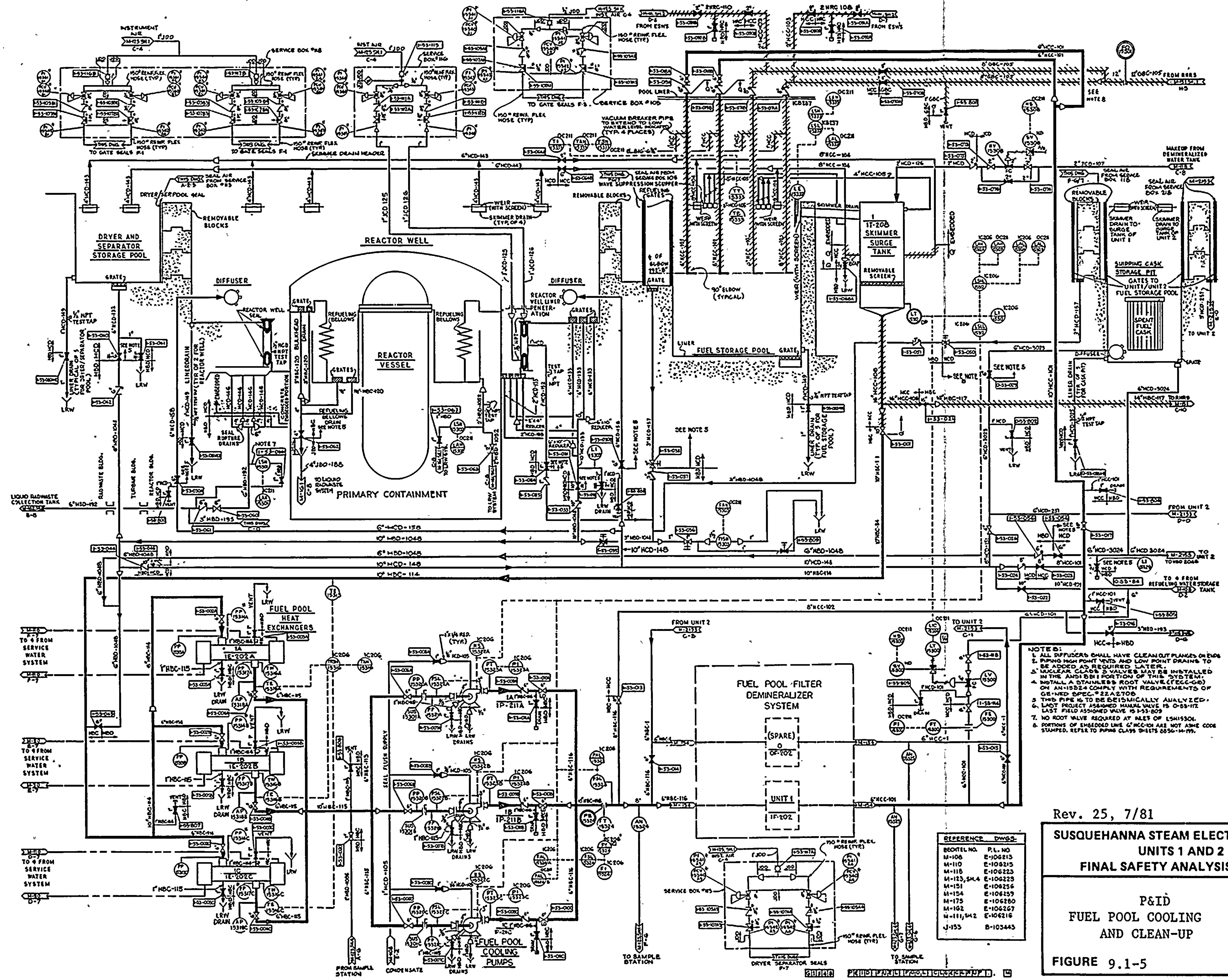
APRM CIRCUIT ARRANGEMENT FOR REACTOR PROTECTION SYSTEM INPUT

FIGURE 7.6-12



NOTES:

1. ALL DIFFUSERS SHALL HAVE CLEANOUT FLANGES ON ENDS
2. PIPING HIGH POINT VENTS AND LOW POINT DRAINS TO BE ADDED AS REQUIRED LATER.
3. NUCLEAR CLASS 3 VALVES MAY BE INSTALLED IN THE ANSI B31.1 PORTION OF THIS SYSTEM.
4. INSTALL A STAINLESS ROOT VALVE (1"ECC-GB) ON AN-15324 COMPLY WITH REQUIREMENTS OF GE-NED SPEC. #22A2708
5. THIS PIPE IS TO BE SEISMICALLY ANALYZED.
6. LAST PROJECT ASSIGNED MANUAL VALVE IS 0-53-112 LAST FIELD ASSIGNED VALVE IS 1-53-809
7. NO ROOT VALVE REQUIRED AT INLET OF LSH15301.
8. PORTIONS OF EMBEDDED LINE 6"HCC-101 ARE NOT ASME CODE STAMPED. REFER TO PIPING CLASS SHEETS 8856-M-199.

REFERENCE DWGS.

BECHTEL NO.	P.L. NO
M-108	E-106213
M-110	E-106215
M-118	E-106223
M-123, SH.4	E-106223
M-151	E-106256
M-154	E-106259
M-175	E-106280
M-162	E-106267
M-111, SH.2	E-106216
J-153	B-103443

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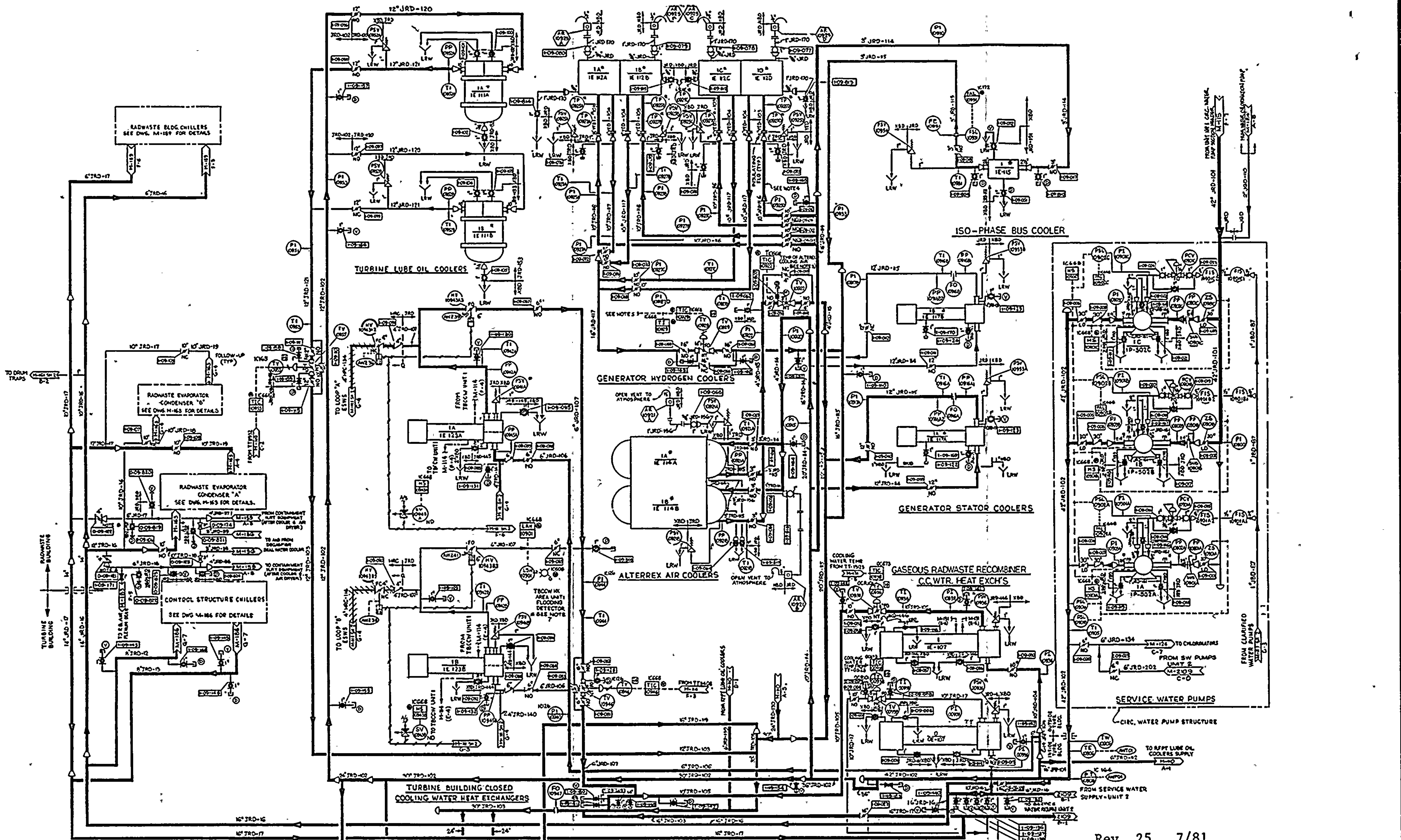
P&ID
FUEL POOL COOLING
AND CLEAN-UP

FIGURE 9.1-5



REFERENCE DWGS.	
BECHTEL NO.	PL. NO.
M-110	E-106215
M-111 SK2	E-106216 SK2
M-114	E-106219
M-115	E-106220
M-117	E-106222
M-124	E-106227
M-163	E 106268
M-131	E-106236
M-2109	E-105909
M-186	E-106291
M-189	E-106294
J-409	D-103521
J-109	B-103399
J-111	B-103401
M-2-E-5	FF-110650

NOTES:

1. REFER TO M-110 FOR OTHER HEAT EXCHANGERS IN THE SW SYSTEM.
2. STATOR COOLING WATER PUMPS 1P216A & 1P216B AND ASSOCIATED CONTROLS SHOWN ON GE DWG. #252R904 AND 112C 4437 (REF. 8856-M2-H4 & M2-E-5). IT-120 VENT AND DRAIN LINES ARE 1½" HBD-5060. SKID PAN DRAIN LINES ARE 1" HBD-5060
3. FROM TT-10192B ON LOOP DIAGRAM J-409 SHEET 4, TEMPERATURE ELEMENT TE-10192B (RTD6N) SUPPLIED BY GE-LSTD (REF. 8856-M2-M4(6)).
4. MISCELLANEOUS VALVES USED ON THIS P & I.D NUMBERED 1-09-001 TO 171, 003, 004, 012, TO 017 DELETED VALVES 1-09-025, 029, 051, 033 TO 035, 043 TO 046, 050, 053, 054, 063, 120, 126, 127, 133 & 134
5. FROM TT 10258 ON LOOP DIAGRAM J-409 SHEET 3, TEMPERATURE ELEMENT TE 10258 (OTG-3A) SUPPLIED BY GE-LSTD (REF. 8856-M2-E-5(3)).
6. ADDED VALVES NUMBERED 0-09-001 THRU 0-09-008, 0-09, 125
7. ALL FLOODING DETECTOR LEVEL SWITCHES AND LIGHTS ARE SEISMICALLY QUALIFIED.
8. THIS REDUCER SHALL BE LOCATED AS CLOSE AS POSSIBLE TO THE VALVE

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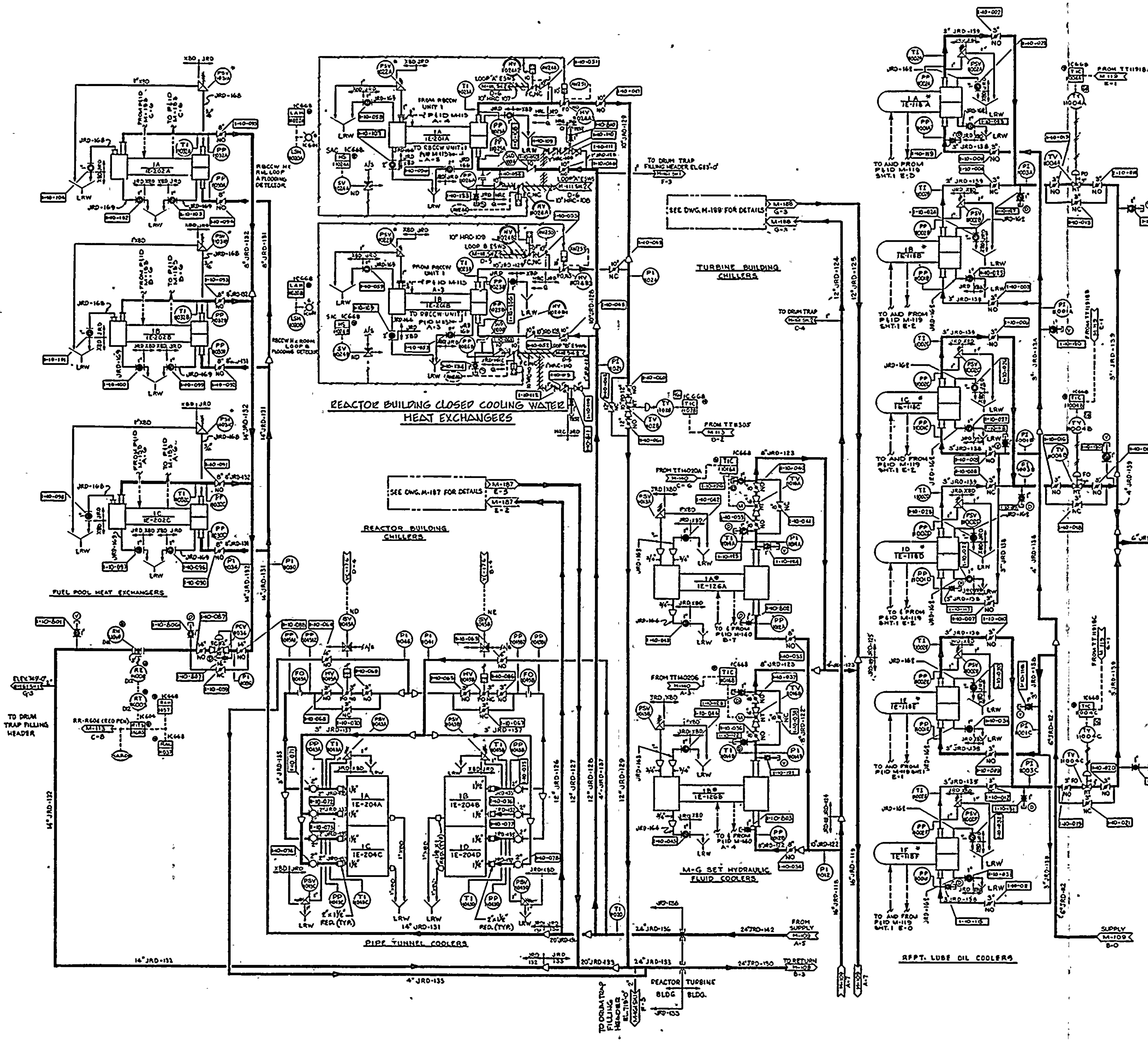
P&ID
SERVICE WATER

FIGURE 9.2-1a



REFERENCE DWGS.	
BECHTEL NO.	PL. NO.
M-109	E-106214
M-113	E-106218
M-125	E-106230
M-176	E-106281
M-187	E-106292
M-188	E-106293
M-153	E-106258
M-111 SH 2	E-106216
J-110	E-103400
M-140	E-106245
M-119 SH.1	E-106224 SH.1

NOTES:

1. SEE P&ID M-109 FOR SERVICE WATER PUMPS AND OTHER HEAT EXCHANGERS IN THIS SYSTEM.
2. MISCELLANEOUS VALVES NUMBERED 001-134 ADDED.

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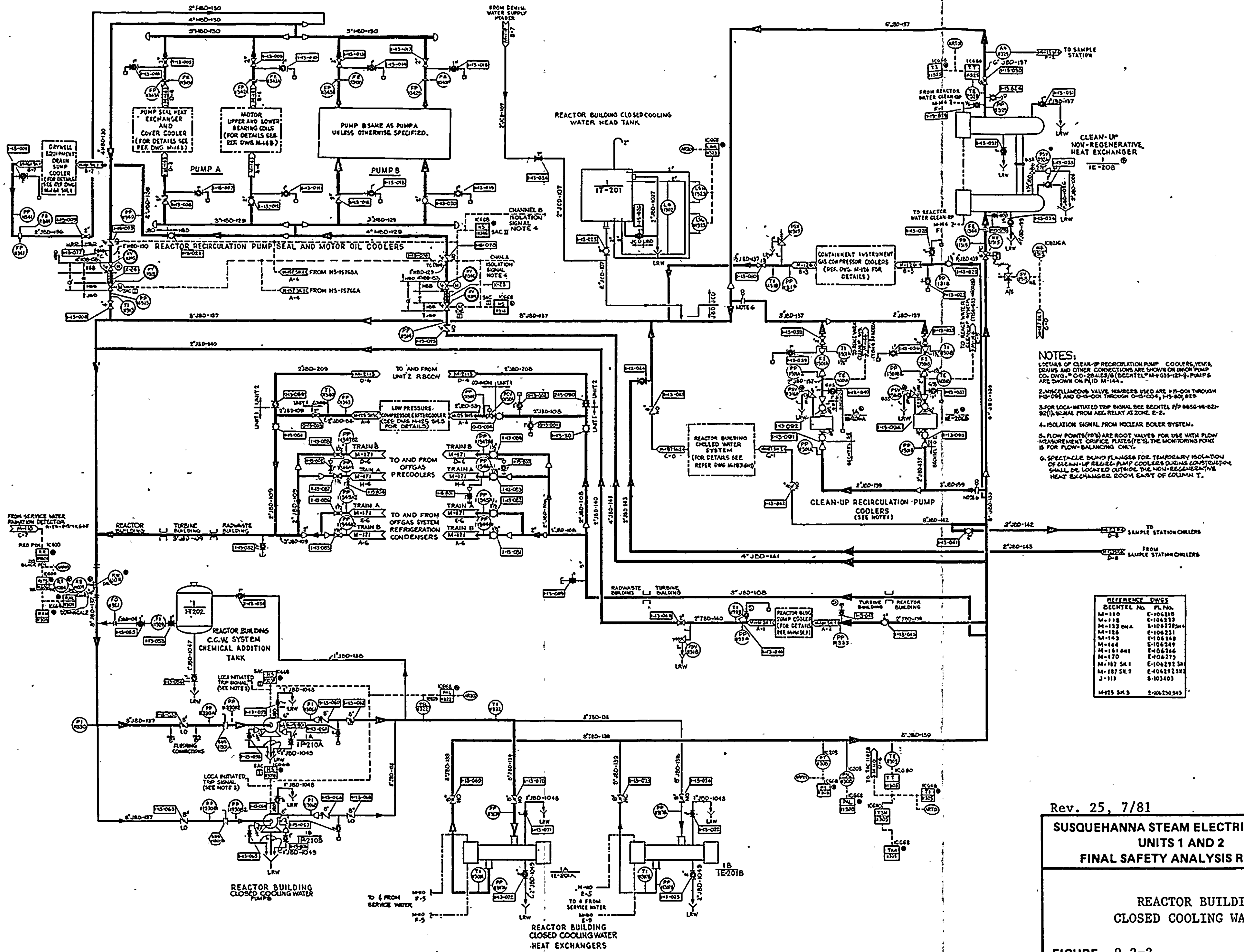
P&ID
SERVICE WATER

FIGURE 9.2-1b



NOTES:

1. DETAILS OF CLEAN-UP RECIRCULATION PUMP COOLERS, VENTS, DRAINS AND OTHER CONNECTIONS ARE SHOWN ON UNION PUMP CO. DWG.# C-D-284125/G (BECHTEL # M-1-G33-127-1). PUMPS ARE SHOWN ON P&ID M-144.

2. MISCELLANEOUS VALVE NUMBERS USED ARE 1-13-001 THROUGH 1-13-095 AND 0-13-001 THROUGH 0-13-004, 1-13-801, 829

3. FOR LOCA-INITIATED TRIP SIGNAL SEE BECHTEL F/D 8856-M-E21-92(1). SIGNAL FROM AUX. RELAY AT ZONE E-2.

4. ISOLATION SIGNAL FROM NUCLEAR BOILER SYSTEM.

5. FLOW POINTS (FP'S) ARE ROOT VALVES FOR USE WITH FLOW MEASUREMENT ORIFICE PLATES (FE'S). THE MONITORING POINT IS FOR FLOW-BALANCING ONLY.

6. SPECTACLE BLIND FLANGES FOR TEMPORARY ISOLATION OF CLEAN-UP RECIRC. PUMP COOLERS DURING CONSTRUCTION SHALL BE LOCATED OUTSIDE THE NON-REGENERATIVE HEAT EXCHANGER ROOM EAST OF COLUMN T.

REFERENCE DWGS

BECHTEL No.	PL No.
M-110	E-106215
M-118	E-106223
M-122 SH 4	E-106228 SH 4
M-126	E-106231
M-143	E-106248
M-144	E-106249
M-161 SH 1	E-106266
M-170	E-106275
M-187 SH 1	E-106292 SH 1
M-187 SH 2	E-106292 SH 2
J-113	B-103403
M-125 SH 3	E-106230, SH 3

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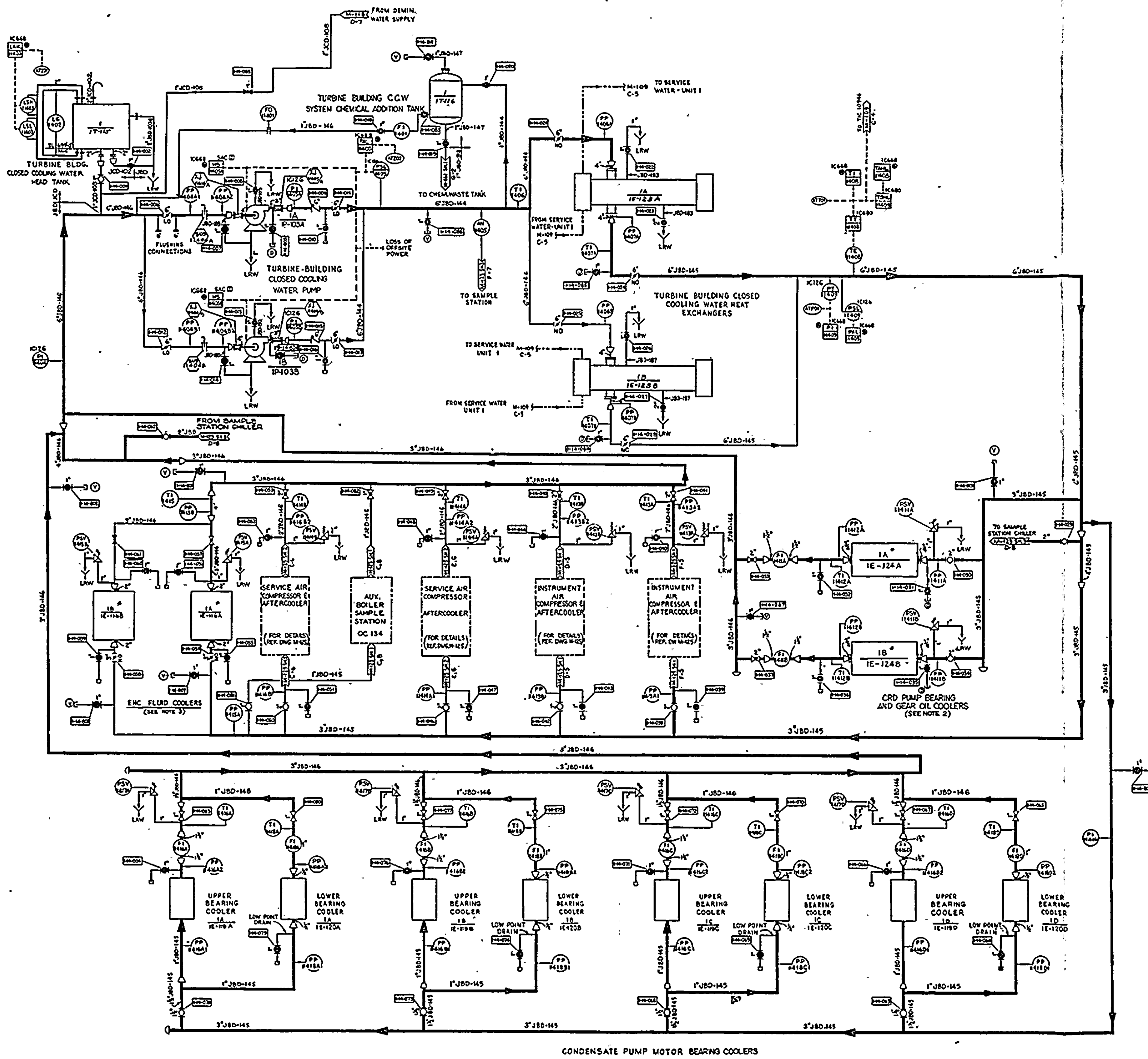
REACTOR BUILDING
CLOSED COOLING WATER

FIGURE 9.2-2



NOTES

1. ~~THERE IS ONLY ONE SERVICE AIR COMPRESSOR FOR UNIT 2.~~
2. DETAIL OF CRD PUMP COOLERS SHOWN ON UNION PUMP CO. DWG # OD-28672L BECHTEL # 8856-M1-OP53-2. PUMPS SHOWN ON P&ID M-146
3. DETAIL OF EHC FLUID COOLERS SHOWN ON GE DWG # 804E373.
4. FOR LOGIC DIAGRAM SEE J-114; LOOP DIAGRAM SEE J-409, J-414.
5. MISC VALVE NUMBERS USED: 1-14-001 THRU 14-087.

REFERENCE DWG'S.	
BECHTEL NO.	PL NO.
M-109	E-106214
M-118	E-106223
M-123 SH.3	E-106228 SH.3
M-125 SH.1	E-106230 SH.1
M-125 SH.2	E-106230 SH.2
M-161 SH.1	E-106266 SH.1
J-114	B-103404
J-409	D-103521 SH.2
J-414	D-103526 SH.2

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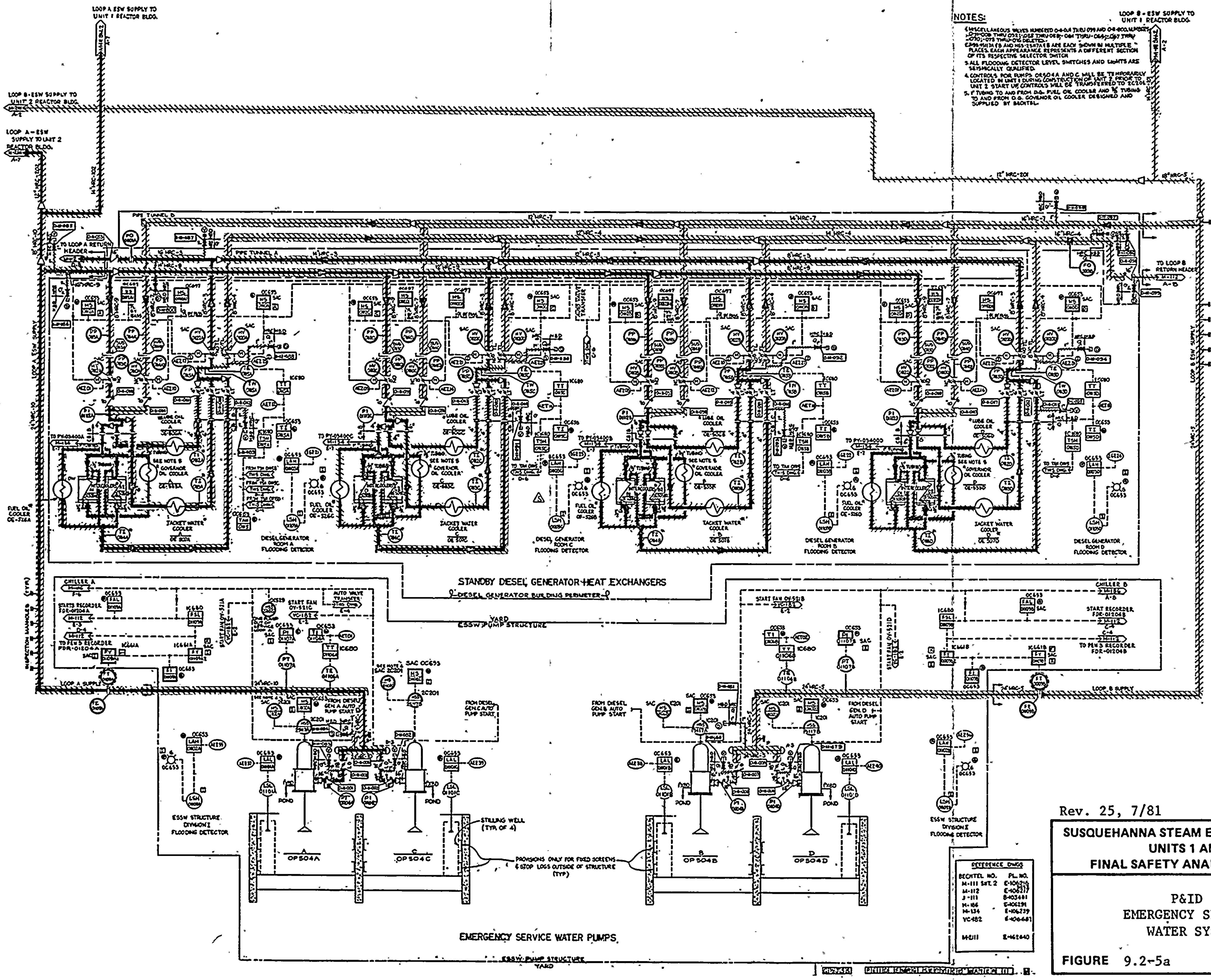
P&ID
TURBINE BUILDING
CLOSED COOLING WATER

FIGURE 9.2-3



REFERENCE DWGS

BECHTEL NO.	PL. NO.
M-111 SHT. 2	E-106216
M-112	E-106217
J-111	8-403481
M-186	E-106291
M-134	E-106239
VC-182	E-106681
M-2111	E-162640

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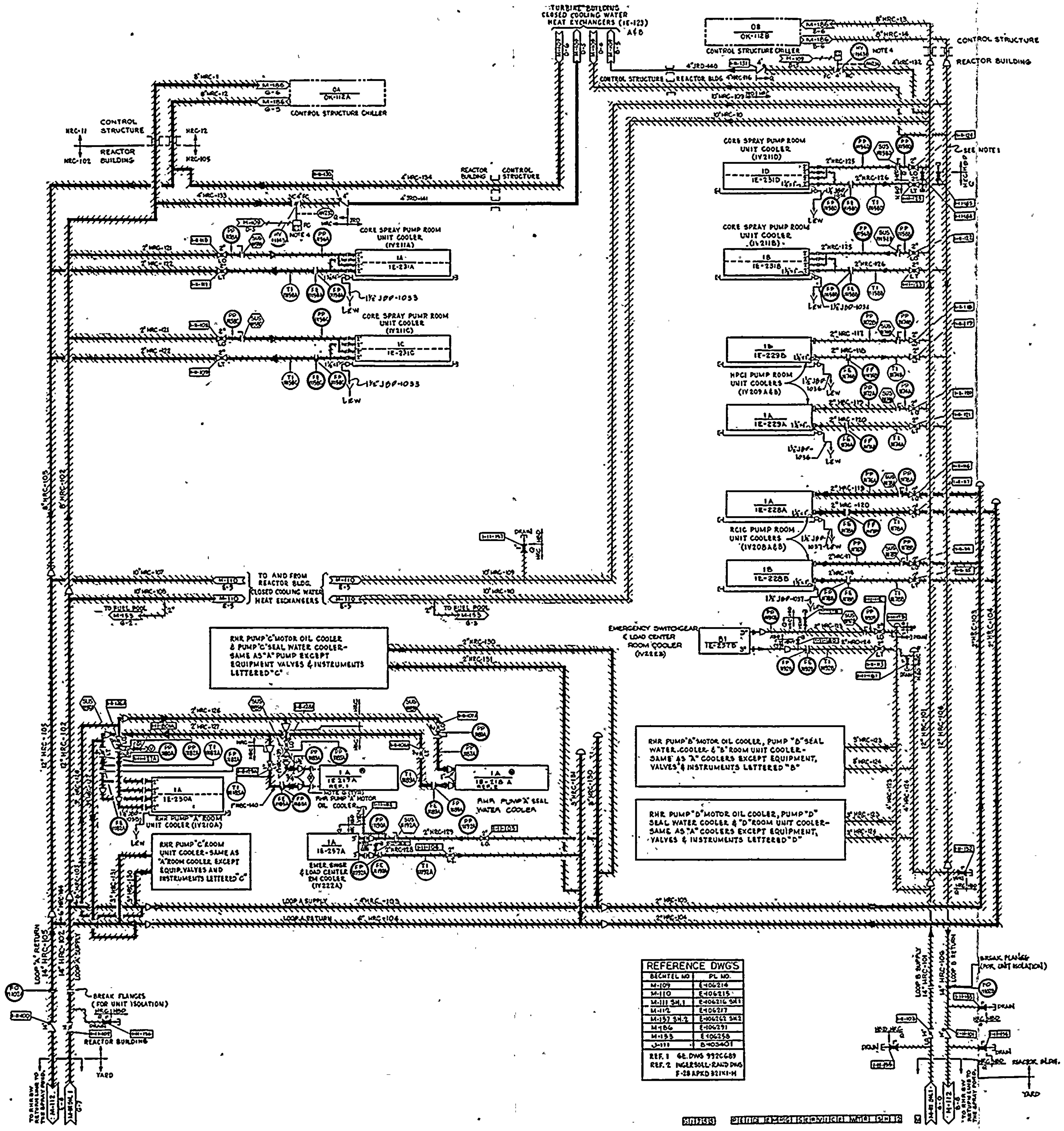
P&ID
EMERGENCY SERVICE
WATER SYSTEM

FIGURE 9.2-5a



REFERENCE DWG'S	
BECHTEL NO	PL NO.
M-109	E-106214
M-110	E-106215
M-111 SH.1	E-106216 SH1
M-112	E-106217
M-157 SH.2	E-106262 SH2
M-186	E-106291
M-153	E-106258
J-111	B-403401
REF. 1	GE. DWG 992C689
REF. 2	INGERSOLL-RAND DWG F-28 APKD 321K1-H

NOTES:

1. ALL HIGH POINT VENTING TO BE BY AUTOMATIC AIR RELEASE VALVES. HIGH POINT VENTING TO BE MINIMIZED.
2. FLOW POINTS FP-XXXX ARE ROOT VALVES FOR USE WITH THE FLOW MEASUREMENT ORIFICE PLATES FE-XXXX. THE MONITORING POINT IS FOR FLOW-BALANCING ONLY.
3. MISCELLANEOUS VALVES NUMBERED H-100 THRU 105, NUMBERS 0-H-156 THRU 177 DELETED
4. HV-11143A&B WILL BE SUPPLIED WITH TAG NO'S HV-10943A1&B1 SINCE THESE VALVES WERE ORIGINALLY SHOWN ON P.&I.D. M-109.
5. THERMOWELLS FOR ALL TEMPERATURE INDICATORS ON THIS DRAWING ARE QUALIFIED "Qp"
6. VENDOR SUPPLIED RED BRASS TO CARBON STEEL INSULATING KIT ¾" NPT.

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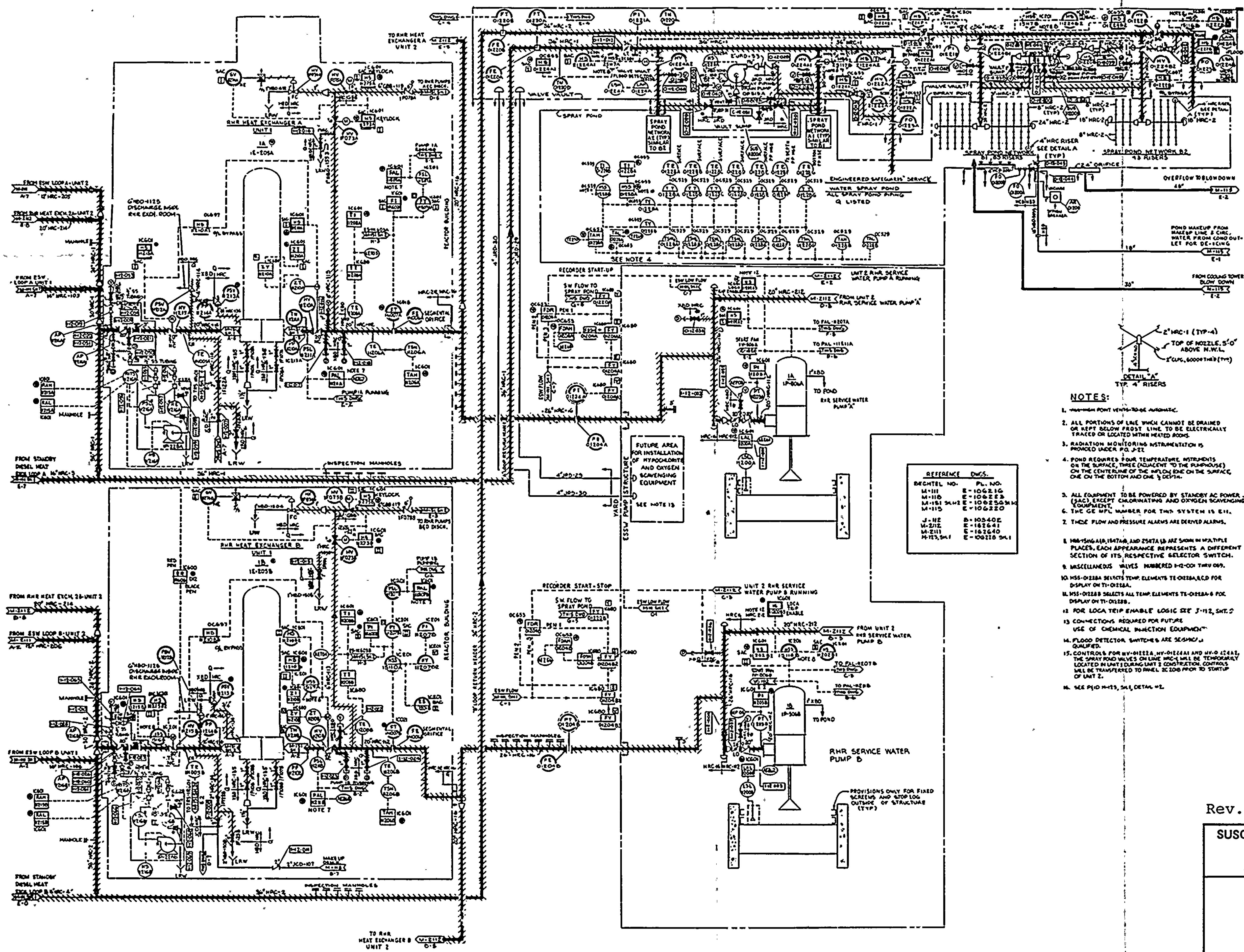
P&ID
EMERGENCY SERVICE
WATER SYSTEM

FIGURE 9.2-5b



DETAIL "A"
TYP. 4" RISERS

NOTES:

1. ~~ALL HIGH POINT VENTS TO BE AUTOMATIC.~~
2. ALL PORTIONS OF LINE WHICH CANNOT BE DRAINED OR KEPT BELOW FROST LINE TO BE ELECTRICALLY TRACED OR LOCATED WITHIN HEATED ROOMS.
3. RADIATION MONITORING INSTRUMENTATION IS PROVIDED UNDER P.O. J-22
4. POND REQUIRES FOUR TEMPERATURE INSTRUMENTS ON THE SURFACE, THREE (ADJACENT TO THE PUMPHOUSE) ON THE CENTERLINE OF THE INFLOW (ONE ON THE SURFACE, ONE ON THE BOTTOM AND ONE ½ DEPTH.
5. ALL EQUIPMENT TO BE POWERED BY STANDBY AC POWER. (SAC) EXCEPT CHLORINATING AND OXYGEN SCAVENGING EQUIPMENT.
6. THE GE MPL NUMBER FOR THIS SYSTEM IS E11.
7. THESE FLOW AND PRESSURE ALARMS ARE DERIVED ALARMS.
8. HSS-15416A&B, 15417A&B, AND 25417A&B ARE SHOWN IN MULTIPLE PLACES. EACH APPEARANCE REPRESENTS A DIFFERENT SECTION OF ITS RESPECTIVE SELECTOR SWITCH.
9. MISCELLANEOUS VALVES NUMBERED 1-12-001 THRU 069.
10. HSS-01228A SELECTS TEMP. ELEMENTS TE-01228A,B,C,D FOR DISPLAY ON TI-01228A.
11. HSS-01228B SELECTS ALL TEMP. ELEMENTS TE-01228A-G FOR DISPLAY ON TI-01228B.
12. FOR LOCA TRIP ENABLE LOGIC SEE J-112, SHT. 2
13. CONNECTIONS REQUIRED FOR FUTURE USE OF CHEMICAL INJECTION EQUIPMENT.
14. FLOOD DETECTOR SWITCHES ARE SEISMICALLY QUALIFIED.
15. CONTROLS FOR HV-01222A, HV-01224A1 AND HV-01224A2, THE SPRAY POND VALVES ON LINE HRC-1 WILL BE TEMPORARILY LOCATED IN UNIT 1 DURING UNIT 2 CONSTRUCTION. CONTROLS WILL BE TRANSFERRED TO PANEL 2C 2016 PRIOR TO STARTUP OF UNIT 2.
16. SEE P&ID M-123, SH.1, DETAIL #2.

REFERENCE DWGS.

BECHTEL NO.	PL. NO.
M-111	E-106216
M-110	E-106223
M-151 SH.2	E-106256 SH.2
M-115	E-106220
J-112	B-103402
M-2112	E-162641
M-2111	E-162640
M-123, SH.1	E-106228 SH.1

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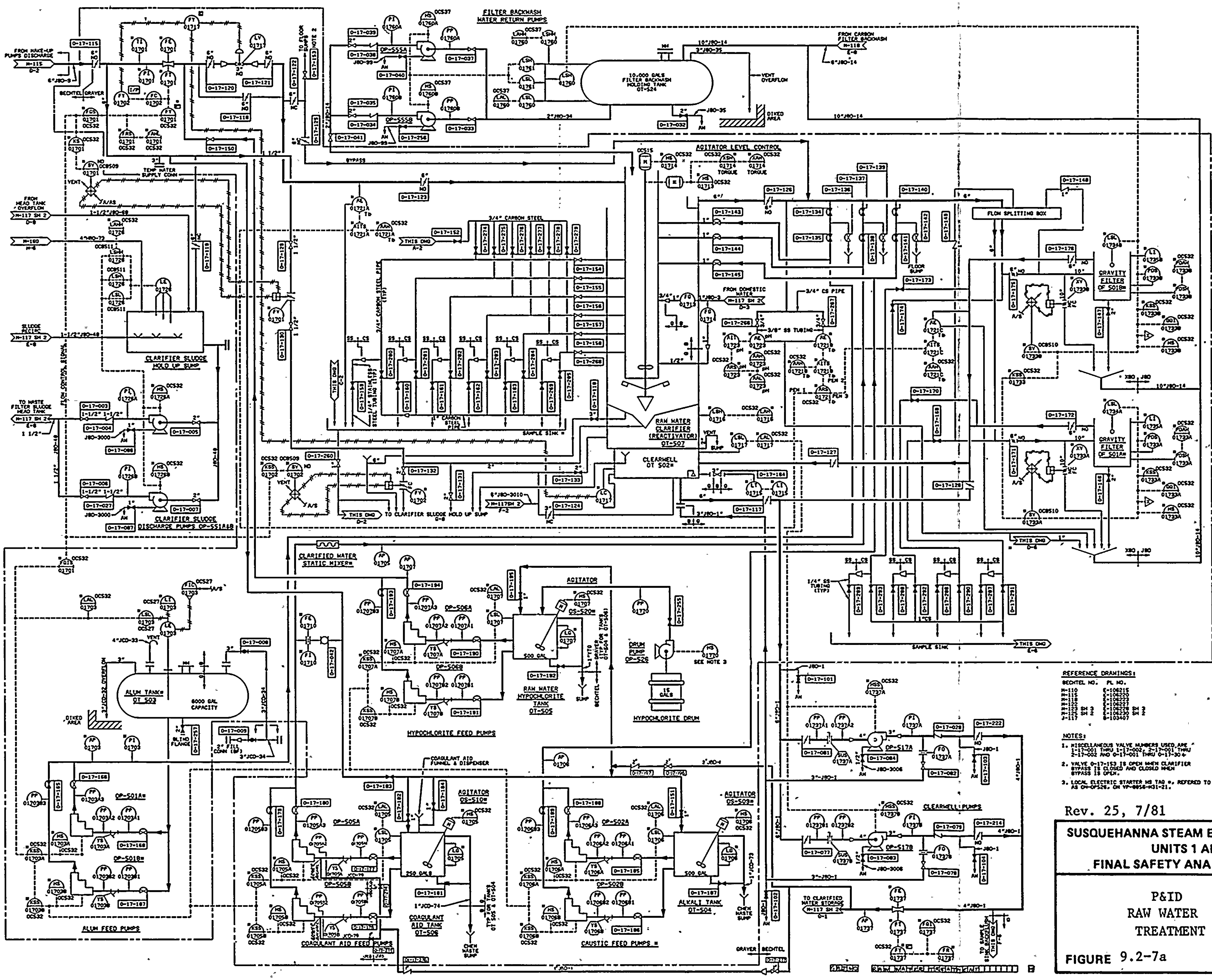
P&ID
RHR SERVICE WATER
SYSTEM

FIGURE 9.2–6



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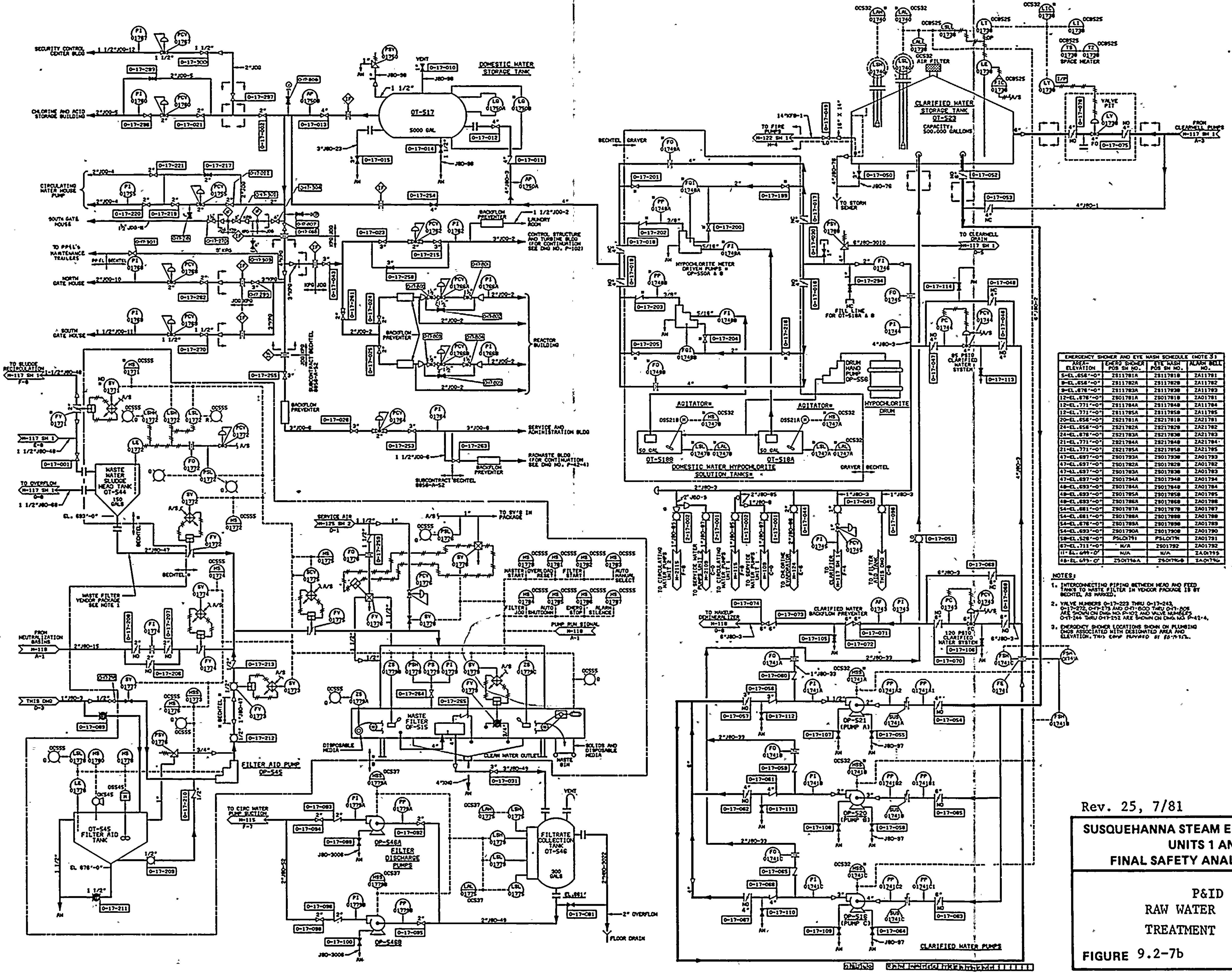
P&ID
RAW WATER
TREATMENT

FIGURE 9.2-7a



EMERGENCY SHOWER AND EYE WASH SCHEDULE (NOTE 3)			
AREA-ELEVATION	EMERG SHOWER POS SH NO.	EYE WASH POS SH NO.	ALARM BELL NO.
5-EL.656'-0"	ZS11781A	ZS11781B	ZA11781
9-EL.656'-0"	ZS11782A	ZS11782B	ZA11782
9-EL.676'-0"	ZS11783A	ZS11783B	ZA11783
12-EL.676'-0"	ZS01781A	ZS01781B	ZA01781
12-EL.771'-0"	ZS11784A	ZS11784B	ZA11784
12-EL.771'-0"	ZS11785A	ZS11785B	ZA11785
20-EL.656'-0"	ZS21781A	ZS21781B	ZA21781
24-EL.656'-0"	ZS21782A	ZS21782B	ZA21782
24-EL.676'-0"	ZS21783A	ZS21783B	ZA21783
21-EL.771'-0"	ZS21784A	ZS21784B	ZA21784
21-EL.771'-0"	ZS21785A	ZS21785B	ZA21785
47-EL.697'-0"	ZS01793A	ZS01793B	ZA01793
47-EL.697'-0"	ZS01782A	ZS01782B	ZA01782
47-EL.697'-0"	ZS01783A	ZS01783B	ZA01783
47-EL.697'-0"	ZS01794A	ZS01794B	ZA01794
48-EL.693'-0"	ZS01784A	ZS01784B	ZA01784
48-EL.693'-0"	ZS01785A	ZS01785B	ZA01785
48-EL.693'-0"	ZS01786A	ZS01786B	ZA01786
54-EL.681'-0"	ZS01787A	ZS01787B	ZA01787
54-EL.681'-0"	ZS01788A	ZS01788B	ZA01788
54-EL.676'-0"	ZS01789A	ZS01789B	ZA01789
54-EL.693'-0"	ZS01790A	ZS01790B	ZA01790
58-EL.528'-0"	PSL01791	PSL01791	ZA01791
67-EL.711'-0"	N/A	ZS01792	ZA01792
11-EL.699-0"	N/A	N/A	ZA01795
48-EL.699-0"	ZS01796A	ZS01796B	ZA01796

NOTES:

1. INTERCONNECTING PIPING BETWEEN HEAD AND FEED TANKS TO WASTE FILTER IN VENDOR PACKAGE IS BY BECHTEL AS MARKED.
2. VALVE NUMBERS 0-17-223 THRU 0-17-243, 0-17-272, 0-17-273 AND 0-17-800 THRU 0-17-805 ARE SHOWN ON DWG NO. P-102 AND VALVE NUMBERS 0-17-244 THRU 0-17-252 ARE SHOWN ON DWG NO. P-42-4.
3. EMERGENCY SHOWER LOCATIONS SHOWN ON PLUMBING DWGS ASSOCIATED WITH DESIGNATED AREA AND ELEVATION. THIS COMP PROVIDED BY [illegible].

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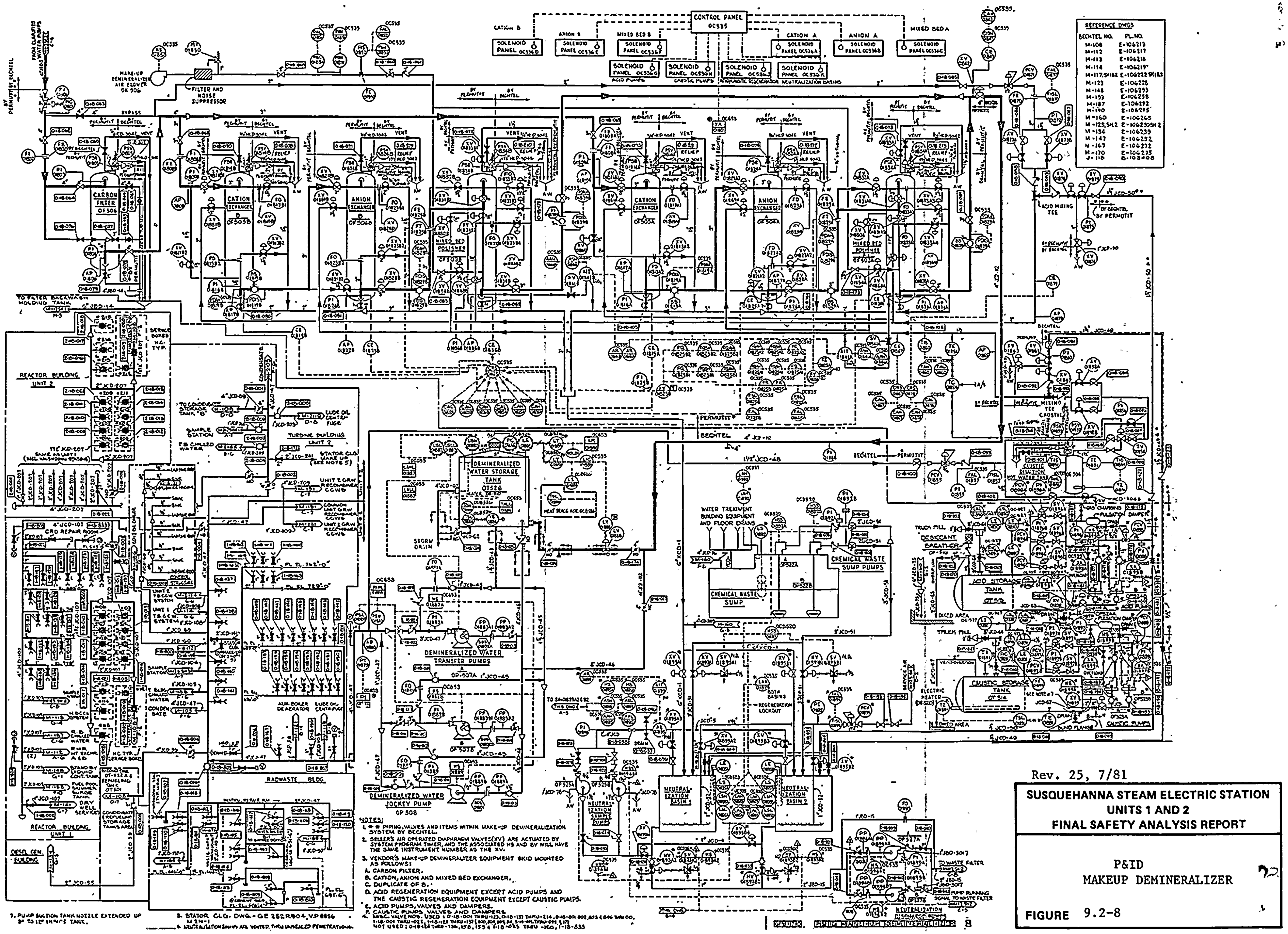
P&ID
RAW WATER
TREATMENT

FIGURE 9.2-7b



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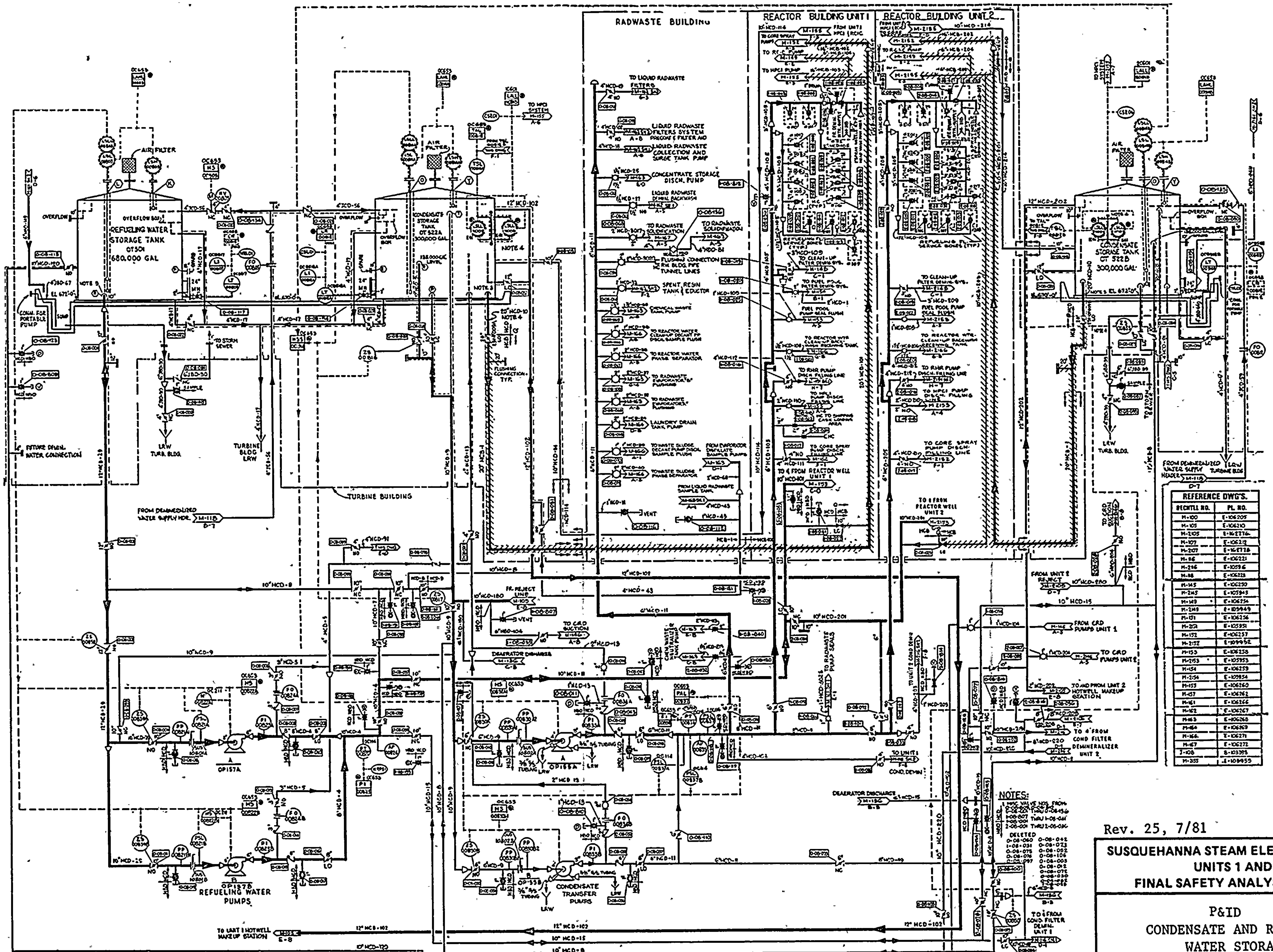
P&ID
MAKEUP DEMINERALIZER

FIGURE 9.2-8



REFERENCE DWG'S.	
BECHTEL NO.	PL NO.
M-100	E-106205
M-105	E-106210
M-2105	E-162776
M-107	E-106212
M-207	E-162778
M-116	E-106221
M-216	E-105916
M-118	E-106223
M-145	E-106250
M-245	E-105945
M-149	E-106254
M-249	E-105949
M-151	E-106256
M-251	E-105951
M-152	E-106257
M-2152	E-109952
M-153	E-106258
M-2153	E-105953
M-154	E-106259
M-2154	E-105954
M-155	E-106260
M-157	E-106262
M-161	E-106266
M-162	E-106267
M-163	E-106268
M-164	E-106269
M-166	E-106271
M-167	E-106272
J-108	8-103398
M-2155	E-109955

NOTE 2.
HV 00806 AND HS 00806 REVISED ON P&I.D. M-108 AS HV 10817 & HS 10817. HV 00807 & HS 00807 REVISED TO P&I.D. M-2105

NOTE 3
A VORTEX BREAKER SHALL BE PROVIDED INSIDE THE TANK AT THE NOZZLE FOR 20"HCD-10.

NOTE 4
THE HEAT TRACING FOR LSLL N002, LSLL N003, 20"HCD-10, 20"HCD-1 SHALL BE REDUNDANT CLASS 1E POWERED (FOR BOTH UNIT 1 AND UNIT 2)

NOTE 5
AN 8" NOZZLE SHALL BE INSERTED INTO THE TANK AT 10"HCD-29 TO PREVENT FLOW RESTRICTION DUE TO ICE FORMATION AT THE TANK WALL.

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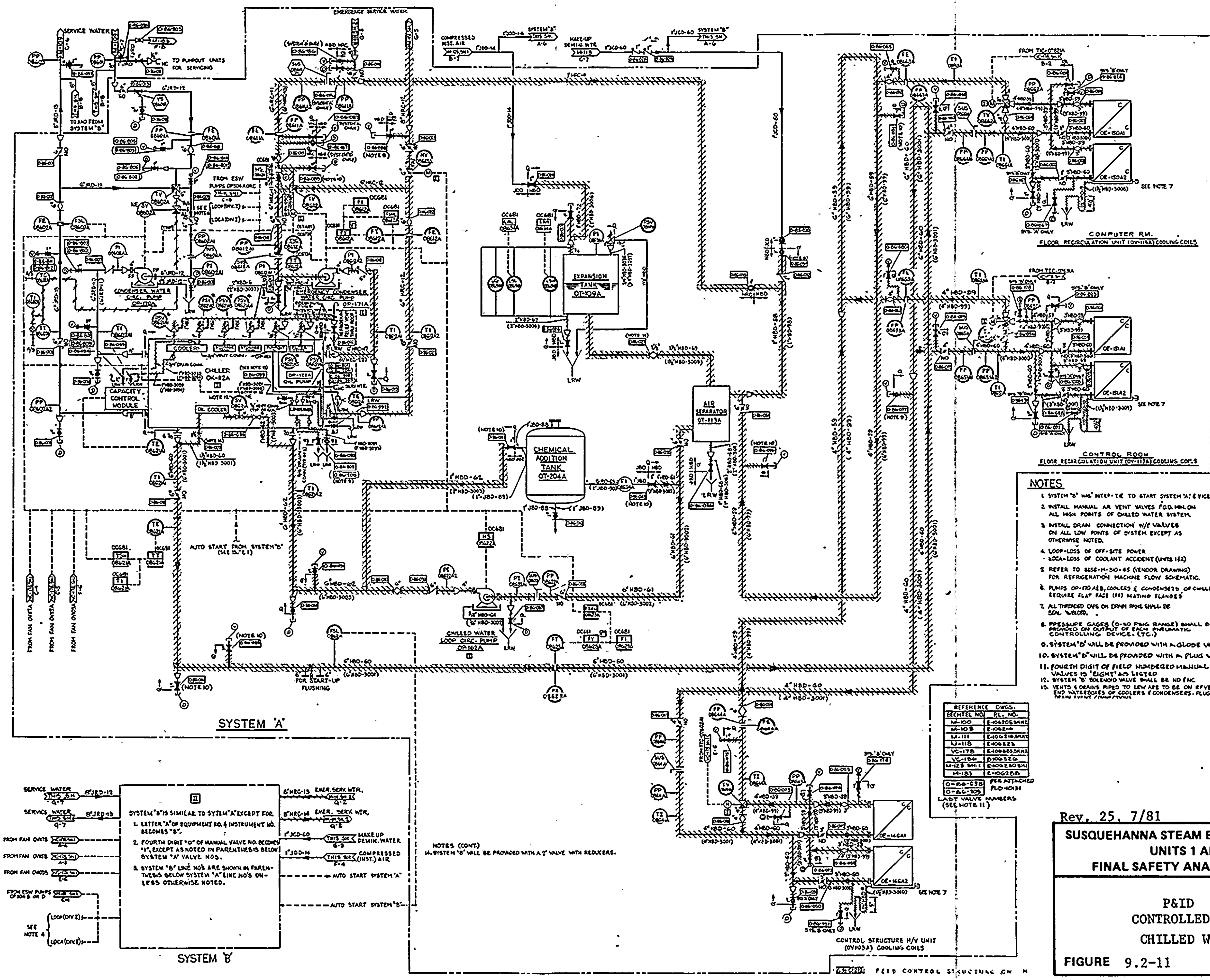
P&ID
CONDENSATE AND REFUELING WATER STORAGE

FIGURE 9.2-9



NOTES

1. SYSTEM "B" HAS INTER-TIE TO START SYSTEM "A" & VICE VERSA.
2. INSTALL MANUAL AIR VENT VALVES F.O.D. MIN. ON ALL HIGH POINTS OF CHILLED WATER SYSTEM.
3. INSTALL DRAIN CONNECTION W/ VALVES ON ALL LOW POINTS OF SYSTEM EXCEPT AS OTHERWISE NOTED.
4. LOOP-LOSS OF OFF-SITE POWER
   LOCA-LOSS OF COOLANT ACCIDENT (UNITS 1&2)
5. REFER TO 8856-M-310-65 (VENDOR DRAWING) FOR REFRIGERATION MACHINE FLOW SCHEMATIC.
6. PUMPS OP-170 A&B, COOLERS & CONDENSERS OF CHILLERS REQUIRE FLAT FACE (FF) MATING FLANGES
7. ALL THREADED CAPS ON DRAIN PANS SHALL BE SEAL WELDED.
8. PRESSURE GAGES (0-30 PSIG RANGE) SHALL BE PROVIDED ON OUTPUT OF EACH PNEUMATIC CONTROLLING DEVICE. (TC.)
9. SYSTEM "B" WILL BE PROVIDED WITH A GLOBE VALVE
10. SYSTEM "B" WILL BE PROVIDED WITH A PLUG VALVE
11. FOURTH DIGIT OF FIELD NUMBERED MANUAL VALVES IS 'EIGHT' AS LISTED
12. SYSTEM "B" SOLENOID VALVE SHALL BE NO & NC
13. VENTS & DRAINS PIPED TO LRW ARE TO BE ON REVERSE END WATERBOXES OF COOLERS & CONDENSERS. PLUG UNUSED DRAIN & VENT CONNECTIONS

NOTES (CONT)
14. SYSTEM "B" WILL BE PROVIDED WITH A 2" VALVE WITH REDUCERS.

REFERENCE DWGS.	
BECHTEL NO.	P.L. NO.
M-100	E-106205 SH. 1&2
M-109	E-106214
M-111	E-106216, SH. 1&2
M-118	E-106223
VC-178	E-106683, SH. 1&2
VC-186	D-106326
M-125 SH. 1	E-106230 SH. 1
M-183	E-106288

0-86-038	PER ATTACHED
0-86-209	FLD-10131

LAST VALVE NUMBERS (SEE NOTE 11)

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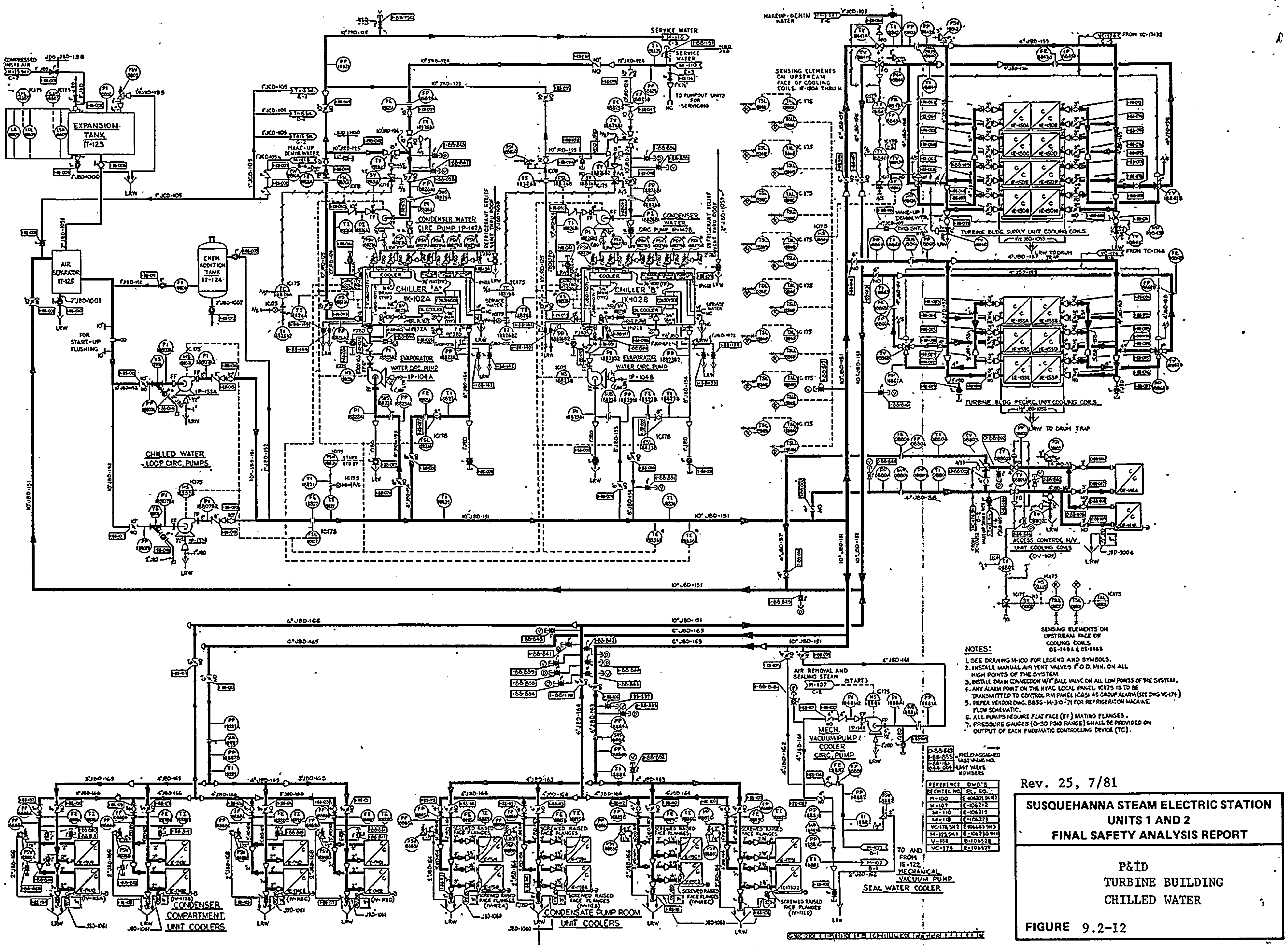
P&ID
CONTROLLED STRUCTURE
CHILLED WATER

FIGURE 9.2-11



NOTES:

1. SEE DRAWING M-100 FOR LEGEND AND SYMBOLS.
2. INSTALL MANUAL AIR VENT VALVES 1" O.D. MIN. ON ALL HIGH POINTS OF THE SYSTEM
3. INSTALL DRAIN CONNECTION W/1" BALL VALVE ON ALL LOW POINTS OF THE SYSTEM.
4. ANY ALARM POINT ON THE HVAC LOCAL PANEL 1C175 IS TO BE TRANSMITTED TO CONTROL RM PANEL 1C651 AS GROUP ALARM (SEE DWG VC-174)
5. REFER VENDOR DWG. 8856-M-310-71 FOR REFRIGERATION MACHINE FLOW SCHEMATIC.
6. ALL PUMPS REQUIRE FLAT FACE (FF) MATING FLANGES.
7. PRESSURE GAUGES (0-30 PSIG RANGE) SHALL BE PROVIDED ON OUTPUT OF EACH PNEUMATIC CONTROLLING DEVICE (TC).

REFERENCE DWG'S	
BECHTEL NO.	PL. NO.
M-100	E-106205 SH 1-2
M-107	E-106212
M-110	E-106215
M-118	E-106223
VC-178, SH 2	E-106683 SH 3
M-125, SH 1	E-106230 SH 1
V-158	B-106528
VC-174	B-106579

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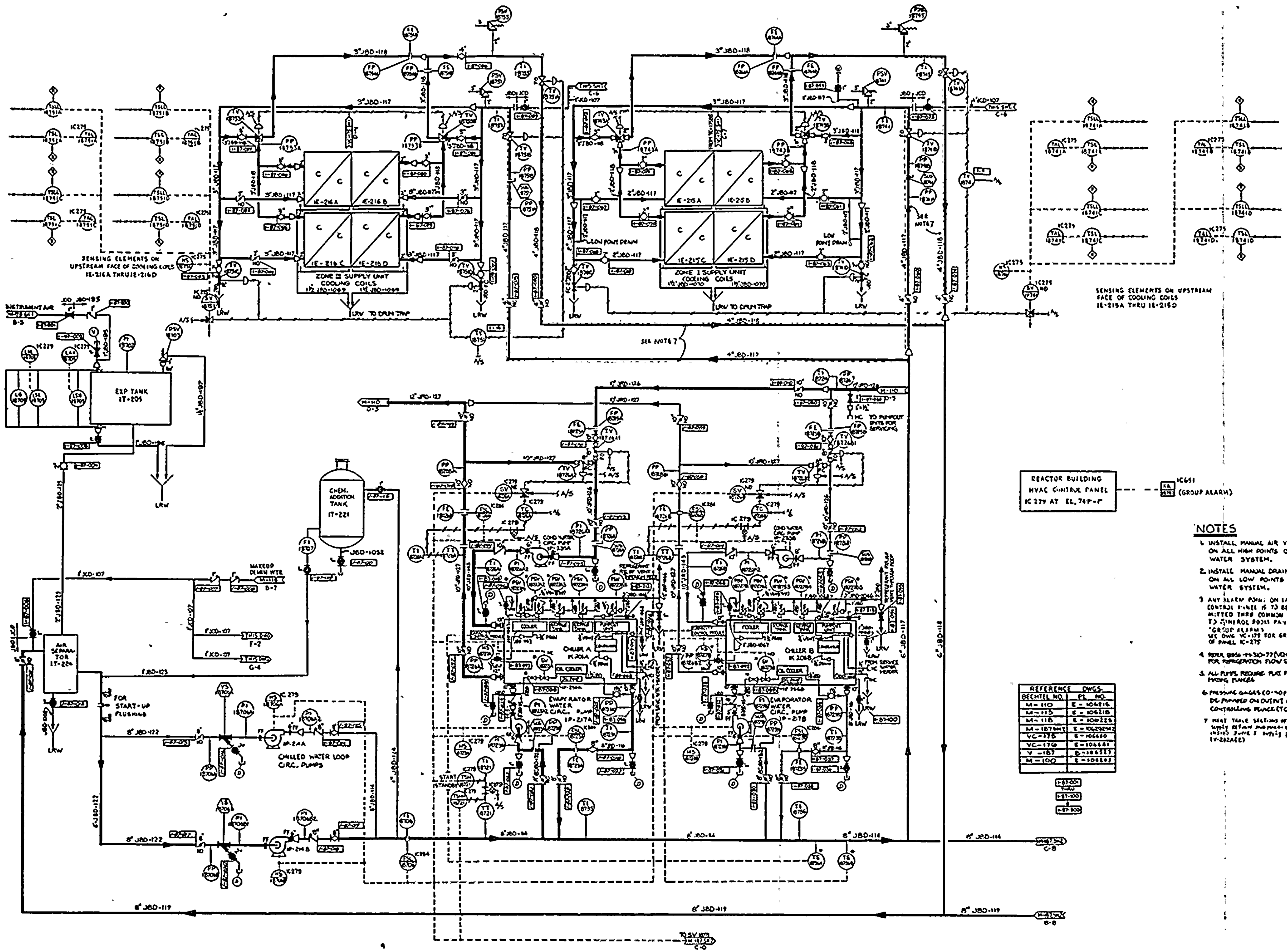
P&ID
TURBINE BUILDING
CHILLED WATER

FIGURE 9.2-12



NOTES

1. INSTALL MANUAL AIR VENT VALVES ON ALL HIGH POINTS OF THE CHILLED WATER SYSTEM.
2. INSTALL MANUAL DRAIN VALVES ON ALL LOW POINTS OF THE CHILLED WATER SYSTEM.
3. ANY ALARM POINT ON EACH LOCAL CONTROL PANEL IS TO BE TRANSMITTED THRU COMMON CHANNEL TO CONTROL ROOM PANEL 1C651 (GROUP ALARM) SEE DWG VC-175 FOR GROUP ALARM OF PANEL 1C-275
4. REFER 8856-44-30-77 (VENDOR DWG) FOR REFRIGERATION FLOW SCHEMATIC.
5. ALL PUMPS REQUIRE FLAT FACE (FF) MATING FLANGES
6. PRESSURE GAGES (0-30 PSIG RANGE) SHALL BE PROVIDED ON OUTPUT OF EACH PNEUMATIC CONTROLLING DEVICE (TC)
7. HEAT TRACE SECTIONS OF CHILLED WATER SUPPLY RETURN AND MAKE-UP WATER LINE INSIDE ZONE I SUPPLY PLENUM (FAN 1V-202A&B)

REFERENCE DWGS.	
BECHTEL NO.	PL NO.
M-110	E-106216
M-115	E-106210
M-116	E-106228
M-187 SH 2	E-106292 SH 2
VC-175	E-106680
VC-176	E-106681
V-187	D-106527
M-100	E-106205

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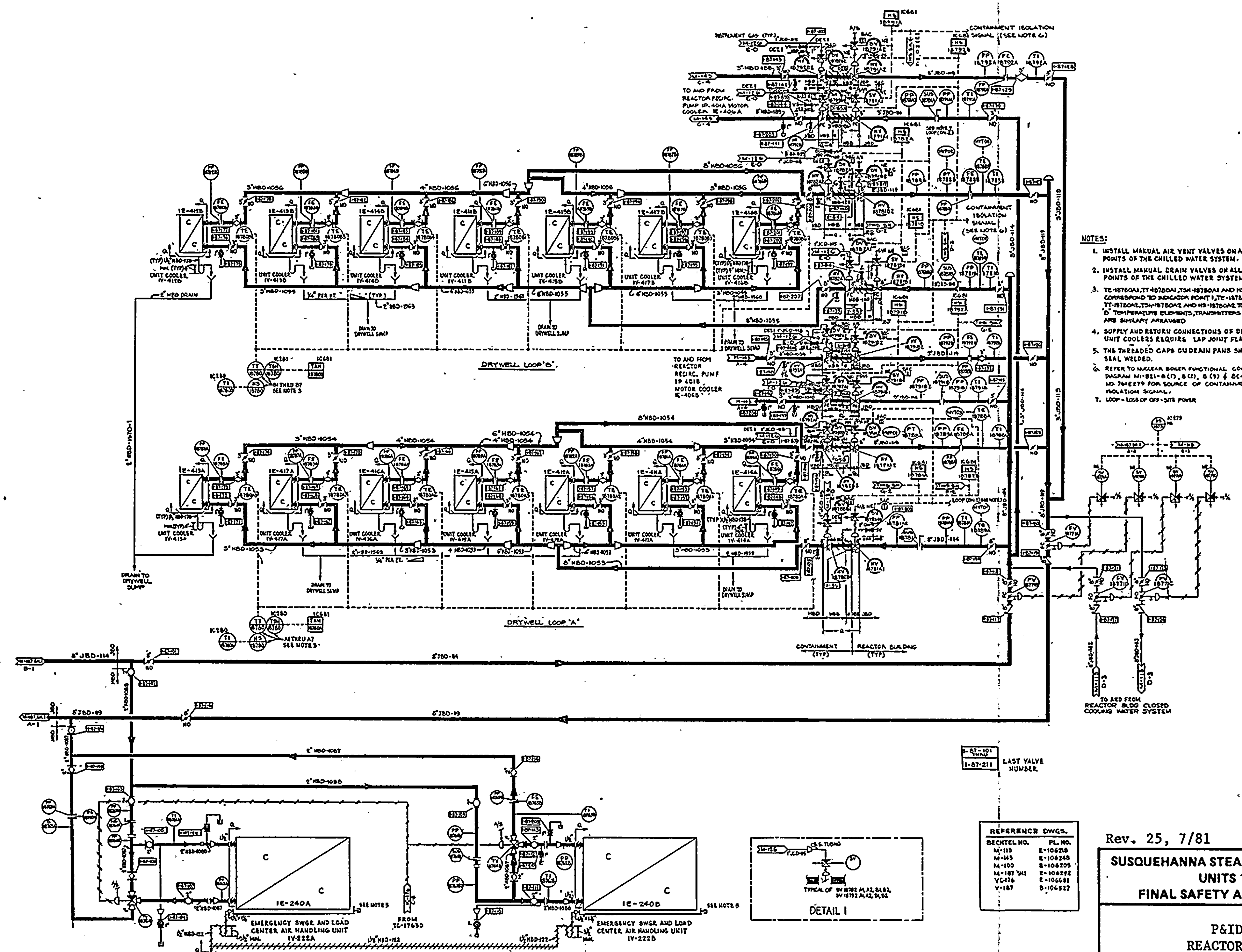
P&ID
REACTOR BUILDING
CHILLED WATER

FIGURE 9.2–13a



NOTES:

1. INSTALL MANUAL AIR VENT VALVES ON ALL HIGH POINTS OF THE CHILLED WATER SYSTEM.
2. INSTALL MANUAL DRAIN VALVES ON ALL LOW POINTS OF THE CHILLED WATER SYSTEM.
3. TE-18780A1, TT-18780A1, TSH-18780A1 AND HS-18780A1 CORRESPOND TO INDICATOR POINT 1, TE-18780A2, TT-18780A2, TSH-18780A2 AND HS-18780A2 TO POINT 2 ETC. "D" TEMPERATURE ELEMENTS, TRANSMITTERS AND SWITCHES ARE SIMILARY ARRANGED
4. SUPPLY AND RETURN CONNECTIONS OF DRYWELL UNIT COOLERS REQUIRE LAP JOINT FLANGES.
5. THE THREADED CAPS ON DRAIN PANS SHALL BE SEAL WELDED.
6. REFER TO NUCLEAR BOILER FUNCTIONAL CONTROL DIAGRAM M1-B21-8(1), 8(2), 8(3) & 8(4), GE. NO. 761E279 FOR SOURCE OF CONTAINMENT ISOLATION SIGNAL.
7. LOOP = LOSS OF OFF-SITE POWER

REFERENCE DWGS.

BECHTEL NO.	PL. NO.
M-113	E-106218
M-143	E-106248
M-100	E-106205
M-187 SH1	E-106292
VC-176	E-106681
V-187	B-106527

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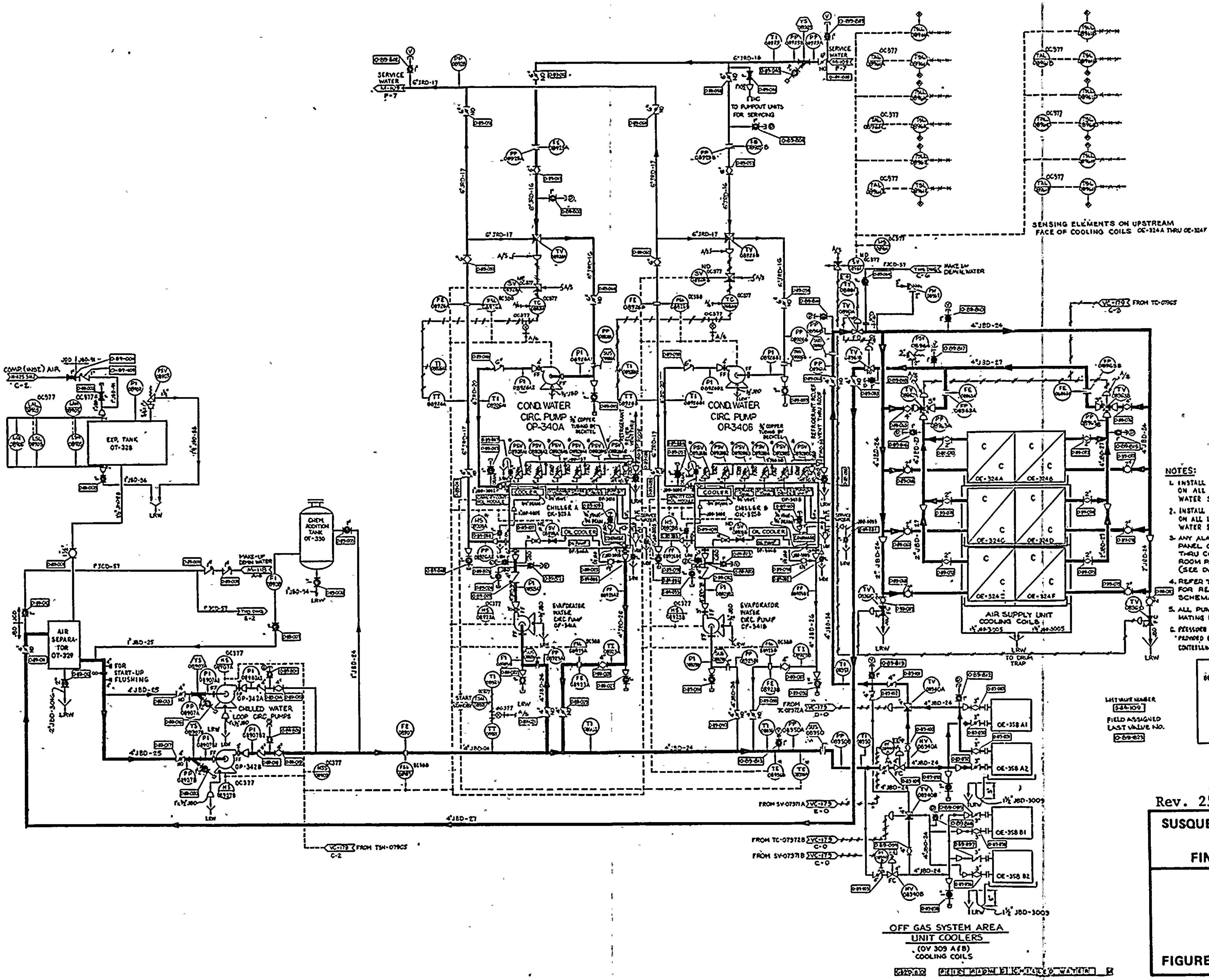
P&ID
REACTOR BUILDING
CHILLED WATER SYSTEM

FIGURE 9.2-13b



NOTES:

1. INSTALL MANUAL AIR VENT VALVES ON ALL HIGH POINTS OF THE CHILLED WATER SYSTEM.
2. INSTALL MANUAL DRAIN VALVES ON ALL LOW POINTS OF THE CHILLED WATER SYSTEM.
3. ANY ALARM POINT ON LOCAL CONTROL PANEL OC377 IS TO BE TRANSMITTED THRU COMMON CHANNEL TO CONTROL ROOM PANEL OC653 (SEE DWG. VC-178)
4. REFER TO VENDOR DWG. 8856-M310-220 FOR REFRIGERATION MACHINE FLOW SCHEMATIC.
5. ALL PUMPS REQUIRE FLAT FACE (FF) MATING FLANGES.
6. PRESSURE GAGES (0-30 PSIG RANGE) SHALL BE PROVIDED ON OUTPUT OF EACH PNEUMATICALLY CONTROLLING DEVICE (TC)

REFERENCE DWGS.	
BECHTEL NO.	PL. NO.
M-189	6-106214
M-118	6-106223
M-125	6-106230
V-169	6-106329
VC-179	6-106684
M-100	6-106209

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P&ID
RADWASTE BUILDING
CHILLED WATER

FIGURE 9.2-14

7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

7.7.1 DESCRIPTION

This subsection discusses instrumentation controls of systems whose functions are not essential for the safety of the plant and permits an understanding of the way the reactor and important subsystems are controlled. The systems include:

(1) Reactor vessel - instrumentation NSSS

(2) Reactor manual control system - instrumentation and controls, NSSS

(3) Recirculation flow control system - instrumentation and controls NSSS

(4) Reactor feedwater system - instrumentation and controls NSSS

(5) Pressure regulator and turbine - generator system - instrumentation and controls non-NSSS

(6) Neutron monitoring system - TIP

(7) Process computer system - instrumentation NSSS

(8) Neutron monitoring system - traversing in-core probe NSSS

(9) Reactor water cleanup system - instrumentation and controls NSSS

(10) Refueling interlocks system 17

(11) Rod block monitor system 18

(12) Nuclear Pressure Relief System - instrumentation & controls

(13) Loose parts monitoring system 22

7.7.1.1 Reactor Vessel - Instrumentation

Figures 5.1-3a and 5.1-3b show the instrument numbers, arrangements of the sensors, and sensing equipment used to monitor the reactor vessel conditions. Because the reactor vessel sensors used for safety systems, engineered safeguards, and control systems are described and evaluated in other portions of this document, only the sensors that are not required for those systems are described in this subsection.

7.7.1.1.1 System Identification

7.7.1.1.1.1 General

The purpose of the reactor vessel instrumentation is to monitor the key reactor vessel operating variables during plant operation.

These instruments and systems are used to provide the operator with information during normal plant operation, startup and shutdown. They are monitoring devices and provide no active power control or safety functions.

7.7.1.1.1.2 Classification

The systems and instruments discussed in this subsection are designed to operate under normal and peak operating conditions of system pressures and ambient pressures and temperatures and are classified as not related to safety.

7.7.1.1.1.3 Reference Design

Table 7.1-2 lists the reference design information. The reactor vessel instrumentation is an operational system and has no safety function. Therefore, there are no safety design differences between this system and those of the reference design facilities. This system is functionally identical to the referenced system.

7.7.1.1.2 Power Sources

The systems and instruments discussed in this subsection are powered from the instrument bus.

7.7.1.1.3 Equipment Design

The instrument sensing lines that the various pressure and level sensors are connected to slope downward from the vessel to the instrument rack with a minimum slope of 1/2-in/ft for those lines which cannot be sloped at the normal minimum of 1 in/ft (including allowance for piping sag), so that air traps are not formed. The instrument lines are self-venting back to the reactor vessel.

7.7.1.1.3.1 Circuit Description

7.7.1.1.3.1.1 Reactor Vessel Temperature

The reactor vessel temperature is determined on the basis of reactor coolant temperature. Temperatures needed for operation and for compliance with the technical specification operating limits are obtained from one of several sources, depending on the operating condition. During normal operation, either reactor pressure and/or the inlet temperature of the coolant in the recirculation loops can be used to determine the vessel temperature. Below the operating span of the resistance temperature detectors in the recirculation loop and above 212°F, the vessel pressure is used for determining the temperature. Below 212°F, the vessel coolant, and thus the vessel temperature, is reasonably well shown by the reactor water cleanup system inlet temperature. These three sources of input are most conveniently available from the process computer. During normal operation, vessel thermal transients are limited via operational constraints on parameters other than temperature.

7.7.1.1.3.1.2 Reactor Vessel Water Level

Figure 7.7-1 shows the water level range and the vessel penetration for each water level range. The instruments that sense the water level are strictly differential pressure devices calibrated to be accurate at a specific vessel pressure and liquid temperature condition. The following is a description of each water level range shown on Figure 7.7-1.

(1) Shutdown water level range: This range is used to monitor the reactor water level during the shutdown condition when the reactor system is flooded for maintenance and head removal. The water level measurement design is the condensate reference chamber leg type that is not compensated for changes in density. The vessel temperature and pressure condition that is used for the calibration is 0 psig and 120°F water in the vessel. The two vessel instrument penetrations elevations used for this water level measurement are located at the top of the RPV head and the instrument tap just below the bottom of the dryer skirt.

(2) Upset water level range: This range is used to monitor the reactor water when the level of the water goes off the narrow range scale on the high side. The design and vessel taps are the same as outlined above. The vessel pressure and temperature condition for accurate indication is at the normal operating point. The upset water level is continuously indicated by a recorder in the control room. The upset range and narrow range recorders are located in close proximity of each other. The upset range upper limit is higher than the narrow range upper limit. Therefore when the indication goes off scale in the upscale direction on the narrow range recorder, water level indication may be read immediately from the upset range recorder. Further information as to the range and main control room indication is discussed in Subsection 7.7.1.4.

(3) Narrow water level range: This range uses for its RPV taps at the elevation near the top of the dryer skirt and the taps at an elevation near the bottom of the dryer skirt. The zero of the instrument is the bottom of the dryer skirt and the instruments are calibrated to be accurate at the normal operating point. The water level measurement design is the condensate reference chamber type, is not density compensated, and uses differential pressure devices as its primary elements. The feedwater control system uses this range for its water level control and indication inputs. For more information as to the range, trip points, number of channels, and control room indication, see the discussion on the feedwater control system, Subsection 7.7.1.4.

(4) Wide water level range: This range uses for its RPV taps at the elevation near the top of the dryer skirt and the taps at an elevation near the top of the active fuel. The zero of the instrument is the bottom of the dryer skirt and the instruments are calibrated to be accurate at the normal power operating point. The water level measurement design is the condensate reference type, is not density compensated, and uses differential pressure devices as its primary elements. Wide range water level is displayed on two redundant recorders located in the main control room.

(5) Fuel zone water level range: This range uses for its RPV taps at the elevation near the top of the dryer skirt and the taps at the jet pump diffuser skirt. The zero of the instrument is the top of the active fuel and the instruments are calibrated to be accurate at 0 psig and saturated condition. The water level design is the condensate reference type, is not density compensated, and uses differential pressure devices as

its primary element. These instruments provide input water level indication.

The condensate reference chamber for the Narrow range, Wide Range, and Fuel Zone water level range is common as discussed in Section 7.3.

In order to decouple the change in measured water level with changes in drywell temperature, the elevation drop from RPV penetration to the drywell penetration will remain uniform for the narrow range and wide range water level instrument lines.

Reactor water level instrumentation that initiates safety systems and engineered safeguards systems is discussed in Sections 7.2 and 7.3. Reactor water level instrumentation that is used as part of the feedwater control system is discussed in Subsection 7.7.1.4. The reactor water level that pertains to this subsection is used to monitor the reactor water level during the shutdown conditions when the reactor system is flooded for maintenance and head removal. The water level design is the condensate chamber reference leg type that is not compensated for change in density. The vessel condition that will provide accurate water level information is 0 psig pressure and ambient temperature. The range of the instrument is from the bottom of the feedwater control operating range to a level over the top of the reactor vessel head.

7.7.1.1.3.1.3 Reactor Core Hydraulics

A differential pressure transmitter indicates core plate pressure drop by measuring the pressure difference between the core inlet plenum and the space just above the core support assembly. The instrument sensing line used to determine the pressure below the core support assembly attaches to the same reactor vessel tap that is used for the injection of the liquid from the standby liquid control system. An instrument sensing line is provided for measuring pressure above the core support assembly. The differential pressure of the core plate is recorded in the main control room.

Another differential pressure device indicates the jet pump developed head by measuring the pressure difference between the pressure above the core and the pressure below the core plate. This indication is indicated in the main control room.

7.7.1.1.3.1.4 Reactor Vessel Pressure

Pressure switches/transducers, indicators, and transmitters detect reactor vessel internal pressure from the same instrument lines used for measuring reactor vessel water level.

The following list shows the subsection in which the reactor vessel pressure measuring instruments are discussed:

(1) Pressure switches/transducers for initiating scram, and pressure switches/transducers for bypassing the main steamline isolation valve closure are discussed in Subsection 7.2.1.1.

(2) Pressure switches/transducers used for HPCI, CS, LPCI, and ADS are discussed in Subsection 7.3.1.1a.1.

(3) Pressure transmitters/transducers and recorders used for feedwater control are discussed in Subsection 7.7.1.4.

(4) Pressure transmitters/transducers that are used for pressure recording are discussed in Subsection 7.5.1.4.2.

7.7.1.1.3.1.5 Reactor Vessel Head Seal Leak Detection

Pressure between the inner and outer reactor vessel head seal ring will be sensed by a pressure switch/transducer. If the inner seal fails, the pressure at the pressure indicator is the vessel pressure and can be read on the panel outside of primary containment. The plant will continue to operate with the outer seal as a backup, and the inner seal can be repaired at the next outage when the head is removed. If both the inner and outer head seals fail, the leak will be detected by an increase in drywell temperature and pressure.

7.7.1.1.3.1.6 Safety/Relief Valve Seal Leak Detection

Thermocouples are located near the discharge of the safety/relief valve seat. The temperature signal goes to a multipoint recorder with an alarm. The alarm will be activated by any temperature in excess of a set temperature signalling that one of the safety/relief valve seats has started to leak.

7.7.1.1.3.1.7 Other Instruments

(1) The steam temperature is measured and is transmitted to a backrow panel in the main control room.

(2) The feedwater temperature is measured and transmitted to a backrow panel in the main control room.

(3) The feedwater corrosion products are monitored and the signal is transmitted to a backrow panel in the main control room for recording.

7.7.1.1.3.2 Testability

Pressure, differential pressure, water level, and flow instruments are located outside the drywell and are piped so that calibration and test signals can be applied during reactor operation, if desired.

7.7.1.1.4 Environmental Considerations

There are no special environmental considerations for the instruments described in this subsection.

7.7.1.1.5 Operational Considerations

7.7.1.1.5.1 General Information

The reactor vessel instrumentation discussed in this subsection is designed to augment the existing information from the engineered safeguards and safety system such that the operator can start up, operate at power, shut down, and service the reactor vessel in an efficient manner. None of this instrumentation is required to initiate any engineered safeguard or safety system.

7.7.1.1.5.2 Reactor Operator Information

The information that the operator has at his disposal from the instrumentation discussed in this subsection is discussed below:

(1) The shutdown flooding water level is indicated in the main control room.

(2) The core plate differential pressure is recorded on one pen of a two pen recorder. The second pen is used for total core flow.

(3) The jet pump developed head is indicated at a local instrument panel.

17 (4) Reactor vessel pressure is displayed on two redundant recorders located in the main control room.

(5) The reactor head inter-seal space pressure indicator
18 shows reactor pressure when the inner reactor head seal fails.

(6) The discharge temperatures of all the safety/relief valves are shown on a multipoint recorder on a backrow panel in the control room. Any temperature point that has exceeded the trip setting will turn on an annunciator indicating that a safety/relief valve seat has started to leak.

(7) The recorder for the feedwater corrosion products monitoring system is located in the main control room. The recorder will turn on an annunciator in the main control room for either a high or low signal.

7.7.1.1.5.3 Set Points

18 The annunciator alarm set points for the safety/relief valve seat leak detection, and feedwater corrosion product monitor are set so the sensitivity to the variable being measured will provide adequate information.

Figure 7.7-1 includes a chart showing the relative indicated water levels at which various automatic alarms and safety actions are initiated. Specific level values are shown in Tables 7.3-1 through 7.3-5. Each of the listed actions is described and evaluated in the subsection of this report where the system involved is described. The following list tells where various level measuring components and their set points are discussed.

(1) Level switches/transducers for initiating scram are discussed in Subsection 7.2.1.

(2) Level switches/transducers for initiating containment or vessel isolation are discussed in Subsection 7.3.1.1a.2.

(3) Level switches used for initiating HPCI, LPCI, CS and ADS and the level switches used to shut down the HPCI/HPCS pump are discussed in Subsection 7.3.1a.

(4) Level switches to initiate RCIC and the level switches to shut down the RCIC pump drive turbine are discussed in Subsection 7.4.1.1.

(5) Level trips to initiate various alarms and trip the main turbine and the feedpumps are discussed in Subsection 7.7.1.4.

7.7.1.2 Reactor Manual Control System - Instrumentation and Controls

7.7.1.2.1 System Identification

7.7.1.2.1.1 General

The objective of the reactor manual control system is to provide the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution can be controlled. The system allows the operator to manipulate control rods.

The reactor manual control system instrumentation and controls consists of the electrical circuitry, switches, indicators, and alarm devices provided for operational manipulation of the control rods and the surveillance of associated equipment.

This system includes the interlocks that inhibit rod movement (rod block) under certain conditions. The reactor manual control system does not include any of the circuitry or devices used to automatically or manually scram the reactor; these devices are discussed in Section 7.2. In addition, the mechanical devices of the control rod drives and the control rod drive hydraulic system are not included in the reactor manual control system. The latter mechanical components are described in Subsection 4.1.3.

7.7.1.2.1.2 Classification

This system is a power generation system, and is classified as not related to safety.

7.7.1.2.1.3 Reference Design

Table 7.1-2 lists reference design information. The reactor manual control system is an operational system and has no safety function. Therefore, there are no safety design differences between this system and those of the reference design facilities. This system is functionally identical to the referenced system.

7.7.1.2.2 Power Sources

Normal

The reactor manual control system receives its power from the 120 V ac instrumentation buses. Each of these buses receives its normal power supply from the appropriate 460 V ac standby power system. (See Subsection 8.3.1.)

Alternate

On loss of normal auxiliary power, the station diesel generators provide backup power to the 460 volt standby ac power systems.

7.7.1.2.3 Equipment Design

7.7.1.2.3.1 General

The following discussions will examine the control rod movement - instrumentation and control aspects of the subject system and the control rod position information system aspects. The "control" descriptions include:

(1) Control Rod Drive - Control System

(2) Control Rod Drive - Hydraulic System

(3) Rod Block Interlocks

The "position" descriptions include:

(1) Rod Position Probes

(2) Display Electronics

Figures 4.6-5a and 4.6-5b show the layout of the control rod drive hydraulic system. Figure 7.7-2 shows the functional arrangement of devices for the control of components in the control rod drive hydraulic system. The logic diagram for the overall reactor manual control system is shown in Figure 7.7-3. Although the figures also show the arrangement of scram devices, these devices are not part of the reactor manual control system. Control rods are moved by admitting water, under pressure from a control rod drive water pump, into the appropriate end of the control rod drive cylinder. The pressurized water forces the piston, which is attached by a connecting rod to the control rod, to move. Three modes of control rod operation are used: insert, withdraw, and settle. Four solenoid-operated valves are associated with each control rod to accomplish the actions required for the operational modes. The valves control the path that the control rod drive water takes to the cylinder.

7.7.1.2.3.2 Rod Movement Controls

7.7.1.2.3.2.1 Control Rod Drive Control System

7.7.1.2.3.2.1.1 Introduction

When the operator selects a control rod for motion and operates the rod insertion control switch as shown in Figure 7.7-4, messages are formulated in the A and B portions of the rod drive control system. A comparison test is made of these two messages, and identical results confirmed; then a serial message in the form of electrical pulses is transmitted to all hydraulic control units (HCU). The message contains two portions, (1) the identity or "address" of the selected HCU, and (2) operation data on the action to be executed. Only one HCU responds to this transmission; it proceeds to execute the rod insertion commands for example. Hence, the two insert valves for the selected rod open, and allow the control rod drive water to follow a path that results in control rod insertion.

On receipt of the transmitted signal as shown in Figure 7.7-4, the responding HCU transmits three portions of a message back to the control structure for comparison with the original message:

(1) its own hard-wire identity "address",

(2) its own operations currently being executed, and

(3) status indications of valve positions, accumulator conditions, and test switch positions.

In a similar manner, rod withdrawal is accomplished by formulating a message containing a different operation code. The responding HCU decodes the message and proceeds to execute the withdrawal command by operation of HCU valves shown in Figures 4.6-5a and 4.6-5b.

In either rod motion direction, the A and B messages are formulated and compared bit by bit (basic word length = 100 microseconds). If they agree, a message is transmitted to the HCU selected by the operator. Continued rod motion depends on receipt of a train of sequential messages because the HCU insert, withdraw, and settle valve control circuits are ac-coupled. The system must operate in a dynamic manner to effect rod motion. Postulated failures within the reactor manual control system generally will result in a static condition within the system, which will prevent further rod motion.

As discussed above, any disagreement between the A and B formulated messages will prevent further rod motion. Electrical noise disruptions will have only a momentary effect on the system unless the duration of the noise source is sufficiently long to disrupt the comparison of the stored "B" message and the "C" acknowledgement a predetermined number of times. In guaranteeing that rod motion is indeed terminated. Operator action is necessary to reset the system to restore normal operation. In Figure 7.7-5, three action loops of the solid-state reactor manual control system are depicted:

(1) The high speed loop (duration = 200 microseconds) alternately

  a) commands the selected control rod, and
  b) either scans a rod for status information or commands a portion of a single HCU self-test.

(2) The medium speed loop (0.045 sec. to 0.062 sec. duration) monitors all control rods in the reactor, one at a time, to update their status display and performs a test on one rod.

(3) The low speed loop (26 sec. to 177 sec. duration) exercises all HCU's one at a time to ensure correct execution of actions commanded. This provides for a continuous, periodic self-test of the entire reactor manual control system.

The rod selection circuitry is arranged so that a rod selection is sustained until either another rod is selected or separate action is taken to revert the selection circuitry to a no-rod-selection condition. Initiating movement of the selected rod prevents the selection of any other rod until the movement cycle of the selected rod has been completed. Reversion to the no-rod-selected condition is not possible (except for loss of control circuit power) until any moving rod has completed the movement cycle.

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Two of the valves on the HCU, labeled "withdraw," permit rod withdrawal. The withdrawal valve that connects the insert drive water supply line to the exhaust water header is the one that is associated with the settle operation. The remaining withdraw valve is associated only with the withdraw operation. The settle mode of control rod operation is provided to decelerate the control rod at the end of either an insert cycle or a withdraw cycle. The settle action smooths out the control rod movement and prolongs the life of control rod drive hydraulic system components. During the settle mode, the withdraw valve associated with the settle operation is opened or remains open while the other three solenoid-operated valves are closed.

During an insert cycle, the settle action vents the pressure from the insert drive water supply line to the exhaust header and thus gradually reduces the differential pressure across the drive piston of the selected rod. During a withdraw cycle, the settle action holds open the discharge path for withdraw water while the withdraw drive water supply is shut off. This also allows a gradual reduction in the differential pressure across the control rod drive piston. After the control rod has slowed down, the collet fingers engage the index tube and lock the rod in position.

The direction in which the selected rod moves is determined by the position of four switches located on the reactor control panel. These four switches, "insert", "withdraw", "continuous insert" and "continuous withdraw", are pushbuttons which return by spring action to an off position.

7.7.1.2.3.2.2.1.2 Insert Cycle

Following is a description of the detailed operation of the reactor manual control system during an insert cycle. The cycle is described in terms of the insert, withdraw, and settle commands emanating from the reactor manual control system. The response of a selected rod when the various commands are transmitted has been explained previously. Figure 7.7-2 can be used to follow the sequence of an insert cycle.

With a control rod selected for movement, depressing the "insert" switch and then releasing the switch energizes the insert command for a limited time. Just as the insert command is removed, the settle command is automatically energized and remains energized for a limited time. The insert command time setting and the rate of drive water flow provided by the control rod drive hydraulic system determine the distance traveled by a rod. The time setting results in a one-notch (6-in.) insertion of the selected rod for each momentary application of a rod-in signal from the rod movement switch. Continuous insertion of a selected control rod is possible by holding the "insert" switch.

A second switch can be used to affect insertion of a selected control rod. This switch is the "continuous insert" switch. By holding this switch "in," the unit maintains the insert command in a continuous, energized state to cause continuous insertion of the selected control rod. When released, the timers are no longer bypassed and normal insert and settle cycles are initiated to stop the drive.

7.7.1.2.3.2.1.3 Withdraw Cycle

Following is a description of the detailed operation of the reactor manual control system during a withdraw cycle. The cycle is described in terms of the insert, withdraw, and settle commands. The response of a selected rod when the various commands are transmitted has been explained previously. Figure 7.7-2 can be used to follow the sequence of a withdraw cycle.

With a control rod selected for movement, depressing the "withdrawal" switch energizes the insert valves for a short time. Energizing the insert valves at the beginning of the withdrawal cycle is necessary to allow the collet fingers to disengage the index tube. When the insert valves are deenergized, the withdraw and settle valves are energized for a controlled period of time. The withdraw valve is deenergized before the settle valve; this tends to decelerate the selected rod. When the settle valve is deenergized, the withdraw cycle is complete. This withdraw cycle is the same whether the withdraw switch is held continuously or momentarily depressed position. The timers that control the withdraw cycle are set so that the rod travels one notch (6-in.) per cycle. Provisions are included to prevent further control rod motion in the event of timer failure.

A selected control rod can be continuously withdrawn if the "withdraw" switch is held in the depressed position at the same time that the "continuous withdraw" switch is held in the depressed position. With both switches held in these positions, the withdraw and settle commands are continuously energized.

7.7.1.2.3.2.2 Control Rod Drive-Hydraulic System Control

Two motor-operated pressure control valves, one air-operated flow control valve, and two solenoid-operated stabilizer valves are included in the control rod drive hydraulic system to maintain smooth and regulated system operation. These devices are shown in Figures 4.6-5a and 4.6-5b. The motoroperated pressure control valves are positioned by manipulating switches in the main control room. The switches for these valves are located close to the pressure indicators that respond to the pressure changes caused by the movements of the valves. The air-operated flow control valve is automatically positioned in response to signals from an upstream flow measuring device. The stabilizer valves are automatically controlled by the energization of the insert and withdraw commands. The control scheme is shown in Figure 7.7-2. There are two drive water pumps which are controlled by switches in the main control room. Each pump automatically stops on indication of low suction pressure.

7.7.1.2.3.2.3 Rod Block Interlocks

The rod block functions are discussed in Subsection 7.6.1a.5.7.

7.7.1.2.3.2.4 Testability

In addition to the periodic self-test mode of system operation, the reactor manual control circuitry can be routinely checked for correct operation by manipulating control rods using the various methods of control. Detailed testing and calibration can be performed by using standard test and calibration procedures for the various components of the reactor manual control circuitry.

7.7.1.2.3.3 Rod Position Information

This subsystem includes the rod position probes and the electronic hardware that processes the probe signals and provides the data described above.

7.7.1.2.3.3.1 Position Probes

The position probe is a long cylindrical assembly that fits inside the control rod drive index tube. It includes fifty-two magnetically operated reed switches, located along the length of the probe and operated by a permanent magnet fixed to the moving part of the hydraulic drive mechanism. As the drive, and with it the control rod blade, moves along its length, the magnet causes reed switches to close as it passes over the switch locations.

The particular switch closed then indicates where the control rod drive, and hence the rod itself is positioned.

The switches are located as follows: one at each of twenty-four notch (even) position; one at each of twenty-four mid-notch (odd) position; two at the fully inserted position (approximately the same location as the "00" notch); one at the fully withdrawn position (approximately the same location as the "48" notch position); and, one at the "overtravel" or decoupled position.

All of the mid-notch or "odd" switches are wired in parallel and treated as one switch (for purposes of external connections), and the two fully-in switches are wired in parallel and treated as one switch. These and the remaining switches are wired in a 5 x 6 array (the switches short the intersections) and routed out in all 11-wire cable to the processing electronics (the probe also includes a thermocouple which is wired out separate from the 5 x 6 array). See Figure 7.7-6.

7.7.1.2.3.3.2 Position Indication Electronics

The electronics consists of a set of "probe multiplexer cards" (one per 4-rod group there the 4-rod group is the same as the display grouping described above), a set of "file control cards" (one per 11 multiplexer cards), and one set of master control and processing cards serving the whole system. All probe multiplexer cards are the same except that each has a pair of plug-in "daughter cards" containing the identity code of one 4-rod group (the probes for the corresponding 4 rods are connected to the probe multiplexer card).

7.7.1.2.3.3.3 System Operation

The system operates on a continuous scanning basis with a complete cycle every 45 milliseconds. The operation is as follows. The control logic generates the identity code of one rod in the set, and transmits it using time multiplexing to all of the file control cards. These in turn transmit the identity with timing signals to all of the probe multiplexer cards. The one multiplexer card with the matching rod identity will respond and transmit its identity (locally generated) plus the "raw" probe data for that rod back through the file control card to the master control and processing logic. The processing logic does several checks on the returning data. First, a check is made to verify that an answer was received. Next, the identity of the answering data is checked against that which was sent. Finally, the format of the data is checked for "legitimacy". Only a single even position or, full-in plus position "00", or full-out plus position "48," or odd, or overtravel, or blank (no switch closed) are legitimate. Any other combination of switches is flagged as a fault.

If the data passes all of these tests, it is (a) decoded and transmitted in multiplexed form to the displays on the Unit Operating Benchboard, and (b) loaded into a memory to be read by the computer as required.

As soon as one rod's data is processed, the next rod's identity is generated and processed and so on for all of the rods. When data for all rods has been gathered, the cycle repeats.

7.7.1.2.4 Environmental Considerations

The reactor manual control system (control and position indication circuitry) is not required for any plant safety function, nor is required to operate in any associated design basis accident or transient occurrence. The reactor manual control circuitry is required to operate only in the normal plant environments during normal power generation operations.

The control rod drives and their hydraulic control units are located in the containment. (See Table 3.11-1).

The logic, control units, and readout instrumentation are located in the control structure. (See Table 3.11-1).

The control rod position detectors are located beneath the reactor vessel in the drywell. The normal design environments encountered in these areas are given in Table 3.11-1.

7.7.1.2.5 Operational Considerations

7.7.1.2.5.1 General Information

The reactor manual control system is totally operable from the main control room. Manual operation of individual control rods is possible with a jog switch to effect control rod insertion, withdrawal, or settle. Rod position indicators, described below, provide the necessary information to ascertain the operating state and position of all control rods. Conditions which prohibit control rod insertion are alarmed with the rod block annuciator.

7.7.1.2.5.2 Reactor Operator Information

Table 7.7-1 gives information on instruments for the reactor manual control system. A large rod information display on the Unit Operating Benchboard is patterned after a top view of the reactor core. (See Figure 7.5-1). The display allows the operator to acquire information rapidly by scanning.

Colored windows provide an overall indication of rod pattern and allow the operator to quickly identify an abnormal indication. The following information for each control rod is presented in the display:

Rod fully inserted (green)
Rod fully withdrawn (red)
Selected rod identification (coordinate position, white)
Accumulator trouble (amber)
Rod scram (blue)
Rod drift (red)

Also dispersed throughout the display, in locations representative of the physical location of LPRM strings in the core, are LPRM lights as follows:

LPRM low flux level (white)
LPRM high flux level (amber)

A separate, smaller hardwired display is located on the standby information panel. A CRT presentation is available on the Unit Operating Benchboard. This latter display presents the positions of the control rod selected for movement and the other rods in the rod group. For display purposes the control rods are considered in groups of four adjacent rods (a "four-rod group") centered around a common core volume monitored by four LPRM strings. Rod groups at the periphery of the core may have less than four rods. The four-rod display shows the positions, in digital form, of the rods in the group to which the selected rod belongs. A blue background on the digital display indicates which of the four rods is selected for movement. On either side of the four-rod position display are indicated the readings of the 16 LPRM channels (four LPRM strings) surrounding the core volume common to the four rods of the group.

The four-rod display allows the operator to better focus his attention to the portion of the core where rod motion is occurring. A full core rod position display would tend to be confusing and difficult to read. The sixteen associated LPRM displays permit the operator to monitor the core reactivity during rod motion. In addition, on demand by the operator, the process computer will provide a print out of all rod positions.

During startup or shutdown, all rods of a given sequence are either fully withdrawn or fully inserted. These patterns are indicated on the full core display with the full-in or full-out lights. In addition to the whole core display, a drifting rod is indicated by an alarm and red light in the control room. The rod drift condition is also monitored by the process computer.

An indication is also provided for rod trend beyond the limits of normal rod movement. If the rod drive piston moves to the "overtravel" position, an alarm is sounded in the control room. The overtravel alarm provides a means to verify that the drive-to-rod coupling is intact because, with the coupling in its normal condition, the drive cannot be physically withdrawn to the overtravel position. Coupling integrity can be checked by attempting to withdraw the drive to the overtravel position.

Accumulator trouble and rod scram indicators are provided to the displays by the rod drive control system. The LPRM high and low flux levels and the sixteen LPRM readings are provided by the power range monitor system. The remaining information to the displays and the position information for the process computer are provided by the rod position information subsystem.

The following main control room lights are provided to allow the operator to know the conditions of the control rod drive hydraulic system and the control circuitry:

Stabilizer valve selector switch position
Insert command energized
Withdraw command energized
Settle command energized
Withdrawal not permissive
Continuous withdrawal
Pressure control valve position
Flow control valve position
Drive water pump low suction pressure (alarm and pump trip)
Drive water filter high differential pressure (alarm only)
Charging water (to accumulator) high pressure (alarm only)
Control rod drive temperature (alarm only)
Scram discharge volume not drained (alarm only)
Scram valve pilot air header high/low pressure (alarm only)

7.7.1.2.5.3 Set Points

The subject system has no safety set points.

7.7.1.3 Recirculation Flow Control System-Instrumentation and Controls

7.7.1.3.1 System Indentification

7.7.1.3.1.1 General

The objective of the recirculation flow control system is to control reactor power level, over a limited range, by controlling the flow rate of the reactor recirculating water.

See Figures 7.7-7 and 7.7-8. The control involves varying the speed of the recirculation pumps by changing the voltage and frequency of the ac supply to each pump motor. The ac supply is provided by a motorgenerator (M-G) set for each pump. Each M-G set consists of a squirrel cage induction motor driving a variable frequency generator through a variable speed converter. The generator output is modulated by varying the slip within the converter. Since flow rate is directly proportional to pump speed, which is proportional to generator speed, generator speed is considered the controlled variable of the system. Either manual or automatic input to the master controller is the reference input to the system. The flow control subsystem is designed to limit the range and rate of change of pump speed, and to otherwise ensure proper operational and equipment protection.

7.7.1.3.1.2 Classification

This system is a power generation system and is classified as not related to safety.

7.7.1.3.1.3 Reference Design

The recirculation flow control system is an operational system and has no safety function; therefore, there are no safety differences between this system and those of the above referenced facilities. This system is functionally identical to the referenced system.

7.7.1.3.2 Power Sources

Normal:

The recirculation flow control system power is supplied by the two 120 Vac instrument buses. Flow control loop A is powered from bus A and flow control loop B from bus B. That portion of the control system which is common to both loops A and B (master controller, etc.) is powered from either bus A or B. Each bus receives its normal power supply from the appropriate 460 Vac normal auxiliary power system. See Subsection 8.3.1 for the detail descriptions.

Alternate:

On loss of normal auxiliary power, the startup transformer provides backup power to the 460 Vac normal auxiliary power systems.

7.7.1.3.3 Equipment Design

7.7.1.3.3.1. General

Reactor recirculation flow is changed by adjusting the speed of the two reactor recirculating pumps. This is accomplished by adjusting the frequency and voltage of the electrical power supplied to the recirculation pump motor (see Figure 7.7-8). Control of pump speed, and thus core flow, is such that at various control rod patterns, different power level changes can be accommodated. For a 100% rod pattern, power change control down to approximately 65 percent of full power is possible by use of flow variation. At other rod patterns, power control is possible over a range of approximately 35 percent of the maximum operating power level for that rod pattern. Thus, the power control range is approximately a constant fraction of operating power but a variable absolute power range.

A lower limit exists on flow control capability, below which automatic control by flow is not permitted. An increase in recirculation flow temporarily reduces the void content of the moderator by increasing the flow of coolant through the core. The additional neutron moderation increases the reactivity of the core, which causes the reactor power level to increase. The increased steam generation rate increases the steam volume in the core with a consequent negative reactivity effect, and a new steady-state power level is established. When recirculation flow is reduced, the power level is reduced in the reverse manner.

Figure 7.7-8 illustrates how the recirculation flow control system operates in conjunction with the turbine controls for automatic load following.

7.7.1.3.3.2 Pump Drive Motor Control

Each recirculation pump motor has its own motor-generator set for a power supply. A variable speed converter is provided between the M-G set motor and generator. To change the speed of the reactor recirculation pump, the variable speed converter varies the generator speed which changes the frequency and magnitude of the voltage supplied to the pump motor to give the desired pump speed. The recirculation flow control system uses a demand signal from either the operator or the main plant turbine generator speed governing mechanism. The demand signal is supplied to the master controller. A signal from the master controller adjusts the speed setting of the speed controller for each M-G set converter. The master controller signal is compared with the actual speed of the generator by the speed controller. The speed controller signal causes adjustment of the speed converter, resulting in a change of the generator speed until the feedback from the generator equals the master controller signal. The reactor power change resulting from the change in recirculation flow causes the initial pressure regulator to reposition the turbine control valves. If the original demand signal was a turbine load/speed error signal, the turbine responds to the change in reactor power level by adjusting the turbine control valves until the load/speed error signal is reduced to zero.

7.7.1.3.3.3 High Frequency Motor - Generator (HFMG) Set

Each of the two M-G sets and its controls are identical. The M-G set can continuously supply power to the pump motor at any speed between approximately 19% and 96% of the drive motor speed. The M-G set is capable of starting the pump and accelerating it from standstill to the desired operating speed when the pump motor thrust bearing is fully loaded by reactor pressure acting on the pump shaft. The main components of the M-G set are a drive motor, a generator, and a variable speed converter, with an actuation device to adjust the converter output speed.

(1) Drive Motor

The drive motor is an ac induction motor which drives the input shaft of the variable speed converter. The motor can operate under electrical supply variations of

5 percent of rated frequency or 10 percent of rated voltage.

(2) Generator

The variable frequency generator is driven by the output shaft of the variable speed converter. During normal operation, the generator exciter is powered by the drive motor. The excitation of the generator is provided from an auxiliary source during pump startup.

(3) Variable Speed Converter and Actuation Device

The variable speed converter transfers power from the drive motor to the generator. The variable speed converter actuator automatically adjusts the slip between the converter input shaft and output shaft as a function of the signal from the speed controller. If the speed controller signal is lost, the actuator causes the speed converter slip to remain "as is." Manual reset of the actuation device is required to return the speed converter to normal operation.

7.7.1.3.3.4 Speed Control Components

The speed control system, Figure 7.7-8, controls the variable speed converters of both motor-generator sets. The M-G sets can be manually controlled individually or jointly. Automatic control is according to the turbine control mechanism load/speed error signal. The master controller and the speed demand limiter are common to the control of both M-G sets. The signal from these two components is fed to two separate sets of control system components, one set for each M-G set. The control system components for each M-G set are the following: a manual automatic transfer station, a speed controller, a function generator, a signal failure alarm, and a startup signal generator.

7.7.1.3.3.4.1 Master Controller

The master controller is a manual/automatic controller that provides a signal to automatically or manually control both recirculation pumps with an interlock to the initial pressure regulator. During automatic load following, the turbine generator control mechanism supplies a load/speed error signal which represents the mismatch between the steam being supplied to the turbine generator and the steam required by the turbine generator to maintain constant speed. This load/speed error signal is supplied to the master controller. During normal automatic operation, the master controller transmits an output signal to the speed controllers of each M-G set. The speed controllers adjust their generator speed, hence pump speed, according to the load/speed error signal requirement. A pressure set point adjustment signal from the turbine controller goes through the interlock switch on the master controller, during automatic operation only, to the initial pressure regulator. This signal allows an immediate response by the turbine generator to the changed load demand. Because of the pressure set point adjustment, the turbine generator steam requirements can be met during the time required for a new power level to be established by the change in recirculation flow. The pressure set point change is effective only while a new reactor power level is being established.

As an example of the use of the pressure regulator set point adjustment, suppose an increase in plant load is demanded by an increased turbine generator requirement. The turbine generator demands more steam to maintain its constant speed. The turbine controller allows the initial pressure regulator to adjust to a lower nuclear system pressure. This temporary decrease in pressure allowed immediately lets the turbine control valves open to meet the turbine generator steam demand while the recirculation flow change is being made. When the recirculation flow change effects the increase in reactor power level, the pressure regulator again controls nuclear system pressure at its original set point.

7.7.1.3.3.4.2 Speed Demand Limiter

The speed demand limiter is an adjustable high/low dual limiter module. It provides a limit on the maximum and the minimum M-G set speeds which can be demanded by the master controller. Normally, the master controller signal is between the speed demand limiter limits, and the signal passes through the speed demand limiter to the manual/automatic transfer stations for each M-G set.

7.7.1.3.3.4.3 Manual/Automatic Transfer Station

The manual/automatic transfer station, one for each M-G set, is manually controlled with a tranfer switch. While the M-G set is being controlled by the master controller, the transfer switch is positioned so that the manual controller is bypassed and the master controller signal goes through the manual/automatic transfer station to the speed controller. During startup, the master controller signal is blocked by the transfer switch and the output signal is generated and controlled by the manual/automatic transfer station by operator adjustment.

7.7.1.3.3.4.4 Speed Controller

The speed controller, one for each M-G set, transmits the signal which adjusts the M-G set variable speed converter. The speed controller compares its set point signal to the feedback signal from the M-G set tachometer and adjusts its output to the speed converter so that the feedback signal from the tachometer is made to equal the set point signal. The speed controller set point signal is received during automatic operation from the master controller, during individual M-G set manual operation from the manual/automatic transfer station, during pump startup from the startup signal generator, and during low feedwater flow from the speed demand limiter.

7.7.1.3.3.4.5 Signal Failure Alarm

If the signal to the M-G set variable speed converter drops below 10 percent normal minimum control signal, the signal failure alarm, of which there is one for each M-G set, actuates an alarm in the main control room and acts to prevent any change of slip within the variable speed converter.

7.7.1.3.3.4.6 Startup Signal Generator

The startup signal generator, one for each M-G set, supplies the set point signal to the speed controller during M-G set startup. This adjusts the M-G set variable speed converter for approximately 50 percent recirculation pump speed.

7.7.1.3.3.4.7 Speed Limiter

The speed limiter, one for each M-G set, is an adjustable high limiter module. The speed controller set point signal is automatically limited by the speed limiter if the recirculation pump main discharge valve is not fully open, or if the feedwater flow is less than 20% of rated flow. The speed controller signal will place the M-G set generator speed at approximately 20% rated speed. If the recirculation pump discharge valve is partly closed, the recirculation pump may overheat. If the feedwater flow is less than 15% of rated flow, sufficient steam voids may be present in the recirculating water to cause cavitation in the recirculation pump or the jet pumps.

7.7.1.3.3.4.8 Recirculation Loop Starting Sequence

Each recirculation loop is independently put into operation by operating the controls of each loop as follows:

(1) The manual/automatic transfer station is switched to manual control and its output signal is adjusted to give the desired generator speed, typically 20 percent of rated speed, after the pump has started.

7.7.1.3.3.4 Testability

The M-G set, master controller, and speed controller are functioning during normal power operation. Any abnormal operation of these components can be detected during operation. The components that do not continually function during normal operation can be tested and inspected for calibration and operability during scheduled plant shutdowns. All the recirculation flow control system components are tested and inspected according to the component manufacturers' recommendations. This can be done during scheduled shutdowns.

7.7.1.3.4 Environmental Considerations

The recirculation flow control system is not required for safety purposes, nor required to operate after the design basis accident. The system is required to operate in the normal plant environment for power generation purposes only.

The only part of the recirculation flow control equipment in the drywell is the pump motor and it is subject to the design conditions environment specified in Table 3.11-1.

The logic control units and instrumentation are located in the main control room and are subject to that environment. Refer to Table 3.11-1.

7.7.1.3.5 Operational Considerations

7.7.1.3.5.1 General Information

Indicators and alarms are provided to keep the operator informed of the status of system and equipment and to permit him to quickly determine the location of malfunctioning equipment. Temperature monitoring of the equipment is recorded and alarmed if safe levels are exceeded. Indicators are provided to show pump power requirements, M-G set speed, recirculation loop flow, valve positions, and analog control signals, all of which determine system status. Alarms are provided to alert the operator to malfunctioning control signals, excessive cooling water temperatures, inability to change pump speed, and status of M-G circulating lube oil supply.

7.7.1.3.5.2 Reactor Operator Information

Indication and alarms are provided to keep the operator informed of the status of systems and equipment, and to quickly determine location of malfunctioning equipment.

Visual display consists of loop flow, valve position, and controller output and input deviation meters. (Figure 7.5-1) Alarms are provided to alert the operator of malfunctioning control signals, inability to change valve position, condition of the hydraulic system, pump, and motor, and temperatures of cooling water. In most cases, alarms are supplemented by light indicators to more closely define the problem area.

7.7.1.3.5.3 Set Points

The subject system has no safety set points.

7.7.1.4 Feedwater Control System-Instrumentation and Controls

7.7.1.4.1 System Identification

7.7.1.4.1.1 General

The Feedwater Control System controls the flow of feedwater into the reactor pressure vessel to maintain the water in the vessel within predetermined levels during all plant operating modes. The range of water level is based upon the requirements of the steam separators (this includes limiting carryover and carryunder, which affects turbine performance), and recirculation pump operation and the need to prevent exposure of the reactor core. The Feedwater Control System employs water level, steam flow, and feedwater flow as a three-element control.

Single-element control is also available based on water level only. Normally, the signal from the feedwater flow is equal to the steam flow signal; thus, if a change in the steam flow occurs, the feedwater flow follows. The steam flow signal provides anticipation of the change in water level that will result from change in load. The level signal provides a correction for any mismatch between the steam and feedwater flow which causes the level of the water in the reactor vessel to rise or fall accordingly.

7.7.1.4.1.2 Classification

This system is a power generation system and is classifed as not related to safety.

7.7.1.4.1.3 Reference Design

Table 7.1-2 lists reference design information. The feedwater control system is an operational system and has no safety function. Therefore, there are no safety differences between this system and those of the above referenced facilities. The subject system is functionally identical to the referenced system.

7.7.1.4.2 Power Sources

The feedwater control system power is supplied by three independent sources such that no single power failure can incapacitate more than one level sensing element. Power for two of the three level sensing channels is supplied from the plant 125 V dc batteries and the other channel is powered from a 120 V ac Instrumentation Power bus.

7.7.1.4.3 Equipment Design

7.7.1.4.3.1 General

During normal plant operation, the feedwater control system automatically regulates feedwater flow into the reactor vessel. The system can be manually operated. (see Figure 7.7-9.)

The feedwater flow control instrumentation measures the water level in the reactor vessel, the feedwater flow rate into the reactor vessel, and the steam flow rate from the reactor vessel. During automatic operation, these three measurements are used for controlling feedwater flow.

The optimum reactor vessel water level is determined by the requirements of the steam separators. The separators limit water carry-over in the steam going to the turbines and limit steam carry-under in water returning to the core. For optimum limitation of carry-over and carry-under, the steam separators require that the reactor vessel water level decrease functionally as reactor power level increases. The water level in the reactor vessel is maintained within ± 1.5 inches of the optimum level. This control capability is achieved during plant load changes by balancing the mass flow rate of feedwater to the reactor vessel with the steam flow from the reactor vessel. The feedwater flow is regulated by controlling the speed of the turbine-driven feedwater pumps to deliver the required flow to the reactor vessel.

7.7.1.4.3.2 Reactor Vessel Water Level Measurement

Reactor vessel narrow range water level is measured by two indentical, independent sensing systems. For each channel, a differential pressure transmitter senses the difference between the pressure caused by a constant reference column of water and the pressure caused by the variable height of water in the reactor vessel. The differential pressure transmitter is installed on lines that serve other systems. (See Subsection 7.7.1.1). The differential pressure signals are used for indication and control. The narrow range level signal from either sensing channel can be selected by the operator as the signal to be used for feedwater flow control. The selected narrow range water level and wide range water level signals are continually recorded in the main control room. A third level sensing channel (upset range) is used in conjection with the two control channels to provide failure tolerant trips of the main turbine and feed pump prime movers. All three reactor level signals and reactor pressure are indicated in the main control room.

7.7.1.4.3.3 Steam Flow Measurement

Steam flow is sensed at each main steamline flow restrictor by a differential pressure transmitter. A signal proportional to the true mass steam flow rate is linearized and indicated in the main control room. The signals are summed to produce a total steam flow signal for indication and feedwater flow control. The total steam flow signal is recorded in the main control room.

7.7.1.4.3.4 Feedwater Flow Measurement

Feedwater flow is sensed at a flow element in each feedwater line by differential pressure transmitters. Each feedwater signal is linearized and than summed to provide a total mass flow signal which is recorded in the main control room.

7.7.1.4.3.5 Feedwater/Level Control

The level controller and the manual/automatic transfer station produce the feedwater control signal. The signal can be controlled either manually or automatically.

The level controller contains a set point deviation meter, and an indicator that displays the corrected level signal, a manual-output control with indicator, and a manually operated level set point scale. Input to the controller is derived from one of two sources. The single-element source is the reactor water level. The three-element source includes measurements of steam flow, feedwater flow, and reactor water level. Either input can be selected. During automatic operation of the feedwater control system, the level controller output is proportional to the level error in the system. During manual operation, output is set and indicated at the controller.

Selection of automatic or manual control is made at the manual/automatic transfer station (one for each feedwater control system). The station is a manual controller with a transfer switch and output indicator. When each system is controlled by its level controller, the transfer switch bypasses the manual controller, and the level controller signal goes through to the turbine-driven feedwater pumps. For manual control, the transfer switch blocks the level controller signal, and the operator provides the feedwater control signal at the manual/automatic transfer station. During normal automatic operation, the optimum reactor vessel water level is automatically determined.

The feedwater control system uses the three-element control signal to maintain reactor vessel water level within a small margin of optimum water level during plant load changes. This signal is obtained as follows. The total steam flow signal and the total feedwater flow signal are fed into a proportional amplifier. The output from this amplifier reflects the mismatch between its input signals. The output is designated as the steam flow/feedwater flow error signal. When steam flow exceeds feedwater flow, the amplifier output is increased from its normal value. The reverse is also true. This amplifier output is fed to a second proportional amplifier, which also receives the reactor vessel water level signal. The reactor vessel water level signal is added to the steam flow/feedwater flow error signal to produce the three-element control signal. Then it is passed on to the level controller as a modified level input. The controller compares this against the set point and provides the final control signal.

7.7.1.4.3.5.1 Interlocks

The level control system also provides interlocks and control functions to other systems. When one of the reactor feed pumps is lost and coincident or subsequent low water level exists, recirculation flow is reduced to within the power capabilities of the remaining reactor feed pumps. This reduction aids in avoiding a low level scram by reducing the steaming rate. Reactor recirculation flow is also reduced on sustained low feedwater flow to ensure that adequate NPSH will be provided for the recirculation system.

Interlocks from steam flow and feedwater flow are used to initiate insertion of the rod worth minimizer block. An alarm on low steam flow indicates that the above rod worth minimizer insertion interlock set point is being approached. Alarms are also provided for (1) high and low water level and (2) reactor high pressure. Interlocks will trip the plant turbine and feedwater pumps in event of reactor high water level.

7.7.1.4.3.6 Turbine-Driven Feedwater Pump Control

Feedwater is delivered to the reactor vessel through turbine-driven feedwater pumps, which are arranged in parallel. The turbines are driven by steam from the reactor vessel. During planned operation, the feedwater control signal from the level controller is fed to the turbine speed control systems, which adjust the speed of their associated turbines so that feedwater flow is proportional to the feedwater demand signal. Each turbine can be controlled by its manual/automatic transfer station. If the feedwater control to the turbine signal is lost, an alarm unit in the feedwater control circuit causes the turbine speed control system to lock the turbine speed "as is" and initiates an alarm in the control room. The level controller, and the manual/auto transfer stations associated with each turbine speed controller, are the "bumpless transfer" types.

7.7.1.4.3.7 Testability

All feedwater flow control system components can be tested and inspected according to manufacturers' recommendations. This can be done prior to plant operation and during scheduled shutdowns. Reactor vessel water level indications from the three water level sensing systems can be compared during normal operation to detect instrument malfunctions. Steam mass flow rate and feedwater mass flow rate can be compared during constant load operation to detect inconsistencies in their signals. The level controller can be tested while the feedwater control system is being controlled by the manual/automatic transfer stations.

7.7.1.4.4 Environmental Considerations

The feedwater control system is not required for safety purposes, nor is it required to operate after the design basis accident. This system is required to operate in the normal plant environment for power generation purposes only. The reactor feed pumps in the turbine building experience the normal design environments listed in Table 3.11-1.

7.7.1.4.5 Operational Considerations

7.7.1.4.5.1 General Information

The level controller is located in the main control room where, at the operator's discretion, the system can be operated either manually or automatically via the manual/auto control selector.

Manual control of the individual feedwater reactor turbine driven pumps is available to the operator in the main control room. This includes control of any low flow feedwater bypass valve that may be used for startup when steam is not available to run the turbine driven reactor feed pumps.

In event of loss of feedwater, the reactor will automatically scram as a result of low water level (trip level 3) Reactor water level will continue to decrease until low water level (trip level 2) is reached. The main steamlines will automatically close and the HPCI and RCIC systems automatically start and water level will be maintained.

7.7.1.4.5.2 Reactor Operator Information

Indicators and alarms, provided to keep the operator informed of the status of the system, are as noted in previous subsections.

7.7.1.4.5.3 Set Points

The subject system has no safety set points.

7.7.1.5 Pressure Regulator and Turbine-Generator Control System

7.7.1.5.1 Power Generation Design Bases

The pressure regulator and turbine-generator control system must maintain a constant turbine inlet pressure (within the range of the regulator controller proportional load setting) with load following capability. In conjunction with the reactor recirculation flow control system, the reactor pressure is controlled from startup, through normal operation, and to shutdown.

The control system must control the speed and the acceleration of the turbine from zero to 100 percent of rated speed.

The control system must match the nuclear steam supply to the steam requirements as determined by the load requirement.

7 A block diagram of the turbine controls is shown in Figure 7.7-15.

7.7.1.5.2 Power Sources

Power for the pressure regulator and turbine-generator control system is supplied by a 120 Vac, 60 Hz, single phase, uninterruptible power supply and a 125 V dc station battery. See Subsections 8.3.1.8 and 8.3.2.1.1.8.

A permanent magnet generator (PMG) on the turbine shaft supplies 115 Vac, 3 phase, 420 Hz for speeds above 1800 RPM.

7.7.1.5.3 Equipment Design

7.7.1.5.3.1 System Description

The turbine-generator control system is a GE Mark I Electrohydraulic Control (EHC) system. Solid state control circuitry in combination with high pressure hydraulic systems provide schemes for turbine steam pressure regulation, steam bypassing to condenser, turbine speed controlling, and load following capability.

7.7.1.5.3.2 Steam Pressure Control

The steam pressure control unit composes the actual main steam pressure with the desired reference pressure, determined by the load requirement, and generates a total steam flow demand.

The pressure reference signal is produced by a motor-operated device that can be operated by local pushbuttons on remote control signals.

The modified pressure error signal is produced redundantly by redundant devices. The two pressure error signals are fed into a gating circuit that accepts the lower pressure as a control signal with the higher becoming the backup.

The steam pressure control unit provides the control valve flow signal and bypass control unit and automatic load following signals.

7.7.1.5.3.3 Steam Bypass System

The steam bypass control unit compares the desired control valve flow signal with the total steam flow signal. The resulting error signal which is biased from 5 to 15% to prevent continuous opening and closing of the bypass valves provides the desired bypass valve flow signal.

The bypass valve jack is a motor-operated device used for setting a bypass valve position reference during startup and shutdown of the reactor. This motor-operated device can be operated by local pushbuttons or remote control signals.

Limit signals are also produced by the maximum combined flow limit and the condenser vacuum pressure switches.

7.7.1.5.3.4 Turbine Speed System/Load Control System

The speed control unit compares the actual turbine speed with the desired speed reference, and the actual acceleration with the desired acceleration reference to provide an error signal to the load control unit.

When the speed reference signal changes by a step, the acceleration control takes over to accelerate the turbine, at the selected rate, to the new speed reference. Upon a decrease of the speed reference, the turbine will coast down with the valves closed. The valves will reopen when the new desired speed is reached.

Because of the extreme importance in safeguarding against overspeed, the speed control unit has two redundant channels. Loss of both speed signals will shut down the turbine.

The load control unit provides flow control signals to the control valves and integral valves, and modified speed error and load reference signals to the automatic load following circuit.

The load reference signal is produced by a motor-operated device that can be operated by local pushbuttons or remote control signals.

The load reference device can be calibrated for rated speed and steam conditions independent of speed regulation. When the generator is not on the line, the load reference signal is a speed adjustment and is used for synchronizing the turbine.

When the generator loses the electrical load, the load control unit initiates the action to rapidly close the control valves and the intercept valves to essentially stop the steam flow to the turbine.

7.7.1.5.3.5 Turbine Generator to Reactor Protection System Interface

Two conditions initiate reactor scram, turbine stop valve closure, and turbine control valve fast closure when reactor power is about 30 percent of rated.

The turbine stop valve closure signal is generated before the turbine stop valves have closed more than 10 percent. This signal originates from position switches that sense stop-valve motion away from fully open. Four limit switches are provided equally among the turbine stop valves. The switches are closed when the stop valves are fully open and open within 10 milliseconds after the setpoint is reached. The switches are electrically isolated from each other and from other turbine plant equipment.

The control valve fast closure signal is generated by four turbine oil line pressure switches which sense hydraulic oil pressure decay. The switches are closed when the valves are open and open within 30 milliseconds after the control valves start to close in a fast closure mode.

Four turbine first-stage pressure switches, which measure equivalent steam flow, are provided for bypassing the stop valve closure and control valve fast closure inputs at reactor power levels below 30 percent.

7.7.1.5.3.6 Turbine-Generator to Main Steam Isolation System Interface

The turbine-generator interfaces the main steam isolation system through the condenser vacuum switches. Four independent main condenser vacuum switches provide isolating signals to the main steam isolation valves. Each vacuum switch has its own isolation (root valve) and pressurizing source connection for testing.

Pressure switch contacts open on low vacuum. Condenser vacuum switches are also discussed in Subsection 7.3.1.1a.2.4.1.13.

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7.7.1.5.3.7 Inspection and Testing

Testing controls are provided for testing the turbine valve reactor protection system interface signal switches.

Each stop valve is individually actuated to the 10 percent closed point.

Stop valves are actuated in pairs to the 10 percent closed point, one pair at a time: 1 and 2, 3 and 4, 1 and 3, and 2 and 4.

One control valve fast closure hydraulic oil pressure switch is actuated at a time by actuating test valves in the pressure switch sensing line.

Each main condenser low vacuum switch is individually tested.

7.7.1.5.4 Environmental Considerations

The turbine-generator control system is required to operate in the normal plant environment for power generation purposes only.

Instruments and controls on the turbine that experience the turbine building normal design environment as listed in Table 3.11-1.

The logic, remote control units, and instrument terminals located in the control structure experience the environment as listed in Table 3.11-1.

7.7.1.5.5 Operational Considerations

Process variables which are controlled by the pressure regulator and speed/load control systems are displayed on the turbine-generator section of the main control board. Manual and automatic control modes for the various turbine-generator operational modes (such as startup, normal operation, and shutdown) are available to the operator from the main control board. Auto display lights are provided to inform the operator of the operating mode of the turbine-generator unit.

In the event of control malfunction during an automatic control mode, control is transferred to the manual mode, with an alarm to alert the operator of the condition.

7.7.1.6 Neutron Monitoring System - Traversing In-core Probe (TIP) Subsystem - Instrumentation and Controls

7.7.1.6.1 System Identification

7.7.1.6.1.1 General

Flux readings along the axial length of the core are obtained by fully inserting the traversing ion chamber into one of the calibration guide tubes, then taking data as the chamber is withdrawn. The data goes directly to the computer. One traversing chamber and its associated drive mechanism is provided for each group of up to nine fixed in-core assemblies.

The control of the subject system is discussed in this section.

7.7.1.6.1.2 Classification

This system is a power generation system, and is classified as not related to safety.

7.7.1.6.1.3 Reference Design

Table 7.1-2 lists reference design information. The subject instrumentation and control system is an operational system and has no safety function. Therefore, there are no safety design differences between this system and those of the reference design facilities. This system is functionally identical to the referenced system.

7.7.1.6.2 Power Sources

The power for the subject system is supplied from the instrument ac power source.

7.7.1.6.3 Equipment Design

7.7.1.6.3.1 General

The number of TIP machines is indicated in Figure 7.6-4. The TIP machines have the following components:

(1) One Traversing in-core probe (TIP),

(2) One drive mechanism,

(3) One indexing mechanism, and

(4) Up to 10 in-core guide tubes.

The subsystem allows calibration of LPRM signals by correlating TIP signals to LPRM signals as the TIP is positioned in various radial and axial locations in the core. The guide tubes inside the reactor are divided into groups. Each group has its own assocaited TIP machine.

7.7.1.6.3.2 Equipment Arrangement

A TIP drive mechanism uses a fission chamber attached to a flexible drive cable (Figure 7.7-14). The cable is driven from outside the drywell by a gearbox assembly. The flexible cable is contained by guide tubes that penetrate the reactcr core. The guide tubes are a part of the LPRM detector assembly. The indexing mechanism allows the use of a single detector in any one of ten different tube paths. The 10th tube is used for TIP cross calibration with the other TIP machines. The control system provides for both manual and semi-automatic operation. Electronics of the TIP panel amplify and display the TIP signal. Core position versus neutron flux is recorded on an X-Y recorder on a backrow panel in the main control room and is provided to the computer. A block diagram of the drive system is shown in Figure 7.7-10. Actual operating experience has shown the system to reproduce within 1.0% of full scale in a sequence of tests (Reference 7.7-1.)

The TIP system equipment is placed outside but must penetrate an area where containment integrity is needed, the following TIP isolation system is provided. A valve system is provided with a valve on each guide tube entering the drywell. These valves are closed except when the TIP is in operation. A ball valve and a cable shearing valve are mounted in the guide tubing just outside the drywell. They maintain the leak tighness integrity of the drywell. A valve is also provided for a nitrogen gas purge line to the indexing mechanisms. A guide tube ball valve opens only when the TIP is being inserted. The shear valve is used only is a leak occurs when the TIP is beyond the ball valve and power to the TIP fails. The shear valve, which is controlled by a manually operated keylock switch, can cut the cable and close off the guide tube. The shear valves are actuated by detonation squibs.

The continuity of the squib circuits is monitored by indicator lights in the main control room. Upon receipt of containment isolation command from the NSSS, all machines are put in automatic full speed withdraw condition, removing the TIP detector from the containment and allowing the ball valves to close. The purge valve is also closed at this time.

7.7.1.6.3.3 Testability

The TIP equipment is tested and calibrated using heat balance data and procedures described in the instruction manual.

7.7.1.6.4 Environmental Considerations

The equipment and cabling located in the drywell are designed for continuous duty up to 150°F and 100% relative humidity.

7.7.1.6.5 Operational Considerations

The TIP can be operated during reactor operation to calibrate the APRMs. The subject system has no safety set points.

7.7.1.7 Plant Computer System-Instrumentation

The plant computer systems are identified below.

7.7.1.7.1 System Identification

The plant computer system consists of the following:

1. Display Control System (DCS)
2. Performance Monitoring System (PMS) for items other than the NSSS computer system.
3. NSSS Computer System.

7.7.1.7.1.1 General Objectives

The objectives of the DCS system are to monitor unit operation, generate graphic displays for operator use and optimize operation surveillance. The objectives of the PMS system are to perform BOP calculations, log data, make historical records, generate graphic displays and alarm status summary display, and provide off line capabilities. The objectives of the NSSS computer system are to provide a quick and accurate determination of core thermal performance; to improve data reduction, accounting, and logging functions; and to supplement procedural requirements for control rod manipulation during reactor startup and shutdown.

7.7.1.7.1.2 Classification

The plant computer system is a power generation system. All three systems are classified as not related to safety.

7.7.1.7.1.3 Reference Design

Table 7.1-2 lists similarities of reference design information for Susquehanna SES compared to other plants. There are no safety-related design differences between the NSSS computer system and those of the reference design facilities. The NSSS software is functionally identical to the reference system. The DCS and PMS systems have no counterpart in the referenced system.

7.7.1.7.2 Power Sources

The power for the plant computer system is supplied from a designated uninterruptible power supply backed up by an engineered safeguard supply (standby power). See Subsection 8.3.1.8.

7.7.1.7.3 Equipment Design

A block diagram of the plant computer system is shown in Figure 7.7-12.

7.7.1.7.3.1 Circuit Description

The Plant Computer System consists of eight processors, common core, and the associated peripheral equipment as shown in Figure 7.7-12. The processors are random access memory units which utilize a 24-bit word and have a basic cycle time of 800 nanoseconds and a maximum total memory access time of 1.2 microseconds.

7.7.1.7.3.1.1 System Hardware Organization

The equipment is grouped to form the Display Control System and the Performance Monitoring System. The Performance Monitoring System is further broken down into several subsystems, Nuclear Steam Supply (NSS), Balance of Plant/Operator I/O (BOP/OIO) and Historical Records/Program Development (HRPD).

These systems communicate through a shared multiported core memory and a supplementary interprocessor interrupt network. The shared memory functions both as a common data storage device and an interprocessor communications buffer. A shared bulk memory is provided for common access of display formats by the BOP/OIO and DCS processors. This capability is provided by a multi-access bulk memory which connects to the BOP/OIO and the two DCS processors.

Each system, except HRPD interfaces to the plant process through remote I/O units, referred to as Remote Analog Units (RAU) and Remote Digital Units (RDU). These units monitor, signal condition the measurements, digitize the analog measurements, and send the resulting data to a local controller and computer interface module. The RDU also has the ability to output analog and digital signals to the process. The computer which accepts these digitized measurement values is the Data Acquisition Processor (DAP). The exception is the NSS processor which acquires the process data directly from the RDU's and RAU's. The DAP is a 4400 process computer with high speed core memory, and a transparent synchronous data link (TSDL) which enables full duplex serial communications to the host processor.

The Host Processor is another 4400 Process Computer with high speed core memory, transparent data links and a number of peripheral devices depending upon the subsystem. Included in these peripheral devices are CRT video monitors/Display Generators which provide a color presentation of plant variables, alarms, etc. to the plant operators.

The processors on DCS are arranged in a redundant configuration with an inactive Host Processor (DCP) as an operational standby to the active Host Processor (DCP). Both Host Processors operate on the same process data from the DAP's with only the active Host Processor being able to communicate with the video display generators.

The DCS is under the surveillance of an independent Test and Reconfiguration Unit (TRU). The TRU determines the operational status of the major elements of the DCS, annunciates this status to the operator, and activates switchover hardware in the event of a failure of the active DCP.

The NSS subsystem inputs Process I/O points directly to the Host Processor from the RDU and RAU. The processor and peripherals are dedicated to calculations and operations associated with the nuclear steam supply. The only data link to the other subsystems is through common memory.

The HRPD subsystem performs the historical recording, retrieval and analysis functions which have no requirement for direct process inputs; therefore, it does not contain RAU's or RDU's. The peripherals contained on this system are also related to it's secondary function of program development.

7.7.1.7.3.1.2 Display Control System

The DCS consists of the following units:

a. Remote Analog Units (RAUs)
b. Remote Digital Units (RDUs)
c. Data Acquisition Processors (DAPs)
d. Display Control Processors (DCPs)
e. Display Generators (DGs)

f. Video Monitors (CRTs)
g. Drum Memories
h. Test and Reconfiguration Unit (TRU)
i. Data Links
j. Operator Interface Hardware

The DCS collects the unit process data and drives the video monitors located on the Unit Operating Benchboard.

The RAUs and RDUs, under control of Data Acquisition Processors, scan the analog and digital measurements from process instrumentation, signal condition the measurements, digitize the analog measurements, and send the resulting data to the DAPs, as directed by the DAPs.

DAPs adjust the data by performing gain compensation, offset correction, digital filtering, sensor drift limiting, and sensor calibration. The data is checked for error conditions, range limits, and significant changes. Data which changes significantly is sent by each DAP to both DCPs. This data is in engineering units.

The DCPs are arranged in a redundant configuration with the inactive DCP an operational standby to the active DCP. Both DCPs operate on the same process data from the DAPs, with only the active DCP being able to communicate with the DGs.

Each DCP updates its data base with data from the DAPs, formats the data in accordance with the formats selected for each video monitor and, in the case of the active DCP, outputs the formatted data to the video monitors through the display generators.

The video monitor (CRT) displays are controlled by either of two operator actions. There is a "Master Display Control Panel" in the center section of the operating benchboard, plus a Format Selection Switch and a System Assignment Switch associated with each CRT. One of the Format Assignment Switch positions will be indicated as the MASTER position. When the individual Format Selection Switch is in the MASTER position, the format displayed will be the one selected by the operator at the "Master Display Control Panel". From this position, the operator can change the formats on all of the CRTs simultaneously and display the format which is appropriate to the operating activity of the Unit. The operator can choose any of the displays available in a system from the local Format Selection Switch.

7.7.1.7.3.1.3 Performance Monitoring System

The PMS consists of the following units:

Remote Analog Units (RAU)
Remote Digital Units (RDU)
Data Acquisition Processor (DAP)
Nuclear Steam Supply Processor (NSSP)
Balance of Plant and Operator Interface Processor (BOPP)
Historical Records and Program Development Processor (HRPDP)
Display Generators (DG)
Video Monitors (CRT)
Video Keyboards
Common Core Unit (CCU)
Drum Memories
Disc Memory
Magnetic Tape Memory
Video Copier
Typewriter (Terminet)
Cassette Memory
Line Printer
Card Reader
Data Links

The RAU's and RDU's scan the analog and digital measurements from plant instrumentation, signal condition the measurements, digitize the analog measurements, and send the resulting data to one of several DAP's.

The DAP is used as a pre-processor to reduce loading on the BOPP by performing the following adjustments to data: gain compensation, offset correction, digital filtering, sensor drift limiting, and sensor calibration. The data is checked for error conditions, range limits, and significant changes. Data which changes significantly is sent to the BOPP in engineering units.

The NSSP performs the same functions as the DAP plus the calculations and operations associated with the nuclear steam supply. The NSSP has a drum memory for bulk storage and typewriters with cassette storage for operator interface.

The BOPP performs the calculations and operations associated with the balance of plant. The BOPP has a drum memoray for bulk storage. It can receive information from the DCS through the CCU and can communicate with the DCS drum memory. It outputs through display generators to video monitors. Line printers are also available for output, and the BOPP has a link with the Power Control Center (PCC) computer. Operator interface is with a typewriter and video keyboards.

The HRPDP peforms the historical recording, retrieval and analysis functions as well as the program development function. It has a disc for recording data received from the CCU. Data from the disc is periodically transferred to HRPD magnetic tape units for long term storage. The HRPDP can output through a display generator to a video monitor which is switchable from BOP. The HRPD also has a line printer for output and a card reader for input.

7.7.1.7.3.2 Testability

The NSSS computer system has some self-checking provisions. It performs diagnostic checks to determine the operability of certain portions of the system hardware and performs internal programming checks to verify that input signals and selected program computations are either within specific limits or within reasonable bounds.

7.7.1.7.4 Environmental Considerations

All the computer equipment, except for peripherals, is designed for continuous duty up to 120°F, 95% relative humidity ambient. The peripherals are designed to operate at 84°F and 60% relative humidity ambient. This equipment is installed in an air-conditioned room.

7.7.1.7.5 Operational Considerations

7.7.1.7.5.1 General Information

The local power density of every 12-inch segment for every fuel assembly is calculated, by the NSSS computer system, using plant inputs of pressure, temperature, flow, LPRM levels, control rod positions, and the calculated fuel exposure. Total core thermal power is calculated from a reactor heat balance. Iterative computational methods are used to establish a compatible relationship between the core coolant flow and core power distribution. The results are subsequently interpreted as local power at specified axial segments for each fuel bundle in the core.

After calculating the power distribution within the core, the computer uses appropriate reactor operating limit criteria to establish alarm trip settings (ATS) for each LPRM channel. These settings are expressed as maximum acceptable LPRM values to which the actual scanned LPRM readings are compared. The scanned LPRM, when exceeding the ATS, will sound an alarm and thereby assist the operator to maintain core operation within permissible thermal limits established by prescribed maximum fuel rod power density and minimum critical power ratio criteria.

The core power distribution calculation sequence is completed periodically and on demand. The sequence requires approximately 10 minutes to execute. Subsequent to executing the program the computer prints a periodic log for record purposes.

Each LPRM reading is ordinarily scanned once per minute. During power level changes, as sensed by a rod withdrawal or by an APRM channel, the scan rate is increased to once every 5 seconds. This fast core monitoring during power level changes is initiated automatically by the NSSS processor and, together with appropriate computational methods, provides nearly continuous re-evaluation of core thermal limits with subsequent modification to the LPRM ATS based on the new reactor operating level. Execution of these rapid computations does not exceed 3 minutes and yields ATS values that are conservative with respect to the more accurate periodic power distribution calculation, which requires approximately 10 minutes to execute. This range of surveillance and the rapidity with which the computer responds to reactor changes permit more rapid power maneuvering with the assurance that thermal operating limits will not be exceeded.

Flux level and position data from the traversing in-core probe (TIP) equipment are read into the computer. The computer evaluates the data and determines gain adjustment factors by which the LPRM amplifier gains can be altered to compensate for exposure-induced sensitivity loss. The LPRM amplifier gains are not to be physically altered except immediately prior to a whole core calibration using the TIP system. The gain adjustment factor computations help to indicate to the operator when such a calibration procedure is necessary.

Using the power distribution data, a distribution of fuel exposure increments from the time of previous power distribution calculation is determined and is used to update the distribution of cumulative fuel exposure. Each fuel bundle is identified by batch and location, and its exposure is stored for each of the axial segments used in the power distribution calculation. These data are printed out on operator demand.

Exposure increments are determined periodically for each quarter-length section of each control rod. The corresponding cumulative exposure totals are periodically updated and printed out on operator demand.

The exposure increment of each local power range monitor is determined periodically and is used to update both the cumulative ion chamber exposures and the correction factors for exposure-dependent LPRM sensitivity loss. These data are printed out on operator demand.

The NSSS computer system provides on-line capability to determine monthly and on-demand isotopic composition for each one-quarter-length section of each fuel bundle in the core. This evaluation consists of computing the weight of one neptunium, three uranium, and five plutonium isotopes as well as the total uranium and total plutonium content. The isotopic composition is calculated for each one-quarter length of each fuel bundle and summed accordingly by bundles and batches. The method of analysis consists of relating the computed fuel exposure and average void fraction for the fuel to computer stored isotopic characteristics applicable to the specific fuel type.

7.7.1.7.5.2 Reactor Operator Information

Major components are arranged as shown in Figure 7.7-13. Functional description and operational arrangement is as follows:

Unit Operating Benchboard (Panel C651) - houses controls, annunciators and displays, including the control rod position display. The primary process displays are computer generated CRT formats from the DCS and PMS computers. All variables in the DCS displays that are required for unit operation, startup and shutdown are displayed on hardwired indicators on either the Unit Operatiq Benchboard or the Standby Information Panel. These variables in both CRT and hardwired displays generally orginate from the same source.

Standby Information Panel (C652) - houses hardwired indicators and recorders required to startup, run, and shutdown the plant without the use of the Display Control System. It is a hardwired backup to the DCS.

Reactor Core Cooling System BB (C601) - houses hardwired indicators, recorders, annunciators and controls for unit BOP system's functions which do not require the operator's immediate attention during normal operation of the power plant. Functions on this panel have been determined to be long time response functions.

Plant Benchboard (C653) - houses hardwired indicators, recorders annunciators and controls for systems which are common to Units 1 and 2. It also houses two CRT's connected to the Performance Monitoring System (PMS).

Unit Monitoring Console (C684) - provides the unit operator sit down surveillance of the Unit Operating Benchboard and access to DCS and PMS CRT displays with the use of a selection keyboard.

Plant Monitoring Console (C683) - provides sit down surveillance of both units and keyboard access to computer functions of both units. There is also computer generated trend recording of variables from either unit.

The annunciator system is a hardwired system which provides the operator with the alarm information required for unit operation, startup, and shutdown. This system is independent of the Plant Computer System although the computer system does provide redundant and auxiliary alarm information as AID's through the DCS and the alarm status summary CRT display from the PMS.

The Display Control System collects unit process information and presents it on nine of the ten video displays (CRTs) on the Unit Operating Benchboard. One of these nine CRTs is normally connected to the PMS computer system and tabulates all actuated alarms. This CRT may be manually switched to the DCS if operating conditions should require. The tenth CRT is connected to the PMS for operator I/O functions.

Approximately 60 display formats are available to the operator to present process information according to operating mode of the plant.

The DCS makes use of redundant computers which are both updated with current information. Either computer may be automatically or manually switched into operation thus maximizing availability. As stated above, the information presented by the DCS is also displayed in parallel on hardwired indicators and recorders, and alarm information presented in DCS displays is redundant or auxiliary to that provided by the hardwired annunciator system. Thus, the DCS is completely backed up with hardwired information in the event of computer failure. While the operator makes primary use of the computer generated displays, he is not dependent on computers.

DCS displays are arranged by system. Each system has a set of formats. Each format is appropriate to an operating mode of the plant. The operator's designation of the plant operating mode by depressing a pushbutton will automatically cause a format appropriate to that mode to be displayed on each system CRT. However, the format on any CRT may be manually selected by the operator.

Each system format uses the bottom lines for Alarm Initiated Displays (AID). When certain variables in the DCS reach a predetermined limit, (generally the limit is prior to an actual alarm or trip limit) the variable appears in the bottom lines along with other preselected variables. This display provides the operator with specific pre-trip information which is designed to allow him to take action to prevent the trip or alleviate its effects.

7.7.1.8 Reactor Water Cleanup (RWCU) System-Instrumentation and Controls

7.7.1.8.1 System Identification

7.7.1.8.1.1 General

The purpose of the reactor cleanup system instrumentation and control is to provide protection for the system equipment from overheating and overpressurization and to provide operator information concerning the effectiveness of operation of the system.

7.7.1.8.1.2 Classification

This is a power generation system and is classified as not related to safety.

7.7.1.8.1.3 Reference Design

Table 7.1-2 lists reference design information. The subject control system is an operational system and has no safety function. Therefore, there are no safety design differences between this system and those of the reference design facilities. This system is functionally identical to the referenced system.

7.7.1.8.2 Power Sources

The RWCU system instrumentation and controls are fed from the plant instrumentation bus. No backup power source is necessary since the RWCU system is not a safety related system. Adequate fuse protection is provided so that a short circuit within the system will have only a local effect which can be easily corrected without interrupting the reactor operation.

7.7.1.8.3 Equipment Design

7.7.1.8.3.1 General

The reactor water cleanup system is described in Subsection 5.4.8. This subsection describes !)t the systems used to protect the resin and the filter-demineralizer. These circuits are shown in Figure 5.4-16 and the operating logic is shown in Figure 7.7-11.

7.7.1.8.3.2 Circuit Description

To prevent resins from entering the reactor recirculation system in the event of a filter-demineralizer resin support failure, a strainer is installed on the outlet of each filter-demineralizer unit. Each strainer is provided with a control room alarm, which is energized by high differential pressure. A bypass line is provided around the filter-demineralizer units for bypassing the units when necessary. Figure 5.4-18 describes the filter-demineralizer instrumentation and control.

Relief valves and instrumentation are provided to protect the equipment against over-pressurization and the resins against overheating. The system is automatically isolated for the reasons indicated when signaled by any of the following occurrences:

(1) High temperature downstream of the nonregenerative heat exchanger - to protect the ion exchange resins from deterioration due to high temperature,

(2) Reactor vessel low water level - to protect the core in case of a possible break in the reactor water cleanup system piping and equipment (see Subsection 7.3.1.1a.2.4.1.1).

(3) Standby Liquid Control System actuation - to prevent removal of the boron by the cleanup system filter-demineralizers,

(4) High cleanup system ambient temperature - (part of the plant leak detection system),

(5) High temperature increase across the system's ventilation ducts - (part of the plant leak detection system),

(6) High change in system inlet flow in comparison to the system outlet flow - (part of the plant leak detection system).

In the event of low flow or loss of flow in the system, flow is maintained through each filter-demineralizer by its own holding pump. Sample points are provided upstream and downstream of each filter-demineralizer unit for continuous indication and recording of system conductivity. High conductivity is annunciated in the main control room. The influent sample point is also used as the normal source of reactor coolant samples. Sample analysis also indicates the effectiveness of the filter-demineralizer units.

7.7.1.8.3.3 Testability

Because the reactor water cleanup system is usually inservice during plant operation, satisfactory performance is demonstrated without the need for any special inspection or testing beyond that specified in the manufacturer's instructions.

7.7.1.8.4 Environmental Considerations

The reactor water cleanup system is not required for safety purposes, nor required to operate after the design basis accident. The reactor water cleanup system is required to operate in the normal plant environment for power generation purposes only.

RWCU instrumentation and controls located in the RWCU equipment area are subject to the environment described in Table 3.11-1.

7.7.1.8.5 Operational Considerations

7.7.1.8.5.1 General Information

The reactor water cleanup system-instrumentation and control is not required for safe operation of the plant. It provides a means of monitoring parameters of the system and protecting the system.

7.7.1.8.5.2 Reactor Operator Information

Refer to the RWCU system instrumentation and control Figures 5.4-16 and 7.7-11.

7.7.1.8.5.3 Set Points

Safety set points associated with the subject system are discussed in Subsection 7.6.1a.4.3.6.

7.7.1.9 Transient Monitoring System for Startup Testing

General Electric Co. (GE) provides as a part of Startup Testing (Low Power and Power Ascension Testing), for purposes of measurement and recording of process transients, a Transient Recording System (STARTREC). STARTREC is brought in for temporary installation and consists primarily of a data acquisition computer and peripherals for control and output (recording).

In Susquehanna SES STARTREC will be located within sight of the plant-operator interface area of the control room, in the Observation Gallery area, elevation 741'. Communications will be provided between STARTREC and the operator interface area.

STARTREC will be removed after startup testing. 14

7.7.1.9.1 Transient Monitoring System (TMS) Description

The equipment for providing and conditioning of transient signals for STARTREC is called the Transient Monitoring System (TMS). The scope of this system includes permanent mounting of devices in NSSS and non-NSSS panels and permanent installation of signal cables from the systems panels to a permanent control room (Observation Gallery) panel. This panel collects and conditions the TMS signals and is called the Transient Monitoring Panel (TMP).

7.7.1.9.1.1 Piping Thermal Expansion Measurement

Measurements of piping thermal expansion and vibration during startup testing is handled by the use of multiplexing signals

directly to STARTREC. Signal conditioning and multiplexers are provided as packages and use a single coaxial cable from each multiplexer (one inside containment and one outside) to STARTREC.
14 Instrumentation and multiplexers are temporarily installed and will be removed at the conclusion of the measurements. Neither the equipment nor the cable used for these measurements are safety related.

7.7.1.10 Refueling Interlocks System-Instrumentation and Controls

7.7.1.10.1 System Identification

The purpose of the refueling interlocks system is to restrict the movement of control rods and the operation of refueling equipment to reinforce operational procedures that prevent the reactor from becoming critical during refueling operations.

This equipment is not required to operate during a seismic event. The operability of the equipment can be verified after a seismic event without jeopardizing safety.

7.7.1.10.2 Power Sources

There is only one source of power for both channels of the logic
17 circuits (see Subsection 7.7.1.10.3.2). However, this power source supplies the Control Rod Drive System as well. A failure of this power supply will prevent any rod motion.

7.7.1.10.3 Equipment Design

7.7.1.10.3.1 Circuit Description

The refueling interlocks circuitry senses the condition of the refueling equipment and the control rods. Depending on the sensed condition, interlocks are actuated to prevent the movement of the refueling equipment or withdrawal of control rods (rod block). Single-failure-proof circuitry is provided to sense the following conditions:

(1) All rods inserted (see Subsection 7.7.1.10.3.2)

(2) Refueling platform positioned near or over the core

(3) Refueling platform hoists fuel-loaded (fuel grapple, frame-mounted hoist, trolley-mounted hoist)

(4) Fuel grapple not fully restricted

(5) Service platform hoist fuel-loaded, and

(6) Reactor Mode Switch in "Refuel" position.

7.7.1.10.3.2 Logic and Sequencing

The indicated conditions are combined in logic circuits to satisfy all restrictions on refueling equipment operations and control rod movement (Figure 7.7-3). A two-channel circuit indicates that all rods are in. The rod-in condition for each rod is established by the closure of a magnetically operated reed switch in the rod position indicator probe. The rod-in switch must be closed for each rod before the all-rods-in signal is generated. This is not the same switch that provides rod position information to the process computer and four rod position display. Both channels must register the all-rods-in signal in order for the refueling interlock circuitry to indicate the all-rods-in condition.

During refueling operations, no more than one control rod is permitted to be withdrawn; this is enforced by a redundant logic circuit that uses the all-rods-in signal and a rod selection signal to prevent the selection of a second rod for movement with any other rod not fully inserted. Control rod withdrawal is prevented by comparison checking between the A and B portions of the reactor manual control system and subsequent message transmission to the affected control rod. The simultaneous selection of two control rods is prevented by the interconnection arrangement of the select push buttons. With the mode switch in the REFUEL position, the circuitry prevents the withdrawal of more than one control rod and the movement of the loaded refueling platform over the core with any control rod withdrawn.

Operation of refueling equipment is prevented by interrupting the power supply to the equipment. The refueling platform is provided with two mechanical switches attached to the platform, which are tripped open by a long, stationary ramp mounted adjacent to the platform rail. The switches open before the platform or any of its hoists are physically located over the reactor vessel to indicate the approach of the platform toward its position over the core.

Load cell readout is provided for all hoists. Indicators display given hoist loads directly to the operator. Load sensing is by hydraulic load cells that use demineralized water as the operating fluid. Associated interlock and load functions are performed by pressure switches that sense the pressure generated by the hydraulic load cells.

The three hoists on the refueling platform and the hoist on the service platform are provided with switches that open when the hoists are fuel-loaded. The switches open at a load weight that is lighter than that of a single fuel assembly. This indicates when fuel is loaded on any hoist.

The telescoping fuel grapple hoist has a geared limit switch. The switch is open any time the grapple has descended more than approximately 4 inches from its fully retracted position. This switch is placed in series with the grapple load switch to ensure interlock operation if the weight of the bottom section of the telescope plus the fuel is less than the preset load.

7.7.1.10.3.3 Bypasses and Interlocks

A bypass for the service platform hoist load interlock is provided. When the service platform is no longer needed, its power plug is removed. This deenergizes the power supply to the hoist. The platform can then be moved away from the core.

Deenergizing the hoist power supply opens the hoist load switches and gives a false indication that the hoist is loaded. This indication prevents control rod withdrawal with the mode switch in the STARTUP or REFUEL positions. A bypass plug allows control rod movement in this situation. The bypass plug is physically arranged to prevent the connection of the service platform power plug unless the bypass plug is removed.

The rod block interlocks and refueling platform interlocks provide two independent levels of interlock action. The interlocks which restrict operation of the platform hoist and grapple provide a third level of interlock action since they would be required only after a failure of a rod block and refueling platform interlock. The strict procedural control exercised during refueling operations may be considered a fourth level of backup.

7.7.1.10.3.4 Redundancy and Diversity

Although the refueling interlocks are not designed nor required to meet the IEEE 279-1971 criteria for Nuclear Power Plant Protection Systems. Failure of the refueling interlocks will neither cause an accident nor prevent safety related systems from performing their protective actions. They are provided for use during planned refueling operations. Criticality is prevented during the insertion of fuel, provided control rods in the vicinity of the vacant fuel space are fully inserted during the fuel insertion. The interlock systems accomplish this by:

(1) Preventing operation of the loaded refueling equipment over the core whenever any control rod is withdrawn.

(2) Preventing control rod withdrawal whenever fuel loading equipment is over the core.

(3) Preventing withdrawal of more than one control rod when the mode switch is in the refuel position.

The refueling interlocks have been carefully designed utilizing redundancy of sensors and circuitry to provide a high level of reliability and assurance that the stated design bases will be met. Each of the individual refueling interlocks discussed above need not meet the single failure criterion because the four essentially independent levels (including procedural control) of protection provide assurance that the design basis will be met. For any of the "situations" listed in Table 7.7-2 a single interlock failure will not cause an accident or result in potential physical damage to fuel or result in radiation exposure to personnel during fuel handling operations.

7.7.1.10.3.5 - Actuated Devices

The refueling interlocks from the Reactor Manual Control System to the refueling equipment trip a relay in the refueling equipment controls which interrupts power to the equipment and prevents it from moving over the core.

The interlocks from the refueling equipment to the Reactor Manual Control System actuate circuitry that provides a control rod block. The rod block prevents the operator from withdrawing any control rods.

7.7.1.10.3.6 Separation

The refueling interlocks are not designed to nor required to meet the IEEE 279-1971 criteria for Nuclear Power Plant Protection Systems. However, a single interlock failure will not cause an accident and are used in conjunction with admistration controls during planned refueling operations.

7.7.1.10.3.7 - Testability

Complete functional testing of all refueling interlocks before any refueling outage will positively indicate that the interlocks operate in the situations for which they were designed. The interlocks can be subjected to valid operational tests by loading each hoist with a dummy fuel assembly, positioning the refueling platform, and withdrawing control rods. Where redundancy is provided in the logic circuitry, tests are performed automatically, on a periodic basis, to assure that each redundant logic element can independently perform its function.

7.7.1.10.4 - Environmental Considerations

Equipment (refueling) will be subjected to the conditions listed in Table 3.11-1 during normal operation. The refueling interlocks are not required to operate under these conditions, but are required to be operable under the conditions listed in Table 3.11-3.

Refueling components are capable of surviving design basis events such as earthquakes, accidents, and anticipated operational occurrences without consequential damage, but are not required to be functional during or after the event without repair.

7.7.1.10.5 - Operational Considerations

7.7.1.10.5.1 General Information

The refueling interlocks system is required only during refueling operations.

7.7.1.10.5.2 Reactor Operator Information

In the refueling mode, the control room operator has an indicator light for "Refueling Mode Select Permissive" whenever all control rods are fully inserted. He can compare this indication with control rod position data from the computer as well as control rod in-out status on the full core status display. Furthermore, whenever a control rod withdrawal block situation occurs, the operator receives annunciation and computer logs of the rod block. He can compare these outputs with the status of the variable providing the rod block condition. Both channels of the control rod withdrawal interlocks must agree that permissive conditions exist in order to move control rods; otherwise, a control rod withdrawal block is placed into effect. Failure of one channel may initiate a rod withdrawal block, and will not prevent application of a valid control rod withdrawal block from the remaining operable channel.

Core flux activity monitoring is provided during refueling by the SRM's and/or dunking chambers which are specified and controlled in Technical Specification 3/4.9.

In terms of refueling platform interlocks, the platform operator has analog type readout indicators for the platform x-y position relative to the reactor core.

The position of the grapple is shown on a digital indicator immediately below the platform position indicators. Analog load cell indications of hoist loads are given for each hoist by locally mounted indicators. Individual push button and rotary control switches are provided for local control of the platform and its hoists. The platform operator can immediately determine whether the platform and hoists are responding to his local instructions, and can, in conjunction with the control room operator, verify proper operation of each of the three categories of interlocks listed previously.

7.7.1.10.5.3 - Set Points

There are no safety set points associated with this system.

7.7.1.11 - Rod Block Monitor (RBM) Subsystem

7.7.1.11.1 - Equipment Design

7.7.1.11.1.1 - Description

The RBM has two channels. Each channel uses input signals from a number of LPRM channels. A trip signal from either RBM channel can initiate a rod block. One RBM channel can be bypassed without loss of subsystem function. The minimum number of LPRM inputs required for each RBM channel to prevent an instrument inoperative alarm is four when using four LPRM assemblies, three when using three LPRM assemblies, and two when using two LPRM assemblies (Figure 7.7-16).

(1) Power Supply

The RBM power is received from the 120 V ac supplies for the RPS. RBM channel A receives power from the ac bus used for RPS Trip System A; RBM channel B receives power from the ac bus used for RPS trip system B.

(2) Signal Conditioning

The RBM signal is generated by averaging a set of LPRM signals. One RBM channel averages the signals from LPRM detectors at the A and C positions in the assigned LPRM assemblies. The second RBM channel averages the signals from the LPRM detectors at the B and D positions. Assignment of LPRM assemblies to be used in RBM averaging is controlled by the selection of control rods. Figure 7.7-16 illustrates the four possible assignment combinations. Note that the RBM is automatically bypassed and the output set to zero if a peripheral rod is selected. If any LPRM detector assigned to an RBM is bypassed, the computed average signal is adjusted automatically to compensate for the number of LPRM input signals.

When a control rod is selected, the gain of each RBM channel output is normalized to an assigned APRM channel. The assigned APRM channel is on the same RPS trip system as the RBM channel. This gain setting is held constant during the movement of that particular control rod to provide an indication of the change in the relative local power level. If the APRM used to normalize the RBM reading is indicating less than 30% power, the RBM is zeroed and the RBM outputs are bypassed.

If the normalizing APRM is bypassed, the normalizing signal is automatically provided by a second APRM. In the operating range, the RBM signal is accurate to approximately 1% of full scale.

7.7.1.11.1.2 Rod Block Trip Function

The RBM supplies a trip signal to the reactor manual control system to inhibit control rod withdrawal. The trip is initiated when RBM output exceeds the rod block set point. There are three parallel rod block set point lines that have an adjustable slope. These lines provide a set point that is a function of the recirculation driving loop flow. The intercepts of these set point lines with rated flow are adjustable. The normal settings are approximately 107% for the upper line, 99% for the intermediate line, and 91% for the lower line. Lights indicate which rod block set point line is active. Two percent below the intermediate and lower rod block set point lines are the set-up-permissive (and setdown) lines. When the power reaches these lines on increasing power, an indicator will light so the operator can evaluate the conditions and manually change to the next higher rod block set point line. On decreasing power, these lines will provide automatic set down. Either RBM can inhibit control rod withdrawal (Figure 7.6-7). Table 7.7-3 itemizes the RBM trip functions.

7.7.1.11.1.3 Bypasses

The operator can bypass one of the two RBMs at any time (see Subsection 7.6.1a.6.3).

7.7.1.11.1.4 Redundancy

The following features are included in RBM design:

(1) Redundant, separate, and isolated RBM channels.

(2) Redundant, separate, isolated rod selection information (including isolated contacts for each rod selection pushbutton) provided directly to each RBM channel.

(3) Separate, isolated LPRM amplifier signal information provided to each RBM channel.

(4) Separate and electrically isolated recirculation flow inputs provided to the RBMs for trip.

(5) Independent, separate, isolated APRM reference signals to each RBM channel.

(6) Independent, isolated RBM level readouts and status displays from the RBM channels.

(7) Mechanical barrier between Channel A and Channel B of the manual bypass switch.

(8) Independent, separate, isolated rod block signals from the RBM channels to the manual control system circuitry.

7.7.1.11.1.5 Testability

The rod block monitor channels are tested and calibrated with procedures given in the applicable instruction manuals. The RBMs are functionally tested by introducing test signals into the RBM channels.

17 7.7.1.11.2 Environmental Considerations

(See description for APRM, Subsection 7.6.1a.5.6.2)

7.7.1.11.3 Operational Considerations

When increasing power, the set-up permissive lamp will light at which time the operator must evaluate conditions before manually changing to the next higher rod block set point line.

7.7.1.12 Nuclear Pressure Relief System

7.7.1.12.1 System Identification

The Nuclear Pressure Relief System, consisting of safety relief valves and associated circuitry, is designed to limit nuclear steam supply system pressure under various modes of reactor operation.

18 7.7.1.12.2 Equipment Design

The Nuclear Pressure Relief System controls and instrumentation consist of manual control/pressure sensor channels each dedicated to its respective safety relief valve and associated valve operator (solenoid operated air pilot valve). The pilot valve controls the pneumatic pressure applied to the air cylinder operator. Upon energizing the pilot valve, pneumatic pressure is directed from the accumulator to act on the air cylinder operator causing the safety relief valve to open. Upon again de-energizing the pilot valve, air in the air cylinder is exhausted and the accumulator is once again isolated via the de-energized pilot valve. An accumulator, one for each valve, is included with the control equipment to store the pneumatic energy for safety relief valve operation. Safety relief valves are automatically initiated by high reactor pressure conditions. Cables from the pressure sensors for vessel pressure are routed to a single logic cabinet in the main control room. Power to the safety relief valves' pilot valves and associated pressure sensors is divided between DC busses A and B, with each bus serving specified safety relief valves and associated circuitry. The logic cabinet provides for appropriate separation of power supply feeders so as to limit the effects of electrical failures.

7.7.1.12.3 Initiating Circuits

Reactor pressure is detected by pressure sensors (one for each valve) which are located in the reactor building. The logic for each valve requires a single sensor trip on vessel pressure to cause safety relief valve actuation.

7.7.1.12.4 Logic and Sequencing

One initiation signal is used for each safety relief valve actuation via each respective pressure sensor output. High vessel pressure indicates the need for safety relief valve actuation to limit nuclear steam supply pressure.

Upon receipt of an initiation signal the pilot air valve is energized thereby opening the safety relief valve. Lights in the main control room indicate when the solenoid-operated pilot valve are energized to open a safety relief valve. The safety relief valves remain open until the system pressure drops below the high pressure setpoint.

Manual system level initiation of a safety relief valve is accomplished by a control switch in either division 1 or division 2 depending on which division is serving a given valve and its associated logic circuitry.

7.7.1.12.5 Bypasses and Interlocks

Bypasses and interlocks are not utilized in the safety relief valve function.

7.7.1.12.6 Redundancy and Diversity

The safety relief valve logic is initiated by high reactor pressure. Though redundancy is not provided for initiating signals to a given safety relief valve, it is provided with separate sensor signals each to different valves. Diversity is not provided.

7.7.1.12.7 Actuated Devices

Safety relief valves are actuated by four methods:

a. Automatically on high reactor pressure via pressure sensors.
b. Manually, by the operator.
c. Mechanically, through spring setpoints.
d. Automatically or manually as part of ADS (Section 7.3.1.1a.1.4)

7.7.1.12.8 Separation

Safety relief valve logic is of single channel design for each valve. Safety relief valves and associated logics are assigned between DC busses A and B. Cable routing, logic circuitry, manual controls and instrumentation are appropriately separated to limit the effects of a single failure.

7.7.1.12.9 Testability

Safety relief valve logic is testable up to and including the sensors and actuated equipment.

7.7.1.12.10 Environmental Considerations

The solenoid valves and their cables and the safety relief valves operators are located inside the drywell and will operate during normal and projected accident environmental conditions. The pressure sensors, which are located within the reactor building will also operate during normal and accident environments.

7.7.1.12.11 Operational Considerations

7.7.1.12.11.1 General Information

The instrumentation and controls of the Nuclear Pressure Relief System are required for normal plant operations to limit nuclear system pressure. When pressure relief action is required, it will be initiated automatically by the circuits described in this section.

7.7.1.12.11.2 Operator Information

A temperature element is installed on the safety relief valve discharge piping approximately three feet from the valve body. The temperature element is connected to a multipoint recorder in the control room to provide a means of detecting safety relief valve leakage during the plant operation. When the temperature in any safety relief valve discharge piping exceeds a preset value, an alarm is sounded in the control room. The alarm setting is far enough above normal (rated power) drywell ambient temperatures to avoid spurious alarms, yet low enough to give early indication of significant safety relief valve leakage. 18

7.7.1.13 Loose Parts Monitoring System

The Loose Parts Monitoring System will monitor, alarm and record the Reactor Vessel acoustics for the presence of internal loose parts in accordance with R.G.1.133 Draft-2 Rev. 1.

The system will monitor the points listed below. When an impact event signal exceeds a selectable amplitude, an alarm will occur and peak impact and impact repetition will automatically be recorded and timed sequentially, for each selected channel.

Eight piezoelectric accelerometers are attached externally to the Reactor Vessel: 22

a. Two mounted approx. 180° apart on or near the main steam lines to monitor the upper head regions.

b. Two mounted approx. 180° apart on or near the feedwater lines to monitor the upper vessel regions.

c. Two mounted approx. 180° apart and at 90° rotation from the upper vessel sensors mounted on or near the recirculation suction lines to monitor the vessel core plate region.

d. Two mounted approx. 90° apart, one on a CRD Housing and the other on the RPV drain piping, to monitor the lower vessel regions.

7.7.2 ANALYSIS

This subsection:

(1) demonstrates by direction or referenced analysis that the subject described systems are not required for any plant safety function, and

(2) demonstrates by direct or referenced analysis that the plant protection systems described elsewhere are capable of coping with all failure modes of the subject control systems.

In response to item (1) above, the following is cited. Upon considering the design basis, descriptions, and evaluations presented here and elsewhere throughout the document relative to the subject system, it can be concluded that these systems do not perform any safety function.

design basis: refer to Subsection 7.1.1
description: refer to Subsection 7.7.1

The individual system analysis in this section concludes that the subject systems are not required for any plant safety action.

For consideration of item (2) above, it is necessary to refer to the safety evaluations in Chapter 15 and Appendix 15A.

In that chapter, it is first shown that the subject systems are not utilized to provide any design basis accident safety function. Safety functions, where required, are provided by other qualified systems. For expected or abnormal transient incidents following the single operation error (SOE) or single component failure (SCF) criteria, protective functions are also shown to be provided by other systems. The expected or abnormal transients cited are the limiting FMEA for the subject systems.

Next, further considerations of situations beyond the SOE and SCF, specified as single active component failure (SACF), are analyzed in Chapter 15 and Appendix 15A. Although these are not design basis requirements, the ability of the plant to provide at least one single protective function, even under these stringent assumptions, is demonstrated.

7.7.2.1 Reactor Vessel Instrumentation

7.7.2.1.1 General Functional Requirements Conformance

The reactor vessel-instrumentation is designed to provide redundant or augmented information to the existing information required from the engineered safeguards and safety systems. The operator utilizes this information to start up, operate at power, shut down, and service the reactor system in an efficient manner. None of this instrumentation is required to initiate or control any engineered safeguard or safety system.

7.7.2.1.2 Specific Regulatory Requirements Conformance

There are no specific regulatory requirements imposed on this reactor vessel instrumentation but the following general considerations are offered:

(1) Conformance with General Design Criteria 13

The reactor vessel information provides the operator with information on the reactor vessel operating variables during normal plant operation and anticipated operational occurrences so that the need to use the safety systems, although ready and able to respond, is minimized. This instrumentation does not serve in any direct controlling functions. Controls that maintain the reactor vessel operating variables within prescribed operating ranges are performed by the:

(a) feedwater system
(b) RCIC system
(c) reactor manual control system or rod control and information system

(2) Conformance with General Design Criteria 24

This instrumentation is not part of or related to any safety system. The circuitry of the safety systems is completely independent of this instrumentation such that failures of this instrumentation will not cause or prevent any action to be initiated by the safety systems.

(3) Conformance to IEEE STD 279, section 4.7

This instrumentation is separate from and independent of the safety systems circuitry. There is no direct circuit-to-circuit or functional interactions between this instrumentation and the safety systems. No single random or multiple failures in this instrumentation can prevent the safety systems from meeting the minimum performance requirements specified in the design basis of that system.

7.7.2.2 Reactor Manual Control System-Instrumentation and Controls

7.7.2.2.1 General Functional Requirements Conformance

The circuitry described for the reactor manual control system is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. The scram circuitry is discussed in Section 7.2. Because each control rod is controlled as an individual unit, a failure that results in energizing of any of the insert or withdraw solenoid valves can affect only one control rod. The effectiveness of a reactor scram is not impaired by the malfunctioning of any one control rod. It can be concluded that no single failure in the reactor manual control system can result in the prevention of a reactor scram and that repair, adjustment, or maintenance of reactor manual control system components does not affect the scram circuitry.

Chapter 15 and Appendix 15A examine the various failure mode considerations for this system. The expected and abnormal transients and accident events analyzed envelope the FMEA associated with this system's components. These include:

(1) control rod withdrawal errors

(2) control rod drop accident

To be very specific, the following is cited:

(1) The RMCS is not required for plant safety functions. The system has no function associated with any design basis accident.

(2) This system is not used for plant shutdown resulting from an accident or nonstandard operational conditions.

(3) The function of the RMCS is to control core reactivity and thus power level. Interlocks from many different sources are incorporated to prevent the spurious operation of drives or undesirable rod patterns throughout all ranges of operation.

(4) This system contains no components, circuits or instruments required for reactor trip or scram. There are no operator manual controls which can prevent scram.

(5) The consequence of improper operator action or the failure of rod block interlocks results in a reactor scram.

(6) The requirements for the portions of RMCS that interface with any safety system function includes tolerance to single failures and component quality.

7.7.2.2.2 Specific Regulatory Requirements

There are no specific requirements imposed on this system, but the following general considerations are offered:

(1) 10CFR50 Appendix A - Criterion 24

No part of the RMCS is required for scram. The rod block functions provided by the NMS are the only instances where the RMCS uses any instruments or devices related to RPS functions. The rod block signals received from the NMS prevent improper rod motion before limits causing reactor scram are reached. Common APRM, IRM, and SRM detectors are used, but the signal is physically and electrically isolated before its use in the reactor manual control system. See Subsections 7.6.1a.5 and 7.6.2a.5 for a description of this interface. This isolation is achieved through two separate relay trip units which prevent any feedback from the reactor manual control system to the reactor protection system. Single failure of a control component therefore will not degrade the protection system.

(2) 10CFR50 Appendix A - Criterion 26

The RMCS is one of the two, independent reactivity control systems as required by this criterion.

7.7.2.3 Recirculation Flow Control System-Instrumentation and Controls

7.7.2.3.1 General Functional Requirements Conformance

The recirculation flow control system is designed so that coupling is maintained between an M-G set drive motor and its generator even if the ac power or a speed controller signal fails. This assures that the drive motor inertia will contribute to the power supplied to the recirculation pump during coastdown of the M-G set after loss of ac power, and that the generator continues to be driven if the speed controller signal is lost.

Transient analyses described in Chapter 15 show that no malfunction in the recirculation flow control system can cause a transient sufficient to damage the fuel barrier or exceed the nuclear system pressure limits, as required by the safety design basis.

The safety design basis of the recirculation flow control loop is that no single component failure shall result in a violation of the plant transient MCPR limit.

The main recirculation process control system is not required to be designed to meet the single failure criterion. Control system failures resulting in complete loss of control signal will result in electrical "locking" of the scoop tube in its last demanded position at the instant of signal loss.

In the case of recirculation control system failures (e.g., transistors, resistors, etc.) causing upscale signal failure, the reactor is protected by high pressure or high flux scram. See Figure 7.7-7. Such faults have been analyzed in Chapter 15 and include both M-G sets going to full speed simultaneously.

Recirculation system flow control failures causing downscale signal failures may cause one or both recirculation M-G sets to go to minimum speed. M-G set speed reduction is limited to not more than 10% per second. Speed reduction of both M-G sets might result from failure of the master controller.

Control component failures such as the speed controller setpoint station, error limiter, speed controller, function generator, and recirculation M-G set speed limiters result in a single M-G set speed reduction at 10% per second. The function generator limits the load demand error signals from the turbine pressure regulator to a normal maximum of 10%.

Recirculation M-G set speed limiters are provided to prevent recirculation pump, valve and jet pumps from operating in regions that would cause cavitation damage to these components.

The recirculation pump valves are treated as conventional remote motoroperated valves. From a circuit viewpoint, each recirculation pump valve is independent of the other, and has its own Unit Operating Benchboard mounted control switch for manual operation. Each valve has open/close travel limit switches and Unit Operating Benchboard pilot lamp indication. (See Figure 7.5-1.)

Chapter 15 and Appendix 15A examine the various failure mode considerations for this system. The expected and abnormal transients and accident events analyzed envelope the FMEA associated with this system's components. These include:

(1) Recirculation flow controller failures

(2) Recirculation pump seizure and pump shaft failure

7.7.2.3.2 Specific Regulatory Requirements

There are no specific regulatory requirements imposed on this system.

7.7.2.4 Feedwater Control System (Turbine-Driven Pumps) Instrumentation and Controls

7.7.2.4.1 General Functional Requirements Conformance

The feedwater control system is a power generation system for purposes of maintaining proper vessel water level. Interlocks are provided to lock the flow changing capabilities in the "as-is" condition in the event of control signal failure. Should the vessel level rise too high, the feedwater pumps and plant main turbine would be tripped. This is an equipment protective action which would result in reactor shutdown by the RPS system as outlined in Section 7.2. Lowering of the vessel level would also result in action of the RPS to shutdown the reactor.

Chapter 15 and Appendix 15A examine the various failure mode considerations for this system relative to plant safety and operational effects. The expected and abnormal transients and accident events analyzed in the appendix envelope the FMEA associated with this system's components. These include:

(1) Loss of all feedwater flow (pumps)

(2) Loss of feedwater heater

(3) Malfunction of feedwater controller

(4) Failure of feedwater line

7.7.2.4.2 Specific Regulatory Requirements Conformance

The feedwater system is not a safety-related system and is not required for safe shutdown of the plant, nor is it required during or after accident conditions.

There are no interconnections with safety-related systems and no specific regulatory requirements are imposed on the system.

7.7.2.5 Pressure Regulator and Turbine-Generator System Instrumentation and Controls

7.7.2.5.1 General Functional Requirements Conformance

Turbine speed and acceleration control is provided by the initial pressure regulator which controls steam throttle valve position to maintain constant reactor pressure. The turbine speed governor overrides the pressure regulator on increase of system frequency or loss of generator load. Excess steam is automatically bypassed directly to the main condenser by the pressure controlled bypass valves.

Provision is made for matching nuclear steam supply to turbine steam requirements. As pressure is lowered by a greater load demand, the pressure regulator sends a proportional signal to the recirculation flow control system which causes an appropriate increase in recirculation flow. Detailed description of conformance to these design bases is contained in Subsection 7.7.1.

Chapter 15 and Appendix 15A examine the various failure mode considerations for this system relative to plant safety and operational effects. The expected and abnormal transient and accidents events analyzed in this appendix envelope the FMEA associated with this system's components. These include:

(1) Failure of pressure regulator

(2) Turbine/generator trips

(3) Main condenser failures

(4) Breaks outside containment

7.7.2.5.2 Specific Regulatory Requirements Conformance

No specific regulatory requirements are imposed on the subject system.

The turbine-generator control system is not a safety related system. Protection systems which are provided as an integral part of the turbine-generator equipment override the turbine-generator control system. In the event of a turbine-generator trip due to a protective action, the control valve fast closure and the stop valve closure inputs to the RPS initiate reactor scram. (See Subsections 7.2.1.1.4.2(d) and 7.2.1.1.4.2(e).)

Pressure regulator malfunction which leads to low turbine inlet pressure is detected by pressure switches provided in the main steam isolation system which in turn initiates closure of the main steamline isolation valves (See Subsection 7.3.1.1a.3). Similarly, high turbine inlet pressure leads to detection of high reactor pressure by the RPS which initiates reactor scram. (See Subsection 7.2.1.1.4(b)).

Control malfunction (e.g., pressure regulation malfunction - upscale) which results in high flow through the turbine control valves and the bypass valves is detected by main steam flow switches provided in the main steam isolation system which initiates closure of the main steam level isolation valves (See Subsection 7.3.1.1a.3) and a subsequent reactor scram. (See Subsection 7.2.1.1.4(f)).

Interfaces between the subject non-safety systems and their components with safety-related systems (RPS, containment isolation control system, etc) are design in such a manner that failure of the non-safety components will not negate the necessary safety system functions.

7.7.2.6 Neutron Monitoring System Traversing In-Core Probe Subsystem (TIP) - Instrumentation and Controls

7.7.2.6.1 General Functional Requirement Conformance

An adequate number of TIP machines is supplied to assure that each LPRM assembly can be probed by a TIP and that one LPRM assembly (the central one) can be probed by every TIP to allow intercalibration. Typical TIPs have been tested to prove linearity. (Reference 7.7-1) The system has been field-tested in an operating reactor to assure reproducibility for repetitive measurements. The mechanical equipment has undergone life testing under simulated operating conditions to assure that all specifications can be met. The system design allows semi-automatic operation for LPRM calibration and process computer use. The TIP machines can be operated manually to allow pointwise flux mapping.

7.7.2.6.2 Specific Regulatory Requirement Conformance

There are no specific regulatory requirements for the TIP subsystem.

7.7.2.7 Plant Computer System - Instrumentation

7.7.2.7.1 General Functional Requirements

The Plant Computer System is designed to provide the operator with certain categories of information as defined in the equipment description (Subsection 7.7.1.7) and to supplement procedural requirements for control rod manipulation during reactor startup and shutdown. The system augments existing information from other systems such that the operator can start up, operate at power, and shutdown in an efficient manner. This system is not required to initiate or control any engineered safeguard or safety-related system.

7.7.2.7.2 Specific Regulatory Requirements Conformance

The plant computer has no specific regulatory requirements.

7.7.2.8 Reactor Water Cleanup System - Instrumentation and Controls

7.7.2.8.1 General Functional Requirement Conformance

The RWCU system is not a safety-related system. Therefore, the instrumentation supplied is for the plant equipment protection and for operator information only.

The cleanup system is protected against overpressurization by relief valves. The ion exchange resin is protected from high temperature by temperature switches upstream of the filter demineralizer unit. One switch activates an alarm while a second switch closes the isolation valve which subsequently trips the cleanup pumps. The isolation valves will also close automatically on a reactor low water level signal and when the standby liquid control system is actuated. The pumps will also trip on high cooling water temperature or low discharge flow.

A high differential pressure across the filter-demineralizer or its discharge strainer will automatically close the units outlet valve after sounding an alarm. The holding pump starts whenever there is low flow through a filter-demineralizer. The precoat pump will not start or stop when the level in the precoat tank is low.

Sampling stations are provided to obtain reactor water samples from the entrance and exit of both filter-demineralizers.

The system control and instrumentation for flow, pressure, temperature, and conductivity are recorded or indicated on a panel in the main control room. Instrumentation and control for backwashing and precoating the filter-demineralizers are on a local panel outside the drywell. Alarms are sounded in the main control room to alert the operator to abnormal conditions.

7.7.2.8.2 Specific Regulatory Requirments Conformance

The subject system has no specific regulatory requirements imposed on it but the following observation is included:

(1) Regulatory Guide 1.56 (6/73)

The Reactor Water Cleanup (RWCU) system provides the recorded conductivity measurements and alarms of influents and effluents of the demineralizers and records of the flow rate through each demineralizer as recommended in the guide.

7.7.2.9 Transient Monitoring System Analysis

7.7.2.9.1 TMS Safety Related Functions

The TMS itself performs no safety function. However, TMS devices are connected in safety related circuits and must maintain the safety related circuit continuity, without disturbance to that circuit, under all conditions.

Where TMS signals are required from safety related circuits, isolation is provided between the safety circuit and the TMS signal by the use of a Validyne Engineering Corp. Remote Carrier Modulator, Model CM249.

7.7.2.9.1.1 TMS Safety Related System Isolation

The Validyne CM249 provides impedance isolation, using transformer coupling, between safety related circuits and TMS circuits. CM249 circuit arrangement provides isolation in compliance with IEEE 279-1971, Section 4.7.2. The CM249 unit has been seismically and environmentally qualified. See Wyle Labs Report (NDQ 783015 Rev. B). 14

A summary of the Validyne CM249 specifications is as follows:

Common Mode Isolation Voltage - 2000 V Peak
Input/Output Dielectric Strength - 2000 VDC, 220 VAC
Insulation Resistance - 10^{10} ohms
Input Impedance - 2 megohms

7.7.2.9.1.2 TMS Wiring Separation

All wiring for the TMS is installed permanently except wiring for piping thermal expansion measurements installed locally from a measuring device to a multiplexer. This thermal expansion wiring is not safety related and will be removed after completion of testing. Wiring from the multiplexers to STARTREC, which is coaxial cable, will be installed in a raceway system as any non-safety related cable. Permanent wiring for the TMS from safety and non-safety systems to the TMP will be as follows:

A. "Wiring required by transient test instrumentation within GE supplied panels is routed to the requirements of A61-4050 Electrical Equipment Separation for Safeguards System."

B. "Cables required by transient test instrumentation is routed
14 through the GE supplied PGCC panel modules in accordance with the requirements of NEDO 10466."

C. Safety related wiring and cables required for transient test instrumentation is run in compliance with criteria set forth in Subsection 3.12 of this FSAR.

7.7.2.10 Refueling Interlocks System-Instrumentation and Controls

7.7.2.10.1 General Functional Requirements Conformance

The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor is subcritical even when the highest worth control rod is fully withdrawn. Refueling procedures are written to avoid situations in which inadvertent criticality is possible. The combination of refueling interlocks for control rods and the refueling platform provides redundant methods of preventing inadvertent criticality even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

17 Table 7.7-2 illustrates the effectiveness of the refueling interlocks. This table considers various operational situations involving rod movement, hoist load conditions, refueling platform movement and position, and mode switch manipulation. The initial conditions in Situations 4 and 5 appear to contradict the action of refueling interlocks, because the initial conditions indicate that more than one control rod is withdrawn, yet the mode switch is in REFUEL. Such initial conditions are possible if the rods are withdrawn when the mode switch is in STARTUP, and then the mode switch is turned to REFUEL. In all cases, correct operation of the refueling interlock will prevent either the operation of loaded refueling equipment over the core when any control rod is withdrawn or the withdrawal of any control rod when fuel-loaded refueling equipment is operating over the core. In addition, when the mode switch is in REFUEL, only one rod can be withdrawn; selection of a second rod initiates a rod block.

7.7.2.10.2 Specific Regulatory Requirements Conformance

No specific regulatory requirements apply to refueling interlocks. The refueling interlocks are designed to be normally energized (fail safe) and single failure tolerant of equipment failures.

IEEE standards do not apply because the refueling interlocks are not required for any postulated design basis accident or for safe shutdown. The interlocks are required only for the refueling mode of plant operation. The requirements of 10 CFR 50 Appendix B are met in the manner set forth in Chapter 17.

There are no specific General Design Criteria requirements for this system.

7.7.2.11 Rod Block Monitor Subsystem

7.7.2.11.1 General Functional Requirement Conformance 17

Motion of a control rod causes the LPRMs adjacent to the control rod to respond strongly to the change in power in the region of the rod in motion. Figures 7.7-17 and 7.7-18 illustrate the calculated response of the two RBMs to the full withdrawal of a selected control rod from a region in which the design limits on power and flow exist.

Because MCPR cannot reach 1.0 until the control rod is withdrawn through greater than half its stroke, the highest rod block set point halts rod motion well before local fuel damage can occur. This is true even with the adjacent and nearest LPRM detector assemblies failed.

7.7.2.11.2 Specific Regulatory Requirement Conformance

IEEE Standards and Regulatory Guides do not apply to the Rod Block Monitor Subsystem because it is not a protection system.

10CFR50 Appendix A

Criteria 24

The RBM provides an interlocking function in the control rod withdrawal portion of the CRD reactor manual control system. This design is separated from the protective functions in the plant to assure their independence.

The RBM is designed to prevent inadvertent control rod withdrawal
17 given an imposed single failure within the RBM. One of the two
RBM channels is sufficient to provide an appropriate control rod
withdrawal block.

In addition, the RBM has been designed to meet "appropriate protection system criteria. . . .acceptable to the Regulatory Staff." (Reference 7.7-2)

7.7.2.12 Nuclear Pressure Relief System - Instrumentation and Controls

7.7.2.12.1 General Functional Requirements Conformance

The Nuclear Pressure Relief System is designed to provide the nuclear steam supply pressure relief function without jeopardy to the safety-related ADS function, discussed in Section 7.3.

7.7.2.12.2 Specific Regulatory Requirements

(1) 10CFR50 Appendix A - Criterion 14.

18 The Nuclear Pressure Relief System provides additional means for
minimizing the probability of abnormal reactor coolant pressure
boundary leakage.

(2) 10CFR50 Appendix A - Criterion 15.

The Nuclear Pressure Relief System is designed to afford adequate additional margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

(3) 10CFR50 Appendix A - Criterion 30.

The components of the Nuclear Pressure Relief System are designed, selected, fabricated, erected and tested to the highest, practical, current industrial standards. The System is designed with temperature sensors for each safety relief valve whereby leaks may be detected and identified in a timely fashion. 18

7.7.2.13 Loose Parts Monitoring System

The LPMS is not a safety-related system. It has been designed in accordance with Regulatory Guide 1.133, Rev. 1, Draft 2. 22

7.7.3 REFERENCES

7.7-1 Morgan, W. R., "In Core Neutron Monitoring System for General Electric Boiling Water Reactors," APED-5706, November 1968 (Rev. April 1969).

7.7-2 Hatch 1 Amendment 7, June 24, 1969, pp. 7-3.0-1 and 7-5.0-1. 17

TABLE 7.7-1

CRD HYDRAULIC SYSTEM PROCESS INDICATORS

Measured Variable	Instrument Type
Total system flow	Flow indicator
Drive water pump suction pressure	Annunciator
Drive water filter differential pressure	Annunciator
Cooling water header pressure	Pressure indicator
Charging water header pressure	Annunciator
Drive water flow rate	Flow indicator
Cooling water header	Flow indicator
Control rod drive temp	Annunciator
Control rod position (normal range)	Rod status display

TABLE 7.7-2

REFUELING INTERLOCK EFFECTIVENESS

SITUATION	REFUELING PLATFORM POSITION	REFUELING TMH*	PLATFORM FMH*	HOISTS FG*	SERVICE PLATFORM HOIST	CONTROL RODS	MODE SWITCH	ATTEMPT	RESULT	
1.	Not near core	UL*	UL*	UL*	UL*	All rods in	Refuel	Move refueling platform over core	No restrictions	
2.	Not near core	UL	UL	UL	UL	All rods in	Refuel	Withdraw rods	Cannot withdraw more than one rod	
3.	Not near core	UL	UL	UL	UL	One rod withdrawn	Refuel	Move refueling platform over core	No restrictions	
4.	Not near core	Any hoist loaded or FG not fully up			UL	One rod withdrawn	Refuel	Move refueling platform over core	Platform stopped before over core	
5.	Not near core	UL	UL	UL	UL	More than one rod withdrawn	Refuel	Move refueling platform over core	Platform stopped before over core	20
6.	Over core	UL	UL	UL	UL	All rods in	Refuel	Withdraw rods	Cannot withdraw more than one rod	
7.	Over core	Any hoist loaded or FG not fully up				All rods in	Refuel	Withdraw rods	Rod block	
8.	Not near core	UL	UL	UL	L*	All rods in	Refuel	Withdraw rods	Rod block	
9.	Not near core	UL	UL	UL	L	All rods in	Refuel	Operate service platform hoist	No restrictions	
10.	Not near core	UL	UL	UL	L	One rod withdrawn	Refuel	Operate service platform hoist	Hoist operation prevented	
11.	Not near core	UL	UL	UL	UL	All rods in	Startup	Move refueling platform over core	Platform stopped before over core	
12.	Not near core	UL	UL	UL	L	All rods in	Startup	Operate service platform hoist	No restrictions	
13.	Not near core	UL	UL	UL	L	One rod withdrawn	Startup	Operate service platform hoist	Hoist operation prevented	

TABLE 7.7-2 (Cont'd.)

REFUELING INTERLOCK EFFECTIVENESS

SITUATION	REFUELING PLATFORM POSITION	REFUELING TMH*	PLATFORM FMH*	HOISTS FG*	SERVICE PLATFORM HOIST	CONTROL RODS	MODE SWITCH	ATTEMPT	RESULT	
14.	Not near core	UL	UL	UL	L	All rods in	Startup	Withdraw rods	Rod block	
15.	Not near core	UL	UL	UL	UL	All rods in	Startup	Withdraw rods	No restrictions	20
16.	Over core	UL	UL	UL	UL	All rods in	Startup	Withdraw rods	Rod block	
17.	Any		Any condition		Any condition	Any condition, reactor not at power	Startup	Turn mode switch to RUN	Rod block	

*LEGEND

TMH - Trolley Mounted Hoist   FMH - Frame Mounted Hoist   FG - Fuel Grapple   UL - Unloaded   L - Fuel Loaded

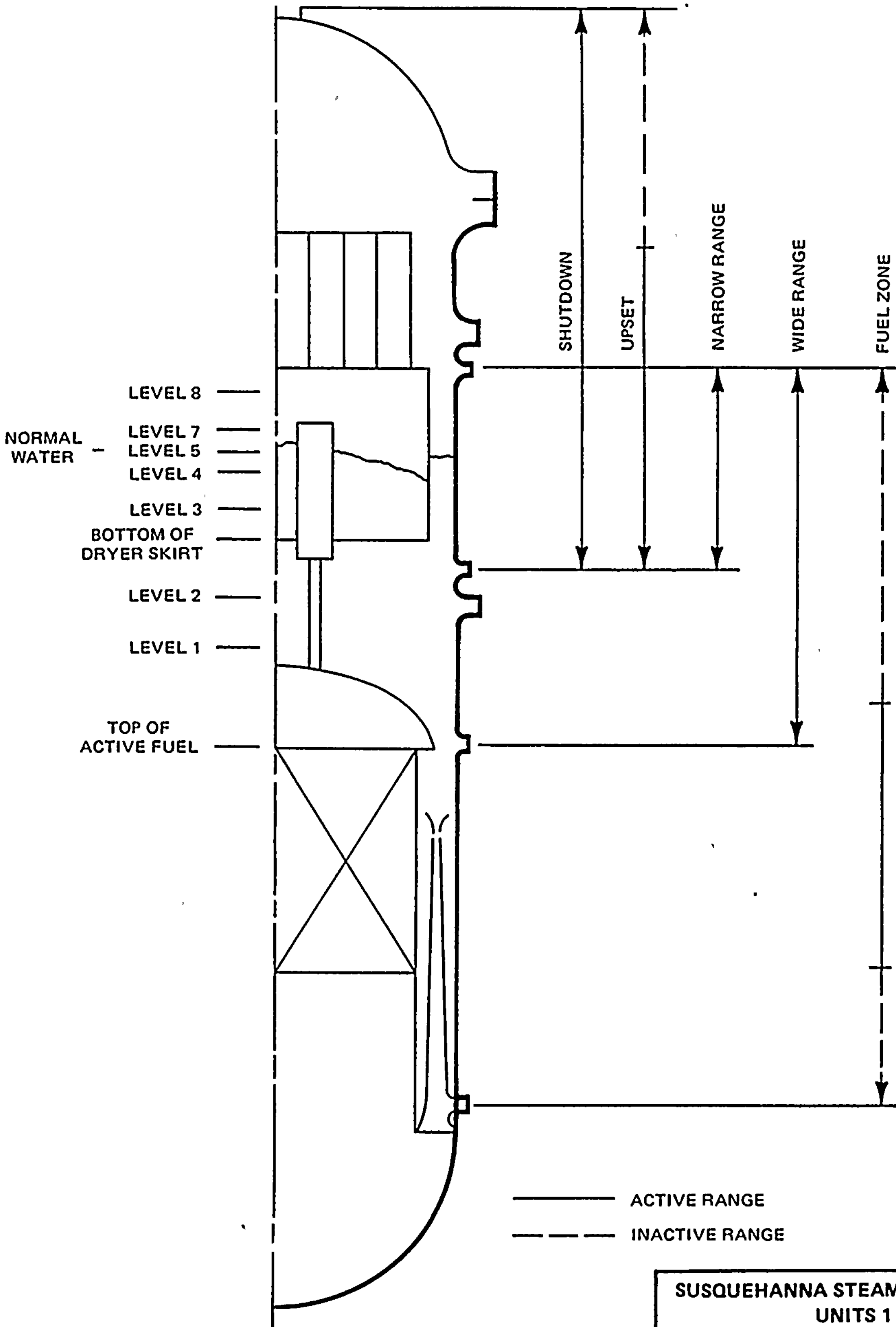
TABLE 7.7-3

RBM SYSTEM TRIPS

TRIP FUNCTION	NOMINAL SETPOINT	TRIP ACTION
RBM upscale (high)	R(.66 Flow + 41%) normal	Rod block, annunciator amber light display
	R(.66 Flow + 33%) intermediate	
	R(.66 Flow + 25%) low	
RBM inoperative	(See Note)	Rod block, annunciator, amber light display
RBM downscale	5/125 FS	Rod block, annunciator, white light display
RBM bypassed	Manual switch or Peripheral rod selected or APRM reference below 30%	White light display

Note:

RBM is operative if module interlock chain is broken, OPERATE-CALIBRATE switch is not in OPERATE position, less than 50% of available LPRM signals are above 3% threshold, or internal logic self-test circuits indicate trouble.



SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

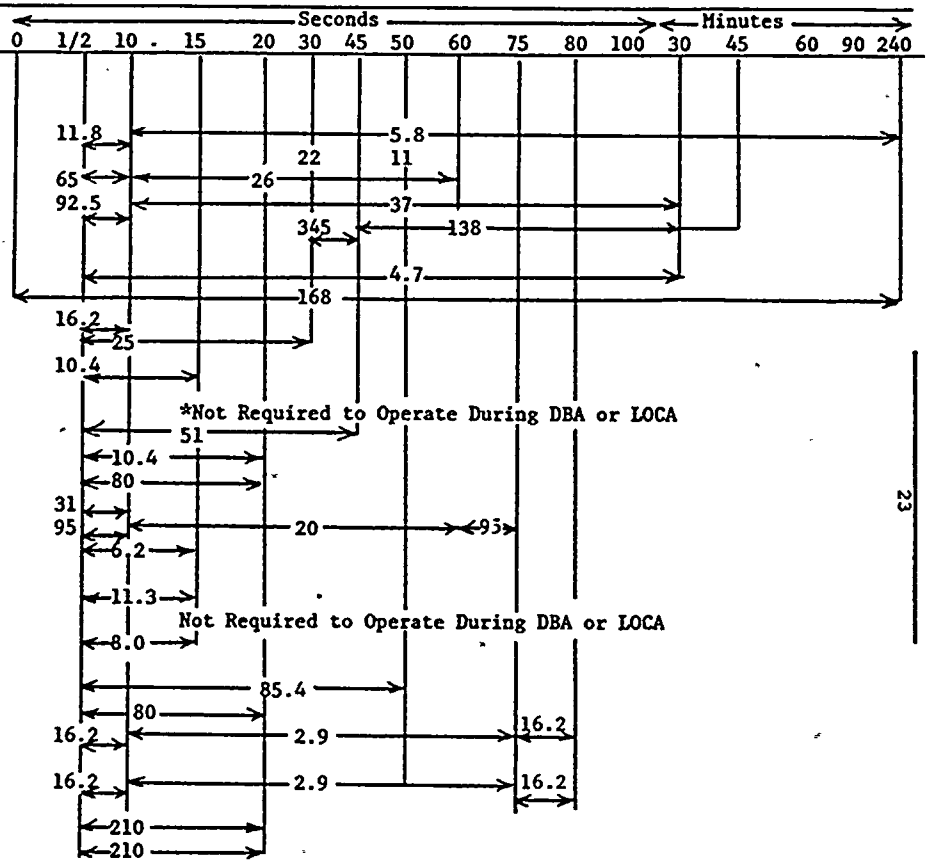
WATER LEVEL RANGE DEFINITION

FIGURE 7.7-1

TABLE 8.3-7 (Con't)

250 VDC Load Cycle (2)

Item	Load Description	Equipment No.	HP	Amperes Inrush	Amperes Full Load
	Division II				
1.	HPCI Barom Condenser Vacuum Pump	IP216	1.5	11.8	5.8
2.	HPCI Vacuum Tank Drain Pump	IP215	3.0	22	11
3.	HPCI Turbine Auxiliary Oil Pump	IP213	7.5	65	26
4.	RFPT "A" Emergency Lube Oil Pump	1P125A	10	92.5	37
5.	Turbine Gen Emerg Bearing Lube Oil Pump	1P112	40	345	138
6.	Reactor Recirc MG Set EBOP	1P155B	1	14.1	4.7
7.	Vital AC Power Supply	1D666		-	168
8.	Main Steam Line Drain Valve	HV-B21-1F019	0.54	16.2	2.9
9.	RWC Recirc Pump Suction Valve	HV-G33-1F004	1.1	25.0	4.0
10.	Supp Pool Water Filter PP Suction Valve	HV-15768	0.33	10.4	2.1
11.	RHR PP Shutdown Cooling Suct Valve	HV-E11-1F008	10.8	210	38
12.	RHR RPV Head Spray OTB2 Iso Valve	HV-E11-1F023	3.0	51	11.5
13.	RHR Radwaste Discharge Valve	HV-E41-1F049	0.36	10.4	2.1
14.	HPCI Test Byp to Cnds Stor Tk Valve	HV-E41-1F011	4.3	80	17
15.	HPCI Min Flow Byp to Supp Pool Valve	HV-E41-1F012	1.0	31	4.8
16.	HPCI Test Byp to Cnds Stor Tk Valve	HV-E41-1F008	5.8	95	20
17.	HPCI Turb Exh to Supp Pool Vac Bkr Valve	HV-E41-1F075	0.13	6.2	0.70
18.	RCIC Vac Relief System Iso Valve	HV-E51-1F084	0.33	11.3	2.0
19.	HPCI Turb Exh to Supp Pool Valve	HV-E41-1F066	0.72	19.5	3.2
20.	HPCI Barom Cond Cooling Water Sup Valve	HV-E41-1F059	0.16	8.0	1.1
21.	HPCI Steam Supply Line Iso Valve	HV-E41-1F003	2.9	85.4	11.1
22.	HPCI Steam Supply to Turbine Valve	HV-E41-1F001	4.3	80	17
23.	HPCI Pump Suction from CNDS Stor Tk Valve	HV-E41-1F004	0.54	16.2	2.9
24.	HPCI Pump Suction from Supp Pool Valve	HV-E41-1F042	0.54	16.2	2.9
25.	HPCI Pump Discharge Valve	HV-E41-1F006	10.9	210	40
26.	HPCI Pump Discharge Valve	HV-E41-1F007	10.9	210	40



Division II Total in Amps: 168 | 1313.7 | 1055 | 1019.1 | 428.7 | 770.7 | 501.7 | 416.3 | 465.3 | 396.9 | 364.5 | 322.8 | 184.8

(1)Unit 1 & 2 250 Vdc Loads are Similar

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9.2 WATER SYSTEMS

9.2.1 SERVICE WATER SYSTEM

9.2.1.1 Design Bases

The Service Water System (SWS) has no safety related function and is designed to remove heat from heat exchangers in the turbine, reactor, and radwaste buildings and to transfer this heat to the cooling towers where it is dissipated.

The SWS is designed to operate during normal plant operation and plant shutdown with offsite power available. The system will not operate on loss of offsite power concurrent with a LOCA.

9.2.1.2 System Description

The SWS is a single loop, which includes three 50 percent capacity, horizontal, centrifugal, single stage pumps, located in the circulating water pump house, operating in parallel (one pump is on standby status) to circulate cool side cooling tower water through the heat exchangers listed in Table 9.2-1 and to discharge it back to the tower by way of the circulating water piping. In most cases the service water flows through the heat exchangers' tubes. The system is shown schematically on Figure 9.2-1a and 9.2-1b.

The water source and heat sink for the SWS is the cooling tower, which dissipates approximately 0.18×10^9 Btu/hr of heat from the system. The system is designed for the total flow of 30,300 gpm at a normal operating pressure of 100 psig. The design pressure is 170 psig. Each of the two generating units is provided with a separate SWS and cooling tower, although the two systems are interconnected so that equipment common to both units can be supplied from either SWS.

The systems' heat exchangers are sized to operate with 95°F service water at the inlets. For accessible areas the pipe is carbon steel with a corrosion allowance of 0.1875 in., while for inaccessible areas 90/10 copper nickel piping is used.

The temperature of fluids cooled by the service water in the respective heat exchangers are regulated by either recirculation of the service water or flow control of the service water.

Recirculation

In this type of regulation the temperature of the fluid being cooled is controlled by adjusting the inlet temperature of the service water. This is done by recirculating some of the warm service water discharging from the respective heat exchanger back into the cool service water entering the heat exchanger. The amount of warm service water recirculated is controlled by a valve that is regulated by a temperature controller in the service water discharge from the heat exchanger.

This type of temperature regulation is used for the following:

a) Control structure chillers

b) Radwaste building chillers

c) Turbine building chillers

d) Reactor building chillers.

Flow Control

In this type of regulation the temperature of the fluid being cooled is regulated by adjusting the flow of service water through the respective heat exchanger. This is done by a control valve located in the service water discharge line from the heat exchanger, which is regulated by a temperature controller that senses the temperature of the cooled fluid.

This type of temperature regulation is used for the following:

a) Generator hydrogen coolers

b) Turbine Building Closed Cooling Water (TBCCW) heat exchangers

c) Reactor Building Closed Cooling Water (RBCCW) heat exchangers

d) Gaseous Radwaste Recombiner Closed Cooling Water (GRRCCW) heat exchangers

e) Main turbine lube oil coolers

f) Reactor feed pump turbine lube oil coolers

g) Alterrex air coolers

h) Reactor recirculation pump M-G set hydraulic fluid coolers

i) Pipe tunnel coolers

The balance of the heat exchangers as listed in Table 9.2-1 have the service water flow adjusted manually to obtain the required fluid temperature.

A back pressure regulator installed in the service water return header from the fuel pool heat exchangers maintains a positive pressure differential between the tube and shell sides of the heat exchangers to prevent possible radioactive contamination of the SWS.

In the case of loss of offsite power, the cooling of the RBCCW heat exchangers and TBCCW heat exchangers can be maually transfered from the SWS to the Emergency Service Water System (ESWS). However, since the heat exchangers are designed for non-essential service, the transfer valves are designed to close on failure of the solenoid valves which control them, ensuring no loss of emergency service water.

9.2.1.3 Safety Evaluation

The SWS operation has no safety related function and failure of the system will not compromise any safety related system or component or prevent a safe nuclear shutdown.

9.2.1.4 Tests and Inspections

The system is hydrostatically tested prior to startup and preoperationally tested in accordance with the requirements of Chapter 14. The standby pump will be tested and put into regular service periodically to ensure system integrity. Standby heat exchangers will be alternated into service on a regular basis.

9.2.1.5 Instrumentation Applications

The suction header of the service water pumps is provided with a pressure indicator and each pump has a pressure indicator on its discharge line. A temperature indicator is located on the common discharge header. The discharge header is monitored for low pressure. If either of the operating pumps fails, the standby pump will start automatically. Low seal water flow to a pump will trip that pump automatically and put it on standby.

Each heat exchanger in the system is provided with a pressure indicator in both the inlet and outlet lines. A temperature indicator is also provided in the outlet lines.

Manually operated throttling valves have been provided downstream of the heat exchangers to control service water flow where heat exchange rate is constant or a wide temperature range can be tolerated. Automatic temperature control valves have been provided where ever it is necessary to keep operating temperatures controlled within a narrow range.

9.2.2 REACTOR BUILDING CLOSED COOLING WATER SYSTEM

9.2.2.1 Design Basis

The Reactor Building Closed Cooling Water (RBCCW) System has no safety related function and is a closed loop system that transfers heat from miscellaneous reactor auxiliary plant equipment to the service water system through the heat exchangers. The plant equipment serviced by the RBCCW system is located in the Reactor and Radwaste Buildings.

The RBCCW system is required to operate during normal operation and on loss of off-site power without occurrence of a loss of coolant accident. In the event that the Reactor Building chillers are unavailable, for reasons other than those of offsite power, or mechanical failure, the RBCCW system is designed to automatically furnish cooling water to the Reactor Building Chilled Water System for drywell cooling. On loss of off-site power the drywell coolers can be manually switched to the RBCCW system.

9.2.2.2 System Description

The RBCCW system consists of two 100 percent capacity cooling water pumps, two 100 percent heat exchangers, one head tank, one chemical addition tank, associated valves, piping and controls as shown on Figure 9.2-2.

System containment penetrations and isolation valves are designed to Seismic Category I and ASME Code Section III, Class 2 requirements at a pressure of 150 psig at 500°F. The system piping located inside containment to and from the Reactor Recirculation Pump and Motor coolers is designed to ANSI B31.1 requirements at a pressure of 150 psig at 500°F. This piping is designed to withstand the SSE such that its failure or loss of function will not impair safety related systems located inside containment. The system piping which is located outside containment is designed to ANSI B31.1 requirements at a pressure of 125 psig at 350°F. All piping is carbon steel.

The RBCCW system provides cooling water to nonessential equipment located in the Reactor and Radwaste Buildings which has the potential to carry radioactive fluids or which require a clean water supply to minimize long term corrosion. The service water in the heat exchanger tube side is maintained by the service water pumps at a higher pressure than the closed loop system in the heat exchanger shell side. In the event of tube failure, the service water would leak into the closed loop system to preclude the possibility of radioactive release to the environment.

During normal operation, one cooling water pump and one heat exchanger are in service. The second pump is on automatic standby. A heat load of 18.4 x 10 Btu/hr is transferred from the closed cooling water system to the service water system in the heat exchanger. During normal plant operation, the RBCCW system furnishes cooling water to the following components:

The following equipment is located in the Reactor Building:

1) Cleanup Non-Regenerative Heat Exchanger

2) Cleanup Recirculation Pump Coolers

3) Reactor Recirculation Pump Seal and Motor Oil Coolers

4) Reactor Building Sump Cooler

5) Sample Station Chillers

6) Containment Instrument Gas Compressor Coolers

7) Drywell Equipment Drain Sump Cooler

The following equipment is located in the Radwaste Building:

1) Low Pressure Compressor and After Cooler

The water is circulated throughout the closed loop by the pump, which is rated at 1100 gpm at 90 ft head. The capacity of cooling water required by each plant component is set by a manual throttling valve on the cooling water outlet of each unit.

The closed loop cooling water temperature leaving the RBCCW heat exchanger is automatically controlled by a motor-operated flow control valve located on the service water side. Automatic control is carried out by a temperature indicating controller which maintains the closed cooling water outlet temperature at a constant 100°F. 19

Upon loss of off-site power without occurrence of a loss of coolant accident, the RBCCW heat exchangers can be manually switched from the service water system to the emergency service 6
water system. The RBCCW pumps start automatically, using standby

ac power furnished by the diesel generators in accordance with the loading sequence. One pump can be taken out of service by remote manual switching. The RBCCW system furnishes cooling water to the Reactor Recirculation pump seal water coolers and motor oil coolers. The drywell coolers (RB chilled water system)
6 can be manually valved in after the RBCCW heat exchanger is manually switched from SW to ESW. A total heat load of 7.3 x 10 BTU/hr. would be transferred from the closed cooling water system to the emergency service water system at this time.

The remainder of the RBCCW system receives a reduced amount of cooling water; therefore, no appreciable heat load is transferred from the other RBCCW users. These users can be isolated manually from the system when required.

During loss of off-site power or loss of both Reactor Building chillers, the cleanup non-regenerative heat exchanger is automatically isolated from the RBCCW system.

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During certain plant operating conditions, such as startup, excess water is normally removed from the reactor by blowdown through the reactor water cleanup system non-regenerative heat exchanger. During blowdown, the heat rejected to the RBCCW system is 25.19 x 10 Btu/hr. At this condition, the second RBCCW heat exchanger is manually put into service to handle this additional, transient heat duty.

The head tank, which is located at the highest point in the system, accommodates thermal expansion and provides ample net postive suction head (NPSH) to the cooling water pumps. The head tank, which has a capacity of 800 gallons, also provides necessary makeup water as required.

The RBCCW supply and makeup is furnished from the demineralized water system. When required, chemicals are added to the system through the chemical addition tank (15 gal. capacity) to maintain a concentration of 500 ppm of chromates for corrosion prevention.

The RBCCW system pumps, heat exchangers, chemical addition tank, and head tank are all located in the Reactor Building.

9.2.2.3 Safety Evaluation

The RBCCW has no safety-related function. Failure of the system will not compromise any safety-related system or component or prevent a safe shutdown of the plant.

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The RBCCW system is not required to operate after a loss-of-coolant accident. The containment isolation valves will close automatically under this condition. Also, the RBCCW pumps receive a loss-of-coolant accident trip signal which prevents pump operation.

9.2.2.4 Testing and Inspection Requirements

The RBCCW system is hydrostatically tested prior to operation. The motor-operated containment penetration valves can be manually closed by the operator in the control room. These valves will be tested to assure that they are capable of opening or closing by operating the manual switches and observing the position lights in the control room.

Test connections are located inside containment to test and verify the leak tightness of the containment penetration isolation valves prior to operation.

The RBCCW system pumps, heat exchangers, head tank, chemical addition tank and piping (to the extent practicable) are located in the Reactor Building to permit periodic inspection during normal operation.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.2.2.5 Instrumentation Requirements

The flow rate of cooling water to all coolers is regulated manually by individual throttling valves on the cooling water outlet from each unit. Flow elements are provided for the coolers located inside the primary containment for initial flow balancing of these components. A temperature indicator is provided on the RBCCW system header outside containment to verify satisfactory cooling of components inside containment. Temperature indicators are provided at the outlet of each cooler located outside primary containment. Test points are furnished across all coolers in the system for pressure measurement.

Continuous radiation monitors are installed in the pump suction header of the RBCCW system. This instrumentation indicates, records, and alarms in the main control room any radioactive leakage into the RBCCW system.

High and low level switches on the RBCCW head tank detect leakage into or out of the system. Switch operation actuates an alarm in the control room. The RBCCW heat exchanger outlet temperature and pressure are monitored. These signals alarm conditions of system high temperature and/or low pressure in the control room.

A low pressure switch is provided on the cooling water pumps discharge header to automatically start the standby pump in the event the system pressure drops below a preset value. The switch also actuates an alarm in the control room.

9.2.3 TURBINE BUILDING CLOSED COOLING WATER SYSTEM

9.2.3.1 Design Basis

The Turbine Building Closed Cooling Water (TBCCW) System has no safety related function and is a closed loop cooling system that transfers heat from miscellaneous turbine plant components to the service water system through the TBCCW heat exchangers.

The TBCCW system is required to operate during normal plant operation. If needed, the system can also operate upon loss of offsite power by remote manual control.

9.2.3.2 System Description

The TBCCW system consists of two 100-percent capacity cooling water pumps, two 100-percent heat exchangers, one head tank, one chemical addition tank, associated valves, piping and controls as shown on Figure 9.2-3. The system is designed to ANSI B31.1 requirements. The system piping is carbon steel designed for a primary rating of 125 psig at 350° F.

The TBCCW system furnishes cooling water to the following turbine plant components:

1) Control Rod Drive Pump Bearing and Oil Coolers

2) Condensate Pump Motor Upper and Lower Bearing Coolers

3) Instrument Air Compressor Coolers

4) Service Air Compressor Coolers

5) EHC Fluid Coolers

6) Sample Station Chillers

7) Auxiliary Boiler Sample Cooler

During normal plant operation, one cooling water pump and one heat exchanger are in service. The second pump is on automatic standby. A heat load of 1.50×10^6 Btu/hr is transferred from

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closed cooling water system to the service water system in the heat exchanger. The water is circulated throughout the closed loop by the pump which is rated at 325 gpm at 130 ft of head. The capacity of cooling water required by each plant component is set by manual throttling valves located on the cooling water outlet of each unit.

The closed loop cooling water temperature leaving the TBCCW heat exchanger is automatically controlled by an air operated flow control valve located on the service water side. Automatic control is carried out by a temperature indicating controller which maintains the closed cooling water outlet temperature at a constant 105° F.

After a loss of offsite power, the pumps can be started by remote manual switching to provide cooling water to the control rod drive pump bearing and oil coolers if required. The TBCCW heat exchangers tube side flow is transferred from the service water system to the emergency service water system by remote switching. A heat load of 0.04×10^6 Btu/hr is rejected to the emergency service water at this time. TBCCW system operation is not required during a loss-of-coolant accident.

The head tank, which is located at the highest point in the system, accommodates thermal expansion and provides ample net positive suction head (NPSH) to the cooling water pumps. The head tank, which has a capacity of 400 gallons, also provides necessary makeup water as required.

The TBCCW supply and makeup is furnished from the demineralized water system. When required, chemicals are added to the system through the chemical addition tank (15 gal. capacity) to maintain a concentration of at least 500 ppm of chromates for corrosion prevention.

The TBCCW system pumps, heat exchangers, chemical addition tank, and head tank are all located in the Turbine Building.

9.2.3.3 Safety Evaluation

Since the TBCCW system has no safety-related function, failure of the system will not compromise any safety-related system or component or prevent a safe shutdown of the plant.

9.2.3.4 Testing and Inspection Requirements

The TBCCW system is hydrostatically tested prior to operation. All portions of the system are accessible for visual examination and inspection during normal operation.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.2.3.5 Instrumentation Requirements

The flow rate of cooling water to all coolers is regulated manually by individual throttling valves on the cooling water outlet from each unit. Temperature indicators are provided at the outlet of each cooler. Test points are furnished across each cooler to measure pressure.

High and low level switches on the TBCCW head tank detect leakage into or out of the system. Switch operation actuates an alarm in the control room. The TBCCW heat exchanger outlet temperature and pressure are monitored. These signals alarm conditions of system high temperature and/or low pressure in the control room.

A low pressure switch is provided on the cooling water pumps discharge header to automatically start the standby pump in the event the system pressure drops below a preset value. The switch also actuates an alarm in the control room.

9.2.4 GASEOUS RADWASTE RECOMBINER CLOSED COOLING WATER SYSTEM

9.2.4.1 Design Basis

The Gaseous Radwaste Recombiner Closed Cooling Water (GRRCCW) System has no safety related function and is a closed loop cooling system that transfers heat from the gaseous radwaste recombiner condensers to the service water system through the GRRCCW heat exchangers.

The GRRCCW system is required to operate only during normal plant operation.

9.2.4.2 System Description

A separate GRRCCW system is provided for each of the three recombiner trains. Each closed cooling water system consists of one cooling water pump, one heat exchanger, one head tank, one chemical addition tank, associated valves, piping and controls as shown on Figure 9.2-4. The system is designed to ANSI B31.1 requirements. The system piping is carbon steel designed for a primary rating of 125 psig at 350°F.

The GRRCCW system furnishes cooling water to only its respective gaseous radwaste recombiner condensers.

During normal operation when a recombiner train is in operation, the heat transferred from its respective GRRCCW system to the service water system is 21 x 10^6 Btu/hr in the heat exchanger. At this time the single cooling water pump and heat exchanger are in operation. The cooling water pump is rated at 1450 gpm at a head of 100 ft. Since the recombiner condensers'are the only components on the GRRCCW system, no throttling valves are required for flow regulation.

The closed cooling water temperature leaving the GRRCCW heat exchanger is automatically controlled by a flow control valve located on the service water side. Automatic control is carried out by a temperature indicating controller which maintains the closed cooling water temperature at a constant 105°F. When a recombiner train is in operation, the closed cooling water pump, which is started by manual initiation in the control room, circulates the cooling water throughout the GRRCCW system.

The head tank, which is located at the highest point in the system, accommodates thermal expansion and provides ample net positive suction head (NPSH) to the cooling water pump. The head tank, which has a nominal capacity of 400 gallons, also provides necessary makeup water as required. 23

The GRRCCW supply and makeup is furnished from the demineralized water system. When required, chemicals are added to the system through the chemical addition tank (15 gallons capacity) to maintain a concentration of 500 ppm of chromates for corrosion prevention.

The GRRCCW system pump, heat exchanger, head tank and chemical addition tank are all located in the Turbine Building.

9.2.4.3 Safety Evaluation

Failure of the GRRCCW system will not compromise any safety-related system or component or prevent a safe shutdown of the plant.

9.2.4.4 Testing and Inspection Requirements

The GRRCCW system is hydrostatically tested prior to operation. All portions of the system are accessible for visual examination and inspection during normal operation.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.2.4.5 Instrumentation Requirements

A temperature indicator is provided at the outlet of the recombiner train. Test points are furnished across the recombiner train for pressure measurement. A flow element has been included in the system piping for flow determination.

High and low level switches on the GRRCCW head tank detect leakage into or out of the system. Switch operation actuates an alarm in the control room. The GRRCCW heat exchanger outlet temperature and pressure are monitored. These signals alarm in the control room conditions of system high temperature and/or low pressure.

The closed cooling water pump operation is controlled by a handswitch located in the control room. A low pressure switch located on the pump discharge header signals an alarm in the control room if system pressure falls below a preset value.

9.2.5 EMERGENCY SERVICE WATER SYSTEM

9.2.5.1 Design Bases

The Emergency Service Water System (ESWS) has a safety related function and is an engineered safeguard system designed to supply cooling water to the emergency diesel generator units, RHR pumps, and to those room coolers required during normal and emergency conditions necessary to safely shut down the plant.

The ESWS is designed to take water from the spray pond (the ultimate heat sink), pump it to the various heat exchangers and return it to the spray pond by way of a network of sprays that dissipate the heat to the atmosphere.

The ESWS is required to supply cooling water to:

a) The RHR pump room unit cooler, the bearing oil coolers, and seal coolers of each RHR pump during all modes of operation of the RHR system.

b) All the heat exchangers associated with the diesel generators during emergency operation and test modes.

c) The room coolers for the core spray (CS) pumps, the high pressure coolant injection (HPCI) pumps, and reactor core isolation cooling (RCIC) pumps during the operation of these systems.

d) The control structure chiller, emergency switchgear and load center cooler, reactor building closed cooling water (RBCCW) heat exchangers, and the turbine building closed cooling water heat exchanger (TBCCW) during emergency operation.

e) Makeup to the fuel pools.

The ESWS starts automatically within 60 sec after the diesel generators receive their start initiation signal, but it can also be started manually from either the main control room or from either of the two remote shutdown panels.

The ESWS is designed to operate during any of the following conditions:

a) Loss of offsite power

b) The operational basis earthquake (OBE)

c) Design high and design low level spray pond conditions.

It is also designed to remain functional following the design Safe Shutdown Earthquake (SSE).

The ESWS has sufficient redundancy so that a single failure of any active component, assuming the loss of offsite power, cannot impair the capability of the system to perform its safety related functions.

The system is designed so that the emergency service water is at a higher pressure than each of the fluids being cooled. This avoids the possibility of any radioactive leakage into the system.

The ESWS is provided with a pipe break detection system to monitor both loss of cooling water and high water level in rooms containing equipment using ESW. Any difference between inlet and outlet flows will be recorded and annunciated. Flooding detectors will annunciate high water level in the rooms.

The ESWS will operate under the conditions set by the design basis accident (DBA) for no less than 30 days with no water makeup to the spray pond. Under these conditions the pond's depth will always be greater than the minimum submergence of 7 ft required by the pumps (see Subsection 2.4.11.5).

Active components of the ESWS can be inspected and tested during plant power generation.

The system is designed for the 40 year life of the plant.

9.2.5.2 System Description

The ESWS is shown schematically in Figures 9.2-5a and 9.2-5b. The system consists of two loops each of which is designed to supply 100 percent of the ESW requirements to both units and the common emergency diesel generators simultaneously. Plant shutdown flow rates are listed in Table 9.2-2. Each loop has two 50 percent capacity, vertical, turbine type, single stage pumps operating at 1780 rpm and rated at 6000 gpm each. These are located in the engineered safeguard service water pumphouse which is built at the edge of the spray pond. Description of the pumphouse is found in Subsection 3.8.4.

The emergency service water flows through the tube side of all heat exchangers, except for the RHR pump seal coolers in which it flows through the shell side. All heat exchangers, except those in the RHR pump motor coolers, and the RHR pump seal water coolers have 90/10 Cu-Ni tubes. The RHR pump motor oil coolers and seal water coolers have type K copper tubes and type 304 stainless tubes respectively.

The supply and return piping is made of carbon steel with a 1/4 in. corrosion allowance. All piping outside of the pumphouse, main plant, and spray pond is buried and it is coated and wrapped for corrosion protection. The buried pipe is predominantly 36 in. diameter, which can be entered and visually examined. Manways with removable blind flanges are provided to allow periodic inspection of the inside of the pipe and also of sample coupons that are installed to provide an indication of the corrosion rate of the carbon steel pipe.

In-service inspection will be in accordance with ASME B&PV Code, Section XI for Section III, Class 3 components. The piping is designed, fabricated, inspected, and tested in accordance with requirements of ASME B&PV Code, Section III, Class 3. The spray pond piping network and ESSW pumphouse are described in Subsections 9.2.7 and 3.8.4.

During normal power generation the ESWS is not operating but is available for shutdown or emergencies. The system is initiated automatically once the emergency diesel generators have started (see Subsection 9.2.5.1).

Each of the two ESWS loops supplies separate equipment in each unit, except in the case of the common emergency diesel generators. This arrangement provides the necessary cooling capacity required by both units while maintaining the redundancy of active components and loops. The emergency diesel generator heat exchangers are connected to both ESWS loops and they can be supplied by either. Motorized valves are installed in each loop so that these heat exchangers can be isolated from a failed loop and still be supplied with cooling water from the other loop. The ESWS return headers to the spray pond are combined with the RHRSWS return headers before discharging into the pond. Loss of one RHRSWS/ESWS loop does not affect the capability of the second loop to safely shut down either or both units during emergency conditions.

Motors of the four ESWS pumps are connected to each of the four diesel generator buses which serve as backup in the case of loss of offsite power. When loss of offsite power occurs, the diesel generators start automatically and these provide emergency power for the pumps and motor operated valves. This transfer from the offsite power source to the standby power supply is automatic.

9.2.5.3 Safety Evaluation

The ESW system, with the exception of the buried piping and the piping in the spray pond, is housed within either the reactor building or the ESSW pumphouse, both of which are Seismic Category I. Tornado protection is discussed in Section 3.3. Flood design is discussed in Section 3.4. Missile protection is discussed in Section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in Section 3.6. Environmental design considerations are discussed in Section 3.11.

Each unit has two 100 percent capacity independent ESWS loops to supply cooling water for plant shutdown. This arrangement ensures that the full heat removal capacity required is available after the postulated active failure of a single component.

Each loop is isolated from the other by barriers, separate trenches, or distance to ensure that simultaneous loss of both loops cannot occur.

Failure of either a motor operated valve, a diesel generator, or pump will not prevent the system from removing the full heat capacity.

Upon loss of power, all safety related components (pumps, valves, and instruments) of this system will automatically be switched to the standby power supply (See Section 8.3).

Except for the Reactor Building Closed Cooling Water (RBCCW) heat exchanger and the Turbine Building Closed Cooling Water (TBCCW) heat exchanger, the entire ESWS including structures, pumps, motors, piping, valves, heat exchangers, and essential instruments are designed in accordance with Seismic Category I requirements. The RBCCW and TBCCW heat exchangers are not Seismic Category I since these are non-essential services.

Since the RBCCW and TBCCW heat exchangers are manually connected to the ESW after loss of off-site power, a failure of the non-safety related piping followed by a single failure in the safety related emergency service water system will not preclude one of the ESW loops from performing its safety function. If the single failure is in the normally closed, fail closed isolation valve, the remaining loop would not be affected.

Operators will verify the integrity of the non-essential piping prior to valving it onto the ESW system

Leak detection in the ESWS is monitored by seismically analyzed flood detectors located in the ESSW structure, in the RBCCW and TBCCW heat exchanger rooms and in each standby diesel generator room. These detectors alarm in the main control room. In addition flow elements measure the quantity of water being discharged by the ESWS pumps and this flow is compared with the flow discharging into the spray pond as measured by other flow elements. Any abnormal discrepancy between these flows is alarmed in the main control room. Flooding effects are discussed in Section 3.4.

The ESWS is designed to include the capability for testing through the full operational sequence that brings the system into operation for reactor shutdown and for LOCA, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

The ESWS pumps, piping, and heat exchangers are sized to provide the flow and cooling capacities required by the various RHR pump and motor coolers during any mode of RHR operation. Tables 9.2-3, 9.2-4 and 9.2-5 show the users and cooling duties on the ESW cooling cycle.

Table 9.2-3 lists all users; Tables 9.2-4 and 9.2-5 relate users to time for two types of shutdown.

Tables 9.2-4 and 9.2-5 are based on one of the four diesels being taken out of operation and placed on standby status after 24 hours of operation. The cooling loads are carried out to 30 days after the shutdown initiation since this is the design life of the ultimate heat sink for operation without make-up water. The operation of all equipment listed at the cooling duty shown represents design conditions. Under actual operating conditions certain pieces of equipment may be shutdown or operated under reduced loads.

9.2.5.4 Tests and Inspections

The ESWS will be hydrotested in accordance with ASME Section III. Pipe welds are subjected to heat treatment, testing, and inspection according to ASME Section III and the material specification.

The system components will be preoperationally tested in accordance with the requirements of Chapter 14.

The ESWS will be tested during normal plant operations in accordance with the requirements of Chapter 16.

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9.2.5.5 Instrumentation Applications

Logics and instrumentation are discussed in Subsection 7.3.1.1b and the displays are discussed in Section 7.5. A complete list of the system's process instrumentation is provided in Table 7.5-1.

The ESWS pumps are designed for remote operation from the control room. One loop can be remotely operated from either of the two remote shutdown panels. Each loop has been provided with a pump discharge pressure transmitter, the indicator for which is in the control room, and each pump chamber is provided with a low level submergence switch which alarms in the control room.

9.2.6 RHR SERVICE WATER SYSTEM

9.2.6.1 Design Bases

The Residual Heat Removal Service Water System (RHRSWS) has a safety related function and is an engineered safeguard system designed to supply cooling water to the residual heat removal (RHR) heat exchangers of both units.

The RHRSWS is designed to take water from the spray pond (the ultimate heat sink), pump it through the RHR heat exchanger, and return it to the spray pond by way of a spray network that dissipates the heat to the atmosphere.

The RHRSWS is designed to provide a reliable source of cooling water for all operating modes of the RHR system including heat removal under post-accident conditions, and also to provide water to flood the reactor core or the primary containment after an accident, should it be necessary.

The RHRSWS is designed to operate under any of the following conditions:

a) Loss of offsite power

b) Design high and design low level spray pond conditions

c) A safe shutdown earthquake (SSE).

The RHRSWS is designed with sufficient capacity and redundancy so that a single failure of any active component, assuming the loss of offsite power, cannot impair the capability of the system to perform its safety related functions.

A radiation monitor is provided in the RHRSW service water discharge piping from each RHR heat exchanger to alarm in the event of high activity level.

The RHRSWS is provided with a pipe break detection system to monitor both loss of cooling water and high water level in rooms containing equipment using RHRSW. Any difference between inlet and outlet flows will be recorded and annunciated. Flooding detectors will annunciate high water level in the rooms.

The RHRSWS is designed with the capability to operate under the conditions set by the design basis accident (DBA) for no less than 30 days without water makeup to the spray pond. The pond's depth will always be greater than the minimum submergence depth of 5 ft required by the pumps. (See Subsection 2.4.11.5)

Active components of the RHRSWS can be inspected and tested during plant power generation.

The system is designed for the 40 year life of the plant.

9.2.6.2 System Description

The RHRSWS is shown schematically in Figure 9.2-6. The system consists of two RHRSWS loops per unit supplying cooling water to each RHR heat exchanger, and each loop is cross-connected between Unit 1 and Unit 2 so that it can supply cooling water to either RHR heat exchanger. Each loop has two 100 percent capacity (for one unit) vertical, turbine type, two stage pumps operating at 1180 rpm, and a rated capacity of 9000 gpm. These are located in the ESSW pumphouse located at the edge of the spray pond. Description of the pumphouse is found in Subsection 3.8.4. The RHR heat exchangers are described in detail in the RHR system.

The RHR service water flows through the tube side of the RHR heat exchangers, the tubes of which are made of corrosion resistant 90-10 Cu-Ni in accordance with ASME Section II, SB-111. To reduce corrosion in the water boxes, demineralized water will be pumped into the heater channels to replace the spray pond water when the RHRSW is not in operation.

The supply and return piping is made of carbon steel with a 1/4 in. corrosion allowance. All piping outside of the pumphouse, main plant, and spray pond is buried and it is coated and wrapped for corrosion protection. The buried pipe is predominantly 36 in. diameter, which can be entered. Manways with removable blind flanges are provided to allow periodic inspection of the inside of the pipe and also of sample coupons that are installed to provide an indication of the corrosion rate of the carbon steel pipe. In-service inspection will be in accordance with ASME B&PV Code, Section XI for Section III, Class 3 components.

The piping is designed, fabricated, inspected, and tested in accordance with requirements of ASME B&PV Code, Section III, Class 3. The spray pond piping network and ESSW pumphouse are described in Subsections 9.2.7 and 3.8.4, respectively.

During normal power generation the RHRSWS is not operating, but is available for normal shutdown or emergencies.

When under emergency conditions, the RHRSW pump motors obtain their power from the standby power supply. The pumps are started manually 10 min after the diesel generators start. Waiting 10 min allows sequential loading of the diesel generators so that they will not be overloaded.

The buried pipe runs of each of the two RHRSWS loops are shared by both units and this provides the necessary capacity required for both units while maintaining the redundancy of active components and loops. Both the cooling water discharging from the RHR heat exchanger and the cooling water headers to the spray pond discharging from the corresponding ESW system are returned to the spray pond in a common header. Loss of one RHRSWS/ESWS loop does not affect the capability of the second loop to safely shut down either or both units during emergency conditions.

Motors of the four RHRSWS pumps are connected to each of the four diesel generator buses that serve as backup in the case of loss of offsite power. When loss of offsite power occurs, the diesel generators start automatically, providing emergency power for the pumps and motor operated valves. This transfer from the offsite power source to the standby power supply is automatic. Although the transfer from offsite power to standby power supply is automatic, the pumps themselves have to be started manually.

To prevent freezing, there is provision for draining the piping in the spray pond.

9.2.6.3 Safety Evaluation

The RHRSW system, with the exception of the buried piping and the piping in the spray pond, is housed within either the reactor building or ESSW pumphouse, both of which are Seismic Category I. Tornado protection is discussed in Section 3.3. Flood design is discussed in Section 3.4. Missile protection is discussed in Section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in Section 3.6. Environmental design considerations are discussed in Section 3.11.

Each generating unit has two independent RHRSWS loops, one for each RHR heat exchanger, to supply cooling water for plant shutdown. This arrangement ensures that the full heat removal capacity required is available after the postulated active failure of a single component.

Each loop is isolated from the other by barriers, separate trenches, or distance to ensure that simultaneous loss of both loops cannot occur.

Failure of either a motor operated valve, a diesel generator, or RHRSW pump will not prevent the system from removing the full heat capacity.

The entire RHRSWS including structures, pumps, motors, piping, valves, heat exchangers, and essential instruments are designed in accordance with Seismic Category I requirements.

The RHRSWS is designed to include the capability for testing through the full operational sequence that brings the system into operation for reactor shutdown and for LOCA, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

The RHRSWS pumps, piping, and heat exchangers are sized to provide the flow and cooling capacities required by the RHR system during any mode of its operation. Tables 9.2-3 thru 9.2-5 show the users and cooling duties on the ESW cooling cycle.

Table 9.2-3 lists all users; Table 9.2-4 and 9.2-5 relate users to time for two types of shutdown.

Table 9.2-4 and 9.2-5 are based on one of the four diesels being taken out of operation and placed on standby status after 24 hours of operation. The cooling load Tables 9.2-4 and 9.2-5 are carried out to 30 days after the shutdown initiation. Thirty days is the design life of the ultimate heat sink (spray pond) for operation without make-up water. The operation of all equipment listed at the cooling duty shown represents design conditions. Under actual operating conditions certain pieces of equipment may be shutdown or operated under reduced loads.

9.2.6.4 Tests and Inspections

The RHRSWS will be hydrotested in accordance with ASME Section III. Pipe welds are subjected to heat treatment, testing, and inspection according to ASME Section III and the material specification.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

The RHRSWS will be tested during norm plant operations in accordance with the requirements of Chapter 16.

9.2.6.5 Instrumentation Applications

Logics and instrumentation are discussed in Subsection 7.3.1.1b and the displays are discussed in Section 7.5. A complete list of the system's process instrumentation is provided in Table 7.5-1.

The RHRSWS pumps are designed for remote operation from the control room. One loop from each unit can be remotely operated from either of the two remote shutdown panels. Each pump has been provided with a discharge pressure transmitter, the indicator for which is in the control room. Each pump chamber is provided with a low level submergence switch that alarms in the control room.

The main water supply line to each heat exchanger is instrumented with control room mounted flow indication, low flow alarm, high temperature alarm, and low pressure alarm. Each heat exchanger has control room operated isolation valves on the inlet and outlet, which remain closed until the system is operated or tested.

Double remotely operated isolation valves are provided on the cross-tie lines between the RHRSW system and the RHR pump discharge for flooding the containment if such action is necessary and no other source of water is available.

9.2.7 ULTIMATE HEAT SINK

The ultimate heat sink has safety related functions and provides cooling water for use in the Engineered Safeguard Service Water system, described in Subsections 9.2.5 and 9.2.6, during ESSW testing, normal shutdown, and accident conditions.

9.2.7.1 Design Bases

The ultimate heat sink is capable of providing sufficient cooling for at least 30 days to (a) permit simultaneous safe shutdown and cooldown of both nuclear reactor units and maintain them in a safe shutdown condition, or (b) mitigate the effects of an accident in one unit, permit safe control and cooldown of the other unit, and maintain it in a safe shutdown condition. Continued cooling beyond 30 days is ensured by use of the makeup pumps to keep the pond at normal water level. The makeup pumps are designed to operate below the historic minimum water level of the Susquehanna River and can be used, in an emergency, down to any river flow at which they can physically draw water. In the event that makeup water from the makeup pumps is not available, additional provisions will be made in the 30 days available to assure continued cooling of the emergency equipment beyond 30 days. These provisions include but are not limited to; re-establishing makeup pump flow to the spray pond, emptying the cooling tower basins into the spray pond, trucking in water from neighboring water sources (such as the Susquehanna River), and providing temporary pumps and/or lines to pump water from neighboring water sources (such as the Susquehanna River, on site storage tanks, well water, etc.). This is in compliance with NRC Regulatory Guide 1.27 Rev. 2 as discussed in Section 3.13.

The ultimate heat sink is also capable of providing enough cooling water without makeup, for a design basis LOCA in one unit with the simultaneous shutdown of the other unit, for 30 days while assuming a concurrent SSE, single failure, and loss of offsite power. This event is evaluated in Subsection 9.2.7.3.1.

The ultimate heat sink consists of at least one highly reliable water source with a capability to perform the safety function required above during and after any one of the following postulated design basis events:

a) The most severe natural phenomena, including the safe shutdown earthquake, tornado, flood, or drought taken individually

b) Nonconcurrent site related events including loss of offsite power, transportation accidents, or oil spills and fires

c) Reasonably probable combinations of less severe natural phenomena and/or site related events

d) Any credible single mechanistic failure of a man-made structure or component.

Codes and standards applicable to the ultimate heat sink are listed in Table 3.2-1.

9.2.7.2 System Description

9.2.7.2.1 General Description

The ultimate heat sink for both units consists of the Susquehanna River and one Seismic Category I spray pond. These water sources ensure that a reliable source of cooling water is available, for shutdown and cooldown of the reactor, and for mitigation of accident conditions. Pertinent design data for ultimate heat sink components is given in Tables 9.2-6 and 9.2-7. 2

The spray pond is initially filled from the Susquehanna River. Pond level is maintained under normal conditions by rainfall on the pond surface (46 inches per year average) and by a small continuous flow of makeup water from the Susquehanna River. The average rainfall will generally exceed the average evaporation from the pond by 2 million gallons per year. This excess rainwater will tend to decrease the concentration of total dissolved solids (TDS) in the pond. This decrease is offset by a 2

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The spray pond is initially filled from the Susquehanna River. Pond level is maintained under normal conditions by rainfall on the pond surface (46 inches per year average) and by a small continuous flow of makeup water from the Susquehanna River. The average rainfall will generally exceed the average evaporation from the pond by 2 million gallons per year. This excess rainwater will tend to decrease the concentration of total dissolved solids (TDS) in the pond. This decrease is offset by a small amount of hot circulating water (less than 100 gpm) which may be used to de-ice the screens at the pump suctions. Thus the concentration of dissolved solids is maintained near that of the pond's initial fill water by a balance of rainwater, makeup from the river, evaporation and de-icing water.

During an emergency, accompanied by the loss of makeup pumps, approximately 40% of the pond water is lost by evaporation over 30 days. This approximately triples the concentration of dissolved solids in the pond. Drift and other losses can increase the total losses to approximately 80% of the initial volume, however, this does not increase the pond concentration of dissolved solids. Assuming the highest initial pond water dissolved solids concentration (See Table 3.3-2 in the Environmental Report Operating License Stage (EROLS)), the pond water can be concentrated to 55% of its initial volume before scaling would occur at 140°F. The emergency condition can persist for over two weeks before a small amount of acid and a proprietary organo-phosphate or equivalent is added to inhibit scale formation. The composition of the pond water is monitored periodically by grab samples.

The PSAR described the flow of cooling tower blowdown through the spray pond and then to the river. This original routing was selected for environmental reasons. The EROLS section 10.10 now states that the cooling tower blowdown bypasses the pond and flows directly to the river. This revised routing has been selected based upon a re-assessment of environmental considerations of flow through the pond. The direct route to the river also eliminates many of the temperature and chemical problems of spray pond water management.

9.2.7.2.2 Component Description

Generally the ultimate heat sink consists of a concrete lined spray pond covering approximately 8 acres and containing 25,000,000 gal of water, and an ESSW intake structure housing four RHRSW pumps and four ESW pumps which pump the water from the pond through their respective loops and back to the pond through a network of sprays located in the pond. The pond and ESSW pumphouse are described in more detail in Subsection 3.8.4.1 and shown on Figures 3.8-94 through 3.8-98.

The spray pond is a Seismic Category I design excavated below grade and has a normally maintained water level of 678.5 ft MSL. The spray pond water volume is adequate for 30 days of cooling without any makeup, as demonstrated in Subsection 9.2.7.3, and is concrete lined to minimize seepage.

The ESSW intake structure which houses the RHRSW pumps and ESWS pumps is located on the spray pond so that a positive water supply is provided at all times to each pump suction. The pumps are the vertical type and the pump pit dimensions are such that the required NPSH for each pump is ensured even at the minimum water level. The spray pond location is shown in Section 1.2. The ESSW intake structure is Seismic Category I and its design is explained in Subsection 3.8.4.

The spray system for the pond consists of one Seismic Category I network for each ESSW service water train. Each network is being sized to provide a uniform pressure of about 7 psig at each spray nozzle throughout the network for a uniform water spray. The nozzle, Spray Engineering Model 1751A, is shown in Figure 9.2-23. The nozzles are precision-cast and are of a design that provides good thermal performance while minimizing drift loss. The nozzles have no internal parts that are susceptible to clogging. The piping in the spray system is designed and installed in accordance with ASME Section III, Class 3.

Four one-third capacity 13,500 gpm motor driven pumps are located in the river intake structure. They supply makeup water to the entire plant through a buried line sized to provide sufficient water to replenish the losses resulting from normal operating plant demands such as the cooling tower basins as well as makeup to the spray pond. An 18 in. makeup connection to the spray pond is tapped off the main 42 in. supply line at a point close to the spray pond. The makeup line, which is also used for de-icing, is arranged in such a manner as to avoid the possibility of water draining from the pond if a failure occurs in the makeup supply system. This line is used to fill the pond initially and to refill it following its use as the result of an emergency. When the pond is not in use the only loss is by evaporation, which is made up by rainfall and a continuous flow of water through a 4 in. bypass line around the closed isolation valve in the 18 in. makeup line. Any excess water in the pond flows over a weir and back to the river. Meteorology for the area indicates that rainfall is expected to add more water to the pond than is lost by evaporation.

9.2.7.2.3 System Operation

Summer Startup and Operation

The conditions that could exist in the pond during summer startup will be less severe than the conditions that could exist during summer operation. No attempt will be made to start up either unit if the pond is not full.

During periods of summer operation with no spraying but high ambient wet bulb temperatures, the pond temperature will approach the equilibrium temperature which is that temperature a stagnant body of water will reach after prolonged exposure to the ambient conditions. The maximum Susquehanna pond temperature was calculated to be 89°F under these conditions.

Under normal plant operating conditions, the maximum pond temperature will approach the equilibrium temperature. Technical Specification limits have been established to limit plant operation if the pond bulk temperature reaches 88°F. This temperature has been calculated to be the maximum allowable starting temperature if the ESSW temperature is to be limited to 95°F under the worst meterological and plant accident conditions.

The total pond water volume is assumed to start at the highest pond temperature reached after exposure to the worst ambient conditions. This assumption is conservative, since at times of generating unit startup, the pond temperature will actually be below this assumed value. A lower pond temperature will increase the quantity of sensible heat that can be absorbed by the water volume. Summer startups will therefore impose no limitation on the ultimate heat sink capability.

Winter Startup and Operation

Startup of either unit will not be implemented unless the spray pond, spray network, and pumping system are available for operation.

At times of subfreezing temperatures, procedures will be enforced to prevent icing of the spray system. These consist primarily of the following:

a) Simultaneous shutdown of both units for refueling or extended maintenance would, if possible, be scheduled to avoid midwinter conditions, thus avoiding the possibility of freezing the ESSW pump suctions after the heat source is exhausted.

b) The total return flow of both the RHRSW and ESW pumps will be first discharged directly into the pond, through a bypass line, without passing through the spray network. This will permit the operation of the pond if nozzles become covered with ice from, for example, a freezing rain. As the water temperature in the pond increases, conduction of heat to the nozzle will melt any accumulated ice. The bypass lines enter above the pond level so that they drain to prevent freezeup and therefore always assure a flow path for the ESW & RHRSW. The bypass lines are located between the two spray networks, approximately 400 feet away from the pump suctions. The physical distance between the pump suctions (which are kept ice free) and the return lines makes the probability of increasing the water temperature of the pump suction above the design maximum temperature due to short circuiting without significantly thawing the pond negligible. An overview of the pond piping and bypass lines are shown on Figure 9.2-24 Sh. 1. The bypass lines are numbered as 36" HRC-1 and 36"HRC-2.

c) Portions of the nozzle header and riser system that are located above pond water level can be drained when not in service. Draining is accomplished by pumping the water out of the pond piping at the low point. The resulting water level in the pipe is more than 2 ft below the pond water level.

d) The majority of the water distribution system associated with the ultimate heat sink will be either buried below the frost line or located inside heated buildings and therefore not exposed to freezing problems.

e) Any sections of the piping which are either not within buildings, or drainable will be electrically traced to protect against freezing. This electrical supply for the tracing is not supplied from the diesel generators since, in the event of auxiliary power loss, heated water will be flowing in the piping that is traced.

The maximum expected ice thickness, assuming there is no heat load on the spray pond, is estimated to be 22 inches, which agrees closely with the maximum expected ice thickness based on probability studies that used field data for colder regions of North America (Ref 9.2.7.9.4).

The extreme weather conditions used for the above analysis were obtained from meteorological records and were based on the month having the lowest average dry bulb temperature. This average temperature was used for the analysis and the resulting estimate of maximum ice thickness is therefore conservative.

With the extreme (cold) meteorological conditions considered, no provision is made to prevent freezing of the spray pond surface if both units were shut down at the same time. However, freezing of the pond when the units are shut down is not a safety concern.

9.2.7.3 Safety Evaluation

The ultimate heat sink spray pond is capable of providing enough cooling water to safely shut down and cool down both reactors, without the addition of makeup water, for 30 days concurrent with any of the following postulated design basis events:

a) SSE, flood or drought.

b) Any single site related event.

c) A reasonably probable combination of less severe natural phenomena and/or site related events.

d) Man-made structural features of the spray pond are designed considering all conceivable failure mechanisms, including the SSE and design basis tornado effects. Conservative allowances are added to the spray pond water volume to account for seepage through the liner. (See Table 9.2-8)

Where the above design events could result in the loss of offsite power, such a loss is assumed. In addition, a single failure is postulated.

The ESSW intake structure is located directly adjacent to the spray pond; therefore, no canals, conduits, or waterways are associated with or required to ensure positive water flow to the suction of the RHRSW pumps and ESW pumps.

The pumps for each loop are in separate closed rooms within the ESSW pumphouse. There are no communication pathways between pump rooms. Internal flooding due to a leakage crack in the moderate energy piping would be mitigated by four 3' by 3' openings with gratings which drain to the spray pond. The pump room doors are at an elevation 3 inches higher than the drains to contain the leakage, estimated at less than 1600 gpm (BTP MEB 3-1), to within the room. Therefore, flooding in one pump room will have no effect on the safe shutdown capabilities of the other loop.

If a tornado passes over the site and causes a loss of water from the spray pond, makeup water will be provided by either the makeup water pumps or in an extreme emergency the cooling tower basins. The minimum operating water elevation in the spray pond will be 678.50' MSL The maximum anticipated elevation will be 681.70' which occurs under PMF conditions. The only postulated exception to these limits is that during the event of a tornado passing over the pond, the water level may temporarily be lower than elevation 678.50' resulting from tornado effects.

The power supply to the motors for the makeup water pumps is provided from an offsite power source through underground cables.

Even if one of the makeup pumps fails, a sufficient flow of water to the spray pond is ensured since each of the four pumps is designed to deliver one-third of the total plant makeup flow requirement.

All spray network headers are located in concrete trenches at the bottom of the spray pond and covered with concrete 18" thick to resist the impact of a tornado missile.

The spray pond and its associated ESSW intake structure are protected from the maximum probable flood level as discussed in Subsection 2.4.8. If a flood requires shutdown of the plant, the spray pond will supply the required cooling water for a 30 day period.

The spray pond is designed to contain the total volume of water required for 30 days of cooling without makeup. After 30 days water will be available from the river for makeup to the pond for long-term cooling. The Susquehanna River is a reliable source of water even during a severe drought (see Section 2.4). As a result of the reliability of the river and the spray pond, a drought has no impact on the operation or shutdown of the plant.

The potential for incapacitating accidents on the site has been evaluated and is discussed in Section 2.2. The physical remoteness of the ultimate heat sink to the avenues used for bulk petroleum transportation makes massive fouling of the heat sink surface by an oil spill unlikely. Vehicles delivering diesel fuel oil to the site will not be permitted to remain in the area of the ultimate heat sink in order to prevent an accident involving the delivery vehicle, which could result in an oil spill.

A fire would have minimal impact upon safe shutdown cooling, inasmuch as the ultimate heat sink and related equipment are largely heat resistant or noncombustible. However smoke detectors are installed inside the ESSW intake structure and CO fire extinguishers are located there. A hydrant is also available adjacent to the structure.

The credible failure of a man-made structural feature will not result in the loss of the ultimate heat sink safety function. The lined spray pond is constructed by excavation and is not subject to catastrophic failure (see Subsection 2.5.5).

9.2.7.3.1 30-Day Transient

The Seismic Category I spray pond has enough water available for at least 30 days without makeup and the design maximum cooling water temperature is 95°F which is based on the worst atmospheric

9.2.7.3.1 30-Day Transient

The Seismic Category I spray pond has enough water available for at least 30 days without makeup and the design maximum cooling water temperature is 95°F which is based on the worst atmospheric conditions on record. Analyses have been performed to demonstrate the ability of the spray pond to meet these criteria.

In analyzing the ability of the spray pond to dissipate the heat rejected from both the RHRSW and ESW systems, alternative 30 day transients have been considered. The method of analysis is presented in Subsection 9.2.7.3.2, and a discussion of the results will be found in Subsection 9.2.7.3.7.

An analysis of the 30 day transient coincident with loss of offsite power to both generating units is presented below.

If both generating units have been operating at full power and a LOCA occurs on one unit, followed by a forced shutdown (without offsite power) on the second, the following sequence of events is assumed to occur:

a) Both reactors would be scrammed and both turbine-generators isolated.

b) The loss of power would cause loss of makeup and circulating water and loss of condenser vacuum on both units.

c) Safeguard equipment, common to both units, would be actuated (four diesels, four ESW pumps).

d) On the unit experiencing the LOCA, all safeguard equipment would be actuated (four core spray pumps, four RHR pumps, ADS, and HPCI).

e) On the unit undergoing the forced shutdown due to loss of offsite power, the RCIC and HPCI system would actuate to hold reactor water level while the safety relief valves limit reactor pressure.

f) All supporting systems associated with the above steps would be brought into service (eg, diesel-generators, emergency service water, RHR service water).

The occurrence of the accident automatically initiates safeguards operation. After 10 minutes, the equipment is operator controlled and, by defining the time these operations are started, the heat rejected to the ultimate heat sink is established. This complicates the analysis and necessitates the study of alternative means of shutdown to determine which result in the limiting heat sink criteria. The most stringent criteria

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conditions on record. Analyses have been performed to demonstrate the ability of the spray pond to meet these criteria.

In analyzing the ability of the spray pond to dissipate the heat rejected from both the RHRSW and ESW systems, alternative 30 day transients have been considered. The method of analysis is presented in Subsection 9.2.7.3.2, and a discussion of the results will be found in Subsection 9.2.7.3.7.

An analysis of the 30 day transient coincident with loss of offsite power to both generating units is presented below.

If both generating units have been operating at full power and a LOCA occurs on one unit, followed by a forced shutdown (without offsite power) on the second, the following sequence of events is assumed to occur:

a) Both reactors would be scrammed and both turbine-generators isolated.

b) The loss of power would cause loss of makeup and circulating water and loss of condenser vacuum on both units. Loss of the main condenser places maximum heat dissipation requirements on the ultimate heat sink.

c) Safeguard equipment, common to both units, would be actuated (four diesels, four ESW pumps).

d) On the unit experiencing the LOCA, all safeguard equipment would be actuated (four core spray pumps, four RHR pumps, ADS, and HPCI).

e) On the unit undergoing the forced shutdown due to loss of offsite power, the RCIC and HPCI system would actuate to hold reactor water level while the safety relief valves limit reactor pressure. After 10 minutes, reactor water level would be maintained by RCIC alone.

f) All supporting systems associated with the above steps would be brought into service (eg, diesel-generators, emergency service water, RHR service water).

The occurrence of the accident automatically initiates safeguards operation. After 10 minutes, the equipment is operator controlled and, by defining the time these operations are started, the heat rejected to the ultimate heat sink is established. This complicates the analysis and necessitates the study of alternative means of shutdown to determine which result in the limiting heat sink criteria.

The maximum head load to the spray pond will occur with a LOCA/Forced Shutdown combination, as opposed to a two unit forced shutdown. Two different LOCA/Forced Shutdown scenarios were developed for this analysis. The shutdown scenario for the minimum heat transfer case assumes the availability of only one division of spray networks, thereby minimizing the amount of heat dissipated to the atmosphere. The shutdown scenario used for the minimum heat transfer case is shown in Tables 9.2-4 and 9.2-21. The shutdown scenario developed for the maximum water loss case assumes the availability of both spray networks, thereby producing the maximum spray evaporation and drift losses. The maximum water loss shutdown scenario is shown in Tables 9.2-5 and 9.2-21a.

The most stringent criteria were used in the analysis and the results demonstrated the ability of the Susquehanna SES spray pond to meet the performance requirements of an ultimate heat sink.

9.2.7.3.2 Methods of Analysis

The analysis is directed at providing sufficient information to define the following three parameters;

a) Pond surface area

b) Pond water volume

c) Nozzle arrangement.

Input Parameters

Heat rejection after the postulated accident during the shutdown sequence is due to decay heat, sensible heat, and auxiliary system heat loads. For this analysis a decay heat curve from the NRC Branch Technical Position APCSB 9.2 was used as presented in Table 9.2-19. The values listed in Table 9.2-19 include fission product and heavy element contributions to the heat generation rate. Sensible heat release is included in the mathematical treatment of the heat removal system model. The diesel generator and auxiliary system heat loads are presented in Table 9.2-20.

The cooling system flow rates released as heat loads to the pond are tabulated in Table 9.2-21 for the RHR and RHR Service Water Systems and in Table 9.2-22 for the Emergency Service Water System.

The initial conditions assumed for the heat removal system model are listed in Table 9.2-23. The results of containment analysis presented in the PSAR were used to determine containment initial conditions (10 minutes after LOCA for this analysis).

The input parameters used in the spray pond thermal efficiency calculations and drift loss calculations are presented in Tables 9.2-24 and 9.2-25, respectively, and are based on spray pond geometry, assumed shutdown sequence, and synthesized meteorology.

Pond Surface Area

Sufficient area is provided to allow the full complement of spray nozzles to be located on the pond surface.

Sufficient area is provided to ensure that the distance of the outermost line of nozzles to the edge of the pond is great enough to prevent unacceptable water losses that result from drift.

Pond Water Volume

Pond water volume has been selected such that the water losses listed in Table 9.2-8 can be experienced over the 30 day transient period while the pond is still able to perform the necessary cooling duty until the end of the 30th day.

Sufficient water is provided to ensure that the sensible heat capacity of this heat sink, together with the cooling ability of the nozzles, are sufficient to keep the temperature of water supplied to the equipment below the design temperature of the equipment. Ensuring that the design temperature is not exceeded is essential in meeting manufacturers' recommendations for equipment and also in limiting the containment transient temperatures that are dependent on the RHR service water temperature.

The pond volume and spray network are designed to limit the maximum temperature of the Engineered Safeguard Service Water to 95°F.

The pond is filled initially with river water, and assuming the pond is not used due to an emergency, no makeup will be required to maintain the level since the average annual rainfall is in excess of the water lost from the pond through evaporation and seepage. Thus the amount of silt deposited on the bottom of the pond will be insignificant and consequently the need to clean the pond will be minimal.

An overflow weir fixes the level of the pond as water is continuously introduced through a 2 in. bypass line around the isolation valve in the 18 in. makeup line; thus, the minimum level is always maintained while either unit is operating. (See Subsection 2.4.8)

Nozzles and Nozzle Arrangement

Nozzles and nozzle arrangement, shown in Figure 9.2-24, are selected such that the optimum heat dissipation is reached, satisfying the following requirements:

a) There are sufficient nozzles to dissipate the maximum heat load resulting from the emergency shutdown operation without allowing the pond temperature to exceed the maximum permissible as discussed above.

b) The nozzles are as close to one another as possible without hindering individual performance.

c) The spray pressure at the nozzles has been selected to optimize the water droplet surface area for heat rejection while minimizing small droplet generation that would increase drift losses. The selected nozzle was chosen because of the experience of the supplier, wide use of this particular nozzle, and a spray pattern close to optimum for minimum drift with maximum thermal dissipation.

d) The piping distribution system supplying the nozzles is arranged to permit isolation of nozzle networks. This will permit startup and shutdown of selected RHRSW or ESW pumps throughout the 30 day transient, while maintaining optimum nozzle pressure.

Correct nozzle pressure is set by throttling of a single valve on each of the two return headers to the pond.

The large number of parameters associated with the above basic variables necessitated the development of analytical models suitable for computer use. There are three principle models and these are outlined below.

9.2.7.3.3 Pond Performance Models

The analysis of the SSES emergency cooling water system is based on three computer models: the spray cooling thermal performance model, the drift loss model, and the system response model. Each of these models are discussed individually.

Use of the three models requires input details on ambient conditions and these have been prepared by PP&L's meteorological consultant, Dames & Moore. A discussion of the use of the meteorology report is presented in Subsection 9.2.7.3.6.

Spray Cooling Thermal Performance Model

The performance of a spray pond depends on many parameters, as pond geometry, drop size spectrum, wind velocity, atmospheric conditions and spray height. All the controlling parameters feasible have been included in the computer model as described briefly below.

The computer model developed for this analysis includes the effects of the following parameters:

1. Drop mean diameter.

2. Wind speed and direction.

3. Air dry bulb temperature.

4. Air wet bulb temperature or relative humidity.

5. Height of nozzles above water level.

6. Pressure drop through the nozzle or height attained by the spray.

7. Dimensions of the spray volume.

8. Water flow rate in spray volume.

For high wind speeds (above 3 mph approximately), the heat transfer mechanism is assumed to be forced convection. For low wind speeds cooling is assumed to be by natural convection only. The individual spray patterns are lumped together to form the spray volume which is divided into a number of increments in the direction of the air movement. The temperature and vapor content of the air in each increment is assumed to be uniform within the increment and is numerically the same as that exiting the preceeding increment. The sprayed water temperature, air temperature, and air moisture content for each increment is calculated and the results combined to yield an average sprayed water temperature for the spray volume. A critical aspect of the calculation is the determination of the evaporation rate within the increment. The empirical work of Ranz and Marshall (Ref. 1 of Question 3.71-18) on droplet heat and mass transfer was used as the basis for the evaporation rate and air temperature calculations. In their experiments, Ranz and Marshall suspended a drop from a capillary tube, supplied a known air flow over the drop surface, and measured the drop temperature, air temperature, drop diameter (held constant with water flow through the capillary tube from a microburet), and make-up flow rate from the microburet. In this way the heat transfer coefficients were derived by correlation with the data.

The increment mass and energy balance used in the calculation of spray cooling efficiency is shown schematically in Figure 9.2-17. Water enters the increment through the spray nozzles (flow rate m_{ws} at temperature T_s) and exits the increment after undergoing mass and energy transfer (flow rate $\dot{m}_{wp}$ at temperature T_{di}). The amount of mass and energy transfered is calculated from heat and mass transfer coefficients derived empirically:

$$N_{NU} = \frac{h_c D}{K} = 2.0 + 0.60\, N_{PR}^{1/3}\, N_{RE}^{1/2}$$

$$N_{SH} = \frac{h_d D}{D_v} = 2.0 + 0.60\, N_{SC}^{1/3}\, N_{RE}^{1/2}$$

where N_{NU} = Nusselt number

N_{SH} = Sherwood number

h_c = Heat transfer coefficient for conduction and convection

h_d = mass transfer coefficient

D = drop diameter

K = thermal conductivity of air-vapor moisture

D_V = diffusivity of vapor in air

N_{PR} = Prandtl number

N_{RE} = Reynolds number

N_{SC} = Schmidt number

The energy transfer rate is the sum of contribution from conduction, convection and evaporation. The lifetime of a drop in the increment, calculated from the pond geometry and other parameters affecting the drop trajectory, is used with the energy transfer rate to determine the temperature of the cooled water leaving the increment. The moisture content of the air leaving the increment is determined from the mass transfer (evaporation) rate and the air flow rate (residence time of the air in the increment). The temperature of air exiting the increment is calculated from an energy balance on the increment. The exit air temperature and moisture content for increment i is used in increment i + 1 to determine the heat and mass transfer rate in that increment. This process is repeated until all the increments have been treated.

At low wind speeds (less than approximately 3 mph), air enters the spray volume from all sides rather than one; therefore, the increment definition used for low wind speeds is rectangular, like a picture frame. The air velocity entering each increment is determined from the density difference between the air-vapor mixture in the increment and the ambient.

Spray efficiencies for wind speeds below 3 mph are calculated assuming natural convection only. Spray efficiencies for wind speeds greater than or equal to 3 mph are calculated assuming forced convection only. This procedure shows good agreement with the test results and avoids excessive conservation.

The results of the calculation described above is a set of cooled water temperatures, one for each increment. Since the air temperature and moisture content for each increment is different, the cooled water temperatures are different. The average cooled water temperature, $\overline{T}$, is calculated.

$$\overline{T} = \sum_{i=1}^{n} \frac{Ti}{n}$$

where

T_i = incremental cooled water temperature,

n = number of increments in the spray volume

The thermal efficiency, E , is calculated from the ambient air wet bulb temperature, T_{wb}, and the water temperature before spraying, T_s.

$$E_{th} = \frac{T_s - \overline{T}}{T_s - T_{wb}}$$

The thermal performance prediction model gives more conservative results than other prediction methods available. The primary conservatism in the model is the lack of convective air motion into the spray volume, (for all but very low wind speeds) which results in lower calculated efficiencies. The convective air motion is most important at low wind speeds; consequently, the degree of conservatism increases as wind speed decreases. Since, thermal performance at low wind speeds is most important, this is a desirable effect as long as the degree of conservatism is not unrealistic. Data taken at existing spray ponds has been used to demonstrate the degree of conservatism of the model.

In the model the temperature of the water being sprayed is calculated using an iterative technique based on the temperature of the pond and the heat addition to the spray water at each increment in time. If the heat dissipated by the sprays is less than that added to the system, the temperature of the water entering the pond after spraying is higher than the bulk pond temperature. As a result, the pond temperature increases until the heat added to the system equals that dissipated by the sprays. As the heat load on the pond decreases with time, the sprays dissipate more heat than is being added and the pond temperature begins to decrease.

Drift Loss Model

During periods of prolonged spray pond operation without makeup water, it is essential that accurate predictions of water consumption are available. The thermal performance model that was developed is used in conjunction with the system model to predict evaporative losses. An independent model has been developed to predict drift losses. A review of the literature revealed no efforts directly applicable to calculation of drift losses from a spray pond. Due to basic system differences, cooling tower drift measurements cannot be applied directly to spray ponds; therefore, a model was developed from principles of analytical mechanics. The following parameters were included in the model:

1. Drop size spectrum

2. Wind speed and direction

3. Elevation necessary for loss of a drop from the pond

4. Distance of each nozzle from the perimeter of the pond in the direction of drift

5. Pressure drop across the nozzle

6. Angle at which water leaves the nozzle

7. Vertical air entrainment of droplets

Drift is caused by the horizontal drag force exerted on small drops as they move relative to the air. A water drop leaves the nozzle with a certain initial velocity and from that time its motion is determined by drag and gravitational forces. By solving the equations of motion the position of each drop is determined as a function of time. When all initial velocities are considered, the positions of drops of the same size that left the nozzle at the same time trace out a locus in the horizontal plane. When drops of similar size are grouped together a locus results for each drop size group.

The loci are concentric circles for a wind velocity of zero, and are somewhat distorted and translated in the wind direction for non-zero wind speeds.

Once the loci have been determined, for a given windspeed, the fraction of flow lost by drift for each drop size group is the ratio of the length of the locus outside the pond perimeter to the total locus length. Since a locus represents the position of drops of a given group no drops from that group are off the locus at that elevation; consequently, the length of the locus is used to calculate loss fraction rather than the enclosed area. The percentage of flow lost by drift is the sum over the drop size groups of the product of drift loss fraction and flow fraction.

$$P = \sum_{i=1}^{N} f_i B_i$$

Where P = percentage of flow lost by drift

f_i = fraction of flow in drop size group i that is lost

B_i = fraction of total flow in drop size group i

n = number of drop size groups

In order to facilitate evaluation of the drag coefficient for each drop size group, the drops are assumed to be spherical. High speed photographs show that drops deviate very little from being spherical, especially in smaller diameters that are most important in drift loss considerations.

Since the draf force on a sphere is proportional to the relative velocity raised to a power between 1 and 2 (depending on the Reynold's number), the resulting equations of motion are non-linear. An approximation is made to allow a solution in closed form in which the draq force is assumed to vary linearly over a certain range of velocities. Two velocity ranges are used and for all velocities the approximation equals or exceeds the actual drag force, thus preserving conservatism by maximizing drift losses.

The linear drag force approximation in combination with Newton's Second Law is used to determine the acceleration of a drop and the acceleration is integrated to determine the position of the drop as a function of time. This is done for both the X and Z directions, shown in Fig. 9.2-18 to determine the coordinates, $X_i(t,0)'$ $Z_i(t,0)'$ of drop position for the ith drop size group as a function of time and initial direction. The motion in Y direction is used for calculation of the drop exposure time only.

In order to find the locus of a given drop size group at the elevation necessary for loss from the pond, the time of flight, or exposure time, must be calculated. The motion of a drop in the y-direction, shown in Figure 9.2-18, is used to calculate the exposure time. Since the water leaves the nozzle in a conical pattern, no draq is applied for the first few feet of travel in the verticle direction to allow maximization of the time in the air, which maximizes drift losses. Drag is applied immediately in calculation of X and Z coordinates in order to maximize drift losses. The vertical position, y(t), is determined as a function of time; subsequently, the elevation necessary for loss from the pond is substituted for the position and the resulting implicit equation is solved for exposure time. There is a different exposure time for each drop size group due to the dependence of drag force on drop diameter.

With the exposure time determined, the locus for each drop size group can be generated by considering all initial velocity directions. A computer program has been written to supply the coordinates of points of each locus. The locus in the X, Z plane for each drop size group is integrated numerically over its length to determine the fraction of the locus, and hence the drop size group, that is beyond the perimeter of the pond. The losses for the different drop size groups are summed to determine the total drift loss percentage from the pond.

The percentage of flow lost due to drift is an input parameter for the system model discussed later. The system model uses it as a loss term in determining the water remaining in the spray pond at any time after the start of operation of the sprays. The drift loss for the SSES spray pond is determined as a function of wind speed, and this information is entered as a table, windspeed versus drift loss, in the system program. The drift loss is determined from the table at each time step in the calculation of system parameters.

System Response Model

In order to predict the response of the emergency cooling water system of the SSES design, it was necessary to develop a computer model of the system. Due to the feedback effects of service water temperature on containment response, the model includes the system from reactor vessel to spray pond. Of particular interest in the transient analysis of the system is the containment temperature, the service water temperature, and the pond water inventory.

In the computer model the system analyzed is represented by a set of simultaneous differential equations resulting from mass and energy balances written for each element of the system. The following assumptions have been made in writing the equations:

1. The absorption of heat by cooling water system equipment and piping during the transient does not significantly contribute to the system response and is therefore neglected.

2. Saturated conditions are assumed to prevail in the containment and reactor pressure vessel.

3. The RHR heat exchanger effectiveness is calculated using equations from Kays and London (Ref. 5 of Question 371.18) when operating as water-water heat exchangers.

4. In the steam condensing mode, heat tranferred through the RHR heat exchangers consists of heat of condensation and some further cooling of the condensate. The total heat transfer rate is determined by the steam flow rate and the RHR heat exchanger effectiveness specific to the steam condensing mode.

5. Flow rates are assumed to change instantaneously when changed.

6. The NRC Branch technical Position APCSB 9-2 decay heat generation curve is used with both fission product and heavy element contributions.

7. The transient analysis is initiated after blowdown at 10 minutes after a LOCA.

8. Complete mixing is assumed where flows are combined in piping.

9. No heat is assumed to be transfered through the containment walls, piping, or spray pond liner.

10. Complete mixing in those elements containing water.

The set of equations that represents the system is solved using a discrete finite differential method. The output of the computer model provides temperatures and water inventories at various points in the system at specified times during the transient analysis. This information is used to plot parameters of interest.

The heat rates, integrated heat rates, pond temperature vs. time, and pond volume vs. time are shown in Figure 9.2-19 through 9.2-22, respectively. The heat rates and integrated heat rates are also presented in tabular form in tables 9.2-30 and 9.2-31.

9.2.7.3.4 Droplet Spectrum Test

Both drift and performance models rely on droplet size input data. A program by nozzle vendor has been established for measuring the droplet size spectrum from the particular nozzle selected for the system and at the particular nozzle pressure chosen. These measurements are based on established high speed photographic techniques. This test program has been completed and the maximum drift has been used in the 30 day transient analyses.

9.2.7.3.5 Discussion of Meteorology

The evaluation of spray pond performance as an ultimate heat sink is based on conservative atmospheric conditions. The basis for selection of these conditions is critical due to the sensitivity of performance to variations in wind speed, temperature, and relative humidity. This requires investigation of two somewhat opposing sets of atmospheric conditions: one that would result in maximum water loss (high wind speed resulting in maximum drift loss) and one that would result in minimum heat transfer (low wind speed, high wet bulb temperature, and high relative humidity). The two sets of conditions have been determined from the available weather data.

The combined effects of wind speed, wet bulb temperature, and dry bulb temperature were considered in selection of the worst time periods. The results of the analyses have been used to synthesize separate 30 day periods of minimum heat transfer and maximum water loss.

Minimum Heat Transfer Case

The ability of the spray pond to reject heat is dependent on both ambient conditions and water temperature at the sprays. It thus becomes important to evaluate the spray water temperature corresponding to any data point condition before analyzing whether that data point is unfavorable for heat transfer.

The metecrological conditions for the minimum heat transfer case are determined using a coefficient of performance that assigns a relative cooling performance value to each set of coincident meteorological conditions. The coefficient of performance is based on the empirical work of Ranz and Marshall (Ref 9.2-1) for cooling of water droplets in air.

The meteorological data used for the minimum heat transfer case was selected to comply with Regulatory Guide 1.27 Rev. 2. A 30 day period of meteorological data was synthesized by having the worst, second worst and third worst days followed by 27 days of the worst 30 day running average period as selected by a coefficient of performance model. The results are summarized in Table 9.2-9.

Maximum Water Loss Case

The major water loss mechanisms that are dependent on meteorology are evaporation and drift loss. A coefficient of water loss was derived based on (a) the work of Ranz and Marshall (Ref 9.2-1) for evaporation loss and (b) the drift loss versus wind speed curve resulting from the drift model of Subsection 9.2.7.3.3. This coefficient was used to determine the worst 30 day period for water loss. The results of the analysis for the worst period for water loss are presented in Table 9.2-10.

The maximum water losses are not necessarily coincident with high ambient temperatures. This results from consideration of both drift and evaporative contributions to water loss. Since the objective is to establish the adequacy of the water supply during periods of high total water loss, the 30 day period of highest total water loss is considered rather than a period of "high temperature and maximum persistent wind speeds".

9.2.7.3.6 Discussion of Results

The intent of this discussion is to compare the analytical results of the transient analysis with all other applicable methods of system sizing. Since the analysis has been done to define three separate design parameters (nozzle efficiency, drift loss, and system response), this discussion of the results will treat each separately.

Nozzle Efficiency

It is easy to measure the cooling effect of the spraying of water from a nozzle; therefore, if sufficient records can be obtained from operating spray ponds, it should be possible to show that the nozzle performance as calculated is equal to or more conservative than the recorded results. Comparison of the conditions measured at Harwood, Canadys, and Rancho Seco Spray Ponds indicated close agreement with the performance predicted analytically (refer to Subsection 9.2.7.6). The analytical results indicate somewhat lesser cooling than was actually measured in both cases and therefore the model provides conservative estimates of spray cooling performance.

A further check on the conservatism of the analytical model has been made by comparing the calculated nozzle performance with other methods that could be found for determining nozzle performance. In every case the model predicted more conservative results (less cooling) than the other methods, and these other methods are not all based on purely theoretical approaches. The Spray Engineering Company, for example, predicts the performance of their nozzles on information obtained from numerous operating installations.

Drift Loss

Since the information currently available on water losses from operating spray ponds includes not only drift but all other losses as well, no valid comparison can be made between the Susquehanna drift model and these total losses monitored for the operating ponds.

The result of drift loss calculations (drift loss versus wind speed) are presented in Figure 9.2-15. This represents the total drift loss percentage from the spray network. The drift loss percentage as a function of nozzle location is presented in Figure 9.2-16. Figure 9.2-15 and Figure 9.2-16 were computed using the Bechtel Drift Loss Program. The results of this program compare favorably with data obtained from the results of tests conducted by the University of California at the Rancho Seco spray pond. (Ref. 2 of Question 371.18). There is a rapid increase in drift losses for nozzles near the perimeter as the wind speed exceeds 15 mph. Consequently, the perimeter of the Susquehanna SES spray pond is designed to be a minimum of 60 feet from the perimeter. This is an added conservatism since the majority of nozzles in that row are more than 60 feet from the perimeter.

System Response

From a detailed description of the system model it can be seen that this model is a series of mass and heat balance equations representing the transfer processes that exist in the reactor heat removal circuits.

The minimum heat transfer analysis for spray pond performance was made using one cooling division (or loop) for 30 days. The results of the minimum heat transfer case and the maximum water loss case are presented in Table 9.2-12. The maximum pond temperature observed during the thirty day minimum heat transfer transient is 95.25°F. It is concluded that this value supports the 95°F ESSW temperature limit for the following reasons. First, the accuracy of this calculation, considering the numerous modeling assumptions and conservativisms involved, render the 0.25°F temperature difference insignificant. Second, the pond temperature only briefly exceeds the 95°F temperature limit before dropping below 95°F. The impact on plant equipment from this 0.25°F temperature difference over a short period of time is expected to be negligible.

The solar evaporation losses calculated are not of a sufficient magnitude to warrant verification tests. Information on total losses, as obtained from the tests described in Subsection 9.2.7.3.5, will provide sufficient verification of this calculation.

9.2.7.3.6.1 Thermal Short-Circuiting

Spray pond thermal short-circuiting due to wind blown spray is not considered to be a design problem. The spray nozzles are located at least 60 feet from the intake. Wind has the greatest displacement effect on small droplets. Wind blown spray that may be blown toward the intake has a smaller droplet size spectrum than that for water leaving the nozzle. It should be noted that the heat transfer rate is inversely proportional to droplet diameter. This smaller droplet size spectrum, combined with the winds necessary to produce the drift and the distance the droplets must travel to the intake assures that the small droplets will be cooled with a closer approach to the wet bulb temperature than the large droplets falling near the nozzles. This has been verified by computations using the thermal performance model and the drift model. Droplet temperatures were computed for those that may land on the pond surface within 10 feet of the intake. The highest hourly wet-bulb temperature of the synthesized worst day (Table 9.2-9), 81°F, and the highest spray water temperature predicted by the system performance model, 120.6°F, was used in this computation. The results are shown in Table 9.2-26 for 2 wind speeds, 20 and 30 mph.

9.2.7.3.7 Discussion of Conservatisms Used

In an attempt to ensure the availability and performance of the pond under all circumstances of ambient and cooling duties, the following conservatisms have been employed in the analysis.

9.2.7.3.7.1 Conservatisms in Meteorology

Time Steps in Meteorology

As suggested in Regulatory Guide 1.27, worst day and worst 30 day running average periods are determined and used to synthesize conservative design meteorology.

Thirty-one years (January 1, 1948 to December 31, 1978) of hourly and three-hourly meteorological observations from the National Weather Service Station at Harrisburg, Pennsylvania provided the meteorological data base. This period includes the highest temperature ever recorded at this station: 107° in July 1966.

Table 2.3-4 of the FSAR presents the monthly and annual average wind speed, dry bulb, and wet bulb temperature for the years 1973 through 1975 for Harrisburg. Table 2.3-5 of the PSAR provides identical information from the Susquehanna site for the years 1973 through 1976. From these tables it can be seen that the wind speed is 1.1 m/sec higher, on the average, at Harrisburg; dry bulb temperatures are 2.9°C higher at Harrisburg; and wet bulb temperatures are 2.2°C higher at Harrisburg. Based upon this comparison it is evident that the use of Harrisburg data in the ultimate heat sink analysis was conservative with respect to conditions to be expected at the site.

Coefficients of performance (water losses and heat losses) were calculated for each meteorological observation. Running averages were then sorted in ascending order for heat losses and desending order for water losses. The worse case (minimum) heat loss and (maximum) water loss were then identified as well as the corresponding meteorological data. These are the meteorological data presented in FSAR Tables 9.2-9 and 9.2-10.

Rainwater Additions

No credit is taken for any rainwater additions to the pond volume over the 30 day transient.

9.2.7.3.7.2 Conservatisms in Operating

Spray Cycling

After the peak heat loads have been experienced and/or when the ambient temperatures are lower than the stringent case considered (winter conditions), it will be possible to cycle the spray system on and off to conserve pond volume. The spray bypass would be available at this time and pond water, returned from the plant, could be injected directly to the pond without going through the sprays. The pond temperature would be allowed to rise under these conditions until further spraying became necessary to limit the cooling water temperature. The use of the bypass instead of the sprays will eliminate drift losses and so curve the pond water.

The system has been designed assuming the spray network is in use for the full 30 day transient.

Shutdown Operations

The sequence of shutdown operations affects the peak pond temperature during the transient. For at least the first 10 minutes actions are assumed to be fully automatic and outside the operator's control. The subsequent operations are operator actuated.

The peak temperature reached in the pond is dependent on the rate at which the heat is dissipated. As discussed in Section 9.2.7.3.1, the shutdown sequences have been developed to provide a worst case realistic combination of heat loads and flows to the spray pond for both the minimum heat transfer and maximum water loss analysis. It should be noted that the forced shutdown unit scenario does not include the use of the RHR steam condensing mode of operation. This mode of operation is not available during a loss of offsite power. A loss of offsite power, with the resulting loss of the main condensor, was assumed to be a governing condition for this analysis. Therefore, the steam condensing mode of RHR was not utilized in the shutdown scenario.

Diesel Operation

The 30 day transient analysis for maximum water loss assumes all four diesels are in continuous service for the first 24 hours. Operator action is assumed after 24 hours to reduce the number of running diesels to three.

9.2.7.3.7.3 Conservatisms in Models

Drift Loss Model

Drag force for motion in the horizontal direction is assumed to be applied immediately, even though the drop is not formed until it is a few feet from the nozzle.

The time the drop is in the air above the elevation necessary for it to be lost from the pond is calculated on the basis that

a) The drag force is conservatively estimated so that the time used in the calculation for the drop to fall from its maximum height is longer than it will be in practice.

b) Using the conservatively estimated drag force to calculate the initial velocity at which the spray rises will result in a higher velocity than will actually occur. Consequently the corresponding initial velocity in the horizontal plane, which was used to calculate the drift loss, will also be higher than the actual horizontal velocity. Thus the calculated drift loss, which was used to help determine the capacity of the pond, will be larger than will actually occur.

c) Vertical airflow is included because it increases the time the drop is in the air and hence the time the drop is exposed to horizontal drag forces. The vertical airflow velocity is determined by correlation of model predictions with experimental results from the Rancho Seco tests (Ref 9.2-2). As a result of the correlation, the combined effects of convective airflow and airflow due to entrainment of air in the sprays themselves are included.

Thermal Performance Model

The time the drop is exposed to the air is calculated assuming no drag force. This reduces the overall exposure time, and hence the heat transfer.

System Model

Water loss due to natural evaporation is calculated from a cooling pond model ignoring the spray volume. This increases the natural evaporation loss because the pond area available for natural evaporation loss is greater than it would be if the pond was being sprayed.

Conservative values of all input parameters, such as suppression pool water volume, were used in the calculations made for the system model. Assumptions 1 and 10 in paragraph 9.2.7.3.3 are also conservatisms in the system model.

9.2.7.4 Tests and Inspections

The ultimate heat sink will be preoperationally tested in accordance with the requirements of Chapter 14 as part of tests P14.1, RHR Service Water, and P54.1, Emergency Service Water.

Pre-operational tests will be performed to verify that the controls for the system are functioning properly and that design flows required for operation can be obtained.

A performance test will be performed on the completed pond with heat input from unit one, to ensure that measured performance equals or exceeds that which is predicted by analysis. A description of the test plan, test procedures, and analysis techniques will be submitted for NRC approval prior to operation of Unit 1.

Pipe welds are subjected to heat treating, testing, and inspection in accordance with ASME Section III, Class 3, and the material specification.

The spray system will be tested regularly during normal plant operations in accordance with the requirements of Chapter 16.

9.2.7.5 Instrumentation Applications

Logics and instrumentation are discussed in Subsection 7.3.1.1b and the displays are discussed in Section 7.5. A complete list of the system's process instrumentation is provided in Table 7.5-1.

All the motorized valves in the spray pond system are designed for remote operation from the control room. One loop can be operated from the remote shutdown panels.

The spray pond has temperature indication and alarms for high water temperature and near freezing water temperature.

9.2.7.6 COMPARISON OF SPRAY POND THERMAL PERFORMANCE RESULTS

In order to verify the conservatism of the spray pond thermal performance model, the model has been applied to a spray pond comparable in size to the one proposed for SSES and on which some performance evaluation tests have been performed. The tests were performed on the spray pond at Canady's Station of South Carolina Electric and Gas Company. In addition to these tests the model has also been applied to a smaller pond with well documented performance, the Rancho Seco Nuclear Power Station of the Sacramento Municipal Utility District. The two sets of test results are discussed separately below.

Canady Station Tests

In order to reduce the temperature rise of the Edisto River due to the Canady Station condenser discharge, a spray pond facility was recently built to lower the temperature of the water returned to the river. The South Carolina Pollution Control Authority required that the river temperature rise not exceed 5° F for all river flow rates. The spray pond was designed to provide the required cooling based on various recommendations. It was found that the expected performance was not realized.

Measurements of wind speed, wind direction, wet bulb and dry bulb temperatures, water temperature before spraying, water temperature just before entering the pond, and pond bulk temperature were taken. The water temperature after spraying was measured just above the surface of the pond at several locations and the average reported. Due to the more extensive instrumentation, the results of the Canady tests must be compared to model predicted values on a point to point basis.

The general description of the Canady station spray pond is given in Table 9.2-27. The original design specified that 120,000 gpm were to be cooled from 102° F to 84° F with a coincident wet bulb temperature of 78° F. Subsequently, the design spray flow rate was increased to 180,000 gpm and the cooled water temperature to 88° F. These design conditions proved optimistic.

The results of the comparison of the Canady Station performance with that predicted by the spray pond thermal performance model are presented in Table 9.2-28. The model predictions agree well with the data and are in general more conservative than the measured efficiencies.

Rancho Seco Tests

At the request of the Atomic Energy Commission, the Sacramento Municipal Utility District (SMUD) arranged to have the Rancho Seco spray ponds tested to verify the ability fo the ponds to meet the design criteria. SMUD asked the University of California, Berkeley, to perform an experience evaluation of the performance of the ponds. Of particular interest from a performance standpoint was the thermal efficiency of the nozzles and drift loss verses wind speed.

The same parameters recorded in the Canady Station tests were also recorded in the Rancho Seco tests, with the addition of accurate pond level measurements. The Rancho Seco test results are compared with model predicted efficiency values on a point to point basis as was done with the Canady Station tests.

The general description of the Rancho Seco spray ponds (2) is given in Table 9.2-27.

The result of the comparison of the Rancho Seco performance tests and model predictions is given in Table 9.2-29. It can be

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observed that the model predictions for performance are more conservative that the measured efficiencies.

References

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3. Brannen, Glenn W. "Spray Pond Testing at Canady's Station", South Carolina Electric and Gas Co., Columbia, S.C. Unpublished.

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9.2.8 RAW WATER TREATMENT SYSTEM

9.2.8.1 Design Basis

The raw water treatment has no safety related function and does not convey radioactive materials.

The raw water treatment system is designed to provide filtered and clarified water at an average effluent turbidity of less than 5 ppm for all plant operating requirements. The filtered and clarified water is furnished to the systems and components listed in Subsection 9.2.7.2.

The system is also designed to effectively filter the waste materials generated from the clarification process and the waste regenerant from the makeup demineralizer system (Subsection 9.2.3).

9.2.8.2 System Description

The raw water treatment system consists of the following:

a) One sludge recirculating type clarifier

b) One chemical feed system designed to inject alum, coagulant aid, hypochlorite, and caustic solutions into the clarifier.

c) Two gravity filters

d) One clearwell, 15,000 gal capacity

e) Two clearwell pumps, each 100 percent capacity

f) One 500,000 gal capacity clarified water storage tank

g) Three clarified water pumps of 100 gpm, 200 gpm, and 300 gpm capacity

h) One clarifier sludge holdup sump of 10,000 gal capacity with two discharge pumps, each 100 percent capacity

i) One 10,000 gal filter backwash holding tank with two discharge pumps

j) One low volume waste treatment filter and associated equipment

k) One 300 gal filtrate collection tank with two discharge pumps

l) Associated piping and controls for all system operations. The piping is carbon steel for the water lines throughout the system and is rated at 125 psig at 350° F. The chemical inlet lines to the clarifier are stainless steel and polyvinyl chloride (PVC) for corrosion protection.

The system is depicted on Figure 9.2-7

The river water turbidity is reduced in the clarifier by the addition of chemicals. Design flow through the clarifier is 300 gpm. The expected normal flow during station operation is 120 gpm. The flow through the clarifier is controlled by a flow modulating valve on the clarifier inlet which is regulated by a clearwell level controller and clarifier inlet flow controller. The opening of this valve is limited to prevent the maximum design flow rate from being exceeded.

The clarifier is a positive internal recirculation upflow unit. All chemical addition shall be in proportion to the inlet flow to the clarifier. An inlet flow recorder with totalizer is used to pace the chemical feed utilizing timers. Backflushing and sludge blowdown from the clarifier is automatic and controlled in proportion to inlet flow. The sludge is directed to the clarifer sludge holdup sump for disposal.

The clarified water flows out of the clarifier to the gravity filters. Normally the flow is split between the two filters. However, the system is designed to allow one filter to pass 300 gpm flow while the other filter is backwashing or out for maintenance. Backwashing of the filter is initiated by pressure drop or a timer. The backwash flow is routed to the backwash holding tank from where the discharge pumps operated by level switches pump the backwash water back to the clarifier for further settling.

The filtered water flows by gravity to the clearwell. One clearwell pump is in continuous operation which sends the water to the clarified water storage tank. Flow to the storage tank is controlled by a flow control valve on the clarified water tank inlet. A controller throttles the inlet valve in proportion to the clarified water storage tank level. A recirculation line from the clearwell pumps discharge header to the clearwell is provided for protection during low flow demand.

A single header from the clarified water storage tank supplies the clarified water pumps at a positive suction pressure. The 100 gpm capacity pump is in continuous operation to furnish the expected normal demand of clarified water. If water demands increase, low pressure switches on the pumps discharge header will start the second and third pump as required to meet the system demands. High pressure switches located on the discharge header will in turn trip the two additional pumps in sequence as water demands decrease. Minimum flow recirculation lines to the storage tank are provided for each pump discharge line for pump protection. The clarified water pumps furnish water for the following use during normal operation:

a) Make Up Demineralizer System

b) Domestic Water System

c) Clarifier Bearing Seals

d) Low Volume Waste Filter Dilution Water for Filter Aid

e) Circulating Water Chlorine Evaporators Makeup

f) Circulating Water Pumps Bearing and Seal Cooling

g) Service Water Pump Bearing and Seal Cooling.

The low volume waste treatment filter collects the solid wastes from the clarifier and from the makeup demineralizer system. Because the two waste materials are different in composition, they are filtered separately. Waste regenerant from the demineralizer's neutralization basin is filtered preferentially over the clarifier sludge. The sludge holdup sump has a large storage capacity and a recirculation line is provided to the sump for sludge return whenever the waste filter is treating the waste regenerant. All filtering operations are automatic. Level switches on both the sludge sump and the neutralization basins operate the discharge pumps. Solid waste material is collected on disposable media and discharged into a hopper for disposal. The filtrate flows to a collection tank from which discharge pumps operated by level switches send the filtrate into the circulating water system.

The waste filter is a flat bed type filter that uses air displacement to produce a 30 mg/l suspended solid content filtrate and a solid cake composition of 50 percent solids on disposable media. The clarifier sludge is pumped to a head tank and filtered on a batch basis. The demineralizer waste regenerant is pumped directly through the filter with a differential pressure switch controlling the air displacement and filter indexing. Filter aid is automatically injected into the filter influent to assist the filtering process.

The raw water treatment equipment is located in the water treatment building. The clarified water storage tank is located in the yard. The storage tank also acts as the primary water source for fire protection with a standpipe in the tank which reserves 300,000 gal of the stored water for fire protection.

9.2.8.3 Safety Evaluation

Failure of the system will not compromise any safety related system or component or prevent a safe shutdown of the plant.

There is sufficient redundancy and sizing in the raw water treatment system to ensure a sufficient supply of clarified water for plant operating conditions and to effectively treat the waste materials from the water treatment processes.

9.2.8.4 Testing and Inspection Requirements

Prior to station operation, the raw water treatment system is operated to furnish clarified water to the makeup demineralizer system for startup operations. This use will verify that all system components and controls function properly.

Since the raw water treatment system and associated equipment is in daily use, no periodic equipment testing is required. All equipment is accessible for observation where inspection during use will ensure the system's operability.

Sample sinks are provided to periodically collect samples and analyze the clarified water quality.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.2.8.5 Instrumentation Requirements

The raw water treatment system is furnished with a control panel located in the water treatment building which is designed for all remote pushbutton control of the clarification process. Automatic control of the clarification, chemical injection, and filtering of the raw water is included in this system panel.

Flow, turbidity, and pH are all monitored to verify system performance and alarm abnormal conditions.

The low volume waste filter is furnished with a system panel that automatically controls the filtering of the waste products. Flows of the two waste streams are recorded. Level switches with low level alarms are furnished for the waste filter head and filter aid tanks.

The clarifier sludge holdup sump is furnished with level switches to indicate high level alarms and to control operation of the discharge pumps. Level switches are also provided on the filtrate collection tank and backwash holding tank for pump control and alarms.

The clarified water storage tank is equipped with level switches to indicate high and low level alarms in the system control panel.

The low level switch also trips the clarified water pumps. Local pressure indicators are provided at the discharge of all pumps in the system for pump head indication.

9.2.9 MAKEUP DEMINERALIZER SYSTEM

9.2.9.1 Design Basis

The makeup demineralizer system has no safety related function and does not convey radioactive materials.

The makeup demineralizer system is designed to provide an adequate supply of demineralized water for the plant operating requirements.

The system is required to provide demineralized water to various plant systems for flushing, cleaning, and filling, prior to operation of the station.

The makeup demineralizers are designed to produce an effluent having the following analysis:

Conductivity, micromho/cm @ 25° C	0.2
Chlorides (as Cl), ppm	0.05
pH @ 25°C	6 to 8
Silica (as SiO_2), ppm	0.005
Total dissolved solids	0.08

9.2.9.2 System Description

The makeup demineralizer system consists of the following:

a) One makeup demineralizer having two trains with each train rated for 120 gpm. Each train consists of a cation exchanger, an anion exchanger, and a mixed bed exchanger.

b) One activated carbon filter rated for a maximum flow of 240 gpm.

c) One demineralized water storage tank of 50,000 gal capacity.

d) One demineralized water jockey pump

e) Two demineralized water transfer pumps

f) Demineralizer regeneration system that includes acid and caustic storage tanks, each 7000 gal capacity, acid and caustic positive displacement pumps, caustic dilution hot water heater and associated piping, valves, and controls.

g) Two 18,000 gal capacity rubber lined concrete neutralization basins complete with two 100 percent capacity sample pumps and two 100 percent capacity discharge pumps.

h) Associated piping and controls for all demineralizer operations. The piping is stainless steel throughout the system and is rated at 125 psig at 350° F.

The complete system is depicted on Figure 9.2-8.

Clarified and filtered river water is supplied to the makeup demineralizer system under pressure from the clarified water pumps (see Subsection 9.2.8). The flow rate through the demineralizers is controlled by a flow control valve located at the inlet of the demineralized water storage tank. A controller throttles the flow control valve in proportion to the demineralized water storage tank level.

Design flow through each demineralizer train is 120 gpm. Expected normal flow during station operation is 60 gpm. The demineralizer is capable of either operating two trains in parallel or having one train operating while the other train is regenerating. During normal operation, one train will be on line with the other train on standby. The standby unit will be placed on line by pushbutton as required. The activated carbon filter preceding the demineralizers removes chlorine residual and is capable of providing service flow to one demineralizer train while simultaneously providing regeneration water to the other train.

When the ion exchange capacity of either the mixed bed vessel or the cation-anion vessels is exhausted, the demineralizer train is automatically removed from service. If exhaustion is indicated by conductivity or silica analyzers, the cation-anion bed or mixed bed undergo an automatic timed rinse. If proper quality is not obtained, the train shuts down automatically. Alarm annunciation of exhaustion is indicated in the demineralizer system control panel.

When regeneration of one of the trains is required, the regeneration operation is initiated manually by pushbutton. The regeneration sequences are controlled automatically after initiation. At the end of the demineralizer regeneration, the train goes into the standby position.

Dilute acid and caustic solutions used in the regeneration process are prepared by in-line dilution of concentrated sulfuric acid and concentrated liquid caustic. The concentrated acid is pumped by one of the 100 percent capacity acid pumps to an acid mixing tee where it is diluted with water from the clarified water storage tank. The concentrated caustic is pumped by one of the 100 percent capacity caustic pumps where it is diluted with heated demineralized water to maintain a caustic solution temperature of 120° F.

The neutralization basins collect the chemical wastes from the regeneration process, in addition to chemical wastes pumped from the chemical waste sump in the water treatment building. All equipment and process drains in the water treatment building are routed to the chemical waste sump. Once regeneration is initiated, air mixing of the basin contents automatically begins to aid in the neutralization. A sample pump also operates automatically to monitor pH during neutralization. If the pH is not within the acceptable limit, the condition is alarmed and acid or caustic solution is added to adjust the pH. After neutralization is complete, basin outlet valves automatically open and one of the two discharge pumps operate to send the solution to the low volume waste treatment filter for disposal. Low level switches stop the sampling and discharge pumps, close the basin outlet valves, and open the inlet valve to accept another regeneration waste influent.

The demineralized water flows to the 50,000 gal capacity storage tank. This water supply is used to fill the condensate storage tanks and the refueling water storage tanks prior to unit operation. The demineralizer water is also used prior to unit operation for plant systems flushing and filling. During normal operation, the demineralized water is used for the following services:

a) Makeup demineralizer acid and caustic dilution

b) Turbine building service connections

c) Turbine building sample station

d) Turbine Building Closed Cooling Water System makeup

e) Turbine Building Chilled Water System makeup

f) Turbine building laboratory

g) Lube oil centrifuge makeup

h) Gaseous Radwaste Recombiner Closed Cooling Water System makeup

i) CRD test pump

j) Reactor building service boxes

k) Reactor building sample station

l) Reactor Building Closed Cooling Water System makeup

m) Reactor Building Chilled Water System makeup

n) RHR heat exchanger tube flushing

o) Drywell services

p) Standby Liquid Control System makeup

q) Fuel Pool Skimmer Surge Tank makeup

r) Radwaste building service connections

s) Radwaste building chilled water system makeup

t) Control Structure Chilled Water System makeup

u) Diesel generator jacket cooling water makeup

v) Condensate and refueling water storage tank makeup

A single header from the demineralized water storage tank supplies the demineralized water jockey pump and transfer pumps at a positive suction pressure. The jockey pump is in continuous operation and is controlled by an on-off hand switch. A recirculation line back to the demineralized water storage tank is provided for prevention of pump overheating on low system demands. The two transfer pumps are controlled by on-off-auto hand switches. During normal operation one transfer pump is in the auto position and the other is in the off position. Low pressure on the pumps discharge header will start the transfer pump in the auto position. This pump will stop when a set high pressure is reached. The second transfer pump can be started manually at any time. Recirculation lines are provided for pump protection. A low level switch on the demineralized water storage tank will stop the jockey and transfer pumps.

The makeup demineralizers and associated equipment are in the water treatment building. The demineralized water storage tank is in the yard and is furnished with an electric heater to prevent freezing.

9.2.9.3 Safety Evaluation

Failure of the system will not compromise any safety related system or component or prevent a safe shutdown of the plant.

9.2.9.4 Testing and Inspection Requirements

Prior to station operation, the makeup demineralizer system is operated to furnish demineralized water for startup operations. This use will verify all system components and controls function properly.

Since the makeup demineralizer system is in daily use, no periodic equipment testing is required. All equipment is accessible for observation where inspection during use will ensure the system's operability.

Grab samples are periodically tested to verify demineralizer performance and to ascertain stored water quality.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.2.9.5 Instrumentation Requirements

The makeup demineralizer system is furnished with a control panel, located in the water treatment building, that is designed for all remote pushbutton type control of the demineralization process.

Automatic and manual control of the regeneration and neutralization processes are also included in the system panel. Flow, conductivity, and silica monitors are provided for each demineralizer train to indicate when the ion exchangers are ready for regeneration. High conductivity alarms and high silica content alarms are provided on the makeup demineralizer system control panel to alert the operator to an abnormal condition.

Pressure, temperature, and conductivity are monitored for the acid and caustic regeneration solutions. Level indicators and low level alarms are provided for the acid and caustic storage tanks.

The neutralization basins are furnished with level switches to indicate high and low level alarms and to control the operation of the sample and discharge pumps. Interlocks are provided in the system to prevent regeneration if both basins are full. The system is also furnished with controls to select the empty basin for neutralization if the other basin is full when a regeneration is initiated.

The demineralized water storage tank is equipped with a level switch that alarms in the main control room to indicate demineralized low water level. This switch also trips the demineralized water jockey and transfer pumps. Pressure switches that alarm in the main control room any conditions of system high or low pressure are furnished on the common discharge header of the demineralized water pumps. Local pressure indicators are provided at the discharge of all pumps in the system for pump head indication.

9.2.10 CONDENSATE STORAGE AND TRANSFER SYSTEM

9.2.10.1 Design Bases

The condensate and refueling water storage system has no safety related function and is designed to perform the following functions:

a) Supply water to fill the reactor well and dryer-separator pool of one unit during refueling operations and to provide storage for this water when refueling is completed.

b) Supply condensate for various processes in the radwaste system and makeup for the Plant systems including the condenser hotwells

c) Supply condensate to the suctions of the HPCI, RCIC, Core Spray and CRD Pumps associated with Units 1 and 2

d) Provide a minimum storage capacity of 135,000 gal for the RCIC and HPCI Pumps associated with each unit

e) Provide the capability to demineralize the water in the refueling water storage tank by pumping it through the condensate demineralizers and returning it to the storage tank

f) Provide storage for condensate rejected from the cycle

g) Provide storage for condensate from the radwaste system

h) Provide the capability to drain the reactor well through the condensate demineralizer and back to the storage tank.

i) Provide the capability for the HPCI and RCIC to recycle water during the test made.

9.2.10.2 System Description

The condensate storage system is shown in Figure 9.2-9. The various flow paths are listed in Table 9.2-13, which also includes the operating modes to achieve these flow paths and a diagram showing the valves referenced by number in the operating modes.

The system consists of the following:

a) One atmospheric condensate storage tank for each unit, each with a capacity of 300,000 gal

b) Two horizontal centrifugal condensate transfer pumps, each full capacity, and rated at 600 gpm

c) One atmospheric refueling water storage tank with a capacity of 680,000 gal, common to both units

d) Two horizontal centrifugal refueling water pumps, each full capacity, and rated at 1500 gpm.

e) Interconnecting piping, valves, instruments and controls.

Condensate Storage Tanks (Units 1 and 2)

These tanks are the prefered source of water for the HPCI and RCIC pumps for both operational use and testing. In addition, they supply water to the core spray pumps but only for testing.

The condensate transfer pumps also take their suction from these tanks to provide water for various services in the radwaste building, the reactor building, and for backwashing the cleanup filter demineralizers and the fuel pool filter demineralizer.

Each condensate transfer pump is rated at 100 percent capacity and normally only one runs. If the discharge pressure of the operating pump falls, the second pump will start automatically. Both pumps can be operated in parallel. Each pump is controlled from the main control room.

Each condensate storage tank maintains a minimum storage of 135,000 gallons to service the associated HPCI and RCIC Pumps during plant operation by use of standpipes and locked closed valves on all other lines.

Makeup is supplied by the demineralized water transfer pumps.

The tanks also act as surge tanks for the condensate systems by receiving any rejected condensate from and making up any deficiency in the heat cycle under the action of the level controls on the condenser hotwell.

Refueling Water Storage Tank

The refueling water storage tank stores the water that is used to fill the reactor well and dryer-separator pool of either Units 1 or 2.

During refueling operations the water is pumped from the storage tank to the respective reactor well and dryer-separator pool by the refueling water pumps that are started and stopped manually. Each pump can be controlled from either the main control room or from the refueling floor thus permitting an operator at either of these locations to operate the pumps. Both pumps are run in parallel.

When refueling is complete the water in the reactor well and dryer separator pool is pumped by the refueling water pumps to the storage tank through the condensate filter-demineralizer. Makeup for the refueling water storage tank is supplied by the demineralized water transfer pumps taking suction from the demineralized water storage tank.

The refueling water storage tank also provides water to fill the spent fuel cask storage pool. This water can be returned to the tank by the refueling water pumps through the condensate filter demineralizer.

9.2.10.3 Safety Evaluation

The Unit 1 condensate storage tank and the refueling water storage tank are located outdoors in the area enclosed by the east wall of the turbine building, the north wall of the reactor building and the west wall of the diesel generator building. The area occupied by the two tanks is surrounded by a wall designed to retain the total volume of water contained in both the refueling water storage tank and the Unit 1 condensate storage tank if both tanks rupture simultaneously. The Unit 2 condensate storage tank, also located outdoors, is surrounded by a wall designed to retain the total volume of water in the tank if it ruptures.

Before any water that collects within the retaining walls is drained, it is monitored for radiation and if the reading is above the acceptable level the water is drained to the liquid radwaste system. If, on the other hand, the reading is below acceptable level the water is drained to the storm sewer.

9.2.10.4 Tests and Inspections

The condensate storage and transfer system is used during Plant operation and requires only visual inspections for leakage or deterioration and to verify operation of the various transfer pumps.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.2.10.5 Instrumentation Applications

Condensate Storage Tanks

Each tank is provided with a level transmitter that operates a pen in a recorder located in the control room. Each condensate storage tank has a separate pen. In addition to the level transmitters, each tank has high and low level switches that alarm in the control room and a low low level switch that trips the condensate transfer pumps if they are running.

Redundant low low level switches are installed in each tank to provide a permissive to allow the HPCI pumps to take suction from the respective reactor suppression pool instead of from the respective condensate storage tank, which is the primary source of water to the HPCI system.

Refueling Water Storage Tank

This tank is provided with a level transmitter that operates a third pen on the control room recorder referred to above. In addition the tank has high and low level switches which alarm in the control room and a low low level switch that trips the refueling water pumps if they are running.

9.2.11 POTABLE AND SANITARY WATER SYSTEMS

The potable water system provides cold and hot water of a quality acceptable for human consumption to plumbing fixtures for the entire plant.

The sanitary waste disposal system treats and disposes waste from all the plumbing fixtures except those which could possibly contain chemicals or radioactivity. Those wastes are handled by the radwaste systems.

9.2.11.1 Design Bases

The potable water system has no safety related function and is designed to prevent radioactive contamination of this system. Before it enters the distribution system, the potable water is filtered and treated in order to prevent harmful physiological effect to plant personnel. This treatment is described in Subsection 9.2.8. Its bacteriological and chemical quality conforms to the requirements of the Pennsylvania Department of Environmental Resources.

Potable water system is designed to provide up to a maximum of 200 gpm during peak demand periods. The potable water is drawn from a 500,000 gallon capacity clarified water storage tank.

Water heaters are provided to supply hot water to the toilet and shower areas and other locations where needed. The storage capacity of the water heaters are based on the maximum hot water demand which is anticipated to occur during the plant personnel shift change during maintenance and refueling operations. Use of only 70 percent of the stored capacity of each unit was assumed due to the addition of cold makeup during the drawdown. A recovery period of 5 hours was used to determine the size of the large electric heaters which are located in the Adminstration Building Control Structure, and the Radwaste Building.

9.2.11.2 System Description

The source of water for the potable water system is from the 500,000 gallon capacity clarified water storage tank. The water before it is stored in the clarified water storage tank is clarified and filtered as described in Subsection 9.2.8.

The potable water system is designated as non-Seismic Category I and includes the clarified water storage tank, clarified water pumps, hypochlorinators, potable water retention tank, electric storage water heaters and the necessary interconnecting piping and valves.

The potable water system is shown in Figure 9.2-7.

9.2.11.2.1 Clarified Water Storage Tank

The clarified water storage tank is a carbon steel tank with a capacity of 500,000 gallons. This tank supplies water for the demineralizer, potable water system and the initial 200,000 gpm for the fire protection water system.

9.2.11.2.2 Clarified Water Pumps

Three pumps, one 100 gpm, one 200 gpm and one 300 gpm capacity, supply clarified water to the makeup demineralizer system and to the potable water system. Normally, the 100 gpm pump is in continuous operation, with the larger pumps being on standby. The 200 gpm and 300 gpm pumps start automatically and in sequence when the system pressure drops to "low" or "low-low", respectively. The pumps will stop automatically when reduced demands cause the system pressure to rise.

The 6-inch line going to the makeup demineralizer system is provided with a backflow preventer using double check valves. The makeup demineralizer water is supplied at 120 psig and this is the only source of pressure in the system. There are no pressure sources to initiate a backflow downstream of the backflow preventer.

9.2.11.2.3 Hypochlorinator

A pressure reducing valve reduces the pressure of the potable water supply to 85 psig downstream of the clarified water pumps.

One of the two water-meter-driven hypochlorinator pumps injects hypochlorite from the hypochlorite solution tank into the potable water system. Each pump has a maximum capacity of 50 gpm. The chlorine feed will be proportional to the potable water flow.

9.2.11.2.4 Potable Water Retention Tank

The chlorine treated water is retained in a 5000-gallon capacity, carbon steel, horizontal tank. Baffles in the tank ensure that the water is retained for a minimum of 35 minutes in order for the chlorine to neutralize living organisms in the water before consumption.

9.2.11.2.5 Hot Water Storage Heaters

Electric storage water heaters of immersion heating element type are provided in four places in the plant. Hot water for the plumbing fixtures in the Control Structure is supplied by one 865-gallon storage capacity electric water heater. One 1300-gallon capacity electric water heater located in the Service and Administration Building will supply all the hot water requirements of the Service and Administration Building. One 530 gallon storage capacity electric water heater will supply hot water to the clothes washing machines and decontamination areas in the Radwaste Building. A 30-gallon capacity electric water heater supplies hot water to the plumbing fixtures in the Water Treatment Building. The pressure tanks of the two large water heaters are constructed and stamped in accordance with the latest ASME Code Section IV for 150 psig design pressure. All four water heaters are wired in accordance with the National Electric Code and are UL listed.

9.2.11.2.6 Valves

ASME code-rated and approved relief valves are provided on all electric storage water heaters for temperature and pressure relief.

Self-actuated pressure reducing regulators are provided in the branch lines supplying each building. Pressures are set to ensure that no plumbing fixture or equipment connection is subjected to a static pressure greater than 65 psig or less than 15 psig.

9.2.11.2.7 Piping

Piping materials used in the potable water distribution system will prevent the introduction of objectionable tastes, odors, discoloration and toxic conditions into the system, and conform to the provisions of the Uniform Plumbing Code.

Piping sizes were designed to limit the flow velocity to a maximum of 8 fps and thus minimize noise, system shock and water hammer. Water hammer arresters, approved and certified by the Plumbing and Drainage Institute, are installed at appropriate locations.

9.2.11.2.8 Sanitary Waste Disposal

All wastes from plumbing fixtures that have no potential for radioactive, oil or chemical contamination are conveyed to the Plant Sewage Treatment Plant.

The sewage treatment plant combines, pulverizes, and aerates the influent sewage and then clarifies it by settling the sludge. The treatment plant then removes the sludge and chlorinates the effluent that discharges to the Susquehanna River. The bacteriological and chemical quality of the effluent conforms to the requirements of the Pennsylvania Department of Environmental Resources.

The sewage treatment system is shown schematically in Figure 9.2-10.

9.2.11.3 Safety Evaluation

The potable water and sanitary waste disposal are not safety-related and are not designed to Seismic Category I requirements.

Failure of this system will not compromise any safety-related system or component or prevent safe shutdown of the plant.

Contamination of the potable water system will be prevented by a combination of air gaps, vacuum breakers and backflow preventers of the reduced pressure zone type.

Backflow preventers are provided on both the 3/4-in. hot and cold water branch lines supplying the Laundry Room and the 1-in. hot and cold water branch lines supplying the Radiation Chemical Laboratory in the Control Structure.

The 2-in. cold water lines that will supply water to the decontamination showers and lavatories in Units 1 and Unit 2 Reactor Buildings are provided with backflow preventers before they enter the buildings.

Backflow preventers are also provided on the 1-1/2-in. hot and cold water lines that will supply water to the decontamination showers and lavatories, clothes washers, service sink and the flushing nozzles of the Radwaste Solidification System in the Radwaste Building.

Decontamination lavatories are provided with faucets that are photocell actuated to automatically close whenever the hands are removed. The spout location provides an air gap of 6-1/2-in. from the flood level. All hose bibb connections to the clothes washers are provided with vacuum breakers. All sink faucets with hose connections are also provided with vacuum breakers. The flushing spray nozzles for the radwaste solidification system discharge to atmospheric pressure and are controlled by normally closed, fail closed valves in addition to the 1-1/2-in. backflow preventer.

Sanitary waste is disposed of in accordance with the requirements of the Pennsylvania Department of Environmental Resources. Potentially contaminated waste from the decontamination showers and lavatories, laundry room, and chemical laboratory is directed to the radwaste treatment system.

9.2.11.4 Tests and Inspections

The potable water piping will be subjected to a hydrostatic test pressure of 100 psig. The system will be disinfected with 50 ppm chlorine for 24 hours. The system will then be drained and flushed with potable water. The sanitary waste piping will be subjected to a hydrostatic test pressure of not less than a ten-foot head of water.

Inspection of the entire system for compliance with the provisions of the Uniform Plumbing Code will be performed.

Periodic tests on the potable water will be performed to determine the residual ppm chlorine content. The sanitary waste effluent will be tested periodically to determine dissolved oxygen, settable solids and pH.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.2.11.5 Instrumentation Application

Pressure controllers are provided to start and stop the clarified water pumps as described in Subsection 9.2.11.2.2. Thermostats, high-temperature limit switches, and temperature gauges are installed on hot water storage heaters. Alarm units, activated upon operation of emergency showers and eyewash units, register local alarms.

A flow meter measures, records and totalizes the effluent flow of the sewer system. An electrical control panel with on-off pushbuttons and indicator lights for all four blower motors, surge tank pump, spray pumps, chlorinator and all other motors of the sewage treatment plant are installed in the sewage treatment control house. Trouble alarm for any motor failure in the treatment plant, including trouble with the motor or compressor of the surge tank effluent pump, and the air pressure shall be transmitted to the main plant control room with local indication of the particular malfunction.

9.2.12 CHILLED WATER SYSTEMS

9.2.12.1 Control Structure Chilled Water System

9.2.12.1.1 Design Bases

The control structure chilled water system is designed to supply chilled water at 44°F to the control room floor cooling system, computer room floor cooling system, and the control structure H&V system. These systems maintain design air temperatures inside the control structure during all modes of plant operation.

The control structure chilled water system is designed so that a single failure of any active component, assuming loss of both offsite power and normal source of cooling water, cannot result in loss of chilled water to the above air conditioning systems during all modes of plant operation.

Codes and standards applicable for the system are listed in Table 3.2-1.

The control structure chilled water system has three subsystems.

a) The chilled water circulation subsystem is safety related and designed to meet Seismic Category I requirements.

   The pressure vessels, piping, pumps, valves, and tanks in this subsystem are designed to quality group D, in accordance with Safety Guide 26, March 1972.

   The system was not designed to quality group C (ASME Section III, Class 3), since the purchase orders for the main components (centrifugal chillers and air handling units) were placed in May and July 1974. Regulatory Guide 1.26 (September 1974), provides an option to design the control room chilled water system to quality group C for the plants whose docket date of application precedes January 1, 1975. Since the system is Q-listed and Seismic Category I, the design meets Regulatory Guide 1.26 with the above exception.

b) The emergency condenser cooling water subsystem.

   This subsystem has a safety related function and is safety related function and is designed to meet Seismic Category I requirements.

   The pressure vessels, piping, pumps, and valves in this subsystem are designed to quality group C (ASME Section III, Class 3) to comply with the design basis of the emergency service water system.

c) The normal condenser cooling water subsystem.

   This subsystem has no safety related function and is not Seismic Category I.

   The pressure vessels, piping, pumps, and valves in the subsystem are designed to quality group D in accordance with Safety Guide 26, March 1972.

9.2.12.1.2 System Description

9.2.12.1.2.1 General Description

The system is common to Units 1 and 2. The system consists of two identical 100 percent capacity chilled water trains. Each train consists of a centrifugal chiller, a chilled water pump, one normal condenser water pump, one emergency condenser water pump, six cooling coils, closed expansion tank, chemical addition tank, air separator, interconnecting piping, instrumentation, and controls.

The system is shown schematically on Figure 9.2-11.

Heat from the six space cooling coils is transferred to the chiller by the circulating chilled water. The heat gained is removed from the chilled water in the chiller evaporator, by a flow of refrigerant which in turn is cooled by the condenser cooling water.

During normal plant operation the source of condenser cooling water is the nonsafety related service water system. Whenever emergency conditions prevail, the safety related Emergency Service Water System (ESWS) provides condenser cooling water.

The normal makeup water supply to the chilled water circulation subsystem is through the manually controlled valve provided in the makeup demineralized water system. The ESWS provides a redundant source of makeup water through a manual control valve.

The chemical addition subsystem, provided to minimize piping corrosion and scale buildup, is manually controlled and is normally isolated. This subsystem is not safety related.

The components of the system are located in the control structure building.

An air separator and an expansion tank are provided to accommodate expansion and contraction in the system due to temperature fluctuations.

9.2.12.1.2.2 Component Description

Design data for major components of the control structure chilled water system are listed in Table 9.2-14.

9.2.12.1.2.3 System Operation

One of the two chilled water trains will be in operation during all modes of plant operation including LOCA. Starting a chilled water pump from the control room initiates the operation of that train operation. Under normal conditions, one chilled water train will be operating and the other train will be on standby.

Chilled water outlet temperature will be maintained at 44°F by automatically positioning the compressor inlet vanes that are controlled by the temperature of the chilled water line leaving the evaporator.

Each of the air handling systems is provided with a thermostatically controlled chilled water three way valve modulated by a temperature controller to match the cooling load requirement. Operation of the standby control structure chilled water train will be automatic on failure of the operating train.

The control structure chilled water system is powered from the emergency power supply system. On loss of offsite power the lead chilled water train will restart automatically according to the diesel generator loading sequence.

If the water level in the closed expansion tank falls below the low level, this condition will be annunciated in the control room.

9.2.12.1.3 Safety Evaluation

The control structure chilled water system is housed within the Seismic Category I control structure. Wind and tornado protection is discussed in Section 3.3. Flood design is discussed in 3.4. Missile protection is discussed in Section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in Section 3.6. Environmental design considerations are discussed in Section 3.11.

The components of the system required for emergency operation are designed to Seismic Category I requirements. The components and supporting structures that are not Seismic Category I, and whose collapse could result in a loss of a required function of the chilled water system through either impact or flooding, are analytically checked to verify that they will not collapse when subject to seismic loading from the Safe Shutdown Earthquake.

The chilled water system capacity is selected and tested to provide adequate heat transfer to allow the air conditioning system to maintain design ambient air temperatures inside the control structure building.

Two separate 100 percent capacity independent systems provide mechanical redundancy. Coupled with the redundancy of electrical design, a failure of any single active component cannot result in a loss of both trains of engineered safety feature equipment, thus enabling a safe shutdown of the plant.

For a failure mode and effect analysis of chilled water system, refer to Table 9.2-15.

9.2.12.1.4 Tests and Inspection

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

Provision is made for periodic inspection of major components to ensure the capability and integrity of the system. Local display devices are provided to indicate all vital parameters (pressures and temperatures) required in testing and inspections.

During normal plant operation, when the ESWS is available, the emergency condenser water pump of the operating chiller can be test operated. A test switch located in the control room can simultaneously stop the normal condenser water pump and start the emergency condenser water pump without stopping the chiller.

9.2.12.1.5 Instrument Applications

The hand control switches of the chilled water pumps and the status indicating lights of the major components of the chilled water systems are located in the control room. A chilled water train is started or set in a standby mode through the control switch of the chilled water pump.

The chilled water system instrumentation is redundant and seismically qualified. Power supplies to the instruments are redundant and connected to the emergency buses.

Indicators for the chilled water temperature and flow as well as the emergency condenser water flow are provided in the control room and will monitor the operation of the chilled water loop during emergency conditions.

The following abnormal conditions are alarmed in the control room:

a) Failure of the chilled water pump

b) Failure of the normal condenser water pump

c) Failure of the emergency condenser water pump

d) High and low water levels at the expansion tank

e) Chilled water high temperature.

9.2.12.2 Turbine Building Chilled Water System

9.2.12.2.1 Design Bases

The turbine building chilled water system has no safety related function.

During normal operation the turbine building chilled water system is designed to supply chilled water at 50°F for maintaining design ambient air temperatures in various areas throughout the turbine building. The system is designed to permit periodic inspection, testing, and maintenance of principal components with a minimum loss of normal operation.

Codes and standards applicable for the system are listed in Table 3.2-1.

9.2.12.2.2 System Description

9.2.12.2.2.1 General Description

Unit 1 turbine building chilled water system supplies chilled water at 50°F to supply unit cooling coils, recirculation unit cooling coils, condensate pump room unit coolers, condenser compartment unit coolers, and access control unit cooling coils. The Unit 2 system is identical except that the access control cooling load is met by the Unit 1 system. Units 1 and 2 systems also provide chilled water to their respective mechanical vacuum pump seal water coolers during startup. The following is a description of the Unit 1 system.

The system consists of two centrifugal water chillers, two evaporator chilled water circulating pumps, two condenser water pumps, two chilled water loop circulating pumps, an air separator, an expansion tank, a chemical addition tank, air cooling coils, interconnecting piping, instrumentation and controls. Each of the chillers and pumps is sized for 100 percent of nominal system capacity.

The system is shown schematically on Figure 9.2-12.

Heat gained by the circulating chilled water is removed in the chiller evaporator by a flow of refrigerant which in turn is cooled by the condenser cooling water.

Condenser cooling water is provided by the service water system.

The makeup water supply is through the manual valve provided in the makeup connection to the demineralized water system.

The chemical addition subsystem provided to minimize piping corrosion and scale buildup is manually controlled.

An air separator and an expansion tank are provided to accommodate expansion and contraction in the system due to temperature change.

The components of the system are located in the turbine building.

9.2.12.2.2.2 Component Description

Design data for major components of the turbine building chilled water system are listed in Table 9.2-16.

9.2.12.2.2.3 System Operation

The turbine building chilled water system operates on a year round basis. One chilled water loop circulating pump, one evaporator chilled water circulating pump, one condenser water pump, and one chiller normally operate, while the others remain on automatic standby. However, if the cooling load exceeds the full capacity of one chiller, the standby chiller will automatically pick up a portion of the load.

The system is started by energizing a selected chilled water loop circulation pump. A preselected set of chiller, evaporator chilled water pump, and condenser cooling water pump are then sequenced in automatically.

The return chilled water is cooled by the operating chiller and recirculated through the chilled water supply loop to the supply air cooling coils. Two temperature sensors located in the chilled water supply main modulate the compressor inlet guide vanes of their respective chillers to maintain constant chilled water supply temperature in the loop. The condenser cooling water outlet temperature is maintained constant by mixing the cooling water supply and return flow using two butterfly valves under the control of the condenser leaving water temperature controller.

Three way mixing valves regulate chilled water flow rate through the supply unit cooling coils, recirculation unit cooling coils, and access control H/V unit cooling coils as described in Section 9.4.

When the temperature of air upstream of any of the cooling coils in the turbine building H/V unit or access control H/V unit drops below the set value of the temperature switches, this is annunciated locally as indication of failure of the upstream heating coil. Further drop of air temperature will be detected by another set of temperature switches which will initiate the draining of water in the cooling coils to prevent freezing. Vacuumn breakers, relief valves, and water connections are provided for draining and manual refilling purposes. Manual switches and valves are provided for coil drain system testing.

High and low levels of the expansion tank are annunciated locally and retransmitted as a trouble alarm to the control room. The tank is filled, drained, and pressurized through manual valves.

9.2.12.2.3 Safety Evaluation

Since the turbine building chilled water system has no safety design basis, no safety evaluation is provided. However the system includes some features that ensure its reliable operation during normal plant operation. These features include redundant components for equipment such as chillers and pumps.

9.2.12.2.4 Test and Inspections

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

Provision is made for periodic inspection of major components to ensure the capability and integrity of the system. Local display devices are provided to indicate vital parameters (pressures and temperatures) required in testing and inspections.

9.2.12.2.5 Instrument Applications

Operation of the water chillers and chilled water pumps will be initiated manually from a local control panel. Automatic standby operation of chillers and pumps is provided.

Indicators on the local panel provide operation status of chillers and pumps. Local pressure and temperature indicators display the appropriate system parameter associated with the chillers and pumps.

The following abnormal conditions are alarmed at the local control panel and retransmitted to the control room as a trouble alarm:

a) Failure of any pump

b) Failure of any chiller

c) High and low water level in the expansion tank

d) Low temperature at the upstream face of the cooling coil in the access control heating/ventilating unit and turbine building supply unit.

9.2.12.3 Reactor Building Chilled Water System

9.2.12.3.1 Design Bases

Portions of the Reactor Building Chilled Water System have safety related functions. The safety related portions include the primary containment piping penetrations and containment isolation valves. The safety related portions are designed to meet Seismic Category I requirements.

The chilled water piping inside the drywell will have no adverse effects on adjacent safety related equipment.

During normal operation the Reactor Building Chilled Water System is designed to maintain normal design air temperatures in various areas in the reactor building, including the emergency switchgear and load center room and the drywell.

The system is also designed to supply chilled water to the reactor recirculation pump motor coolers inside the drywell to maintain motor temperature within allowable limits.

The Reactor Building Chilled Water System is designed to permit periodic inspection, testing, and maintenance of principal components with a minimum loss of normal operation.

The Reactor Building Closed Cooling Water System provides a backup to the portion of the Reactor Building Chilled Water System that serves the primary containment, to maintain drywell temperatures and provide reduced cooling to the recirculation pump motor coolers during loss of offsite power or failure of the Reactor Building Chilled Water System.

Codes and standards applicable for the System are listed in Table 3.2-1.

9.2.12.3.2 System Description

9.2.12.3.2.1 General Description

Unit 1 Reactor Building Chilled Water System supplies chilled water at 50°F to the Zones I and III air supply unit cooling coils, emergency switchgear and load center room air handling units, reactor recirculation pump motor coolers, and drywell unit coolers.

The Unit 2 system is identical except that Zone I (Unit 1 portion of the cooling load) is replaced by Zone II for Unit 2.

The following discussion is applicable for the Unit 1 system.

The system consists of two centrifugal water chillers, two evaporator chilled water circulating pumps, two condenser water pumps, two chilled water loop circulating pumps, an air separator, an expansion tank, a chemical addition tank, air cooling coils, interconnecting piping, instrumentation, and controls. Each of the chillers and pumps is sized for 100 percent of system capacity.

The system is shown schematically on Figure 9.2-13a and 9.2-13b.

Heat gained by the circulating chilled water is removed in the chiller evaporator by a flow of refrigerant, which in turn is cooled by the condenser cooling water.

Cooling water to the chiller condensers is provided from the plant Service Water System. Makeup water supply is through a manual valve provided in the branch connection from the demineralized water supply system.

The chemical addition subsystem, provided to minimize piping corrosion and scale buildup, is manually controlled and is not safety related.

An air separator and an expansion tank are provided to accommodate expansion and contraction in the system due to variations in temperature.

The components of the system are located in the reactor building.

9.2.12.3.2.2 Component Description

Design data for major components of the Reactor Building Chilled Water System are listed in Table 9.2-17.

9.2.12.3.2.3 System Operation

The Reactor Building Chilled Water System operates continuously. One chilled water loop circulating pump, one evaporator chilled water circulating pump, one condenser water pump, and one chiller normally operate while the redundant units remain on automatic standby. However, if the cooling load exceeds the full capacity of one chiller, the standby chiller will automatically pick up a portion of the load.

The system is started by energizing a selected chilled water loop circulation pump. A preselected chiller, evaporator chilled water pump, and condenser cooling water pump are then automatically started in sequence.

The return chilled water is cooled by the operating chiller and recirculated through the chilled water supply loop to the air cooling coils.

Two temperature sensors in the chilled water main supply header modulate the compressor inlet guide vanes of their respective chillers to maintain constant chilled water supply temperature in the loop. A temperature controller maintains constant condenser cooling water outlet temperature by modulating butterfly valves on the cooling water supply, and return.

Three way valves are used for regulating the chilled water flow rate through the cooling coils except those in the drywell unit coolers and recirculation pump motor coolers, which are balanced for constant flow.

When the temperature of air upstream of any of the cooling coils in the Zone I and Zone III supply air unit drops below the set value of the temperature switches, this is annunciated as indication of failure of the upstream heating coil. Further drop of air temperature will be detected by another set of temperature switches which will initiate the draining of water in the cooling coils. There are two sets of unit coolers with seven unit coolers in each set for the drywell. Each set is on a separate piping loop, and each is sized for the normal load. This redundancy will ensure 100 percent cooling capacity for the drywell, if either loop fails. Two separate piping connections supply chilled water to recirculation pump motor coolers A and B.

On failure of both chillers, the flow valves on the chilled water connections to the drywell coolers close and the interconnecting valves on the Reactor Building Closed Cooling Water System open automatically to supply cooling water to the drywell units except on loss of offsite power when this transfer is performed manually after transfer of RBCCW heat exchanger to the ESW system. The flow valves in the Chilled Water System will return to their normal positions when one of the two chillers restores the supply of chilled water or the offsite power is restored.

During a DBA, there will be no chilled water or reactor building closed cooling water supply to the drywell, because the containment isolation valves will be closed.

During refueling operations, the chilled water available for the Zone III cooling coils is increased by shutting off the supply to the recirculation pump motor coolers. This will provide more comfortable working conditions for plant personnel on the refueling floor if plant shutdown occurs during periods of high outside air temperature.

9.2.12.3.3 Safety Evaluation

The operation of the Reactor Building Chilled Water System has no safety related function. The isolated portions of the containment are safety related as described in Subsection 6.2.4.

The chilled water piping inside the drywell has been examined to ensure that in the event of a SSE it will have no adverse effects on adjacent safety related equipment.

9.2.12.3.4 Test and Inspections

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

Provision is made for periodic inspection of major components to ensure the capability and integrity of the system. Local display devices are provided to indicate vital parameters (pressures and temperatures) required in testing and inspection.

9.2.12.3.5 Instrument Applications

Operation of the water chillers and chilled water pumps will be initiated manually from a local control station. Automatic standby operation of chillers and pumps is provided.

Indications are displayed at the local control panel to show operation status of chillers and pumps. Local pressure and temperature indicators display the appropriate system parameter associated with the chillers and pumps.

Chilled water flow into and out of the containment will be controlled by isolation valves that will close automatically on isolation signal or power failure at the valve. For complete discussion on containment isolation, refer to Section 6.2.4.

The following abnormal conditions are alarmed at the local control panel and retransmitted to the control room as a trouble alarm:

a) Failure of any pump

b) Failure of any chiller

c) High and low water level at the expansion tank

d) Low temperature at Zones I and III cooling coils

e) Drywell cooling coils outlet water high temperature

9.2.12.4 Radwaste Building Chilled Water System

9.2.12.4.1 Design Bases

The Radwaste Building Chilled Water System has no safety related function.

The Radwaste Building Chilled Water System is designed to supply chilled water at 48°F to six cooling coils in the supply air unit and to four coils in the off-gas area unit to maintain design ambient air temperatures in various areas of the radwaste building.

The system is designed to permit periodic inspection, testing, and maintenance of principal components with a minimum loss of normal operation.

Codes and standards applicable for the system are discussed in Table 3.2-1.

9.2.12.4.2 System Description

9.2.12.4.2.1 General Description

The system is common to both Units 1 and 2. It consists of two centrifugal water chillers, two evaporator chilled water circulating pumps, two condenser water pumps, two chilled water loop circulating pumps, an air separator, an expansion tank, a chemical addition tank, air cooling coils, interconnecting piping, and controls. Each chiller and pump is sized for 100 percent of nominal system capacity. The system is shown schematically on Figure 9.2-14.

Heat gained by the circulating chilled water is removed in the chiller evaporator refrigerant which in turn is cooled by the condenser cooling water.

Cooling water to the chiller condenser is provided from the normal Service Water System.

The makeup water supply comes through the manual valve provided in the makeup connection to the demineralized water system.

The chemical addition subsystem provided to minimize piping corrosion and scale buildup is manually controlled.

An air separator and an expansion tank are provided to accommodate expansion and contraction in the system due to temperature change.

The components of the system are located in the radwaste building.

9.2.12.4.2.2 Component Description

Design data for major components of the Radwaste Building Chilled Water System are listed in Table 9.2-18.

9.2.12.4.2.3 System Operation

A preselected chilled water circulating pump and a chiller with its associated evaporator and condenser water circulating pumps start automatically when the outside air temperature reaches 52°F. If the cooling load exceeds the full capacity of the operating chiller, the standby chiller will automatically pick up a portion of the load.

The system can be manually started by energizing a selected chilled water loop circulation pump. A preselected set that includes the chiller, evaporator chilled water pump, and condenser cooling water pump is then sequenced in automatically.

The return chilled water is cooled by the operating chiller and recirculated through the chilled water supply loop to the air cooling coils. Two temperature sensors located downstream of the chillers in the chilled water supply main modulate the compressor inlet guide vanes of their respective chillers to maintain constant chilled water supply temperature in the loop.

A temperature controller maintains constant condenser cooling water outlet temperature by modulating a three way valve between the cooling water supply and return line.

Three-way valves are also provided for regulating the chilled water flow rate through the air supply unit and off-gas area unit cooling coils.

When the temperature of air upstream of any of the air supply unit cooling coils drops below the set value of the temperature switches, this is annunciated at the local control panel as indication of failure of the upstream heating coil. Further drop of air temperature will be detected by another set of temperature switches, which will initiate the draining of water in the cooling coils to prevent freezing. Vacuum breakers, relief valves, and water connections are provided for draining and manual refilling purposes. Manual switches and valves are provided for coil drain system testing.

High and low levels of the expansion tank are annunciated locally and as a trouble alarm to the control room. The tank is filled, drained, and pressurized through manual valves.

9.2.12.4.3 Safety Evaluation

Because the Radwaste Building Chilled Water System has no safety design basis, no safety evaluation is provided. However, the system has features that ensure its reliable operation during plant normal operation. These features include redundant equipment such as chillers and pumps. Additional features include fail-safe positions on the system controls and equipment safety controls.

9.2.12.4.4 Tests and Inspections

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

Provision is made for periodic inspection of major components to ensure the capability and integrity of the system. Local display devices are provided to indicate vital parameters (pressures and temperatures) required in testing and inspections.

9.2.12.4.5 Instrument Applications

Operation of the chilled water system will be initiated manually from a local control panel. Automatic standby operation of chillers and pumps is provided.

Indications displayed on the local control panel give the operating status of chillers and pumps. Local pressure and temperature indicators display the appropriate system parameters associated with the chiller and pumps.

The following abnormal conditions are alarmed at the local control panel and retransmitted to the control room as a trouble alarm:

a) Failure of any pump

b) Failure of any chiller

c) High and low water level in the expansion tank

d) Low air temperature at the upstream side of the air supply unit cooling coils.

Table 9.2-1

LIST OF COOLERS SUPPLIED COOLING WATER BY THE SERVICE WATER SYSTEM

1. Generator Stator Coolers
2. Generator Hydrogen Coolers
3. Alterrex Air Coolers
4. Iso-Phase Bus Coolers
5. Gaseous Radwaste Recombiner CCW Heat Exchangers
6. Turbine Building CCW Heat Exchangers
7. Turbine Lube Oil Coolers
8. Reactor Feed Pump Turbine Lube Oil Coolers
9. Reactor Recirculation Pump M-G Set Fluid Coolers
10. Reactor Building CCW Heat Exchangers
11. Fuel Pool Heat Exchangers
12. Pipe Tunnel Coolers
13. Turbine Building Chillers
14. Reactor Building Chillers
15. Control Structure Chillers
16. Radwaste Building Chillers
17. Radwaste Evaporator Condensers
18. Radwaste Ambient Charcoal Chillers

TABLE 9.2-2

EMERGENCY SERVICE WATER SYSTEM FLOWS
(EACH LOOP) (TWO GENERATING UNITS)

Component	Plant Shutdown Flowrate (gpm)
1. Emergency Diesel Generator Heat Exchangers	4840
2. RHR Pump Seal Coolers	72
3. RHR Pump Motor Bearing Oil Coolers	24
4. RHR Pump Room Unit Coolers	504
5. Core Spray Pump Room Unit Coolers	160
6. HPCI Pump Room Unit Coolers (1)	112
7. RCIC Pump Room Unit Coolers (2)	80
8. Control Structure Chiller	740
9. Emergency Switchgear and Load Center Cooler	90
10. RBCCW Heat Exchanger	2800*
11. TBCCW Heat Exchanger	490*
12. Makeup to Fuel Pools	120
TOTAL	9920 (GPM (Division I)[3] 9952 GPM (Division II)[3]

[1] Division II (Loop B) only
[2] Division I (Loop A) only
[3] Maximum Demand Per Loop

*Non-essential service and on one loop only

TABLE 9.2-3

DEFINITION OF ESW FLOWS AND HEAT LOADS FOR UNITS 1 & 2

	Component	No. of Users Per Loop Unit 1	No. of Users Per Loop Unit 2	ESW Flow Per User (gpm)	Total ESW Flow Per Loop (gpm)	Cooling Duty Per User (x106 BTU/Hr.)	
1.	Standby diesel-generator htx	4 Common Total		1,210	4,840	9.555	
2.	RHR pump room unit cooler	2	2	126	504	.51	
3.	RHR pump motor bearing oil cooler	2	2	6	24	.05	
4.	RHR pump seal cooler	2	2	18	72	.1	
5.	Core spray pump room unit cooler	2	2	40	160	.2	
6.	HPCI pump room unit cooler	2	2	28	112(1)	.14	
7.	RCIC pump room unit cooler	2	2	20	80(2)	.100	
8.	Control structure chiller	1 Common per loop		740	740	3.7	
9.	Emergency switchgear & load center cooler	1	1	45	90	0.2	
*10.	RBCCW heat exchanger	1	1	1,400	2,800		
	a) drywell coolers					5.60	
	b) clean up recirc pump coolers					.7	7.3
	c) reactor recirc pump and motor coolers					1.0	
*11.	TBCCW heat exchanger	1	1	245	490	.04	
	a) CRD pump unit coolers						
12.	Makeup to fuel pools	1	1	60	120	No pond duty	
	Total Loop Flow				9,920 GPM (Division I)(3) 9,952 GPM (Division II)(3)		

(1) Division II (Loop B) Only.
(2) Division I (Loop A) Only.
(3) Maximum Demand Per Loop.
* On One Loop Only.

TABLE 9.2-4

ESW COOLING DUTY FOLLOWING DBA FOR MINIMUM HEAT TRANSFER CASE (LOSS OF ALL AUXILIARY POWER FOLLOWED BY A SINGLE UNIT LOCA)

Time		Operating Safeguards	LOCA Unit	S/D Unit		Cooling Duties On ESW	Number In Use LOCA Unit	Number In Use S/D Unit	Total ESW Cooling Duty ($X10^6$ BTU/Hr.)
0-10 Mins.	1)	RHR Pumps	4-LPCI	0	1)	Diesel-generator Hx	4 - Common		38.22
	2)	RHR Hx's	0	0	2)	Control Structure Cooler Chiller	1 - Common		3.7
	3)	CS Pumps	4	0	3)	Em. Switchgear and Load Center	1	1	0.400
	4)	ADS Valves	Closed	Open	4)	RBCCW Hx	-	0	0
	5)	RCIC Pumps	0	1	5)	TBCCW Hx	-	0	0
	6)	HPCI Pumps	0	1	6)	RHR Room Cooler	2	-	1.02
	7)	RHR SW Pumps	0	0	7)	RHR Motor Oil Cooler	4	-	.200
	8)	ESW Pumps	All 4 Run		8)	RHR Pump Seal Cooler	4	-	.400
	9)	Diesels	All 4 Run		9)	C.S. Room Cooler	2	-	.40
					10)	HPCI Room Cooler	-	1	.140
					11)	RCIC Room Cooler	-	1	.10
								Total	44.58
10-30 Mins.	1)	RHR Pumps	2-Cont Spray	1 Pool Cool	1)	Diesel-generator Hx	4 - Common		38.22
	2)	RHR Hx's	1	1	2)	Control Structure Cooler	1 - Common		3.7
	3)	CS Pumps	4	0	3)	Em. Switchgear and Load Center Cooler	1	1	0.400
	4)	ADS Valves	Closed	Open	4)	RBCCW Hx	-	0	0
	5)	RCIC Pumps	0	1	5)	TBCCW Hx	-	0	0
	6)	HPCI Pumps	0	0	6)	RHR Room Cooler	1	1	1.02
	7)	RHR SW Pumps	1	1	7)	RHR Motor Oil Cooler	2	1	.15
	8)	ESW Pumps	All 4 Run		8)	RHR Pump Seal Cooler	2	1	.30
	9)	Diesels	All 4 Run		9)	C.S. Room Cooler	2	-	.40
					10)	RCIC Room Cooler	-	1	.100
								Total	44.29

TABLE 9.2-4 (cont'd)

ESW COOLING DUTY FOLLOWING DBA FOR MINIMUM HEAT TRANSFER CASE (LOSS OF ALL AUXILIARY POWER FOLLOWED BY A SINGLE UNIT LOCA)

Time	Operating Safeguards	LOCA Unit	S/D Unit	Cooling Duties On ESW	Number In Use LOCA Unit	Number In Use S/D Unit	Total ESW Cooling Duty ($X10^6$ BTU/Hr.)
30 Mins.- Approx.3 hrs.	1) RHR Pumps	2-Cont Spray	1 Pool Cool	1) Diesel-generator Hx	4 - Common		38.22
	2) RHR Hx's	1	1	2) Control Structure Chiller	1 - Common		3.7
	3) CS Pumps	2	0	3) Em. Switchgear and Load Cooler	1	1	.400
	4) ADS Valves	Closed	Open	4) RBCCW Hx	-	0	0
	5) RCIC Pumps	0	1	5) TBCCW Hx	-	0	0
	6) HPCI Pumps	0	0	6) RHR Room Cooler	1	1	1.02
	7) RHR SW Pumps	1	1	7) RHR Motor Oil Cooler	2	1	.15
	8) ESW Pumps	All 4 Run		8) RHR Pump Seal Cooler	2	1	.30
	9) Diesels	All 4 Run		9) C.S. Room Cooler	1	-	.20
				10) RCIC Room Cooler	-	1	.100
						Total	44.09
Approx.3 hrs.- 24 hrs.	1) RHR Pumps	2-Cont Spray	1 Shutdown Cool	1) Diesel-generator Hx	4 - Common		38.22
	2) RHR Hx's	1	1 Shutdown Cool	2) Control Structure Chiller	1 - Common		3.7
	3) CS Pumps	2	0	3) Em. Switchgear and Load Center Cooler	1	1	0.400
	4) ADS Valve	Closed	Closed	4) RBCCW Hx	-	1	7.300
	5) RCIC Pumps	0	0	5) TBCCW Hx	0	1	.04
	6) HPCI Pumps	0	0	6) RHR Room Cooler	1	1	1.02
	7) RHR SW Pumps	1	1	7) RHR Motor Oil Cooler	2	1	.150
	8) ESW Pumps	All 4 Run		8) RHR Pump Seal Cooler	2	1	.300
	9) Diesel	All 4 Run		9) C.S. Room Cooler	1	-	.20
						Total	51.33

TABLE 9.2-4 (cont'd)

ESW COOLING DUTY FOLLOWING DBA FOR MINIMUM HEAT TRANSFER CASE (LOSS OF ALL AUXILIARY POWER FOLLOWED BY A SINGLE UNIT LOCA)

Time		Operating Safeguards	LOCA Unit	S/D Unit		Cooling Duties On ESW	Number In Use LOCA Unit	Number In Use S/D Unit	Total ESW Cooling Duty ($X10^6$ DTU/Hr.)
24 hrs. - 30 days	1)	RHR Pumps	2-Cont Spray	1 Shutdown Cool	1)	Diesel-generator Hx	3 - Common		28.665
	2)	RHR Hx's	1-Cont Spray	1 Shutdown Cool	2)	Control Structure Chiller	1 - Common		3.7
	3)	CS Pumps	2	0	3)	Em. Switchgear and Load Center Cooler	1	1	0.400
	4)	ADS Valve	Closed	Closed	4)	RBCCW Hx	-	1	7.300
	5)	RCIC Pumps	0	0	5)	TBCCW Hx	1	1	.04
	6)	HPCI Pumps	0	0	6)	RHR Room Cooler	1	1	1.02
	7)	RHR SW Pumps	1	1	7)	RHR Motor Oil Cooler	1	1	.15
	8)	ESW Pumps	3 Run		8)	RHR Pump Seal Cooler	1	1	.30
	9)	Diesel	3 Run		9)	C.S. Room Cooler	2	-	.20
								Total	41.775

*It is assumed that after four hours the shutdown unit RBCCW and TBCCW Heat Exchangers are manually brought into service to assist the shutdown operations.

TABLE 9.2-5

ESW COOLING DUTY FOLLOWING DBA FOR MAXIMUM WATER LOSS CASE (LOSS OF ALL AUXILIARY POWER FOLLOWED BY A SINGLE UNIT LOCA)

Time	Operating Safeguards	LOCA Unit	S/D Unit	Cooling Duties On ESW	Number In Use LOCA Unit	Number In Use S/D Unit	Total ESW Cooling Duty ($X10^6$ BTU/Hr.)
0-10 Mins.	1) RHR Pumps	4-LPCI	0	1) Diesel-generator Hx	4 - Common		38.22
	2) RHR Hx's	0	0	2) Control Structure Cooler Chiller	1 - Common		3.7
	3) CS Pumps	4	0	3) Em. Switchgear and Load Center	1	1	0.400
	4) ADS Valves	Closed	Open	4) RBCCW Hx	-	-	0
	5) RCIC Pumps	0	1	5) TBCCW Hx	-	-	0
	6) HPCI Pumps	0	1	6) RHR Room Cooler	2	-	1.02
	7) RHR SW Pumps	0	0	7) RHR Motor Oil Cooler	4	-	.200
	8) ESW Pumps	All 4 Run		8) RHR Pump Seal Cooler	4	-	.400
	9) Diesels	All 4 Run		9) C.S. Room Cooler	2	-	.4
				10) HPCI Room Cooler	-	1	.140
				11) RCIC Room Cooler	-	1	.1
						Total	44.58
10-30 Mins.	1) RHR Pumps	4-Cont Spray	2 Pool Cool	1) Diesel-generator Hx	4 - Common		38.22
	2) RHR Hx's	2	2	2) Control Structure Cooler	1 - Common		3.7
	3) CS Pumps	4	0	3) Em. Switchgear and Load Center Cooler	1	1	0.400
	4) ADS Valves	Closed	Open	4) RBCCW Hx	-	-	0
	5) RCIC Pumps	0	1	5) TBCCW Hx	-	-	0
	6) HPCI Pumps	0	0	6) RHR Room Cooler	2	2	2.04
	7) RHR SW Pumps	2	2	7) RHR Motor Oil Cooler	4	2	.3
	8) ESW Pumps	All 4 Run		8) RHR Pump Seal Cooler	4	2	.6
	9) Diesels	All 4 Run		9) C.S. Room Cooler	2	-	.4
				10) RCIC Room Cooler	-	1	.100
						Total	45.76

TABLE 9.2-5 (cont'd)

ESW COOLING DUTY FOLLOWING DBA FOR MAXIMUM WATER LOSS CASE (LOSS OF ALL AUXILIARY POWER FOLLOWED BY A SINGLE UNIT LOCA)

Time	Operating Safeguards	LOCA Unit	S/D Unit	Cooling Duties On ESW	Number In Use LOCA Unit	Number In Use S/D Unit	Total ESW Cooling Duty ($X10^6$ BTU/Hr.)
30 Mins.- 6 hrs.	1) RHR Pumps	4-Cont Spray	2 Pool Cool	1) Diesel-generator Hx	4 - Common		38.22
	2) RHR Hx's	2	2	2) Control Structure Chiller	1 - Common		3.7
	3) CS Pumps	2	0	3) Em. Switchgear and Load Cooler	1	1	.400
	4) ADS Valves	Closed	Open	4) RBCCW Hx	-	-	0
	5) RCIC Pumps	0	1	5) TBCCW Hx	-	-	0
	6) HPCI Pumps	0	0	6) RHR Room Cooler	2	2	2.04
	7) RHR SW Pumps	2	2	7) RHR Motor Oil Cooler	4	2	.3
	8) ESW Pumps	All 4 Run		8) RHR Pump Seal Cooler	4	2	.6
	9) Diesels	All 4 Run		9) C.S. Room Cooler	1	-	.2
				10) RCIC Room Cooler	-	1	.100
						Total	45.56
6 hrs.- 24 hrs.	1) RHR Pumps	4-Cont Spray	2	1) Diesel-generator Hx	4 - Common		38.22
	2) RHR Hx's	2	1 Shutdown Cool 1	2) Control Structure Chiller	1 - Common		3.7
	3) CS Pumps	2	0 Pool Cool	3) Em. Switchgear and Load Center Cooler	1	1	0.400
	4) ADS Valve	Closed	Closed	4) RBCCW Hx	-	1	7.300
	5) RCIC Pumps	0	0	5) TBCCW Hx	-	1	.04
	6) HPCI Pumps	0	0	6) RHR Room Cooler	2	2	2.04
	7) RHR SW Pumps	2	2	7) RHR Motor Oil Cooler	4	2	.3
	8) ESW Pumps	All 4 Run		8) RHR Pump Seal Cooler	4	2	.6
	9) Diesel	All 4 Run		9) C.S. Room Cooler	1	-	.2
						Total	52.8

TABLE 9.2-5 (cont'd)

ESW COOLING CUTY FOLLOWING DBA FOR MAXIMUM WATER LOSS CASE (LOSS OF ALL AUXILIARY POWER FOLLOWED BY A SINGLE UNIT LOCA)

Time		Operating Safeguards	LOCA Unit	S/D Unit		Cooling Duties On ESW	Number In Use LOCA Unit	Number In Use S/D Unit	Total ESW Cooling Duty ($X10^6$ DTU/Hr.)
24 hrs. - 30 days	1)	RHR Pumps	2-Cont Spray	1 Shutdown Cool	1)	Diesel-generator Hx	3 - Common		28.665
	2)	RHR Hx's	1-Cont Spray	1 Shutdown Cool	2)	Control Structure Chiller	1 - Common		3.7
	3)	CS Pumps	2	0	3)	Em. Switchgear and Load Center Cooler	1	1	0.400
	4)	ADS Valve	Closed	Closed	4)	RBCCW Hx	-	1	7.300
	5)	RCIC Pumps	0	0	5)	TBCCW Hx	-	1	.04
	6)	HPCI Pumps	0	0	6)	RHR Room Cooler	1	1	1.02
	7)	RHR SW Pumps	1	1	7)	RHR Motor Oil Cooler	2	1	.15
	8)	ESW Pumps	3 Run		8)	RHR Pump Seal Cooler	2	1	.30
	9)	Diesel	3 Run		9)	C.S. Room Cooler	1	-	.20
								Total	41.775

*It is assumed that after four hours the shutdown unit RBCCW and TBCCW Heat Exchangers are manually brought into service to assist the shutdown operations.

TABLE 9.2-20

EMERGENCY SERVICE WATER SYSTEM HEAT LOADS (BOTH UNITS)

(X10& Btu/hr)

I. Minimum Heat Transfer Case (One Spray Division for 30 days, second Division on bypass for 30 min.)

Time	Loop A	Loop B	Diesels Operating
0-10 min.	43.43	1.15	4
10-30 min.	43.43	1.67	4
30 min.-4 hrs.	44.09	1.47	4
4 hrs.-1 day	51.33	1.47	4
1 day-30 days	41.78	0	3

II. Maximum Water Loss Case

Time	Loop A	Loop B	Diesels Operating
0-10 min.	43.43	1.15	4
10-30 min.	43.43	1.67	4
30 min.-4 hrs.	44.09	1.47	4
4 hrs.-1 day	51.33	1.47	4
1 day-30 days	41.78	0	3

TABLE 9.2-21

RHR AND RHR SERVICE WATER SYSTEM FLOW RATES
MINIMUM HEAT TRANSFER CASE

Time	Parameter	LOCA	S/D
10-30 min.	Core Spray (2 loops)	12,700	0
	Safety Relief valves & RCICS	0	(1)
	RHRHX (shell side) Flow, Loop A	11,260	10,000
	RHRHX (shell side) Flow, Loop B	0	0
	RHRHX (tube side) Flow, Loop A	9,000	9,000
	RHRHX (tube side) Flow, Loop B	0	0
30-240 min.	Core Spray (1 loop)	6,350	0
	RHRHX (tube side) Flow, Loop A	11,260	9,000
	RHRHX (tube side) Flow, Loop B	0	0
	RHRHX (shell side) Flow, Loop A	9,000	10,000
	RHRHX (shell side) Flow, Loop B	0	0
4 hrs-30 days	Core Spray (1 loop)	6,350	0
	RHRHX (tube side) Flow, Loop A	11,260	9,000
	RHRHX (tube side) Flow, Loop B	0	0
	RHRHX (shell side) Flow, Loop A	9,000	10,000
	RHRHX (shell side) Flow, Loop B	0	0

NOTE (1) Flow Rate Determined by cooldown rate.

(2) LOCA Unit RHR Flow Rates denote containment spray mode of operation.

(3) S.D. Unit RHR Flow Rates denote suppression pool cooling mode of operation for approximately the first three hours of the transient followed by shutdown cooling mode of operation for the remainder of the thirty-day period.

TABLE 9.2-21a

RHR AND RHR SERVICE WATER SYSTEM FLOW RATES
MAXIMUM WATER LOSS CASE

Time	Parameter	LOCA	S/D
10-30 min.	Core Spray (2 loops)	12,700	0
	Safety Relief valves & RCICS	NA	(1)
	RHRHX (shell side) Flow, Loop A	11,260	10,000
	RHRHX (shell side) Flow, Loop B	11,260	10,000
	RHRHX (tube side) Flow, Loop A	9,000	9,000
	RHRHX (tube side) Flow, Loop B	9,000	9,000
30-360 min.	Core Spray (1 loop)	6,350	0
	RHRHX (tube side) Flow, Loop A	9,000	9,000
	RHRHX (tube side) Flow, Loop B	9,000	9,000
	RHRHX (shell side) Flow, Loop A	11,260	10,000
	RHRHX (shell side) Flow, Loop B	11,260	10,000
6-24 hrs.	Core Spray (1 loop)	6,350	0
	RHRHX (tube side) Flow, Loop A	9,000	9,000
	RHRHX (tube side) Flow, Loop B	9,000	9,000
	RHRHX (shell side) Flow, Loop A	11,260	10,000
	RHRHX (shell side) Flow, Loop B	11,260	10,000
1-30 days	Core Spray (1 loop)	6,350	0
	RHRHX (tube side) Flow, Loop A	9,000	9,000
	RHRHX (tube side) Flow, Loop B	0	9,000
	RHRHX (sheel tube) Flow, Loop A	11,260	10,000
	RHRHX (sheel side) Flow, Loop B	0	10,000

NOTE (1) Flow Rate Determined by cooldown rate.

(2) LOCA Unit RHR Flow Rates denote containment spray modes of operation.

(3) S.D. Unit RHR Flow Rates denote two divisions of suppression pool cooling operation for approximately the first three hours of the transient followed by one division of shutdown cooling operation and one division of pool cooling operation for the remainder of the thirty days.

TABLE 9.2-22

EMERGENCY SERVICE WATER SYSTEM FLOW RATES
(MHT AND MWL CASES)

Time	Loop A	Loop B
0-4 hrs.	6,510	962
4 hrs.-1 day	8,155	962
1-30 days	7,065	481

TABLE 9.2-25

INPUT PARAMETERS FOR DRIFT LOSS

Drop Size Spectrum

From Spray Engr. Co. "Droplet Size Spectrum Tests Report"[3]

Dia.	Volume Fraction,%
200	0.05
260	0.05
300	0.10
330	0.10
365	0.10
400	0.10
425	0.20
460	0.30
520	0.40
580	0.60
640	2.00
855	3.00
1000	3.00
1190	5.00
1340	5.00
1650	10.00
2000	10.00
2290	10.00
2800	20.00
3600	15.00
4000	15.00

Nozzle Distance from Perimeter of Spray Pond in Feet

a	60
b	67
c	75.75
d	82.75
e	91.50
f	98.50
q	107.25
h	114.25
i	123.00
j	130.00
k	138.75
l	145.75
m	154.50
n	161.50
0	170.25
p	177.25

TABLE 9.2-27

SIZE COMPARISON OF SPRAY POINDS INVESTIGATED

	Canadys	SSES	Rancho Seco
Water Flow Rate, gpm	180,000	55,968 (max)	16,500
Nozzles	1,800 @ 100 gpm	1,056 @ 53 gpm	304 @ 53 gpm
Water Pressure at Nozzle, psig	7	7	7
Pond Length, feet	-	1,250	330
Width, feet		425to225	165
Depth, feet	10	10.5	5
Volume (approx.) $\times 10^6$ gallons	32.4	25.0	5.7
Area Acres	10	8	1.3

TABLE 9.2-29

PERFORMANCE COMPARISON OF RANCHO SECO TEST RESULTS AND MODEL RESULTS

Date	Time	TWB,°F	TDB,°F	W.S.,MPH	T,°F	Rancho Seco % Efficiency	Model Prediction Efficiency
5/19/73	1424	61.0	81.5	13.0	79.8	41.4	41.35
5/19/73	1615	61.5	81.0	12.5	80.0	47.1	42.62
5/20/73	0400	51.0	55.0	4.9 (5.3)*	77.4	33.9	27.12
5/20/73	0630	48.5	52.0	2.8	77.4	28.8	22.95
5/20/73	1000	56.6 (56.5)*	65.0	6.2 (6.0)*	77.6	30.8	29.65
5/20/73	1148	57.5	71.0	6.2 (6.5)*	78.6	34.7	31.52
5/18/73	1500	72.4	95.0	7.0	80.0	38.3	37.46
5/18/73	1700	69.7	93.0	6.5 (6.6)*	81.1 (81.2)*	36.6	36.55
5/18/73	1900	66.6	85.7	8.5 (8.4)*	80.7	50.8	38.90
5/18/73	2200	60.9	72.3	3.4 (3.8)*	80.1 (80.3)*	33.6	25.56
7/19/73	2300	60.2	69.3	2.6 (3.8)*	79.7	28.5	25.66
5/20/73	2300	54.2	58.0	1.0	101.4	37.4	30.89
5/20/73	2330	53.0	57.0	1.2 (1.6)*	98.5 (100.0)*	36.5	34.13
5/20/73	2400	52.0	56.0	1.2 (1.3)*	97.8 (97.9)*	38.3	31.49
5/20/73	0330	49.0	53.0	1.4 (1.0)*	101.7	36.7	29.75
5/20/73	0415	48.0	51.0	0.80 (0.59)*	97.4	36.1	26.04

TABLE 9.2-30

HEAT RATE BTU'S/HR X 10+6

&TIME	DECAY HT RATE	AUXILIARY RATE	SENSIBLERATE*	TOTAL RATE
11.000	6.37800+08	4.42908+07	-5.51298+08	1.31051+08
15.000	5.99844+08	4.42908+07	-5.06106+08	1.38306+08
20.000	5.20392+08	4.42908+07	-4.19454+08	1.45520+08
30.000	4.67892+08	4.42908+07	-3.62748+08	1.49729+08
40.000	4.27596+08	4.42908+07	-3.12948+08	1.59758+08
50.000	3.96276+08	4.42908+07	-2.71560+08	1.69348+08
60.000	3.71748+08	4.42908+07	-2.38314+08	1.78075+08
90.000	3.24204+08	4.42908+07	-1.69146+08	1.99748+08
120.00	2.97816+08	4.42908+07	-1.25052+08	2.17484+08
150.00	2.80896+08	4.42908+07	-9.30900+07	2.32564+08
180.00	2.68428+08	4.42908+07	7.47840+07	3.88275+08
240.00	2.49456+08	4.42908+07	2.17026+07	3.16081+08
300.00	2.34336+08	5.13300+07	9.49320+06	2.95750+08
360.00	2.21592+08	5.13300+07	8.72880+06	2.82213+08
480.00	2.01324+08	5.13300+07	1.28790+07	2.66061+08
600.00	1.86204+08	5.13300+07	1.56972+07	2.53736+08
720.00	1.74744+08	5.13300+07	1.65240+07	2.43085+08
900.00	1.62240+08	5.13300+07	1.54728+07	2.29498+08
1200.0	1.48704+08	5.13300+07	1.17804+07	2.12234+08
1440.0	1.41372+08	5.13300+07	9.39300+06	2.02504+08
1800.0	1.33128+08	4.15500+07	6.38520+06	1.81422+08
2160.0	1.26564+08	4.15500+07	4.36842+06	1.72825+08
2880.0	1.15900+08	4.15500+07	3.41928+06	1.61192+08
4320.0	9.99288+07	4.15500+07	3.15408+06	1.44919+08
5760.0	8.86488+07	4.15500+07	2.69976+06	1.33166+08
7200.0	8.04516+07	4.15500+07	2.03448+06	1.24288+08
8640.0	7.43544+07	4.15500+07	1.41624+06	1.17554+08
11520.	6.59820+07	4.15500+07	9.75240+05	1.08727+08
14400.	6.04692+07	4.15500+07	6.48660+05	1.02875+08
21600.	5.18244+07	4.15500+07	3.90606+05	9.39528+07
28800.	4.60548+07	4.15500+07	2.22810+05	8.80038+07
36000.	4.15992+07	4.15500+07	1.56222+05	8.34714+07
43200.	3.80028+07	4.15500+07	92070.	7.98042+07

* Negative numbers indicate sensible heat is being stored and positive numbers indicate sensible heat is being released.

&Time in min.

9.3 PROCESS AUXILIARIES

9.3.1 COMPRESSED AIR SYSTEMS

9.3.1.1 Instrument Air System

9.3.1.1.1 Design Bases

The instrument air system has no safety related function. The instrument air system is designed to provide a continuous supply of filtered, dry, and oil free air for all pneumatic instruments and controls in the plant except those described in Subsections 9.3.1.4 and 9.3.1.5.

Each compressor unit is powered from a separate electrical bus.

Codes and standards applicable to the compressed air system are listed in Table 3.2-1. The compressed air system is designed and constructed in accordance with quality group D specifications.

9.3.1.1.2 System Description

General Description

Units 1 and 2 instrument air systems are identical. The following discussion is for the Unit 1 system. The system includes two identical 100 percent capacity trains consisting of an air intake filter silencer, a compressor unit, an aftercooler, an air receiver, two parallel prefilters, a dryer unit, and two parallel afterfilters. The two trains are connected in parallel by a common header that branches into the instrument air subsystems. All of the above components and a common alarm and control panel for each unit are located in the turbine building. The system is shown schematically in Figure 9.3-1.

The major components and the design data of the compressed air system are presented in Table 9.3-1.

System Operation

During normal unit operation, one of the two instrument air compressors will be selected as the lead compressor for continuous operation, automatically loaded or unloaded in response to the instrument air system demand. The other instrument air compressor will serve as a standby. The standby compressor will start automatically if the lead compressor fails, or if its continuous operation cannot meet the instrument air system demand.

The service air compressors serve as backup to the instrument air compressors. The interconnecting valves between the two systems must be opened manually. The Unit 2 instrument air system also serves as backup to the entire Unit 1 system and vice versa. The interconnecting valves must be operated manually.

The Unit 1 system supplies instrument air to common areas such as the control structure, radwaste building, diesel generator building, circulating water pump house, and chlorination building.

9.3.1.1.3 Safety Evaluation

Most instruments required for the operation of engineered safety features are operated electrically. Instrument air operated components, which are essential for the safe shutdown of the plant, are designed to assume the safe position upon loss of air pressure. Their energy source for safety operation is not the nonsafety related instrument air. The list of such components is shown in Table 9.3-2. The operation of the containment isolation valves is described in Subsection 6.2.4. For a failure mode and effect analysis, See Table 9.3-3.

The compressed air system is switched automatically to the standby ac power supply during a loss of offsite power. Both Unit 1 and 2 compressors are tripped off the standby ac power source upon receiving a "LOCA signal" from either operating unit coincident with loss of offsite power. One or more compressors may be restarted manually when the "LOCA signal" is no longer present.

9.3.1.1.4 Tests and Inspections

The compressors, aftercoolers, receivers, prefilters, dryer units, afterfilters, and the control panel are shop inspected and tested. Air compressors and associated components on standby are checked and operated periodically. Air filters are periodically inspected for cleanliness, and the desiccant in the dryer units is sampled every six months to verify its useful life. If the sample is dark brown or black in color, the desiccant is replaced.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.3.1.1.5 Instrumentation Application

Instrumentation is provided for each instrument air compressor train to monitor and automatically control each compressor's operation.

Switches monitoring the following parameters alarm and trip their respective compressor: oil low pressure, oil high temperature, discharge air high temperature, cooling water high temperature, cooling water low pressure, and intercooler separator liquid high level. Instrumentation temperature is indicated locally on the compressor discharge and the moisture separator. Local instrument air pressure is indicated on the air receiver. Malfunction of the operating compressor and pressure loss in the main header will be annunciated in the control room. All of the compressor controls, including start-stop and load-unload are furnished in the local panel. Hand switches for starting or stopping the units are provided in the control room panel.

The instrument air dryer package has the following controls and instrumentation. The prefilters and afterfilters each have pressure differential indicators. Each of the dryer towers has a pressure indicator, temperature indicator, and high and low temperature switches for heating element control and low temperature alarm.

9.3.1.2 Service Air System

9.3.1.2.1 Design Bases

The service air system has no safety-related functions. It is designed to provide compressed air for service air outlets located throughout the plant and as a backup system for instrument air.

Codes and standards applicable to the service air system are listed in Table 3.2-1. The compressed air system is designed and constructed in accordance with quality group D specifications.

9.3.1.2.2 System Description

9.3.1.2.2.1 General Description

The Unit 1 system includes two identical 100 percent capacity air compressing trains, each consisting of an air intake filter/silencer, a compressor unit, an aftercooler and an air receiver. The two trains are connected in parallel by a common header that branches into the service air subsystems. All of the above components with their common alarm and control panels are in the turbine building. The system is shown schematically on Figure 9.3-2.

The Unit 2 system includes only a single compressing train identical to those of Unit 1. The system is schematically shown on Figure 9.3-3. The Unit 2 system serves as backup to the Unit 1 system and vice versa. The interconnecting valves must be operated manually.

The major components and their design data of the service air system are presented in Table 9.3-4.

System Operation

During normal plant operation of Unit 1, one of the two compressors will be selected as the lead compressor and will operate, automatically, being loaded or unloaded in response to the service air system pressure. The other service air compressor will serve as a standby. The standby compressor will start automatically if the lead compressor fails, or if its continuous operation cannot meet the service air system demand.

The Unit 2 compressor normally runs automatically and is loaded or unloaded in response to service air system demand. When it fails, Unit 1 standby compressor will be manually started to supply Unit 2 service air demand after manually opening the valve interconnecting the two systems.

9.3.1.2.3 Safety Evaluation

As the service air system has no safety design basis, no safety evaluation is provided.

9.3.1.2.4 Tests and Inspections

The compressors, aftercoolers, receivers, and the control panels are shop inspected and tested prior to installation. Air compressors and associated components on standby are checked and operated periodically. Air filters are periodically inspected for cleanliness.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.3.1.2.5 Instrumentation Application

Instrumentation is provided for each service air compressor train to monitor and automatically control each compressor's operation.

Switches monitoring the following parameters alarm and trip their respective compressor: lubricator failure, oil low pressure, oil high temperature, discharge air high temperature, cooling water high temperature, cooling water low pressure, and intercooler separator liquid high level. Local temperature indicators are provided in the compressor discharge. Local pressure indication is located on the service air receiver. Low pressure of the service air header is indicated and annunciated in the control room. All of the compressor controls, including start-stop and load-unload, are furnished in the local panel.

Hand switches for starting or stopping the units are provided in the control room panel. There is also a common trouble alarm in the control room panel.

9.3.1.3 Radwaste Building Low Pressure Air System

9.3.1.3.1 Design Bases

The radwaste building low pressure air system has no safety related function.

The radwaste building low pressure air system is designed to provide filtered oil free low pressure compressed air for the liquid radwaste filters and the liquid radwaste demineralizer, as these processes require. This system serves also as a backup source of air for the cement silo should its integral blower fail.

Codes and standards applicable to the low pressure air system are listed in Table 3.2-1. The low pressure air system is designed and constructed in accordance with quality group D requirements.

9.3.1.3.2 System Description

General Description

The system includes two intake filter silencers, one compressor, one aftercooler with moisture separator, and one air receiver. The system is shown schematically on Figure 9.3-4.

The major components and their design data are presented in Table 9.3-5.

System Operation

The system operates intermittently based on air demand from the liquid radwaste processing system equipment. The demand on the system will be as follows:

a) Radwaste filters cake drying: 30 min to 1 hr duration - four times a week, quantity 700 scfm

b) Demineralizer - 20 min duration, once a year, demand 325 scfm

c) Cement silo - 35 scfm. This demand will occur only if the cement silo blower fails.

Generally, not more than one of the above four demands will occur at one time.

The compressor has an auto dual capacity control system. During periods when air is being used, the compressor runs and is loaded or unloaded automatically in response to the system pressure. When the demand for air ceases, the compressor stops automatically after a set time interval, restarts, and loads and unloads again if the demand resumes. The compressor can be started manually from the radwaste building control room or from the local panel mounted on the compressor skid.

The system is common to both Units 1 and 2.

There is no standby provision because of the intermittent operation of the system.

Any abnormal operating condition of the low pressure air system will be annunciated in the radwaste control room on panel OC-301.

9.3.1.3.3 Safety Evaluation

The low pressure compressed air system has no safety related function and no safety evaluation is provided.

9.3.1.3.4 Tests and Inspections

The compressor, aftercooler, receiver, and the control panel are shop inspected and tested prior to installation.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.3.1.3.5 Instrumentation Applications

Instrumentation is provided for the low pressure air compressor train to monitor and automatically control the compressor's operation.

Switches monitoring the following parameters alarm and trip the compressor: oil low pressure, oil high temperature, discharge air high temperature, cooling water high temperature, and cooling water low pressure. Local temperature indication is on the compressor discharge and the moisture separator. Local pressure indication is on the low pressure air receiver. All of the compressor controls, ie, start-stop and load-unload, are furnished in the compressor package.

9.3.1.4 River Intake Structure Compressed Air System

9.3.1.4.1 Design Bases

The river intake structure compressed air system has no safety-related function. The river intake structure compressed air system provides a continuous supply of dry, filtered, oil free air for pneumatic instruments and controls and for limited service air use inside the river intake structure building.

Codes and standards applicable to the compressed air system are shown in Table 3.2-1. The compressed air system is designed and constructed in accordance with quality group D specifications.

9.3.1.4.2 System Description

General Description

The river intake structure compressed air system is common to both Units 1 and 2. The system includes two identical duplex air compressor packages and a on dryer unit. Each duplex unit contains two identical compressors with motors and aftercoolers; a common air receiver, instrumentation, controls, interconnecting piping, and electrical wiring. The duplex air compressor packages are connected in parallel by a common header from which service air can be drawn for limited maintenance use. The dryer unit is also fed from this common header. The individual instrument air branch connections are tapped from the dryer outlet header. All of the above components are located in the river intake structure. The system is shown schematically on Figure 9.3-4.

The major components and their design data are presented in Table 9.3-6.

System Operation

During normal plant operation, one of the two duplex units will be selected as the lead unit for continuous operation. The two compressors within the duplex will alternate, with each compressor automatically stored or stopped in response to the instrument air system pressure. The other duplex unit will serve as a standby. The standby unit will start automatically if the lead unit fails, or if its continuous operation cannot meet the air system demand.

9.3.1.4.3 Safety Evaluation

The river intake structure compressed air system has no safety function.

For plant availability purposes, there is a redundancy on the compressor train. Failure of this system will not endanger the operation of any safety related instruments and controls.

9.3.1.4.4 Tests and Inspections

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.3.1.4.5 Instrumentation Application

Instrumentation is provided for each compressor of the duplex unit to monitor and automatically control its operation. A low cylinder oil level switch trips the compressor on low oil level. Local pressure indication is provided on each air receiver. All of the compressor controls, ie, start-stop, load-unload, are furnished in each compressor package.

The river intake structure air dryer packages have local temperature and pressure indicators and controls for automatic operation. A failure-to-cycle alarm is provided in the dryer unit panel.

The air header low pressure switch alarms locally and starts the standby duplex compressor set. A low-low pressure switch annunciates locally if the standby set cannot maintain system pressure. Any trouble alarm in the local panel is transmitted to the control room as a common alarm.

9.3.1.5 Containment Instrument Gas System

9.3.1.5.1 Design Basis

Unit 1 and Unit 2 containment instrument gas systems are identical. The Unit 1 system is described in this Subsection.

The containment instrument gas system is designed to provide filtered, dry, oil-free instrument gas to the pneumatic devices located inside the drywell and suppression chamber.

Portions of the containment instrument gas system are safety related as shown in Figure 9.3-5.

The safety related portions included are containment penetrations, the emergency backup nitrogen storage system and the gas distribution piping to the six main steam relief valves that are part of the Automatic Depressurization System (ADS).

The system provides instrument gas at 150 psig for the safety related main steam relief valves with ADS function and at 90 psig for all other pneumatic devices inside the containment. The safety related backup nitrogen storage system is maintained at 2200 psig.

The safety related nitrogen storage system contains adequate gas in storage for 30 days after a postulated DBA.

The normal supply of compressed gas is not safety related; however, it has 100 percent redundancy on its major components.

Each compressor unit is powered from a separate electrical bus.

Codes and standards applicable to the instrument gas systems are listed in Table 3.2-1. All of the instrument gas systems except the safety related portions are designed and constructed in accordance with quality group D specifications. The safety related portions are designed and constructed to quality group C specifications except the storage bottles, and connection fittings. These storage bottles conform to Department of Transportation (DOT) Standards, Title 49, Section 178.37, Specification 3AA and ICC Specifications. The connection fittings are standard stainless steel tubing connection assemblies.

Although these bottles provide the gas supply for the safety related function, such bottles complete with shut off valves and connection fittings are not readily available as Q listed and N stamped. However, the manufacturing and testing of these tanks conform to DOT and ICC standards, which are in excess of those required by ASME Section III, Class 3.

9.3.1.5.2 System Description

General Description

Containment instrument gas is a recyling system that, for normal operation, takes suction from the drywell atmosphere.

For normal operation the system includes one intake screen filter, one inlet moisture separator, two inlet gas filters, two full capacity gas compressors, two gas aftercoolers with moisture separators, two gas receivers, two gas dryer systems, two outlet gas filters, two pressure reducing stations, one instrument gas accumulator, associated piping, valves, controls and instruments.

For emergency operation, the system includes two loops of high pressure bottled nitrogen. Each loop consists of nitrogen storage bottles, pigtails, station valves, manifold, shut off valve, two stage regulator, and other instruments and controls as shown on Figure 9.3-5.

The system is shown schematically on Figure 9.3-5. Table 9.3-7 is a list of pneumatically operated devices in the Containment Instrument Gas System. Some of these valves are required for safe shutdown, and they assume the safe position in the event of a loss of instrument gas pressure. Some of these valves have individual safety related accumulators with redundant normal compressed gas supply. Valves designated for ADS function have safety related individual accumulators with redundant normal compressed gas supply and emergency backup supply.

The major components of the instrument gas system and their design data are presented in Table 9.3-8.

System Operation

During normal unit operation, the compressor controls are designed to permit automatic start and stop operation of one or two compressors in response to system demand. One compressor normally provides the instrument gas needs. The second compressor serves as a standby for abnormal instrument gas demands.

When the normal gas pressure in the piping headers leading to the ADS function relief valves falls below 150 psig because of the failure of both of the compressors or because of containment isolation, the high pressure (2200 psig) nitrogen storage bottles automatically provide instrument gas at 150 psig to the ADS function main steam relief valves. Instrument gas pressure from the storage bottles is reduced to 150 psig for transmission to the ADS function relief valve accumulators.

9.3.1.5.3 Safety Evaluation

Failure of the nonsafety related portions of the compressed gas system does not impair the operation of Engineered Safety Feature (ESF) Systems or the integrity of containment isolation during the accidents described in Chapter 15. For a failure mode and effect analysis, see Table 9.3-9.

Pneumatically operated devices, which are essential for the safe shutdown of the plant, are designed to operate in the safe position upon loss of gas pressure or they are provided with individual accumulators and/or a backup source of safety related high pressure nitrogen gas.

The compressed gas system's nonsafety related compressors are switched automatically to standby electrical power. These compressors are tripped of the standby ac power source upon receiving a LOCA signal from either operating unit coincident with loss of offsite power. The compressors may be started manually when the LOCA signal is no longer present.

The system is housed within the reactor building. Wind and tornado protection is discussed in Section 3.3. Flood design is discussed in Section 3.4. Missile protection is discussed in Section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in Section 3.6. Environmental design considerations are discussed in Section 3.11.

The compressed gas penetrations of the containments are designed, fabricated, and installed in accordance with the requirements of ASME Section III, Class 2 and Seismic Category I to prevent release of radioactive materials in the event of an accident. These penetrations will function as part of the Containment Isolation System, discussed in Subsection 6.2.4.

Because of the provisions of redundant system components and separate sources of electric power for the normal compressed gas system, the failure of a component or an electric power supply will not interrupt the operation of the Containment Instrument Gas System.

9.3.1.5.4 Tests and Inspections

The compressors, aftercoolers, receivers, prefilters, desiccant chambers, afterfilters, and the control panel are shop inspected or tested prior to installation.

During normal plant operation, gas compressors and associated components on standby are checked and operated periodically. Gas filters are inspected for cleanliness, and the desiccant is inspected for its useful life.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.3.1.5.5 Instrumentation Applications

Instrumentation is provided for each compressor train to monitor and automatically control each compressor's operation.

A compressor suction line pressure switch shuts down the compressors if the suction line pressure drops, as following a containment isolation. A suction line temperature switch shuts down the compressors if high temperature gas, such as steam from a ruptured pipe inside the containment, is drawn into the suction line.

In the compressor packages, compressor lube oil pressure, gas discharge temperature, and cooling water temperatures and pressure are monitored and will alarm locally and shut down their respective compressor if abnormal conditions are measured. A control room alarm, actuated by low-low pressure, monitors the instrument gas receiver of each compressor. Pressure switches on the header start their respective compressor if the compressor is in standby mode.

A pressure transmitter on the header transmits to a pressure indicator in the main control room. Two local pressure gages indicate the pressure in the manifold of each safety related instrument gas supply bottle header. A pressure switch on each header annunciates safety related header low pressure in the main control room.

Reduced pressure instrument gas is provided via a pressure reducing valve. Local and control room indication of this pressure is provided, as well as local pressure indication on the instrument gas accumulator.

9.3.2 PROCESS SAMPLING SYSTEM

The process sampling system is provided to monitor the operation of plant equipment and to provide information needed to make operational decisions.

The process sampling system provides remote sampling facilities and the capability for sampling fluids of various process systems during normal plant power operation and shutdown conditions.

The monitoring of gaseous and liquid process streams for nuclear radiation is covered separately in Section 11.5.

9.3.2.1 Design Bases

The portion of the process sampling system running from the reactor coolant system to the first isolation valve outside the containment is constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Class 1. Other sample piping, from the point where it connects to the process system and including the first process shutoff valve (root valve), will be the same classification as the system piping to which it connects. For ASME III, Class 1, 2, and 3 systems the sampling piping downstream from the root valve will be ASME III, Class 3 up to and including the isolation valve above the sample station.

All ASME Section III Class 1, 2 and 3 sample piping and valves are designed to Seismic Category I requirements.

Lines connected to reactor water or main steam systems are of sufficient length to permit decay of short lived nuclides so that sampling personnel will not be unnecessarily exposed to radiation. Additionally, shielding is installed at points on sampling piping to further curtail exposures (as described in Chapter 12) and ensure that they be kept below the limits of 10CFR20.

The process sampling system is designed to ensure that representative samples of all appropriate process fluids will be obtained.

Process sampling system piping is large enough to avoid being clogged by anticipated solids. Piping size is minimized to permit effective line purging with a minimum loss of fluid volume.

The process sampling system is designed so that the sample stations will not affect plant safety.

The process sampling system is designed to provide the capability to conduct continuous analysis as well as analysis of discrete samples (grab samples).

The process sampling system is designed to prevent hazards to operating personnel from high pressure, temperature, or radiation levels of the process fluid during all modes of operation.

The process sampling systems for each unit is designed to be functionally similar but operationally independent.

9.3.2.2 System Description

The process sampling system is illustrated schematically by Figures 9.3-6 thru 9.3-9. Locations of sample points are shown on the appropriate system piping and instrumentation diagrams for the systems to be sampled. The process sampling system consists of sampling lines, heat exchangers, sample vessels, sample sinks, and analysis equipment and instrumentation.

Sampling stations are located in the reactor, turbine, and radwaste buildings. The liquid radwaste collection sample station and the auxiliary boiler sample station are common for Units 1 and 2. The reactor and turbine building sample stations are operationally independent systems with the following exception: the spare fuel pool filter demineralizer effluent sample and the common offgas recombiner closed loop cooling water sample are located in Unit 1 stations.

Local grab samples rather than permanently installed sample lines to a control sampling station are provided for process points that require weekly sampling and are in zones where radioactivity is less than 15 mrem/hr (radiation Zones I, II, or III).

Samples of reactor feedwater, reactor recirculation water, main steam, and fuel pool water are routed to the reactor building sampling station. Samples of condensate and samples from secondary cooling water loops are brought to the turbine building sample station of each unit.

The reactor and turbine building stations are equipped with automatic monitors that continuously determine the critical parameters in the samples drawn from process lines.

Grab samples can also be taken periodically, in the laboratory or by portable instruments, from each point at each station to determine chlorides and other components. Portable instruments are also used for periodic calibration of the automatic monitors.

When working with grab samples, the operator is protected by a fume hood that is exhausted through the ventilation ductwork.

Sample flow rates to the monitors can be read and adjusted with a valve to provide the following conditions:

a) Ensure turbulence in the sample line to prevent plate-out of radioactive materials

b) Minimize lag time to monitors

c) Slow the sample flow rate as required for the decay of radioactive nitrogen prior to entering the stations

d) Minimize the waste of high purity water as well as input to the radwaste system.

Representative samples are drawn from process lines by sample nozzles extending into or from the pipe. Where practicable, a sample probe is located after a run of straight process pipe of about 10 but no less than three pipe diameters. On horizontal process pipes, the connection is made on the side rather than on the bottom of the pipe. Sample lines are as short as possible, avoiding traps, dead legs, and dips upstream of the sample stations. The connecting tubing is sized for optimal flow rates to the stations.

At each station, samples are automatically conditioned for pressure and temperature to the needs of the monitoring instruments and as required for the operators' safety. A constant temperature bath ensures that all samples are measured at the reference temperature of 25°C. Condensate sample waste is returned to the condenser hotwell. Fuel pool water sample waste is returned to the fuel pool. All other sample wastes are returned to the radwaste collection system.

Primary sample coolers, pressure reducers, and secondary coolers condition the samples to safe levels of pressure and temperature.

Prior to taking discrete samples, the sample line is purged (to the sample sink) with the fluid to be sampled so that a representative sample may be obtained. Sampling lines used for continuous samples do not require an additional purging prior to taking a sample.

Tubing loops in the sample stations (for decontamination) can be purged by connecting the demineralized water hose to the grab sample outlet while shutting off the sample inlet valve.

All sample panels are open in the back with tubing and components visible and accessible, so that leaks are easily detected and repaired.

9.3.2.3 Safety Evaluation

The process ssampling lines connected to the reactor coolant system are designed in accordance with Seismic Category I requirements as specified in Section 3.2 through the first isolation valve outside containment. Process sampling lines connected to other Seismic Category I components are Seismic Category I from the component through the first normally shut valve.

Sample lines that penetrate the containment are provided with isolation valves in accordance with 10CFR50, Appendix A, General Design Criteria 55.

The sampling system is designed to limit the discharge flows, under normal operation and during postulated malfunctions or failures, to preclude any fission-product release leading to exposures that exceed the site boundary limits stipulated in 10CFR20. Adequate safety features are provided to protect personnel and prevent the spread of contamination from the sampling station.

The sample station is composed of closed systems; grab samples will be taken under a chemical fume hood to preclude radiological hazard. The hood will maintain a constant air velocity of 150 ft/min through the hood working face to ensure that airborne contamination will not enter the room under operating conditions. The fume hood exhaust air will be exhausted to the buildings equipment compartment exhaust system. The sampling sinks are provided with demineralized water for washdown. The sinks drain to the liquid radwaste system.

The process sampling system is not required to function during an accident, nor is it required to prevent or mitigate the consequences of an accident.

9.3.2.4 Testing and Inspection

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

Most components will be used regularly during power operation, yielding cumulative data that will ensure the performance of the sampling system. Also, grab samples will be used to periodically test, calibrate, and check instrument response. Normal operating procedures at the stations provide for:

a) Adjusting pressure, temperature, and sample flow controls

b) Calibrating the monitors

c) Inspecting and cleaning conductivity, pH, and other sensors

d) Inspecting tubing and valves for leakages

e) Inspecting and cleaning sample coolers

f) Inspecting chiller-compressor stations (for the secondary sample coolers).

9.3.2.5 Instrumentation Applications

Local pressure, temperature, and flow indicators are used to facilitate manual operation and to verify sample conditions before samples are drawn.

The monitors used are solid state electronic instruments of standard industrial design. The analytical variables are recorded in the sample stations or in the main control room.

The main variables pertaining to the quality of the reactor coolant are entered into the computer. All monitored variables have alarm trips that signal when preset limits have been exceeded. Common trouble alarms are transmitted to the main control room.

9.3.3 EQUIPMENT AND FLOOR DRAINAGE SYSTEM

The Equipment and Floor Drainage System (EFDS) is provided throughout the plant to collect liquid wastes from their points of origin and transfer them to the Liquid Waste Management System, the plant discharge water treatment facilities, or the Storm Drainage System.

9.3.3.1 Design Bases

The EFDS is capable of handling the maximum expected influent. The turbine, reactor, circulating water pump, and diesel generator buildings influent is based upon 5 min of Fire Protection System operation. For the drywell and radwaste building the maximum expected leakage from equipment provides the design base.

The EFDS in the chlorine evaporator and sulfuric acid storage building is designed to drain rainwater from the acid unloading pad and from the open sides of the building.

The transformer gravel pits are sized to retain the oil contained in the transformers, in addition to the water volume from 15 min operation of the Deluge Fire Protection System.

The water treatment building acid unloading pad drainage system is designed to catch all acid leakage from the delivery trucks.

To prevent back flow into the Engineered Safety Features (ESF) equipment rooms, normally closed manual valves are provided in each drain line from those rooms.

Seismic Category I level switches, which are designed per IEEE 279 and 308 standards, alarm in the control room on ESF room high water level.

The EFDS is designed and arranged so that no inadvertent introduction of radioactive or potentially radioactive fluid to the segregated Sanitary and Storm Drainage Systems will occur.

Sump and drain tank pumps are designed to discharge at a flow rate adequate to keep the sumps and drain tanks from overflowing because of the expected influents outlined above. A backup pump is provided for each sump and drain tank, except for the condenser area transfer sump and the pipe tunnel sump. Backup pumps are started if the water level rises above the first pump start level.

The drywell equipment drain tank drains by gravity. The drain tank's discharge valves automatically open when a predetermined high level in the tank is reached. The discharge valves close at a predetermined low level.

Normally closed equipment and piping drains and vents discharging occasionally into the EFDS do not control the sizing of the system.

The sump and drain tank live capacities (water volume between pump start and stop levels) are based on not less than 5 min pump-out time with one pump.

The inlet pipes to the sumps are a minimum of 1 ft submerged at all times to prevent back gassing.

Vent lines from sumps containing potentially radioactive wastes are connected to the building ventilation systems.

Static oil interceptors with oil sumps precede the low conductivity sumps in the turbine and reactor buildings. Drainage lines from areas that are required to maintain an air pressure differential but drain to the same header are provided with water seals. Sequenced makeup water is provided to the water seals to maintain the air pressure differential. Where they penetrate the containment, the drywell floor drain sump pump and equipment drain tank discharge lines, including the containment isolation valves, are safety related.

9.3.3.1.1 Codes and Standards

The Equipment and Floor Drainage Systems are designed, fabricated, and installed in accordance with the requirements of the applicable codes and standards shown in Section 3.2, Table 3.2-1.

9.3.3.2 System Description

9.3.3.2.1 General Description

The combined Equipment and Floor Drainage Systems provided for collection of various liquid wastes are shown on Figures 9.3-10, 9.3-11, and 9.3-12. The chemical waste sump of the water treatment building is shown on Figure 9.2-8.

a) For potentially radioactive liquid wastes:

1) The Liquid Radwaste (LRW) Collection System collects potentially radioactive liquid wastes at atmospheric pressure from equipment and floor

drainage of the drywells, containments, reactor buildings, turbine building and radwaste building.

2) The Chemical Radwaste (CRW) Collection System collects corrosive, potentially radioactive liquid wastes at atmospheric pressure from the wash-down areas, sample stations, chemical equipment, and floor drains in the turbine and radwaste buildings. Nonradioactive, high conductivity wastes from the auxiliary boiler blow-down lines and the turbine building closed cooling water chemical addition tanks, at atmospheric pressure, are also collected by the CRW System.

3) The Detergent Radwaste (DetRW) Collection System collects potentially radioactive liquid wastes at atmospheric pressure from the wash-down areas, personnel decontamination stations, and laundry facilities in the reactor buildings, control building, and radwaste building.

The drainage sources and expected inputs from areas of potential radioactivity are shown in Table 11.2-1.

b) For nonradioactive liquid wastes:

1) Oily Waste Drainage Systems collect liquid wastes from the nonradioactive equipment areas in which oil is expected to be present. These areas include the circulating water pumphouse, diesel generator building, transformer areas, lube and diesel oil storage tank areas, oil circuit breaker areas, and auxiliary buildings.

2) Acid Waste Drainage Systems collect liquid wastes containing nonradioactive chemicals and corrosive substances from equipment and floor drains in the chlorine evaporator and sulphuric acid storage building and the water treatment building.

3) Sanitary Drainage Systems collect liquid wastes from all plumbing fixtures of the plant outside the restricted access areas.

4) Storm Drainage Systems collect water resulting from precipitation on all building roofs and areaways, paved and unpaved surface areas outside the buildings.

The radioactive and nonradioactive EFDS consist of collection piping, equipment drains, floor drains, vents, traps, cleanouts, collection sumps, sump pumps, tanks, oil separators, and instrumentation.

The arrangement is such that the nonradioactive drain systems serve only nonrestricted areas where no radioactivity potential is present, exclusive of the water closet and urinal wastes in the access control area that are collected by the Sanitary Drainage System.

The potentially radioactive wastes from personnel decontamination facilities and floor drains in the access control area are collected by the Detergent Radioactive Waste Collection System.

9.3.3.2.2 Component Description

Components of the Equipment and Floor Drainage Systems are described in the following paragraphs. Major components, such as sumps and sump pumps are shown in Table 9.3-10.

In areas of potential radioactivity, the collection system piping for liquid and detergent waste is made of carbon steel; the Chemical Waste Collection System piping is made of stainless steel. The piping has a slope that induces waste to flow in the piping at a velocity of not less than 2 fps. Equipment drainage piping terminates not less than 3 in. above the finished floor at each location where the discharge from equipment is collected.

Surface Drain Units (SDU) in the Oily Waste System are provided with backwater valves to prevent spread of potential fires.

All floor drains are installed with rims flush with the low point elevation of the finished floor. Floor drains in areas of potential radioactivity are welded directly to the collection piping and provided with threaded T-handle plugs of the same material. The T-handle plugs are used to seal off the floor drains for pressure testing of the drainage systems. They may be installed to prevent aspiration of radioactive particulates into the normally dry systems.

Inlets to all drainage systems (except those in areas of potential radioactivity, and those in rainwater and clean drainage service) are provided with a vented P-trap water seal to minimize entry of vermin, foul odors, and toxic, corrosive, or inflammable vapors into the building. Air pressure vent lines to the outside atmosphere are provided downstream of the P-traps to prevent excessive backpressures that could cause blowout or siphonage of the water seal. Normally, traps are not installed on inlets in areas of potential radioactivity in order to reduce the potential for accumulation of radioactivity in the traps, and because of the difficulty of maintaining a water seal in the trap.

Cleanouts are provided (when practicable) in all collection system piping where the change of direction in horizontal runs is

90 degrees, at offsets where the aggregate change is 135 degrees or greater, and at maximum intervals of 50 ft. Cleanouts for the potentially radioactive collection systems are welded directly to the piping.

Sources of the Laundry Radwaste and Chemical Radwaste Systems, which are too low in elevation to drain by gravity to the designated collection tank in the radwaste building, drain to local laundry and chemical drain tanks in the turbine building. From these drain tanks, the wastes are pumped to the main laundry and chemical waste tanks in the radwaste building.

All sumps are recessed in concrete located at the lowest elevation of the area served. Except for the drywell sumps, pipe tunnel drain sump, turbine building condenser area and chemical sumps, which are provided with removable steel cover sections, all sumps have access manholes.

Equipment and piping drains are separated by an air gap from the drainage piping to prevent pressurization of the drainage piping.

Where necessary for contamination control, splash guards are provided over air gaps.

9.3.3.2.3 System Operation

The various wastes drain to the appropriate sumps or tanks by gravity. Except for the condenser area sump pump, each sump pump starts automatically when a predetermined high level in its sump is reached; the sump pump stops at a predetermined low water level.

Potentially radioactive wastes are pumped to waste collection tanks in the radwaste building.

Leaks inside the drywell drain to the drywell floor drain sumps, except for piped valve and pump seal leak-offs which are routed to the drywell equipment drain tank for identification.

At the operator's option the manual valves in these seal leak-off lines can be closed to make use of the double seals on the leakage monitored valves and pumps. Leakages through the second seals, which are collected in the drywell floor drain sumps, are not identified to their source.

Floor and equipment drains from the condenser area are routed to the condenser area sump, which overflows to the turbine building central area oil separator. The overflow pipe contains an isolation valve that automatically closes when any of the following conditions occurs:

a) Activation of any of the three wet pipe sprinkler systems in the condenser area

b) High water level in the condenser area sump

c) Oil carry-over into the turbine building central area sump.

In case of fire or a large leakage from the circulating water piping, the condenser area transfer pump is started manually and the waste is pumped to the condensate tank berm for storage pending final disposition. Depending on the quality of the wastewater, it is then discharged either to the storm sewer or to the Liquid Radwaste System.

The Oily Waste System collects liquid that enters surface drain units (SDU) located in areas with no sources of potentially radioactive wastes, and where the possibility for oil spillage exists. The oily wastes are conveyed by gravity to oil separators of either the API or baffle type.

The oily waste in the circulating water pumphouse flows through an API type oil interceptor of 90 percent oil removal efficiency, and the effluent is pumped to a baffle type oil separator. Baffle type oil separators provide an effluent with a total oil concentration of less than 10 ppm, conforming to the requirements of Pennsylvania's Department of Environmental Resources. The clarified effluent discharges to the circulating water pump house sump. Water collected in the sump is discharged to the circulating water pump suction line. Oil collected in the oil separator is periodically pumped into a portable drum for disposal.

Floor and equipment drains from the four diesel generator bays are collected in the diesel generator building sump after passing through a baffle type oil separator. The clarified effluent is pumped to the storm sewer.

Equipment drains from the main turbine bearings and the turbine lube oil centrifuges are piped directly to the turbine building outer area oil sumps. The turbine lube oil reservoir rooms are recessed from the normal floor level in order to contain all the lube oil in case of a tank rupture. The floor drains from the turbine lube oil reservoir rooms can be conveyed directly or through the oil interceptor to the oil sump by opening one of the two normally closed valves on two drainage branch lines.

The Acid Waste System collects liquid waste containing chemicals and corrosive substances discharged by laboratory fixtures and equipment. The Acid Waste System also serves the floor drains, which are located in the water treatment building, and conveys the liquid waste directly or by means of the chemical waste sump pumps to a pair of neutralization basins. Floor drains from the acid storage and chlorine evaporator building containing acid contamination are collected in a sump and automatically pumped to the neutralization basins, or if it is determined that the sump contains only rainwater, to the storm sewer.

The Sanitary Drainage System collects liquid wastes and some entrained solids discharged by all plumbing fixtures located in areas with no sources of potentially radioactive wastes, oily wastes, or acid wastes and conveys them to a sewage treatment facility described in Subsection 9.2.4.

The drain lines were designed to accommodate fire protection system design flow when actuated.

9.3.3.3 Safety Evaluation

With the exception of the drywell equipment drain and drywell floor drain sump discharge pipe penetrations through the primary containment and the associated isolation valves, the failure of the EFDS cannot affect plant safety. The drywell floor drain sumps and the associated leak detection instrumentation are designed to Seismic Category I. Pump operability is not required for the functioning of the differential level Drywell Floor Drain Leak Detection System.

Each of the six pump rooms (ECCS and RCIC) is provided with a separate drain line to the reactor building sump inlet header. A normally closed manual valve is provided in each drain line outside the pump room to prevent backflow. Seismic Category I level instrumentation provides for main control room alarms if the water level in the pump rooms rises above a preset value.

9.3.3.4 Tests and Inspections

All waste collection piping was hydrostatically tested prior to its embedment in concrete. Potentially radioactive drainage piping was tested to 75 psig, in accordance with ANSI B31.1.0. Nonradioactive oily, acid, sanitary, and storm drainage piping was tested to the equivalent of a 10 ft head of water for at least 15 min. The operability of Equipment and Floor Drainage Systems can be checked by normal use and through the instrumentation provided in the sumps and the main control room.

9.3.3.5 Instrumentation Application

High and low level switches are provided in each sump. For sumps having two pumps, the level switch will actuate the second pump at a higher level. The first pump to start is alternated on each pumping cycle to equalize run times. Table 9.3-10 shows the usage factors resulting from this provision.

The drywell equipment drain tank drains by gravity. The drain tank's discharge valves automatically open when a predetermined high level in the bank is reached. The discharge valves close at a predetermined low level.

Oil sumps are equipped with level switches and high level alarms in the main control room.

To detect leaks, a level alarm will be provided in the main control room for each ESF equipment room.

The drywell floor drain sump and the drywell equipment drain tank temperatures are indicated, and a high alarm is annunciated on a local panel in the reactor building of each unit.

The levels in the drywell floor drain sumps and drywell equipment drain tanks are recorded, and a high level alarm is annunciated in the main control room. Refer to Subsection 5.2.5 and Section 7.6 for further details of the Leak Detection System.

9.3.4 CHEMICAL AND VOLUME CONTROL SYSTEM

Not applicable to BWR's.

9.3.5 STANDBY LIQUID CONTROL SYSTEM

9.3.5.1 Design Bases

The standby liquid control system is a special safety system and is designed in accordance with Seismic Category I requirements. It shall meet the following safety design bases:

(a) Backup capability for reactivity control shall be provided, independent of normal reactivity control provisions in the nuclear reactor, to be able to shut down the reactor if the normal control ever becomes inoperative.

shutdown margin, to assure complete shutdown from the most reactive condition at any time in core life.

(c) The time required for actuation and effectiveness of the backup control shall be consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown conditions. A fast scram of the reactor or operational control of fast reactivity transients is not specified to be accomplished by this system.

(d) Means shall be provided by which the functional performance capability of the backup control system components can be verified periodically under conditions approaching actual use requirements. Demineralized water, rather than the actual neutron absorber solution, can be injected into the reactor to test the operation of all components of the redundant control system.

(e) The neutron absorber shall be dispersed within the reactor core in sufficient quantity to provide a reasonable margin for leakage for imperfect mixing.

(f) The system shall be reliable to a degree consistent with its role as a special safety system; the possibility of unintentional or accidental shutdown of the reactor by this system shall be minimized.

9.3.5.2 System Description

The standby liquid control system (see Figure 9.3-13) is manually initiated through a single keylock switch in the main control room to pump a boron neutron absorber solution into the reactor if the operator determines the reactor cannot be shut down or kept shut down with the control rods. The keylocked control room switch is provided to assure positive action from the main control room should the need arise. Procedural controls are applied to the operation of the keylocked control room switch.

The boron solution tank, the test water tank, the two positive displacement pumps, the two explosive valves, the two pump suction valves, and associated local valves and controls are located in the reactor building. The liquid is piped into the reactor vessel and discharged near the bottom of the core shroud so it mixes with the cooling water rising through the core (see Section 5.3).

The specified neutron absorber solution is sodium pentaborate ($Na_2 B_{10} O_{16} 10H_2O$). It is prepared by dissolving stoichiometric quantities of borax and boric acid in demineralized water. An air sparger is provided in the tank for mixing. To prevent system plugging, the tank outlet is raised above the bottom of the tank.

The SLC system is sized to deliver enough sodium pentaborate solution into the reactor (see Figure 9.3-14) to assure reactor shutdown.

The saturation temperature of the recommended solution is 59°F at the low level alarm volume and approximately 49°F at the tank overflow volume (see Figure 9.3-15). The equipment containing the solution is installed in a room in which the air temperature is to be maintained within the range of 70° to 100°F. In addition, a heater system maintains the solution temperature at 75° to 85°F to prevent precipitation of the sodium pentaborate from the solution during storage. High or low temperature, or high or low liquid level, causes an alarm in the control room.

Each positive displacement pump is sized to inject the solution into the reactor in 50 to 125 minutes, independent of the amount of solution in the tank. The pump and system design pressure between the explosive valves and the pump discharge is 1400 psig. The two relief valves are set slightly under 1400 psig. To prevent bypass flow from one pump in case of relief valve failure in the line from the other pump, a check valve is installed downstream of each relief valve line in the pump discharge pipe.

The two explosive-actuated injection valves provide assurance of opening when needed and ensure that boron will not leak into the reactor even when the pumps are being tested.

Each explosive valve is closed by a plug in the inlet chamber. The plug is circumscribed with a deep groove so the end will readily shear off when pushed with the valve plunger. This opens the inlet hole through the plug. The sheared end is pushed out of the way in the chamber; it is shaped so it will not block the ports after release.

The shearing plunger is actuated by an explosive charge with dual ignition primers inserted in the side chamber of the valve. Ignition circuit continuity is monitored by a trickle current, and an alarm occurs in the control room if either circuit opens. Indicator lights show which primary circuit opened.

The SLC system is actuated by a three-position keylocked switch on the control room console. This assures that switching from the "off" position is a deliberate act. Switching to either side starts an injection pump, actuates both of the explosive valves, and closes the reactor cleanup system outboard isolation valve to prevent loss or dilution of the boron.

A light in the control room indicates that power is available to the pump motor contactor and that the contactor is deenergized (pump not running). Another light indicates that the contactor is energized (pump running).

Storage tank liquid level, tank outlet valve position, pump discharge pressure, and loss of continuity on the explosive valves indicate that the system is functioning. If any of those items indicate that the liquid may not be flowing, the operator may immediately change the other switch status to "run" thereby activating the redundant train of the SLC system. The local switch will not have a "stop" position. This prevents the isolation of the pump from the control room. Pump discharge pressure and valve status are indicated in the control room.

Equipment drains and tank overflow are not piped to the radwaste system but to separate containers (such as 55-gal. drums) that can be removed and disposed of independently to prevent any trace of boron from inadvertently reaching the reactor.

Instrumentation consisting of solution temperature indication and control, solution level, and heater system status is provided locally at the storage tank. Table 9.3-11 contains the process data for the various modes of operation of the SLC.

9.3.5.3 Safety Evaluation

The standby liquid control system is a reactivity control system and is maintained in an operable status whenever the reactor is critical. The system is expected never to be needed for safety reasons because of the large number of independent control rods available to shut down the reactor.

To assure the availability of the SLC system, and to facilitate maintenance and testing, two sets of the components required to actuate the system - pumps and explosive valves are provided in parallel redundancy.

The system is designed to bring the reactor from rated power to a cold shutdown at any time in core life. The reactivity compensation provided will reduce reactor power from rated to zero level and allow cooling the nuclear system to room temperature, with the control rods remaining withdrawn in the rated power pattern. It includes the reactivity gains that result from complete decay of the rated power xenon inventory. It also includes the positive reactivity effects from eliminating steam voids, changing water density from hot to cold, reduced Doppler effect in uranium, reducing neutron leakage from boiling to cold, and decreasing control rod worth as the moderator cools.

The minimum average concentration of natural boron in the reactor to provide adequate shutdown margin, after operation of the SLC system, is 660 ppm. Calculation of the minimum quantity of sodium pentaborate to be injected into the reactor is based on the required 660 ppm average concentration in the reactor coolant including recirculation loops, at 70°F and reactor normal water level. The result is increased by 25% to allow for imperfect mixing and leakage. Additional sodium pentaborate is provided to accommodate dilution by the RHR system in the shutdown cooling mode. This concentration will be achieved if the solution is prepared as defined in Subsection 9.3.5.2 and maintained above saturation temperature.

Cooldown of the nuclear system will require a minimum of several hours to remove the thermal energy stored in the reactor, cooling water, and associated equipment. The controlled limit for the reactor vessel cooldown is 100°F per hour, and normal operating temperature is approximately 550°F. Use of the main condenser and various shutdown cooling systems requires 10 to 24 hours to lower the reactor vessel to room temperature (70°F); this is the condition of maximum reactivity and, therefore, the condition that requires the maximum concentration of boron.

The specified boron injection rate is limited to the range of 6 to 25 ppm per minute. The lower rate assures that the boron is injected into the reactor in approximatley two hours. This resulting reactivity insertion is considerably quicker than that covered by the cooldown. The upper limit injection rate assures that there is sufficient mixing so that boron does not recirculate through the core in uneven concentrations that could possibly cause reactor power to rise and fall cyclically.

The SLC system is required to be operable in the event of a station power failure, therefore the pumps, heaters, valves, and
controls are powered from or connectable to the standby a-c power 17
supply. The pumps and valves are powered and controlled from separate buses and circuits so that a single electrical failure
will not prevent injection of sodium pentaborate on demand. 10

The SLC system and pumps have sufficient pressure margin, up to the system relief valve setting of approximately 1400 psig, to assure solution injection into the reactor above the normal pressure in the bottom of the reactor. The nuclear system relief and safety valves begin to relieve pressure above approximately 1100 psig. Therefore, the SLC system positive displacement pumps cannot overpressurize the nuclear system.

Only one of the two standby liquid control pumps is needed for system operation. If a redundant component (e.g., one pump) is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation can continue during repairs. The time during which one redundant component upstream of the explosive valves may be out of operation should be consistent with the following: the probability of failure of both the control rod shutdown capability and the alternate component in the SLC system; and the fact that nuclear system cooldown takes several hours while liquid control solution injection takes approximately two hours. Since this probability is small, considerable time is available for repairing and restoring the SLC system to an operable condition while reactor operation continues. Assurance that the system will still fulfill its function during repairs is obtained by demonstrating operation of the operable pump.

In the event of a loss of the thermostatically-controlled storage tank heater "A", a low temperature alarm would eventually be annunciated in the control room and would alert the operator to control storage tank temperature manually from the local panel by means of the mixing heater "B".

A low-temperature alarm will also annunciate in the control room
10 if there is a loss of the suction piping heat tracing. The alarm setpoint is sufficiently above saturation temperature of the sodium pentaborate solution such that, even in the unlikely event that ambient temperature is below 70°F, sufficient time will be available to enable the operating personnel to take appropriate temporary measures to heat the suction piping before precipitation occurs.

The SLC system is evaluated against the applicable General Design Criteria as follows:

<u>Criterion 2</u>: The SLCS is located in the area outside of the primary containment (drywell) and below the refueling floor. In this location it is protected by the containment and compartment walls from external natural phenomena such as earthquakes, tornadoes, hurricanes and floods and internally from effects of such events and internal postulated events.

<u>Criterion 4</u>: The SLCS is designed for the expected environment in the containment and specifically for the compartment in which it is located. In this compartment, it is not subject to the more violent conditions postulated in this criterion such as missiles, whipping pipes, and discharging fluids. This system is only called upon to perform a pseudo-safety function under normal operation conditions.

<u>Criterion 21</u>: Criterion 21 is applicable to protection systems only. The SLC system is a reactivity control system and should be evaluated against Criterion 29.

<u>Criterion 26</u>: The SLCS is the second reactivity control system required by this criterion. The requirements of this criterion do not apply within the SLCS itself.

<u>Criterion 27</u>: This criterion applies no specific requirements onto the SLCS and, therefore, is not applicable. See the General
17 Design Criteria Section (Section 3.1) for discussion of combined capability.

Criterion 29: The SLCS pumps and valves cutboard of the isolation valves are redundant. Two pumps, and two injection valves are arranged and cross-tied such that operation of any one of each results in successful operation of the system. The SLCS also has test capability. A special test tank is supplied for providing test fluid for the yearly injection test. Pumping capability may be tested at any time. A trickle current continuously monitors continuity of the firing mechanisms of the injection squib valves.

The SLC system is evaluated against the applicable regulatory guides as follows:

Regulatory Guide 1.26 Revision 2: Because the SLCS is a reactivity control system, all mechanical components are at least Quality Group B. Those portions which are part of the Reactor Cooling Pressure Boundary are Quality Group A. This is shown in Table 3.2-1.

Regulatory Guide 1.29 Revision 1: All GE supplied components of the SLCS which are necessary for injection of neutron absorber into the reactor are Seismic Category I. This is shown in Table 3.2-1.

Since the SLC system is located within its own compartment within the reactor building, it is adequately protected from flooding, tornadoes, and internally and externally generated missiles. SLC system equipment is protected from pipe break by providing adequate distance between the seismic and non-seismic SLC system equipment where such protection is necessary. In addition, appropriate distance is provided between the SLC system and other piping systems. Where adequate protection cannot be assured, barriers have been considered to assure SLC system protection from pipe break (See Section 3.6).

It should be noted that the SLC system is not required to provide a safety function during any postulated pipe break events. This system is only required under an extremely low probability event where all of the control rods are assumed to be inoperable while the reactor is at normal full power operation. Therefore, the protection provided is considered over and above that required to meet the intent of APCSB 3-1 and MEB 3-1.

This system is used in a couple of special plant capability demonstration events cited in Appendix A of Chapter 15. Specifically Events 51, 52, and 53 which are extremely low probability non-design basis postulated incidents. The analyses given there are to demonstrate additional plant safety consideration far beyond reasonable and conservative assumptions.

A system-level, qualitative-type failure mode and effects analysis is presented in Subsection 15A.6.6.

9.3.5.4 Testing and Inspection Requirements

Operational testing of the SLC system is performed in at least two parts to avoid inadvertently injecting boron into the reactor.

With the valve from the storage tank closed and the valves to and from the test tank opened, demineralized water in the test tank can be recirculated by locally starting either pump.

During a refueling or maintenance outage, the injection portion of the system can be functionally tested by valving the suction line to the test tank and actuating the system from the control room. System operation is indicated in the control room.

After functional tests, the injection valve shear plugs and explosive charges must be replaced and all the valves returned to their normal positions as indicated.

After closing a local locked-open valve to the reactor, leakage through the injection valves can be detected by opening valves at a test connection in the line between the containment isolation check valves. Position indicator lights in the control room indicate that the local valve is closed for tests or open and ready for operation. Leakage form the reactor through the first check valve can be detected by opening the same test connection in the line between the Containment Isolation Check Valves when the reactor is pressurized.

The test tank contains demineralized water for approximately 3 minutes of pump operation. Demineralized water from the makeup system or the condensate storage system is available for refilling or flushing the system.

Should the boron solution ever be injected into the reactor, either intentionally or inadvertently, then after making certain that the normal reactivity controls will keep the reactor subcritical, the boron is removed from the reactor coolant system by flushing for gross dilution followed by operating the reactor cleanup system. There is practically no effect on reactor operations when the boron concentration has been reduced below approximately 50 ppm.

The concentration of the sodium pentaborate in the solution tank is determined periodically by chemical analysis. Electrical supplies and relief valves are also subjected to periodic testing (see Chapter 16).

The SLC system is preoperationally tested in accordance with the requirements of Chapter 14.

9.3.5.5 Instrumentation Requirements

The instrumentation and control system for the SLC is designed to allow the injection of liquid poison into the reactor and the maintenance of the liquid poison solution well above the saturation temperature. A further discussion of the SLC instrumentation may be found in Chapter 7.

TABLE 9.3-2

INSTRUMENT AIR SYSTEM

PNEUMATICALLY OPERATED VALVES WHICH HAVE A SAFETY FUNCTION(1)

SYSTEM AND FIGURE NUMBER	LOCATION	DESIGN FUNCTION	NORMAL POSITION	FAIL POSITION	SAFE POSITION
Service Water Figure 9.2-1a and 9.2-1b HV-10943A2	Turbine building closed cooling water heat exchanger outlets	Heat removal from turbine building closed cooling water heat exchangers	Closed	Closed	Closed
HV-10943B2	Turbine building closed cooling water heat exchanger outlets	Heat removal from turbine building closed cooling water heat exchangers	Closed	Closed	Closed
HV-10943A3	Turbine building closed cooling water heat exchanger outlets	Heat removal from turbine building closed cooling water heat exchangers	Open	Open	Open
HV10943B3	Turbine building closed cooling water heat exchanger outlets	Heat removal from turbine building closed cooling water heat exchangers	Open	Open	Open
Emergency Service Water Figures 9.2-5a and9.2-5b HV-11143A	Turbine building closed cooling water heat exchanger inlet	Heat removal from turbine building closed cooling water heat exchangers	Closed	Closed	Closed
HV-11143B	Turbine building closed cooling water heat exchanger inlet	Heat removal from turbine building closed cooling water heat exchangers	Closed	Closed	Closed
Reactor Core Isolation Cooling Figure 9.2-6 HV-1F088	Steam supply	Bypass	Closed	Closed	Closed
HV-1F025	Steam drain line	Isolation	Open	Closed	Closed
HV-1F026	Steam drain line	Isolation	Open	Closed	Closed
LV-1F054	Steam drain line	Bypass	Closed	Closed	Closed
RCIC Turbine-Pump Figure 9.2-6 HV-1F004	Drain line on the RCIC vacuum tank condensate pump	Drain isolation	Closed	Closed	Closed
HV-1F005	Drain line on the RCIC vacuum tank condensate pump	Drain isolation	Open	Closed	Closed

TABLE 9.3-2 (Continued)

SYSTEM AND FIGURE NUMBER	LOCATION	DESIGN FUNCTION	NORMAL POSITION	FAIL POSITION	SAFE POSITION
HPCI Turbine Pump Figure 9.2-6 HV-1F026	Drain line on the HPCI turbine pump	Drain isolation	Closed	Closed	Closed
HV-1F025	Drain line on the HPCI turbine pump	Drain isolation	Open	Closed	Closed
Containment Atmos. Control PV-05719	Nitrogen supply line to primary containment	Supply shut-off (isolation)	Closed	Closed	Closed
Control Rod Drive XV-1F011	Scram discharge volume piping	Vent valve isolation	Open	Closed	Closed
XV-1F010	Scram discharge volume	Drain line isolation	Open	Closed	Closed
XV-126	Scram inlet valve	Scram inlet	Closed	Open	Open
XV-127	Scram exhaust valve	Scram exhaust	Closed	Open	Open
FV-1F002A	Man/auto station drive water pump discharge	Flow control	Open	Closed	Closed
FV-1F002B	Man/auto station drive water pump discharge	Flow control	Open	Closed	Closed

TABLE 9.3-2 (Continued)

SYSTEM AND FIGURE NUMBER	LOCATION	DESIGN FUNCTION	NORMAL POSITION	FAIL POSITION	SAFE POSITION
Reactor Building HVAC System Figure 9.4-6, 9.4-7, and 9.4-8	Fan discharge (V212A&B)	Isolation	Open	Closed	Closed
HD-17564A&B	Zone III supply fan discharge (1V212B)				
HD-17586A&B	Zone I supply fan discharge (1V202A&B)	Isolation	Open	Closed	Closed
HD-17576A&B	Zone I exhaust fan inlet (1V205A&B)	Isolation	Open	Closed	Closed
HD-17502A&B	Zone III exhaust system inlet (1V213A&B)	Isolation	Open	Closed	Closed
HD-17514A&B	Zone III filtered exhaust system (1V217A&B)	Isolation	Open	Closed	Closed
HD-17524A&B	Zone I equipment comp. exhaust system (1V206A&B)	Isolation	Open	Closed	Closed
HD-17508A&B	On duct from Zone I or II drywell purge to SGTS	Isolation	Open	Closed	Closed
TV-07550A&B	Fire protection isolation valve for SGTS	Isolation	Closed	Closed	Closed
HV-07551A1,2,3&4 HV-07551B1,2,3&4	SGTS drain valves	Drain off fire protection water	Closed	Closed	Closed
HD-17534A thru H	R.B. air locks	Isolation	Open	Closed	Closed
HD-07543A&B	R.B. recirculation system outlet	Interconnection between SGTS & Recirculation System	Closed	Open	Open
HD-17601A&B HD-17602A&B HD-17657A&B	R.B. recirculation system inlet	Interconnection between Recirculation System RB Duct	Closed	Closed	Closed
PDD-07554A&B PDD-07543A&B	R.B. SGTS inlet from R.B. Recirculation	R.B. negative pressure (No Dulation)	Open	As Is	As Is

TABLE 9.3-2 (Continued)

SYSTEM AND FIGURE NUMBER	LOCATION	DESIGN FUNCTION	NORMAL POSITION	FAIL POSITION	SAFE POSITION
Control Structure HVAC System Figure 9.4-2		Fire protection			
TV-07813A TV-07813B	Water spray to the activated charcoal filters	Water spray isolation	Closed	Closed	Closed
HD-07824A1,B1 HD-07824A3,B3 HD-07824A5,B5	Return air to units 0V103A&B	Isolation	Open	Closed	Closed
HD-07824A2,B2 HD-07824A4,B4 HD-07824A6,B6	Supply air from units 0V103A&B	Isolation	Open	Closed	Closed
HD-07802A&B	Outside air to units 0V103A&B	Isolation	Open	Closed	Closed
HD-07833A&B	Control room floor relief fan inlet	Isolation	Open	Closed	Closed
HD-07873A&B	Control room kitchen exhaust fan inlet	Isolation	Open	Closed	Closed
HD-07872A&B	Control room toilet fan inlet	Isolation	Open	Closed	Closed

(1) A complete list of pneumatically operated valves required for containment isolation is found in Table 6.2-12

TABLE 9.3-4

SERVICE AIR SYSTEM DESIGN PARAMETERS
FOR UNIT 1(1)

Compressor Units	
Quantity	2(1)
Capacity, each, scfm	440
Discharge pressure, psig	125
Cooling water: flow rate, gpm	13
temp. in °F	105
temp. out °F	125
Aftercoolers	
Quantity	2(1)
Capacity, each, scfm	440
Operating pressure, psig	125
Cooling water: flow rate, gpm	10.8
temp. in °F	105
temp. out °F	125
Receivers	
Quantity	2(1)
Capacity, each ft^3	223
Design pressure, psig	139
Design temperature, °F	200

(1) The design parameters for Units 1&2 are identical except the quantity of each component for Unit 2 is one.

TABLE 9.3-5

LOW PRESSURE AIR SYSTEM DESIGN PARAMETERS
(COMMON TO UNITS 1 & 2)

Compressor	
Quantity	1
Capacity, scfm	700
Discharge pressure, psig	35
Cooling water: flow rate, gpm	7
temp. inlet °F	105
temp. outlet °F	125
Aftercooler	
Quantity	1
Capacity, scfm	700
Operating pressure, psig	125
Cooling water: flow rate, gpm	19
temp. inlet °F	105
temp. outlet °F	125
Receiver	
Quantity	1
Capacity, ft^3	151
Design pressure, psig	125
Design temperature, °F	200

TABLE 9.3-7

LIST OF INSTRUMENT GAS OPERATED DEVICES

1. Four main steam isolation valves.
2. Sixteen main steam relief valves, including six valves with auto depressurizing function (ADF).
3. One recirculation sample line valve.
4. Two RHR check valves.
5. Two equalizing valves for RHR check valves.
6. Two core spray check valves.
7. Two equalizing valves for core spray check valves.
8. Five tip indexing mechanisms.
9. Ten vacuum relief valves.
10. Eight reactor building chilled water valves.
11. Containment leak chase detection system.
12. One Reactor Core Isolation Cooling (RCIC) steam line equalizing valve.
13. One High Pressure Coolant Injection (HPCI) steam line equalizing valve.

9.4 AIR CONDITIONING, HEATING, COOLING, AND VENTILATION SYSTEMS

9.4.1 CONTROL ROOM AND CONTROL STRUCTURE HVAC SYSTEMS

The following systems are covered under this subsection.

a) Control Room Floor Cooling System

b) Computer Room Floor Cooling System

c) Control Structure H&V System

d) Emergency Outside Air Supply System

e) SGTS Equipment Room H&V Systems

f) Battery Rooms Exhaust System

g) Smoke Removal System

h) Access Control and Lab Area Supply System

i) Lab Fume Hood Makeup and Exhaust Systems

All HVAC systems in the control structure are common systems which are shared by two power plant units (Unit 1 and Unit 2).

9.4.1.1 Design Basis

9.4.1.1.1 Control Room Floor Cooling System (0V-117)

This system provides ventilation, cooling, and control of environmental conditions in the control room and associated areas on the 728 ft elevation of the control structure. The system is designed to accomplish the following objectives during normal plant operation as well as under emergency conditions:

a) Maintain the space temperature at 75°F ± 5°F, to control the air movement for personnel comfort and to ensure the operability of control room equipment and instruments under normal and design basis accident conditions

b) Maintain the space relative humidity at 50 percent ± 5 percent for personnel comfort and equipment performance under normal operation

c) Maintain a positive pressure above atmosphere to inhibit air leakage into the control room during high radiation isolation modes

d) Isolate the space and divert the outside air supply through the emergency outside air filter system when high radiation is detected in the outside air

e) Recirculate and clean up room air when chlorine is present in the outside air

f) Monitor radiation and detect chlorine in the outside air supply

g) Operate during normal, shutdown, and design basis accident conditions without loss of function.

The control room floor cooling system (0V-117) has a safety related function and is designed to meet the Seismic Category I requirements. The kitchen exhaust fan, toilet exhaust fan, reheat coils, and the humidification systems are not safety related.

9.4.1.1.2 Computer Room Floor Cooling System (0V-115)

This system provides ventilation, cooling, and control of environmental conditions in the computer room, lower relay room, maintenance rooms, and Uninterruptible Power Supply (UPS) rooms. The cooling system is designed to:

a) Maintain the space temperature at 75°F ± 10°F (except the UPS rooms which are 104°F maximum), to control air movement for personnel comfort and to ensure the operability of the computer equipment under normal conditions.

b) Maintain the space relative humidity at 50 percent ±10 percent as required for computer performance under normal operation.

9.4.1.1.3 Control Structure H&V System (0V-103)

This system serves all elevations within the control structure envelope (refer to Figure 9.4-1), except elevation 728 ft (control room floor) and elevation 697 ft (computer room floor).

The system is designed to accomplish the following objectives during normal plant operation as well as under DBA conditions:

a) Maintain temperatures in the various spaces within specified limits.

b) Meet the specified cooling and ventilation requirements to ensure the operability of the equipment and instruments without loss of function.

c) Maintain a positive pressure above atmosphere to inhibit air leakage into the control structure envelope.

d) Isolate the control structure envelope and divert the outside air supply through the emergency outside air filter system during accident conditions.

9.4.1.1.4 Emergency Outside Air Supply System (OV-101)

This system is designed to:

a) Filter radioactivity from the outside air supply

b) Recirculate and clean up room air when chlorine is present in the outside air

c) Maintain the specified outside air supply to the control room and control structure envelope during accident conditions

d) Maintain a positive pressure above atmospheric to inhibit air leakage into the control room during radiation isolation

e) Operate during and after design basis accident and reactor building isolation mode conditions without loss of function

f) Provide radiation monitoring and chlorine detection of outside air supply.

The emergency outside air supply system has safety related functions and is designed to Seismic Category I requirements.

9.4.1.1.5 SGTS Equipment Room H&V Systems

The SGTS equipment room heating system (OV-144) and ventilation system (OV-118) are designed to:

a) Maintain temperatures in the space within a range suitable for equipment performance

b) Maintain adequate air flow for ventilation.

The SGTS equipment room is located at Elevation 806 ft. All ductwork and equipment has safety related functions and is designed to Seismic Category I requirements.

9.4.1.1.6 Battery Room Exhaust System (0V-116)

The function of the battery room exhaust system is to maintain design temperature and pressure conditions and provide adequate airflow for ventilation. The battery room exhaust system is designed to ensure that hydrogen concentrations remain within acceptable limits.

9.4.1.1.7 Smoke Removal Exhaust System (0V-104)

The purpose of the smoke removal system is to exhaust smoke and gas after a fire has been extinguished from areas in the control structure between the elevations of 697 ft-0 in. and 771 ft-0 in. including the control room.

The system has no safety related function. However, the isolation dampers in the top of the duct shaft wall and those in the control structure envelope boundary are of Seismic Category I construction. The connecting ductwork between these isolation dampers is also of Seismic Category I design.

9.4.1.1.8 Access Control and Lab Area Supply System (0V-105)

This system serves the access control and laboratory area at elevation 676 ft-0 in. of the control structure. This area is located outside the control structure envelope boundary. The equipment that serves this area is located in turbine building Unit 1 at elevation 762 ft-0 in. (H&V equipment room). The system has no safety related function and is designed to accomplish the following objectives during normal plant operation:

a) Maintain temperature in the various areas within personnel comfort limits, (75°F ± 10°F)

b) Maintain adequate airflow for comfort and ventilation.

c) Maintain space pressure at approximately atmospheric.

9.4.1.1.9 Lab Fume Hood Makeup and Exhaust Systems(0V-106 and 0V-114)

These systems have no safety related function. The design basis is to accomplish the following objectives during normal plant operation:

a) Maintain air balance for fume hoods

b) Filter contaminated air from fume hoods and exhaust it through turbine building Unit 1 exhaust vent to the atmosphere.

The laboratory fume hoods are located in the control structure at elevation 676 ft-0 in., the filter units at elevation 686 ft-0 in., and the exhaust fan at elevation 806 ft-0 in.

The laboratory fume hood makeup air unit is located in the turbine building Unit 1 H&V equipment room at elevation 762 ft-0 in. All the above lab fume hood systems equipment is located outside the control structure boundary.

9.4.1.2 System Description

9.4.1.2.1 Control Room Floor Cooling System (0V-117) and Computer Room Floor Cooling System (0V-115)

The control room floor and computer room floor cooling systems (0V-117 and 0V-115 respectively) are symmetrically designed. One serves the control room and the other serves the computer room. Both systems are shown on Figures 9.4-1 and 9.4-2. Design parameters for the control room floor and computer room floor cooling systems are listed in Table 9.4-2.

Each system is served by two 100 percent capacity redundant air handling units (one operating and one on standby). Each unit contains a ventilation filter bank, chilled water cooling coils, a centrifugal fan, and a fan outlet damper. The two units are connected to a common Seismic Category I supply and return duct system that distributes supply air throughout the space and returns room air to the units. The conditioned air is cooled by water cooling coils. The chilled water supply system is described in Subsection 9.2.12. The control room and the computer room air conditioning equipment is located within a Seismic Category I structure. All equipment in each redundant system is powered from an independent Class IE power source.

The chilled water for the cooling coils in each system is supplied by a Seismic Category I, independent chilled water supply system. The chilled water systems are interlocked with their respective supply air fans in the same division.

When the chiller train starts, the fans on the same division automatically start. Failure of any of these fans is annunciated in the control room and also trips the chilled water system and the fans in that division. The standby chilled water and air systems start automatically.

Redundant temperature switches are provided at the suction side of the fans of both systems. When the suction trip air temperature for the fans is high, the operating fan and its associated chiller train is tripped and the standby chiller train and its associated fans all started simultaneously.

Fan selector switches in the control room panel allow manual selection of systems.

Each air system is provided with an air temperature controller that regulates the temperature of the return air. The controller will modulate a three-way mixing valve to control chilled water flows through the cooling coils.

For further description of the chilled water system see Subsection 9.2.12.

Each system supplies a minimum quantity of outside air and recirculates conditioned air to maintain space requirements; space humidity will be controlled during normal operations.

The outside air is taken from an outside air intake system that is described in Subsection 9.4.1.2.4. The control room floor and computer room floor cooling systems are supplied with outside air through a branch duct from the outside air intake system. A preset quantity of outside air is provided for these systems. This branch outside air duct is equipped with an electric duct heater. A duct mounted thermostat regulates a controller to modulate the leaving air temperature to a minimum of 50°F. The duct heater unit is Seismic Category I, but the control for the heater is not safety related. During emergency operation, this heater is not required to operate; the outside air will be heated by the emergency outside air system.

A four step humidification system is provided for the control room and computer room. A humidistat mounted in the control room return air duct regulates a step controller to maintain humidity in the control room. The humidification is designed for normal operation only and is not a safety related system.

The humidification system is supported independantly except for the steam distributor which is mounted in the duct. In the event of a DBA if the distributor fails, it will not affect the operation of the control room HVAC system.

The plant security office and locker rooms are equipped with duct mounted heating coils that operate during normal operation only and are not safety related. These reheat coils are controlled by space thermostats set at 75°F. The heating coils are interlocked with the system supply fan.

The control structure envelope is designed to be maintained at a positive pressure of approximately 1/8 in. water gauge above atmospheric pressure during the high radiation isolation phase and normal operation. The TSC and DSC are equipped with a nonsafety related duct mounted reheat coil for normal operations. The reheat coil is modulated by a space thermostat to control space temperature and is interlocked with the system supply fan.

During normal operation the pressurized air in the control room will be relieved through a fan (OV-119). The fan is not safety related.

The operation of the control room HVAC system during the isolation phase is described in Subsection 9.4.1.2.4.

9.4.1.2.2 Computer Room Floor Cooling System

See Subsection 9.4.1.2.1 for description of Computer Room Floor Cooling System.

9.4.1.2.3 Control Structure H&V Systems (OV-103)

The system is shown on Figures 9.4-1 and 9.4-2 . Design parameters are listed in Table 9.4-2.

The system consists of two 100 percent capacity redundant air handling units (one operating and one standby). Each unit contains a ventilation filter bank, two electric heating coils, two chilled water cooling coils, a centrifugal fan, and a fan discharge damper. The units are connected to a common Seismic Category I supply and return duct system that distributes supply air throughout the areas served and returns room air to the units. The control structure H&V equipment is located within a Seismic Category I structure. All components in each redundant system are powered from an independent Class 1E power source.

The control structure H&V systems supplies a fixed quantity of outside air through the outside air duct system and recirculates conditioned air to maintain space requirements. In each space the return air, exhaust air, and exfiltration will be balanced to:

a) Provide supply air to maintain specified temperature conditions in the battery room floor, elevation 771 ft and furnish ventilation air to the battery room exhaust system (OV-116). For further discussion see Subsection 9.4.1.2.6.

b) Maintain space temperature conditions in the following areas:

Elevation 783 ft -	H&V equipment room
Elevation 753 ft -	Upper relay rooms, upper cable spreading rooms, and electrician's office
Elevation 741 ft -	Technical Support Center
Elevation 714 ft -	Lower cable spreading rooms

The control structure H&V system is designed to handle the heating and cooling load for the spaces mentioned above.

The chilled water systems that supply the control structure H&V unit cooling coils and their operation are described in Subsection 9.2.12.

In each fan system air temperature controllers, (one heating and one cooling), sense the temperature in the return air. The heating controller regulates the output of the electric heating coil through an SCR (Silicon Control Rectifier). Chilled water flow through the cooling coil is modulated by a three way mixing valve controlled by the cooling controller.

9.4.1.2.4 Control Structure Emergency Outside Air Supply System (OV-101)(CSEOASS)

This system consists of two 100 percent redundant Seismic Category I filter trains complete with fans as described in Subsection 6.5.1.2. Each redundant system is powered from an independent Class IE power source.

The system as shown on Figures 9.4-1 and 9.4-2. Design parameters are listed in Table 9.4-2.

Each filter train is connected to a common Seismic Category I duct system. The emergency outside air supply system is located within a Seismic Category I structure.

When the emergency outside supply system is in operation the volume of air flowing in the main supply duct is continuously indicated and recorded in the control room. The upstream HEPA filter pressure differential is also continuously recorded in the control room. The loss of airflow will automatically trip and isolate the operating train and start the standby train. Both loss of airflow and high pressure differential across the above filter are alarmed in the control room. Temperature detectors monitor the temperature of the charcoal adsorber. The pre-ignition temperature (set at 190°F) is alarmed in the control room and indicated locally on the unit's heat detection control panel which is located on elevation 806 ft. The ignition temperature (set at 450°F) is also alarmed in the control room and indicated locally. In addition, the ignition temperature signal will automatically trip the train and open the fire protection water deluge valves (see Subsection 6.5.1.2). Note also that there are two valves in series. The air operated valve adjacent to the filter housing is Seismic Category I. The upstream valve is a standard fire protection deluge valve.

The temperature differential across the filter train is also monitored. When this temperature differential increases to 30°F it is alarmed in the control room. When it decreases to approximately 10°F this is also alarmed in the control room and also trips the supply fan. The low temperature differential is normally an indication of the failure of the electric heater.

The outside air for the control room floor cooling system (OV-117), computer room floor cooling system (OV-115), control structure H&V system (OV-103), and the SGTS equipment floor ventilation system are taken from a common outside air intake. The outside air intake is missile protected and connected to Seismic Category I design duct systems.

During normal operation, the outside air is drawn through the ducts and distributed to each system as described in above.

When high radiation is detected at the outside air intake, this is annunciated in the control room, and the outside air is automatically diverted through the emergency outside air filter system (OV-101). The isolation dampers at the control room relief air duct are closed and all the nonsafety related systems are tripped. The smoke removal system dampers are normally closed.

When chlorine is detected at the outside air intake, the high chlorine signal is annunciated in the control room. All isolation dampers in the control structure envelope close automatically. This includes:

a) All exhaust isolation dampers

b) Control room relief air duct isolation dampers

c) All outside supply air isolation dampers including the normal and emergency outside air dampers.

The entire control structure is isolated before chlorine reaches the intake isolation dampers. All the nonsafety related systems are tripped. The smoke removal system dampers are normally closed. After control room isolation is initiated, the emergency outside air system (OV-101) can be started up and operated manually to recirculate and clean up space air in the control room. The outside air intake dampers remain closed during this mode of operation.

The high chlorine signal stops the battery room exhaust fans (OV-116 A and B).

When reactor building isolation is initiated, as described in Subsection 9.4.2.1, the emergency outside air system will automatically operate in the radiation isolation mode.

9.4.1.2.5 SGTS Equipment Room Heating and Ventilating Systems (OV-144 and OV-118)

The equipment in both the heating and the ventilating systems is 100 percent redundant. The systems are shown on Figures 9.4-1 and 9.4-3 . Design parameters are listed in Table 9.4-2. Each redundant system is powered from an independent Class IE power source.

The redundant equipment is connected to common Seismic Category I ductwork systems. There are two redundant heating system units each containing a ventilation filter, an electric heating coil, and a centrigual fan with a discharge damper. The two redundant ventilation systems each contain a fan and discharge damper. The heating units (0V-144) recirculate room air and the space thermostat controls the heating coil to maintain a minimum room temperature of 40°F.

The SGTS equipment room is ventilated by outside air that is introduced into the room through the outside air intake duct systems and exhausted by the ventilation system (0V-118) through the SGTS vent duct to atmosphere. The exhaust fan is controlled by a room thermostat that is set at 100°F.

9.4.1.2.6 The Battery Rooms Exhaust System (0V-116)

The system consists of redundant exhaust fans and redundant isolation dampers. Each redundant system is powered from an independent Class IE power source.

The system is shown on Figures 9.4-1 and 9.4-3 . Design parameters are listed in Table 9.4-2.

The individual battery rooms are not equipped with independent fans. The system is designed to operate with one fan on standby and one operating. The branch ducts to each battery room are connected into a common duct system.

The battery rooms' makeup air is introduced by the control structure H&V system (0V-103). The exhaust fan (0V-116) system is designed to exhaust air from each battery room and discharge through the SGTS vent duct to the atmosphere.

When chlorine is detected at the outside air intake, the chlorine isolation signal will trip the exhaust fan and close all redundant isolation dampers.

9.4.1.2.7 Smoke Removal System (0V-104)

The control structure smoke removal system is composed of two 100 percent capacity redundant centrifugal fans, normally open fire dampers, normally closed control dampers, and associated ductwork and control.

The system is shown on Figures 9.4-1 and 9.4-3 . Design parameters are listed in Table 9.4-2.

The hand control switches and associated status indicating lights of the fans and dampers are located in the control room on the fire protection control board.

Fire dampers are provided for each floor area under supervisory control. The dampers are normally closed. When fire has been detected and suppressed, the smoke removal fan is manually switched on and the redundant fan is put into automatic standby. Smoke from affected floor areas is then purged by manually opening the appropriate fire dampers. If the operating fan fails it is alarmed in the control room and the standby unit automatically starts. Smoke is exhausted to the turbine building exhaust vent.

The smoke removal system will not be operated during accident conditions.

9.4.1.2.8 Access Control and Lab Area Supply System (0V-105)

The system is shown on Figures 9.4-1 and 9.4-3. Design parameters are listed in Table 9.4-2.

The access control and laboratory area supply unit contains a ventilation filter bank, an electric heating coil, chilled water cooling coils, and a centrifugal fan.

The system is designed to maintain a temperature of 75°F ± 5°F for personnel comfort and provides makeup air during normal operation.

The control switch, located in a local panel, starts the supply fan. The system discharge air temperature controller directly controls the amount of chilled water flowing into the chilled water cooling coils. This temperature controller also controls the output of the electric heating coil through a step controller.

Control structure isolation signals will shut down the supply fan, which will in turn stop all other interlocked systems (see Subsection 9.4.1.5).

There are seven zones in this system; each zone contains an electric reheat coil. A zone thermostat will maintain each zone temperature. Each zone reheat coil is interlocked with the supply fans.

The control switch and associated status indicating lights and instruments are located in a local control panel installed in the turbine building Unit 1, H&V equipment room.

A unit heater in the personnel access corridor entry at elevation 676 ft-0 in. will temper infiltration from the entrance. The unit heater will maintain temperature under the control of a local thermostat.

9.4.1.2.9 Lab Fume Hood Makeup Air System (0V-106), Contaminated Filter Units Exhaust System (0V-114) and Hood Exhaust Filter Systems

These systems are shown on Figures 9.4-1 and 9.4-3. Design parameters are listed in Table 9.4-2.

The laboratory fume hood makeup air system consists of an air handling unit equipped with a ventilation filter bank, an electric heating coil and a centrifugal fan, with the associated ductwork, dampers and controls. The makeup air supply unit supplies auxiliary air type fume hoods that are located in the control structure laboratories at elevation 676 ft-0 in.

The fan control is interlocked with the contaminated filter units exhaust fans. When the fan is in operation the system discharge air temperature controller controls the output of the electric heating coil, to provide tempered makeup air to the fume hood.

9.4.1.2.10 Control Room Toilet (0V-107), Control Room Kitchen (0V-108), Access Control Area Toilet (0V-112), and Access Control General Area (0V-113) Exhaust Fan Systems

These systems are shown on Figures 9.4-1 and 9.4-3. Design parameters are listed in Table 9.4-2.

The hand control switches of all the fans are located on the local control panel. The operation of these fans is strictly manual and they will operate only if the access control and lab area supply fan (0V-105) is operating. Since the control room may be directly exposed to the outside environment through the control room toilet and kitchen exhaust systems, fail-closed, redundant isolation dampers in series are installed at the intake of the toilet and kitchen exhaust fans. These isolation dampers are automatically closed by the high outside air radiation or chlorine signals.

9.4.1.2.11 Radiation Chemical Laboratory, Sample Room and Decontamination Area Hood Exhaust Filter Systems

These systems are shown on Figures 9.4-1 and 9.4-3. Design parameters are listed in Table 9.4-2.

Each hood exhaust filter train consists of a pre-filter, HEPA filter, charcoal filter, fire detection system, filter train inlet and outlet dampers, and associated controls and instrumentation.

The inlet and outlet dampers are manually operated through a hand switch located at the hood or on a local control station as in the case of the decontamination area hood exhaust filter system.

Local differential pressure indicator is provided across each filter and high differential pressure across the HEPA filter is alarmed at the local control panel.

The fire detection system has four temperature sensors. Two sensors to monitor pre-ignition (set at 190°F) and ignition (set at 450°F) temperatures are embedded in the charcoal filter. Two more identical sensors are located at the air outlet end of the filter train. The pre-ignition and ignition temperatures alarm directly in the control room. At ignition temperature, the inlet and outlet dampers are automatically closed to isolate the whole filter train, and the water deluge valve is opened to flood the charcoal filter.

Temperature indicators are provided at the local control panel to monitor the temperature of the inlet and outlet air of the filter train. In addition, high temperature differential between the inlet and outlet air of the filter train is alarmed at the local control panel.

9.4.1.3 Safety Evaluation

All safety related control structure and control room systems are designed to maintain functional integrity during a design basis accident. Each system is provided with redundant equipment and controls to maintain uninterrupted room air circulation, cooling and heating for personnel comfort and instrument functioning. All equipment is located within the control structure, a protected Seismic Category I structure. During loss of offsite power, standby power is available from the standby diesel generators for the continued operation of all safety related equipment.

The single failure criteria for active safety related equipment are met by using redundant equipment and controls and automatically switching from one redundant system to the other. Active equipment such as fans, controls, dampers, pumps, and chillers are redundant. Passive system components such as supply and return ductworks systems are common.

For failure mode and effect analysis see Tables 9.4-16 through 9.4-21 for safety related modes of operation.

All ductwork and supports for the safety related systems meet the Seismic Category I requirements.

The control room HVAC system is designed to maintain environmental conditions within the space as specified for habitability and equipment operation under the normal and abnormal operating conditions. All equipment in the system is designed to Seismic Category I requirements, except the humidification equipment.

A radiation monitoring system is provided in the outside air intake to detect high radiation and initiate measures to ensure that personnel safety and equipment functions are not impaired and that the requirements of 10CFR20 are satisfied. In the event of a high radiation condition, the normal outside air supply to the system is diverted through the emergency outside air filter train before being delivered to the control room. All isolation dampers except those on the battery room exhaust and emergency outside air intake will be closed. These operations will be annunciated in the control room.

The emergency outside air filter train and the control room shielding envelope are designed to limit the occupational dose level as required by General Design Criterion 19 of 10CFR50.

The introduction of a predetermined quantity of outside air maintains the control room and the other areas served by the control room floor system at a positive pressure with respect to surrounding areas. This positive pressure is maintained during all the plant operating conditions except when the system is in the recirculation mode.

Chlorine detectors are provided at the outside air intake to detect chlorine in the environment. In the event of a high chlorine concentration, all isolation dampers will be closed, including those on the outside air intakes, and all nonsafety related fan systems will be tripped. The control room HVAC system will be manually switched to the recirculation mode to cycle room air through the emergency outside air filter train (charcoal adsorber) system.

A smoke removal system is provided with a capability of purging any one of the floor areas under supervisory control.

9.4.1.4 Tests and Inspections

The control room HVAC system and its components are thoroughly tested in a program consisting of the following:

a) Factory and component qualification tests (see Table 9.4-1)

b) Onsite preoperational testing (see Chapter 14)

c) Onsite subsequent periodic testing (see Chapter 16).

Written test procedures establish minimum acceptable values for all tests. Test results are recorded as a matter of performance record, thus enabling early detection of faulty performance.

All equipment is factory inspected and tested in accordance with the applicable equipment specifications, codes, and quality assurance requirements. Refer to Table 9.4-1 for details of inspection and testing.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.4.1.5 Instrumentation Requirements

The control switches and the associated status indicating lights of all safety related equipment of the control room and control structure HVAC systems are located in the control room. Control switches and indicating lights of all isolation dampers are located on local control panels. Except for the isolation dampers between Units 1 and 2, status indicating lights of all isolation dampers are duplicated in the control room.

Although the control switches of isolation dampers are remote from the control room, the redundant isolation dampers are always in series and are designed to fail safe in the closed position. In addition, the redundant isolation signals of the isolation dampers are wired so that they override their corresponding control switch.

The control switches and status indicating lights of all nonsafety related equipment, except the smoke removal system, are located on the equipment, on local panels, or on local control stations. Control switches and indicating lights of the smoke removal system fans and dampers are located on the fire protection panel in the control room.

All safety related equipment failures, such as fans failing to establish airflow when required, are alarmed in the control room on one of two separate annunciators (one annunciator for Division I equipment and one for Division II). In addition, the following are alarmed in the control room:

a) High chlorine gas in the outside air

b) High radiation in the outside air

c) High-high radiation in the outside air (upscale)

d) Outside air radiation detection systems failure

e) High temperature (pre-ignition) in a charcoal adsorber of the Control Structure Emergency Outside Air Supply Systems (CSEOASS)

f) High-high temperature (ignition) in a charcoal adsorber of the CSEOASS

g) High pressure differential across an upstream HEPA filter of the CSEOASS

h) High temperature differential across a filter train of the CSEOASS

i) Normal outside air supply isolation damper failed closed in the absence of control structure isolation signals

j) Loss of control power to the electronic instruments

k) Battery rooms isolation damper failed closed in the absence of high chlorine isolation signal.

The outside air radiation level is continuously recorded in the control room.

Failure of nonsafety related equipment is alarmed on local control panels and is retransmitted to the control room as a trouble alarm.

All safety related equipment with maintained contacts type control switches have automatic input to the bypass indication systems (see Section 7.5) when the switch is in the "OFF" position.

Instruments of the safety related systems are seismically qualified and redundant to meet the single failure criteria. In particular, the emergency outside air supply systems (atmosphere cleanup) are instrumented to comply with the requirements of Regulatory Guide 1.52. Airflow in these systems is indicated, recorded, and alarmed (loss of flow) in the control room. Upstream HEPA filter pressure differentials are recorded and alarmed (high pressure differential) in the control room.

9.4.2 REACTOR BUILDING VENTILATION SYSTEM

The following systems are covered under this subsection:

a) Reactor building HVAC systems for normal operation

b) Safety related and RCIC air cooling systems include; 1) Emergency Core Cooling Systems (ECCS) and RCIC pump rooms unit coolers and 2) emergency SWGR room and load center room cooling units (for normal and emergency operation).

The ESF reactor building recirculation system is covered in Subsection 6.5.3, and the standby gas treatment system (SGTS) is described in Subsection 6.5.1.

9.4.2.1 Reactor Building HVAC Systems for Normal Operation

The secondary containment is divided into three isolated ventilation zones. Zones I and II surround respective Units 1 and 2 containments below the floor at elevation 779 ft-1 in. Zone III includes Units 1 and 2 secondary containments above the floor at elevation 779 ft-1 in. including the refueling floor. (See Figures 9.4-4 and 9.4-5).

This section discusses Unit 1 secondary containment HVAC systems (Zones I and III of Unit 1). The Unit 2 secondary containment HVAC systems (Zones II and III of Unit 2) are identical to those described for Unit 1.

Each of the ventilation zones is provided with independent HVAC systems designed to operate during plant normal operation and during shutdown. Zone III systems will function during normal fuel handling and storage operation. The recirculation system and SGTS will be used after a fuel handling accident.

9.4.2.1.1 Design Basis

The reactor building HVAC system is designed to accomplish the following objectives during stable and transient operating conditions, from start-up to full load to shutdown:

a) Provide filtered outside air at approximately 2.9 air changes per hour

b) Maintain air flow from areas of lesser to areas of greater potential contamination

c) The building will not exceed the maximum temperature of 100°F in the refueling floor, 120°F in the reactor building portion of the steam pipe tunnel, and 104°F or 100°F in the various remaining areas of the building, except for a maximum of 110°F in the fuel pool filter demineralizer rooms, cleanup filter demineralizer rooms, in the surge tank vault, and regenerative and nonregenerative heat exchanger rooms

d) The building's minimum temperature will not be below 60° F

e) Maintain the secondary containment at minimum negative pressure of approximately 0.25 in. wg.

f) Supply ventilation or purge air to the primary containment

g) Provide ventilation, cooling, and heating to the ECCS pump rooms during normal plant operation. For safety related cooling see Subsection 9.4.2.2.

h) Filter air exhausted from areas of greater potential contamination (equipment rooms - all zones)

i) Monitor radiation in the unfiltered air from the Zone III exhaust system (V-213), and isolates the Zone III portion of the secondary containment on a high radiation signal

j) Provide for radiation sampling in the reactor building exhaust vent

k) Provide for a transit time of exhaust air from the radiation monitors to the isolation dampers of Zone III unfiltered exhaust system, greater than the damper closing time plus the radiation monitor response time

l) Isolate appropriate ventilation zone or zones and start the recirculation system upon receipt of the reactor building isolation signal

m) Isolate supply and exhaust ducts of rooms containing high energy pipelines after a pipe break.

The portion of the reactor building ventilation system that is associated with the recirculation system is safety related. The remaining portion of the ductwork within the secondary containment boundary is not safety related; however, it is seismically designed and analyzed to ensure that it will not damage the safety related equipment and systems. Safety classifications are shown on the airflow diagrams, Figures 9.4-4 and 9.4-5.

Monitoring of radiation levels in the spent fuel pool is discussed in Subsection 12.3.4.

9.4.2.1.2 System Description

The air flow diagrams for the reactor building are shown on Figures 9.4-4 and 9.4-5. System design parameters are listed in Table 9.4-3. Cooling water is supplied to the air cooling coils in the HVAC systems by the reactor building chilled water system described in Subsection 9.2.12. The controls and instrumentation associated with each system are an integral part of that system. The instruments and controls are shown on Figures 9.4-6 through 9.4-9.

All the equipment of the Unit 1 air handling systems are located in two H&V equipment rooms (El. 779 ft-1 in. and 799 ft-1 in. east of the spent fuel pool). The two rooms are outside the secondary containment boundary.

Access to any zone from outdoors, to H&V equipment rooms, or access between the zones is through air locks with airtight doors on the potentially contaminated side and conventional doors on the clean side. The air lock is continuously exhausted.

Zone I Supply Unit System (V-202) and Zone III Supply Unit System (V-212)

Each system supplies the respective zone with conditioned 100 percent outdoor air.

Each system includes, in the direction of air flow: outdoor air intake; filters bank, four 35 percent capacity each electric heating coils; four 35 percent capacity each, chilled water cooling coils; two 100 percent capacity fans; two disc type isolation dampers, in-series, distribution ductwork with dampers, supply air outlets, and associated controls.

Zone I Equipment Compartment Exhaust System (V-206) and Zone III Filtered Exhaust System (V-217)

Each system exhausts air from the respective zone equipment compartments and from rooms with the higher potential for radioactive contamination.

Each system includes, in the direction of air flow: distribution ductwork with exhaust registers and dampers; two disc type isolation dampers, in-series, two filter trains, approximately 55 percent capacity each; two 100 percent capacity fans; system discharge ductwork connecting to the reactor building exhaust vent, and associated controls. Each filter train contains prefilters, upstream HEPA filters, charcoal adsorber (6 in. deep vertical bed), and downstream HEPA filters.

Zone I Exhaust System (V-205) and Zone III Exhaust System (V213)

Each system exhausts air from the respective zone areas of lesser radioactive contamination potential. Each system includes, in the direction of air flow: distribution ductwork with exhaust registers and dampers; two disc type, isolation dampers in-series; two 100 percent capacity fans, discharge ductwork connecting to the reactor building exhaust vent, and associated controls.

V-201 Recirculation System (See Subsection 6.5.3.2.)

During normal plant operation the reactor building ventilation systems maintain the design temperature and pressure in the respective zones of the secondary containments of Units 1 and 2. The supply air systems (V-202 and V-212) supply the respective zone with constant air volume. Fan discharge dampers of the exhaust systems V-205 and V-213 are modulated by appropriate pressure differential controllers to maintain a negative pressure of approximately 0.25 in. wg in the secondary containment. The air flow of exhaust systems V-206 and V-217 is controlled by their pressure differential controllers to maintain the air flow from areas of lesser to areas of greater potential contamination.

Each supply system is provided with two air temperature controllers to control the temperature of the air leaving the fans (one for heating and one for cooling). The output of the electric heating coils is regulated by a step controller. Chilled water flow through the cooling coils is modulated by three-way mixing valves controlled by the cooling controller. On supply system V-212 (Zone III) there is a manual cooling controller override to provide full capacity cooling during the refueling operation.

All panel mounted instruments and controls, including fan manual switches, are installed on local control panels. A group alarm from each panel is annunciated in the control room. In addition a "no ventilation" alarm for each zone is annunciated in the control room.

The chilled water cooling coils are protected from freeze-up by two sets of temperature switches (low and low-low) mounted on the face of each coil. (See Subsection 9.2.12.)

Two back draft isolation dampers (BDID) in-series, are provided on supply and exhaust ducts of selected rooms (see Figure 9.4-4) housing ECCS pumps or containing high energy piping. Each BDID is provided with a pressure differential switch that trips the release mechanism to close the damper on sensing high pressure inside the room.

The trip circuits are connected to uninterruptable dc power supply.

Only one fan of each system is running during plant normal operation. On loss of air flow from the running fan, its associated discharge damper closes and the standby fan starts automatically.

Failure of both fans on any one system to establish air flow will result in an automatic shutdown of the remaining ventilation systems in that zone. The loss of the zone ventilation is alarmed in the control room.

Redundant radiation monitors are provided on three branch ducts of V-213 (Zone III exhaust system). A high radiation signal from any monitor will automatically isolate Zone III as described in Subsection 9.4.2.1.3. Exhaust air transit time between the monitors and the V-213 system isolation dampers is greater then the combined time of damper closure and the monitor response.

The systems' intake louvers and exhaust vents are not safety related and are outside the secondary containment; therefore, no provisions for missile protection are made for these components.

The primary containment is purged at a rate of 10,500 cfm. This amount of air is diverted to the primary containment from the Zone I supply system. From there the air is filtered through the SGTS and exhausted to the environment.

The reactor building exhaust vent is provided with a radiation sampler. High radiation level in the exhaust air is alarmed in the control room.

9.4.2.1.3 Safety Evaluation

The Reactor Building Ventilation system is housed within the Seismic Category I reactor building. Wind and tornado protection is discussed in Section 3.3. Flood design is discussed in Section 3.4. Missile protection is discussed in Section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in Section 3.6. Environmental design considerations are discussed in Section 3.11.

The secondary containment isolation is the only active safety related function of the normal operation of the reactor building HVAC system. The system passive safety related function is the use of related ductwork in the reactor building recirculation mode which is discussed in Subsection 6.5.3.

The isolation dampers which are used for secondary containment isolation are redundant (two in series), fail close, disc type dampers, operated by spring loaded air cylinder. If an active failure disables one of the two dampers, the other one is able to perform the isolation function.

All hand control switches and indicating lights for safety related isolation dampers are located in the control room.

The reactor building ventilation system is started manually from the local control panel. The primary containment purge supply air damper is manually operated from the control room.

The appropriate ventilation zones of the secondary containment are automatically isolated and the recirculation system is actuated upon receipt of one of the following signals:

Signal	Isolates Zone(s)
High radiation in the refueling floor exhaust ducts	III
High radiation in the railroad access shaft exhaust duct (Unit 1 only)	III
High pressure in the drywell	I* & III
Low reactor water level	I* & III
A manual signal from the control room	III or I* & III

* Or Zone II if the signal is from Unit 2 drywell or reactor.

Any of the above isolation signals, will result in the following automatic sequence for the affected zone or zones:

a) Trip all running ventilation fans and prevent standby units from operating

b) Close normally open isolation dampers (two in-series separating safety related from nonsafety related portions of each system)

c) Open normally closed isolation dampers (two, in parallel), on each duct connecting the recirculation system fans into the ventilation system ductwork to be used in the recirculation mode of operation

d) Start the recirculation system (Subsection 6.5.3)

e) Start the SGTS (Subsection 6.5.1)

During the plant normal or emergency operation the following events will result in the secondary containment's not being maintained at a pressure below atmospheric:

a) Loss of offsite power (emergency operation)

b) A normally opened isolation damper on any of the supply or exhaust systems failing in a closed position results in a system trip

c) Loss of the reactor building ventilation due to failure, malfunction of system components.

The loss of the reactor building's ventilation will be alarmed in the control room, and the isolation of the affected ventilation zone(s) of the secondary containment may be initiated manually. As a result, the preferred air flow from areas of lesser to areas of higher potential contamination may not be maintained; however, the affected secondary containment will be maintained at a negative pressure of approximately 0.25 in. wg.

Each ventilation system is provided with two 100 percent capacity fans. When failure of a running fan or its discharge damper is detected by a flow switch, the respective standby fan will automatically start. On failure of a fan or its discharge damper, the preferred air flow pattern will not be affected.

The failure of a BDID in a closed position will result in a loss of ventilation for the equipment room affected and trouble alarm on a local HVAC panel. Each trouble alarm will be sounded in the control room as the panel group alarm. Indicating lights on the local panel will identify the failed damper, which can be manually reset to the open position.

Refer to Subsection 9.4.2.1.5 for a list of abnormal conditions which are alarmed on the local HVAC control panels.

High outlet air and water temperature from each drywell unit cooler is indicated, as an alarm light, on a local control panel and annunciated in the control room.

The operational degradation of ventilation system components can be detected by direct equipment status indication (indicating lights for damper position, fan running status) or can be concluded based on abnormal temperature, differential pressure, alarms, and indication. Corrective action can then be taken.

9.4.2.1.4 Tests and Inspections

All tests and inspections described in Table 9.4-1 apply to the reactor building HVAC systems, which are used during normal operation.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.4.2.1.5 Instrumentation Requirements

The following systems or equipment are provided with hand switches and status indicating lights in the control room:

a) Each isolation damper, except for air locks isolation dampers

b) Each ECCS pump room unit cooler

c) Each SWGR cooling unit.

Hand switches and status indicating lights for the balance of the air handling systems, and for the air locks isolation dampers are located on local HVAC control panels.

The following alarms are annunciated in the control room:

a) Fan failure, each safety related fan

b) High supply air temperature for emergency SWGRs

c) High radiation in Zone III exhaust ducts, and the downscale signal from radiation monitors

d) High and low flow conditions in the reactor building exhaust vent radiation sampler

e) High flow in ducts interconnecting Zone I or II ventilation systems with the recirculation system fans. This is to detect isolation dampers which fail open in the ventilation zone which is not being recirculated.

f) Loss of ventilation in any of the three ventilation zones (Zone I, II or III)

g) Group alarm on closure of any of the steam flooding backdraft isolation dampers

h) Group alarm from each HVAC local control panel

i) Manually induced inoperability of the safety related systems is alarmed and continuously indicated in the control room on bypass indication system

j) Pre-ignition and ignition temperatures of charcoal absorbers.

In addition, the following conditions are alarmed on the local HVAC control panels and transmitted to the control room as a group alarm:

a) Fan failure, each non-safety related fan

b) High pressure drop across filters

c) High or low pressure in the zone or one of the potentially contaminated areas

d) Low air temperature entering cooling coils (freeze protection), on coils handling outside air

e) High pressure differential across the upstream HEPA filter bank of each filter train

f) Pre-ignition and ignition charcoal absorber temperatures

g) High temperature differential across charcoal absorber bed

All instruments and controls performing safety related functions are qualified to the Seismic Category I requirements.

The redundancy and separation of instrumentation and controls conforms to the redundancy and separation of the equipment they control or monitor.

9.4.2.2 Safety Related and RCIC Air Cooling Systems

The following equipment and systems are covered under this heading:

1) The RHR, HPCI, RCIC, and core spray pump rooms unit coolers.

2) Emergency SWGR cooling units with associated ductwork.

9.4.2.2.1 Design Basis

The above cooling systems are designed to:

a) Maintain temperature at a maximum of 130°F in the ECCS pump rooms after a DBA.

b) Maintain emergency SWGR room temperature below a maximum of 130° F after a DBA and 104° F during plant normal operation.

The coolers, associated ductwork, and supporting structures are safety related and are Seismic Category I.

9.4.2.2.2 System Description

General

The safety related air cooling systems are shown on the reactor building, Zone I, air flow diagram (Figure 9.4-4). See Table 9.4-4 for the system design parameters. All coolers are supplied with emergency service water. The emergency SWGR room cooling units also contain chilled water cooling coils, for use during normal operation. The controls and instrumentation associated with each system are an integral part of that system. The instruments and controls are shown on Figure 9.4-6.

ECCS and RCIC Pump Room Unit Coolers

Each ECCS and RCIC pump room unit cooler recirculates and cools the respective room air, and is capable of carrying the following cooling loads:

a) RHR and core spray pump room coolers - total cooling load associated with operation of a single ECCS pump (one out of two in each room)

b) RCIC and HPCI pump room coolers - the total room cooling load.

Each unit cooler consists of a cabinet with a cleanable emergency service water cooling coil, a direct drive vane-axial fan mounted outside of the cabinet, and except for RHR pump room coolers, a sheetmetal transition section with a supply air register. The unit coolers are mounted adjacent to the pumps they serve, and they start automatically when the pump starts. Each cooler is also provided with a hand switch in the control room for manual operation. During plant normal operation, the reactor building ventilation system is used to maintain the design conditions in the ECCS and RCIC pump rooms (see Subsection 9.4.2.1).

Each pair of RCIC and HPCI room coolers is provided with additional hand selector switches in the control room for selection of the lead and standby units. In addition each cooler is provided with a temperature switch to transfer to the standby unit on detection of high air temperature at the discharge of the running unit and to annunciate this condition in the control room.

Emergency SWGR and Load Center Room Cooling Units

Two 100 percent capacity cooling units are provided for the emergency SWGR and load center rooms. Each unit consists of a cabinet with the following components, in the direction of the air flow: prefilters, emergency service water cooling coil, a chilled water cooling coil, and a belt driven centrifugal fan. The air discharge of each unit is connected to a common supply air duct.

Air enters the unit inlet directly from the surrounding area. Duct penetrations for the supply air, and the transfer grilles for the return air to and from each room, are redundant and parallel, and are furnished with fire protection dampers. During normal operation chilled water flow through the coil is modulated by a three-way mixing valve controlled by the discharge air temperature controller.

After a DBA the chilled water is not available, and the emergency service water cooling coils are used. Emergency service water flow through the coils is unrestricted with no supply air temperature or water flow control.

Only one cooling unit is running during plant normal or emergency operation. When loss of air flow or high discharge air temperature from the running unit is detected that unit's discharge damper closes and the fan is tripped. The standby unit starts automatically. Both the high temperature and the running unit trip are alarmed in the control room.

Each unit is provided with a three position (auto, start, stop) hand switch in the control room, a flow switch and a temperature switch both mounted on a common supply air duct.

9.4.2.2.3 Safety Evaluation

For failure mode and effect analysis see Table 9.4-5, for safety related modes of operation.

All units, ductwork and supports, and other systems components, except for discharge air temperature pneumatic control loop, meet Seismic Category I requirements and single failure criteria.

9.4.2.2.4 Tests and Inspections

With the exception of items (4) and (7) through (9) all tests and inspections described in Table 9.4-1 apply to the coolers and associated ductwork system.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.4.2.2.5 Instrumentation Requirements

The discussion for instrumentation may be found in Subsection 9.4.2.1.5.

9.4.3 RADWASTE BUILDING VENTILATION SYSTEM

9.4.3.1 Design Bases

The Radwaste Building HVAC systems have no safety related functions.

The Radwaste Building Heating, Ventilating, and Air Conditioning (HVAC) systems are designed to operate during normal operations and accomplish the following objectives:

a) Provide a supply of filtered and tempered outside air to all areas of the building

b) Maintain airflow from areas of lesser to areas of greater potential contamination

c) Maintain the building spaces below the following maximum temperatures:

General Areas	100°F
Equipment Rooms	104°F
Tank Rooms	120°F

d) Maintain the building minimum temperature of 40°F

e) Maintain the building at a slightly negative pressure to minimize exfiltration to the outside atmosphere

f) Filter through charcoal and particulate filters all air exhausted from:

The tank vent system

Liquid radwaste filters

Liquid radwaste demineralizer

Spent resin tank

g) Discharge all air exhausted from the space through particulate filters to the turbine building exhaust vent.

9.4.3.2 System Description

The airflow diagrams for the radwaste building are shown on Figures 9.4-10 and 9.4-11. Systems design parameters are listed in Table 9.4-6. Cooling water is supplied to the air cooling coils of the HVAC supply unit by the radwaste building chilled water system described in Subsection 9.2.12. The instrumentation and controls, shown on Figure 9.4-12, are considered an integral part of their systems.

The building HVAC supply and exhaust units are located in the equipment rooms on elevation 691 ft 6 in. The tank exhaust system fan and filters are located in the filter room on elevation 646 ft 0 in.

The supply system contains two 100 percent capacity fans, a housing containing one bank of particulate filters, a bank of electric heating coils, and a chilled water cooling coil. Filtered and tempered air is distributed throughout the building in quantities designed to maintain required temperatures and airflow toward areas of higher potential contamination. The building exhaust system contains two 100 percent capacity fans and two 50 percent capacity filter housings, each with a bank of high efficiency particulate filters (HEPA) and a bank of prefilters upstream. This exhaust system is balanced to maintain the flow of air within the building as described.

In addition to the Building Supply System a Recirculation System supplies cooling air to the off-gas area. The Recirculation System includes a unit cooler which is interlocked with the Building Supply System.

The tank exhaust system provides a means of filtering and venting air from tanks and equipment housed in the radwaste building. A single fan and filter train are employed for this purpose. An electric duct heater upstream of the filter is used to lower the humidity of the air, as necessary, to ensure proper filter operation. The filters, in the direction of air flow are prefilter, HEPA, and charcoal. Since the flow of air from tanks and equipment varies, space air is admitted as required to maintain system volume.

Both exhaust systems use the same duct to transport the filtered air to the turbine building exhaust vent.

The building exhaust system and the supply system are interlocked so that complete failure of either system will shut down the entire building ventilation system. This condition is a "total loss of radwaste building ventilation" and is alarmed directly in the control room.

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The control switch of each of the fans in the building exhaust system is on a local control panel. The two exhaust fans are interlocked so that failure of the operating fan will automatically start the standby unit, isolate the failed unit, and alarm in the local control panel. This system is manually started.

The variable inlet vanes of each exhaust fan enable the system to vary exhaust airflow to maintain the building at a slightly negative pressure.

Each of the two filter trains is manually set up for operation by opening the train's inlet and outlet dampers through the control switch on the local control panel.

The two supply system fans are interlocked in the same manner as the exhaust fans. The standby automatically starts on failure of the operating fan. Fan failure is alarmed at the local control panel.

When the supply system is operating, the system discharge air temperature controller directly controls the amount of chilled water entering the systems cooling coils. This temperature controller also controls the output of the banks of electric heaters.

The tank exhaust system starts to operate automatically in conjunction with the building exhaust and supply systems. The control switch of this system's fan is on the local control panel and fan failure is alarmed on the same panel. High temperature and high-high temperature in the charcoal adsorber are alarmed directly in the control room. The tank exhaust system filter train high differential temperature is alarmed on the local panel.

When the charcoal adsorber approaches ignition temperature, the high-high temperature switch trips the fan and opens the fire protection deluge system valve; this floods the housing containing the charcoal bed.

The electric heater in the duct upstream of the filter train operates only if the filter train exhaust fan is running. The electric heater is used to limit the humidity of the air entering the charcoal filter.

9.4.3.3 Safety Evaluation

The failure of the radwaste building HVAC systems or their components will not compromise any safety related system or prevent a safe shutdown of the plant.

The charcoal filters contain fire detection instruments which annunciate high and high-high charcoal temperature on the HVAC panel in the control room. The high-high temperature signal causes the filter deluge system to flood the housing to extinguish the fire. The exhaust air is checked for radiation by the radiation monitors in the turbine building exhaust vent.

9.4.3.4 Tests and Inspection

The system will be preoperationally tested in accordance with the requirements of Chapter 14. Maintaining normal conditions verifies that the system is performing properly during operation.

9.4.3.5 Instrumentation Requirements

All hand control switches of the radwaste building HVAC systems are on the local control panels in the radwaste building. The local panels have annunciators which transmit a trouble alarm to the control room if any abnormal condition exists in the radwaste building HVAC systems.

The following abnormal conditions are alarmed at the local panel:

a) All fan failures

b) High or low indoor/outdoor pressure differential

c) High pressure differential across the filter of the supply system

d) High temperature differential across the tank exhaust system filter train

e) High pressure differential across the HEPA filter of the tank exhaust system

f) Phase B overcurrent on the exhaust fan motors feeder breakers

g) Supply system electric heater trouble

h) High pressure differential across the HEPA filters of the exhaust system.

The following are alarmed directly in the control room:

a) Loss of ventilation in the radwaste building

b) High and high-high temperatures in the tank exhaust system charcoal adsorber.

A pressure differential controller, with pressure sensors inside and outside the building, modulates the variable inlet vanes of the exhaust fans to maintain the building at a slightly negative pressure. A pressure differential indicator is also provided on the local control panel.

Pressure differential indicators are also provided locally at all the filters including the tank exhaust system charcoal adsorber.

A temperature sensor is provided at the supply system outside air intake plenum which automatically operates the radwaste building chilled water systems (see Subsection 9.2.12).

9.4.4 TURBINE BUILDING VENTILATION SYSTEM

9.4.4.1 Design Basis

The turbine building heating, ventilating, and air conditioning (HVAC) systems have no safety related functions.

The turbine building HVAC systems are designed to operate during normal operation and accomplish the following objectives:

a) Provide a supply of filtered and tempered air to all areas of the building

b) Maintain airflow from areas of lesser to areas of greater potential contamination

c) Maintain building spaces below the following maximum temperatures:

General Areas	104°F
Electrical Rooms	104°F
Mechanical Areas	120°F

d) Maintain the building minimum temperature of 40°F

e) Maintain the Turbine Building except the generator bay area, at a slightly negative pressure to minimize exfiltration to the outside atmosphere

f) Recirculate and cool space air to reduce exhaust volume

g) Exhaust air from potentially contaminated spaces through particulate and charcoal filters

h) Discharge all exhaust air through the turbine building exhaust vent

i) Provide cooling air to the motor generator sets.

9.4.4.2 System Descriptions

The airflow diagram for the turbine building HVAC systems are shown on Figure 9.4-13. System design parameters are listed in Table 9.4-7. Cooling water is supplied to the HVAC cooling coils by the turbine building chilled water system described in Subsection 9.2.12. The instruments and controls shown on Figure 9.4-14, should be considered an integral part of the system.

The turbine building supply unit and associated return fans are located in the H&V equipment room at elevation 762 ft 0 in. This room also contains the recirculation unit, the filtered exhaust unit, and the MG set cooling unit. The condenser area unit coolers are installed in the condenser area at elevation 676 ft 0 in. The condensate pump room unit coolers are located in the condensate pump room at elevation 656 ft 0 in.

The systems described are for the Unit 1 turbine building. Unit 2 systems are similar.

Supply System (V101)

The supply system unit housing contains two 100 percent capacity fans, a bank of chilled water cooling coils, a bank of electrical heating coils, and a bank of particulate filters. The unit is connected to a ductwork system with outlets, dampers, and controls to distribute tempered air throughout the building to maintain temperatures and airflows so that they meet the stated requirements. The air entering the supply unit contains at all times sufficient outside air for ventilation. This minimum quantity of outside air will be increased up to 100 percent of system airflow when outside air temperature makes this practicable.

The supply system operates only if the return air system (V104) is operating. The control switches of the supply fans are on the local control panel. The air supply system is put into operation by manually starting one supply fan and setting up the second supply fan in standby mode. The standby fan starts automatically on failure of the operating fan.

Return Air System (V104)

The return air system housing contains two 100 percent capacity fans that, through the associated ductwork system, exhaust air from clean areas. Depending on requirements of the supply system (V101), this air may be either exhausted directly to the turbine building vent or returned to the intake of the supply unit.

The return air fans control switches are on the local control panel. The system is put into operation by manually starting one return fan and setting up the second fan in standby mode. The standby fan starts automatically on failure of the operating fan. The return air system is tripped on failure of the supply system (V101) or the filtered exhaust system (V106).

Recirculation System (V105)

The recirculation unit housing contains two 100 percent capacity fans, a chilled water cooling coil and a bank of particulate filters. The housing connects to both a supply and a return ductwork system complete with outlets and dampers.

The recirculation system supplies and returns air to and from areas as required for cooling but does not affect access and clean areas.

The hand control switches of the recirculation system fans are located on the local control panel. The system is put into operation by manually starting one fan and setting up the second fan in standby mode. The standby fan starts automatically on failure of the operating fan.

Filter Exhaust System (V106)

The filtered exhaust system contains two 100 percent capacity fans and two filter housings, of 50 percent capacity each. Each filter housing contains prefilters, downstream HEPA filters, charcoal filters, and upstream HEPA filters. Air from potentially contaminated areas in the turbine building is routed through the filtered exhaust system before it is discharged to the atmosphere via the turbine building exhaust vent.

The filtered exhaust system operates only if the supply system (V101) is operating. The control switches of the filtered exhaust system fans are on the local control panel. The system is put into operation by manually starting one fan and setting up the second fan in standby mode. The fan operates with the two filter trains. The standby fan starts automatically on failure of the operating fan.

MG Set Cooling System (V103)

The MG set cooling system unit housing contains two 100 percent capacity fans and a bank of particulate filters. The air entering the MG set cooling unit contains at all times sufficient outside air to cool the MG sets. This minimum quantity of outside air will be increased to 100 percent of system air when outside air makes it practicable. The system exhausts directly to the turbine building exhaust vent.

The control switches of the MG set cooling system fans are located in the control room on the same board as the controls and instrumentation of the MG sets. This system is put into operation by manually starting one fan and setting up the second fan in standby mode. The standby fan starts automatically on failure of the operating fan.

Condenser Area Cooling (V113)

The condenser area unit cooler system consists of two pairs of unit coolers. Each unit cooler is sized for 50 percent of the load and its housing contains a fan and a bank of chilled water cooling coils. Each pair of unit coolers discharges cooled air through a common duct to the condenser area.

The control switch of each fan is on a local control panel. One fan on each pair of unit coolers is manually started and the remaining units set up in standby mode. Each pair of unit coolers is controlled by a room thermostat that automatically starts the standby unit when the temperature rises to 120°F. The standby unit also starts automatically on failure of the operating unit.

Condensate Pump Room Cooling (V112) 2

The condensate pump room unit cooling system consists of four 2
unit coolers. Each unit cooler is sized for 33.3 percent of the 14
load and its housing contains a fan and a bank of chilled water
cooling coils. Each pair of unit coolers discharges cooled air
through a common duct to the condensate pump room.

The control switch of each fan is on a local control panel.
Under normal operating conditions three fans are manually
started. The fourth unit is set in standby mode. The standby 3
unit starts automatically on failure of any of the three
operating units or when the pump room temperature rises to 104°F. 2

9.4.4.3 Safety Evaluation

The turbine building HVAC systems have no safety related functions.

The turbine building HVAC systems are designed to maintain airflows from clean areas to potentially contaminated areas and from areas of potentially lower level contamination to areas of potentially higher level contaminations, then through a filter exhaust system.

All systems are provided with redundant fans; upon any failure of any operating fan, the standby fan will be automatically started.

The main exhaust charcoal filters contain a fire detection system that annunciates on the control room panel when high temperature occurs. If high-high (ignition) temperature is detected in the charcoal, the deluge system fills the filter housing with water to extinguish the fire. The exhaust air is monitored for radiation by the radiation detection system in the exhaust vent outlet.

9.4.4.4 Tests and Inspections

All components are tested and inspected as separate components and as integrated systems. After the ductwork system is installed and airflows are measured and adjusted to meet design requirements, all instruments are calibrated to the design conditions. The system will be preoperationally tested in accordance with the requirements of Chapter 14.

Periodic flow measurements will be taken to verify the design condition in order to ensure operability and integrity of the system.

9.4.4.5 Instrumentation Requirements

All the hand control switches of the turbine building HVAC equipment are on the local control panel in the turbine building, except for the control switches of the fans in the MG set cooling system which are located in the control room. The local control panel has an annunciator that transmits a trouble alarm to the control room if any abnormal condition occurs in the turbine building HVAC systems exits.

The following abnormal conditions are alarmed at the local control panel:

a) All fan failures

b) High temperatures at the condenser area

c) High temperatures at the condensate pump rooms

d) High pressure differential across the upstream HEPA filter of the filtered exhaust system

e) High differential temperature across the filter train of the filtered exhaust system

f) High and low building pressures

g) High and low pressure differential between the generator area (clean) and condenser area (contaminated)

h) High pressure differential across the filter of the supply system

i) High differential pressure across the filter of the recirculation system

j) High differential pressure across the filter of the MG set cooling system

k) Supply fans (V101), return fans (V104), exhaust fans (V106), and MG set cooling fans (V103) motor feeder breakers phase B overcurrent

l) Supply system electric heater trouble.

The following abnormal conditions are alarmed directly in the control room:

a) Loss of ventilation in the turbine building

b) Pre-ignition and ignition temperatures of the filtered exhaust system charcoal filters

c) Turbine building exhaust vent radiation sampling station sample airflow high or low

d) Turbine building exhaust air high radiation

e) Turbine building exhaust air radiation monitoring system component failure (downscale)

f) Turbine building exhaust air high-high radiation (upscale)

Two radiation indicators and a radiation recorder in the control room continuously monitor the air exhausted from the turbine building.

Local pressure differential indicators are provided across all the filters. Pressure differential indicators are also provided on the local control panels to monitor the building pressure as well as differential pressures between clean and contaminated areas.

The condenser area and the condensate pump rooms are provided with room thermostats. These thermostats automatically start their associated unit coolers when the temperature rises above set point.

Inlet and outlet air temperatures of the recirculation and supply systems and the filter trains of the filtered exhaust system are displayed on temperature indicators at the local control panel.

9.4.5 PRIMARY CONTAINMENT VENTILATION SYSTEM

9.4.5.1 Design Basis

The primary containment air cooling systems are designed to accomplish the following objectives during stable and transient operating conditions from start-up to full load to shutdown:

a) Maintain temperatures in the various spaces within specified limits. The general drywell area will be maintained at an average temperature of 135° F, maximum not to exceed 150° F. The control rod drive area design temperature is 135° F, while maximum allowable temperature is 165° F. The area around the recirculation pump will be maintained at 128° F. The drywell head area design temperatures are 135° F average and 150° F maximum.

b) Provide for the primary containment air purge (See Subsection 9.4.2.)

c) Prevent concrete structures within the containment from exceeding the maximum design temperature of 150°F locally.

d) During post LOCA conditions selected drywell air cooler systems are designed to mix the drywell atmosphere to prevent hydrogen concentration build-up. This is the only safety related function performed by the dry well unit coolers.

The cooling systems including the ductwork that services the CRD area, the head area, and one of the systems serving the general drywell area are safety related. All other cooling systems are seismically analyzed to ensure that they present no hazard to the safety related equipment and systems. Safety classification and seismic categories are shown on Figure 9.4-15. Pipe whip has not been considered if the pressure differential between the Drywell atmosphere and the inside of the duct is less than 6" H_2O and the fluid density is low.

9.4.5.2 System Description

The air flow diagram for the drywell is shown in Figure 9.4-15. The duct layout is shown in Figure 9.4-22. Design parameters are shown in Table 9.4-8. Cooling water to the air cooling coils is provided by the reactor building chilled water system or, on loss of offsite power, the reactor building closed cooling water system.

The controls and instruments associated with each system are shown on Figure 9.4-16, and should be considered as an integral part of that system.

The drywell air flow system contains 14 unit coolers (7 pairs), 1 (8000 CFM) fan per cooler, and 1 cooling coil per cooler. Each cooler has an individually ducted supply system. Functionally the unit coolers are arranged in pairs for standby operation. The two units of each pair are physically separated. During the high speed mode of operation units on standby will start automatically on loss of air flow in the running cooler.

The unit coolers are assigned (in pairs) to specific areas of the drywell as follows:

V411A&B - RPV support skirt flange area and reactor shield annulus. Cooling air is supplied through two ring headers, each with 12 evenly spaced penetrations through the reactor shield feeding outlets in the skirt flange area. From there the air is forced through the annulus between the RPV insulation and the shield and is exhausted to the drywell general area. Each supply air opening is furnished with a dispersing plate to prevent direct impingement of cold air against the RPV skirt.

V414A&B - <u>Safety related systems</u> - serving the RPV head space and the main steam relief valve area. Cooling air is supplied to the head area through two outlets 180° apart for maximum air mixing. The air is exfiltrated through four openings in the seal plate into the top of the drywell.

V415A&B - Safety related systems - serving the CRD area and the lower portion of the drywell. Cooling air is supplied from two openings 180° F apart, located in the lower elevation of the CRD pipe space. CRD pipe openings, at the top of the space, are used to allow the air to exfiltrate into the drywell general area.

V412A&B, V413A&B, V416A&B, V417A&B - Systems V416A&B are safety related. Each system, except V417A&B, supplies its total air flow at the top of the drywell directly below the seal plate. Systems V417A&B supply a portion of the cooling air in the vicinity of the main steam relief valves. The supply duct outlets, at the top of the drywell, are arranged tangentially around the RPV, so that an even circular air flow pattern is maintained for cooling and air mixing.

The cooling units are connected to ductwork on the discharge side only. Return air enters unit inlet directly from the space. Physically the units are dispersed around the RPV in the lower section of the drywell between el 704 ft-0 in. and 714 ft-0 in.

The unit coolers are provided with two speed motors for operation at low speed under the drywell post LOCA conditions and during the intergrated leak rate test, and at high speed under normal conditions. The impellors are subject to a 125% overspeed test to provide assurance that they will not generate missiles.

Each drywell unit cooler is provided with the following controls (See Figure 9.4-16 for control diagrams):

a) A four position switch - high; low; auto high; stop - (located in the control room)

b) A pressure differential switch across each fan (flow detection function) - (local)

c) Temperature sensors on inlet and outlet - (local)

d) Temperature sensor on cooling coil leaving water line - local (See Subsection 9.2.8.)

e) Local high outlet air temperature alarm, and a group alarm in the control room

f) Local high chilled water outlet temperature alarm, and a group alarm in the control room

g) Fan failure alarm in the control room

h) Fan starter switch bypass indication in the control room, for the safety related unit coolers only.

In addition a common local temperature indication and alarm panel, for both air and chilled water, is in the reactor building outside the primary containment.

Ambient temperature of various areas of the primary containment is monitored (See Subsection 6.2.1.1). High temperature in these CRD areas will automatically start the CRD area standby unit cooler. High temperature detected in the drywell area outside the CRD area will automatically start all six (6) standby unit coolers. In the event that the average air temperature in the drywell cannot be maintained with the standby units, the reactor will be shutdown in accordance with the technical specifications.

All coolers, including standbys, can be operated manually from the control room, if necessary to control the drywell temperature.

All operating modes of the drywell coolers are shown in Table 9.4-9.

9.4.5.3 Safety Evaluation

The Primary Containment Ventilation system is housed within the Seismic Category I reactor building. Wind and tornado protection is discussed in Section 3.3. Flood design is discussed in Section 3.4. Missile protection is discussed in Section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in Section 3.6. Environmental design considerations are discussed in Section 3.11.

Low speed operation of coolers V414A&B, V415A&B, and V416A&B is the only safety related function of the system. For failure mode and effect analysis see Table 9.4-10.

For high speed operation, during normal plant operation, none of the unit coolers have any safety related function. Fans are started manually. If a fan fails to start or fails during operation, the standby fan starts automatically when on "Auto-High" mode. The failure is annunciated in the control room. This is accomplished by use of pressure differential switches across each fan.

To ensure continuous operation during loss of offsite power, all drywell unit cooler fans and controls are on the emergency power supply. Units A are on Division I Power Supply and Units B are on Division II Power Supply.

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9.4.5.4 Tests and Inspection

With the exceptions of items (4) and (6) through (9), all the tests and inspections described in Table 9.4-1 apply to the drywell unit coolers.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.4.5.5 Instrumentation Requirements

Each unit cooler is controlled from the control room by a four position starter switch.

Failure of any fan is alarmed in the control room. For each cooler, high air and water discharge temperatures are alarmed individually on the local control panel, and in the control room, as a group alarm. Safety related cooler fan starter switch bypass is indicated in the control room.

Fan starter switch circuit for the safety related coolers are safety related. All other controls and instrumentation, including alarms, are not safety related. The safety related switches are qualified to Seismic Category I requirements. Isolating relays are provided to separate safety related from non safety related control circuits.

9.4.6 REFUELING AND SPENT FUEL AREA VENTILATION SYSTEM

The refueling and spent fuel area ventilation system is part of Zone III ventilation system described in Subsection 9.4.2.

The following features are provided to control air distribution in the spent fuel area in order to reduce concentration and spread of airborne radioactive contaminants within the refueling floor:

a) Exhaust air registers high over the spent fuel pool, in addition to general exhaust registers on the west wall of the refueling floor. (See Figure 9.4-17).

b) The exhaust air ducts from the two register locations are provided with control dampers, controlled by a common, two position selector switch in a local HVAC control panel. The two positions of the switch set the dampers in alternate combinations of minimum and maximum flow.

Depending on the pool water temperature and the refueling floor exterior enclosure surface temperatures, the appropriate exhaust air flow combination can be selected to make use of prevailing convection currents induced by those conditions.

9.4.7 DIESEL GENERATOR BUILDING VENTILATION SYSTEM

9.4.7.1 Design Basis

The H&V system for the diesel generator building has a safety related function. It is designed to maintain a suitable environment for the diesel generators and their accessories during all modes of operation. To ensure proper diesel generator operation, each of the four emergency diesel generator rooms is individually ventilated and heated not to exceed a maximum design room temperature of 120° F and a minimum design room temperature of 72° F.

9.4.7.2 System Description

Each diesel generator room is provided with a separate ventilation system as shown in Figure 9.4-18 and 9.4-19. Each system is designed to modulate outside/return air flow ratio from 0 to 100 percent depending on the respective room cooling demand. Design parameters are listed in Table 9.4-11.

Each supply fan starts with its associated diesel or when the room temperature, sensed by a start temperature switch, exceeds approximately 95°F and continues to run after the diesel stops until the temperature in the room is below the stop thermostat cutout setting approximately 78° F. The fan discharge air temperature is controlled by modulating outside air intake, exhaust, and recirculation dampers. Ventilation is provided by infiltration when the diesels and fans are off. The discharge air controller starts to modulate the dampers when the temperature exceeds 92°F. Circulating fans, located in the basement of each diesel generator room, circulate air between the basement and main floor. These fans are manually started by a local hand switch. These fans are not safety related. Heating for each room is provided by thermostatically controlled electric unit heaters that operate when the room temperature falls below approximately 72° F. The basement is heated with electric wall heaters, which are controlled by individual thermostats. The heating systems are not safety related.

The ventilation and combustion air is protected from dust by locating the air intake/combustion air filter in a separate compartment inside the building about 25 ft. above the grade (676'-0) elevation, see Fig. 9.5.27. Further dust protection is provided by covering the ground around the diesel generator and Turbine Building with grass, gravel and asphalt. The west side of the Turbine Building will be covered with gravel up to the centerline of the plant, the east side for about 300 ft. and to south for 200 ft. from the building will be covered with grass and asphalt.

Considering the location of the air intake, the combustion air dust exposure is minimum.

During the Unit 2 construction some of the yard grading will be in progress in the summer of 1982; however, the location of this work is about 700 ft. south from the air intake filters. All yard work around the Diesel Generator Building will be completed prior to startup.

The fans are not run except during engine operation. During engine operation the louvers are controlled by room temperature to minimize outside air requirements.

In addition to the design features which minimize the impact of dust on the diesel generator operation, the preventative maintenance program for the control cabinets include requirements for cleaning out dust accumulation when the equipment is checked.

9.4.7.3 Safety Evaluation

Each of the four diesel generator fan systems is located in a separate room within the Seismic Category I Diesel General Building. The ventilation system required for heat removal from each room is safety related and designed to Seismic Category I requirements. For failure mode and effect analysis see Table 9.4-12. Wind and tornado protection is discussed in Section 3.3. Flood design is discussed in Section 3.4. Missile protection is discussed in Section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in Section 3.6. Environmental design considerations are discussed in Section 3.11.

9.4.7.4 Test and Inspection

With exception of items 4 thu 12 all tests and inspection described in Table 9.4-1 apply to safety related components of the diesel generator ventilation system, with the addition of the manufacturer's short motor test report. The test reports provide documentation for each ventilation fan motor, giving manufacturer's data for the following tests:

a) Running light current (no load)

b) Power input

c) High potential

d) Bearing inspection

e) Calculated locked rotor current.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.4.7.5 Instrumentation Requirements

Each diesel generator ventilation system can be individually controlled from the control room. The system can be started manually or automatically when in the auto mode, the diesel start signal or the tripping of the high temperature switch causes the ventilation system to operate. In addition to the fan start and stop temperature switches a low temperature and two high temperature switches in each diesel room alarms in the control room in case of abnormal temperature conditions. Failure of any ventilation component resulting in loss of air flow also alarms in the control room.

9.4.8 ENGINEERED SAFEGUARD SERVICE WATER PUMPHOUSE VENTILATION SYSTEM

9.4.8.1 Design Basis

The H+V system for the ESSW pumphouse has a safety related function. It is designed to maintain a suitable environment for the ESW and RHRSW pumps and their associated accessories. To insure proper pump operation each of the two separate areas is individually ventilated and heated not to exeed a maximum design temperature of 104°F and a minimum design temperature of 60°F.

9.4.8.2 System Description

Each of the ESW and RHRSW pumps is provided with a separate ventilation system as shown on Figure 9.4-18 and 9.4-19. Each system is designed to modulate outside/return airflow ratio from o to 100 percent depending on the cooling demand. Design parameters are listed in Table 9.4-13. Each supply fan starts with its associated pump, or when temperature sensed by a start temperature switch exceeds approximately 95°F, and continues to run when the pump is shut off until the temperature in the room is below the stop thermostat cut-out setting (67° F). The fan discharge air temperature is controlled by modulating the outside air intake, exhaust and recirculation dampers. The exhaust and intake dampers are designed to fail close in order to prevent freeze up in the event of damper control malfunction. Ventilation of the pump house when the pumps are not in operation will be provided by infiltration. Heating for each room is provided by thermostatically controlled electric unit heaters, which operate when the room temperature falls below 60° F.

9.4.8.3 Safety Evaluation

The eight fans and components serving the ESSW pumphouse are located in the Seismic Category I ESSW pumphouse structure, which is divided into two areas with a missile proof separation wall. The ventilation systems required for heat removal are safety related and designed to Seismic Category I requirements. The heating system is not safety related.

For failure mode and effect analysis see Table 9.4-14. Wind and tornado protection is discussed in Section 3.3. Flood design is discussed in Section 3.4. Missile protection is discussed in Section 3.5. Protection against dynamic effects associated with the postulated rupture of piping is discussed in Section 3.6. Environmental design considerations are discussed in Section 3.11.

9.4.8.4 Test and Inspections

With exception of items 4 thru 12, all test and inspections described in Table 9.4-1 apply to safety-related components of the ESSW pumphouse ventilation system, with the addition of the short motor manufacturer's test report. These test reports provide documentation for each ventilation fan motor giving manufactuer's data for the following tests:

a) Running light current (no load)

b) Power input

c) High potential

d) Bearing inspection

e) Calculated locked rotor current

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.4.8.5 Instrumentation Requirements

Each ESSW pumphouse ventilation system can be individually controlled from the control room. The system can be started manually or automatically. When in the auto mode, the corresponding pump start signal or the tripping of the high temperature switch will cause the ventilation system to operate. In addition to the fan start and stop temperature switches a high temperature switch is located in the vicinity of each pump along with a low temperature detector for each of the two pumping bays. The tripping of any of the switches, alarms the control room of abnormal temperature conditions. Failure of any ventilation component, resulting in loss of air flow, also alarms in the control room.

9.4.9 CIRCULATING WATER PUMPHOUSE AND WATER TREATMENT BUILDING HVAC

9.4.9.1 Design Basis

The HVAC systems for the circulating water pumphouse and water treatment building have no safety related functions and are only used to maintain a suitable environment for the circulating and service water pumps, the water treatment equipment and all associated accessories.

To ensure proper pump operation, the circulating water pump room is heated and ventilated not to exceed a maximum design temperature of 104° F, during plant operation, and a minimum design temperature of 40° F during plant shutdown. The diesel driven fire pump room (located in the circulating water pumphouse) is designed to maintain a minimum of 70° F when the pump is not in operation, and a maximum temperature of 107° F during pump operation. The water treatment building minimum design temperature is 60° F, with the exception of the lab, toilet, and janitor closet, where the minimum design temperature is 72° F. Ventilation is provided for the water treatment building.

9.4.9.2 System Description

The circulating and service water pump area is ventilated by 20 roof mounted exhaust fans, controlled by their individual thermostats. Outside air is drawn through the louvers/dampers, which are located at ground level on the front side of the building, and exhausted through the roof. Each fan and its associated damper is controlled by a thermostat, which is located adjacent to a pump motor.

The fire pump room is normally ventilated by a roof mounted ventilation fan which also exhausts air from the sump area of the pumphouse. During operation of the diesel fire pump, the inlet dampers open to provide outside air for combustion and cooling and when the room temperature reaches 90° F a thermostat will start the exhaust fan. When the temperature reaches 90° F and the diesel is not running, the thermostat will open the intake damper and start the exhaust fan.

The water treatment building is ventilated by a number of different fan systems. The main system contains two in-line axial fans, one operating continuously, the other being on standby. Air is exhausted from the acid storage room and the water treatment rooms (Elevations 676'-0" and 693'-0"), by the main system. Tempered air is drawn through transfer grills into the acid storage and water treatment rooms from the circulating pump room. Outside air is also drawn into the water treatment room (el. 693'-0"). If the water treatment room (el. 693'-0") temperature exceeds 90°F, a roof mounted exhaust fan starts and draws in additional outside air through a three position damper cooling the area.

The toilet and janitor's closet are ventilated by a common roof exhaust fan. The laboratory is also ventilated with a roof mounted exhaust fan.

See Figures 9.4-20 and 9.4-21 . Design parameters are listed in Table 9.4-15.

Heating for the circulating water pumphouse and the water treatment building is provided by electric unit heaters, base board heaters and cabinet convectors. Each of these heating devices is controlled by its individual thermostat.

9.4.9.3 Safety Evaluation

The HVAC systems for the circulating water pumphouse and water treatment building are not safety related, however, there is a redundant fan for the main ventilation system in the water treatment building. Two redundant unit heaters have been designed as back up heaters to assure 60° F temperature for the acid storage room.

9.4.9.4 Test and Inspections

All equipment will be tested after installation, to verify its design conditions. The system will be preoperationally tested in accordance with the requirements of Chapter 14.

9.4.9.5 Instrumentation Requirements

The HVAC systems in the circulating water pumphouse, and water treatment building are controlled by local thermostats or by locally mounted switches, which can override the thermostat. There are no HVAC systems in this building that can be controlled from the control room. High temperature in the vicinity of each circulating water pump is detected and alarmed individually on a local control panel and in the control room as a group alarm.

TABLE 9.4-1

VENTILATION SYSTEMS TESTS AND INSPECTIONS

1) GENERAL

a) All safety related components identified in Table 3.2-1 are designed, fabricated, installed, and tested under quality assurance requirements in accordance with Appendix B to 10CFR50.

b) For systems that must perform a safety related function, periodic in-service testing of all fans, valves, controls, and instrumentation in the systems will be performed. All motor-operated valves and dampers will be tested by opening and closing the valve or damper.

c) Equipment in Seismic Category I systems is required by specification to meet the seismic requirements for this project. Before each equipment item is shipped, the supplier of that item is required to submit an adequate analysis or applicable test data as evidence of compliance, which is approved by PP&L.

d) Systems designed to meet Seismic Category I requirements are subjected to a program of plant and field testing.

e) All standby units will be tested at periodic intervals to verify the operation of essential features. Periodic tests of the activation circuitry and the system components will be conducted during normal plant operation.

2) FANS

All centrifugal and propeller fans are tested in accordance with the AMCA Standard Test Code for Air Moving Devices, Bulletin 210. Vane axial fans are tested in the field for flow and pressure requirements. Blade setting adjustments are made to correct flow rates when necessary.

3) MOTORS

All motors are built, designed, rated, and tested in accordance with NEMA-MG-1. Category I motors will have certification for the NEMA tests required in Publication No. MG-1. Motors used within the containment comply with IEEE 334.

4) HEATING COILS

The heating coils are furnished in accordance with the requirements of UL 1096 and the National Electrical Code, Article 424. The heating coils are installed according to the National Fire Protection Association Pamphlets 90A and 90B.

5) COOLING COILS

The cooling coils are furnished in accordance with ASHRAE 33 and ARI Standard 410. All chilled water, service water, and emergency service water coils are hydrostatically and pneumatically tested. Category I coils are seismically qualified by analysis or testing on a shaker table. The Emergency Service Water Cooling coils have been tested in accordance with ASME subsection 3 ND-6200 and ND-6300.

6) MIST ELIMINATORS

All eliminators are built in accordance with MSAR 71-45, "Entrained Moisture Separators for Fine (1 to 10 microns) Water-Airstream Service". The eliminators are Seismic Category I and have been seismically analyzed.

7) PARTICULATE FILTERS (Supply Air)

The particulate filters are UL Class 1 approved under UL 900. The filter efficiency and performance is in accordance with ASHRAE Standard 52-68. The airflow resistance of the particulate filters is less than 0.35 in. wg (clean) and 1.0 in. wg (dirty) at rated flow (2000 cfm). The filters have an efficiency rating of 85 to 90% by dust spot test on atmospheric dust.

8) PREFILTERS (Used in Series with HEPA Filters)

The prefilters are certified to meet the standards for UL Class 1 filters. The airflow resistance of the prefilters at rated flow is less than 0.3 in. wg (clean) and 0.9 in. wg (dirty). The prefilters have an efficiency rating of 80 to 95 percent by the dust spot test on atmospheric dust.

9) HEPA FILTERS

a) Qualification Tests Prior to Installation

The HEPA filters are constructed in accordance with MIL-F-51079A, Filter Medium, Fire Resistant, High Efficiency, and MIL-F-51068C, (Filter, Particulate High-Efficiency, Fire Resistant). The filters are Type IIC (SGTS) and IIB (all others). The minimum tensile strength of the filter media is at least

2.5 lb/in. of width in accordance with the requirements of MIL-F-51079A.

The filter medium is securely fastened to the sides and ends of the filter frame with adhesive to seal the edges of the medium to the filter frame. Patching of holes or tears in the medium is not permitted.

The assembled filters are type tested in accordance with the requirements of UL 586 (High Efficiency Air Filter Units) to minimize fire hazards. The filters are approved UL Class 1.

Each filter has been tested for flow resistance at rated flow. The filter resistance does not exceed the rated pressure drop of 1 in. wg under this condition.

The filters have been rough handled with the Q110 Vibrating Machine, DLA 26-18-67, examined for damage and the DOP penetration determined in accordance with Section 4.3.4.1.

All filters have been subjected to acceptance tests made by an NRC quality assurance station. The filter efficiency exceeds 99.97 percent when tested with monodispersed, thermally generated DOP aerosol having a mean particle size of 0.3 micron.

Filters selected at random from the manufacturer's production line have been subjected to moisture, overpressure resistance, and filter dust loading tests in order to initially qualify the filters. The moisture and overpressure resistance tests were preformed in accordance with MIL-F-51068.

Each filter has been individually tested by the appropriate NRC quality assurance station at 100 percent and 20 percent of the rated capacity.

b) Preoperational Tests for Acceptance (performed in filter train housing)

Visual and dimensional checks of the housing and mounting frames were made in the field for conformance with design specifications. Nonconforming items are rejected and replaced with acceptable equipment.

After installation, in-place testing of the HEPA filter efficiency was conducted in accordance with Section 10 of ANSI N510-1975 (formerly ANSI N101.1-

1972). The tests are conducted at the rated airflow, using the DOP aerosol test equipment, test procedures, and test reports specified in ANSI N510-1975. The overall filtration efficiency is not less than 99.97 percent. When leaks that would result in inability to meet the specified system parameters exist, they are located and repaired by welding. The system is then tested again to ensure conformance with acceptance criteria.

10) CHARCOAL ADSORBERS

Charcoal adosrbers were tested as follows:

a) Qualification Tests Prior to Installation

1) Representative samples, taken from each batch of charcoal used for filling the adsorbers, were tested for adsorption efficiencies of radioactive elemental iodine, radioactive particulate iodine, and radioactive methyl iodine. The test methods were comparable to those shown in the Oak Ridge Laboratory Publication NSIC-65. The iodine loading in the test gas stream was about 0.01 mg/m³. The removal efficiencies and residence times were at least as follows for relative humidities up to 70 percent:

2 3/16 in. (0.25 sec. res. time)	95% elemental 95% organic
6 in. (0.75 sec. res. time)	99% elemental 99% organic
8 in. (1.0 sec. res. time)	99% elemental 99% organic

Calculations have been done to demonstrate that the residence times shown above are met.

2) Each charcoal adsorber cell was tested for leakage using the test method presented in ANSI 510 (formerly AEC Report DP1082). The tracer gas used in the test was either R-112 or R-11, (tetrachlorodifluorothane or trichlorofluoromethane). The tracer gas was mixed into the rated airflow in accordance with the above procedure. Leakage paths were identified and blocked by welding, as necessary to meet the limiting requirements on leakage. The pressure drop across the cell was measured during the tracer gas test.

TABLE 9.4-1 (Continued)

3) The percentage of impregnation on the charcoal as well as type was verified by random lot sampling.

4) In addition, tests were conducted in accordance with Paragraph 4.3 and Table #4 of RDT M16-1T to determine the following:

a) Particle size

b) Ignition temperature

c) Apparent density

d) Moisture content

e) Carbon tetrachloride activity.

b) In-place Testing of Adsorber

1) Refrigerant (R-11 or R-112) was introduced into the upstream side of the adsorber at a concentration of approximately 20 ppm at rated airflow. The downstream concentration was less than 0.25 percent of the upstream 20 ppm. No more than four tests were conducted on any given charcoal adsorber cell. No radioactive isotopes were used in the efficiency tests performed on the charcoal adsorbers.

2) The installed carbon adsorber filter bank was visually and dimensionally checked for conformance to the design specifications.

11) FILTER HOUSINGS

In addition to the housing manufacturer's shop tests, a field performance test was given to each housing. The leakage rate for each housing was less than 0.1 percent of the rated airflow in cubic feet per minute at 125 percent of the negative design pressure (-0.25 in. wg).

12) FILTER IN-SERVICE TESTS AND INSPECTIONS

a) The air filtering systems are subject to in-place testing before initial startup and after each filter or adsorber change. The test interval not to exceed one year, in accordance with the recommendations of ORNL-NSIC-65.

b) The periodic testing of the filter banks ensures that the filter bank performance is not degraded, through normal use or during standby service, to a level of below that assumed in the accident analyses. The test methods and sensitivities are the same as or equal to those for initial acceptance of the system components. Should the test results indicate that the performance of a component has fallen to the level assumed by the accident analyses, the component is replaced.

c) The results of all tests are made available upon completion of performance and acceptance by PP&L.

d) The following filter in-service tests and inspections are performed at regular intervals during plant life to determine that the filtration systems are functioning correctly:

1) With the fan running, readings of the differential pressure gages, mounted on the filter plenum are observed and recorded.

2) Prefilters are replaced when the pressure drop across them reaches 1.0 in. water column.

3) HEPA filters are replaced when the pressure drop across them reaches 3.0 in. water column.

4) Field leak tests are conducted after each change of HEPA filters in a system.

5) Field leak tests of HEPA filter banks are made with cold-generated dioctylphtholate, using a light-scattering aerosol photometer for measuring percentage penetration. An efficiency of less than 99.97 percent requires corrective action, as stated previously.

6) Corrective action after a leak test may consist of increasing the contact pressure on a seal, or replacement of a cell or cells. After corrective action is taken, an additional leak test is made.

7) Test of successive canisters of charcoal in the airstream of the charcoal adsorbers will be made every 6, 8, or 12 months (depending on carbon bed depth) and after the charcoal adsorber bank is installed. The test procedures are the same as used during initial batch qualification, for elemental

iodine and methyl iodide attenuating capacity. Tests for hardness, ignition temperature, and radioactivity are not made on these samples.

13) DUCTWORK

a) Leakage tests on all ductwork were conducted during construction.

b) All air distribution systems were tested and balanced to provide design air quantities at each outlet within a tolerance of ± 10 percent.

c) Category I ductwork is supported by seismically designed duct hangers.

d) All Category I ductwork is seismically designed and based on the analysis and test results which were conducted by Bechtel Power Corporation in April, 1976.

The test reports were based on:

1. Structural Design of Class I Seismic HVAC Ducts.

2. Report on testing of HVAC Duct Specimens

14) CONTROLS

a) All controls and instrumentation were tested prior to plant operation.

b) In-service tests and inspection procedures were incorporated in the plant operations manual and are performed at regular intervals during the life of the plant to show that the instruments are functioning properly. Recalibration, when necessary, is made at that time.

15) BRANCH TECHNICAL POSITION-ETSB NO. 11-2

All secondary filter systems used for normal ventilation exhaust, comply to Branch Technical Position - ETSB No. 11-2 Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Reactor Plants, with the exception of the following paragraphs (References are to BTP-ETSB 11-2)

a) Reference: Para 2.a Moisture separators are used only where moisture impingement may be a problem. Heaters are used to lower the relative humidity (R.H) when the ambient exceeds 70% R.H. None of

TABLE 9.4-1 (Continued)

the secondary non-safety related filter systems require either a moisture separator or heater to reduce the moisture content or lower the R.H.

b) Reference: Para. 2.c. The pertinent pressure drop which is instrumented to signal, alarm, and record in the control room is the pressure drop across the first HEPA filter.

c) Reference: Para 2.e. Overall design considerations include reduction of radiation exposures during routine maintenance and testing insofar as effectually possible. It is envisioned, however, that workers will not handle filter units after a design basis accident and will thereby avoid exposures associated with immediate post-accident filter handling. Accordingly, no efforts were made toward a unitized atmosphere cleanup train design in the interest of accident exposure reduction.

d) Reference: Para 3.c. Since none of the HEPA filters separators are exposed to potential iodine removal spray, the units are not designed for contact with the spray.

e) Reference: Para 3.d. In this section and all others where reference is made to ORNL-NSIC-65, the reference is understood to be ERDA 76-21 or ANSI N509 where appropriate.

f) Reference: Para 4.c. The spacing requirement is applicable to systems requiring operator access to remove filters and adsorber trays. Where unnecessary, the space is not provided, eg, gasketless carbon adsorbers which are filled and emptied externally.

g) Reference: Para 4.d. The length of pipe associated with manifolding would promote plate-out of the constituents of the sampled gas stream, thereby resulting in erroneous test results. The test probes are located in readily accessible locations; a minimum run of piping is used and manifolding is not employed.

TABLE 9.4-2

CONTROL STRUCTURE HVAC SYSTEMS DESIGN PARAMETERS

ITEM	H&V SYSTEM 0V-103	COMPUTER RM 0V-115	SAFETY RELATED SYSTEMS BATTERY RM 0V-116	CONTROL RM 0V-117	SGTS EXH SYS 0V-118	SGTS HTNG SYS 0V-144
Type	Air handling	Air handling	Indiv. fans	Air handling	Indiv. fans	Air handling
Number of Units	Unit 2	Unit 2	2 fans	Unit 2	2 fans	Unit 2
Flow rate each, CFM	24,450	30,000	4,000	26,020	3,500	3,000
Fan						
Type	Centrif.	Centrif.	Centrif.	Centrif.	Centrif.	Centrif.
Drive	Belt	Belt	Belt	Belt	Belt	Belt
No. of fans per unit	1	1	2 per sys	1	2 per sys	1
No. of running fans	1	1	1	1	1	1
Static pressure, each, in.	4.7	4.4	3	4.4	3	2.6
Motor hp, each	40	40	5	40	5	3
Cooling Coil			N/A		N/A	N/A
No. of coils per unit	2	2	-	2	-	-
Cooling capacity each, Btu/hr	1,025,680	775,635	-	509,471	-	-
Heating Coils		N/A	N/A	N/A	N/A	
No. of coils per unit	1	-	-	-	-	1
Heating capacity each, Btu/hr	443,690 (130 kw)	-	-	-	-	102,000 (30 kw)
Filters			N/A		N/A	
Quantity and size, in.	8-24x12x12 48-24x24x12	8-24x12x12 48-24x24x12	-	8-24x12x12 48-24x24x12	-	6-24x24x12
Pressure drop, in. wg						
Clean	0.5	0.5	-	0.5	-	0.5
Dirty	1	1	-	1	-	1
Efficiency(1), %	90%	90%	-	90%	-	90%

(1) Dust spot test on atmospheric dust

TABLE 9.4-2 (Continued)

ITEM	NON-SAFETY RELATED SYSTEMS SMOKE REMOVAL 0V-104	ACCESS & LAB EXH 0V-105	LAB FUME HOOD 0V-106	TOILET EXHAUST 0V-107	KITCHEN EXH 0V-108	ACCESS TOILET EXH 0V-112	ACCESS GEN EXH 0V-113
Type	Indiv. fans	Air handling	Air handling	Indiv. fans	Indiv. fans	Indiv. fans	Indiv. fans
Number of Units	2 fans	1	1	1	1	1	1
Flow rate each, cfm	6,000	10,510	1,950	350	200	1,200	5,000
Fan							
Type	Centrif.	Centrif.	Centrif.	Centrif.	Centrif.	Centrif.	Centrif.
Drive	Belt	Belt	Belt	Belt	Belt	Belt	Belt
No. of fans per unit	2 per sys.	1	1	1	1	1	1
No. of running fans	1	1	1	1	1	1	1
Static pressure each, in.	3	4.7	2.6	2.5	2.5	3	3.5
Motor hp, each	7.5	15	3	1	1	1.5	7.5
Cooling Coil	N/A		N/A	N/A	N/A	N/A	N/A
No. of coils per unit	-	2	-	-	-	-	-
Cooling capacity each, Btu/hr	-	1,050,000	-	-	-	-	-
Heating Coils	N/A			N/A	N/A	N/A	N/A
No. of coils per unit	-	1	1	-	-	-	-
Heating capacity each, Btu/hr	-	850,000 (250 kw)	119,000 (35 kw)	-	-	-	-
Prefilters	N/A			N/A	N/A	N/A	N/A
Quantity and size, in.	-	3-24x12x12 15-24x24x12	4[1]-16x25x2	-	-	-	-
Pressure drop, in. wg							
Clean	-	.5	0.18	-	-	-	-
Dirty	-	1	0.5	-	-	-	-
Efficiency[2], %	-	90%	[3]70-80%	-	-	-	-

[1] High velocity filters
[2] Dust spot test on atmospheric dust
[3] ASHRAE Standard 52-68 test method

TABLE 9.4-2 (Continued)

ITEM	SAFETY RELATED EMERG. O/A SUPP. OV-101	NON-SAFETY RELATED CONTAMINATED FILTER UNITS EXHAUST SYS.[5] OV-114			
Type	Built-up unit		Built-up unit		
Number of units	2		2		
Flow rate each, cfm	6,000		8,600[4]		
Fan					
Type	Centrif.		Centrif.		
Drive	Belt		Belt		
No. of fans per unit	1		1		
No. of running fans	1		1		
Static pressure each, in.	10		10		
Motor hp, each	20		30		
Cooling Coil					
No. of coils per unit	N/A	N/A	N/A	N/A	N/A
Cooling capacity each, Btu/hr	N/A	N/A	N/A	N/A	N/A
Heating Coils					
No. of coils per unit	2	N/A	N/A	N/A	N/A
Heating capacity each, Btu/hr	102,000 (30 kw)	N/A	N/A	N/A	N/A
Prefilters					
Quantity and size, in.	6-24x24x12	2-24x24x12	2-24x24x12	2-24x24x12	2-24x24x12
Pressure drop, in. wg					
Clean	0.3	0.2	0.2	0.2	0.2
Dirty	0.9	0.9	0.9	0.9	0.9
Efficiency[1], %	95	95	95	95	95
HEPA filter, upstream					
Quantity and size, in.	6-24x24x12	2-24x24x12	2-24x24x12	2-24x24x12	2-24x24x12
Pressure drop, in. wg					
Clean	1.0	0.8	0.8	0.8	0.8
Dirty	3.0	3.0	3.0	3.0	3.0
Efficiency[2], %	99.97	99.97	99.97	99.97	99.97
Charcoal Adsorber					
Type	Vertical bed	Horizontal tray	Horizontal tray	Horizontal tray	Horizontal tray
Depth, in.	4	2 3/16	2 3/16	2 3/16	2 3/16
Filter media	Impregnated activated carbon	Impregnated activated carbon	Impregnated activated carbon	Impregnated activated carbon	Impregnated activated carbon
Pressure drop, in. wg	2.2	1.2	1.2	1.2	1.2
Efficiency					
Removing inorganic iodine, %	99	95	95	95	95
Removing organic iodine, %	99	95	95	95	95

TABLE 9.4-2 (Continued)

ITEM	SAFETY RELATED EMERG. O/A SUPP. OV-101	NON-SAFETY RELATED CONTAMINATED FILTER UNITS EXHAUST SYS.(5) OV-114			
HEPA filter, downstream	(3)	N/A	N/A	N/A	N/A

(1) Dust spot test on atmospheric dust
(2) By MIL Standard 282 DOP test method 0.3
(2) All design parameters same as HEPA, upstream
(4) 0F133 - 1,400 cfm
0F136 - 600 cfm
0F139 - 600 cfm
0F142 - 700 cfm
Unfiltered Exh - 2,200
Other - 3,100
(5) System includes four banks of filters
(6) 70% Relative Humidity

TABLE 9.4-6

RADWASTE BUILDING HVAC SYSTEMS DESIGN PARAMETERS

ITEM	SUPPLY SYSTEM OV-301	OFF GAS AREA UNIT COOLERS OV-309 A & B	FILTERED EXH OV-302	FILT LRW TANK VENT OV-304	
Type	Built-up unit	Air Handling Unit	Built-up unit	Built-up unit	
Number of Units	2	2	2	1	
Flow rate each, cfm	50,000	26,000	50,000[1] fan 25,000[1] filter	1,000	
Fan					
Type	Centrif.	Centrif.	Centrif.	Centrif.	
Drive	Belt	Belt	Belt	Belt	
No. of fans per unit	1	1	1	1	
No. of running fans	1	1	1	1	
Static pressure, each, in.	4	3.8	12	6	
Motor hp, each	50	30	150	3	
Cooling Coil					
No. of coils per unit	6	2	N/A	N/A	3
Cooling capacity each, Btu/hr	292,000	1,085,000	N/A	N/A	
Heating Coils					
No. of coils per unit	6	N/A	N/A	1	
Heating capacity each, Btu/hr	1,020,000 (300 kW)	- -	N/A	25,500 (7.5 kW)	
	2 Filter banks each containing	- -	2 Filter banks each containing		
Prefilters					
Quantity and size, in.	24-24x24x12	N/A	30-24x24x12	1-24x24x12	
Pressure drop, in. wg					
Clean	.5	-	0.3	0.3	
Dirty	1	-	0.9	0.9	
Efficiency[2], %	85	-	95	95	
HEPA filter, upstream	N/A	N/A			
Quantity and size, in.	-	-	30-24x24x12	1-24x24x12	
Pressure drop, in. wg					
Clean	-	-	1	1	
Dirty	-	-	3	3	
Efficiency[3], %	-	-	99.97	99.97	
Charcoal filter	N/A	N/A	N/A		
Type	-	-	-	Horizontal tray	
Depth, in.	-	-	-	2 3/16	
Filter media	-	-	-	Impregnated activated charcoal	
Pressure drop, in. wg	-	-	-	1.2	

TABLE 9.4-

(Continued)

ITEM	SUPPLY SYSTEM OV-301	OFF GAS AREA UNIT COOLERS OV-309 A&B	FILTERED EXH OV-302	FILT LRW TANK VENT OV-304
Efficiency				
Removing inorganic iodine, %	-	-	-	99.9 min
Removing organic iodine, %	-	-	-	99.5 min
HEPA filter, downstream	N/A	N/A	N/A	N/A

(1) Normally two filter units operate in conjunction with one fan, each fan rated at 50,000 cfm
(2) Dust spot test on atmospheric dust
(3) By MIL Standard 282 DOP test method on 0.3 micron particles

TABLE 9.4-7

TURBINE BUILDING HVAC SYSTEMS DESIGN PARAMETERS

ITEM	SUPPLY SYS V-101	MG SET CLNG SUPP V-103	RETURN SYS V-104	RECIRC SYS V-105	FILT EXHAUST V-106	COND PP RM CLNG V-112	COND UNIT CLRS V-113
Type	Built-up unit	Built-up unit	Built-up unit	Built-up unit	Built-up unit	Unit coolers	Unit coolers
Number of units	2	2	2	2	2	4	4
Flow rate each, cfm	135,000	88,000	110,000	50,000	40,000(1) fan 20,000(1) filter	20,000	24,000
Fan							
Type	Centrif.	Centrif.	Centrif.	Centrif.	Centrif.	Centrif.	Centrif.
Drive	Belt	Belt	Belt	Belt	Belt	Belt	Belt
No. of fans per unit	1	1	1	1	1	1	1
No. of running fans	1	1	1	1	1	2	2
Static pressure each, in.	5.5	4	4	6	18	2	2.5
Motor hp, each	200	100	125	75	200	20	20
Cooling Coil		N/A	N/A		N/A		
No. of coils per unit	8	-	-	6	-	2	2
Cooling cap. each, Btu/hr	600,800	-	-	430,000	-	1,080,000	1,040,000
Heating Coils		N/A	N/A	N/A	N/A	N/A	N/A
No. of coils per unit	8	-	-	-	-	-	-
Heating cap. each, Btu/hr	765,000 (225 kW)	-	-	-	-	-	-
Prefilters (Filters)					2 filter housings, each containing		
Quantity and size, in.	56-24x24x12	36-24x24x12	-	24-24x24x12	21-24x24x12	-	-
Pressure drop, in. wg			N/A			N/A	N/A
Clean	0.5	0.5	-	0.5	0.3	-	-
Dirty	1.0	1.0	-	1.0	0.9	-	-
Efficiency(2), %	85	85	-	85	95	-	-
HEPA filter, upstream	N/A	N/A	N/A	N/A		N/A	N/A
Quantity and size, in.	-	-	-	-	21-24x24x12	-	-
Pressure drop, in. wg							
Clean	-	-	-	-	1.0	-	-
Dirty	-	-	-	-	3.0	-	-
Efficiency(3), %	-	-	-	-	99.97	-	-
Charcoal adsorber	N/A	N/A	N/A	N/A		N/A	N/A
Type	-	-	-	-	Vertical bed	-	-
Depth, in.	-	-	-	-	6	-	-
Filter media	-	-	-	-	Impregnated	-	-

TABLE 9.4-7

(Continued)

ITEM	SUPPLY SYS V-101	MG SET CLNG SUPP V-103	RETURN SYS V-104	RECIRC SYS V-105	FILT EXHAUST V-106	COND PP RM CLNG V-112	COND UNIT CLRS V-113
					activated charcoal		
Pressure drop, in. wg	-	-	-	-	3.1	-	-
Efficiency							
Rem. inorganic iodine,%	-	-	-	-	99.9 min		
Rem. organic iodine, %	-	-	-	-	99.5 min		
HEPA filter, downstream	N/A	N/A	N/A	N/A	(4)	N/A	N/A

(1) Normally two filter units operate in conjunction with one fan, each fan rated at 40,000 cfm.
(2) Dust spot test on atmospheric dust
(3) By MIL Standard 28 DOP test method on 0.3 micron particles
(4) All design parameters same as HEPA, upstream

TABLE 9.4-8

PRIMARY CONTAINMENT UNIT COOLERS DESIGN PARAMETERS

Item	Safety Related			Nonsafety Related			
	RPV Head Area V-414 A&B	CRD Area V-415 A&B	Drywell Gen. Area V-416 A&B	RPV Annulus V-411 A&B	Drywell Gen. Area V-412 A&B	Drywell Gen. Area V-413 A&B	Drywell Gen. Area V417 A&B
Type	Built-up	Built-up	Built-up	Built-up	Built-up	Built-up	Built-up
Number of units	2	2	2	2	2	2	2
Flow rate each, cfm	8,000	8,000	8.000	8,000	8,000	8,000	8,000
Fan							
Type	Vane-axial	Vane-axial	Vane-axial	Vane-axial	Vane-axial	Vane-axial	Vane-axial
Drive	Direct	Direct	Direct	Direct	Direct	Direct	Direct
No. of fans per unit	1	1	1	1	1	1	1
No. of running fans(1)	1	1	1	1	1	1	1
External static pressure, each, in.	2.5	2.5	2.5	2.5	2.5	2.5	2.5
High speed, rpm	1,770	1,770	1,770	1,770	1,770	1,770	1,770
Low speed, rpm	870	870	870	870	870	870	870
Motor hp, each Hi/Lo speed	10/5	10/5	10/5	10/5	10/5	10/5	10/5
Cooling Coil							
No. of coils per unit	1	1	1	1	1	1	1
Cooling capacity each, Btu/hr	520,000	520,000	520,000	520,000	520,000	520,000	520,000
Heating Coils	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Filters(2)							

(1) See Table 9.4-9 for other operating modes.
(2) "Throw-away" type roughing filters two in. thick are used during plant construction only. No filters are used during normal plant operation.

TABLE 9.4-11

DIESEL GENERATOR BLDG HVAC SYSTEMS DESIGN PARAMETERS

Item	Vent System OV-512[1]	Basement Vent. OV-511[1]
Type	Vane Axial	Tubular
Flow rate each, cfm	96,000	3,000
Fan		
Type	Propeller	Propeller
Drive	Direct	No. 9 Susp. Horiz.
No. of fans per diesel gen.	4	1
No. of running fans	1	1
Static pressure, each, in.	1	0.25
Motor hp, each	40	1.5
Cooling Coil	N/A	N/A
Heating Coils	N/A	N/A
Filters	N/A	N/A

(1) Typical for each emergency diesel generator

TABLE 9.4-13

ESSW PUMPHOUSE HVAC SYSTEMS DESIGN PARAMETERS

Item	Ventilation Sys. 1V-506(1)	Ventilation Sys. 0V-521(2)
Type	Vane Axial	Vane Axial
Flow rate each, cfm	12,500	10,000
Fan		
Type	Propeller	Propeller
Drive	Direct	Direct
No. of fans per ESSW Unit	2	4
No. of running fans	X1(3)	X1(3)
Static pressure, each, in.	1	1
Motor hp, each	5	5
Cooling Coil	N/A	N/A
Heating Coils	N/A	N/A
Filters	N/A	N/A

(1) Typical for each unit ESSW pump

(2) Typical for each common ESSW pump

(3) Additional fan(s) adjacent to the running fan(s), may be started, manually from the control loop, or automatically by the high temperature switch, when required to provide additional cooling.

TABLE 9.4-15

CIRC. WTR. PUMPHOUSE HVAC SYSTEMS DESIGN PARAMETERS

Item	Diesel Rm. Clng. 0V-506	Fire PP. & Sump PP. Rm. 0V-503	Toilet Roof Exh. 0V-501	Lab. Roof Exh. 0V-504	Circ. PP. Rm. Exh.(1) IV-503A&B IV-501A-H	Wtr. Trmt. Rm. Vent 0V-505	Wtr. Trmt. Rm. Vent 0V-502
Type	Roof Mounted	Roof Mounted	Roof Mounted	Roof Mounted	Roof Mounted	Roof Mounted	Tubular
Number of Units	1	1	1	1	10	1	1
Flow rate each, cfm	7,000	2,300	200	200	18,500	6,000	10,000
Fan							
Type	Centrif.	Centrif.	Centrif.	Centrif.	Propeller	Centrif.	Ducted Propeller
Drive	Belt	Belt	Belt	Belt	Belt	Belt	No. 9 Ceiling Susp
No. of fans per unit	1	1	1	1	1	1	2
No. of running fans	1	1	1	1	As req.	1	As req.
Total pressure, each, in.	0.5	0.5	0.5	0.5	0.5	0.5	1
Motor hp, each	2	.75	.25	.25	5	1.5	5
Cooling Coil	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Heating Coils	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Filters	N/A	N/A	N/A	N/A	N/A	N/A	N/A

(1) These are Unit 1 fans. An equal number exists for Unit 2.

TABLE 9.4-18

CONTROL ROOM & CONTROL STRUCTURE HVAC SYSTEMS
CONTROL STRUCTURE H&V SYSTEMS
FAILURE MODE AND EFFECT ANALYSIS

PLANT OPERATING MODE	SYSTEM COMPONENT	COMPONENT FAILURE MODE	EFFECT OF FAILURE ON THE SYSTEM	FAILURE MODE DETECTION	EFFECT OF FAILURE ON PLANT OPERATION
Emergency	Power Supply	Total loss of offsite power (LOOP)	None. The systems are redundant and are powered from separate standby diesel generators.	Alarm in the control room	No loss of safety function
Emergency (LOCA or high chlorine gas or radiation in outside air with or without LOOP)	Fans (0V103)	Loss of one fan	None. The standby unit automatically starts.	Alarm in the control room	No loss of safety function
Emergency (LOCA or high chlorine or radiation in outside air with or without LOOP)	Fans (0V103) outlet dampers	Damper failure	None. The dampers are redundant and are designed to fail safe in the closed position. When the damper fails close it trips and isolates its associated fan and the standby unit automatically starts.	Alarm in the control room	No loss of safety function
Emergency (LOCA or high chlorine gas or radiation in outside air with or without LOOP)	Isolation dampers (HD-07802) (HD-07824	Damper failure	None. Redundant isolation dampers are in series and are designed to fail safe in the closed position. Only one damper is needed to close and effectively isolate.	Damper position indication in the control room	No loss of safety function

TABLE 9.4-18 (continued)

PLANT OPERATING MODE	SYSTEM COMPONENT	COMPONENT FAILURE MODE	EFFECT OF FAILURE ON THE SYSTEM	FAILURE MODE DETECTION	EFFECT OF FAILURE ON PLANT OPERATION
Emergency (LOCA or high chlorine or radiation in outside air with or without LOOP)	Cooling coils	Loss of cooling coil due to leaks or rupture	None. The redundant full capacity unit train is put into operation	Eventual loss of chilled water alarm in the control room	No loss of safety function
Emergency (LOCA or high chlorine or radiation in outside air with or without LOOP)	Electric heating coils	Failure of heating coil	None. The electric heating coils are not required to operate during emergency operation.	Temperature indicators at the duct and in local control panels	No loss of safety function
Emergency (High chlorine or radiation in outisde air or Zone I, II or III isolation signals with or without LOOP)	Isolation dampers between Units 1 & 2 (HD-07824)	Damper failure	None. The redundant dampers are in series and are designed to fail in the safe closed position.	Damper position indication in the control room	No loss of safety function

TABLE 9.4-19

CONTROL ROOM & CONTROL STRUCTURE HVAC SYSTEMS
EMERGENCY OUTSIDE AIR SUPPLY SYSTEMS
FAILURE MODE AN EFFECT ANALYSIS

PLANT OPERATING MODE	SYSTEM COMPONENT	COMPONENT FAILURE MODE	EFFECT OF FAILURE ON THE SYSTEM	FAILURE MODE DETECTION	EFFECT OF FAILURE ON PLANT OPERATION
Emergency	Power Supply	Total loss of offsite power (LOOP)	None. The systems are redundant and are powered from separate standby diesel generators.	Alarm in the control room	No loss of safety function
Emergency (High outside air radiation or Zones I, II or III isolation signals)	Fans (OV-101)	Loss of one fan	None. The standby fan automatically starts.	Alarm in the control room	No loss of safety function
Emergency (High chlorine in outside air)	Fans (OV-101)	Loss of one fan	The systems operation during this condition is strictly manual. Therefore, the redundant train can be manually started on failure of the operating train.	Alarm in the control room and fan status indicating lights also in the control room	No loss of safety function
Emergency (High outside air radiation or Zones I, II or III isolation signals)	Fan outlet dampers (HD-07811)	Damper failure	None. The dampers are designed to fail close. When the damper fails closed it trips its associated train and isolates the entire filter train. The standby train will then automatically start.	Alarm in the control room	No loss of safety function
Emergency (High chlorine in outside air)	Fan outlet dampers (HD-07811)	Damper failure	The systems operation during this condition is strictly manual. The dampers are designed to fail closed. When the damper fails closed it trips its associated fan and isolate the entire filter train. The redundant train can be manually started on failure of the operating train.	Alarm in the control room	No loss of safety function
Emergency (High outside air radiation or Zones I, II or III isolation signals)	Emergency outside air dampers (HD-07812, HD-07814)	Damper failure	These are two sets of redundant dampers in parallel. Because they are in parallel failure of one does not affect the system. These dampers are designed to fail in the closed position because they are used for isolation during high chlorine condition.	Damper position indication in the control room	No loss of safety function

TABLE 9.4-19 (CONTINUED)

PLANT OPERATING MODE	SYSTEM COMPONENT	COMPONENT FAILURE MODE	EFFECT OF FAILURE ON THE SYSTEM	FAILURE MODE DETECTION	EFFECT OF FAILURE ON PLANT OPERATION
Emergency (High chlorine)	Emergency outside air dampers (HD-07812, HD-07814)	Damper failure	None. These dampers are designed to fail safe in the closed position. This system can only be manually operated during high chlorine. Therefore, before operation these dampers are checked to make sure they are fully closed.	Damper position indication in the control room	No loss of safety function
Emergency (High outside air radiation or Zones I, II or III isolation signals)	Recirculation inlet inlet dampers (HD-07813)	Damper failure	None. These dampers are designed to fail closed and are required to be closed during this mode of operation.	Damper position indication in the control room	No loss of safety function
Emergency (High chlorine)	Recirculation inlet dampers (HD-07813)	Damper failure	None. During this mode of operation the system is manually operated. If the damper fails closed the train fails and a fan failure due to loss of flow is alarmed in the control room. The redundant train can then be manually started.	Fan failure alarm and damper position indication in the control room	No loss of safety function
Emergency (High radiation in outside air or Zones I, II or III isolation signals)	Electric heating coil (OE-143)	Heater failure	None. High temperature (pre-ignition) is alarmed in the control room. At a higher temperature (ignition) the train is tripped and isolated. The standby train automatically starts. When the heater fails (no heat) the train is also tripped and the standby train automatically starts.	Alarms in the control room (pre-ignition and ignition temperatures)	No loss of safety function
Emergency (High chlorine)	Electric heating coil (OE-143)	Heater failure	None. Under this mode of operation the train is operated manually. High temperature (pre-ignition) is alarmed in the control room. A higher temperature (ignition) is also alarmed in the control room. Ignition temperature and loss of heaters (no heat) trip the train. The redundant train is then manually started.	Alarms in the control room	No loss of safety function

TABLE 9.4-19 (CONTINUED)

PLANT OPERATING MODE	SYSTEM COMPONENT	COMPONENT FAILURE MODE	EFFECT OF FAILURE ON THE SYSTEM	FAILURE MODE DETECTION	EFFECT OF FAILURE ON PLANT OPERATION
Emergency (High radiation or chlorine in outside air or Zones I, II or III isolation signals)	Prefilter, downstream and upstream HEPA filters	High differential pressure across any of these components	None. If any of these filters is completely clogged, air flow will be lost and the standby unit will automatically start.	Local differential pressure indicators. Pressure differential across the upstream HEPA filter is recorded and alarmed in the control in compliance with Regulatory Guide 1.52.	No loss of safety function
Emergency (High radiation in outside air or Zones I, II or III isolation signals)	Charcoal absorbers (OF-125)	High temperature (ignition temperature)	None. Pre-ignition temperature is alarmed in the control room. At a higher temperature (ignition) the fire protection deluge water valves are opened, the whole train is tripped & isolated and the standby train automatically starts.	Pre-ignition and ignition temperature alarms in the control room	No loss of safety function
Emergency (High chlorine)	Charcoal absorbers (OF-125)	High temperature (ignition temperature)	None. Under this mode of operation the train is operated manually. Pre-ignition temperature is alarmed in the control room. At a higher ignition temperature the train is tripped and isolated and alarmed in the control room. The redundant train can then be manually operated.	Pre-ignition and ignition alarms in the control room	No loss of safety function
Emergency (High chlorine or radiation in outside air or Zones I, II or III isolation signals)	Q-listed fire protection backup deluge water valve (TV-07813)	Valve failure	None. These values are designed to fail closed and are used to backup the non-seismically qualified deluge valves.	Alarm in the contorl room	No loss of safety function

CHAPTER 10

STEAM AND POWER CONVERSION SYSTEM

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CHAPTER 10

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10.4 OTHER FEATURES OF THE STEAM AND POWER CONVERSION SYSTEM

10.4.1 MAIN CONDENSER

10.4.1.1 Design Bases

The main condenser has no safety-related function and is designed to:

1) Condense and deaerate the exhaust steam from the main turbine and reactor feed pump turbines.

2) Accept and deaerate the drains from the feedwater heaters and other components in the heat cycle.

3) Serve as a heat sink for the turbine bypass steam, extraction steamline dump drains and heat cycle relief valve discharges.

4) Retain for a minimum of two minutes the condensate formed during full load operation to allow radioactive decay prior to returning the condensate to the cycle.

10.4.1.2 Description

The main condenser is a triple-pressure deaerating type comprising three (3) separate shells, one Low Pressure (LP), one Intermediate Pressure (IP) and one High Pressure (HP). Each of these shells is connected to the exhaust of one of the three low pressure turbines by a rubber expansion joint which is secured between two steel frames, one being welded to the turbine exhaust and the other to the condenser. The condensers are provided with deaerating hotwells which remove air plus hydrogen and oxygen formed in the turbine steam due to the disassociation of water in the reactor.

These non-condensible gases are concentrated in the air cooling sections of the condensers, where they are removed by the steam jet air ejector and discharged to the gaseous radwaste system. (See Subsection 10.4.2)

The steam exhausted to the condenser is condensed by water which is circulated through the condenser tubes by pumps which take their suction from the cooling tower basin (see Subsection 10.4.5).

The oxygen content of the condensate will not exceed 0.005 cc/l as measured at the discharge of the condensate pumps with an air in leakage of up to 75 cfm. Design parameters for the condensers are shown in Table 10.4-1. The condenser is designed and built to the standards of the Heat Exchange Institute and the manufacturer's standard practice.

10.4.1.3 Safety Evaluation

The main condensers are not required either to support the safe shutdown of the reactor or to perform in the operation of any reactor safety features.

The anticipated inventory of radioactive contaminants during both operation and shutdown is discussed in Sections 11.1 and 11.3. The shielding and controlled access arrangement for the main condensers is described in Section 12.3.

10.4.1.3.1 Radioactive Gases

Under normal operation, these gases are removed by the air ejector and delivered to the gaseous radwaste system. To prevent unacceptable accidental releases of radioactivity to the environment, the ventilation system maintains a slight vacuum in the condenser area. Any radioactive gases which leak out of the condenser are removed by the ventilation system and processed through charcoal filters before being vented out of the turbine building stack. Any hydrogen which accumulates in the condenser during operation is removed as described in Subsection 10.4.1.2. There will be no significant buildup of hydrogen in the main condenser during shutdown as it will be isolated from potential sources of hydrogen.

10.4.1.3.2 Condenser Leakage

If one or more condenser tubes develop leaks or if a tube to tubesheet joint fails, the circulating water will be forced into the condensate, thus raising its conductivity. To detect this condition, conductivity cells are installed in the condenser hotwell and in tubesheets which alarm if the conductivity rises above acceptable limits.

The water boxes and crossover piping are made of carbon steel and designed for a pressure of 90 psig with a 1/8" corrosion allowance. A system of cathodic protection can be added if it is deemed necessary at a later time. The condenser tubes, baffles, distribution headers, impingement plates, spray pipes, etc., are type 304 stainless steel.

All high-velocity or flashing steam-and-water mixtures such as drains, dumps and turbine bypass blowdown connections are provided with suitable type 304SS impingement baffles and/or perforated distribution pipes to prevent tube erosion and preclude cutting of structural members.

10.4.1.3.3 Circulating Water System Rupture

The presence of any water accumulation in the condenser area is detected by level switches which are mounted on the shielding wall at various points around its perimeter.

These switches will alarm in the control room. Since remedial action is dependent on the rate of increasing water level, analog level rate measurement is provided. This indication will provide the operator with sufficient information to determine if an orderly unit shutdown is possible.

The flooding caused by a major leak such as the complete rupture of a rubber expansion joint will be contained within the concrete shielding walls which surround the condenser area and which are designed to withstand the possible 20 feet of differential water pressure they could experience in the event of a major rupture.

The doors which provide access through the shielding walls are watertight, and, in addition, minimum clearances are provided around penetrations through the walls so as to limit the quantity of water leaking out of the condenser area in the event that it becomes flooded. During a major rupture of an expansion joint, the water could rise from the condenser area floor at el. 656 to grade at el. 676, at which level it will spill out through the doors. There is no safety related equipment in the Turbine Building below grade and there are no penetrations below grade between the turbine building and the Reactor Building. Regular inspections will be made of all the rubber expansion joints in an effort to detect any signs of deterioration which could lead to a major rupture, and any time the system is subjected to a major transient, an additional inspection will be made.

The level instruments will be tested on a regular basis to ensure that they are in good working order. If the hotwell of the condenser ruptured, the 100,000 gallons of condensate released would flood the condenser area to a depth of 18 inches. However, neither a major rupture of the circulating water system nor a rupture of the condenser hotwell will have any effect on any safety-related system since no safety-related systems are located within this area.

10.4.1.4 Tests and Inspections

The steam side of each condenser will be hydrostatically tested in the field by completely filling each shell with water and inspecting all accessible welds and surfaces for leakage and/or excessive deflection.

The circulating water side of each condenser will be hydrostatically tested in the field to a pressure of 95 psig, and all joints and surfaces will be inspected for leaks. The tube to tube sheet joints will also be inspected for leaks at this time. In addition, all the tubes will be subjected to a pneumatic test at 150 psig minimum and nondestructively tested by the eddy current method in accordance with ASTM Standards A249 and A450.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

10.4.1.5 Controls and Instrumentation

10.4.1.5.1 Condenser

To determine that the condenser does not exceed recommended pressures while it is in operation, pressure switches are installed which alarm at 24.2 inches Hg vacuum and trip the turbine at 21.7 inches Hg vacuum. In addition, separate switches are installed which close to prevent the bypass valves from opening if the vacuum in the condenser is low. They will remain closed until the vacuum exceeds the switch setting by 1 inch Hg. These switches are set at 7 inches Hg vacuum.

The accuracy of all the switches referred to above is ±1/2 percent. To ensure that the level of the condensate in the hotwell remains within acceptable limits, level transmitters are installed which regulate two sets of makeup and reject control valves to maintain the hotwell level. The 4-inch makeup and 3-inch reject valves regulate the flow of condensate between the condenser and the condensate storage tanks during normal steady state operation and small transients. During large transients in the system, the regulation of the condensate flow between the condenser and the condensate storage tanks is transferred to the 12-inch makeup and 8-inch reject control valves, respectively. Pressure switches are provided to trip the MSIV's closed in the event that condenser pressure rises to approximately 21 inches Hg.

Motor operated bypass valves are provided for both the make up and reject control valves. If either or both of these control valves fail the make up and/or reject flow will be regulated remote manually by hand switches in the control room. A high-low level switch is provided to alert the control room operator of abnormal hot well levels.

10.4.1.5.2 Condensate Contamination

If any condenser tubes develop leaks at the tubesheets, a leakage collection trough under each tube sheet and a conductivity cell associated with each trough will detect this leakage and alarm locally and in the control room.

Condenser tube rupture is detected by conductivity cells located in the lower tray of each condenser. These cells also alarm locally and in the control room.

10.4.2 MAIN CONDENSER EVACUATION SYSTEM

10.4.2.1 Design Bases

The Main Condenser Evacuation system has no safety-related function and is designed to:

a) Establish vacuum on the condenser during startup

b) Remove the noncondensible gases from the main condenser and discharge them to the gaseous radioactive waste system

c) Condense any steam removed from the condenser with the noncondensible gases and return the condensate to the condenser.

10.4.2.2 System Description

The major components of the main condenser evacuation system are a mechanical vacuum pump and steam jet air ejectors. These function as follows:

Mechanical Vacuum Pump - The mechanical vacuum pump is used during startup to establish vacuum in the condenser once the turbine glands have been sealed with clean steam and to discharge the air drawn from the condenser to atmosphere through the plant ventilation stack.

The mechanical vacuum pump and its suction valve are operated remotely from the control room. The suction valve is automatically closed upon a main steam high radiation signal. The pump is shutdown on high radiation or low seal water flow to the pump. A water separator removes the water droplets from the noncondensible gases before discharging them to atmosphere.

The capacity of the vacuum pump is 4000 scfm and it is designed to evacuate the main condenser to 5 inches Hg Absolute in 95 minutes.

Since the pump components are manufacturer's standard items, they are not ordered per a specific ASTM, ANSI or ASME code or standard.

Steam Jet Air Ejectors (SJAE) - Once vacuum has been established by the mechanical vacuum pump, the steam jet air ejector is placed in service to maintain vacuum and the mechanical vacuum pump is shut down. The air ejector is a full capacity 2-stage unit with four 25 percent capacity first stage ejectors and one full capacity second-stage ejector with a rated capacity at 130°F and 14.7 psia of 75 scfm air, 150 scfm hydrogen and 80 scfm oxygen.

The four first-stage steam jet ejectors continously remove noncondensible gases and some steam from the condenser and discharge them to the intercondenser where the carryover steam is condensed and returned to the condenser. The gases are then removed from the intercondenser by the second-stage ejector and discharged to the gaseous radioactive waste system together with the second-stage ejector motive steam. This steam and gas mixture eliminates the possibility of an explosion in the line even though a mixture rich in hydrogen is present. The ejectors require motive steam at 110 psig to operate. This is obtained either from main steam, which is nominally at 950 psig and must therefore be reduced in pressure through pressure reducing valves, or from steam produced by the auxiliary boilers through a separate control valve. There are two redundant pressure-reducing valves, either of which can be used to provide motive steam. Steam for additional off-gas dilution is provided in a bypass piping loop around the second stage air ejector. Control is provided by two full capacity parallel flow control loops. The controls are designed to provide an effluent hydrogen concentration of approximately 4% by volume at 100% rated reactor power which maintains the downstream equipment within the equipment design temperature limits. Upon a reduction of approximately 10% of the required steam flow, the first stage air ejector gas inlet valves close.

Condensate taken from the condenser hotwell by the condensate pumps and discharged to the condensate system is used as the cooling medium for the air ejector intercondenser.

Both stages of air ejectors have nominal pressure ratings of 15 psig. The intercondenser has a pressure rating of 50 psig. An intercondenser relief valve is provided and is set to relieve full flow at 16.5 psig. Based on B31.1 allowable stresses and maximum allowable pressure equations, the first stage ejectors are designed for 280 psig and the second stage ejector for 250 psig. The first stage ejectors and intercondenser operate at less than 10 psia. The second stage ejectors operate between 10 psia and 2 psig. 2

The intercondenser is manufactured to Section VIII of the ASME Boiler and Pressure Vessel Code. The complete system is shown on Figure 10.4-9.

10.4.2.3 Safety Evaluation

The noncondensible gases in the main condenser are removed by the air ejectors and delivered to the gaseous radioactive waste system. These gases normally include the following: nitrogen-16, oxygen-19 and nitrogen-13, plus the radioactive noble gas parents of strontium-89, strontium-90 and cesium-137. The

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largest contribution to the main condenser's off-gas activity will come from the nitrogen-16 source.

For an evaluation of radioactive contaminants in the effluent from the steam jet air ejectors and the associated doses, see Section 11.3.

10.4.2.4 Tests and Inspections

All tests and inspections for the air ejectors and the intercondenser are in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code.

The vacuum pumps were tested as completely assembled units.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

10.4.2.5 Controls and Instrumentation

Both the steam jet air ejectors and the mechanical vacuum pump are controlled remotely from the control room. To put the air ejectors into service, the motive steam inlet and discharge valves for the secondary ejector must be opened first. This action allows the primary ejectors steam isolating valve to be opened which in turn provides the permissives to allow the main steam isolating valves, the pressure-reducing valves and the air removal isolating valves to be opened. This sequence of operation prevents the intercondenser from being overpressurized.

The noncondensible gases in the outlet line from the intercondenser are monitored by a pressure transmitter and a temperature transmitter both of which read out in the control room. In addition, a pressure switch is provided which on high pressure in the line alarms on a local panel and indicates as a trouble alarm in the control room. The discharge line from the secondary ejector is provided with a pressure tansmitter which reads out in the control room and a pressure switch which on high pressure indicates as a trouble alarm in the control room.

The pressure in the motive steamline to both the primary ejector and secondary ejector is sensed by a pressure switch which alarms on high pressure at a local panel and indicated in the control room as a trouble alarm. A pressure indicator in the control room also monitors the steam supply.

The vacuum pump, the seal water pump and the air suction valve are all operated from a common handswitch in the control room.

When this switch is energized the seal water pump starts and the air suction valve opens. After 150 seconds the vacuum pump starts providing the flow switch in the seal water supply to the vacuum pump indicates there is sufficient flow.

A flow switch is located in the line supplying seal water to the vacuum pump from the seal water cooler. This switch trips the pump in the event of low seal water flow to the pump.

Both the steam jet air ejectors and the mechanical pump will be shut down automatically on receipt of a main steam high radiation signal.

10.4.3 STEAM SEAL SYSTEM

10.4.3.1 Design Bases

The steam seal system has no safety-related function and is designed to provide a continuous supply of clean (nonradioactive) steam to main turbine shaft seals, the stem packings of the stop valves, control valves, combined intermediate valves and bypass valves, the shaft seals of the reactor feed pump turbines and the stem packing of the reactor feed pump turbines, stop and control valves.

The purpose of this sealing steam is to prevent air leaking into the cycle and radioactive steam leaking out of the cycle into the Turbine Building.

The steam seal evaporator, in which the clean steam is generated, is designed, fabricated, tested and stamped in accordance with Section VIII of the ASME Boiler and Pressure Vessel Code. The steam seal system piping material is carbon steel and is equivalent to ASTM A106 Grade B, ANSI B31.1.

10.4.3.2 Description

The clean (nonradioactive) steam for the seals is generated in the steam seal evaporator by heating condensate with radioactive steam. The evaporator is a horizontal shell and U-tube heat exchanger in which the radioactive steam is passed through the U-tube bundle which is totally immersed in condensate. The condensate for the evaporator is taken from the condensate system from a point downstream of the condensate demineralizer, and the radioactive heating steam is taken either from a main steamline or from the extraction steamline to heaters 1-3A, 1-3B and 1-3C. A backup source of sealing steam is available from the auxiliary boilers. For more detail on the operation of these various steam sources, see subsection 10.4.3.5.

The sealing function is provided by supplying nonradioactive steam at 19 psia to the turbine shaft seals and various valve stems. Of the sealing steam entering the turbine shaft seals, some leaks inward towards the turbine, but some leaks towards the outside, where it enters a vent annulus which is maintained at a slight vacuum (about 5 inches of water) by the steam packing exhauster. A small amount of air is drawn into this vent annulus from the outside, and this air, together with the nonradioactive sealing steam, is drawn to the steam packing exhauster where the steam is condensed and returned to the condenser. The cooling medium for condensing the steam is condensate in the main feedwater system. The saturated air is discharged by the exhauster to the Turbine Building vent. With this arrangement, all radioactive steam is completely contained within the turbine loop.

10.4.3.3 Safety Evaluation

The clean steam seal system is designed to provide the required quantity of sealing steam under all modes of operation. The individual components of the system have been designed to ensure as far as possible that the saturated air going to the steam packing exhauster is nonradioactive. In the event of a failure in the steam seal system, an alternate source of clean sealing steam is the auxiliary boilers. Therefore, there is no radioactive leakage path to the environment. Steam from these boilers will be fed directly into the steam seal supply header after its pressure is reduced to 19 psia by a pressure reducing valve.

Protection against overpressure is provided for both the heating steam (tube) side and the condensate (shell) side of the evaporator by the installation of relief valves. Similarly, the steam seal supply header is protected against overpressure by relief valves. All these relief valves discharge to the main condensers.

The steam packing exhauster condenser is provided with two full-capacity exhausters, which are mounted on the condenser so that there will be a backup exhauster for the one which is operating.

10.4.3.4 Tests and Inspections

The steam seal evaporator is fabricated and tested to Section VIII, Division I of the ASME Boiler and Pressure Vessel Code.

The piping furnished by the Owner is inspected and tested in accordance with ANSI B31.1 and the piping furnished by the turbine vendor, General Electric, is inspected and tested in accordance with their own standards.

Both the shell side and the tube side of the steam packing exhauster condenser are hydrostatically tested to 1.5 times their design pressure.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

10.4.3.5 Controls and Instrumentation

Evaporator Steam Heating System - During low load operation (startup and shutdown), the heating steam for the steam seal evaporator is taken from the main steam lines ahead of the turbine stop valves. Its pressure is reduced by throttling the two evaporator steam feed valves which are 4 inches and 8 inches, respectively. With main steam pressures of 25 percent or more of rated pressure, 965 psia, the 4-inch regulating valve will automatically supply sufficient heating steam to the evaporator to generate the required flow of clean sealing steam to seal the shaft packings when these packings have normal clearances. With very low main steam pressures or worn packings, the 8 inch regulating valve automatically operates in parallel with the 4 inch valve to provide the additional heating steam required to produce the required clean steam seal flow in the evaporator. As the load on the turbine increases, extraction steam pressure also increases and when it is approximately 5 psi higher than the main steam pressure downstream of the regulating valves a differential pressure switch actuates and allows the BTU's in the extraction steamline to open and supply heating steam to the steam seal evaporator. Both regulating valves fail open on loss of supply air and if a failure occurs, the evaporator can be controlled from the control room manually by throttling a motor operated valve for a short time until the air supply can be reestablished or the auxiliary steam supply can be put into service.

Evaporator Feed Water System - The level of condensate in the evaporator is maintained within the required limits by a control valve, the operation of which is regulated by signals from a remote proportional controller. In addition to this control valve, the evaporator feedwater isolating valve, which is normally open, and the evaporator feedwater bypass valve, which is normally closed, can be used to control the flow of condensate to the evaporator if the automatic control system malfunctions. The automatic control system is monitored by the evaporator low level and high level alarm. In addition, a high level switch closes the evaporator feedwater isolation valve. The operation of all isolation and bypass valves is performed by hand switches located in the control room. The appropriate alarms and level indicators read out in the control room.

Steam Seal Header - The clean sealing steam generated in the evaporator enters the steam seal supply header through the pressure control valve which is set to maintain the pressure of the sealing steam at about 4 psig (19 psia). Redundancy for this pressure control valve is provided by a motor operated bypass valve which would be regulated manually from the control room to maintain the pressure of the sealing steam if the pressure control valve fails.

A pressure transmitter in the steam seal header which reads out in the control room will indicate to the operator when to adjust the bypass valve. The steam seal header is also provided with a high-low pressure switch which alarms in the control room.

If there should be a malfunction in the primary steam seal supply system which causes the pressure in the steam seal header to drop, a pressure switch will open the isolating valve and admit auxiliary steam to the steam seal header at 4 psig. This pressure is regulated automatically by another pressure control valve located in the auxiliary steamline.

Evaporator Drain System - The heating steam to the evaporator condenses in the tube bundle and collects in the evaporator drain tank. Under normal operation, the low level controller positions the level control valve to regulate the drain flow to feedwater heaters 2A, 2B and 2C and maintain the drain tank level. If for any reason the flow cannot be handled by this primary drain system and the level in the drain tank continues to rise, the high level controller will open a second level control valve which will allow the excess drains to flow to the high pressure condenser, thereby restoring the level in the drain tank to normal.

In addition to the level controllers, the drain tank is provided with low and high level switches which alarm in the control room.

10.4.4 TURBINE BYPASS SYSTEM

10.4.4.1 Design Bases

The turbine bypass system has no safety-related functions.

The turbine bypass system is designed to bypass main steam directly to the condenser to control the pressure in the reactor during the following modes of operation;

1) Reactor vessel heat up to rated pressure and subsequent cooldown.

2) Turbine run up and run down.

3) Power operation when the quantity of steam generated by the reactor exceeds that required by the turbine.

The piping which connects the main steamlines to the inlet of the bypass valve chest is designed in accordance with ASME Section III Class 2 requirements and the piping connecting the discharge of the bypass valves to the condensers is designed to ANSI B31.1.

The bypass valves are required to have regulation capability and a fast-opening response approximately equivalent to the fast closure of the turbine stop and control valves.

The bypass values are designed to GE's nuclear standards.

10.4.4.2 System Description

The turbine bypass system Figure 10.4-1 consists of:

a) Bypass valves

b) Piping between the main steamlines and the bypass valve inlets

c) Piping between the discharges of the bypass valves and the condensers

d) Pressure reducer assemblies.

The bypass valve chest consists of five separate bypass control valves mounted in individual compartments of a common valve chest. The valves are globe type with the stems arranged so that they reach the outside of the chest through the discharge chamber of the respective valve. This minimizes leakage when the valves are closed since it is necessary only to seal the stem against condenser vacuum. The valves open sequentially. When used during normal start up and shutdown only the No. 1 and No. 2 bypass valves are used. However, in the event of a full load rejection, such as would occur if the generator circuit breaker opened, it is necessary that all 5 valves open to bypass 25 percent of the turbine valves wide open flow, which is the maximum design flow of the bypass valves.

Bypass steam flows from the main steamlines through a 24 inch header, which is upstream of the bypass valves, divides into two 18 inch headers each of which is connected to the valve chest at opposite ends. The discharge connections of the bypass valves are piped individually to the condensers in 10" lines and, in order to reduce the pressure at which the bypassed steam enters the respective condenser, a pressure reducer assembly is installed in each bypass valve discharge line. The pressure reducing assembly consists of a fabricated piece of pipe 6 feet long which increases in size over its length from 10 inch nominal pipe size at the inlet end to 18 inch nominal pipe size at the outlet. This assembly contains four pressure breakdown plates along its length.

The condensers are designed to accommodate the maximum turbine bypass flow (3,711,000 lb/hr total), which is reduced to 250 psig at 1191.5 BTU/Hr through the pressure breakdown assemblies, without increasing the turbine backpressure to the turbine trip setting of 7" Hga. The backpressure will exceed the design backpressure of 5" Hga for an estimated 30 seconds maximum with a peak of 6.1" Hga. The bypass flow enters the condenser below the tube bundle to assure steam desuperheating and prevent steam at 150°F from reaching the turbine.

10.4.4.3 Safety Evaluation

With 7 inches HgA and vacuum decreasing in the condenser, the bypass valve vacuum trip pressure switches will close to prevent the bypass valves from opening. These pressure switches will remain closed until the vacuum exceeds the bypass valve trip setting (7" HgA) by 1 inch (6" HgA). At this time the circuit automatically resets and permits the bypass valves to open.

There are no safety related components in the vicinity of the bypass piping. A high energy line failure in the turbine bypass system could cause a trip of the main turbine either due to high condenser pressure because of increased air in leakage at the break or a possible break of EHC system piping due to steam impingement. Loss of EHC cannot cause overspeed but may cause trip.

Failure of the bypass valves to open for any reason, such as a mechanical malfunction or insufficient vacuum in the condenser, will cause the pressure in the reactor to increase, ultimately scramming the reactor and lifting the safety relief valves which discharge the excess steam to the suppression pool. A bypass system failure will have no adverse effect on a safety related components or systems.

The effect of such a situation on the reactor coolant system is described in Chapter 15.

10.4.4.4 Tests and Inspections

The piping upstream of the bypass valve chest, which connects the main steamlines to the bypass valve inlets, is inspected and tested according to ASME Section III, Class 2. The piping downstream of the bypass valve chest, which connects the discharge of each bypass valve individually to the respective condenser nozzle, is inspected and tested according to ANSI B31.1.

Each pressure reducer assembly is hydrostatically tested to 1400 psig by General Electric.

During normal plant operation each bypass valve can be tested from controls on the EHC panel in the control room to ensure it is functioning correctly.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

10.4.4.5 Controls and Instrumentation

Controls and valves are designed so that the bypass valves steam flow is shut off if the control system loses its electric power or hydraulic system 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% second of reactor rated flow.

Upon turbine trip or generator load rejection, the start of the bypass valve steam flow will not be delayed more than 0.1 second after the start of the stop valve or the control valve fast closure motion. A minimum of 80% of the rated bypass capacity will be established within 0.3 second after the start of the stop valve or the control valve closure motion. For more detail, refer to Subsection 7.7.1.5.

10.4.5 CIRCULATING WATER SYSTEM

10.4.5.1 Design Bases

The circulating water system has no safety-related functions and is designed to remove the latent heat from the main condenser and sensible heat from the service water system and dissipate both in a hyperbolic natural draft cooling tower.

Heat load from main condenser	$= 7.89 \times 10^9$ BTU/Hr
Heat load from service water system	$= \underline{0.18 \times 10^9}$ BTU/Hr
Total heat load for which Cooling Tower is designed	$= 8.07 \times 10^9$ BTU/Hr

10.4.5.2 System Description

The circulating water system consists generally of the following major components.

a) 1 - Cooling Tower rated to remove a heat load of 8.07×10^9 Btu/Hr

b) 4 - 25% capacity circulating water pumps each rated 112,000 gpm at 100 ft head.

c) 1 - Condenser Tube Cleaning System comprising

2 - Ball strainer sections

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4 - Ball circulating pumps each rated 260 gpm at 65ft head

4 - Ball collecting baskets

16 - Ball injection nozzels

2 - Control Panels

d) Piping, valves, controls and instrumentation.

Circulating water from the cold water outlet of the cooling tower basin is delivered to the suctions of the four circulating water pumps through two 108 inch diameter pipes. Each of these pipes supply one pair of pumps. Each pair of pumps discharge to a 96 inch header and the two 96 inch headers are run underground into the condenser area. Here each of these headers divides into two 78 inch lines one pair being connected to the A and B quadrants and the other to the C and D quadrants circulating water inlet connections of the Low Pressure (LP) Condenser. The circulating water discharging from the LP condenser flows into the Intermediate Pressure (IP) Condenser and subsequently to the High Pressure (HP) Condenser in series. From the discharge of the HP Condenser each pair of 78 inch lines is combined into one of two 96 inch headers which return the circulating water to the cooling tower.

The circulating water pumps are the horizontal centrifical type and driven by electric motors. Motorized butterfly valves are provided in the suction and discharge of each pump so that the pump can be isolated if necessary. Motorized butterfly valves are also provided in each of the 78 inch circulating water lines, one at the inlet to the LP Condenser and another at the outlet from the HP Condenser. These permit any of the four quadrants on the circulating water side (tubeside) of the condensers to be isolated in the event of tube leakage. Rubber expansion joints are provided at the suction and discharge of each circulating water pump and at the inlet and outlet connections of each of the three condensers.

The condenser tube cleaning system is built into the circulating water system. It operates by injecting foam rubber balls, whose diameter is slightly larger than that of the tubes, into the four CW lines entering the LP Condenser. The flow of water drives these balls through the tubes of each condenser in turn and since they are larger in diameter than the tubes they scour the inside of the tubes & keep them free of algae, etc. The balls emerge in the CW lines at the outlet from the HP Condenser and they are collected in full flow strainers which are installed in the 96 inch CW discharge lines.

Recirculation pumps draw off the balls from the strainers together with a small quantity of water and reinject them back into the LP Condenser inlet. The strainers automatically reverse for backwashing on high pressure drop across the strainer. However, before backwashing is started the balls are removed from service automatically and gathered in specially designed ball collectors.

The circulating water is chlorinated to prevent the formation of biological growth. Sulfuric acid is added to the system on an intermittent basis to control pH.

Water lost from the system due to evaporation and drift from the cooling tower is replenished by water pumped from the Susquehanna River by the make up pumps.

The service water system is described in Subsection 9.2.1.

10.4.5.3 Safety Evaluation

The potential for flooding due to an expansion joint rupture is discussed in Subsection 10.4.1.

The opening and closing times of the circulating pump discharge valves have been arranged to minimize system transients when a pump is started. On loss of one of the four circulating water pumps the turbine load will be automatically set back to approximately 60% of that for the VWO condition.

Failure of the cooling towers will not affect any safety-related systems.

10.4.5.4 Tests and Inspections

The condenser is tested as described in Subsection 10.4.1.3. The pumps, butterfly valves, expansion joints and tube cleaning system are all tested by their respective manufactures prior to shipment.

Hydrostatic and leakage tests are performed on the circulating water pipe in accordance with AWWA standards.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

10.4.5.5 Controls and Instrumentation

Level elements are located upstream and downstream of the screens in the cooling tower basin outlet. These initiate high and low level alarms. A differential level switch alarms on high level differential across the screens indicating that they require clearing. Low low level switches sense level in the basin outlet downstream of the screens and upstream of the inlet to the 108 in. diameter CW pump suction piping. On low low level these switches alarm and at the same time close the suction and discharge valves of each pumps and trip the pump sequentially.

The pressure at both the suction and discharge of each pair of CW pumps is monitored by pressure transmitters which read out in the control room.

The bearings of each pump and motor are provided with thermocouples that alarm on high temperature. In addition the temperatures of the windings of each motor are monitored by RTD's that alarm on high temperature.

Pressure transmitters are installed in each of the four CW inlet connections on the LP Condenser and in the four outlet connections of the HP Condenser. These are interconnected to monitor the differential pressure across the three condensers on the CW side. Resistance Temperature Detectors (RTD's) are also installed in these locations to monitor the temperature of the circulating water.

A level controller senses the level in the cooling tower basin, and operates a level control valve, which is in the makeup pump discharge line, to maintain the basin level.

10.4.6 CONDENSATE CLEANUP SYSTEM

10.4.6.1 Design Bases

The condensate demineralizer system has no safety-related functions and is designed to maintain the condensate at the required purity by removal of the following contaminants:

a) Products resulting from corrosion that occur in the main steam and turbine extraction piping, feedwater heater shells, and drains

b) Suspended and dissolved solids that may be introduced by small leakages of circulating water through condenser tubes

c) Fission and activation products that are entrained in reactor steam and retained in condensate leaving the hotwell

d) Solids carried into the condenser by makeup water and miscellaneous drains.

The system design is based on the influent concentrations given in Table 10.4-2.

At 4800 gpm per vessel, and with the influent quality listed in Table 10.4-2, the ion exchangers effluent will not exceed the following quality:

a)	Conductivity at 25°C	0.1 micromho/cm
b)	pH at 25°C	6.5 to 7.5
c)	Silica (SiO_2)	5 ppb
d)	Iron, total (Fe)	5 ppb
e)	Copper (Cu)	2 ppb
f)	Nickel (Ni)	2 ppb
g)	Chloride (Cl)	1 ppb
h)	Total metallic impurities*	9 ppb

*Total metallic residue retained on a 0.45 micron film filter.

Piping is furnished in accordance with ANSI B31.1.0. Pressure vessels that fall within the jurisdiction of ASME Section VIII are furnished in accordance with that Code.

The design pressure of the condensate demineralizer system is 740 psig at 150°F, which is above the shut-off head of the condensate pumps.

10.4.6.2 System Description

The condensate demineralizer system (Figure 10.4-2) is designed to purify condensate continuously at 131°F and 550 psig at a flow rate of 28,800 gpm. Each demineralizer vessel has a flow capacity of 4,800 gpm and is capable of operating at flow rates up 5,760 gpm for short periods.

10.4.6.2.1 Condensate Demineralizer System

The condensate demineralizer system consists of a battery of seven ion exchangers, each containing a bed of mixed resin in the proportion of two parts cation resin to one part anion resin by volume. Six exchangers are in service at one time during normal conditions. The seventh exchanger is held on standby for replacement of an inservice unit at the end of its service run and in the event of an abnormal condenser leak. The condensate demineralizers are piped directly into the feedwater cycle and receive condensate under pressure from the condensate pumps.

Regeneration of a specific demineralizer unit occurs when one of three endpoints is reached:

1. Total flow through a unit reaches a preset limit (130,000,000 gallons),

2. Pressure drop across a unit reaches a preset limit (50 PSID from inlet header to outlet header),

3. Conductivity measurements at the outlet of each unit reach a preset level (0.1 μ mhos/cm).

Based on a total throughput of 130,000,000 gallons, regeneration frequency is approximately every 19 days for each resin bed based on influent quality listed in Table 10.4-2 at full load.

These endpoints have been chosen in order that each resin bed be taken out of service prior to reaching an unacceptable level of operation. In particular, the conductivity measurements provide indication that a specific bed may be ionically exhausted in order that it may be regenerated before an unacceptable level of overall condensate water quality is reached.

The control room alarm setpoint for the outlet of each demineralizer vessel, indicating resin bed exhaustion, is 0.1μ mho/cm as is the control room alarm setpoint for the condensate effluent header. Condensate influent to the demineralizers is alarmed in the control room at 0.2μ mhos/cm conductivity.

The resin beds are transferred from the ion exchangers to the external regeneration system for cleaning and chemical regeneration. A spare charge of resins is held in the external regeneration system for immediate replacement of an exhausted bed in an ion exchanger so that the exchanger may be made available promptly for replacement of another exhausted exchanger.

10.4.6.2.2 External Regeneration System

The system provided for cleaning and chemical regeneration of the resins used in the condensate demineralizer is shown in Figure 10.4-3. It consists essentially of three vessels: a cation, an anion, and a resin storage tank. The cation tank also serves as a resin receiving, resin cleaning, and resin separation tank, through which exhausted resins are transferred from the ion exchanger to the regeneration system. Interlocks are provided so that an off-line demineralizer cannot detect condensate pressure unless it is isolated from the external regeneration system. In addition, an isolation valve in the resin transfer line will automatically close if high pressure occurs in that line. The regeneration system is designed for 75 psig and 150°F.

The removal of crud accumulation on the resins is accomplished by a cycle of draining, air backwashing, and rinsing in the cation tank. The regeneration system is designed for use with an ultrasonic resin cleaner and space and connections are provided so that one may be added later. The cleaned resins are transferred back to the original ion exchange vessel for further ion exchange.

Resins in need of complete regeneration are transferred to the cation regeneration tank and cleaned as described in the preceding paragraph. The anion and cation resins are then separated by backwashing before the anions are transferred to the anion regeneration tank. At the end of regeneration the resins are mixed and stored in the resin storage tank.

10.4.6.2.3 Acid and Caustic Dilution Systems

Solutions of acid and caustic required for regeneration of cation and anion resins are prepared by in-line dilution of 66 degree Baume' sulfuric acid and 50 percent sodium hydroxide pumped from bulk storage tanks below the regeneration equipment.

Approximately 5-1/2 percent concentration of acid solution is required to regenerate the cation resins. The strong acid is mixed in a mixing tee with clean condensate as needed. Water is supplied at a constant rate by condensate transfer pumps through a pressure control valve.

Approximately 5-1/2 percent concentration of caustic at 120°F is required to regenerate the anion resins. Strong caustic is mixed with dilution water at 120°F in a mixing tee as needed. Dilution water is produced by blending 180°F water from the caustic dilution hot water tank with cool water.

10.4.6.2.4 Waste System

Three types of wastes are segregated from the regeneration waste discharge. These are: high conductivity, low conductivity and low solids content, and low conductivity and high solids content.

High conductivity wastewater (conductivity above 100 micromhos) is channeled to the chemical waste neutralizer tanks where it is neutralized and pumped to the radwaste evaporators for distillation. Low conductivity condensate from this process is returned to the condensate storage tank.

Low conductivity, low solids wastewater is channeled to the turbine buildings outer area sump where it is pumped to the liquid radwaste collection tanks.

Low conductivity, high solids wastewater (greater than 3 JTU) is channeled to the regeneration waste surge tanks. The tanks are designed with cone bottoms. From there the wastewater is pumped at 35 gpm to the waste sludge phase separator.

See Section 11.2 for the effect of the Condensate Cleanup System on the radwaste system.

10.4.6.3 Safety Evaluation

The equipment and controls in the condensate demineralizer system are of the same design and operational integrity as those in the radwaste system.

Spare capacity is provided in the system to negate the possibility of difficulties in handling radioactive waste when the system is operating. If the radwaste handling system approaches design capacity, such as when condenser tube leaks require maximum rates of regeneration of ion exchangers, the unit load is reduced to eliminate the possibility of exceeding operational limits.

The effluent water quality stated in Subsection 10.4.6.1 will not be exceeded with an 11.5 gpm condenser leak when circulating water contains 1000 ppm of total dissolved solids. The system will sustain an effluent conductivity of 0.15 micromho with a 46 gpm condenser leak when circulating water contains 1000 ppm of total dissolved solids. The circulating water quality used in the design of the CCS is given in Table 10.4-3. Conductivity is recorded at 12 locations in the condenser and at analysis stations located on the common influent and effluent header to the condensate demineralizer system, on the discharge of each ion exchange vessel, and at the discharge header of the reactor feed pumps. High conductivity alarms are provided to alert the plant operators to an abnormal condition.

Treated condensate conductivity levels are maintained within the limits of Table 2 of Regulatory Guide 1.56, Rev. 1 in the following manner:

Individual demineralizer vessel outlet conductivity is monitored and continuously recorded locally at the Turbine Building Sample Station Control Panel. High conductivity, indicating ionic exhaustion, is alarmed locally and at the main control room panel. The high conductivity alarm setpoint is 0.1 µmhos/cm, thus an ionically exhausted resin bed is removed from service and regenerated before reaching the Table 2 of Regulatory Guide 1.56, Rev. 1 lower limit of 0.2µmhos/cm. In addition, a regenerated resin bed being brought on line is automatically recycled to the condenser prior to being placed in service to ensure that the vessel is not brought on stream at high conductivity levels.

The combined demineralizer outlet conductivity is also monitored and continuously recorded locally at the Turbine Building sample station control panel. High conductivity of the combined effluent is alarmed locally and at the main control room panel at 0.1 µ mhos/cm. Since each vessel is alarmed when conductivity reaches 0.1 µmhos/cm. The likelihood of the combined effluent reaching the alarm point is remote except under conditions of a large condenser leak. However, demineralizer inlet conductivity is monitored in the same manner as the outlet flows, with an alarm setpoint of 0.2 µmhos/cm indicating condenser leakage. (Table 2 of Regulatory Guide 1.56, Rev. 1 lower limit is 0.5 mhos/cm).

The condensate demineralizer system is designed to operate in a manner such that corrective action is initiated prior to reaching the lower limits of Table 2 of Regulatory Guide 1.56, Rev. 1.

The values shown in Table 10.4-4 are arrived at assuming 100 percent removal efficiency for all dissolved principal fission and corrosion activation products. However, because the removal efficiency for suspended solids is 50 to 75 percent, the overall removal of corrosion activation products is somewhat less than 100 percent for the system. Tables 10.4-4 and 10.4-5 provide the design bases for radiation shielding in the condensate demineralizer area.

The effluent strainer in the discharge from each ion exchanger protects the feedwater system against a massive discharge of resins in the event of an underdrain failure.

10.4.6.4 Tests and Inspections

Piping is inspected and tested in accordance with ANSI B31.1.0. All pressure vessels are hydrostatically tested to 1.5 times their design pressure.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

10.4.6.5 Controls and Instrumentation

The condensate demineralizer and regeneration systems are controlled from a local control panel for all modes of operation, including transfer of resins for cleaning and returning these resins to the exchange vessel, or transfer of resins for cleaning and regeneration and transferring the previously regenerated and stored resins to the exchanger for standby.

The conductivities are monitored by a multipoint recorder for the following:

a) Influent and effluent of the condensate polishing demineralizer system

b) Effluent from each condensate polishing demineralizer.

In addition, conductivity alarms are provided to alert the operator for off-normal conditions. Resin condition is monitored in accordance with Regulatory Guide 1.56. A differential pressure transmitter is provided to monitor the differential pressure across the condensate demineralizer system. Flow transmitters, recorders, and flow totalizers are provided in the effluent of each condensate polishing demineralizer.

10.4.7 CONDENSATE AND FEEDWATER

10.4.7.1 Design Bases

The condensate and feedwater systems have no safety-related functions and are designed to return condensate from the condenser hotwell to the reactor at the required flows, pressure, and temperature. The systems are designed to automatically maintain the water levels in the reactor and the condenser hotwell during steady state and transient conditions.

Piping from the condenser hotwell up to but not including the outermost containment isolation valve is furnished in accordance with ANSI B31.1. Feedwater piping from the outermost primary containment isolation valve up to but not including the valve just outside the containment is designed in accordance with ASME, Section III, Class 2. Feedwater piping from the containment isolation valve to the reactor is designed in accordance with ASME, Section III, Class 1. The feedwater heaters and drain coolers are furnished in accordance with ASME Section VIII.

The condensate and feedwater systems from the condenser hotwell up to but not including the outermost primary containment isolation valve are not safety related. The feedwater system from the outermost primary containment isolation valve to the reactor is safety related. For the isolation criteria between this system and the reactor coolant boundary see Subsection 6.2.4. In-service inspection is performed in accordance with ASME Section XI for that portion of the feedwater system furnished in accordance with ASME Section III. The condensate and feedwater systems are stress analyzed for the forces and moments that result from thermal growth. The ASME Section III and feedwater systems are stress analyzed for the forces and moments that result from thermal growth. The ASME Section III feedwater pipe located inside the reactor building main steam pipe tunnel and inside the drywell is Seismic Category I.

The condensate and feedwater system is designed to permit continued operation of the plant at reduced power without reactor trip upon loss of one or two of the four condensate pumps, one of the three reactor feed pumps, or one of the three strings of feedwater heaters.

10.4.7.2 System Description

The condensate and feedwater systems are shown in Figures 10.4-4 and 10.4-5, respectively.

Four, 25 percent capacity, vertical, 10-stage, canned suction, constant speed, motor driven, centrifugal condensate pumps take a common suction from the condenser hotwell and discharge into a common header. The condensate flow is then directed through the steam jet air ejector (SJAE) condenser and the steam packing exhauster (SPE) condenser and then into the condensate demineralizer's common influent header. Condensate then proceeds through the condensate demineralizer system and discharges into a common effluent header.

The sealing water for the condensate pump glands is taken from the pump discharge. It passes through a magnetic separator and cyclone separator before entering the glands. The leak off from the glands is piped to the liquid radwaste system.

A condensate recirculation system is furnished to maintain a minimum flow through each condensate pump, through the SJAE and SPE condensers and through the condensate demineralizers during low load operation. A recirculation flow of 2000 gpm per condensate pump is automatically maintained by a control valve downstream of the condensate demineralizers. Block and bypass valves are provided around the control valve for maintenance accessibility during plant operation. Recirculated condensate is returned to the condenser.

A hotwell makeup and reject system maintains the condenser hotwell level during steady state and transient conditions. Condensate, downstream of the condensate demineralizer effluent header, is rejected to the condensate storage tank to decrease the hotwell level or condensate from the storage tank is drained into the condenser to make up the hotwell level. A small amount of reject during steady state conditions is normal. A combination of small and large control valves, with block and bypass valves, are used to control the makeup and reject flows.

The small control valve controls the smaller normal flows whereas the larger control valve is used during larger transient flow. The control valves are controlled by water level in the condenser hotwell. (Refer to Subsection 10.4.1.2.)

Condensate from the condensate demineralizer effluent header divides and passes through three parallel strings of feedwater heaters. The feedwater passes through the tube side of an external drain cooler for heater one, then heaters one through five, respectively. Heaters one and two are located in the neck of the main condenser shells. The vents, drains and extraction steam side of the feedwater heaters are discussed in Subsection 10.4.10. Feedwater downstream of the heaters is combined into a common header from which three, 1/3 capacity, turbine driven, variable speed, barrel type, double suction, centrifugal, reactor feed pumps take suction. The reactor feed pumps discharge the feedwater into a common header for distribution through the reactor containment isolation valves and into the reactor.

Injection water for the reactor feed pump seals is taken from the effluent header of the condensate demineralizers. The inboard seals drain to number 5 feedwater heater while the outboard seals drain to the condenser..

Each reactor feed pump has a recirculation line connected to the main condenser. This recirculation line is tapped off the discharge line between the reactor feed pump and its check valve, and it is used to maintain a minimum flow through the feed pump at startup and low load operation, to avoid pump vibration and high running temperatures. Proportional flow control valves in each recirculation line will be regulated by a controller that senses the flow through the pumps by flow elements located in the pump discharge line and recirculation line and uses pump differential pressure as a set point reference. The control valves open proportionally as flow drops below 50 percent of the pump's best efficiency flow at that speed.

The reactor feed pumps are driven by variable speed, multistage turbines that receive steam from either the main steam cross-connection header or the crossover piping downstream of the moisture separators. During normal full power operation the turbine drivers use low pressure crossover steam. High pressure main steam is used during startup, low load, or transient conditions when crossover steam is either not available or is of insufficient pressure. The exhaust steam from each turbine driver is piped to the main condenser.

Before starting the reactor, the feedwater lines between the condensate demineralizer and the reactor are flushed with condensate to remove any crud present. To do this, approximately 50 percent of valves wide open flow is pumped through these lines by the condensate pumps, bypassing the reactor feed pumps and recirculating the flow to the condenser through the cleanup line. This ensures proper reactor water quality during startup and establishes the feedwater flow prior to admitting water into the reactor vessel.

The feedwater flow branches into two separate lines inside the reactor building. Primary containment isolation in each branch is provided by a motor operated stop check valve for the outermost containment valve and a testable check valve just outside the containment wall. A check valve and motor operated gate valve are located just inside the containment. (Subsection 6.2.4.)

Normally, the four condensate and three reactor feed pumps are in service together with all three strings of feedwater heaters. The system is designed so that a minimum of 68 percent of the rated feedwater flow can be maintained, without a reactor scram, with only two condensate pumps or two reactor feed pumps or two feedwater heater strings in service.

Both the condensate pumps and reactor feed pumps are designed to provide the maximum required design flows plus adequate margin to account for both 10 percent transients and 5 percent pump wear. Adequate margin is provided in the net positive suction head requirements to ensure noncavitating performance under all operating and runout conditions.

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 provided as necessary (see Section 12.1 for details). The condensate and feedwater system is designed to minimize leakage, with welded construction used where practicable. Relief valve discharges and operating vents are handled through closed systems.

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 removing from service and bypassing equipment and sections of the system.

An abnormal operational transient analysis of the loss of feedwater heater string is included in Chapter 15.

The probability of releasing radioactivity to the environment due to a pipe break outside the primary containment is minimized by the containment isolation valves. The primary containment prevents the release of radioactivity to the environment should a feedwater line break occur inside the primary containment.

The nonseismic portions of the condensate and feedwater system, ie, those portions upstream of the outermost containment isolation valves, are not essential for safe shutdown of the plant. If a pipe break occurs in the nonseismic piping, the reactor level will fall and on low-low level the high pressure coolant injection pumps will be started automatically and a reactor trip will be initiated. The location of equipment in the turbine building in the vicinity of the feedwater piping is such that no safety related component could be flooded by a rupture in the feedwater lines.

10.4.7.4 Tests and Inspections

That portion of the feedwater system designed to ASME Section III Class 1 or 2 is inspected and tested in accordance with Articles 5000 and 6000 of the appropriate sections. That portion of the condensate and feedwater system designed to ANSI B31.1 is inspected and tested in accordance with Paragraphs 136 and 137.

Performance tests are made on all condensate and reactor feed pumps in accordance with ASME Power Test Codes for Centrifugal Pumps, PTC 8.2.

The casings of the condensate and reactor feed pumps are hydrostatically tested to 1.5 times their shutoff discharge pressures. The shell and tube side of all feedwater heaters and drain coolers are hydrostatically tested to 1.5 times their design pressure in accordance with ASME Section VIII.

Before initial operation, the completed condensate and feedwater system will receive a field hydrostatic test and inspection in accordance with the applicable code. Periodic tests and inspections of the system are performed in conjuntion with scheduled maintenance outages.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

10.4.7.5 Controls and Instrumentation

Controls are provided to maintain condenser hotwell level so that on high level the condensate pump discharges the surplus condensate to the condensate storage tank while on low level makeup from the storage tank is admitted to the system.

Each feedwater heater is provided with two level transmitters to maintain the correct level of condensate in the respective heater shell.

One transmitter modulates the valve which controls the normal drain flow from the respective heater to the next lower pressure heaters. The other transmitter modulates the high level dump valve and this operates only when the water level in the heater shell rises above the high level setting of the transmitter regulating the normal drain flow. The dump valve discharges directly into the main condenser. On loss of control air the normal drain valve fails closed while the dump valve fails open. This prevents the flooding of the next lower pressure heater.

In addition to two level transmitters, each of the three higher pressure feedwater heaters has a high-high level switch and each of the two lower pressure heaters has two high-high level switches. Operation of the high-high level switches in the top three heaters closes the isolation valves in the respective extraction steamlines to prevent water induction into the turbine. Operation of one of the high-high level switches on the two lower pressure heaters closes the drain valve from the preceding heater. If after this action the water level continues to rise, the second high-high level switch will isolate the entire heater string on the feedwater side.

The reactor is filled initially by the condensate pumps, through the start-up control valves which bypass the reactor feed pumps and allows the condensate pumps to discharge directly into the reactor. During this period the level in the reactor is maintained by the smaller start-up valve which is modulated manually from the control room.

When reactor pressure approaches condensate pump pressure a reactor feed pump is started and feedwater is supplied to the reactor through the large start-up control valve which is regulated by single element control (reactor level). This mode of control will operate until the reactor is up to approximately 20 percent of power level. At this point the RFP discharge valves are opened, the large start-up control valve closed, and the feedwater placed on 3 element control which regulates the feedwater flow by adjusting the speed of the reactor feed pumps.

Monitoring systems including pressure indicators, flow and temperature indicators, and alarms for abnormal conditions are provided in the control room to ensure the proper operation of system components.

The reactor recirculation pumps will automatically run back reactor power to 68 percent upon loss of permissive if any one of the following occurs:

a) Low pressure at discharge of any condensate pump

b) Isolation of one string of feedwater heaters due to high-high water level in feedwater heaters one or two (Subsection 10.4.10)

c) Low flow to the reactor at any reactor feed pump discharge.

Flow elements are provided at all major points along the condensate and feedwater system, in the feedwater heater drains, and in the makeup and reject lines. Abnormal flows will indicate pipe breaks or tube leaks.

The feedwater control system is described in Subsection 7.7.1.4.

10.4.8 STEAM GENERATOR BLOWDOWN SYSTEM (PWR)

Not applicable to BWR.

10.4.9 AUXILIARY FEEDWATER SYSTEM (PWR)

Not applicable to BWR

10.4.10 EXTRACTION STEAM AND FEEDWATER HEATER DRAIN AND VENT SYSTEM

10.4.10.1 Design Basis

The Extraction Steam and Feedwater Heater Drain and Vent System has no safety related function.

a) Extraction Steam System

The extraction steam system is designed to supply steam from intermediate stages of the main turbine to closed feedwater heaters to heat the feedwater to 387°F at VWO load. The system is designed and meets the requirements of the ASME standard, "Recommended Practices for the Prevention of Water Damage to Steam Turbines Used for Electric Power Generation, Part 2 Nuclear Fueled Plants", to prevent water induction into the turbine.

b) Feedwater Heater Drain and Vent System

The feedwater heater and drain cooler vents and drains system is designed to accomplish the following objectives during steady and transient loads from startup to full load to shutdown:

1) Remove non-condensable gases continuously from the condensed drains to assure good heat transfer over the tube surfaces.

2) Cascade drains continuously from each heater to the next lower pressure heater and thence to the main condenser while maintaining the desired water level in all heaters.

3) In the event of excessively high level in any heater, dump the drains from that heater directly to the condenser.

4) Prevent water backing up into the turbine through the extraction lines to heaters 2A, 2B and 2C.

These systems are designed and installed in accordance with the requirements of applicable codes, and standards shown in Table 3.2-1.

The design pressures and temperatures of the feedwater heaters is shown in Table 10.4-6.

10.4.10.2 System Description

The extraction steam and feedwater heater drain and vent system is shown in Figures 10.4-6 thru 10.4-8. This system includes:

Bleeder trip valves

Extraction steamline motor operated isolating valves

Extraction steamline drain valves

Associated piping, valves, controls, and instrumentation

Extraction Steam System

Under normal operation, steam is extracted from the turbine to the shell side of the feedwater heaters to heat feedwater flowing on the tube side. Five stages of heating are provided. The pressure, temperature, and flow of the extraction steam vary in accordance with the load on the turbine.

Bleeder trip valves (BTV), extraction steam isolation valves and associated drain valves are provided in the extraction lines as required by the turbine manufacturer to prevent turbine overspeed due to flashing steam flowing back into the turbine in the event of a turbine trip, and also to prevent water induction into the turbine in the event of heater tube failure.

The extraction lines to feedwater heaters 3A, B & C, 4A, B & C, and 5A, B & C each contain a bleeder trip valve and in addition each has a motor-operated isolating valve located downstream of the BTVs.

No non-return valves are installed in the extraction lines to heaters 1A, B & C and 2A, B & C since the use of antiflash baffles in these heaters eliminates the need for them.

Each of the extraction lines to heaters 3A, B & C, 4A, B & C, and 5A, B & C is provided with two drain lines, one on the turbine side of the respective bleeder trip valve and the other between the bleeder trip valve and the respective isolating valve.

The drain lines on the turbine side of the bleeder trip valves in the extraction lines are interlocked with their respective bleeder trip valves and isolating valves such that when either of these valves close the drain valves open.

The valves in the drain lines between the bleeder trip valves and the isolating valves will open on either turbine trip or high-high level in the respective heater. The high-high level signal and turbine trip will also close the respective isolating valve.

Once closed the isolating valve can only be opened by the operator and only when the water level has returned to normal.

In all cases the interlocks can be overridden by hand switches so that at start up the drains can be operated in accordance with GE's instructions.

The drains from Heaters 1A, B & C flow continuously through unvalved drain lines to their respective drain cooler. From there the drains discharge into the main condenser, entering at a point below the elevation of the heaters, thus preventing flooding of the heaters after a main turbine trip or as a result of large feedwater heater tube leakage.

The bleeder trip valves are provided with side-pilot air cylinders that, on loss of air pressure following either high feedwater heater level or main turbine trip, provide a closing impulse to the bleeder trip valves. This closing impulse is primarily to insure that the bleeder trip valve disc is not "hung-up" from long periods of operation in the open position. A test switch is provided for each bleeder trip valve for checking of the valve's operability with the positive closing cylinder.

Moisture Separator Drains

Drains from each moisture separator are collected in an integral drain tank and then discharged to Feedwater Heaters 4A, B, C, or to the condenser as dictated by level controllers on the drain tank.

Feedwater Heater Vents and Drains System

During startup, the large volume of air in the heater shell is vented into the main condenser through a remote manual operated valve in the startup vent header from each heater.

During plant operation, a manually set throttling valve in the operating vent header from each heater passes air and noncondensable gases continuously with minimum steam carryover, into the main condenser.

The nozzle connection in the operating vent line on each heater is sized to choke the flow in the line such that an orifice or throttling valve is not necessary.

The heater drain normal-level and high-level (dump) control valves for each heater are each controlled by an individual level transmitter and associated controller. Normally, only the heater normal-level drain valve is modulated to maintain heater shell level.

The high-level (dump) control valve opens only when the water level rises above the high level setting of the normal level controller and passes the drains directly to the condenser.

Loss of control air results in loss of control over the heater water level. To prevent sudden flooding of the next lower pressure heater on control air failure, the normal-level control valve is designed to fail closed on loss of air and the dump valve to fail open on loss of air. Both control valves have handjacks for manual positioning of the valve.

The heater drain lines from feedwater heaters 1A, B & C and associated external drain coolers to the condenser shells serve as loop seals; therefore drain control valves are not required. This arrangement precludes water from backing-up through Heaters 1A, B and C extraction lines into the main turbine should large feedwater tube leaks occur. Flow from each drain cooler is measured by a venturi flowmeter and the signal is fed to the computer to detect tube leakage.

Shell relief valves are provided to prevent shell over-pressure in the event of tube leakage and closing of the bleeder trip valves in the extraction lines.

Feedwater heaters 1A, 1B, 1C, 2A, 2B, and 2C each have a high-high level switch which when operated will close the respective drain valve from the preceding heater and annunciate an alarm. If the level in the respective heater continues to rise after the alarm has been annunciated a separate level switch will operate to isolate the entire heater string on the feedwater side.

10.4.10.3 Safety Evaluation

During operation, radioactive steam is present in the extraction steam piping and feedwater heater shells. Shielding and controlled access are provided as necessary (see Section 12.3 for details). Both systems are designed to minimize leakage, with welded construction used where practicable.

10.4.10.4 Tests and Inspections

Each feedwater heater and drain cooler will receive a shop hydrostatic test of 1.5 times its design pressure. All tube joints of feedwater heaters are to be shop leak tested. Prior to initial operation, the condensate and feedwater system receives a field hydrostatic test and inspection to varify the system integrity. Periodic tests and inspections of the system are performed in conjunction with scheduled maintenance outages.

The system will be preoperationally tested in accordance with the requirements of Chapter 14.

10.4.10.5 Controls and Instrumentation

Instruments and controls are provided to measure extraction steam temperature and pressure. Controls are provided to actuate automatically the extraction steam check valves, shut-off valves, and drain valves as described in Subsection 10.4.10.2.

Instrumentation and controls are provided for regulating the heater drain flow rate to maintain the proper condensate level in each feedwater heater shell or heater drain tank. High-level alarms and automatic dump action on high level and automatic isolation of a heater on high-high level are also provided.

Alarms are provided in the control room to alert the operator to any necessary action.

10.4.11 AUXILIARY STEAM SYSTEM

10.4.11.1 Design Bases

The Auxiliary Steam System has no safety related function; is non-Seismic Category I, and is designed to operate independently of the Nuclear Steam Supply System. The system provides clean steam to various plant processes.

The Auxiliary Steam Boilers are designed in accordance with Section 1, ASME Boiler Vessel Code, and in compliance with applicable state and local regulations.

The system is designed to provide the operational flexibility necessary to accommodate the varying steam demands during all modes of operation.

10.4.11.2 System Description

The Auxiliary Steam System consists of two electrode steam boilers with integral recirculation pumps, one deaerator with feedwater heater and recycle pump, two boiler feed pumps, two boiler blowdown separators, two chemical feed tanks and pumps, and all associated piping, valves, controls, and instrumentation. All major components of the Auxiliary Steam System are in the Unit 1 turbine building. The boiler controls are designed for automatic operation. The piping system is designed for single or dual boiler operation.

A pressure reducing station maintains system pressure. Final pressure reduction is accomplished, as required, by individual valves adjacent to the equipment served.

The Auxiliary Steam System provides a backup for the clean steam seal system. The boilers will be cycled on and off via a header pressure switch which will maintain the boilers at or near
15 operating pressure and temperature during no load periods. In case of sudden demand for steam the ramped "on" cycle is overridden by a supplementary header pressure switch to permit full output in less than 20 seconds.

3 Auxiliary steam is also required for the radwaste evaporators and off-gas recombiner.

During initial plant startup, auxiliary steam will be used to test the HPCI, RCIC, and RFP turbines, to operate the steam jet air ejectors, and for condenser hotwell deaeration.

The condensate from any potentially contaminated source will be routed to the radwaste system or the main steam condensate tank.
3 Condensate from some steamline traps will be returned to the deaerator.

Makeup for the auxiliary steam boilers is taken from the demineralized water system through the deaerator.

The boiler water conductivity controller regulates conductivity by either activating the chemical feed pump on low conductivity or causing the boiler to blow down to the radwaste system on high conductivity.

The only connections between the Auxiliary Steam System and Seismic Category I systems are those used for preoperational testing of the HPCI and RCIC turbines. The connections are made through removable pipe spools. Before startup the pipe spools are removed and the Auxiliary Steam System is disconnected from any Seismic Category I system for plant operation. The connections between the Auxiliary Steam System and non-Seismic Category I portions of the Main Steam System are protected by normally closed valves.

10.4.11.3 Safety Evaluation

The Auxiliary Steam System is designed such that a failure of the system will not compromise any safety related system or prevent a safe reactor shutdown. The system is protected against high pressure, low and high boiler water level, overcurrent, and other malfunctions.

10.4.11.4 Test and Inspections

The Auxiliary Steam System will be proven operable by its use during startup and normal plant operations. The system will be preoperationally tested in accordance with the requirements of Chapter 14.

10.4.11.5 Instrumentation Application

Each boiler is equipped with temperature, pressure, and level indicators. Flow in each boiler outlet and pressure in the main header is indicated in the control room and on the boiler local panel.

The boiler feed pumps are operated by the boiler level controllers.

Abnormal water level, conductivity or pressure is alarmed on the local panels and as a group alarm in the control room. 3

TABLE 10.4-2

INFLUENT CONCENTRATIONS TO THE
CONDENSATE DEMINERALIZER SYSTEM

(Parts Per Billion)

Constituents	Normal Operation	Start-Up
Iron (Fe)		
Soluble	5	40
Insoluble	50	1000
Nickel (Ni)		
Soluble	5	30
Insoluble	5	100
Copper (Cu)	5	10
Chloride (Cl)	10(1)	10
pH at 25°C	6.5 to 7.5	6 to 8
Conductivity at 25°C	0.2 mho/cm	0.5 mho/cm
Radioactivity Ci/ml	10 to 10	10 to 10

(1) For equipment specification only because, without condenser tube leak, chlorides are much lower.

CHAPTER 11.0

RADIOACTIVE WASTE MANAGEMENT

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11.2 LIQUID WASTE MANAGEMENT SYSTEMS

11.2.1 DESIGN BASES

The objective of the Liquid Waste Management System (LWMS) is to collect, process, store, and monitor for reuse or disposal all potentially radioactive liquid wastes.

The LWMS is capable of recycling the majority of potentially radioactive wastes to condensate quality. Sufficient treatment equipment is available to process the liquid waste from both nuclear units without impairing the operation or availability of the plant.

Liquid wastes that cannot be processed to meet the quality requirements for recycling are solidified along with process concentrates and solid wastes for offsite shipment and disposal.

During operation there may be a buildup of the plant condensate inventory due to the introduction of recycled liquid from regeneration chemicals, auxiliary steam for equipment testing and startup, demineralized water for decontamination, and domestic water for laundry purposes. Evaporation losses from the fuel pools, equipment leakages discharged by the building ventilation systems, and water contained in the solidified waste decrease the condensate inventory.

The excess water is released in a controlled and monitored manner into the cooling tower blowdown line for dilution.

The expected radionuclide activity concentrations in the reactor water corresponding to fuel defects that result in 60,000 μCi/sec noble gas activity for one reactor unit after a 30 minute delay are shown in Table 11.2-9. The design basis radionuclide activity concentrations corresponding to fuel defects that result in 100,000 μCi/sec noble gas activity for one reactor unit after a 30 minute delay are shown in Tables 11.1-1 through 11.1-5.

Subsection 15.7.3 discusses the liquid radwaste tank rupture accident.

The LWMS is designed so that no potentially radioactive liquids can be directly discharged to the environment unless they have been monitored and diluted. This results in radionuclide activity concentrations in the offsite release and radiation exposures to individuals and the general population (on an annual averaging basis) within the limits of 10CFR20 and 50 (Table 11.2-13).

The LWMS is designed to keep the exposure to the general population and plant personnel during normal operation and maintenance ALARA.

Redundant and backup equipment, alternate process routes, interconnections, and spare volumes are designed into the system to provide for operational and unanticipated surge waste volumes due to refueling, abnormal leakage rates, decontamination activities, LRWM equipment down time, maintenance, and repair.

The expected daily inputs and activities to each of the three subsystems is shown in Tables 11.2-1 and 11.2-2. An evaluation of the causes for the maximum expected inputs for each subsystem shows that operational modes exclude, and the unlikely occurrence of the same failure in both units minimizes, the potential for coincidental maximum input from both units into the same subsystem. Concurrent refuelings or cold startups are not design bases for the plant.

The usage factors for pumps and processing equipment provided in Table 11.2-3 show sufficient reserve capacities for the maximum expected inputs.

Table 11.2-3 shows the design parameters of the LWMS equipment. Major LWMS components are located in separate shielded compartments according to their radiation level and considering accessibility for maintenance and repair while operating redundant components of the system. Rooms of components containing significant amounts of liquid radwaste are provided with elevated door thresholds to minimize potential spread of contamination from leaks. All liquid radwaste tanks are located below ground level. (Figures 11.2-1 through 11.2-7, 9.3-10 and 9.3-11) . Instrumentation and controls are designed and located to minimize exposure to the operating personnel.

Floor drains and sloped floors are provided in equipment rooms to control the spread of contamination from leakage. Refer to Subsection 2.4.13.3 for the analysis of accidental release of radioactive waste to the groundwater.

The LWMS has no nuclear safety related function as a design basis.

The seismic and quality group classifications of the LWMS components and piping and the radwaste building are listed in Section 3.2.

All tanks located outside reactor containment and containing radioactive materials in liquids are designed tc prevent uncontrolled releases of radioactive materials due to spillage in buildings or from outdoor storage tanks. The following design

a) Automatically and manually operated valves and instrumentation located outside the large radioactive volume containing equipment rooms, unless required by the process

b) Sequencer controlled valve setting and pump operations upon manual initiation of main process steps

c) Automatic flushing of subsystems after process termination

d) Manual override provisions for all sequencer operated and interlocked components

e) Manholes and access ladders on storage tanks

f) Remote manual drain valves on storage tanks

g) Low point piping and equipment drains in isolable portions of systems

h) Internal decontamination spray nozzle in the spent resin tank

i) Condensate flushing connections on all major piping routes

j) All vents of radwaste tanks, filters, demineralizer and evaporator condensers, air tightly routed to the building ventilation system filters. A slight negative pressure against atmosphere is maintained in these components when vented.

k) Welded connections where practical. Line sizes over 2 in. are butt welded to avoid crud traps.

l) Pumps provided with mechanical seals with flush connections

m) Pump baseplates with drip lips.

The expected and design basis radionuclide activity inventories of liquid radwaste system components containing significant amounts of radioactive liquids are shown in Table 11.2-5, 11.2-6 and 11.2-7 and are based upon the following assumptions:

a) Expected flow rates for streams shown on Figure 11.2-8, as given in Tables 11.2-4 and 11.2-10. The radionuclide activity concentrations in a given pipe of the LWMS vary depending on the origin of the stream. For example, the inlet pipe to the radwaste evaporator will contain, at one time, waste from the chemical waste collector tank, at another time waste from the chemical waste neutralizer tank. If the decay times due to residence in the tanks are conservatively ignored, then the highest radionuclide activity concentrations in each pipe are obtained by dividing the appropriate tank activity inventory by the tank contents.

b) Reactor water isotope activity concentrations as listed in Tables 11.1-2 through 11.1-5 for design conditions, and Table 11.2-9 for expected conditions

c) Emission rates of noble gases as listed in Table 11.1-1 for design conditions, and Table 11.2-9 for expected conditions

d) Two percent of the halogens in the reactor water carries over into the condensate

e) One-tenth of one percent of the nonhalogen radioisotopes in the reactor water carries over into the condensate

f) For five minutes after release from the reactor vessel nozzles, the decay daughter products of xenon and krypton of the off-gas are deposited in the condensate flow

g) Radwaste inputs, total activities, and component parameters based on data from operating plants (Dresden, KRB, etc), data collected by GE, and design data for Susquehanna SES as shown in Tables 11.2-1 through 11.2-3 for expected averages during normal operation

h) Decontamination factors within the liquid radioactive waste system for:

Holdup:	radioactive decay
Filtration:	1.0, no decontamination
Evaporation:	10^3 for iodine
	10^4 for all other isotopes
Demineralization:	10 for cesium and rubidium
	10^2 for anions and other isotopes

i) While a process stream is collecting in a collection tank the isotopes already in the tank are undergoing radioactive decay.

j) Four hours are required to complete sampling of a sample tank. The sample tank is isolated during this period.

k) All water volumes are homogenous mixtures.

l) One refueling shutdown per year per unit not occurring simultaneously.

All atmospheric liquid radwaste tanks are provided with an overflow connection of at least the size of the largest inlet connection. Common overflow from laundry drain tanks (OT-311A&B) to Chemical Radwaste funnel is exempt from this requirement. The two tanks are interconnected by a 4" overflow line. See Section 11.2.2.4. The overflow is connected below the tank vent and at least one inch above the high level alarm trip point. Overflow liquid is routed to a redundant tank or the nearest atmospheric drainage point.

Processed wastes are collected in sample tanks prior to their reuse as condensate quality water or monitored discharge into the cooling tower blowdown pipe for dilution before entering the Susquehanna River. These measures minimize the potential for uncontrolled release due to operator error or equipment malfunction.

Control and monitoring of radioactive release in accordance with General Design Criteria 60 and 64 of Appendix A to 10CFR50 is discussed in Subsection 11.2.3 and Section 11.5.

11.2.2 SYSTEM DESCRIPTIONS

11.2.2.1 General

The Liquid Waste Management System serves both reactor units and consists of three basic subsystems, each for collecting, processing, storing, monitoring, and disposing of specific types of liquid wastes according to their conductivity, chemical composition, and radioactivity. These subsystems are:

a) Liquid Radwaste Processing

b) Liquid Radwaste Chemical Processing

c) Liquid Radwaste Laundry Drain Processing

Waste influent to each of the subsystems is collected in batch tanks to allow for quality and volume monitoring before processing.

Recirculation of the collection and sample tank contents while isolated or being pumped out minimizes settling of suspended solids and provides representative grab samples. The recirculation lines with stroke limited valves guarantee a minimum pump flow for cooling in case a pump discharge valve is closed.

Recirculation and pump-out of all process tanks is remote manually initiated and ceases upon a low level signal. This protects the pumps from cavitation. Manual fill selection of multiple tanks is provided. In the automatic mode, the tanks are filled sequentially to their high level. High level alarms and level indication over the live volume range are provided in the radwaste control room.

Simultaneous filling of one tank and mixing, sampling, or processing of another is possible through separate suction and recirculation lines and pumps for each tank.

A local pressure gage is provided in each pump discharge line. A more detailed description of the instrumentation and controls of the liquid waste management systems is contained in Section 7.7.

Suction lines of multiple pump and tank arrangements are cross-connected to provide backup capability. Manual valves and individual controls for all automatic valves and pumps also allow transfer of waste between tanks, complete pump-out of tanks for maintenance or repair, system flushing with condensate, and bypassing of process equipment.

The following subsections describe additional features of each of the three subsystems.

11.2.2.2 Liquid Radwaste Processing Subsystem

Refer to Figures 11.2-9 and 11.2-10.

During normal plant operation, high and low purity wastewater originating from potentially radioactive equipment leakage, floor drains, and other sources throughout the plant (see Table 11.2-1) is routed through local sumps or directly to the Liquid Radwaste Processing Subsystem in the radwaste building.

Three sets of not individually isolable twin tanks collect the plant low conductivity ($\leq 100\ \mu$mho/cm at 25°C) waste in batches. A radwaste building control room alarm annunciates when two out of three tank sets are unavailable. A main control room alarm announces when all three collection tank sets are unavailable and the wastewater is routed into the surge tanks. These additional two twin sets of tanks provide surge capacity for unanticipated high waste volumes. They feature internal mixing eductors and are associated with one common pump.

Processing of liquid waste is normally done by separate filtering and ion exchanging. High conductivity waste inadvertently collected in the liquid radwaste tanks can be pumped to the chemical waste tank.

Condensate quality water in the sample tanks is transferred to the condensate storage tank by the three liquid radwaste sample tank pumps. Excess water is discharged through the monitored discharge pipe into the cooling tower blowdown pipe.

Radwaste Filters

Two vertical centrifugal dry cake discharge radwaste filters are provided for:

a) Filtering of low conductivity liquid radwaste

b) Dewatering of phase separator sludges and spent demineralizer resins.

The two radwaste filters are piped in parallel and in series. Normally only one filter is used for filtering of liquid waste with the second one available for dewatering service or used as a backup to the radwaste demineralizer.

For the filtering mode, a thin precoat layer of diatomaceous earth (DE) or powdered resin is deposited on the stacked filter screen plates. Normal filtering flow is 200 gpm which is 2/3 gpm per sq. ft. Based on operating experience, an adjustable amount of filter aid may be injected into the waste inlet stream to extend the filter run length over the full allowable differential pressure range up to 90 psi.

The precoat and filter aid pumps and tanks are supplied with the filters and are located in a normally accessible area.

The approximately 5 w/o solids containing phase separator sludges and spent demineralizer resins are separately dewatered at a flow rate of 50 to 100 gpm.

When used as radwaste demineralizer backup, the filter plates are precoated with powdered ion exchange resin.

The filtering/dewatering process is terminated upon a high differential pressure alarm across the filter or the maximum allowable cake thickness between the filter plates. The latter can be observed through an illuminated sight glass in the filter vessel shell and a filter run timer set accordingly.

Flow controllers keep the flow rate independent of the increasing pressure drop over a filter run length.

The effluent from the radwaste filters contains ≤ 1 ppm suspended solids of 1 micron or larger size provided the feed contains ≤ 100 ppm of 1 micron size particles or ≤ 1000 ppm of 5 micron size particles.

Upon termination of a filter run, the filter vessel is drained to the waste sludge phase separator and the cake dried by blowing 700 scfm of 120°F saturated air at 30 psig through it. This reduces the water content of the cake prior to solidification and disposal. The length of the drying periods is established by experience and the dew point instrument alarm in the drying air exhaust line set accordingly. The humid air passes through the radwaste mist eliminator before entering the radwaste tank filter described in Subsection 9.4.3.

The dried filter cake is spun off the filter plates by motorized rotation of the vertical stacking shaft, and a scraper at the vessel bottom discharges it through a vertical chute into the waste mixing tank of the solidification system described in Section 11.4.

After a brief backflushing into the waste sludge phase separator, the filter is filled and ready for a fresh precoat.

For a cake drying time of 1 hr the filter is back in service in approximately 4 hr.

Radwaste Demineralizer

The filtered liquid waste is processed through one 140 ft^3 nonregenerated deep bed demineralizer before entering the liquid radwaste sample tanks. A normal flow rate of 200 gpm is maintained through the 4:1 to 1:1 cation to anion ratio ion exchange resin bed. The effluent conductivity is indicated, recorded, and alarmed at 0.5 and 1.0 mho/cm at 25°C in the radwaste control room. The differential pressure between the vessel inlet and outlet is also indicated and alarmed over an adjustable range up to 25 psi.

The differential pressure and a level indication is provided on a local instrument rack and in the radwaste control room.

The demineralizer inlet valve will close automatically upon > 1 mho/cm at 25°C conductivity in the effluent, high differential pressure, or loss of control air or power. Exhausted or fouled ion exchange resins are sluiced to the spent resin tank for subsequent dewatering in the radwaste filters and solidification.

Fresh resin beads are manually loaded through a hopper above the demineralizer vessel and mixed inside by compressed air. The air is vented through the radwaste mist eliminator to the radwaste tank filter described in Subsection 9.4.3.

Total outage time for removal and replacement of resins is approximately 6 hr.

11.2.2.3 Liquid Radwaste Chemical Processing Subsystem

Refer to Figure 11.2-11.

High conductivity ($\geq$100 mho/cm at 25°C) wastes from potentially radioactive sources throughout the plant listed in Table 11.2-1 (Chemical RadWaste) are routed via local sumps or directly to the liquid radwaste chemical processing subsystem. Except for the chemical waste neutralizer tanks located in the turbine building, all components of this subsystem are located in the radwaste building.

Diluted sulfuric acid and sodium hydroxide from the condensate deep bed demineralizer regeneration process are collected in two twin sets of chemical waste neutralizer tanks associated with each reactor unit.

Each tank set accommodates two batches of regeneration chemicals. One of the two redundant chemical waste neutralizer tank pumps recirculates the chemicals through internal mixing eductors in each tank while local grab samples may be taken. Remote pH indication with high and low point alarms are provided in the radwaste control room. In order to bring the pH value of the chemical waste in the neutralizer tanks within the required range of six to eight for evaporation, small amounts of sulfuric acid and sodium hydroxide are injected from the acid and caustic storage tanks through separate lines.

Various chemical solutions originating from laboratory, equipment, and sample rack drains and decontamination stations throughout the plant are collected in the chemical waste tank located in the radwaste building. Auxiliary boiler blowdown waste is also collected in the chemical waste tank due to the possibility of radioactive contamination from evaporator tube leaks. The chemical waste tank contents are recirculated by one of the two redundant chemical waste tank pumps while remote grab samples may be taken on the radwaste building sample rack.

Remote pH indication with high and low point alarms are provided in the radwaste control room. The pH value of the chemical waste tank contents is adjusted in the same manner as that of the chemical waste neutralizer tanks. Pumps for chemical wastes are provided with automatic gland seal flushing with condensate.

Equipment and piping containing evaporator concentrate is heat traced to 130°F to prevent precipitation of solids from the liquid and plugging of the lines.

During operation of a radwaste evaporator, distillate is continuously collected in the evaporator distillate sample tank or the LRW sample tanks. One of two redundant evaporator sample tank pumps recirculates the tank contents through tank internal mixing eductors while remote grab samples are taken at the radwaste sample rack. Distillate meeting condensate quality is then discharged to the condensate storage tank. Distillate that does not meet the quality requirements is returned to the chemical waste tank or the liquid radwaste collection tanks. Surplus distillate may be discharged to the Susquehanna River by way of the monitored discharge pipe.

Evaporator concentrate is directly discharged in batches to the waste mixing tank of the radwaste solidification system described in Section 11.4 or temporarily stored in the evaporator concentrate storage tank.

This tank is associated with a single pump for recirculation and transfer of the concentrate to the waste mixing tank of the solidification system.

The controls and instrumentation of the liquid radwaste chemical processing subsystem are as described in Subsection 11.2.2.1 except as follows.

The chemical waste neutralizer, chemical waste, and evaporator distillate sample tanks are equipped with level recording instrumentation in place of indicating instruments to provide evaporator performance records.

Inlet flow and pH value to each evaporator is indicated in the radwaste control room with an alarm for high and low pH.

Radwaste Evaporators

Two radwaste evaporators are piped in parallel for simultaneous operation and as backup to each other. Depending on the concentration in the shell, each radwaste evaporator can process 15 to 30 gpm of radioactive waste. Concentration is limited by precipitation of solids out of the solution and increased carry-over of iodine and other volatile activity into the distillate to approximately 25 w/o. The contents of one neutralizer tank or the chemical waste tank can be processed through one or both evaporators at the same time.

Each evaporator can separately process the contents of one tank provided the suction streams are not mixed in cross-over lines.

The radwaste evaporators are of the forced circulation design with bowed titanium tubes for chill-shock descaling. A manhole permits access to the shell for clean-out.

Heating steam is provided from the two auxiliary boilers in the turbine building, allowing both evaporators to operate during reactor shutdowns.

An electric heater is provided in the evaporator shell to keep the concentrate in solution during steam interruptions and startups. The evaporators are designed for automatic unattended process operation until the desired bottom concentration, as determined by on-line indication or local grab sampling, is obtained. Pump-out of the cooled concentrate as a batch, startup, and blowdown require attendance of an operator at the radwaste control room panel.

Influent to the evaporators is controlled by the level in the shell to keep the tubes submerged.

Through-put (distillate produced and feed rate) is manually set and automatically controlled by the flow of cooling water to the distillate condenser. The evaporators operate at 0-5 psig and are of fail safe design, recirculating the process streams internally when isolated.

The heating steam of the evaporators is collected and cooled in condensate return tanks for reuse in the auxiliary boilers.

A pump for recycling and returning of this condensate to the auxiliary boiler deaerator is provided with each tank. The discharge stream is monitored and, upon high conductivity that indicates an evaporator tube leak, it is diverted to the liquid radwaste collection tanks.

Service water (cooling tower water quality) is used to cool the evaporator distillate and the auxiliary steam condensate.

Instrumentation and controls of the evaporator assemblies are located in the radwaste control room and include: evaporator shell (concentrate) level indication with high and low alarms, concentrate temperature indication with low alarm, concentrate recirculation flow low alarm, shell pressure indication with high and low alarm, distillate conductivity indication with high alarm, distillate temperature indication with high and low alarms, evaporator condenser level indication with high and low alarms, condensate return tank level indication with high and low alarms, evaporator condenser cooling water inlet and outlet temperature indication.

The evaporator shell is shielded by a concrete block wall to reduce operator exposure during maintenance and repair of the evaporator condenser, the concentrate or distillate pumps and instrumentation located in a local rack on the evaporator assembly skid.

11.2.2.4 Liquid Radwaste Laundry Processing Subsystem

Refer to Figure 11.2-12.

Detergent-containing wastewaters from the white and blue laundries in the control structure and the radwaste building, and from various equipment washdown stations and personnel decontamination facilities throughout the plant, are routed to the liquid radwaste laundry processing subsystem located in the radwaste building.

The bulk of the input to this subsystem originates from clothes washers using domestic water.

The wastewater processed in this system is expected to be normally of low radioactivity as shown in Table 11.2-13 (Laundry Radwaste). Influent to one of the two laundry drain tanks is selected from the radwaste control room.

The two tanks are interconnected by a 4" overflow line below the overflow connection piped to the chemical radwaste sump.

Each tank is associated with a pump for recirculation through internal mixing eductors or processing of the contents through one of two cartridge type filters.

The pumps are protected by coarse strainers in the suction lines. Both pumps and filters can be operated simultaneously. Cross-connections are provided to serve either or both filters by one pump.

An internal mixing eductor in the laundry drain sample tank ensures a representative grab sample on the radwaste sample rack.

Effluent from the sample tank is normally discharged by one or both laundry drain sample tank pumps through the monitored discharge pipe into the cooling tower blowdown pipe. High conductive filtrate can be transferred to the chemical waste tank. A return line allows recycling of sampled water back to the laundry drain tanks.

The controls and instrumentation of the liquid radwaste laundry processing subsystem are as described in Subsection 11.2.2.1, except as follows.

The laundry drain and laundry drain sample tanks are equipped with level recording instrumentation instead of indicating instruments to provide performance records of the laundry drain filters. High differential pressure through the wye strainers in the laundry drain tank pump suction lines is alarmed in the radwaste control room.

Laundry Drain Filters

Two cartridge type filters are piped in parallel for simultaneous operation.

The purpose of these filters is to remove particulate contamination such as lint from the laundry drain waste at a normal flow rate of 25 gpm per filter assembly. The maximum flow rate per filter assembly is 135 gpm with a removal efficiency of 98 percent for 30 micrometer and 100 percent for 49 micrometer particles on an absolute basis.

The corrugated epoxy impregnated cellulose fiber cartridges are replaced when the pressure differential alarms in the radwaste control room trip at a maximum set point of 70 psi.

Replacement of the filter cartridges is done manually because of the low expected radioactivity. Swing bolted housing closures and lift rings facilitate replacement of the cartridges.

Depending on the activity level, the spent cartridges are disposed in the compacted solid waste or in the solidified radwaste described in Section 11.4.

11.2.3 RADIOACTIVE RELEASES

During liquid processing by the LWMS, radioactive contaminants are removed so that the bulk of the liquid is restored to clean water, which is either recycled in the plant or discharged to the environment. The radioactivity removed from the liquids is concentrated in filters, ion exchange resins, and evaporator bottoms. These concentrated wastes are sent to the Solid Radwaste System (SRS) for solidification, packaging, and eventual shipment to a licensed burial ground. If the liquid is to be recycled back to the plant, it must meet the purity requirements for condensate makeup (Subsection 9.2.6). If the liquid is to be discharged, the activity concentration must be consistent with the discharge criteria of 10CFR20. Normally, most of the liquid passing through the liquid radwaste and chemical processing systems is recycled in the plant. However, the treatment in these systems is such that these liquids can be discharged from the plant after monitoring if required by plant water balance considerations. Tritiated water will be discharged from the systems consistent with the discharge criteria of 10CFR20. Normally the liquid passing through the laundry drain processing system is discharged directly, in accordance with 10CFR20 guidelines; however, it may be processed through the chemical processing system if necessary.

The resulting doses from radioactive effluents will be within the guideline values of Appendix I to 10CFR50. In addition to the radioactivity limitations on releases, water quality standards for discharge and heat content may necessitate recycling of the water, rather than discharging. A detailed analysis of these subjects is presented in Subsection 5.1.1 of the Environmental Report.

Although the plant discharges vary as stated above, this analysis assumes the following which are consistent with NUREG 0016:

a) Discharge of 1 percent of the liquid radwaste processing stream

b) Discharge of 5 percent of the chemical processing stream

c) Discharge of 100 percent of the laundry drain processing stream.

The assumptions and parameters used to calculate the yearly activity releases are given in Table 11.2-8. The yearly activity releases for each waste stream and the total are given in Table 11.2-13.

Design and administrative controls are incorporated into the LWMS to prevent inadvertent releases to the environment. Controls include administrative procedures, operator training, redundant discharge valves, discharge radiation monitors that trip alarms and automatic discharge valve closure (see Section 11.5). Prior to any discharging, activity concentrations are measured in samples taken from the various sample tanks. The discharge header receives effluents from the discharge points in the LWMS shown on Figure 11.2-13. A single line is provided for radioactive plant discharges to minimize the potential for operator error.

The processed liquid radwaste that is not recycled in the plant is discharged into the cooling tower blowdown pipe on a batch basis at up to 280 gpm from the Liquid Radwaste Processing System (LRPS), 50 gpm from the Liquid Radwaste Chemical Processing

System (LRCPS), and 10 gpm from the Liquid Radwaste Laundry Drain Processing System (LRLPS). Flow is controlled by a flow control valve. Therefore, the actual flow could be substantially less. The total cooling tower blowdown flow of 10,000 gpm for both units dilutes above discharge rates by at least a factor of 35 for the LRPS, 200 for the LRCPS, and 1000 for the LRLPS. This dilution occurs within the site boundary and is used in determining specific activity concentrations for the releases. These concentrations and a comparison to 10CFR20 limits are given in Table 11.2-14.

11.2.4 ESTIMATED DOSES

Dose calculations to assure compliance with Appendix I to 10 CFR Part 50 based on the liquid source term described above were performed in accordance with USNRC Regulatory Guide 1.109 by use of the USNRC computer code "LADTAP". To these purposes doses were calculated to a maximum individual consuming aquatic biota and receiving shoreline exposure at the edge of the initial mixing zone and drinking water from the nearest downstream supply (Danville). Input data for these calculations are given in Table 11.2-15. The calculated doses were 1.47 mrem/yr to the total body of an adult and 2.39 mrem/yr to the bone of an adult. These doses are well within the Appendix I design guides of 3.0 and 10.0 mrem/yr to the total body and any organ, respectively. (See Sections 5.2 of ER for further discussions).

TABLE 11.2-1

EXPECTED DAILY INPUTS AND ACTIVITIES TO THE LIQUID WASTE MANAGEMENT SYSTEM FROM TWO UNITS

Source	Expected Average From Two Units In Normal Operation (gpd)	Primary Coolant Activity Fraction[4] (PCA)	Maximum Expected With One Unit Cold Startup, One Unit Normal Operation (gpd)	Notes
Liquid Radwaste				
Drywell Equipment Drains	4400	1.0	4400	Recirc pump seal leak is only expected source
Drywell Floor Drains	4400	1.0	4400	
Unident. floor drains	(1000)		(1000)	
Drywell cooler drains	(1400)		(1400)	
Steam valve seal leaks	(800)		(800)	
Recirc valve seal leaks	(1200)		(1200)	
Reactor Building Drains	12,124	.01	12,124	
Unident. floor drains	(4000)		(4000)	
Cleanup pump seal leaks	(100)		(100)	
Scram valve seal leaks	(540)		(540)	
Scram valve intern. leaks	(3800)		(3800)	
Steam valve seal leaks	(800)		(800)	
Sample drains	(2880)		(2880)	
RCIC & HPCI line drains	(4)		(4)	
Turbine Building Central Area Drains	6900	.01	6900	
Unident. floor drains	(2000)		(2000)	
Cond. pump seal leaks	(1920)		(1920)	
CRD pump seal leaks	(100)		(100)	
Sample drains	(2880)		(2880)	
Turbine Building Outer Area Drains	7863	.01	120,000	
Unident. floor drains	(2000)		(2000)	
Cond. demin. resin cleaner	(5863)[1]		(118,000)	Max. for back-to-back cond. demin. resin cleaning without regen.
Radwaste Building Drains	2080	.01	2080	
Unident. floor drains	(2000)		(2000)	
Off-gas system drains	(80)		(80)	
RWCU Phase Sep. Decantate	648[1]	.002	2530	
Waste Sludge Phase Sep. Decantate	1997[1]	.002	51,200	Max. for back-to-back cond. demin. resin cleaning without regen.
Cond. Demin. Resin Cleaner				
Fuel Pool Backwash				
Radwaste Filter Drain	250[1]			
RWCU Discharge			56,000	CRD coolant, bypass and

TABLE 11.2-1 (Continued)

Source	Expected Average From Two Units In Normal Operation (gpd)	Primary Coolant Activity Fraction[4] (PCA)	Maximum Expected With One Unit Cold Startup, One Unit Normal Operation (gpd)	Notes
				thermal expansion water during cold startup (normally routed to cond. hotwell)
	----------------		----------------	
TOTAL	40,262		259,634	
Chemical Radwaste				
Cond. Demin. Regeneration Chem. Neutralizing Tanks	2952	[2]	52,334	
Lab and decon. drains	1000	0.02	1000	
Aux. boiler blowdown	153		1786	Max. for two boilers blowdown of 1% over 24 hr
	----------------		----------------	
TOTAL	3605		54,720	
Laundry Radwaste				
Laundry and Decontamination Drains	1300	[3]	6000	

(1) These inputs are averaged. The expected batch sizes and frequencies are shown in Table 11.2-2.
(2) See Table 11.2-7.
(3) See Table 11.2-11.
(4) See Tables 11.1-through 5 and 11.2-9 (Reactor Coolant).

TABLE 11.2-2

EXPECTED BATCHED INPUTS TO THE LIQUID RADWASTE SYSTEM FROM THE SOLID RADWASTE SYSTEM FOR NORMAL OPERATION OF TWO UNITS

Source	First Intermediate Collectors Input, Batch Size Each/Time	Second Intermediate Collectors Input, Batch Size Each/Time	Liquid Radwaste Collection Tank Input From Each Second Intermed. Collector, Batch Size Each/Time
Four RWCU F/D's	Two cleanup backwash receiving tanks, 1000 gal/3.4 days	Alternating at 60 days' interval for one of two cleanup phase separators 2200 gal/3.4 days	2200 gal/3.4 days
Two Fuel Pool F/D's	One fuel pool backwash receiving tank, 880 gal/9 days	One waste sludge phase separator, 1950/18 days	
Fourteen Cond. Demineralizers	Two regen. waste surge tanks, 7588 gal/9.26 days	One waste sludge phase separator, 7588 gal/4.63 days	9250 gal/4.63 days
Two Radw. Filters	-	One waste sludge phase separator, 500/2 days	
Averaged Total Input to Radwaste Collection Tanks From Solid Radwaste System			2648 gal/day

TABLE 11.2-3

LIQUID WASTE MANAGEMENT SYSTEM COMPONENT DESCRIPTION

A. PUMPS	EQUIPMENT NOS.	TYPE	QUANTITY	MATERIAL	CAPACITY EACH, GPM	TDH FT.	USAGE FACTOR NORMAL	DRIVER HP	DESIGN PRESSURE/TEMP. PSIG/°F
Liquid Radwaste Processing									
Collection Tank	0P-301A,B,C	Horiz. Centr.	3	SS	280	300	0.11(1)	50	150/155
Surge Tank	0P-302	Horiz. Centr.	1	SS	280	300	--	50	150/155
Sample Tank	0P-305A,B,C	Horiz. Centr.	3	SS	280	170	0.11(1)	30	150/155
Filter Precoat	0P-324A,B	Horiz. Centr.	2	SS	357	97	0.015	20	150/155
Filter Aid Proport.	0P-303,311	Reciprocating	2	SS/Hypalon	1.48	230	0.05	1	175/150
Liquid Radwaste Chemical Processing									
Neutralizing Tank	1P-130A,B	Horiz. Centr.	2	SS	50	175	0.02(1)	10	150/155
Neutralizing Tank	2P-130A,B	Horiz. Centr.	2	SS	50	175	0.02(1)	10	150/155
Chem. Waste Tank	0P-326A,B	Horiz. Centr.	2	SS	20	100	0.04(1)	5	150/155
Conc. Storage Tank	0P-328	Horiz. Centr.	1	SS	20	180	0.012(1)	10	150/155
Evap. Dist. Sample Tank	0P-327A,B	Horiz. Centr.	2	SS	50	180	0.052(1)	10	150/155
Evap. Concentrate	0P-329A,B	Horiz. Centr.	2	SS	50	70	0.075(1)	15	150/223
Evap. Distillate	0P-330A,B	Horiz. Centr.	2	SS	36	217	0.012(1)	8	150/125
Evap. Condensate Return	0P-333A,B	Horiz. Centr.	2	SS	50	155	0.012(1)	7.5	150/212
Liquid Radwaste Laundry Drain Processing									
Collection Tank	0P-318A,B	Horiz. Centr.	2	SS	25	220	0.036(1)	15	150/155
Sample Tank	0P-319A,B	Horiz. Centr.	2	SS	10	120	0.09(1)	5	150/155

TABLE 11.2-3 (Continued)

C. PROCESSING EQUIPMENT	EQUIPMENT NOS.	TYPE	QUANTITY	SIZE EACH	MATERIAL	CAPACITY EACH, GPM	USAGE FACTOR NORMAL	DESIGN PRESSURE/TEMP.
Liquid Radwaste Processing								
Liquid Radwaste Filter	0F-302A,B	Vert., Centr. Drycake Discharge	2	300 ft^2	SS	200	0.14	150/150
Liquid Radwaste Demin.	0F-301	Mixed Bed, Non-regen.	1	140 ft^3	SS	200	0.17	150/140
Mist Eliminator	0T-319	Vert. cycle Mech. separ.	1		SS	(700scfm)	0.015	150/250
Liquid Radwaste Chemical Processing								
Radwaste Evaporator	0E-302A,B	Horiz., Bowed Tubes. Forced Circulation	2	1200 ft^2 1500 gal	Shell: SS Channel: CS Tubes&Sheets: Ti	30/15	0.075	Shell: 50/300 Tubes: 65/350
Evap. Absorption Tower	0E-304A,B	Wire Mesh Trays	2	5'diam	SS	30	0.04	50/300
Evap. Condenser	0E-303A,B	Horiz., U-tubes	2	620 ft^2	SS	Shell: 16,400 lb/hr Tubes:620,000 lb/hr		Shell: 50/300 Tubes: 150/200
Evap. Heating Steam Cond. Return Tank	0T-333A,B	Horiz., U-tubes	2	40 ft^2	Shell; Tubes & Sheet: SS; Channel: CS	Shell: 21,000 lb/hr Tubes: 68,500 lb/hr	0.04	Shell: 65/350 Tubes: 150/200
Liquid Radwaste Laundry Drain Processing								
Laundry Drain Filter	0F-313A,B	Vert., Cyl., Fiber Cartridge	2	247 ft^2	Shell: SS Cartridge: Epoxy Impregn.Cellulose	135 maximum, 25 for 100% removal of 49 microns	0.018	150/250

TABLE 11.2-3 (Continued)

B. TANKS	EQUIPMENT NOS.	TYPE	QUANTITY	MATERIAL	LIVE/NOMINAL CAPACITY, EACH, GAL	DESIGN PRESSURE/TEMP. PSIG/°F
Liquid Radwaste Processing						
Collection	0T-302A thru F	Vert. Cyl.	6	SS	11100/15000	Atmos./200
Surge	0T-304A thru D	Vert. Cyl.	4	SS	11100/15000	Atmos./200
Sample	0T-303A thru F	Vert. Cyl.	6	SS	11100/15000	Atmos./200
Filter Precoat	0T-305	Vert. Cyl.	1	SS	860/1280	Atmos./200
Filter Aid	0T-310	Vert. Cyl.	1	SS	540/800	Atmos./200
Liquid Radwaste Chemical Processing						
Neutralizing	1T-130A,B	Horiz. Cyl.	2	SS	14000/15000	Atmos./200
Neutralizing	2T-130A,B	Horiz. Cyl.	2	SS	14000/15000	Atmos./200
Collection	0T-314	Vert. Cyl.	1	SS	11800/15000	Atmos./200
Evap. Dist. Sample	0T-321	Vert. Cyl.	1	SS	5700/7500	Atmos./200
Evap. Conc.	0T-322	Vert. Cyl.	1	SS	3730/5000	Atmos./200
Liquid Radwaste Laundry Drain Processing						
Collection	0T-311A,B	Vert. Cyl.	2	SS	820/1000	Atmos./200
Sample	0T-312	Vert. Cyl.	1	SS	1420/1500	Atmos./200

TABLE 11.2-3 (Continued)

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(1) Usage factors for these pumps consider recirculation of one tank (set) volume prior to discharge and continuous recirculation for the evaporator system. For the usage factors, flow from the liquid radwaste collection and sample tank pumps is the same as the liquid radwaste filter capacity (200 gpm).

Usage factor of processing equipment includes time for precoating, backwashing, discharging, etc.

TABLE 11.2-4

LIQUID RADWASTE SYSTEM FLOWS (Refer to Figure 11.2-5)

Stream No.		Normal Operation of Both Units Averaged Number and Batch Frequency	Volume/ Batch (gal)	Nominal Flow Rate (gpm)	Normal Operation of Both Units Av. Volume/ day (gpd)	Maximum Number and Batch Frequency	Maximim Volume/ Day (gpd)	Comment
1.	To Liquid Radwaste Filter & Demineralizer (From Coll. and Surge Tanks)	1/0.55 days	22,200	200	40,412	1/0.125 days	177,049	Maximum during cold startup of one unit
2.	To Liquid Radwaste Sample Tanks (From Radw. Demineralizer)	1/0.55 days	22,200	200	40,412	1/0.125 days	177,049	Maximum during cold startup of one unit
3.	To Condensate Storage Tanks (From Sample Tanks)	1/0.55 days	22,200	200	40,412	1/0.125 days	177,049	Maximum during cold startup of one unit
4.	To Plant Discharge Pipe (From Sample Tanks)	-	22,200	200	-	1/1.71 days	13,000	Maximum during startup of one unit with two hydrogen recombiners and condenser hotwell deaerator on aux. steam
5.	To Radwaste Evaporators (From Neutr. Tanks)	1/4.63 days	13,668	50	2,960	1/0.26 days	51,943	Maximum during back-to-back regeneration of one unit
6.	To Radwaste Evaporators (From Chem. Waste Tank)	1/10.3 days	11,850	20	1,153	1/0.54 days	22,200	Maximum corresponds to one liquid radwaste collection tank set
7.	To Distillate Sample Tank (From Evaporators)	2/3.1 days	5,700	30	3,702	1/0.12 days	47,786	Volumes are 90% of evaporator inlet
8.	To Condensate Storage Tanks (From Dist Sample Tank)	2/3.1 days	5,700	50	3,702	1/0.12 days	47,786	Maximum during back-to-back regeneration of one unit
9.	To Plant Discharge Pipe (From Distillate Sample Tank)	-	5,700	50	-	1/7.25 days	788	Excess distillate from regeneration chemical neutralization, lab drains, and aux-boiler blowdown. Back-to-back regen. of one unit for max
10.	To Laundry Drain Filters	1/0.63 days	820	25	1,300	1/0.14 days	6,000	Maximum during startup of

TABLE 11.2-4 (Continued)

Stream No.		Normal Operation of Both Units Averaged Number and Batch Frequency	Volume/ Batch (gal)	Nominal Flow Rate (gpm)	Normal Operation of Both Units Av. Volume/ day (gpd)	Maximum Number and Batch Frequency	Maximim Volume/ Day (gpd)	Comment
	and Sample Tanks (From Laundry Drain Coll. Tanks)							one unit
11.	To Plant Discharge Pipe (From Laundry Drain Sample Tanks)	1/1.1 days	1,420	10	1,300	1/0.22 days	6,000	
12.	To Cooling Tower Blowdown Line into River	-	-	200	2,000	-	288,000	Normal includes excess water from introduction of other than condensate water to plant inventory (laundry water, demin. water, chemicals, etc) Max. is permissible limit.
13.	To Evaporator Concentrate Storage Tanks or Waste Mixing Tanks (From Evaporator)	1/4.67 days	1,500	50	321	1/0.27 days	5,220	Volumes are 10% of evaporator inlet. Batch volume is evap. hold-up volume.

TABLE 11.2-5

EXPECTED RADIONUCLIDE ACTIVITY INVENTORIES OF LIQUID RADWASTE SYSTEM COMPONENTS (Curies Per Component)[2]

Isotope	Liquid Radwaste Collector Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizer Tank	Chemical Waste Collector Tank	Evaporator Distillate Sample Tank	Evaporator Concentrates Storage Tank	Laundry Drain Collector Tank	Laundry Drain Sample Tank	Liquid Radwaste Demineralizer
TRITIUM									
H-3	4.2-1	4.2-1	-	-	-	-	-	-	-
CORROSION PRODUCTS									
Na-24	8.5-2[1]	8.5-3	-	8.1-3	-	-	-	-	-
P-32	1.9-3	1.9-4	-	1.8-4	-	-	-	-	-
Cr-51	4.7-2	4.7-3	-	4.5-3	-	-	-	-	-
Mn-54	5.6-4	5.6-5	-	5.4-5	-	-	4.3-5	7.5-7	-
Mn-56	4.7-1	4.7-2	-	4.5-2	-	-	-	-	-
Fe-55	9.4-3	9.4-4	-	8.9-4	-	-	-	-	-
Fe-59	2.8-4	2.8-5	-	2.7-5	-	-	-	-	-
Co-58	1.9-3	1.9-4	-	1.8-4	-	-	1.7-5	2.9-6	-
Co-60	3.8-3	3.8-4	-	3.6-4	-	-	3.9-5	6.8-6	-
Ni-63	9.4-6	9.4-7	-	8.9-7	-	-	-	-	-
Ni-65	2.8-3	2.8-4	-	2.7-4	-	-	-	-	-
Cu-64	2.8-1	2.8-2	-	2.7-2	-	-	-	-	-
Zn-65	1.9-3	1.9-4	-	1.8-4	-	-	-	-	-
Zn-69	1.9-2	1.9-3	-	1.8-3	-	-	-	-	-
FISSION PRODUCTS									
Br-83	2.8-2	2.8-4	2.02-1	2.7-3	9.0-5	6.18-3	-	-	6.5-3
Br-84	4.7-2	4.7-4	7.38-2	4.5-3	3.2-5	-	-	-	3.2-10
Br-85	2.8-2	2.8-4	4.16-3	2.7-3	1.9-6	-	-	-	-
I-131	4.7-2	4.7-4	2.66+1	4.5-3	1.2-2	6.70+1	2.6-6	4.5-6	6.9+1

TABLE 11.2-5 (Continued)

Isotope	Liquid Radwaste Collector Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizer Tank	Chemical Waste Collector Tank	Evaporator Distillate Sample Tank	Evaporator Concentrates Storage Tank	Laundry Drain Collector Tank	Laundry Drain Sample Tank	Liquid Radwaste Demineralizer
I-132	2.8-1	2.8-3	1.89+0	2.7-2	8.4-4	4.58-2	-	-	9.9-2
I-133	1.9-1	1.9-3	1.13+1	1.8-2	5.0-3	1.77+1	-	-	1.9+1
I-134	4.7-1	4.7-3	1.21+0	4.5-2	5.4-4	1.33-5	-	-	1.9-5
I-135	1.9-1	1.9-3	3.72+0	1.8-2	1.7-3	1.98+0	-	-	2.1+0
Rb-89	4.7-2	2.35-2	1.78-4	4.5-3	8.0-8	-	-	-	-
Cs-134	2.8-4	1.4-4	1.59-2	2.7-5	7.2-7	2.26-2	5.6-5	9.7-5	1.4+0
Cs-136	1.9-4	9.5-5	1.92-3	1.8-5	8.6-8	4.88-3	-	-	3.9-1
Cs-137	6.6-4	3.3-4	2.04-2	6.3-5	9.2-7	5.40-2	1.0-4	1.7-4	3.3+0
Cs-138	9.4-2	4.7-2	1.66-4	8.9-3	7.4-8	-	-	-	7.6-10
Sr-89	9.4-4	9.4-6	9.32-2	8.9-5	4.2-6	2.44-1	-	-	4.0+0
Sr-90	5.6-5	5.6-7	7.84-3	5.4-6	3.4-7	2.08-2	-	-	3.1-1
Sr-91	3.8-2	3.8-4	5.38-2	3.6-3	2.4-6	4.70-2	-	-	9.9-1
Sr-92	9.4-2	9.4-4	3.76-2	8.9-3	1.7-6	1.90-3	-	-	4.0-2
Y-89m			9.32-6		4.2-10	2.44-5	-	-	4.0-4
Y-90			7.34-3		3.4-7	1.97-2	-	-	2.9-1
Y-91	3.8-4	3.8-6	3.84+0	3.6-5	1.7-4	1.01+1	-	-	2.8+0
Y-91m			3.16-2		1.4-6	3.04-2	-	-	6.4-1
Y-92	5.6-2	5.6-4	6.70-2	5.4-3	3.0-6	1.80-2	-	-	3.8-1
Y-93	3.8-2	3.8-4	5.72-2	3.6-3	2.6-6	5.36-2	-	-	1.1+0
Zr-93			4.06-9		-	1.09-8	-	-	1.6-7
Zr-95	6.6-5	6.6-7	6.88-3	6.3-6	3.0-7	1.82-2	6.1-6	1.1-5	2.9-1
Zr-97	4.7-5	4.7-7	1.18-4	4.5-6	5.4-9	1.67-4	-	-	3.5-3
Nb-95	6.6-5	6.6-7	8.46-3	6.3-6	3.8-7	2.24-2	8.6-6	1.5-5	3.5-1

TABLE 11.2-5 (Continued)

Isotope	Liquid Radwaste Collector Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizer Tank	Chemical Waste Collector Tank	Evaporator Distillate Sample Tank	Evaporator Concentrates Storage Tank	Laundry Drain Collector Tank	Laundry Drain Sample Tank	Liquid Radwaste Demineralizer
Nb-95m	-	-	1.28-4	-	5.8-9	3.42-4	-	-	5.3-3
Nb-97	-	-	1.18-4	-	5.4-9	1.80-4	-	-	3.7-3
Nb-97m	-	-	1.10-4	-	5.0-9	1.55-4	-	-	3.3-3
Nb-98	3.8-2	3.8-4	-	3.6-3	-	-	-	-	8.8-7
Mo-99	1.9-2	1.9-4	1.86-1	1.8-3	8.4-6	4.18-1	-	-	8.8+0
Tc-99	-	-	1.72-7	-	-	4.58-7	-	-	6.7-6
Tc-99m	1.9-1	1.9-3	3.30-1	1.8-2	1.5-5	4.58-1	-	-	9.9+0
Tc-101	8.5-1	8.5-3	2.92-2	8.1-2	1.3-6	-	-	-	-
Tc-104	7.5-1	7.5-3	3.34-2	7.1-2	1.5-6	-	-	-	-
Ru-103	1.9-4	1.9-6	1.66-2	1.8-5	7.4-7	4.36-2	6.1-7	1.1-6	7.3-1
Ru-105	1.9-2	1.9-4	1.24-2	1.8-3	5.4-7	2.92-3	-	-	6.1-2
Ru-106	2.8-5	2.8-7	3.72-2	2.7-6	1.7-7	9.88-3	1.0-5	1.7-5	1.5-1
Rh-103m	1.8-4	1.8-6	1.63-2	1.7-5	7.4-7	4.26-2	-	-	7.2-1
Rh-105	1.9-2	1.9-4	1.23-2	1.8-3	5.4-7	2.74-2	-	-	5.8-1
Rh-105m	-	-	2.96-3	-	1.3-7	7.02-4	-	-	1.5-2
Rh-106	2.8-5	2.8-7	3.72-3	2.7-6	1.7-7	5.88-3	-	-	1.5-1
Te-129	-	-	1.99-2	-	9.0-7	5.22-2	-	-	8.9-1
Te-129m	3.8-4	3.8-6	3.10-2	3.6-5	1.4-6	8.14-2	-	-	1.4+0
Te-131	-	-	7.50-4	-	3.4-8	1.41-4	-	-	-
Te-131m	9.4-4	9.4-6	4.16-3	8.9-5	1.9-7	7.74-3	-	-	-
Te-132	9.4-5	9.4-7	1.08-3	8.9-6	4.8-8	2.50-3	-	-	5.3-2
Ba-139	9.4-2	9.4-4	1.92-2	8.9-3	8.6-7	2.14-4	-	-	4.5-4
Ba-140	3.8-3	3.8-5	1.63-1	3.6-4	7.4-6	4.16-1	-	-	1.6+1
Ba-141	9.4-2	9.4-4	4.16-3	8.9-3	1.9-7	-	-	-	-

TABLE 11.2-5 (Continued)

Isotope	Liquid Radwaste Collector Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizer Tank	Chemical Waste Collector Tank	Evaporator Distillate Sample Tank	Evaporator Concentrates Storage Tank	Laundry Drain Collector Tank	Laundry Drain Sample Tank	Liquid Radwaste Demineralizer
Ba-142	5.6-2	5.6-4	1.53-3	5.4-3	6.8-8	-	-	-	-
La-140	3.7-3	3.7-5	1.61-1	3.5-5	7.2-6	4.28-1	-	-	1.7+1
La-141	9.4-2	9.4-4	4.16-3	8.9-3	1.9-7	7.48-4	-	-	1.6-2
La-142	4.7-2	4.7-4	1.22-2	4.5-3	5.4-7	3.12-5	-	-	-
Ce-141	2.8-4	2.8-6	2.56-2	2.7-5	1.2-6	6.74-2	-	-	1.2+0
Ce-143	2.8-4	2.8-6	1.38-3	2.7-5	6.2-8	2.64-3	-	-	5.6-2
Ce-144	2.8-5	2.8-7	3.66-3	2.7-6	1.6-7	9.72-3	-	-	1.5-1
Pr-143	3.8-4	3.8-6	1.84-2	3.6-5	8.2-7	4.74-2	-	-	9.2-1
Pr-144	2.8-5	2.8-7	3.66-3	2.7-6	1.6-7	9.72-3	-	-	1.5-1
Nd-144	-	-	-	-	-	-	-	-	-
Nd-147	2.8-5	2.8-7	1.07-3	2.7-6	4.8-8	2.74-3	-	-	5.5-2
Pm-147	-	-	3.24-5	-	-	8.74-5	-	-	1.1-3
W-187	2.8-3	2.8-5	-	2.7-4	-	-	-	-	-
Np-239	6.6-2	6.6-4	5.48-1	6.3-3	2.4-5	1.20+0	-	-	2.5+1
Pu-239	-	-	2.26-6	-	-	6.10-6	-	-	8.9-5

(1) Typical: 8.5-2 means 8.5×10^{-2}

(2) Values are curies per component filled to its live capacity.

TABLE 11.2-6

DESIGN BASIS RADIONUCLIDE ACTIVITY INVENTORIES OF LIQUID RADWASTE SYSTEM COMPONENTS (Curies Per Component)[2]

Isotope	Liquid Radwaste Collector Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizer Tank	Chemical Waste Collector Tank	Evaporator Distillate Sample Tank	Evaporator Concentrates Storage Tank	Laundry Drain Collector Tank	Laundry Drain Sample Tank	Liquid Radwaste Demineralizer
TRITIUM									
H-3	4.2-1	4.2-1	-	-	-	-	-	-	-
CORROSION PRODUCTS									
Na-24	1.9-2	1.9-3[1]	-	1.8-3	-	-	-	-	-
P-32	1.9-4	1.9-5	-	1.8-5	-	-	-	-	-
Cr-51	4.7-3	4.7-4	-	4.5-4	-	-	-	-	-
Mn-54	3.8-4	3.8-5	-	3.6-5	-	-	2.9-5	5.0-7	-
Mn-56	4.7-1	4.7-2	-	4.5-2	-	-	-	-	-
Fe-55	-	-	-	-	-	-	-	-	-
Fe-59	7.5-4	7.5-5	-	7.1-5	-	-	-	-	-
Co-58	4.7-2	4.7-3	-	4.5-3	-	-	4.3-4	7.3-5	-
Co-60	4.7-3	4.7-4	-	4.5-4	-	-	4.9-5	8.5-6	-
Ni-63	-	-	-	-	-	-	-	-	-
Ni-65	2.8-3	2.8-4	-	2.7-4	-	-	-	-	-
Cu-64	-	-	-	-	-	-	-	-	-
Zn-65	1.9-5	1.9-6	-	1.8-6	-	-	-	-	-
Zn-69	-	-	-	-	-	-	-	-	-
FISSION PRODUCTS									
Br-83	1.4-1	1.4-3	1.00+0	1.3-2	4.6-4	3.10-2	-	-	3.2-2
Br-84	2.5-1	2.5-3	3.98-1	2.4-2	1.8-4	-	-	-	1.8-9
Br-85	1.6-1	1.6-3	2.36-2	1.5-2	1.1-5	-	-	-	-
I-131	1.2-1	1.2-3	6.94+1	1.2-2	3.2-2	1.74+2	6.8-6	1.2-5	1.7+2

TABLE 11.2-7

RADIONUCLIDE ACTIVITY CONCENTRATIONS IN CONDENSATE DEMINERALIZER REGENERATION CHEMICAL WASTE (µCi/gm)

Isotope	Expected	Design Basis
Halogens		
Br-83	3.8-3*	1.9-2
Br-84	1.3-3	7.4-3
Br-85	7.8-5	4.4-4
I-131	5.0-1	1.3+0
I-132	3.6-2	1.4-1
I-133	2.2-1	9.4-1
I-134	2.2-2	1.1-1
I-135	7.0-2	4.6-1
Fission Products		
Rb-89	3.4-5	-
Cs-134	3.0-4	8.6-4
Cs-136	3.6-5	2.0-4
Cs-137	3.8-4	1.3-3
Cs-138	3.2-5	9.0-4
Sr-89	1.8-3	5.4-2
Sr-90	1.5-4	5.6-3
Sr-91	1.0-3	1.7-2
Sr-92	7.0-4	7.8-3
Y-89m	1.8-7	5.4-6
Y-90	1.4-4	5.4-3
Y-91	7.2-2	8.4-3
Y-91m	6.0-4	1.0-2
Y-92	1.3-3	7.8-3
Y-93	1.1-3	-
Zr-93	-	-
Zr-95	1.3-4	7.4-4
Zr-97	2.2-6	1.4-5
Nb-95	1.6-4	9.4-4
Nb-95m	2.4-6	1.4-5
Nb-97	2.2-6	1.4-5
Nb-97m	2.0-6	1.3-5
Mo-99	3.6-3	3.8-2
Tc-99	3.2-9	4.0-8
Tc-99m	6.2-3	7.8-2
Tc-101	5.4-4	8.6-4
Tc-104	6.4-4	-
Ru-103	3.2-4	3.0-4
Ru-105	2.2-4	-
Ru-106	7.0-5	6.0-5
Rh-103m	3.0-4	3.0-4
Rh-105	2.4-4	-
Rh-105m	5.6-5	-

TABLE 11.2-7 (Continued)

Isotope	Expected	Design Basis
Rh-106	7.0-5	6.0-5
Te-129	3.8-4	3.8-4
Te-129m	5.8-4	5.8-4
Te-131	1.4-5	-
Te-131m	7.8-5	-
Te-132	2.0-5	1.0-4
Ba-139	3.6-4	5.8-3
Ba-140	3.0-3	7.0-3
Ba-141	7.8-5	1.3-3
Ba-142	2.8-4	8.2-1
La-140	3.0-3	6.8-3
La-141	7.8-5	1.3-3
Ce-141	4.8-4	1.5-3
Ce-143	2.6-5	3.0-5
Ce-144	7.0-5	8.0-4
Pr-143	3.4-4	3.4-4
Pr-144	7.0-5	8.0-4
Nd-144	-	-
Nd-147	2.0-5	9.4-5
Pm-147	6.2-7	2.8-6
W-187	-	-
Np-239	1.0-2	3.6-1
Pu-239	4.2-8	1.5-6

* Typical: 1.9-3 means 1.9×10^{-3}

TABLE 11.2-8

ASSUMPTIONS AND PARAMETERS USED FOR EVALUATION OF RADIOACTIVE RELEASES

ITEM	VALUE OR REFERENCE	SOURCE
1. GENERAL		
a) Maximum core thermal power (MWT) evaluated for safety considerations	3440	FSAR 4.1
b) Total quantity of tritium released from one unit (Ci/yr)	89	NUREG 16
2. NUCLEAR STEAM SUPPLY SYSTEM		
a) Total steam flow (lb/hr) for valve wide open (VWO)	1.4+7(1)	FSAR 5.1
b) Mass of reactor coolant (lb) in vessel at full power	3.8+5	FSAR 5.1
3. REACTOR WATER CLEANUP SYSTEM		
a) Average flow rate (lb/hr)	1.33+5	FSAR 5.4.8
b) Powdex demineralizer size (lb of dry resin incl. 10 w/o crud)	30	FSAR 5.4.8
c) Replacement frequency (days)	3.4	FSAR 11.2
d) Backwash volume (gal/event)	1000	FSAR 11.2
4. CONDENSATE DEMINERALIZERS		
a) Average flow rate (lb/hr) total for 6 vessels (VWO)	1.4+7	FSAR 5.1
b) Deep bed demineralizer size (ft^3 of resin per vessel)	276	FSAR 10.4.6

(1) Typical: 1.4+7 means 1.4×10^7

(2) The system design regeneration frequency of 9.26 days is due to the use of a non-ultrasonic resin cleaning process prior to regeneration (see Section 10.4.6). Suspended solids from the resin cleaning are transferred to the solid radwaste system and do not contribute to the liquid radioactive release. The radioactive effluent from the regeneration (excluding corrosion products) is processed normally through the evaporators in the liquid radwaste chemical processing system.

(3) Spent resins from the radwaste demineralizer are sluiced to the solid radwaste system.

TABLE 11.2-8 (Continued)

ITEM			VALUE OR REFERENCE	SOURCE
	c)	Number of demineralizers	6 plus 1 out of service	FSAR 10.4.6
	d)	Regeneration frequency (days)(2)	9.26	FSAR 11.2
	e)	No ultrasonic resin cleaning(2)	N/A	-
	f)	Regenerant volume (gal/event)	14,000	FSAR 11.2
		Regenerant activity (μCi/gm)	Table 11.2-7	FSAR 11.2
5.	LIQUID WASTE PROCESSING SYSTEM			
	a)	1) Sources, flow rates, and expected activities in flow streams	Tables 11.2-4;11.2-7;11.2-10; 11.2-13	FSAR 11.2
		2) Holdup times for collection, processing, and discharge	Table 11.2-11	FSAR 11.2
		3) Capacities of tanks and processing equipment	Table 11.2-3	FSAR 11.2
		4) Decontamination factors	Table 11.2-12	NUREG 16, FSAR 11.2
		5) Fraction from each stream discharged		
		Liquid Radwaste Processing System	.01	NUREG 16, FSAR 11.2
		Liquid Radwaste Chemical Processing System	.05	NUREG 16, FSAR 11.2
		Liquid Radwaste Laundry Drain Proc. Sys.	1.0	NUREG 16, FSAR 11.2
		6) Radwaste demineralizer regeneration frequency (days)	None(3)	FSAR 11.2
		Radwaste demineralizer regeneration volume (gal/event)	None(3)	FSAR 11.2
		7) Liquid source terms for normal operation (Ci/yr)	Table 11.2-14	FSAR 11.2
	b)	P&IDs and process flow drawings for liquid radwaste systems	Figures 11.2-8 through 11.2-12	FSAR 11.2
6.	MAIN CONDENSER AND TURBINE GLAND SEAL AIR REMOVAL SYSTEMS			
	a)	Holdup time for offgas prior to offgas treatment system (hr)	0.15	FSAR 11.3
	b)	Description of offgas treatment system	FSAR 11.3	FSAR 11.3
	c)	Offgas treatment system		
		1) Mass of charcoal (lb)		FSAR 11.3
		2) Operating/dew point (°F)		FSAR 11.3
		3) Dynamic adsorption coeff. Xe,Kr(cm^3/g)		FSAR 11.3
	d)	Gland seal steam flow (lb/hr) and source	25,000 (normal) steam from condensate	FSAR 10.4.3
	e)	Radioactive iodine reduction systems for the		

TABLE 11.2-8 (Continued)

ITEM		VALUE OR REFERENCE	SOURCE
	gland seal system	N/A	N/A
	f) P&IDs and process flow drawings for gaseous waste systems	Figures 11.3-1 through 11.3-5	FSAR 11.3
7.	VENTILATION AND EXHAUST SYSTEMS		
	a) Provisions to reduce releases in individual buildings	Table 11.3-4	FSAR 9.4, 11.3
	b) Decontamination factors in individual buildings	Table 11.3-4	NUREG 16, BTP ETSB 11-2
	c) Release rates Ci/yr	Table 11.3-1 11.3-3	FSAR 11.3
	d) Release points - heights, temperatures, size, and shape of orifices	Figure 11.3-5	FSAR 11.3
	e) Containment purge frequency (per year)	4	FSAR 11.3
8.	EXPECTED RADIONUCLIDE ACTIVITY CONCENTRATIONS IN REACTOR COOLANT AND MAIN STEAM USED FOR EVALUATION OF RADIOACTIVE RELEASES	Table 11.2-9	NUREG 16, FSAR 11.2.3

TABLE 11.2-13

EXPECTED YEARLY ACTIVITY RELEASED FROM LIQUID RADWASTE MANAGEMENT SYSTEMS (Ci/yr)
USED FOR EVALUATION OF COMPLIANCE WITH APP. I OF 10CFR50

	LRW Processing System	LRW Chemical Processing System	Total LRW	Adjusted Total	Detergent Wastes	Total
Fission Products (Cont'd.)						
Zr-95	-	-	1.0-5	2.0-5	-	2.0-5
Nb-95	-	-	1.0-5	2.0-5	-	2.0-5
Nb-98	4.0-5	-	4.0-5	9.0-5	-	9.0-5
Mo-99	1.05-3	2.0-5	1.07-3	2.43-3	-	2.43-3
Tc-99m	4.55-3	2.0-5	4.57-3	1.04-2	-	1.04-2
Te-101	1.0-5	-	1.0-5	2.0-5	-	2.0-5
Ru-103	1.0-5	-	1.0-5	4.0-5	1.4-4	1.8-4
Rh-103m	1.0-5	-	1.0-5	4.0-5	-	4.0-5
TC-104	3.0-5	-	3.0-5	6.0-5	-	6.0-5
Ru-105	2.9-4	-	2.9-4	6.6-4	-	6.6-4
Rh-105m	2.9-4	-	2.9-4	6.6-4	-	6.6-4
Rh-105	9.0-5	-	9.0-5	2.1-4	-	2.1-4
Te-129m	2.0-5	1.0-5	3.0-5	7.0-5	-	7.0-5
Te-129	1.0-5	1.0-5	2.0-5	5.0-5	-	5.0-5
Te-131m	5.0-5	-	5.0-5	1.0-4	-	1.0-4
Te-131	1.0-5	-	1.0-5	2.0-5	-	2.0-5
I-131	2.82-3	5.554-2	5.835-2	1.325-1	6.0-5	1.325-1
Te-132	1.0-5	-	1.0-5	1.0-5	-	1.0-5
I-132	1.88-3	-	1.88-3	4.26-3	-	4.26-3
I-133	8.25-3	1.66-3	9.91-3	2.25-2	-	2.25-2
I-134	7.7-4	-	7-7-4	1.74-3	-	1.74-3
Cs-134	1.8-4	1.0-5	1.8-4	4.1-9	1.3-2	1.34-2
I-135	4.32-3	-	4.32-3	9.82-3	-	9.82-3
Cs-136	1.1-4	-	1.1-4	2.6-4	-	2.6-4
Cs-137	4.1-4	1.0-5	4.3-4	9.7-4	2.4-2	2.5-2
Ba-137m	3.8-4	1.0-5	4.0-4	9.0-4	-	9.0-4
Cs-138	3.0-4	-	3.0-4	6.8-4	-	6.8-4
Ba-139	2.7-4	-	2.7-4	6.2-4	-	6.2-4
Ba-140	2.3-4	4.0-5	2.7-4	6.1-4	-	6.1-4
La-140	4.0-5	4.0-5	8.0-5	1.9-4	-	1.9-4
La-141	1.0-4	-	1.0-4	2.4-4	-	2.4-4

TABLE 11.2-16

TANKS OUTSIDE REACTOR CONTAINMENT WHICH CONTAIN POTENTIALLY RADIOACTIVE LIQUIDS[1]

TANK	EQUIPT. NO.	LOCATION[2]	LEVEL INSTRUMENTS[3]	OVERFLOW PROVISIONS[4]
Liquid Radwaste Processing System:				
Collection Tk	OT-302A thru F	646-J-9	I, AHH (OC301)	Paired tanks O/F to LRW funnel
Surge Tk	OT-304A thru D	646-J-7	I, AHH (OC301)	Paired tanks O/F to LRW funnel
Sample Tk	OT-303A thru F	646-J-5	I, AHH (OC301)	Paired tanks O/F to LRW funnel
Filter Precoat Tk	OT-305	676-G-9	AH (OC307)	O/F routed to Waste Sludge Phase Separator
Filter Aid Tk	OT-310	676-G-8	AH (OC307)	O/F routed to Waste Sludge Phase Separator
Spent Resin Tk	OT-324	646-G-6	I, AH (OC323)	O/F to LRW funnel
LRW Chemical Processing System:				
Neutralizer Tk	1T-130A,B	656-M-12	R, AH (OC301)	Paired tanks O/F to CRW.
	2T-130A,B	656-M-45	R, AH (OC301)	Paired tanks O/F to CRW.
Collection (Chem Waste) Tk	OT-314	646-M-8	R, AH (OC301)	O/F to CRW funnel (or pump suction)
Evap. Dist. Sample Tk	OT-321	646-L-5	R, AH (OC301)	O/F to CRW funnel (or pump suction)
Evap. Conc. Storage Tk	OT-322	646-M-5	I, AH (OC301)	O/F to CRW funnel (or pump suction)
LRW Laundry Drain Processing:				
Collection Tk	OT-311 A,B	646-N-10	R, AH (OC301)	A tank overflows to B tank; B tank has O/F to CRW funnel (or pump suction).
Sample Tk	OT-312	646-L-9	R, AH (OC301)	O/F to CRW funnel
LRW Collection System:				
Chemical Drain Tk	OT-114	656-K-26	AHH (1C692)	O/F to LRW funnel
Laundry Drain Tk	OT-115	646-K-28	AHH (1C692)	O/F to LRW funnel

TANK	EQUIPT. NO.	LOCATION(2)	LEVEL INSTRUMENTS(3)	OVERFLOW PROVISIONS(4)
Solid Radwaste Collection System:				
RWCU Phase Separator	OT-318 A,B	646-G-4	I, AHH (C323)	O/F to LRW funnel (or pump suction).
Waste Sludge Phase Sep	OT-331	646-G-5	I, AHH (OC323)	O/F to LRW funnel (or pump suction).
Regeneration Waste Surge Tank	1T-106 A,B	656-N-15	AHH (1C121, OC323)	Combined O/F to LRW funnel.
	2T-106 A,B	656-N-43	AHH (2C121, OC323)	Combined O/F to LRW funnel.
RWCU Backwash Receiving Tank	1T-225	749-P-23	I, AHH (OC323)	O/F to LRW funnel.
	2T-225	749-P-35	I, AHH (OC323)	O/F to LRW funnel.
Fuel Pool Backwash Receiving Tank	OT-203	762-Q-28	I (OC323, OC307); AH (OC323)	O/F to LRW funnel.
Air Removal & Sealing Stm:				
Mech. Vacuum Pump Water Separator/Silencer	1T-107	656-J-27	I	O/F to LRW funnel.
	2T-107	656-J-32	I	O/F to LRW funnel.
Cnds. & Refuel Water Storage:				
Refuelding Water Storage Tk	OT-501	Outside	I (OCB517); R, AHL (OC653)	RFWST and Cnds Sto Tk "A" have a common dike and sump. Sump can be drained to TBLRW (or storm sewer) or a portable pump can be connected to installed flanged suction pipe. Tank has O/F to TBLRW.
Condensate Storage Tk A	OT-522A	Outside	I (OCB518A); R, AHL (OC653)	
Condensate Storage Tk B	OT-522B	Outside	I (OCB518B); R, AHL (OC653)	Dike and sump. Sump can be drained to TBLRW (or storm sewer) or a portable pump can be connected to installed flanged suction pipe. Tank has O/F to TBLRW funnel.

TANK	EQUIPT. NO.	LOCATION(2)	LEVEL INSTRUMENTS(3)	OVERFLOW PROVISIONS(4)
FPC & CU Skimmer Surge Tk	1T-208	779-R-28	I (1C206); AH (1C206, 0C211)	Skimmer Surge Tank is interconnected with Fuel Storage Pool via Skimmer Drain
	2T-208	779-R-30	I (2C206); AH (2C206, 0C211)	
Fuel Pool Filter/Demineralizer:				
Resin Tk	0T-202	779-Q-27	None	Overflows to Fuel Pool Backwash Receiving Tk
Precoat Tk	0T-201	779-P-27	None	Overflows to Fuel Pool Backwash Receiving Tk
Clean Up F/D Precoat Tk	1T-209	774-P-24	AH (1C040)	Batch-and high-level O/Fs to RWCU Backwash Receiving Tk (or Chem Waste Tk).

NOTES

(1) This table does not include water inventories in steam-cycle pressure vessels.

(2) Location of tank is indicated by elevation and nearest column-line.

(3) I = Indicator
AH = High level alarm
AHH = High-high level alarm
R = Recorder
AHL = High-low level alarm
G = Sight gage
Panel-board location of instrument is shown enclosed by parentheses.

(4) O/F = Overflow
LRW = Liquid Radwaste System
CRW = Chemical Radwaste System

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

11.3.1 DESIGN BASES

11.3.1.1 Design Objective

The gaseous waste management systems (GWMS) are designed to process and control the release of gaseous radioactive wastes to the site environs so that the total radiation exposure of persons in offsite areas is as low as reasonably achievable and does not exceed applicable guidelines. This is to be accomplished while maintaining the occupational exposure as low as reasonably achievable and without limiting plant operation or availability.

11.3.1.2 Design Basis

The gaseous waste systems are designed to limit offsite doses from routine station releases to significantly less than the limits specified in 10CFR20, and to operate within the dose objectives established in 10CFR50, Appendix I.

The design basis and maximum expected source terms correspond to 100,000 and 60,000 µ Ci/sec respectively of noble radiogas after a 30 minute delay. Table 11.3-1 lists the quantities of nuclides expected to be released to the environs when operating at the maximum expected failed fuel levels. The expected doses to individuals at or beyond the site boundary are shown in Subsection 11.3.4 and Environmental Report Subsection 5.2.4.2.

A description of the major equipment items in the offgas system is provided in Table 11.3-5. The seismic and quality group classifications of the GWMS components, piping and structures housing them are listed in Section 3.2.

Conservative analyses similar to those presented in Ref 11.3-1 demonstrate that equipment failure cannot result in doses exceeding acceptable guidelines; thus, neither the offgas system nor the buildings housing the equipment were designed to meet Seismic Category I requirements; however, the offgas structure walls are part of the total structural shear wall system and were analyzed to withstand the effects of earthquakes.

The failure of the Ambient Temperature Charcoal Offgas Treatment system is analyzed in Subsection 15.7.1.1. The related failure of the steam jet air ejector lines and failure of the main turbine gland sealing system are analyzed in Subsections 15.7.1.3 and 15.7.1.2 respectively.

11.3.2 SYSTEM DESCRIPTIONS

11.3.2.1 Offgas System

Noncondensible radioactive offgas is continuously removed from
5 the main condenser by the steam jet air ejector (SJAE) during
plant operation. This is the major source of gaseous releases
from the plant and is larger than all other sources combined.
5 The SJAE offgas will normally contain activation gases,
principally N-16, O-19, and N-13. The N-16 and O-19 have short
half-lives and are readily decayed. The N-13, with a 10-minute
half-life is present in small amounts that are further reduced by
5 delay. The SJAE offgas will also contain various isotopes of the
radioactive noble gases Xe and Kr, precursors of biologically
significant Sr-89, Sr-90, Ba-140, and Cs-137. 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 leaks. An offgas
5 system has been provided to treat these radioactive sources.
This system utilizes catalytic recombination and charcoal
adsorption as discussed below.

5 The building layout and equipment location of the offgas system
components is shown on Figures 11.2-3 through 11.2-7.

11.3.2.1.1 Process Flow Description

The noncondensible gases in the main turbine condenser are
removed by a two stage steam jet air ejector (SJAE) and
discharged to the offgas recombiner system. During startup,
clean auxiliary steam maybe used to drive the SJAE and the
18 recombiner system to minimize operation of and untreated
noncondensible releases from the mechanical vacuum pump. After
5 startup, pressure reduced steam from the main steam line is used.

Because of the limited motive steam capacity of the second stage SJAE, additional dilution steam to maintain the H_2 concentration below 4% by volume in the offgas stream, bypasses around the ejector nozzle to the discharge. This arrangement allows adjusting the total dilution steam flow without sacrificing SJAE performance. The offgas-steam flow then enters the associated or the common standby catalytic recombiner system through an electrically heat traced piping manifold. This prevents condensation of the dilution steam particularly during cold start-up. The purpose of the recombiner system is to reduce the offgas volume and eliminate the potential for explosion by controlled recombination of the radiolytic hydrogen with oxygen to less than 1% concentration by volume on a dry basis of 5 scfm air flow and less than 0.5% concentration for an air flow of at least 10 scfm.

The offgas first passes through the recombiner preheater in order
to minimize the moisture contents prior to entering the catalyst
bed. The recombination process takes place inside the recombiner
vessel which is electrically preheated during standby to a range 18
of 240 F to 270 F by strip heater on the outside. The
reaction temperature is approximately 800°F.

The moisture in the offgas leaving the recombiner vessel is
removed in the recombiner condenser where the offgas is cooled to
150°F. A motive steam jet then boosts the saturated gas stream 18
pressure from below to slightly above atmospheric pressure.

5

The reduced pressure main or auxiliary motive steam used in the 18
motive jet is then removed from the offgas stream in the motive
steam jet condenser and the 150°F offgas passes through a delay
pipe from the recombiner system in the turbine building to the
ambient temperature charcoal offgas system in the radwaste
building.

The pressure differential between the condensers in the
recombiner systems and the main condenser is sufficient to drain 18
the condensate without additional motive force to the main
condenser, while the delay pipe is drained by level controlled 18
valves to the turbine building radwaste sump.

The delay line varies in diameter from 8 to 16 in. and is
approximately 600 ft in length. At the design flow rate of 18
30 scfm, this pipe provides for approximately nine minutes of
decay of the radioactive products in the offgas stream prior to
entering the adsorption train.

After exiting this line, the gas mixture passes through a HEPA
prefilter to remove any radioactive particulate material that may
be entrained in the process stream. The gas is then cooled to
approximately 40°F by a refrigerated chiller unit and reheated to
approximately 65°F to prevent condensation. Moisture and 5
temperature instrumentation measure the process conditions
downstream of the chiller to monitor the performance of the water
removal assemblies and to guard against degraded charcoal
performance that might result from either an increase in the
moisture content or temperature of the gas.

Prior to entering the main charcoal vessels, the process stream passes through a sacrificial guard bed. The principal function of this guard bed is to absorb impurities that may be entrained in the process gas that might adversely affect the performance of the charcoal adsorbent. Each guard bed has been sized to absorb the moisture that might result from a failure of the chiller over a period of 48 hours. This design feature, in conjunction with the moisture and temperature instrumentation, should provide adequate protection against the contamination of the charcoal adsorber bed. Differential pressure indication is provided as a backup to the moisture instrumentation.

After passing through the guard bed, the gas enters the main
charcoal adsorption bed. This bed, operating in a controlled
temperature vault, selectively adsorbs and delays the xenon and
krypton from the bulk carrier gas. This delay on the charcoal
permits the xenon and krypton to decay in place. After
undergoing a sufficient decay in the charcoal vessels, the
18| process stream passes through a HEPA outlet filter, where
radioactive particulate matter and possible charcoal fines are
retained. This stream is continuously monitored and an alarm
5| will annunciate any abnormal releases from this system.

The process stream is then directed to the turbine building
5| ventilation exhaust duct where it is diluted with minimum
| 42,000 scfm of air prior to being released from the top of the
reactor building. Table 11.3-1 indicates the estimated annual
release rates from the turbine building of various isotopes.

11.3.2.1.1.1 Process Flow Diagram

Figure 11.3-1 is the process flow diagram for the system. The
process data for startup and normal operating conditions are
5| contained in Table 11.3-8.

11.3.2.1.1.2 Process and Instrumentation Diagram (P&ID)

18| The P&ID is shown as Figures 10.4-9, 11.3-3, A&B and 11.3-4.

11.3.2.1.1.3 Process Design Parameters

The krypton and xenon holdup times are closely approximated by
the following equation:

18|

$$T = \frac{K_D M}{V} \qquad \text{(Equation 11.3-1)}$$

Where:

T = holdup time of a given gas

18| K_D = dynamic adsorption coefficient for a given noble gas

M = mass of charcoal adsorber

V = flow rate of the carrier gas.

Conservative dynamic adsorption coefficients of 420 cc/gm for xenon and 23.7 cc/gm for krypton were assumed for the charcoal adsorbent material. They were derived by adjusting the values presented in NUREG 0016 for the temperature and humidity conditions of the Susquehanna SES offgas process stream.

Dynamic adsorption coefficients for xenon and krypton have been reported by Browning (Ref. 11.3-2). General Electric has performed pilot plant tests at their Vallecitos Laboratory, and the results were reported at the Twelfth AEC Air Cleaning Conference (Ref. 11.3-6). Further data on a similar system operating at ambient temperature are reported in Ref 11.3-3.

The temperature adjustment was obtained by a straight-line interpolation of the coefficients provided, in NUREG 0016 for the following data points: 77°F, dew point 0°F and 0°F, dew point -20°F. The moisture content of a gas mixture at these two points 18
is relatively low and thus the variations in adsorption coefficients between these points is mainly a function of temperature. The coefficients thus obtained were adjusted to reflect the effects of moisture content in a manner consistent with that employed in NUREG 0016.

With a design condenser air in-leakage of 30 scfm, and above adsorption coefficients this system provides a design delay of 32 hours for krypton and 23.7 days for xenon. Since the expected condenser in-leakage is below the design value, the actual delay times should be several times longer than the design delay times. Table 11.3-1 lists isotopic activities at the discharge of the turbine building exhaust vent.

After passing thru the recombiner section, the off gas stream consists primarily of the air in-leakage from the main condenser. The air in-leakage design basis is conservatively assumed at 18
30 scfm. The Sixth Edition of the Heat Exchange Institute Standards for Steam Surface Condensers (Ref 11.3-4, paragraph 5.16(c)(2)) indicates that with certain conditions of stable operation and suitable construction, noncondensibles should not exceed 6 scfm for large condensers. Dresden-2, Fukushima-1, Tsuruga, and KRB have all operated at 6 scfm or less.

5

11.3.2.2 Component Description

5 ### 11.3.2.2.1 Recombiner System

The offgas treatment system is divided into two sections to
5 facilitate plant arrangement: the recombiner system and the
charcoal offgas system. Three recombiner assemblies are located
18 in the turbine building in a shielded area below the main
condenser steam jet air ejectors. Each recombiner assembly
18 5 consists of the following major components: a recombiner
preheater, recombiner vessel, recombiner condenser, motive steam
jet, motive steam jet condenser and a condensate cooler.

One recombiner assembly is primarily designated for the service
5 of each nuclear unit and the third assembly is a common standby
to both units. Each recombiner assembly is sized to accommodate
5 the design flow from one nuclear unit. The piping and valve
18 manifold upstream of the recombiner assemblies permit the
transfer of the offgas stream between a unit designated assembly
5 and the common standby recombiner assembly.

The materials of construction, design pressures and temperatures, and the design codes for the components associated with the recombiner assemblies are listed in Table 11.3-5.

11.3.2.2.2 Charcoal Offgas System

After the radiolytic hydrogen and oxygen have been removed from the process stream by the recombiner assembly, the remaining gas enters a delay line which is approximately 600 ft in length and varies in size from 8 to 16 in. The purpose of this delay line is to permit the large quantity of N-16 to decay to a reasonable activity concentration prior to entering the charcoal adsorption portion of the offgas system. Although there is a separate delay line for each recombiner assembly, these lines are joined into a single common header in the radwaste building. However, the process offgas stream from each unit is segregated by the use of isolation valves that are installed in this common header.

After entering the common inlet header, the gas mixture from each
18 unit can be directed to either of two parallel equipment
subtrains consisting each of an inlet HEPA prefilter, a water
removal/temperature reduction assembly, and a charcoal guard bed.
The utilized charcoal adsorption train of each offgas treatment
18 5 system is primarily designated for the service of the associated
nuclear unit. Each adsorption train consists of five charcoal
adsorber beds in series. The trains and subtrains are isolable at
18 both the inlet and outlet by remotely operated valves. Following

subsections describe the various equipment associated with each 18
system.

11.3.2.2.3 Inlet HEPA Filter 5

As the carrier gas exits the delay line and enters an equipment 5
subtrain, particulates are removed by the HEPA filter with an
efficiency of at least 99.97 percent for particulates, sizes 0.3
microns and larger. Each filter will be individually tested
using the DOP method at 100 and 20 percent of the design flow
rate. DOP test connections have been provided for in line
testing of these filters at the time of initial installation or
subsequent replacement.

The filter element is housed in a carbon steel pressure vessel 18
designed to the code requirements of ASME Section VIII, Division
1. The materials of construction, the design pressure, and the
design temperature for these vessels is listed in Table 11.3-5.

11.3.2.2.4 Water Removal/Temperature Reduction Assembly

The offgas flow, after passing through the HEPA inlet filter, is
first processed through a precooler using reactor building closed
cooling water as the cooling medium. This heat exchanger is 5 18
designed to reduce the temperature of 150 scfm of offgas during
startup from 150 to 114°F and is built in accordance with TEMA
Standard Class C, Type BEU.

A chiller further reduces the offgas stream temperature to 40°F 18
at the design condenser air in-leakage rate. Refrigerant R-12
flows in the tube side and the offgas in the shell side. Water
cooled refrigeration condensing units are provided for each
chiller. This design eliminates the problems generally 18
associated with a system circulating chilled glycol, such as
leakages between the sides of the heat exchanger, leakages of
glycol solution from pump seals, etc. Also, the direct expansion
refrigeration approach eliminates the use of circulation pumps,
which should increase system reliability. The refrigeration
condensing units are located away from the precooler/chiller
assembly in a low radiation area. 5 18

The condensate from the precooler and chiller is collected in a 18
drain pot on the chiller. Since the accumulation rate of
condensate is expected to be very small, an on-off type level
control has been incorporated into the design. The condensate is 18
directed back to the main condenser. Malfunction of the level
control system may result in some of the offgas returning to the
main condenser, thereby preventing an uncontrolled release into
the radwaste building.

18 5 The offgas sides of the precooler and the chiller have been constructed of stainless steel to reduce the amount of corrosion products that might increase maintenance personnel doses or decrease system reliability.

11.3.2.2.5 Guard Beds

The offgas stream leaving the water removal assembly is reheated
18 5 to 60°F by electric heat tracing prior to entering the guard bed. The moisture content is then measured in order to monitor the performance of the water removal equipment. If the moisture content exceeds a preset level, an alarm is initiated in both the main control room and the local radwaste building control room.

The guard bed serves two basic functions: to provide delay time for the formation of long-lived particulate daughter products
18 from the short-lived gaseous Xenon isotopes in the offgas stream and to absorb any moisture that may be entrained in the stream.

18 The guard bed contains approximately 1280 lb of activated charcoal. The guard bed is sized to absorb moisture that could result from a failure of the chiller over a period of 48 hours.
18 5 An air drying/purge system has been provided to dry the guard bed should it become contaminated with water.

5 The moisture monitor at the discharge of each guard bed will indicate when the guard bed is approaching saturation and corrective measures can be taken prior to any contamination of the main charcoal adsorber bed.

18 The carbon steel guard bed vessel is designed to the code requirements of ASME Section VIII, Division 1. The materials of construction, the design pressure and temperature of these vessels are listed in Table 11.3-5.

11.3.2.2.6 Main Charcoal Adsorber Bed

5 Each adsorber train contains five tanks of activated charcoal
18 which are connected in series. These tanks is to provide sufficient delay of the radioactive noble gases Xenon and Krypton to permit releases to the environment that will satisfy the requirements of Appendix I to 10CFR50. The design delay times and the assumptions used to calculate these times are presented in Subsection 11.3.2.1.1.3.

18 The temperature of the charcoal is kept below 65°F, which is well below its ignition temperature, thus precluding overheating or fire and the consequential release of radioactive materials. It

has been determined that the heat generation rate during shutdown is not sufficient to present a charcoal ignition hazard.

The adsorbers are located in shielded vaults which are maintained at a temperature below 65°F by one of two 100% capacity air conditioning systems that remove the decay heat generated in the adsorbers, any heat introduced by the process stream and through the vault walls. The back-up air conditioning unit is activated automatically upon failure of the operating unit. Failure of the operating unit actuates a group alarm in 18
the main control room and at a local control panel. In the unlikely event that both air conditioning units are unable to function, the radioactive emissions from the offgas system would increase slightly; however, the releases to the environs would still be well below acceptable limits for the condenser air in-leakage normally expected.

Channeling in the charcoal adsorbers is prevented by maintaining a high bed-to-particle diameter ratio (approximately 750). Underhill (Ref 11.3-4) has stated that channeling or wall effects may reduce efficiency of the holdup bed if this ratio is not greater than 12. During installation of the charcoal, the adsorber vessels may be vibrated from the outside to minimize 5 | 18
voids and to increase the bulk density.

There are no provisions for bypassing the charcoal adsorbers during any mode of operation, except during the first stage of evacuation of the main condenser by the mechanical vacuum pump. 5

The ability of the charcoal to delay the noble gases can be evaluated by comparing activity measured and recorded by the process activity monitors downstream of the SJAE intercondenser 18
and at the exit of the outlet HEPA filters.

Each charcoal vessel is 8 ft in diameter and 26 ft in overall height. The five tanks that constitute the adsorber bed contain 76 tons of activated charcoal, which will provide in excess of 41 days delay for xenon under the design air in-leakage condition of 18
30 scfm. This assumes the use of realistic values for the adsorption coefficients for the charcoal as demonstrated by the offgas system supplier.

The carbon steel charcoal vessels are designed to the code 18
requirements of ASME Section VIII, Division 1. The materials of construction, the design pressure, and the design temperature of these vessels are listed in Table 11.3-5.

11.3.2.2.7 Outlet HEPA Filter

After the offgas stream exits the main charcoal bed, it passes 18
through a HEPA filter where any entrained particulates or

charcoal dust are collected. The removal efficiency of this HEPA
filter is 99.97 percent for particulate size 0.3 micron and
18 larger. Similar to the inlet HEPA filters described in
Subsection 11.3.2.2.3, these filters are individually tested
18 using the DOP method. The outlet HEPA filter is larger in size
than the inlet filter to accommodate the full design startup flow
rate of 300 scfm.

The offgas stream exiting the outlet HEPA filter is continuously
18 monitored for the level of radioactivity and activates an alarm
in both the main control room and the local radwaste control room
18 in the event of abnormal radioactive releases. The offgas stream
is then directed to the turbine building exhaust duct and
released through the exhaust vent on top of the reactor building.

11.3.2.2.8 Instrumentation and Control

5 The offgas system is monitored by means of flow, temperature,
pressure, and humidity instrumentation, and by hydrogen analyzers
5 to verify specified operation and control, and to ensure that the
hydrogen concentration is maintained below the flammable limit.
Figures 10.4-9 and 11.3-3 show the process parameters that are
monitored to alarm in the main control room and the local
radwaste control room, as well as whether the parameters are
recorded or just indicated.

5 A sample chamber with redundant radiation monitor placed
downstream of the SJAE intercondenser continuously records and
indicates gaseous radioactivity release from the reactor and
therefore provides information in the main control room on the
condition of the fuel cladding and the inlet activity to the
18 recombiner system and the charcoal adsorbers. Redundant high and
high-high alarms which are also displayed in the main control
room.

Experience with boiling water reactors has shown that the
calibration correction factor of the offgas radiation monitors
change with the isotopic content. The isotopic content can
change depending on the presence or absence of fuel cladding
leaks in the reactor, the nature of the leaks, and the holdup
time prior to release. Because of these variations, the monitors
are periodically calibrated against grab samples.

Grab samples can be retrieved downstream of the SJAE
intercondenser at the DOP test connections of the inlet and
outlet HEPA filters of the charcoal offgas system and at a
connection on the offgas pipe leading to the exhaust vent on the
reactor building. The combined second stage SJAE motive and
dilution steam flow is measured and recorded by redundant
instruments on the local recombiner control panel with low and

low-low flow annunciation. Indication and low-low annunciation 18
by a group alarm is provided on the main control room panel.

The temperature of the recombiner catalyst bed is monitored by
three RTD's with each output switchable to one indicator
thermocouples. An alarm is provided to annunciate temperature
conditions in excess of the process design value. The inlet
temperature to the recombiner is monitored by redundant RTD's and 5
low alarms annunciate when the temperature falls below the point
where adequate recombination of the radiolytic hydrogen and
oxygen would occur. Each recombiner assembly is heat traced
during standby. The common standby recombiner assembly is heat 18
traced and monitored to ensure its availability in case the unit
designated becomes inoperative.

The recombiner inlet and outlet temperatures are recorded and low
high and high-high alarms annunciate on the local panel while
indication and high-high and low temperature alarms are
annunciated on the main control room panel. Level controlled 5
valves are used in the drain lines from the recombiner and steam
jet condenser shells to the common condensate cooler which, in
turn drains to the main condenser. High condensate level alarms
are annunciated at the local control panel, with a group alarm on 5
the main control room panel.

The motive steam jet suction pressure is regulated by a butterfly
valve in order to keep the recombiner condenser pressure above
the main condenser pressure, thus allowing drainage of the
condensate without a motive device.

5

Two redundant thermal conductivity type hydrogen analyzers are
used to measure the hydrogen content of the offgas process stream
at the discharge of each recombiner assembly. The hydrogen
concentration from each analyzer is recorded and two high alarms
annunciate at the local control panel. High-high hydrogen
concentation alarms are provided both at the local and main
control room panels while indication is provided on the main 5 18
control room panel only. Each hydrogen analyzer can be
independently calibrated with the redundant one in operation.

The hydrogen analyzer systems continuously withdraw samples of
the offgas, analyze the hydrogen content, and return the sample 18
gas to the recombiner assembly. A hydrogen level of 2 percent
detected by either analyzer will alarm and annunciate both on the
local and by a group alarm on the main control room panel, while
4 percent detected by either analyzer will alarm and cause 5
closure of the SJAE offgas suction valves thus eliminating the 18
potential for build-up explosive misture in the offgas system.

Offgas system flow measurements are made downstream of the water
removal assemblies in the charcoal offgas treatment system with 5
indication and high flow alarm at a local and main control room 18
panel, recording on the local panel only.

The dilution steam flow will be adjusted to a minimum for 100%
18 5 power while maintaining the allowable in- and outlet hydrogen
concentrations, in order to keep the recombiner vessel outlet
temperature at max. 850°F.

5 11.3.2.2.9 Leakage of Gases from Offgas System

Leakage of radioactive gases from the offgas system is limited by
the use of welded construction wherever practicable and by using
18 valves of bellows stem seal design or employing double steam
5 packing with leak off connection being routed to the main
condenser or pressurized with instrument air.

The offgas system operates at a maximum of 5 psig during startup
and less than 2 psig during normal plant operations. The
18 5 differential pressure to the atmosphere is therefore small.

All drains from the various heat exchangers associated with the
recombiner and charcoal offgas system are directed back to the
main turbine condenser. Because of the low elevation the drains
from the delay line are routed through a drain pot with two level
5 control valves in series into the radioactive turbine building
sump. This minimized the potential for offgas escape into the
building in case of valve malfunctioning. Alarm and redundant
18 level control instrumentation is also provided.

11.3.2.3 Typical Operating Modes

11.3.2.3.1 Standby

During standby mode the recombiner system is isolated from the
18 offgas stream. The assembly steam supply and preheater bleed
steam supply valve as well as the condensate cooler drain valve
5 are open. For the common standby recombiner these valves are
aligned to and from either reactor unit. This in conjunction
with the electrical offgas inlet line heat tracing keeps the
system at approximately 300°F thus preventing condensation when
switching the offgas stream from an operating recombiner to the
18 standby one.

Depending on the air in leakage to the main condenser this
transfer is to be performed within approximately 10 minutes in
order to keep the condenser pressure below the allowable 5 inches
of mercury absolute.

Cooling water is normally maintained to the standby recombiner
18 assembly and the precooler and refrigeration condenser of the

charcoal offgas treatment system subtrains not in operation. The refrigeration system for the chiller is placed in the standby 18
mode and will start operation upon demand.

11.3.2.3.2 Prestart

In the prestart mode the motive steam jet steam supply valve and the recombiner system discharge valve are open in addition to the valves opened during standby. Motive steam may be from the auxiliary or nuclear boiler. The motive steam is condensed in the motive steam jet condenser while the recombiner system components are evacuated, ready for offgas admission.

The water removal precooler and chiller as well as the charcoal bed vaults of the ambient temperature charcoal offgas system must be at the required operating temperature and all valves in the normal operation status.

11.3.2.3.3 Normal Operation

Prior to placing the recombiner system from the prestart into the normal operation the following permissives must be present:

- o Recombiner inlet temperature not low 5
- o Recombiner outlet temperature not high-high
- o Recombiner condenser cooling water flow not low
- o Motive steam jet condenser cooling water flow not low
- o Recombiner system outlet hydrogen concentration not high-high

Except for the outlet hydrogen concentration permissive logic which requires only one out of two trips, each permissive is 18
incorporated into the controls of the recombiner system by a two out of two logics allowing opening of the offgas inlet valves to the first stage SJAE upon establishing steam flow through the second stage SJAE and the dilution steam bypass.

Closing of the first stage SJAE offgas inlet valves occurs when 18
any of the above permissive trips or two out of two trip signals of:

- o Dilution steam (2nd stage SJAE motive & bypass) flow low-low.

The recombiner system inlet valve closes upon a recombiner condenser outlet temperature high-high which in turn automatically opens a bypass valve recycling the SJAE discharge back to the main condenser. This allows switchover to the standby recombiner without interrupting the SJAE motive steam flow within the period determined by the rise of the main condenser pressure.

11.3.2.3.4 Startup

During start-up two inlet HEPA filters, water removal skids and
5 guard beds may be operated in parallel to accommodate the increased offgas flow of up to 300 scfm consisting mainly of air. During this limited time period the dew point of the gas at the chiller outlet may rise above the design temperature of 40°F without adverse effect on the charcoal adsorption coefficient.

11.3.2.3.5 Equipment Malfunction

Malfunction analysis, indicating the consequences and design precautions taken to accommodate failure of various components of the offgas system, is presented in Table 11.3-6.

11.3.2.4 Other Radioactive Gas Sources

There are three general areas that contain gaseous radioactive sources: the primary and secondary containment, the turbine building, and the radwaste building. The description of the ventilation systems for these buildings is presented in Section 9.4. The building volumes, flow rates, sources, and other information required to calculate the airborne concentrations of radioactive materials are contained in Subsections 12.2.2, 12.3.3, and 12.4.

11.3.2.4.1 Primary and Secondary Containment

Gaseous radioactive effluents can emanate from several sources. Leakage into the drywell and wetwell of the primary containment will be contained until containment atmosphere is purged in preparation for maintenance. This atmosphere will be processed through the charcoal filters of the SGTS prior to any release to the plant environs.

As indicated in Section 9.4, the two reactor buildings and the common refueling floor area have been designated as HVAC Zones I, II and III, respectively. Each of these zones has been divided into equipment compartment areas, where radioactive leakage may be expected, and other areas which contain non-radioactive equipment, accessways and the refueling floor. The exhaust air from the equipment compartment where radioactive leakage may occur is discharged through exhaust systems containing six inch deep charcoal filters. The air from the other areas is usually released unfiltered; however, if high concentrations do occur, the air can be recirculated and a small fraction discharged through the SGTS until the high concentration condition can be corrected. 5

In the Appendix I evaluation it was assumed that all radioactive releases from the reactor building are processed through the charcoal and HEPA filters before release to the atmosphere. There may be small quantities of radioactivity released unfiltered from the refueling floor area and the spent fuel pool, especially during the early stages of refueling. However, the quantities of iodine and particulates released from this source are expected to be much less than the releases from equipment leakage, equipment maintenance, drywell purge, and the vessel head lifting operation, all of which are filtered. Considering the uncertainties in the calculation of the reactor building releases and the conservative use of a 90% efficiency for the charcoal filtration systems, it is expected that the actual releases from the reactor building to the atmosphere should be lower than the estimates used in this evaluation.

The main steam relief valves are vented to the suppression pool. The activity released from the actuation of these relief valves will be contained in the primary containment until its atmosphere is purged through the SGTS in preparation for personnel access. Table 11.3-9 shows the frequency and quantity of steam discharged 5
to the suppression pool for abnormal occurrences.

11.3.2.4.2 Radwaste Building

Leakage into the radwaste building atmosphere will be processed through a prefilter and HEPA filter. In addition a charcoal filtration system will process the exhausts from the major radwaste system tanks. The valves associated with the adsorber portion of the offgas system have double packing to prevent leakage through the valve stem.

11.3.2.4.3 Turbine Building

As indicated in Subsection 9.4-4, the turbine building ventilation system contains a filtration system with HEPA filters and a six inch deep charcoal filter. Building air from those areas of the turbine building where equipment leakage and airborne activity could be expected is processed through the filtration system before it is released through the turbine building vent exhaust to the atmosphere. Air from non-contaminated areas is released through the turbine building vent exhaust without filtration.

The process valve stem leakage collection system is used for the
5 collection of any stem packing leakage from all steam valves in
the turbine building. Leakage from these valves is directed to
the main turbine condenser and processed through the offgas
system prior to being released to the environs.

In the past, the steam packing exhaust has presented a source of
gaseous radioactive releases in some BWR plants. However, at
this station an auxiliary source of clean steam is provided for
18 gland seal purposes from the steam seal evaporator. Therefore,
essentially no activity is released from this system. Subsection
10.4.3 provides a detailed description of the gland seal steam
system.

During the startup of each plant, air is removed from the main turbine condenser by a mechanical vacuum pump. This vacuum pump discharges to the turbine building ventilation exhaust system. A radiation detector continuously monitors the effluent from the turbine building exhaust system and an alarm is actuated upon the detection of a high radiation level.

11.3.3 RADIOACTIVE RELEASES

The activity released from the various vents will be monitored to ensure that the airborne concentrations at offsite locations will be below the limits of 10CFR20.106. In addition, the yearly releases will be kept as low as is reasonably achievable in order to meet the dose guidelines of Appendix I to 10CFR50.

An evaluation of the gaseous radioactive releases was done to
18 show compliance with the guidelines referenced above. The
assumptions used in this evaluation are summarized in Table 11.2-
8 , and are given in Tables 11.3-2 and 11.3-3 for gaseous
releases. The building vent locations, shape, effluent velocity,
and heat input are given on Figure 11.3-5. The calculated annual
releases are given in Table 11.3-1.

It is expected that the actual releases from the plant will be
18 lower than those referenced above due to the more realistic
parameters associated with the equipment described in this
Chapter. The charcoal filtration systems which reduce the
airborne radioactive releases are summarized in Table 11.3-4.

11.3.4 ESTIMATED DOSES

Dose calculations to assure compliance with Appendix I to 10 CFR Part 50 based on the gaseous source term referenced above were 18
performed in accordance with USNRC Regulatory Guide 1.109 by use of the USNRC computer code "GASPAR" (dated 2/20/76). One change was made to the code to allow for nuclide specific decay correction for transit time to the point of exposure. For this purpose maximum doses were calculated for an individual residing 18
at the point of highest dose. Input data for these calculations are given in Table 11.3-7. The critical location for noble gases is the south-west site boundary. The calculated doses were 8.97 mrad/yr beta air dose and 6.78 mrad/yr gamma air dose for noble gases. The Appendix I design guides are 20 and 10, respectively. Noble gas doses to the total body and skin were calculated as 4.38 mrem/yr and 9.63 mrem/yr. The respective limits for these pathways are 5.0 mrem/yr and 15.0 mrem/yr. For doses resulting from radioiodine and particulates, the critical location is the nearest dairy farm in the north-west sector. The Appendix I design guide for this pathway is 15 mrem/yr to any organ. The dose from this source was calculated as 5.47 mrem/yr. All calculated doses are within the appropriate Appendix I design guide. (See Section 5.3 of Environmental Report for further discussion of dose assessment and cost benefit analysis).

11.3.5 REFERENCES

11.3-1 NEDO-10734, "A General Justification for Classification of Effluent Treatment System Equipment as Group D" (February 1973).

11.3-2 ORNL-CF-59-6-47, "Removal of Fission Product Gases From Reactor Offgas Stream by Adsorption", W.E. Browning, R.E. Adams, and R.D. Ackley (June 11, 1959).

11.3-3 "Design of Fission Gas Holdup Systems", Proceedings of the 11th AEC Air Cleaning Conference, D. Underhill et al (1970).

11.3-4 "Standards for Steam Surface Condensers", Sixth Edition, Heat Exchange Institute, N.Y., N.Y. (1970).

11.3-5 NUREG 0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-Gale Code)", U.S. Nuclear Regulatory Commission (April 1976).

11.3-6 D. P. Siegwarth, "Measurement of Dynamic Adsorption Coefficients for Noble Gasses on Activated Carbon," 12th AEC Air Cleaning Conference.

TABLE 11.3-1

ANNUAL GASEOUS RELEASES PER UNIT

	Annual Building or Component Release (Ci/Yr)					
Nuclide	Reactor Building	Turbine Building	Radwaste Building	Steam Jet Air Ejector	Mechanical Vacuum Pump	Total Annual Release
H-3						8.0+1(1)
Kr-83m	0	0	0	0	0	0
Kr-85m	6.0	1.4+1	0	5.6+2	0	5.8+2
Kr-85	0	0	0	2.7+2	0	2.7+2
Kr-87	6.0	2.6+1	0	0	0	3.2+1
Kr-88	6.0	4.6+1	0	1.1+2	0	1.6+2
Kr-89	0	0	0	0	0	0
Xe-131m	0	0	0	5.3+1	0	5.3+1
Xe-133m	0	0	0	3.0+0	0	3.0+0
Xe-133	1.32+2	5.0+1	1.0+1	5.3+3	2.3+3	7.8+3
Xe-135m	9.2+1	1.3+2	0	0	0	2.2+2
Xe-135	6.8+1	1.3+2	4.5+1	0	3.5+2	5.9+2
Xe-137	0	0	0	0	0	0
Xe-138	1.4+1	2.9+2	0	0	0	3.0+2
I-131	3.4-2	3.8-3	5.0-2	0	3.0-2	1.2-1
I-133	1.36-1	1.5-2	1.8-1	0	0	3.3-1
Cr-51	6.0-6	2.6-5	9.0-5	0	0	1.2-4
Mn-54	6.0-5	1.2-6	3.0-4	0	0	3.6-4
Fe-59	8.0-6	1.0-6	1.5-4	0	0	1.6-4
Co-58	1.2-5	1.2-6	4.5-5	0	0	5.8-5
Co-60	2.0-4	4.0-6	9.0-4	0	0	1.1-3
Zn-65	4.0-5	4.0-7	1.5-5	0	0	5.5-5

TABLE 11.3-1 (Continued)

Annual Building or Component Release (Ci/Yr)

Nuclide	Reactor Building	Turbine Building	Radwaste Air Building	Steam Jet Air Ejector	Mechanical Vacuum Pump	Total Annual Release
Sr-89	1.8-6	1.2-5	4.5-6	0	0	1.8-5
Sr-90	1.0-7	4.0-8	3.0-6	0	0	3.1-6
Zr-95	8.0-6	2.0-7	5.0-7	0	0	8.7-6
Sb-124	4.0-6	6.0-7	5.0-7	0	0	5.1-6
Cs-134	8.0-5	6.0-7	4.5-5	0	3.0-6	1.3-4
Cs-136	6.0-6	1.0-7	4.5-6	0	2.0-6	1.3-5
Cs-137	1.1-4	1.2-6	9.0-5	0	1.0-5	2.1-4
Ba-140	8.0-6	2.2-5	1.0-6	0	1.1-5	4.2-5
Ce-141	2.0-6	1.2-6	2.6-5	0	0	2.9-5

(1) $8.0+1 = 8.0 \times 10^1$

TABLE 11.3-2

ASSUMPTIONS AND PARAMETERS USED FOR EVALUATION OF GASEOUS RELEASES

1.	Reactor coolant and main steam radionuclide concentrations	Table 11.2-8
2.	Radionuclide releases from buildings before treatment (from NUREG 16)	Table 11.3-3
3.	Radioiodine input into the main condenser offgas system	5 Ci/yr
4.	Main condenser vacuum pump release	Isotope / Ci/yr: Xe-133 2300 Xe-135 350 I-131 0.03
5.	Charcoal Delay System Parameters	
	a) Mass of charcoal, M, (lb)	152,000
	b) Operating/dew point temperatures (°F)	60-65/40
	c) Dynamic adsorption coefficient, K, (cm^3/g) for Xe/Kr	420/23.7
	d) Number of condenser shells, N	3
	e) Charcoal holdup time T(hrs)	.262 x M x K/(10 x N)
6.	Gland seal system flow rate lb/hr Activity level	25,000 Clean steam
7.	Offgas holdup time before the charcoal delay system (hr)	0.16
8.	Filtration systems on building ventilation	Table 11.3-4

TABLE 11.3-4

FILTER TRAINS USED TO CONTROL GASEOUS EFFLUENTS

Filter	Flow Rate (cfm)	Filter Components*	Charcoal Thickness (in.)	Iodine Removal Efficiency (%)
Containment/Aux Bldg				
Unit 1 - Area I, below refueling floor	16,000	P-H-C-H	6	99
Unit 2 - Area II, below refueling floor	16,000	P-H-C-H	6	99
Units 1&2 - Area III refueling floor	4,000	P-H-C-H	6	99
Drywell purge - through SGTS	10,500	M-P-H-C-H	8	99
Turbine Bldg	20,000	P-H-C-H	6	99
Radwaste Bldg	30,000	P-H	0	70
Tank exhaust filter	1,000	P-H-C-H	2	70

* M - Moisture separator
P - Prefilter
H - HEPA
C - Charcoal

TABLE 11.3-5

OFF GAS SYSTEM MAJOR EQUIPMENT DESCRIPTION
(Design Codes and Standards are provided in Table 3.2-1)

EQUIPMENT	EQUIPMENT NUMBERS	TYPE	QTY.	MATERIAL	CAPACITY	SIZE	DESIGN PRESSURE/ TEMP. PSIG/°F	
				RECOMBINER SYSTEM				
Preheater	OE/1E/2E-136	Shell and Straight tubes, BEM	3	Shell,Channel: CS Tubes, Sheet: SS	776,500 Btu/hr	21 0 sq. ft.	Shell side: 450/450 Tube side: 300/450	18
Recombiner Condenser	OE/1E/2E-137	Shell and U-tubes, BEU	3	Shell, Tubes Sheet: SS; Channel: CS	12.1×10^6 Btu/hr	970 sq. ft.	Shell side: 300/950 Tube side: 150/150	18
Motive Steam Jet Condenser	OE/1E/2E-134	Shell and U-tubes, BEU	3	Shell, Channel: CS; Tubes, Sheet: SS	7.9×10^6 Btu/hr	960 sq. ft.	Shell side: 300/300 Tube side: 150/150	18
Condensate Cooler	OE/1E/2E-152	Shell and U-tubes, BEU	3	Shell, channel: CS; Tubes, Sheet: SS	540×10^3 Btu/hr	138 sq. ft.	Shell side: 300/250 Tube side: 150/150	5 18
Recombiner Vessel	OS/1S/2S-125	Vertical Cyl.	3	SS	2700 Lbs. Catalyst	60" Dia.,74" High	300/950	
Catalyst Bed	----	Engelhard Deoxo A-18363	3	Ceramic based Precious Metal Pellets	Inlet:208,900 SCFH total, 3.89 Vol.% H_2 Outlet: 1 Vol % H_2 based on 5 SCFM Dry Air	60" Dia., 23" Deep	Outlet: 765°F	18
Motive Steam Jet	OE/1E/2E-133		3	SS	Suction: 300 SCFM Air 123 SCFM Steam Disch: 300 SCFM Air 1189 SCFM Steam	Suct: 6", Disch: 4", Motive 2 1/2"	300# Per ANSI	18

TABLE 11.3-5, Continued

AMBIENT TEMPERATURE CHARCOAL OFF GAS SYSTEM

EQUIPMENT	EQUIPMENT NUMBERS	TYPE	QTY.	MATERIAL	CAPACITY	SIZE	DESIGN PRESSURE/ TEMP. PSIG/°F
Delay Piping	GBC-106, 1,206		3	CS		8" / 16"	300/150
Inlet HEPA Filter	1F/2F-301A,B	Pleated Cylinder	4	Vessel: CS Cartridge: Glass Fiber	150 SCFM at 3.3 inch-H_2O	Vessel: 10"dia., 29" high Cartr: 6 1/2"dia., 8" high	450/200
Outlet HEPA Filter	1F/2F-302	Pleated Cylinder	2		300 SCFM at 2.7 inch H_2O	Vessel: 14"dia., 36" high Cartr: 11" dia., 8" high	310/150
Precooler	1E/2E-302A,B	Shell and U-tubes	4	Shell side: CS Tube Side: SS	86,930 Btu/hr	41.3 sq. ft.	Shell Side: 150/150 Tube Side: 695/200
Chiller Vessel	1E/2E-303A,B	Shell and Helical Coils	4	Shell Side: SS Tube Side: SS	43,000 Btu/hr	108 Sq. ft.	Shell side: 520/200 Tube Side: 150/125
Guard Bed Vessel	1T/2T-303 A,B	Vertical Cyl.	4	CS	0.64/tons of activated charcoal	30" dia. x 122" high	410/150
Charcoal Adsorber Vessel	1T/2T-306 thru 310	Vertical Cyl.	10	CS	15.35 tons	99" dia. x 254" high	375/150
Refrigeration Compressor; Condenser	1K/2K-321 A,B 1E/2E-301 A,B	Reciprocating Semi Hermetic	4	CI / Freon 12	43,500 Btu/hr.	3 HP	350/150
Charcoal Bed	----	Union Carbide MBQ 6 x 8	10	Mineral Base (Coal)	Adsorpt.coeff. for 40 SCFH Air } 7Xe:735 KR:52	On 8 Mesh, Max. 97%	----

5 18

18

TABLE 11.3-6 (Continued)

Equipment Item	Malfunction	Consequences	Design Precautions	
		biner condenser. Steam consumption would increase.	until the leakage was in excess of the capacity of the level control system at which point there would be drain pot high level alarms actuated.	
5. Recombiners	Catalyst gradually deactivates	Temperature profile changes through catalyst. Eventually excess hydrogen would be detected by hydrogen analyzer, recorded, and alarmed.	Temperature probes in the recombiner beds and exit of each recombiner. The hydrogen analyzer would indicate high hydrogen concentrations and provide a recording to compare to previous performance. If catalyst degraded significantly, a high hydrogen concentration alarm might be actutated.	18
	Catalyst gets wet	Hydrogen recombination falls off and higher hydrogen concentrations are detected downstream by hydrogen analyzers. Eventually the gas mixture could become combustible.	Temperature probes in recombiners and at outlet and hydrogen analyzers are provided, which annunciate locally and in the main control room. Automatic closure of the SJAE offgas inlet valves with further increase in hydrogen concentration.	5
6. Holdup piping between recombiner assembly and charcoal adsorber portion of offgas system	Excessive corrosion of line	Leakage of gaseous and liquid radioactive products into piping tunnel.	Area radiation monitors in turbine and radwaste buildings. The piping tunnel is provided with floor drains which would direct leakage to liquid radwaste system.	5
	Level valve fails to open	Accumulation of condensate in pipe	Redundant level control instrumentation with high level annunciator in the main control room. Manual bypass valve can be opened around one of the two redundant level valves.	5
	Level valve fails to close	Offgas leakage to turbine building sump.	Valves are spring loaded fail closed types. Two valves in series are provided. Area	18

TABLE 11.3-8

OFFGAS SYSTEM FLOWS (Refer to Fig. 11.3-1)
(PPH 1 SCFM)

Stream No.	Press. Psia	Temp. °F	Design Normal Operation (30 SCFM Air)			PPH/SCFM Total	Start-up (300 SCFM Air; Cond. Pressure 5" Hg)						
			H_2 % Vol. Conc.	PPH/SCFM H_2O	PPH/SCFM Non-Condens.		Press. Psia	Temp. of	H_2 % * Vol. Conc.	PPH/SCFM H_2O	PPH/SCFM Non-Concens.	PPH/SCFM Total	18
1. 2nd Stage SJAE Inlet	5.0	141	7.9	757/1480	526/234	1283/1714							
2. 2nd Stage SJAE Outlet	15.7	272	3.9	9350/3278	526/234	9876/3512	15.7	270		9350/3278	1395/ 300	10745/3578	
3. Recomb. Inlet	14.2	273	3.9	9350/3278	526/234	9876/3512	13.8	270		9350/3278	1395/ 300	10745/3578	
4. Recomb. Outlet	10.6	760	0.001	9737/3414	140/ 30	9877/3444	9.9	720		9350/3278	1395/ 300	10745/3578	
5. Recomb. Cond. Outlet	10.4	140	0.12	30/10.5	140/ 30	170/ 40.5	9.7	140		352/ 123	1395/ 300	1747/ 423	5 18
6. Motive Steam Jet Outlet	15.2	274	0.005	3070/1077	140/ 30	3210/1107	18.6	239		3392/1189	1395/ 300	4787/1489	
7. Motive Jet Cond. Outlet	15.2	140	0.14	18/6	140/ 30	158/ 36	18.5	140		158/ 55	1395/ 300	1553/ 355	
8. Precooler Inlet	15.1	140	0.14	18/6	140/ 30	158/ 36	17.7	140		79/ 28	698/ 150	777/ 178	
9. Chiller Outlet	15.0	40	0.17	1/0.2	140/ 30	141/ 30	16.7	72		8.1/3.6	698/ 150	706/ 154	
10. Guard Bed Inlet	15.0	65	0.17	1/0.2	140/ 30	141/ 30	16.7	72		8.1/3.6	698/ 150	706/ 154	

* During cold start-up a negligible amount of hydrogen is introduced into the main condenser.

TABLE 11.3-9

FREQUENCY AND QUANTITY OF STEAM DISCHARGED TO SUPPRESSION POOL

Event	Frequency Category	Quantity of Steam lbs/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	4,000
5. Inadvertent SRV Opening	Moderate	4,000
6. Trip of Both Rcirc. Pumps	Moderate	16,000
7. Turbine Trip	Moderate	11,000
8. Generator Load Rejection	Moderate	11,000
9. Pressure Regulator Failure - Open	Moderate	256,000*
10. Pressure Regulation Downscale Failure	Moderate	42,000
11. Recirc. Flow Control Failure - Decreasing	Moderate	16,000
12. Inadvertent Closure - All MSIV	Moderate	290,000*
13. Loss of Condenser Vacuum	Moderate	289,000*
14. Feedwater Control Failure - Max. Demand	Moderate	25,000
15. Loss of Auxiliary Transformer	Moderate	254,000*
16. Loss of All Grid Connections	Moderate	265,000*
17. Turbine Trip w/o Bypass	Infrequent	300,000*
18. Generator Load Rejection w/o Bypass	Infrequent	300,000*
19. Stuck Open SRV	Moderate	800,000*

NOTES: Events 1, 2 and 3 based on steam flow quantity during test mode per RCIC Syst. Process Diag.

Events 4 and 5 assuming test and inadvertent opening at 150-600 psi reactor pressure for 30 seconds.

Events 6 through 18 based on event description from Grand Gulf FSAR Chapter 15.

Event 19 based on results from 251 Standard Plant Suppression Pool Response Analyses.

* Isolation event. It is assumed that S/RV cycling is terminated at T=30 minutes and reactor depressurized by both RHR heat exchangers in steam condensing mode.

11.4 SOLID WASTE MANAGEMENT SYSTEM

11.4.1 DESIGN BASES

The objective of the Solid Waste Management System (SWMS) is to control, collect, handle, process, package, and temporarily store prior to offsite shipping the wet waste sludges and the evaporator concentrate generated by the Liquid Radwaste System, the Reactor Water Cleanup System, Fuel Pool Cleanup System, and the Condensate Cleanup System. Contaminated solids such as HEPA and cartridge filters, rags, paper, clothing, tools, and equipment are also disposed of in the SWMS.

The system is capable of receiving, processing, solidifying, or compacting the solid radioactive waste inputs as shown in Tables 11.4-1 and 11.4-2 for permanent offsite disposal. There is no liquid plant discharge from the SWMS. The SWMS is designed to minimize the volume of solidified or compacted waste for offsite shipment and burial.

Storage space for approximately one month's volume of solidified wet waste is provided in the radwaste building. Most dry solid waste is expected to be sufficiently low in activity to permit temporary storage in unshielded, cordoned-off areas adjacent to the waste compactor.

The SWMS is designed to package radioactive solid wastes for offsite shipment and burial in accordance with the requirements of applicable NRC and DOT regulations including 10CFR71 and 49CFR170 through 178. This results in radiation exposures to individuals and the general population within the limits of 10CFR20 and 50.

The SWMS does not have any safety related functions. Redundant and backup equipment, alternate routes, and interconnections are designed into the system to provide for operational occurrences such as refueling, abnormal leak rates, decontamination activities, SWMS equipment down time, maintenance and repair. Table 11.4-3 shows the design parameters of the SWMS equipment.

Equipment locations, room designs, drainage, ventilation, and design features of components are consistent with those shown in Section 11.2, to reduce maintenance, equipment down time, leakage, gaseous releases of radioactive materials to the building atmosphere, or to otherwise improve the system operations.

Remote controls and viewing systems are used to keep exposure to the personnel as low as reasonably achievable.

Section 15.1 analyzes the radwaste tank rupture accident applicable to both the liquid and solid radwaste management systems.

The seismic and quality group classifications of the SWMS components and piping and the structures housing it are listed in Section 3.2.

The expected and design inventories of individual radionuclides in components containing significant amounts of radioactive material are shown in Tables 11.4-5, 11.4-6 and 11.4-7.

11.4.2 SYSTEM DESCRIPTION

11.4.2.1 General

The SWMS consists of two subsystems: one for the collection, processing, and solidification of wet solids such as filter material slurries and evaporator concentrates, and the other for collecting and packaging of dry solids such as contaminated filter media, clothing, equipment, tools, paper, and plastic sheeting.

Except for condensate demineralizer regeneration waste surge tanks in the turbine building and the RWCU and fuel pool backwash receiving tanks in the reactor buildings, all SWMS equipment serves both reactor units and is located in the radwaste building.

11.4.2.2 Wet Solid Waste Subsystem

The wet solid waste subsystem shown on Figures 11.4-1 and 11.4-2 collects suspended solids slurries from six sources and a concentrated chemical solution from the radwaste evaporators described in Subsection 11.2.2.3. These wastes are solidified to a freestanding product. As shown on the flow diagram (Figure 11.4-3 and Table 11.4-4)) some of the waste inputs are collectively processed due to their expected similar activities. The solidification process with cement and catalyst is identical for all wet solid wastes.

Spent Resins

The spent ion exchange resin deep bed from the radwaste demineralizer is periodically sluiced to the spent resin tank for radioactive decay and storage until a radwaste filter is available for dewatering.

Sufficient capacity is provided in the spent resin tank for several batches of radwaste demineralizer resins and an infrequent batch of spent resins from the condensate demineralizer system of either reactor unit.

The vent and overflow nozzles of the spent resin tank are equipped with 30 mesh screens to minimize spread of particulate contamination to the radwaste tank vent system. A spray nozzle with spherical pattern located in the tank center allows remote internal washdown. A manhole and external ladder provide access to the tank interior.

Associated with the spent resin tank is the spent resin transfer pump, which is of the progressing cabity (Moyno) type. This pump is used to mix and transfer the spent resin tank contents. Normally liquid radwaste collection tank water is added to the spent resin tank to dilute the spent resin to a pumpable slurry, with condensate also being available as a backup water source. The spent resin transfer pump is used to mix the tank contents by recirculating tank fluid through internal tank mixing eductors located near the bottom of the tank. After the resins are in suspension, a portion of the spent resin transfer pump discharge is directed to either one of the two radwaste filters for removal and dewatering of the resins from the slurry at a flow rate of 50 to 100 gpm. The spent resin transfer pump is sized to provide continuous recirculation of the tank contents during tank pumpout to keep the resin in the tank in suspension.

The spent resin transfer pump and associated valves are separated from the spent resin tank by a shield wall to permit maintenance access.

The amount of spent bead resins being dewatered at one time is expected to be limited by the space between the filter screen plates rather than by the differential pressure across the filter screens. A demineralizer resin bed must therefore be dewatered in several batches.

The cake drying and discharging cycle of the radwaste filter is identical to the one described in Subsection 11.2.2.2 for the liquid radwaste filtering mode.

Regeneration Waste and Fuel Pool Filter Demineralizer Backwash Slurry

A twin set of interconnected conical bottom tanks is close to each condensate demineralizer resin cleaning system in the turbine building. These tanks provide surge capacity for the fluctuating waste flow during the resin scrub cycles. The corrosion products containing waste stream is continuously recirculated through tank internal mixing eductors and a 35 gpm partial flow is discharged to the waste sludge phase separator by one of two redundant in-line pumps. This inlet flow rate to the waste sludge phase separator allows continuous settling of the suspended solids, while the maximum 200 ppm solids-containing supernatant overflows into an internal standpipe and is transferred by the waste sludge phase separator decant pump to the liquid radwaste collection tanks.

Exhausted powdered ion exchange resins from the fuel pool filter demineralizers of both reactor units are backwashed into a common receiving tank in the reactor building of Unit 1. Two batches of exhausted fuel pool filter demineralizer resins can be collected in the backwash receiving tank. Compressed air at a flow rate of approximately 75 scfm is then injected through a diffusor at the tank bottom for approximately 30 minutes to agitate the slurry before and while the tank is gravity drained to the waste sludge phase separator in the radwaste building.

The waste sludge phase separator also receives drainage from the radwaste filter vessels prior to and following the filter cake drying and discharging cycle.

A mode selector switch allows isolation of the waste sludge phase separator for extended settling periods. This mode is used with resin fines carry-over in the condensate resin scrubbing waste inflow or when receiving fuel pool filter demineralizer backwash slurry. Interlocks are provided to prevent fuel pool filter demineralizer backwash inlet to the phase separator, when in the continuous decant mode to minimize the slow settling powdered ion exchange resin from entering the standpipe.

The elevation of the decant nozzle on the phase separator allows collection of approximately 500 gal of sludge on the slanted bottom.

Before sludge is transferred to the dewatering filter by the waste sludge discharge mixing pump, it is diluted in a phase separator volume of water to a pumpable concentration. Internal mixing eductors are driven by recirculated tank fluid.

Automatic flushing of slurry carrying lines and mechanical seals of slurry pumps is provided.

Because of the pressure drop and volume limitations of the radwaste filters several dewatering batches may be necessary for one phase separator sludge load.

The cake drying and discharging cycle of the filter in the dewatering mode is identical to the one described in Subsection 11.2.2.2 for the liquid radwaste filtering mode.

Reactor Water Cleanup Filter Demineralizer Backwash Slurry

A backwash receiving tank is close to the reactor water cleanup filter demineralizer system of each reactor unit in the reactor buildings. Two batches of exhausted powdered ion exchange resins from a RWCU filter demineralizer can be collected in each tank.

Compressed air at a flow rate of approximately 75 scfm is then injected through a diffuser at the tank bottoms for approximately 30 minutes to agitate the slurry before and while the tanks are gravity drained to one of two reactor water cleanup phase separators in the radwaste building.

The sludge holding capacity of one RWCU phase separator allows collection of two months' backwash sludges from both RWCU demineralizer systems at normal frequency.

During this period, sufficient settling time for the suspended solids in each backwash slurry batch is allowed before the maximum 200 ppm solids-containing supernatant is transferred by the reactor water cleanup decant pump to the radwaste collection tanks.

Alternating at two months intervals, each RWCU phase separator is first in the sludge collecting and then in the isolated mode to allow radioactive decay of isotopes with short half-lives.

This provision reduces operator exposures in subsequent processing steps and facilitates handling and shielding for offsite disposal. The sludge holding capacity on the slanted bottom of each RWCU phase separator is 750 gal. Additional decant nozzles are provided for adjustment of the phase separation height.

Before the sludge is transferred to the dewatering filter by the reactor water cleanup sludge discharge mixing pump, it is diluted in a phase separator volume of water to a pumpable concentration.

Internal mixing eductors are driven by recirculated tank fluid.

Automatic flushing of slurry carrying lines and mechanical seals of slurry pumps is provided.

Wet Radwaste Solidification and Packaging

Dried filter cakes discharged by the radwaste filters and chemical concentrates discharged by the radwaste evaporators are immobilized in portland cement with sodium silicate as a catalyst to induce immediate jelling and to prevent formation of free water on the surface in the containers.

The mixture is packaged in disposable radwaste containers for offsite burial.

A waste mixing tank beneath each radwaste filter receives, by gravity flow, the dried waste discharged from the filtering and dewatering process.

A maximum 1000 lb of approximately 50 percent by weight solids containing filter cake is discharged from a radwaste filter into the waste mixing tank. While operating the mechanical agitator in the tank, the remainder of the mixing tank volume is then filled with evaporator concentrate of minimum 130°F to produce a pumpable slurry. This mixture, or evaporator concentrate only, which is transferred from the concentrate storage tank to one of the heat traced waste mixing tanks, is now ready for solidification.

The progressing cavity (Moyno) process feed pump beneath each waste mixing tank transfers the slurry to the throat of a progressing cavity (Moyno) mixing pump. Dry portland cement is also metered from a storage silo at an adjustable rate into the mixing pump throat. The mixing pump kneads the waste slurry and portland cement and extrudes it through a fill pipe with pneumatically retractable splash guard into a waste container.

An adjustable amount of sodium silicate is metered by a progressing cavity (Moyno) pump into the waste-cement mixture at the fill pipe outlet. This shortens the initial setup period and improves the shipping efficiency (ratio of waste volume to solidified product volume). Flushing water mixed with sodium silicate may be routed into the waste containers. A drip funnel swings under the fill pipe and drains residual flush water to a settling container and sump. Two redundant process trains are provided in three separately shielded rooms. Each train consists of a mixing tank with conical bottom, agitator, internal decontamination spray nozzles, heat tracing, sonic level detector, temperature sensor and remote manual emergency drain valve, the associated process feed pump, mixing pump with remote manual cranking capability and internal decontamination spray nozzles, cement screw conveyor, associated piping, valves and instrumentation. Process feed pump discharge branch lines permit transfer of wastes between the two mixing tanks. The sealed cement silo with pneumatic conveying and dust control system, sonic level detector, diverting valve, adjustable speed rotary valve, the one sodium silicate tank and pump, and the container fill pipe with sonic level detector and associated piping, valves, and instrumentation are located in normally accessible areas or are of low maintenance design, therefore serving both solidification equipment trains. The cement silo and sodium silicate tank are filled by a bulk truck from the hose connection outside the cement silo building. Depending upon the experienced specific activity, the waste-cement mixture is filled into either a nominal 200 ft^3 or a 50 ft^3 disposable steel container or other suitable containers. The complete solidification of waste is maintained by testing similar waste process fluid and solidification agents prior to the start of actual solidification. From this the process parameters are formed for each type of process fluids which determines the operational boundary conditions. Operation of the solidification system adheres to these boundary conditions which are recorded for individual batches, thereby assuring complete solidification. If prior to the drum capping operation, free surface water should be found in a waste container, it can be readily absorbed by the addition of a small quantity of Portland cement to the waste container before capping. This can be accomplished by use of a small paper bag of Portland cement being added to the container at the swipe/pickup station by use of the swipe test tool. This tool can be used to pick up and drop in the container a preloaded bag of cement. These containers are provided with a fill neck, lid with compressive gasket and clamp ring identical to that of a standard 55 gal drum per DOT 17H Specification. The 50 ft^3 cylindrical container further qualifies as Type A per DOT regulations. A remotely controlled capping machine adjacent to the fill station secures a watertight lid on the container.

This is followed by a decontamination spray station with roll-up entrance and exit doors to minimize spread of contamination. A long handled pair of tongs protruding through the shield wall in the container swipe/pickup station allows swipe sampling of the container top and two sides.

The waste containers are transferred from the loading to the fill, capping, washdown and swipe/pickup station on an endless cable propelled rail dolly with container adapters as necessary to adjust container height to match the fixed height of the fillport and capper. An idler cable allows manual retrieval of the cart, should the motorized winch or endless cable fail. Three pushbutton stations are provided for the cart control with position indicating lights on the control panel.

The solidification system is step-controlled from a panel with mimic display located in a shielded area adjacent to the container transfer aisle. The filling and capping operation can be observed through lead glass shielding windows, while loading of the containers onto and picking them up from the cart is observed on a closed circuit TV monitor with cameras attached to the radwaste building crane. Additional viewing mirrors and TV cameras are installed as necessary to observe the manual swipe sampling. A branch line from the phase separator slude discharge pumps, a return line to the liquid radwaste collection tanks, and a vent line are provided to the radwaste building truck loading area for hookup of a truck mounted sludge dewatering device or solidification system.

The formulation ranges of solidification agents required to achieve dry solidification of the various waste materials will be described in the UNI topical report. The formulation established during shop testing of this system utilizing non-radioactive wastes is given in Table 11.4-8.

Table 11.4-8 is provided as a guide in determining the proper waste mixtures for processing. However, the data will be adjusted during system operation to reflect the actual batch sizes of the filter cake, its moisture content, composition, and the evaporator concentrate waste material concentration.

Radwaste Building Crane

A remotely controlled bridge crane is provided in the solid waste handling area of the radwaste building. It is used for loading of empty waste containers onto the rail dolly, transferring filled containers from the rail dolly into temporary storage compartments and onto a shielded truck for offsite disposal and for disassembling the radwaste filters.

The lifting deck on the crane is suspended by four independent cables reeved by two cable drums. This minimizes swaying during lifting and unbalancing of the load should one cable fail. Four pneumatically operated twist lock pins on the lifting deck engage four compatible receptacles with lead-ins on the waste container tops and storage compartment lids. This allows crane operation from any one of three stationary control pendants at a time or from the main control console, located in normally accessible areas. The crane bridge and trolley actuate limit switches along their runways, indicating by lights on the control console when the lifting deck center reaches the container loading, pickup, storage compartment, and truck loading positions.

The crane bridge and trolley travels are interlocked with the lifting deck elevation to prevent interference with shield walls and the container loading and pickup shaft walls.

Closed circuit TV cameras on the crane trolley and the lifting deck transmit images of the relative crane position to a portable monitor in the console to assist in remote operation of the crane when handling filled waste containers or compartment lids. Target marks on the container and compartment lids facilitate positioning of the lifting deck. Proper engagement of the twist lock pins in the receptacles is indicated by a light on the control console and each pendant. A slack cable indication light is provided on the control console only.

The trolley and bridge speeds are controllable in 10 increments from 5 to 50 ft per minute while the hoist speed is controllable in 10 increments from 1.5 to 15 ft per minute. The motor generator set for speed control allows dynamic braking of the hoist. Two additional, independent solenoid operated holding brakes are provided.

A standard 25 ton crane hook can be mounted on the lifting deck for general use of the radwaste building crane.

Waste Container Storage and Offsite Disposal

Filled waste containers are separately stored in covered concrete compartments for radioactive decay prior to offsite disposal.

The number of compartments allows storage of more than one month's anticipated solidified waste volume for normal operation of both reactor units considering refueling. The storage capacity consist of four (4) shielded compartments for the 50 ft^3 drums and twelve (12) shielded compartments for the 200 ft^3 drums. Each compartment will contain one drum. Shielding of the storage compartments reduces the radiation in the adjacent crane control area to less than 2.5 mR/hr. A monitoring room is provided for final swipe sampling and radiation surveying of the filled waste containers prior to loading them onto a truck. Sufficient lifting height is provided to place a large waste container into a top entry shield cask on a truck.

11.4.2.3 Dry Solid Waste Subsystem

The dry solid waste subsystem collects contaminated filter media, clothing, equipment, tools, paper, plastic sheeting, and miscellaneous radioactive wastes that are not amenable to solidification prior to packaging.

Depending upon the activity level, the physical size, and the material, three handling and packaging procedures for dry solid wastes are used. Except for irradiated reactor internals and the HEPA filter elements at the in and outlet to the ambient temperature charcoal offgas treatment system, the dry solid waste is expected to allow temporarily unshielded handling without exceeding the dose limits of 10CFR20. Normally the dry solid wastes are disposed of in licensed burial grounds after radioactive decay at the plant.

Irradiated Reactor Internals

Irradiated reactor internals being replaced are removed from the RPV underwater and stored for radioactive decay in the fuel pool. Subsection 9.1.4.2.2 describes reactor vessel and in-vessel servicing equipment used for handling reactor components.

An estimated average of 7 percent (14) of the control rod blades are removed from each reactor annually and are stored on hangers on the fuel pool walls or in racks interspersed with the spent fuel racks. Offsite shipping is done in spent fuel shipping casks.

Approximately 50 percent (22) of the power range monitor (PRM) detectors are replaced in each reactor annually. The replacement procedure is described in Subsection 9.1.4.2.2. Spent in-core detectors and dry tubes are transferred on a refueling platform auxiliary hoist underwater to the spent fuel pool.

A pneumatically operated cutting tool supplied with the nuclear steam supply system allows remote cutting of the in-core detectors and dry tubes on the work table in the fuel pool. The cut in-core monitors and dry tubes and other small sized reactor internals are shipped offsite in suitable containers and/or shielded casks that can be loaded underwater.

A trolley mounted disposal cask with internal cable drum is supplied with the nuclear steam supply system for spent source and intermediate range neutron monitor (SRM,IRM) detector cables retrieved from underneath the reactor, and the traverse in-core probe (TIP) wires retrieved into the reactor building.

<u>HEPA Filter Elements of the Offgas System</u>

The inlet and outlet HEPA filter elements of the ambient temperature charcoal offgas system are housed in four pressure vessels at the inlet and two pressure vessels at the outlet of each Unit's system. Due to the high activity expected, the inlet filter vessels are designed for remote shielded removal of the elements into an in-plant transfer cask. The annual number of disposed HEPA filter elements from the offgas system is shown in Table 11.4-2. The size of the individual filter elements allows disposal in a container with 55 gal drum size opening. To reduce the waste volume, the filter elements can be placed in a wet solid waste container prior to filling with the waste cement mixture. A floor hatch above the wet solid waste container decontamination station is provided for this purpose.

The offgas HEPA filter cartridges are prevented from floating to the top of the waste containers by added weights, thus taking advantage of the cement shielding.

Filter Media from Ventilation Systems and the Laundry Drain <u>System, Miscellaneous Contaminated Dry Solid Wastes</u>

Administrative procedures provide for frequent radiation monitoring and periodic replacement of the ventilation and laundry drain system filter media to limit the dose to maintenance personnel during handling to as low as reasonably achievable (ALARA). Redundant filter trains further allow shutdown of one train for decay of the radioactive isotopes in the filter media before replacement. Portable charcoal removal and loading systems are employed for packaging exhausted charcoal beds in 55 gal drums.

Pre- and HEPA filter elements are manually retrieved from the filter housings and wrapped in dust-tight plastic bags.

Where possible, the dry filter media waste and miscellaneous wastes from the rad lab instrumentation shop and decontamination stations are compressed into 55 gal drums by a hydraulic compactor with vent hood in the radwaste building. A fan on the compactor keeps the 55 gal drum interior at a slight vacuum with discharge through a HEPA filter to the building ventilation duct.

A special compactor may be installed later for compacting the 24 in. ventilation filters into standard 55 gal drums.

The averaged annual volumes of uncompacted dry solid wastes originating from dry filter media is shown in Table 11.4-2. The averaged annual volume of charcoal waste is derived from the bed depth, number of test canisters provided, and required test frequency per NRC Regulatory Guide 1.52. These volumes may be up to six times higher for any given year due to removal of the longest time exposed test canister. The averaged annual volume of prefilter, HEPA, and laundry drain cartridge filter waste is estimated from previous plant experience. The 55 gal drums are stored adjacent to the hydraulic compactor with storage capacity available for twenty (20) 55 gal drums.

Volumes of miscellaneous wastes may vary widely depending on the housekeeping in the plant. Compacting reduces the miscellaneous waste volume approximately by a factor of three.

The averaged annual dry solid waste volume, when partially compacted, is expected to be approximately 20 percent of the total solid waste volume including solidification agents. This compares with the results of the "Survey and Evaluation of Handling and Disposing of Solid Low-Level Nuclear Fuel Cycle Wastes" published by the AIF in October 1976.

TABLE 11.4-1

INPUTS TO THE WASTE MIXING TANKS

SOURCE	WASTE TYPE (NOTE 4)	EXPECTED AVERAGED BATCH SIZE AND FRE-QUENCY (FT^3/DAYS)	MAX. EXPECTED BATCH SIZE AND FRE-QUENCY (FT^3/DAYS)	EXPECTED YEARLY VOLUME (FT^3) (NOTE 1)	EXPECTED YEARLY SOLIDI-FIED VOLUME USING PORTLAND CEMENT AND SODIUM SILICATE WITH ADDITION OF EVAP. CONC. (FT^3)	NOTES
Radwaste Demineralizer	Dewatered spent ion exchange resin beads	140/40	140/6.2	1281	7103	2
Radwaste Filters	Dewatered fouled D.E. incl. crud	20/19.2	20/3	385	1397	2
Waste Sludge Phase Sep.		21.55/4.85	21.55/0.212	1622	5604	2
Cond. Demin. Resin Cleaners	Dewatered corrosion prod. crud and D.E. precoat and body feed	(13.34/4.63)	(13.34/0.22)	(1052)		
Fuel Pool Cleanup F/D	Dewatered (40 w/o) spent powdered ion exchange resin, corrosion products crud and small amounts of D.E.	(14.05/9)	(14.05/9)	(570)		
RW Cleanup Phase Sep's	Dewatered spent powdered ion exchange resin, corrosion products crud and small amounts of D.E.	69.3/60	86/60	422	1991	2
Radwaste Evaporators		200/4.67	200/0.276	15645	5781 (Note 1)	3
Cond. Demin. Regeneration	Sodium sulfate (12 w/o) Corrosion products, crud.	(182.8/4.63)	(188/0.27)	(14411)		
Labs and Decon. Stations	Various chemicals	(1.34/1)	(1.34/1)	(488)		
Auxiliary Boilers	Crud, corrosion products	(2.05/1)	(24/1)	(746)		

1) Dewatered resin and sludge volumes as shown in column (5) will be mixed with evaporator concentrate resulting in a 15 w/o suspended solids mixture prior to solidification. The excess evaporator concentrate (4574 ft^3) is solidified separately as shown in column (6).

2) Max. Expected Batch Size and Frequency is for one unit in cold startup, the other unit in normal operation.

3) Max. Expected Batch Size and Frequency is for one unit with cond. demin. back to back regenerating, the other unit in normal operation.

TABLE 11.4-1 (Continued)

4)	Bulk density of dewatered spent resin beads (50 w/o moisture; 3.6 w/o crud):	45.9 lb/ft^3.
	Bulk density of dewatered fouled D.E. (43 w/o moisture; 43.7 w/o crud):	32.1 lb/ft^3.
	Bulk density of dewatered crud	100 lb/ft^3
	Bulk density of dewatered powdered ion exch. resins (60 w/o moisture):	45 lb/ft^3
	Density of evaporator concentrate (12 w/o Sodium sulfate):	70.35 lb/ft^3
	Density of dry portland cement (particulate)	94 lb/ft^3
	Density of sodium silicate (SiO /Na_2O ratio 3.22):	86.8 lb/ft^3

TABLE 11.4-3

SOLID WASTE MANAGEMENT SYSTEM COMPONENT DESCRIPTION

A. PUMPS	EQUIPMENT NOS.	TYPE	QUANTITY	MATERIAL	CAPACITY EACH, gpm	TDH ft	USAGE FACTOR NORMAL	DRIVER hp	DESIGN PRESSURE/ TEMP. psig/°F
Solid Radwaste Collection									
Spent Resin Transfer	0P-320	Prog. Cavity (Moyno)	1	SS	200	369	0.010	60	160/140
RW Cleanup Phase Sep. Decant	0P-336	Horiz. Centr.	1	SS	50	155	0.012	10	150/155
RW Cleanup Phase Sep. Sludge Disch.	0P-334	Horiz. Centr.	1	SS	200	272	0.008	30	150/155
Waste Sludge Phase Sep. Decant	0P-331	Horiz. Centr.	1	SS	50	155	0.031	10	150/155
Waste Sludge Phase Sep. Disch	0P-332	Horiz. Centr.	1	SS	200	272	0.03	30	150/155
Cond. Demin. Regen. Waste Transfer	1P-106A,B	Inline Centr.	2	SS	85	53	0.009	3	150/105
Cond. Demin. Regen. Waste Transfer	2P-106A,B	Inline Centr.	2	SS	85	150	0.009	10	150/105
Radwaste Solidification									
Process Feed	0P-304A,B	Prog. Cavity (Moyno)	2	SS/BUNA N	20	(setp.3 psig)	0.007	2	125/180
Mixing	0P-307A,B	Prog. Cavity (Moyno)	2	SS/BUNA N	32	(setp.3 psig)	0.007	3	125/180
Sodium Silicate	0P-309	Prog. Cavity (Moyno)	1	CS/BUNA N	2.9	(setp.180 psig)	0.014	1/2	125/105
Cement Feed	0S-305	Rotary Valve	1	C.I.	(110 lb/min.)		0.014	1	

TABLE 11.4-3 (Continued)

B. TANKS	EQUIPMENT NOS.	TYPE	QUANTITY	MATERIAL	LIVE/ NOMINAL CAPACITY, EACH,gal	DESIGN PRESSURE/ TEMP.psig/°F
Solid Radwaste Collection						
Spent Resin	OT-324	Vert. Cyl.	1	SS	7150/750	Atmos./200
RW Cleanup Backw. Receiving	1T-225	Vert. Cyl.	1	SS	2335/3000	Atmos./200
RW Cleanup Backw. Receiving	2T-225	Vert. Cyl.	1	SS	2335/3000	Atmos./200
Fuel Pool Backw. Receiving	OT-203	Vert. Cyl.	1	SS	1815/2500	Atmos./200
Cond. Demin. Regen. Waste Surge	1T-106A,B	Vert. Cyl.	2	CS	3300/3390	Atmos./105
Cond. Demin. Regen. Waste Surge	2T-106A,B	Vert. Cyl.	2	CS	3260/3200	Atmos./105
RW Cleanup Phase Separator	OT-318A,B	Vert. Cyl.	2	SS	3100 Supern./750 Sludge/7400	Atmos./200
Waste Sludge Phase Separator	OT-331	Vert. Cyl.	1	SS	7330Supern./500 Sludge/10780	Atmos./200
Radwaste Solidification						
Waste Mixing	OT-307A,B	Vert. Cyl.	2	SS	700/750	Atmos./200
Cement Silo	OT-306	Vert. Cyl.	1	CS	3950/4325	Atmos./100
Sodium Silicate	OT-309	Vert. Cyl.	1	CS	1150/1270	Atmos./200
HSA Waste Container	OS-321	4x4ft Cyl.	As Req.	CS	353/380	Per DOT 7A
LSA Waste Container	OS-320	6ft cube	As Req.	CS	1415/1570	Atmos./250

C. MISC. EQUIPMENT	EQUIPMENT NOS.	TYPE	QUANTITY	MATERIAL	CAPACITY		hp
Radwaste Building Crane	OH-301	Bridge & Trolley	1	CS	25 tons	48ft lift 168ft runway 34ft span	50
Waste Compactor	OS-313	Vert.hydr. piston	1	CS/SS	10 tons	14-1/2 in.stroke	5
Waste container transfer cart.	OS-315	Rail, cable	1	CS	20 tons	63.75ft travel	1
Waste container storage compartments							
Large	-	(Top entry)	12	concr.	For 6ft cube cont.		
Small	-	(with lid)	4	concr.	For 4x4ft Cyl. cont.		

TABLE 11.4-4

SOLID WASTE MANAGEMENT SYSTEM FLOWS (Refer to Figure 11.4-3)

Stream No.	Stream (NOTE 1)	Averaged Number and frequency of batch (NOTE 2)	Volume/Batch gal	Flow rate gpm	Weight of dewatered solids per batch, lb (NOTE 3)	Max. expected number and frequency of batch (NOTE 4)	Notes
1.	To Spent Resin Tank (from Radwaste Demin.)	1/40.6 days	1910	96	6535	1/6.3 days	Volume includes 28 gpm sluice water for 20 min.
2.	To Radwaste (Dewatering) Filter (From Spent Resin Tank)	7/40.6 days	2850	50	934	7/6.3 days	Volume includes flushing
3.	To Waste Sludge Phase Separator (From Regen. Waste. T.)	1/4.63 days	7588	35	190	1/0.22 days	Density of dewatered solids assumed at 100 lb/ft.^{3}
4.	To Waste Sludge Phase Separator (From Fuel Pool Backw Tank)	1/18 days	1950	By Gravity	350	1/18 days	
5.	To Radwaste (Dewatering) Filter (From Waste Sludge Ph. Sep.)	1/4.85 days	10100	50	290	1/0.33 days	Frequency determined by RW filter capacity and based on 6.9 w/o of powdered ion exchange resin and crud settled as 500 gal. of sludge. 100 gal. flush water incl. Average density of dewatered solids 74 lb/ft.^{3} (not including filter precoat and body feed)
6.	To RWCU Phase Sep. (From RWCU Backw. Tanks)	1/2.93 days	2200	By Gravity	114	1/0.69 days	
7.	To Radwaste (Dewatering) Filter (From RWCU Phase Sep.)	5/60 days	6625	50	467	5.4/60 days	Frequency based on 20 w/o of powdered ion exchange resin and crud settled as 750 gal. of sludge. 100 gal. flush water incl.
8.	To Waste Mixing Tanks (From Radwaste Filter)	1/4.73 days	165	By Gravity	673	1/0.32 days	Waste sludge phase sep. crud powdered ion exchange resin and dewatering process D.E.

TABLE 11.4-4 (Continued)

Stream No.	Stream (NOTE 1)	Averaged Number and frequency of batch (NOTE 2)	Volume/Batch gal	Flow rate gpm	Weight of dewatered solids per batch, lb (NOTE 3)	Max. expected number and frequency of batch (NOTE 4)	Notes
9.	To Waste Mixing Tanks (From Radwaste Filter)	7/40.6 days	152	By Gravity	934	7/6.3 days	Radwaste Dem. Resins
10.	To Waste Mixing Tank (From Radwaste Filter)	1/19.2 days	151.5	By Gravity	650	1/3 days	Radwaste Filter d.e. and crud
11.	To Waste Mixing Tank (From Radwaste Filter)	5/60 days	103.6	By Gravity	546	5.4/60 days	RWCU Demin. powdered ion exchange resin, crud and dewatering process D.E.
12.	To Waste Mixing Tanks (From Evap.Conc. Tank)	1/2.34 days	780	20	7336	1/0.14 days	Weight based on tenfold concentration of Na2SO4 and aux. boiler blowdown, 100-fold concentration of lab drains. Max. for cond. dem. of one unit back to back regen.
13.	To Large Waste Containers (From Waste Mixing Tanks)	1/3.41 days	1415	20	Solidified 16017	1/0.21 days	Solidified with Portland cement and sodium silicate. Shipping efficiency 88.5%
14.	To Small Waste Containers (From Waste Mixing Tanks)	1/8.49 days	353	20	Solidified 3794	1/7.86 days	Solidified with Portland cement and sodium silicate. Shipping efficiency 88.4%.
15.	To Hydraulic Press/ Trash Compactor (From Labs and Decon. Stations)	1/1.33 days	165	-	Variable		Per AIF (NESP) survey and evaluation of handling and disposing of solid low level nuclear fuel cycle wastes, Oct. 1976, contaminated trash contributes 20% to annual solid waste volume
16.	To 55 Gal. Drum (From Hydraulic Press)	1/1.33 days	55	-	Variable		

TABLE 11.4-4 (Continued)

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1) Filter and Demineralizer wastes are solidified with addition of evaporator concentrate.

2) Batch frequencies are from total number of equipment.

3) Dewatered ion exchange resins contain 50-60 w/o interstitial water

4) Maximum expected numbers and frequencies of batches are for one unit cold startup one unit normal operation, except for stream 12.

TABLE 11.4-8

TYPICAL DEWATERED FILTER CAKE SLURRY RATIOS

WASTE	APPROX.* DENSITY #/FT^3	APPROX. MOISTURE CONTENT OF FILTER CAKE W/O	ASSUMED VALUE OF DEWATERED WASTE FT^3	VALUE** OF 25 W/O Na_2SO_4 EVAP CONCENTRATE TO BE ADDED-FT^3	15 W/O SOLID SLURRY MIXED VOLUME FT^3	OT-307*** TANK LEVEL READING % FULL
Crud & D. E.	25	60	10	18.15	21.2 (158 gal)	19
Fouled Powdex & Solka Floc	45	55	10	32.7	39.0 (291 gal)	35
Fouled DE	21	60	10	15.25	17.85 (133.5 gal)	17
Bead Resins	50	60	10	36.3	43.7 (326 gal)	39

* These data are subject to verification in field with actual equipment performance.

** Concentrate volume required to produce 15 w/o dewatered solids slurry. This is the maximum solids concentration to be used in normal processing.

*** If the moisture content of the filter cake varies from the value given above, adjustment of the dilution liquid added is required.

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The process and effluent radiological monitoring and sampling systems are provided to monitor releases of radioactive material in the plant gaseous and liquid process and effluent streams in order to control these releases.

11.5.1 DESIGN BASES

11.5.1.1 Design Objectives

The design objectives of the systems described in this section are to generally conform with the requirements of General Design Criteria 60, 63 and 64; and Regulatory Guide 1.21, Rev. 1, 6-74.

Certain of the effluent systems described provide mitigating circuits for the Engineered Safety Feature Systems. These systems are designed to be in compliance with IEEE 279-1971 to assure performance of the protective action required. References to Chapter 7 are provided for these systems.

Provision for monitoring postulated accidents is primarily provided by the measuring range of the channel provided. These ranges are noted in table 11.5-1

11.5.1.2 Design Criteria

Design Criteria are General Desin Criteria 60, 63 and 64, Regulatory Guide 1.21, Rev. 1, 6-74, and IEEE 279-1971.

11.5.2 PROCESS AND EFFLUET RADIOLOGICAL MONITORING SYSTEM DESCRIPTION

11.5.2.1 Continuous Process and Effluent Monitor Systems

Process and effluent radiological monitoring systems (RMS) are identified below. Systems which provide safety related functions are noted and appropriate references are made. One influent system is listed since it is the same basic equipment.

(1) Gaseous effleunt stream monitoring:

a) Reactor Building Vent Exhaust Sampler

b) Turbine Building and Radwaste Vent Exhaust Sampler and Sample RMS.

c) Standby Gas Treatment Vent Exhaust Sampler.

d) Standby Gas Treatment Vent Exhaust RMS: a safety related system also discussed in Subsections 7.3.1.1b.4 and 9.4.2

e) Refueling Floor Wall Duct Exhaust RMS: a safety related system also discussed in Subsections 7.3.1.1b.5 and 9.4.2

f) Refueling Floor High Exhaust Duct RMS: a safety relates systemalso discussies in Subsections 7.3.1.1b.5 and 9.4.2

g) Railroad Access Exhaust Duct RMS: a safety related system also discussed in Subsections 7.3.1.1b.5 and 9.4.2

h) Outside Air Intake (Influent) RMS: a safety related system also discussed in subsections 7.3.1.1b.7, 9.4.2 and Section 6.4.

(2) Liquid effluent monitor systems

a) Plant Radwaste Effluent
b) Service Water Discharge

(3) Gaseous process streams monitoring

a) Offgas Pretreatment
b) Offgas Post-treatment
c) Main Steamline Radiation Monitoring System (see Subsection 7.3.1)

4) Liquid Process Monitor Systems

a) RHR Service Water RMS

11.5.2.1.1 Reactor Building Vent Exhaust Sampler

This system samples effluent from reactor building ventilation exhaust from zones as noted on Figures 9.4-7, 9.4-8 and 9.4-9. A sample is continuously extracted using an isokinetic probe designed for the vent. The sample is passed through particulate and adsorbent filters and returned to the exhaust vent.

Periodic removal of the filters for laboratory analysis will provide data for required effluent reports.

A remotely operated purge of the system is provided.

11.5.2.1.2 Turbine Building and Radwaste Vent Exhaust Sampler and Sample RMS

This system samples effleunt from the turbine building exhaust vent which serves the turbine building ventilation and the radwaste building as shown on figure 9.4-14. This effluent includes the processed and filtered reactor gaseous radwaste.

A continuous sample is extracted via an isokinetic probe designed for the vent. The sample is monitored in a local sample panel to minimize sample line length.

Dual GM tubes monitor the sample for radiation in parallel with the filtering of the sample through particulate and adsorbent filters. The filters are periodically removed for laboratory analysis.

The two GM tubes are used in identical instrument channels each consisting of the detector, a log count rate meter, and one pen of a two pen recorder located in the main control room. A check source and purge operated from the control room provide the ability to check proper operation. Each circuit is provided with two upscale and one downscale trip points to indicate high radiation and malfunction respectively. Control room alarms are provided for these trips. Ranges of instruments and setpoints are provided per the technical specifications. See table 11.5-1 for details.

11.5.2.1.3 Standby Gas Treatment System Vent Sampler and Sample RMS

The system monitors radioactivity in the Standby Gas Treatment System Vent prior to release to the environment. A continuous sample is extracted using an isokinetic probe designed for the vent. The sample is passed through the standby gas treatment sample panel for monitoring and filtering and returned to the exhaust vent. The sample panel has a pair of filters (one for particulate collection and one for halogen collection). This sampling and monitoring system is the same as the turbine building vent sampling system.

11.5.2.1.4 Standby Gas Treatment Vent Exhaust Radiation Monitoring System

The system monitors gamma radiation in the exhaust vent of the standby gas treatment system. Two redundant instrument channels constitute this system. Automatic trip of the active standby gas treatment system and the standby train is actuated upon detection of high radiation in the exhaust vent.

(1) The detectors are mounted inside the exhaust vent.

(2) The sensors are Geiger-Muller tubes with converter units. Refer to table 11.5-1.

(3) Two redundant, independent instrument channels are designed each with a detector/converter unit mounted locally and an indicator trip unit in the control structure. The radiation measurement is displayed on the indicator trip unit and both channels are recorded in the main control room on a two pen recorder.

The trip circuit detects two upscale trips (high-high and high radiation) and one downscale/inoperative trip.

(4) The following annunciators are located in the control room:

a) Standby Gas Exhaust Vent High-High Radiation

b) Standby Gas Exhaust Vent High Radiation

c) Standby Gas Exhaust Vent Monitor Downscale

5) The power source for instrument channel A is from the reactor protection system 120 V ac supply bus A.

Instrument Channel B is powered from RPS bus B.

6) Set points for alarms are defined in the technical specification (Chapter 16).

11.5.2.1.5 Refueling Floor Wall Exhaust Duct Radiation Monitoring System

This system monitors the radiation level in the exhaust duct from the refueling floor prior to its discharge to the atmosphere through the reactor building vent.

Refer to Figure 9.4-8 for system design.

(1) The detection assemblies are located in the exhaust ducting upstream of the inboard isolaticn damper. The distance from the inboard isolation damper is defined by the ventilation flow transport time at maximum design flow rate from the detector location to the inboard isolation damper, which is greater than the total time to respond to a trip level radiation for complete closure of the inboard isolation damper.

(2) The detector assembly is a Geiger-Muller tube with a converter unit.

(3) Two redundant independent instrument channels are provided. Each channel consists of a local detector and converter unit, which transmits a signal to the control room indicator and trip unit. Two trip circuits monitor the upscale (high-high)/inoperative condition and the downscale condition. The upscale trip indicates high radiation and the downscale trip indicates instrument trouble. The upscale trip initiates closure of the reactor building Zone III ventilation outboard isolation dampers, starts the SGTS and the reactor building recirculation system (see Section 6.5). Refer to Section 7.3 for isolation logic. The radiation level of each channel is recorded on a two pen recorder.

(4) Measurement ranges are defined in Table 11.5-1.

(5) Two alarms for high radiation (high and high-high) and one alarm for downscale trip are located in the control room. Actuation ofisolation dampers is discussed in Section 7.3.

(6) Reactor protection system power bus A is the power source for one instrument channel. RPS bus B is power source for the other channel. Refer to Chapter 8.

(7) The monitors are readily accessible for calibration, inspection and maintenance. A portable gamma source may be used for testing the instrumentation.

11.5.2.1.6 Refueling Floor High Exhaust Duct Radiation Monitoring System (Unit 1 only)

This system monitors the radiation level exterior to the fuel pool ventilation exhaust duct. The system is identical to the refueling floor wall exhaust radiation monitoring system with the same channel trip logic and protective action initiation. Refer to Figure 9.4-8 and Table 11.5-1 for system configuration.

11.5.2.1.7 Railroad Access Exhaust Duct Radiation Monitoring System (Unit 1 only)

This system monitors the radiation level at the air exhaust duct prior to the reactor building vent. The system design is identical to the refueling floor wall radiation monitoring system. Table 11.5-1 identifies type of detector, location and instrument ranges. Refer to Section 7.3 for reactor building isolation initiation. Figure 9.4-8 documents the system configuration.

11.5.2.1.8 Control Structure Outside Air Intake Radiation Monitoring System

The radioactivity of the outside air intake for the control structure is continuously monitored to detect airborne radioactive material which enters the heating, ventilating, and air conditioning system for the control structure. An increase in gamma radiation is detected by the two redundant instrument channels. The system provides a trip signal to initiate the emergency intake air supply system for the control room habitability engineered safety feature. Refer to Section 7.3 for actuation of this safety system.

(1) The two redundant monitors are located in the outside intake air plenum.

(2) The radiation detection system uses a Geiger-Muller tube with a converter unit.

(3) Two redundant, independent instrument channels are provided, each with an indicator and trip unit in the control structure. The radiation measurement is displayed on a four decade logarithmic scale. A high radiation reading initiates the upscale trip circuit and contacts for alarm and protective action are provided. Power failure or component malfunction causes downscale trip initiation.

    Each channel measurement is recorded on a two pen recorder in the control room.

(4) Instrument ranges and scale information are in Table 11.5-1.

(5) High/high radiation and downscale/inoperative alarms are displayed in the control room. The two pen recorder initiates a control room high radiation alarm set at a slightly lower level than the high/high alarm.

At high/high radiation level, the trip circuit initiates the emergency outside air intake system as described in Section 7.3.

(6) Each channel is powered by its separate and independent reactor protection system power supply, channel A from RPS bus A, and channel B from RPS bus B.

Refer to Chapter 8 for electrical power distribution.

(7) Set points for alarms and controls are per the technical specification. Refer to Chapter 16.

(8) The instrumentation is readily accessible for calibration, inspection, and maintenance. A portable gamma source unit may be used for testing and calibration.

11.5.2.1.9 Liquid Radwaste Effluent RMS

This system monitors the gross gamma activity in the discharge line from liquid radwaste prior to discharge into the cooling tower blowdown line.

The system monitors a sample of the discharge effluent by use of a side stream of the discharge. The side stream pipe is located downstream of the last point of waste admission to the discharge, and upstream of the first isolation valve. The sample flow is pumped by the Liquid Radwaste Effluent RMS through a shielded sample chamber and back to the discharge line upstream of the flow control valve (Fig. 11.2-12)

The Liquid Radwaste Effluent RMS consist of:

1. a flow element to measure total discharge flow rate. The flow rate is recorded on a recorder on panel OC301 in radwaste control room. A signal is sent to a radiation calcultor to be combined with a signal from the radiation element system to calculate the integrated release in μci. This calculated release is recorded on pen two of a two pen recorder.

2. a gamma sensitive scintillation detector contained in the shielded sample chamber. The count rate is converted to μ ci/ML and recorded on pen one of a two pen recorder as the instantaneous release rate. The signal is also sent to the radiation calculator to calculate the integrated release in μci.

The monitoring instrumentation provides three trips, all of which terminate radwaste discharge. The trips are high radiation, downscale and low sample flow. The high radiation and downscale trips initiate alarms in the main control room and the low sample flow trip initiates an alarm in the radwaste control room. A check source is provided for periodic testing of the monitoring system.

All controls, alarms, set point adjustment and indicators are located in the radwaste control room, except for trouble alarms located in the main plant control room. Two isolation valves installed in series are provided. These valves are air operated and fail closed. All controls and permissive interlocks require contact closures to open the valves, thereby providing fail safe valve control. No emergency power is provided.

System sensitivity meets the requirements of Regulatory Guide 1.21 Rev. 1, 6-74 of 1×10^{-7} µCi/ml by gross radioactivity measurement.

Refer to Section 11.2 for a discussion of consentrations, compositions, flows and measurements.

11.5.2.1.10 Service Water Discharge Radiation Monitoring System

The objective of this system is to detect radioactive material leakage to the service water from any source.

(1) The service water monitor is located on the downstream side of the fuel pool heat exhangers prior to discharge to the cooling tower. (See Figure 9.2-1b).

(2) A scintillation detector is provided (See Table 11.5-1).

(3) The detection assembly is mounted in a piping T-type flange in the process flow. Shielding is provided to ensure adequate sensitivity to the process radiation. A local pulse amplifier transmits the signal to the radiation monitoring unit in the control structure (upper relay room). The radiation monitor provides three trip points: One upscale (high-high radiation), one upscale (high radiation), and one downscale/inoperative. High radiation and downscale trips are alarmed in the control room. The radiation level can be observed in the control room on a strip chart recorder.

(4) Table 11.5-1 shows instrument ranges

(5) No emergency power is provided.

(6) Calibration, maintenance and testing is based on the technical specification in Chapter 16.

11.5.2.1.11 Offgas Pretreatment Radiation Monitoring System

This system monitors radioactivity in the condenser offgas discharge after it has passed through the steam jet air ejector (SJAE), see Figure 10.4.9. 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. It is then passed through a sample chamber and a sample panel before being returned to the suction side of the 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 to background radiation 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 sample chamber is monitored by three channels. Each channel has a gamma-sensitive ionization chamber mounted external to the sample chamber. Two channels have logarithmic radiation monitors which provide system alarm output. The third channel has a linear radiation monitor for recording in the control room. The two logarithmic channels are recorded on a separate two pen recorder.

Power is supplied from the 125 V nondivisional bus for the logarithmic channels, from 24 V bus A for the linear channel, from the 120 V instrument bus for the recorders, and from a local 120 V bus for the sample and vial sampler panels.

The logarithmic radiation monitors have 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 actuates a control room annunciator: offgas high-high, offgas high, and offgas downscale/inoperative. High or low sample line flow measured at the sample panel actuates a control room offgas sample high-low flow annunciator.

A correlation between the observed activity and the monitor reading permits calibration of the monitor. The radiation level output by the monitor may be 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 from the bottle to a solenoid-operated sample valve. The solenoid-operated sample valve is opened to allow offgas to enter the bottle. The bottle is then removed and the sample is analyzed in the counting room with a multichannel gamma pulse height analyzer to determine the concentration of the various noble gas radionuclides.

11.5.2.1.12 Offgas Post Treatment Radiation Monitoring System

This system monitors the radioactive level in the offgas piping downstream of the offgas system charcoal bed. By comparing this measurement with the offgas pretreatment radiation level, the effectiveness of the offgas treatment system can be determined.

(1) The detector is located in a snowplow sampler mounted on the downstream piping.

(2) A Geiger-Muller detection is used to measure gross gamma radiation.

(3) The instrumentatioh consists of a Geiger-Muller tube, a log rate meter, an alarm switch, a strip chart recorder in the radwaste control room, and an indicating meter in the main control room.

(4) Instrument range is shown in Table 11.5-1.

(5) One high radiation alarm is in the radwaste control room, and one high radiation alarm is in the main control room.

(6) No emergency power is provided for this system.

11.5.2.1.13 Main Steamline 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 channel trip signals to the reactor protection system to initiate protective action. Refer to Subsections 7.2.1 and 7.3.1.

(1) Four radiation monitors are located near the main steam lines just downstream of the outboard main steam line isolation valves in the space between the primary and

secondary containment walls. The detectors are geometrically arranged so that this system is capable of

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detectiqn significant increases in radiation level with any number of main steam lies in operation.

(2) Each monitor has a gamma sensitive ion chamber with 3.7×10^{-10} amp/R/hr sensitivity.

(3) The system consists of four redundant instrument channels. Each ion chamber detector provides a signal to a control room log-radiation monitor with meter and auxiliary trip unit. One two pen recorder powered from the 120 V instrument bus allows the output of any two channels to be recorded by the use of selection switches. The channels A and C are physically and electrically independent of channels B and D.

(4) Table 11.5-1 lists the range of the detectors.

(5) 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 monitor. A high-high or inoperative trip in the radiation monitor results in a channel trip in the auxiliary unit, which is an input to the reactor protection system (RPS). A RPS logic trip from MSL channel input results in initiation of main steamline isolation valve closure, reactor scram, and mechanical vacuum pump shutdown and discharge valve closure. A high trip initiates a main control room annunciator common to all channels. A downscale trip actuates a MSL downscale control room annunciator common to all channels. High and low trips do not result in a channel trip. Each channel has a control room display of the measured radiation level.

(6) Power for two channels (A and C) is supplied from the RPS bus A and for the other two channels (B and D) from the RPS bus B. Refer to Chapter 8 for more detail on emergency power supply.

(7) Set points for alarms and controls are discussed in Subsection 7.3.1.

(8) Calibration, maintenance, inspection, decontamination and replacement of radiological instruments is based on the technical specifications.

11.5.2.1.14 RHR Service Water Radiation Monitoring System

This system is designed to detect primary coolant leakage into the RHR service water during operation of the RHR heat exchanger. The two RHR heat exchangers are each rated for 100 percent of reactor shutdown operation. In the event of a leakage in one heat exhanger, the redundant unit could be placed into service by the control room operator.

A radiation detector is located on the downstream piping of each RHR heat exchanger.

The description of instrumentation and locations of annunciators, power supplies, etc, are similar to those outlined in Subsection 11.5.2.1.10. Table 11.5-1 identifies the instrumentation.

11.5.2.1.15 Reactor Building Closed Cooling Water Radiation Monitoring System

The radiation monitor system detects leakage from the reactor water cleanup system through the nonregenerate heat exchanger system. Any increase of the radiation level is an indication of leakage.

The detector is located in the suction header piping of the RBCCW pumps.

Table 11.5-1 identifies the provided instrumentation.

11.5.2.2 Routine Sampling

The requirements of the system design bases for routine continuous and discrete sampling of radioactivity are satisfied by a system of liquid, gaseous, and airborne samplers, laboratory equipment for sensitive radio-chemical analyses, and a program of procedures for obtaining and analyzing representative samples when and where appropriate. This subsection provides a description of system hardware and procedures in general, including the type of sampling equipment used, the procedures to obtain representative samples, and analytical procedures. Table 11.5-2 is a tabulation of basic information describing each of the radioactivity sampling locations, including the basis for selecting the location, expected process flows, sampling frequency, analytical procedure, and expected monitor sensitivites. Table 11.5-3 gives the expected composition and concentration of nuclides in routine effluent samples.

Sampling equipment and procedures are provided to assure that representative samples are obtained. Prior to sampling, large tanks of liquid waste are well-mixed in as short an interval as practicable to assure that any sediments or particulate solids are distributed uniformly in the waste mixture. Sample lines are flushed for a sufficient period of time prior to sample extraction in order to remove sediment deposits and air and gas pockets. A sample is taken before discharge to determine the isotopic mixture and concentration of the tank. During discharge, a composite sample is taken of the effluent mixed with cooling tower blowdown to assure that the individual sample was indeed representative of the tank. Periods of collection are kept as short as pracicable, and polyethylene collection bottles are used to preclude the loss of radioactive material by deposition on the walls of the sample container or volatilization of potentially volatile material. Periodic checks are performed to identify any such changes.

Effluent ventilation vents are sampled continuously and isokinetically for radioactive gases, particulates, and iodines. Particulate and iodine sampler filters are replaced and removed for analysis weekly for all continuous airborne radiation monitors and samplers. A gas sample will be taken monthly or if the gaseous monitor count rate shows a significant change.

11.5.2.2.1 Analytical Procedures

Samples of process and effluent gases and liquids are analyzed in the laboratory by the following techniques:

A. Gross beta counting

B. Gross alpha counting

C. Gamma spectrometry

D. Liquid scintillation counting

E. Radiochemical separations

Instrumentation available in the laboratory for the measurement of radioactivity includes:

A. End-window GM counter

B. Gas flow proportional counter

C. Liquid scintillation counter

D. Gamma spectrometer

1. GeLi detctor

2. NaI detector

3. Multichannel analyzer system

Gross beta and gross alpha analysis of all liquid effluent samples are performed with the gas flow proportional counter. In both cases, samples are evaporated onto stainless steel planchets for counting. Sample volume and counting time are chosen to give the desired sensitivities. Corrections are made for sample-detector geometry, sample self-absorption, and other parameters as necessary to assure accuracy.

Gross beta and gross alpha analysis of air particulate is performed by counting of filters with the gas flcw proportinal counter.

Gamma spectrometry is used for isotopic analysis of liquid, gaseous, and airborne particulate and iodine samples. A high-efficiency, high-resolution GeLi detector and a NaI detector are available for resolving complex gamma spectra.

Gaseous tritium samples are collected and counted on the liquid scintillation conter.

11.5.3 EFFLUENT MONITORING AND SAMPLING

General Design Criteria 64, "Monitoring Of Radioactivity Releases," is implemented using the equipment and systems described in Subsection 11.5.2. With respect to the specific areas, discharges and environs mentioned in General Design Criteria 64, the following subsections apply.

11.5.3.1 Containment Atmosphere

Monitoring of containment atmosphere for radioactivity is described in subsections 5.2.5, and 7.6.1b.1.4. Description is given of the systems which can detect leakage of radiation from the vessel and piping to the primary containment. Monitoring is continuous and is applicable to normal operations, anticipated ocurrences and accident conditions.

11.5.3.2 Reactor Building

The reactor building contains components and piping which are used for the recirculation of LOCA fluids. Radiation monitoring in this space consists of ventilation duct monitors, (refer to subsections 11.5.2.1.4 through 11.5.2.1.7) and the area radiation monitors, (refer to subsection 12.3.4 and table 12.3-7 channels 1 thru 16). The descriptions for this equipment apply for normal, operating, anticipated occurancies and accident conditions.

11.5.3.3 Effluent Discharge Paths

Monitoring of plant effluent discharge paths is described in Subsection 11.5.2.

11.5.3.4 Plant Environs Measurement

Properational Environs measurements are discussed in Chapter 6 of the Susquehanna S.E.S. Environmental Report. Operational Environs Measurements are discussed in Appendix F of the Environmental Technical Specifications.

11.5.4 PROCESS MONITORING AND SAMPLING

Systems monitoring gaseous process streams are described in Subsections 11,5,2,1,11 through 11.5.2.1.13.

Systems monitoring liquid process streams are described in 11.5.2.1.14 and 11.5.2.1.15.

11.5.4.1 Process Monitoring and Sampling Systems

These systemts implement General Design Criteria 60 with respect to automatic closure of isolation valves as described in the following subsections.

11.5.4.1.1 Plant Radwaste Effluent RMS

This monitoring system will initiate isolation of two effluent discharge valves. Description is provided in 11.5.2.1.9.

11.5.4.1.2 Service Water Discharge RMS

This system does not provide initiation of any isolation function.

11.5.4.1.3 Offgas Pretreatment RMS

There is no provision for the offgas pretreatment monitor system to automatically stop this effluent from reaching the turbine building exhaust vent.

11.5.4.1.4 Offgas Post TReatment RMS

There is no provision for the offgas post treatment monitor system to automatically stop the effluent from reaching the turbine building exhaust vent.

11.5.4.1.5 Main Steamline RMS

Detection of high radiation by this monitor system is a trip to the Reactor Protection System and Primary Containment Isolation (Section 7.2 and Subsection 7.3.1.1a).

11.5.4.1.6 RHR Service Water RMS

There is no provision for isolation initiation by this System.

11.5.4.1.7 Reactor Building Closed Cooling Water RMS

There is no provision for isolation initiation by this system.

11.5.4.2 Process Monitoring and Sampling Systems

These systems implement General Design Criteria 63, "Monitoring Fuel and Waste Storage," with respect to radiation levels in radioactive waste process systems. Radiation levels are measured only at points in the actual discharge lines in the case of liquids and at the discharge vents in the case of gases (Turbine

TABLE 11.5-1

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS

MONITORED PROCESS	NO. OF CHANNELS	DETECTOR TYPE	DETECTOR LOCATION	CHANNEL RANGE	UPSCALE SET POINT WARNING ALARM	UPSCALE SET POINT TRIP	SCALE
A. Safety-Related Systems							
Main Steamline	4	Ionization chamber	Immediately downstream of last main steam valve	$1-10^6$ mr/hr	later	technical specification	6 dec. log
Refuel Floor Wall Exhaust	2	Geiger-Muller tube	Exhaust duct upstream of exhaust ventilation isolation damper	0.01 mr/hr to 100 mr/hr	later	technical specification	4 dec. log
Refuel Floor High Exhaust	2	Geiger-Muller tube	Exhaust duct upstream of exhaust ventilation isolation damper	0.01 mr/hr to 100 mr/hr	later	technical specification	4 dec. log
Emergency Outside Air Intake	2	Geiger-Muller tube	Outside air intake plenum	0.01 mr/hr to 100 mr/hr	later	technical specification	4 dec. log
Railroad Access Shaft Exhaust	2	Geiger-Muller tube	Exhaust duct upstream of exhaust ventilation isolation damper	0.01 mr/hr to 100 mr/hr	later	technical specification	4 dec. log
B. Systems Required for Plant Operation							
Service Water Discharge	1	Scintillation	Effluent pipe prior to discharge into other systems	10^{-1} to 10^6 counts/sec	above background	not applicable	7 dec. log
Reactor Building Closed Cooling Water	1	Scintillation	Suction header to closed cooling water pumps	10^{-1} to 10^6 counts/sec	above background	not applicable	7 dec. long
RHR Service Water	2	Scintillation	Process pipe downstream of heat exchanger	10^{-1} to 10^6 counts/sec	above background	not applicable	7 dec. log

TABLE 11.5-1 (Continued)

MONITORED PROCESS	NO. OF CHANNELS	DETECTOR TYPE	DETECTOR LOCATION	CHANNEL RANGE	UPSCALE SET POINT WARNING ALARM	UPSCALE SET POINT TRIP	SCALE
Liquid Waste Effluent (Plant)	1	Scintillation	Effluent pipe prior to discharge				
Offgas Pretreat	3	Ionization chamber	Sample line	1 to 10^6 mr/hr	later	not applicable	6 dec. log/ 6 dec. linear
Turbine Building/ Vent Exhaust	2	Geiger-Muller tubes	Sample line	10 to 10^6 counts/min	later	not applicable	5 dec. log
Standby Gas Treatment Vent Exhaust (Sampler)	2	Geiger-Muller tube	Sample line	10 to 10^6 counts/min	later	not applicable	5 dec. log
Standby Gas Treatment Vent Exhaust	2	Geiger-Muller tubes	Exhaust vent	0.01 to 100 mr/hr	later	not applicable	4 dec. log
Offgas Post Treatment	1	Geiger-Muller tube	Charcoal bed discharge	0 - 10^6 counts/min	later	not applicable	

TABLE 11.5-2

RADIOLOGICAL ANALYSIS SUMMARY OF ROUTINE EFFLUENT SAMPLING

Sample Location	Basis For Location	Expected Process Flow	Sampling Frequency	Analytical Procedure	Expected Sensitivities
Reactor Building Vents (isokinetic probe)	Determination of identity. concentration & quanity of radionuclides released	168,250 cfm (each unit)	Once per month or a) at the time of a significant power surge b) in the case of a 50% change in monitor count rate c) when there is a change in monitor count rate from an unknown cause	Analyzed for gamma emitting nuclides	10^{-4} µCi/CC (for each isotope)
Unit 1 Turbine Building Vent (isokinetic probe)	Determination of identity, concentration & quanity of radionuclides released	385,000 cfm	Continuous	Gross gamma activity	Sufficient to permit measurement of a small fraction of the activity which would result in (1) an annual air dose of 10 millirads at any location near ground level at or beyond the site boundary and (2) an annual air dose of 20 millirads due to beta radiation at any location near ground level at or beyond the site boundary
			Once per month or a) at the time of a significant power surge b) in the case of a 50% change in monitor count rate c) when there is a change in monitor count rate from an unknown cause	Analyzed for gamma emitting nuclides	10^{-4} µCi/CC (for each nuclide)

TABLE 11.5-2 (Cont'd)

Sample Location	Basis For Location	Expected Process Flow	Sampling Frequency	Analytical Procedure	Expected Sensitivities
Unit 2 Turbine Building Vent (isokinetic probe)	Determination of identity, concentration & quanity of radionuclides released	240,000 cfm	Continuous	Gross gamma activity	Sufficient to permit measurement of a small fraction of the activity which would result in (1) an annual air dose of 10 millirads at any location near ground level at or beyond the site boundary and (2) an annual air dose of 20 millirads due to beta radiation at any location near ground level at or beyond the site boundary
			Once per month or a) at the time of a significant power surge b) in the case of a 50% change in monitor count rate c) when there is a change in monitor count rate from an unknown cause	Analyzed for gamma emitting nuclides	10^{-4} µCi/CC (for each nuclide)
Standby Gas Treatment Exhaust Vent	Determination of identity, concentration & quanity of radionuclides released	7,500 cfm	Continuous	Gross gamma activity	Sufficient to permit measurement of a small fraction of the activity which would result in (1) an annual air dose of 10 millirads at any location near ground level at or beyond the site boundary and (2) an annual air dose of 20 millirads due to beta radiation at any location near ground level at or beyond the site boundary

TABLE 11.5-2 (Cont'd)

Sample Location	Basis For Location	Expected Process Flow	Sampling Frequency	Analytical Procedure	Expected Sensitivities
Standby Gas Treatment Exhaust Vent (Con't)			Once per month or a) at the time of a significant power surge b) in the case of a 50% change in monitor count rate c) when there is a change in monitor count rate from an unknown cause	Analyzed for gamma emitting nuclides	10^{-4} μCi/CC (for each nuclide)
Common Station Liquid Effluent Pipe	Determination of identity, concentration & quanity of radionuclides released	200 gpm	Continuous during discharge	Gross activity monitored during discharge	1×10^{-6}
				Grab sample prior to discharge to identify gamma emmiting nuclides	As per R.G. 1.21 Rev. 1 June 1974 for grab samples

12.1.3.1 Procedure Development

Station procedures will be prepared, reviewed, and approved in accordance with Section 13.5.

12.1.3.1.1 ALARA Procedures

To assure adequate emphasis on the necessity to minimize personnel exposures, ALARA procedures will be prepared as a subcategory of Health Physics procedures. These procedures implement considerations of such topics as ALARA Training, ALARA review of applicable Radiation Work Permits (RWP), worker feedback, special task training and evaluation of proposed changes in applicable facilities or equipment. ALARA procedures will provide the necessary basis for instruction of station personnel in the mechanisms available to minimize personnel exposures.

12.1.3.1.2 Station Procedures

Administrative requirements will be implemented to assure that applicable procedures developed by other plant disciplines have adequately incorporated the principle of minimizing personnel exposure. Station administrative documents will describe the criteria for selection of those procedures and revisions that will be reviewed by Health Physics. Recommendations made by Health Physics will normally be resolved with the appropriate plant discipline prior to submission for final review and approval.

12.1.3.2 Station Organization

As described in Subsection 12.5.1, the Station organization provides the Health Physics Supervisor direct access to the Superintendent of Plant to assure uniform support of Health Physics and ALARA requirements. This organization will allow the Superintendent of Plant direct involvement in the review and approval of specific ALARA goals and objectives as well as review of data and dissemination of information related to the ALARA program.

The organization also provides a Health Physics Engineer who is normally free from routine Health Physics activities to implement the Station ALARA program. This individual is primarily responsible for coordination of Station ALARA activities and will routinely interface with first line supervision in radiation work planning and post job review.

12.1.3.3 Operating Experience

The Radiation Work Permit process described in Subsection 12.5.3.2 will provide a mechanism for collection and evaluation of data relating to personnel exposure. Information collated by systems and/or components and job function will assist in evaluating design or procedure changes intended to minimize future radiation exposures.

12.1.3.4 Exposure Reduction

Specific exposure reduction techniques that will be employed at Susquehanna SES are described in Subsection 12.5.3.2. Procedures will assure that applicable station activities are completed with adequate preparation and planning; work is performed with appropriate Health Physics recommendations and support; and results of post job data evaluation are applied to implement improvements.

In addition, the Health Physics staff, will at all times be vigilant for ways to reduce exposures by soliciting employee suggestions, evaluating origins of plant exposures, investigating unusual exposures, and assuring that adequate supplies and instrumentation are available.

PP&L management will perform periodic reviews of station programs to assure workers are receiving adequate instruction in ALARA and Health Physics requirements. Implementation of the Health Physics program, selected procedures, and past exposure records will also be reviewed. Management will perform formal reviews of the Susquehanna SES Health Physics program at least once every three years and results will be forwarded to the Superintendent of Plant, ALARA Review Committee and appropriate members of corporate management. The results of management reviews may also include recommendations on mechanisms which may reduce personnel exposure. The Superintendent of Plant will respond to noted recommendations or deficiencies and corrective action or improvements will be verified during subsequent reviews.

12.1.4 REFERENCES

12.1-1 T.D. Murphy, WASH-1311, UC-78, A Compilation of Occupational Radiation Exposure from Light Water Cooled

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

Specific design features for maintaining personnel exposures ALARA are discussed in this subsection.

12.3.1.1 Common Equipment and Component Designs for ALARA

This subsection describes the design features used for several general classes of equipment or components. These classes of equipment are common to many of the plant systems; thus, the features employed for each system to maintain minimum exposures are similar and are discussed by equipment class in the following paragraphs.

Filters: Whenever practicable, filters that accumulate radioactive material are supplied with the means either to backflush the filter remotely or to perform cartridge replacement with semiremote tools (ie, long handled tools). For cartridge filters, adequate space is provided to allow removing, cask loading, and transporting the cartridge to the solid radwaste area.

Demineralizers: Demineralizers for radioactive systems are designed so that spent resins can be remotely and hydraulically transferred to spent resin tanks prior to solidification and that fresh resin can be loaded into the demineralizer remotely. Underdrains and downstream strainers are designed for full system pressure drop. The demineralizers and piping are designed with provisions for being flushed. Strainers are installed in the vent lines to prevent entry of spent resin into the exhaust duct.

Evaporators: Evaporators are provided with chemical addition connections to allow the use of chemicals for descaling operations. Space is provided to allow uncomplicated removal of heating tube bundles. To the extent practicable, the more radioactive components are separated from those that are less radioactive by a shield wall.

Pumps: Wherever practicable, pumps, in radioactive areas are purchased with mechanical seals to reduce seal servicing time. Pumps and associated piping are arranged to provide adequate space for access to the pumps for servicing. Small pumps are installed in a manner that allows easy removal if necessary. All pumps in radioactive waste systems are provided with flanged connections for ease in removal. Generally, pump casings are provided with drain connections for draining the pump for maintenance.

Wherever practicable, valves for clean, nonradioactive systems are separated from radioactive sources and are located in readily accessible areas.

Manually operated valves in the filter and demineralizer valve compartments required for normal operation and shutdown are equipped with reach rods extending through or over the valve gallery wall. Personnel are not required to enter the valve gallery during flushing operations. The valve gallery shield walls are designed for maximum expected filter backflush activities during flushing operations.

For most larger valves (2-1/2 in. and larger) in lines carrying radioactive fluids, a double set of packing with lantern ring is provided. A stuffing box is also provided with a leak-off connection that may be piped to a drain header. Full ported valves are used in systems expected to contain radioactive solids.

Special valve designs with minimum internal crevices are normally used where CRUD trapping could become a problem, especially for piping carrying spent resin or evaporator bottoms.

Piping: The piping in pipe chases is designed for the lifetime of the unit. The number of valves or instrumentation in the pipe chases has been reduced to maximum extent practicable. Where radioactive piping is routed through areas that require routine maintenance, pipe chases are normally provided to reduce the radiation contribution from these pipes. Wherever practicable, piping containing radioactive material is routed to minimize radiation exposure to the unit personnel.

Floor Drains: Floor drains and properly sloped floors are provided for each room or cubicle, within which are serviceable components containing radioactive liquids. If a radioactive drain line must pass through a zone lower than that at which it will terminate, proper shielding is provided. Local gas traps or porous seals are not used on radwaste floor drains. Gas traps are provided at the common sump or tank.

Lighting: Multiple electric lights are provided for each cell or room containing highly radioactive components so that the burnout of a single lamp will not require entry and immediate replacement of the defective lamp. Normally, incandescent lights that require less time for servicing are provided to minimize personnel exposure. The fluorescent lights used in some areas do not require frequent service because of the increased life of the tubes.

HVAC: The HVAC system design provides for rapid replacement of the filter elements and housings.

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Sample Stations: Sample stations for routine sampling of process fluids are located in accessible areas. Shielding is provided at the sample stations as required to maintain radiation zoning in proximate areas and minimize personnel exposure during sampling. Ventilation, drains or other means of contamination control are provided where necessary. The counting room and laboratory facilities are described in Section 12.5.

Clean Services: Whenever practicable, active components of clean services and equipment such as compressed air piping, clean water piping, ventilation ducts, and cable trays are not routed through radioactive pipeways.

12.3.1.2 Common Facility and Layout Designs for ALARA

This subsection describes the design features used for standard type plant processes and layout situations. These features are used in conjunction with the general equipment designs described in Subsection 12.3.1.1 and include the features discussed in the following paragraphs.

Valve Galleries: Valve galleries are provided with shielded entrances for personnel protection. In many cases the valve galleries are divided by shielding or distance into subcompartments that service only two or three components and are further subdivided by fin walls so that personnel are only exposed to the valves and piping associated with one component at any given location. Threshold berms and floor drains are provided to control radioactive leakage. To facilitate decontamination in valve galleries, concrete surfaces are covered with a smooth surfaced coating which will allow easy decontamination.

Piping: Pipes carrying radioactive materials are routed through controlled access areas zoned for that level of activity. Each piping run is analyzed to determine the potential radioactivity level and surface dose rate. Where radioactive piping must be routed through corridors or other low radiation zone areas,
1 shielded pipeways are normally provided. Whenever practicable, valves and instruments are not placed in radioactive pipeways. Whenever practicable, equipment compartments contain only piping associated with equipment in that compartment.

1 Where practicable piping is designed to minimize low points and dead legs. Drains are provided on piping where low points and dead legs cannot be eliminated. Where possible, thermal expansion loops are raised rather than dropped. In radioactive systems, the use of

nonremovable backing rings in the piping joints is minimized to eliminate a potential crud trap for radioactive materials. Piping carrying resin slurries or evaporator bottoms is run vertically as much as possible.

Whenever possible, branch lines having little or no flow during normal operation are connected above the horizontal midplane of the main pipe.

Field Run Piping: All routing of radioactive process piping, large and small, is reviewed by the design engineering office.

Penetrations: To minimize radiation streaming through penetrations, as many penetrations as practicable are located with an offset between the source and the accessible areas. If offsets are not practicable, penetrations are located as far as possible above the floor elevation to reduce the exposure to personnel. If these two methods are not used, then baffle shield walls or grouting the area around the penetration are provided.

Contamination Control: Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination. Equipment vents and drains from certain radioactive systems are piped directly to the collection system instead of allowing any contaminated fluid to flow across to the floor drain. All-welded piping systems are used on contaminated systems to the maximum extent practicable to reduce system leakage and crud buildup at joints. The valves in some radioactive systems are provided with leak-off connections piped directly to the collection system.

Decontamination of potentially contaminated areas within the plant is facilitated by the application of suitable smooth surfaced coatings to the concrete floors and walls.

Floor drains with properly sloping floors are provided in potentially contaminated areas of the plant. In addition, radioactive and potentially radioactive drains are separated from nonradioactive drains.

Systems that become highly radioactive, such as the radwaste slurry transport system, are provided with flush and drain connections. Certain systems have provisions for chemical and mechanical cleaning prior to maintenance.

Equipment Layout: In systems where process equipment is a major radiation source (such as fuel pool cleanup, radwaste, condensate demineralizer, etc), pumps, valves, and instruments are normally separated from the process component. This allows servicing and maintenance of items in reduced radiation zones. Control panels are located in lowest practicable radiation zones.

Major components (such as tanks, demineralizers, and filters) in radioactive systems are isolated in individual shielded compartments insofar as practicable.

Provision is made on some major plant components for removal of these components to lower radiation zones for maintenance.

Labyrinth entrance way shields or shielding doors are provided for compartments from which radiation could stream to access areas and exceed the radiation zone dose limits for those areas. For potentially high radiation components (such as filters and demineralizers), completely enclosed shielded compartments with hatch openings are used.

Equipment in nonradioactive systems that requires lubrication is located outside radiation areas. Wherever practicable, lubrication of equipment in radiation areas is achieved with the use of tube type extensions to reduce exposure during maintenance.

Figures 12.3-1 to 12.3-7 provide layout arrangements for demineralizers, filters, spent resin storage tanks, hydrogen recombiners, sample racks, and their associated valve compartments or galleries.

Exposure from routine in-plant inspection is controlled by locating, whenever possible, inspection points in shielded low background radiation areas. Radioactive and nonradioactive systems are separated as far as practicable to limit radiation exposure from routine inspection of nonradioactive systems. For radioactive systems, emphasis is placed on space and ease of motion in a shielded inspection area. Where longer times for routine inspection are required and permanent shielding is not feasible, sufficient space for portable shielding is normally provided. In high radiation areas where routine surveillance is required, remote viewing devices are provided when practicable.

Facilities for Handling Sealed and Unsealed Radioactive Material: As discussed in Subsection 12.2.1.9, special material used in the radiochemistry laboratory require the design of special handling equipment. For unsealed materials, the following is provided:

a) Exhaust hoods that exhaust to the ventilation system are located in areas such as sample stations and the radiochemistry laboratory.

b) Decontamination facilities, radiochemistry laboratory, controlled zone shop, instrument repair shops and washdown area are situated at various locations in the plant and are described in Subsection 12.5.2.

c) An area for the repair and maintenance of removed control rod drives is provided in the reactor building in close proximity to the control rod drive removal hatch.

12.3.1.3 Radiation Zoning and Access Control

Access to areas inside the plant structures and plant yards is regulated and controlled. Each radiation zone defines the radiation level range to which the aggregate of all contributing sources must be attenuated by shielding.

All plant areas are categorized into radiation zones according to expected radiation levels and anticipated personnel occupancy, with consideration given toward maintaining personnel external exposures ALARA and within the standards of 10CFR20. Each room, corridor, and pipeway of every plant building is evaluated for potential radiation sources during normal operation and shutdown; for maintenance occupancy requirements, and for general access requirements to determine appropriate zoning. Radiation zone categories used and their descriptions are given in Table 12.3-1 and the specific zoning for each plant area is shown on Figures 12.3-8 through 12.3-27. All frequently accessed areas, ie, corridors, are shielded for Zone I and Zone II access.

The control of ingress or egress of plant operating personnel to controlled access areas and procedures employed to ensure that radiation levels and allowable working time are within the limits prescribed by 10CFR20 as described in Section 12.5.

12.3.1.4 Control of Activated Corrosion Products

In order to minimize the radiation exposure associated with the deposition of activated corrosion products in reactor coolant and auxiliary systems, the following steps have been taken:

(1) The reactor coolant system consists mainly of austenitic stainless steel, carbon steel and low alloy steel components. Nickel content of these materials is low, and it is controlled in accordance with applicable ASME material specifications.

A small amount of nickel base material (Inconel 600) is employed in the reactor vessel internal components. Inconel 600 is required where components are attached to the reactor vessel shell and the coefficient of expansion must match the thermal expansion characteristics of the low alloy vessel steel. Inconel 600 was selected because it provides the proper thermal expansion characteristics, adequate corrosion resistance and can be readily fabricated and welded.

(2) Materials employed in the reactor coolant system are purchased to ASME material specification requirements. No special controls on levels of cobalt impurities are specified.

(3) Hardfacing and wear materials having a high percentage of cobalt are restricted to applications where no satisfactory alternate materials are available. Studies currently are being made to determine whether any alternate low cobalt alloys are satisfactory for long term use in nuclear reactor applications. To date, no satisfactory replacement materials have been found.

(4) A high temperature filtration system was not employed in the Reactor Water Clean-up System. The reasons for this included:

a. Lack of quantitative data on the removal efficiency for insoluble cobalt by the high temperature filter;

b. Uncertainty in the deposition model including the relative effectiveness of cobalt removal on deposition rate;

c. Doubtful cost-effectiveness in an area where other methods under study (such as decontamination) may prove better at reducing dose rates while also being more cost-effective.

(5) Items 1, 2, and 3 above also apply to valve materials in contact with reactor coolant. Valve packing materials are selected primarily for their properties in the particular environment.

(6) Subsections 12.1.2.2, 12.3.1.1, and 12.3.1.2 describe the various flushing, draining, testing, and chemical addition connections which have been incorporated into the design of piping and equipment which handle radioactive materials. If decontamination is to be performed, these connections would be used for that purpose.

(7) The plant is designed with a 1% mixed resin, pressure precoat clean-up system for the primary coolant in the reactor and a full flow deep bed condensate demineralizer system for the feedwater. See Figures 10.4-2, 10.4-3, 5.4-16 and 5.4-18.

Water is used as the primary shield material for areas above the spent fuel transfer and storage areas.

Special features employed to maintain radiation exposures ALARA in routinely occupied areas such as valve operating stations and sample stations are described in Subsections 12.3.1.1 and 12.3.1.2.

12.3.2.2.1 Reactor Building Shielding Design

During reactor operation, the steel-lined, reinforced concrete drywell wall and the reactor building walls protect personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and associated equipment within the reactor building. The reactor vessel shield wall, drywell wall, and various equipment compartment walls together with the reactor building walls minimize the radiation levels outside the reactor building.

Where personnel and equipment hatches or penetrations pass through the drywell wall, additional shielding is designed to attenuate the radiation to below the required level defined by the radiation zone outside the drywell wall during normal operation and shutdown and to acceptable emergency levels as defined by 10CFR50 during design basis accidents.

12.3.2.2.2 Reactor Building Interior Shielding Design

Inside Drywell Structure: Areas within the drywell are designed as Zone V areas and are normally inaccessible during plant operation. The reactor vessel shield provides shielding for access in the drywell during shutdown, and reduces the activation of and radiation damage to drywell equipment and materials.

Outside Drywell Structure: The drywell wall is designed to reduce radiation levels in normally occupied areas of the reactor building from sources within the drywell to less than the maximum level for Zone II.

Penetrations and hatch openings in the drywell wall are shielded, as necessary, to meet adjacent area radiation zoning levels. Shielding requirements for the personnel, equipment, and CRD removal hatch openings are shown on Figure 12.3-19 in the areas numbered 412, 413, and 402, respectively. Drywell piping and electrical penetrations are shielded by providing either local shields within the penetration assembly or a shielded penetration room. Shielded piping penetration room locations and bulk shielding requirements are shown on Figures 12.3-18 through 12.3-

20. These rooms, numbered 202, 204, 205; 403, 411; 501, 504, 506, 515; are designated radiation Zone V during reactor power operation and are provided with personnel access controls. Electrical penetrations which are not located within these rooms are provided with supplementary local shielding as needed to meet outside zoning levels. Six inches of lead, in addition to the self-shielding by the electrical cables, is furnished in each electrical penetration assembly to attenuate drywell radiation sources.

The components of the reactor water cleanup (RWCU) system described in Section 5.5 are located in shielded compartments which are designed as Zone V, restricted access areas. Shielding is provided for each piece of equipment in the RWCU system consistent with its postulated maximum activity Subsection 12.2.1 and with the access and zoning requirements of the adjacent areas. This equipment includes:

a) Regenerative heat exchanger

b) Nonregenerative heat exchanger

c) RWCU pumps and piping

d) RWCU filter demineralizers and holdup pumps

e) RWCU backwash receiving tank, pumps, and piping.

The traversing in-core probe (TIP) system is located inside a shielded compartment to protect personnel from the neutron activated portion of the TIP cable.

Main steamlines are located within shielded structures from the drywell wall to the reactor building wall.

Spent fuel is a primary source of radiation during refueling. Because of the extremely high activity of the fission products contained in the spent fuel assemblies and the proximity of Zone II areas, shielding is provided for areas surrounding the fuel transfer canal and pool to ensure that radiation levels remain below zone levels specified for adjacent areas.

After reactor shutdown, the Residual Heat Removal (RHR) System pumps and heat exchangers are in operation to remove heat from the reactor water. It is anticipated that the radiation levels in the vicinity of this equipment will temporarily reach Zone V levels due to corrosion and fission products in the reactor water. Shielding is designed to attentuate radiation from RHR equipment during shutdown cooling operations to levels consistent with the radiation zoning requirements of adjacent areas. Adequate shielding will also be provided to maintain radiation

Water is also used as shielding material above the steam dryer and separator storage area. Concrete walls and water in the pool are designed to provide Zone II dose rates in adjacent accessible | 1
areas during storage of the dryer and separator.

The Fuel Pool Cooling and Cleanup (FPCC) System (See Section 9.1) shielding is based on the maximum activity discussed in Subsection 12.2.1 and the access and zoning requirements of adjacent areas. Equipment in the FPCC system to be shielded includes the FPCC heat exchangers, pumps and piping, filter | 15
demineralizers, and backwash receiving tank.

12.3.2.2.3 Radwaste Building Shielding Design

Shielding is provided as necessary around the following equipment in the radwaste building to ensure that the radiation zone and access requirements are met for surrounding areas.

a) Laundry drain tank and pumps

b) Chemical waste tank and pumps

c) Radwaste evaporators

d) Radwaste evaporator tanks and pumps

e) Liquid radwaste collection tanks and pumps

f) Liquid radwaste surge tanks

g) Liquid radwaste sample tanks and pumps

h) Reactor water cleanup phase separator and pumps

i) Waste sludge phase separator and pumps

j) Spent resin tank

k) Waste filling and capping station

l) Waste drum transfer and storage areas

m) Liquid radwaste demineralizer and piping

n) Waste mixing tanks

o) Liquid radwaste filters

p) Gaseous radwaste equipment.

12.3.2.2.4 Turbine Building Shielding Design

Radiation shielding is provided around the following equipment in the turbine building to ensure that zone access requirements (Figures 12.3-10 through 12.3-15) are met for the following surrounding areas:

a) Condensate filter demineralizers and piping

b) Regeneration waste surge tanks and pumps

c) Chemical waste neutralizing tanks and pumps

d) Reactor feed pump turbines and piping

e) Condensate pumps and piping

f) Main condensers and hotwell

g) Mechanical vacuum pump

h) Recombiners and piping

i) Steam packing exhauster

j) Condensate demineralizer resin regeneration tanks

k) Air ejectors and gland steam condensers

l) Feedwater heaters, heater drains, and piping

m) Main steam piping

n) Steam seal evaporator and drain tank

o) Moisture separator and drain tanks

p) High pressure and low pressure turbines

q) Offgas piping.

Areas within most of these shield walls have high radiation levels and limited access.

12.3.2.2.5 Control Room Shielding Design

Figures 12.3-9 and 12.3-28 represent layout and isometric drawings of the control room, showing its relationship to the reactor building.

The design basis loss-of-coolant accident (LOCA) dictates the shielding requirements for the control room. Shielding is provided to permit access and occupancy of the control room under LOCA conditions with radiation doses limited to 5 rem whole body from all contributing modes of exposure for the duration of the accident, in accordance with 10CFR50 Appendix A, General Design Criterion 19.

The design basis LOCA is described in Subsection 15.1.13 and is based on Regulatory Guide 1.3. The direct radiation from airborne fission products inside the reactor building would contribute less than 105 mrem to personnel inside the control room for the 30-day period following the LOCA, based on radioactivity sources described in Subsection 12.2.1.6.

The parameters used in the demonstration of the control room habitability are listed below and in Regulatory Guide 1.3. (The ventilation system parameters are listed in Subsection 12.3.3).

For all isotopes that escape from the drywell to the reactor building, no credit is taken for shielding by the internal structures in the reactor building. Shielding credit is taken for the reactor and control building walls. For all isotopes that remain within the drywell, shielding credit is taken for the drywell wall.

12.3.2.2.6 Diesel Generator Building Shielding Design

There are no radiation sources in the diesel generator building; therefore, no shielding is required for the building.

12.3.2.2.7 Miscellaneous Plant Areas and Plant Yard Areas

Sufficient shielding is provided for all plant buildings containing radiation sources so that radiation levels at accessible areas outside buildings are minimized. Plant yard areas which are frequently occupied by plant personnel are accessible during normal operation and shutdown. These areas are surrounded by a security fence and closed off from areas accessible to the general public.

12.3.2.2.8 Counting Room Shielding

Because the counting room contains sensitive instruments to radioactivity measurements, it is imperative that the background radiation levels are minimized. To accomplish this, no flyash was used in the concrete mix for the walls and slabs surrounding the counting room. Flyash normally contains a relatively large amount of slow decaying radioactive isotopes. In addition, the shield walls and slabs were sized to maintain a background radiation level of less than 130 mrem/year for anticipated operational occurrences and 45 mrem/year for normal operation.

12.3.2.3 Shielding Calculational Methods

The shielding thicknesses provided to ensure compliance with plant radiation zoning and to minimize plant personnel exposure are based on maximum equipment activities under the plant operating conditions described in Subsection 12.2.1. The thickness of each shield wall surrounding radioactive equipment is determined by approximating as closely as possible the actual geometry and physical condition of the source or sources. The isotopic concentrations are converted to energy group sources using data from standard Refs 12.3-1 through 12.3-5.

The geometric model assumed for shielding evaluation of tanks, heat exchangers, filters, demineralizers, and evaporators is a finite cylindrical volume source. For shielding evaluation of piping, the geometric model is a finite shielded cylinder. In cases where radioactive materials are deposited on surfaces such as pipe, the latter is treated as an annular cylindrical surface source. Typical computer codes that are used for shielding analysis are listed in Table 12.3-2. Shielding attenuation data used in those codes include gamma class attenuation coefficients (Ref. 12.3-6), gamma buildup factors (Ref 12.3-7), neutron-gamma multigroup cross sections (Ref 12.3-20), and albedos (Ref 12.3-12).

The shielding thicknesses are selected to reduce the aggregate computed radiation level from all contributing sources below the upper limit of the radiation zone specified for each plant area. Shielding requirements are evaluated at the point of maximum radiation dose through any wall. Therefore, the actual anticipated radiation levels in the greater region of each plant area is less than this maximum dose and therefore less than the radiation zone upper limit.

Where shielded entryways to compartments containing high radiation sources are necessary, labyrinths or mazes are designed using a general purpose gamma-ray scattering code G-33 (Ref. 12.3-11). The mazes are constructed so that the scattered dose rate plus the transmitted dose rate through the shield wall from all contributing sources is below the upper limit of the radiation zone specified for each plant area.

12.3.3 VENTILATION

The plant heating, ventilating, and air conditioning (HVAC) systems are designed to provide a suitable environment for personnel and equipment during normal operation and anticipated operational occurrences. Parts of the plant HVAC systems perform safety related functions.

12.3.3.1 Design Objectives

The systems are designed to operate such that the in-plant airborne activity levels for normal operation (including anticipated operational occurrences) in the general personnel access areas are within the limits of 10CFR20. The systems operate to reduce the spread of airborne radioactivity during normal and anticipated abnormal operating conditions.

During post accident conditions the ventilation system for the plant control room provides a suitable environment for personnel and equipment and ensures continuous occupancy in this area. The plant ventilation systems are designed to comply with the airborne radioactivity release limits for offsite areas during normal operation.

12.3.3.2 Design Criteria

Design criteria for the plant HVAC systems include the following:

a) During normal operation and anticipated operational occurrences, the average and maximum airborne radioactivity levels to which plant personnel are exposed in restricted areas of the plant is ALARA and within the limits specified in 10CFR20. The average and maximum airborne radioactivity levels in unrestricted areas of the plant during normal operation and anticipated operational occurrences will be ALARA and within the limits of Appendix B, Table II of 10CFR20.

b) During normal operation and anticipated operational occurrences, the dose from concentrations of airborne radioactive material in unrestricted areas beyond the site boundary will be ALARA and within the limits specified in 10CFR20 and 10CFR50.

c) The plant siting dose guidelines of 10CFR100 will be satisfied following those hypothetical accidents, described in Chapter 15, which involve a release of radioactivity from the plant.

d) The dose to control room personnel shall not exceed the limits specified in General Design Criterion 19 of Appendix A to 10CFR50 following those hypothetical accidents, described in Chapter 15, which involve a release of radioactivity from the plant.

12.3.3.3 Design Guidelines

In order to accomplish the design objectives, the following guidelines are followed wherever practicable.

12.3.3.3.1 Guidelines to Minimize Airborne Radioactivity

a) Access control and traffic patterns are considered in the basic plant layout to minimize the spread of contamination.

b) Equipment vents and drains are piped directly to a collection device connected to the collection system instead of allowing any contaminated fluid to flow across the floor to the floor drain.

c) All-welded piping systems are used on contaminated systems to the maximum extent practicable to reduce system leakage. If welded piping systems are not used, drip trays are provided at the points of potential leakage. Drains from drip trays are piped directly to the collection system.

d) The valves in some systems are provided with leak-off connections piped directly to the collection system.

e) Suitable coatings are applied to the concrete floors of potentially contaminated areas to facilitate decontamination.

fluids. A stuffing box is also provided with a leak-off connection that may be piped to a drain header. Where practicable, metal diaphragm or bellows seat valves are used on those systems where essentially no leakage can be tolerated.

g) Contaminated equipment has design features that minimize the potential for airborne contamination during maintenance operations. These features may include flush connections on pump casings for draining and flushing the pump prior to maintenance or flush connections on piping systems that could become highly radioactive.

h) Exhaust hoods are used in plant areas to facilitate processing of radioactive samples by drawing contaminants away from the personnel breathing areas and into the ventilation and filtering systems.

i) Equipment decontamination facilities are ventilated to ensure control of released contamination and minimize personnel exposure and the spread of contamination.

12.3.3.3.2 Guidelines to Control Airborne Radioactivity

a) The airflow is directed from areas with lesser potential for contamination to areas with greater potential for contamination under normal conditions.

b) In building compartments with a potential for contamination, a greater volumetric flow is exhausted from the area than is supplied to the area to minimize the amount of uncontrolled exfiltration from the area.

c) Floor and equipment drain collector tank vents are piped to a collection header and processed by the tank vent filter system.

d) Exhaust air is routed through prefilter and HEPA filters or a combination of prefilter, HEPA and charcoal filters where necessary before release to the atmosphere to reduce onsite and offsite airborne concentrations.

e) Air is supplied to each principal building via separate supply intakes and duct systems.

f) Redundant, Seismic Category I systems and components are provided for portions of the ventilation system required for safe shutdown of the reactor and to mitigate the consequences of design basis accidents. Included herein as the plant control room ventilation system, the reactor building recirculation system, the standby gas treatment system, and coolers are selected engineered safety feature equipment rooms.

g) Air being discharged from potentially contaminated areas of the Turbine Building and the Reactor Building is passed through prefilters, HEPA filters and charcoal adsorbing filters. Air being discharged from the Radwaste Building is passed through prefilters and HEPA filters. Means are provided to isolate these areas upon indication of contamination to prevent the discharge of contaminants to the environment.

h) Suitable containment isolation valves are installed in accordance with General Design Criteria 54 and 56, including valve controls, to ensure that the containment integrity is maintained. See additional discussion in Subsections 3.1.2.5.5, 3.1.2.5.7 and 6.2.4.

12.3.3.3.3 Guidelines to Minimize Personnel Exposure from HVAC Equipment

a) Ventilation fans and filters are provided with adequate access space to permit servicing with minimum personnel radiation exposure. The HVAC system is designed to allow rapid replacement of components.

b) Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts to the maximum extent practicable. Welded seams are used to join ductwork segments and internal obstructions are avoided wherever practicable.

c) Access and service of ventilation systems in potentially radioactive areas are provided by component location to minimize operator exposure during maintenance, inspection, and testing as follows:

1) The outside air supply units and building exhaust system components are enclosed in ventilation equipment rooms. These equipment rooms are located

unit for anticipated maintenance, testing, and inspection. Filter-adsorber units generally comply with the access and service requirements of Regulatory Guide 1.52. (Refer to response to Regulatory Guide 1.52 in Section 3.13).

Local cooling equipment, servicing the building requirements, will normally be located in areas of low contamination potential radiation Zones I or II.

d) While the majority of the activity in the filter train is removed by simply removing the contaminated filters, further decontamination of the internal structure is facilitated by the proximity of electrical outlets for operation of decontamination equipment, and water supply for washdown of the interior, if necessary. Drains are provided on the filter housing for removal of contaminated water.

e) Active elements of the atmospheric cleanup systems are designed to permit easy removal.

f) Access to active elements is direct from working platforms to simplify element handling. Space is provided on the platforms for accommodating safe personnel movement during replacement of components, including the use of necessary material-handling facilities and during any in-place testing operations.

g) The clear space for doors is a minimum of 3 ft by 7 ft.

h) The filters are designed with replaceable 2 ft by 2 ft units that are clamped in place against compression seals. The filter housing is designed, tested, and proven to be airtight with bulkhead type doors that are closed against compression seals.

i) Filter systems in which radioactive materials could accumulate to produce signficant radiation fields external to the ductwork are appropriately located and shielded to reduce exposure to personnel and equipment.

j) Filters in all systems are changed based upon the airflow and the pressure drop across the filter bank. Charcoal adsorbers are changed based on the residual adsorption capacity of the bed as measured by the testing of carbon samples taken from the removable cannisters located in the carbon bed. The testing of the carbon adsorbers and all other components is described in Table 9.4-1.

12.3.3.4 Design Description

The ventilation systems serving the following structures are assumed to be potentially radioactive and are discussed in detail in Section 9.4.

a) Reactor building

b) Radwaste building

c) Turbine building.

Although the control room is considered to be a nonradioactive area, radiation protection is provided to ensure habitability (see Section 6.4).

Ventilation system design parameters for the four systems are given in Tables 12.3-3 through 12.3-6.

A typical layout of a potentially radioactive filter unit is given on Figure 12.3-3.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

12.3.4.1 Area Radiation Monitoring

The area radiation monitoring system supplements the personnel and area radiation survey provisions of the plant health physics program described in Section 12.5 to ensure compliance with the personnel radiation protection guidelines of 10CFR20, 10CFR50, 10CFR70, and Regulatory Guides 1.21, 8.2, 8.8 and 8.12 as discussed below.

The area radiation monitors function to:

a) Alert plant personnel of abnormally high radiation levels which, if unnoticed, could result in inadvertent exposures.

b) Inform the control room operator of the occurrence and approximate location of abnormal radiation level increase.

c) Comply with the requirements of 10CFR50 Appendix A, General Design Criterion 63 for monitoring fuel and waste storage and handling areas.

d) Assist in the detection of unauthorized or inadvertent movement of radioactive material in the plant.

e) Supplement other systems including process radiation monitoring, leak detection, etc., in detecting abnormal migrations of radioactive material in or from the process streams.

The area radiation monitoring system has no function related to the safe shutdown of the plant, or to the quantitative monitoring of the release of radioactive material to the environment.

12.3.4.1.1 Criteria for Area Monitor Selection

The following design criteria are applicable to the area radiation monitoring system.

Energy Dependence: The detector-indicator and trip unit should be responsive to gamma radiation over an energy range of 80 KeV to 7 MeV. The energy dependence should not exceed ±20 percent of point for an exposure rate of approximately 50 mr/hr from 100 KeV to 3 MeV and there should be response from 80 KeV to 7 KeV.

Accuracy: The overall accuracy within the design range of temperature, humidity, line voltage, and line frequency variation should be such that the actual reading relative to the true reading, including susceptibility and energy dependence (100 KeV to 3 MeV), should be within 9.5 percent of equivalent linear full scale recorder output for any decade.

Reproducibility: At design center the reading shall be reproducible within ±10 percent of point with constant geometry.

Environmental Conditions

	Sensor Location		Control Room	
Parameter	Design Center	Range	Design Center	Range
Temperature (degrees C)	25	0 to 60	25	5 to 50
Relative Humidity (Percent)	50	20 to 100	50	20 to 90

12.3.4.1.2 Criteria for Location of Area Monitors

Generally, area radiation monitors are provided in areas to which personnel normally have access and for which there is a potential for personnel unknowingly to receive high radiation doses (e.g., in excess of 10CFR20 limits) in a short period of time because of system failure or improper personnel action. Any plant area that meets one or more of the following criteria is monitored:

a) Zone I areas which, during normal plant operation, including refueling, could exceed the radiation limit of 0.5 mrem/hr upon system failure or personnel error or which will be continuously occupied following an accident requiring plant shutdown

b) Zone II areas where personnel could otherwise unknowingly receive high levels of radiation exposure due to system failure or personnel error

c) Area monitors are in accordance with General Design Criterion 63 of 10CFR50 Appendix A.

12.3.4.1.3 System Description (Area Radiation Monitoring)

General

The area radiation monitoring system is shown in diagram form in Figure 12.3-29. Each channel consists of a combined sensor/converter unit, a local auxiliary unit (readout with visual and audible alarm), a combined indicator/trip unit, a shared power supply, and a shared multipoint recorder. The location of each area radiation detector is indicated on the shielding and zoning drawings, Figures 12.3-8, 12.3-27, and is listed in Table 12.3-7.

Circuit Description

Sensor/Converter: Each sensor/converter contains all silicon semiconductors in sealed enclosure with a Cooke-Yarborough courtyard circuit which combines a long integrating time constant at low radiation levels with fast overall response at high radiation levels.

Auxiliary Unit: Each auxiliary unit gives instant local readout at the sensor location with a visual alarm. An audible alarm is connected to the auxiliary unit to alert personnel of excessive area radiation.

Indicator and Trip Unit: The indicator and trip unit provides channel control for the area radiation monitoring system. Its circuitry provides an upscale trip that indicates high radiation and a downscale trip that may indicate instrument trouble or loss of power. The module has an analog readout, a low and high trip indicating light, a trip test device, an alarm reset and an output for a multipoint recorder.

Ranges and Sensitivity: Ranges and sensitivities are selected for each location based on the anticipated radiation level as provided by experimental measurments of levels in similar plants and shielding calculations. Refer to Table 12.3-7 for detail.

Accuracy: The overall accuracy is such that the actual reading relative to the true reading is within ±7.5 percent of equivalent full scale.

12.3.4.1.4 Area Radiation Monitoring Instrumentation

Power Sources: The power source for the area radiation monitoring system is the 120 V ac instrument bus. The area radiation monitor instrumentation is powered by a high and low voltage electrically regulated power supply capable of handling up to 10 channels. The system has no emergency power supply.

Alarm Set Points: Refer to Table 12.3-7.

Recording Devices: One 40-channel multipoint recorder is located in the system cabinet for Unit 1 and common circuits. One 40-channel multipoint recorder is provided for Unit 2.

Location of Devices: Refer to details in Table 12.3-7.

Readouts and Alarms: Local readouts, visual and audible alarms are provided for each monitoring channel. The indicator/trip units and the multipoint recorders located in the system cabinet in the control structure (Upper Relay Room).

The following annunciators are located in the main control room to alert the operator:

a Reactor Building Area High Radiation (Units 1 and 2)

b Turbine Building Area High Radiation (Units 1 and 2)

c Radwaste Building Area High Radiation

d Refueling Floor Area High Radiation (Units 1 and 2)

e Spent Fuel Pool Area High Radiation (Units 1 and 2)

f Reactor Building Common Area High Radiation q Administration Building Area High Radiation

h Control Structure Area High Radiation

i Area Radiation Monitoring Downscale (for any sensor)

12.3.4.1.5 Safety Evaluation

The area radiation monitoring system is designed to operate unattended for extended periods and is designed for high reliability. Failure of one monitor has no effect on any other.

The system is not essential for safe shutdown of the plant, and serves no active emergency function during operation. The system is not safety related and is constructed to Quality Group D Requirements.

12.3.4.1.6 Calibration Method and Testability

Facilities for calibrating these monitor units are provided, which include a test unit designed for use in the adjustment procedure for the area radiation monitor sensor and converter unit. These provide several gamma radiation levels between 1 and 250 mrem/hr. The calibration unit source is Co^{60}.

A cavity in the calibration unit receives the sensor and converter unit. A window through which radiation from the source emanates is located on the back wall of the cylindrical lower half of the cavity. A chart on each calibration unit indicates the radiation levels available from the unit for the various control settings.

An internal trip test circuit, adjustable over the full range of the trip circuit, is provided. The test signal is fed into the indicator and trip unit input so that a meter reading is provided in addition to a real trip. All trip circuits are the latching type and must be manually reset at the front panel in the control room.

The radiation monitors will be calibrated at regular time intervals in accordance with station procedures.

12.3.4.2 Airborne Radioactivitiy Monitoring

Refer to Subsections 12.5.2.6.3 and 12.5.3.5.4 for information on air borne radioactivity monitoring.

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TABLE 12.3-2

LIST OF COMPUTER CODES USED IN SHIELDING DESIGN CALCULATIONS

GRACE 1:	Multigroup, multiregion, gamma-ray attentuation code used to compute gamma heating and gamma dose rates in slab geometry (Ref 12.3-13).
GRACE 2:	Multigroup, multiregion, gamma-ray attentuation code used to compute the dose rate or heat generation rate for a spherical or a cylindrical source with slab or concentric shields (Ref 12.3-14).
ANISN:	Multigroup, multiregion code solving the Boltzman transport equation for neutrons or gamma-rays in one dimension slab, cylindrical, or spherical geometry (Ref 12.3-8).
SDC:	Multigroup, multiregion, Kernal integration gamma-ray, shield design code which calculates dose rates for 13 geometry options (Ref 12.3-10).
QAD:	Multigroup, multiregion, three-dimensional, point Kernal code which calculates fast neutron and gamma-ray dose and heat generation rates (Ref 12.3-9).
NAP:	Determines neutron activation and gamma emission source strengths as a function of neutron exposure and decay time (Ref 12.3-15).
MORSE-CG:	Three-dimensional Monte Carlo neutron and gamma ray general transport code (Ref 12.3-16).
DOT III:	Two-dimensional neutron, gamma ray, discrete ordinate, transport code (Ref 12.3-17).
ORIGIN:	Isotope generation and depletion code which solves equations of radioactive growth and decay for isotopes of arbitrary coupling (Ref 12.3-19).
G^3:	A general purpose gamma-ray scattering code (Ref 12.3-11).

TABLE 12.3-4

TURBINE BUILDING VENTILATION SYSTEM DESIGN FEATURES

Area	Radiological Safety Features	Exhaust Air Flow Rate (cfm)
General Personnel Access Areas	Two 100% supply fans, two 100% exhaust fans.	40,000[1]
Equipment Areas	Two 100% exhaust fans. Exhausts from all turbine areas are continuously filtered. All process valves 2 1/2 in. and larger are connected to a valve stem leak-off collection system.	40,000[1]

(1) Total normal exhaust from turbine building is 40,000 cfm.

TABLE 12.3-6

CONTROL ROOM VENTILATON SYSTEM DESIGN FEATURES

Operation Mode	Radiological Safety Features	Air Flow Rate (cfm)	Exposure to Airborne Concentrations
Normal	Two 100% supply recirculation fans	26,000	Background
Accident	Two 100% supply fans. Automatic/ manual switch to emergency intake and filtering system on high activity signal, manual recirculation and filterin on high chlorine signal.		Less than allowable limits set in Code of Federal Regulations. Whole body < 5 rem Thyroid < 30 rem

TABLE 12.3-7

AREA RADIATION MONITORING SYSTEM UNIT 1

Channel No.	Monitor Description	Bldg.	Approx. Loc.	Elev.	Range (mR/hr)	Set Point (mR/hr)
1	Chan 1 RX Bldg Residual heat removal area	RB	T/22	645'	0.1-1000	200
2	Chan 2 RX Bldg RCIC pump turbine room	RB	T/21	645'	0.01-100	30
3	Chan 3 RX Bldg HPCI pump turbine room	RB	S/21	645'	0.01-100	30
4	Chan 4 RX Bldg Radwaste sump area	RB	S/28	645'	0.1-1000	200
5	Chan 5 RX Bldg Contr. rod drive Hyd. Units north	RB	R/21	719'	0.1-1000	100
6	Chan 6 RX Bldg Contr. rod drive Hyd. Units south	RB	R/29	719'	0.1-1000	100
7	Chan 7 spare					
8	Chan 8 RX Bldg. Cleanup recirc. pump access area	RB	R/21	749'	0.01-100	5
9	Chan 9 RX Bldg Penetration room	RB	T/25	749'	0.1-1000	200
10	Chan 10 RX Bldg Fuel pool pump room	RB	Q/28	749'	0.1-1000	100
11	Chan 11 RX Bldg Refueling control area	RB	Q/27	779'	0.01-100	5
12	Chan 12 RX Bldg Recirculation fan room	RB	U/27	799'	0.01-100	5
13	Chan 13 RX Bldg NW Fuel Pool	RB	Q/27	799'	0.01-100	5

TABLE 12.3-7 (Continued)

AREA RADIATION MONITORING SYSTEM UNIT 1

Channel No.	Monitor Description	Bldg.	Approx. Loc.	Elev.	Range (mR/hr)	Set Point (mR/hr)
14	Chan 14 RX Bldg Spent fuel pool	RB	S/27	818'	0.1-1000	50
15	Chan 15 RX Bldg refueling floor area	RB	M/25	818'	0.01-100	5
16	Chan 16 RX Bldg Access to remote shutdown panel	RB	P/21	670'	0.01-100	5
17	Chan 17 TB Bldg Condensate pumps area	TB	J/26	656'	0.01-100	5
18	Chan 18 TB Bldg RFPT area	TB	L/21	676'	0.01-100	5
19	Chan 19 TB Bldg Air ejector room	TB	H/25	676'	0.1-1000	300
20	Chan 20 TB Bldg Feedwater heater area	TB	L/20	699'	0.1-1000	100
21	Chan 21 TB Bldg Reactor recirc pump M.G. area	TB	M/20	729'	0.01-100	5
22	Chan 22 TB Bldg generator bay area	TB	G/26	729'	0.01-100	5
23	Chan 23 TB Bldg Heat and vent. equipment room	TB	L/23	762'	0.01-100	5
24	Chan 24 TB Bldg Turbine front end	TB	G/16	729'	0.01-100	5
25	Chan 25 RX Bldg Residual heat removal area	RB	T/22	645'	0.1-1000	200
26	Chan 26 RX Bldg TIP drive area	RB	Q/22	719'	0.1-1000	200

TABLE 12.3-7 (Continued)

AREA RADIATION MONITORING SYSTEM UNIT 1

Channel No.	Monitor Description	Bldg.	Approx. Loc.	Elev.	Range (mR/hr)	Set Point (mR/hr)
27	Chan 27 Admin Bldg Lobby	ADM BLDG	P/11	676'	0.01-100	5
28	Chan 28 Admin Bldg Corr. 2nd floor	ADM BLDG	P/11	691'	0.01-100	5
29	Chan 29 RW Bldg Corridor pers. access area	RW	K/3	646'	0.1-1000	100
30	Chan 30 RW Bldg Opt. surveillance control area	RW	G/8	646'	0.1-1000	100
31	Chan 31 RW Bldg Corridor to collection tank	RW	J/12	646'	0.1-1000	100
32	Chan 32 RW Bldg Controlled zone shop	RW	K/12	676'	0.1-1000	100
33	Chan 33 RW Bldg Corr. to storage equipment area	RW	J/9	676'	0.1-1000	100
34	Chan 34 RW Bldg Storage and equipment area	RW	G/6	676'	0.1-1000	100
35	Chan 35 RX Bldg Shipping cask storage area	RB	S/29	818'	0.01-100	5
36	Chan 36 RX Bldg Railroad access area	RB	U/29	670'	0.01-100	5
37	Chan 37 Ctr. Twr. Standby gas treatment room	CTR TWR	R/27	806'	0.01-100	5
38	Chan 38 Ctr. Twr. Rad. chem. laboratory	CTR TWR	M/27	676'	0.01-100	5

TABLE 12.3-7 (Continued)

AREA RADIATION MONITORING SYSTEM UNIT 1

Channel No.	Monitor Description	Bldg.	Approx. Loc.	Elev.	Range (mR/hr)	Set Point (mR/hr)
39	Chan 39 Ctr. Twr. Control room	CTR TWR	L/29	729'	0.01-100	5
40	Chan 40 Admin Bldg Access to turb bldg	ADM BLDG	N/12	676'	0.01-100	5

TABLE 12.3-7 (Continued)

AREA RADIATION MONITORING SYSTEM UNIT 2

Channel No.	Monitor Description	Bldg.	Approx. Loc.	Elev.	Range (mR/hr)	Set Point (mR/hr)
1	Chan 1 RX Bldg Residual heat removal area	RB	T/31	645'	0.1-1000	200
2	Chan 2 RX Bldg RCIC pump turbine room	RB	T/30	645'	0.1-1000	30
3	Chan 3 RX Bldg HPCI pump and turbine room	RB	S/30	645'	0.1-1000	30
4	Chan 4 RX Bldg Radwaste sump area	RB	S/36	645'	0.1-1000	200
5	Chan 5 RX Bldg Contr. rod drive Hyd. Units north	RB	R/30	719'	0.1-1000	100
6	Chan 6 RX Bldg Contr. rod drive south	RB	R/37	719'	0.1-1000	100
7	Chan 7 Spare					
8	Chan 8 RX Bldg Cleanup recirc pump access area	RB	R/37	749'	0.01-100	5
9	Chan 9 RX Bldg Penetration room	RB	T/33	749'	0.1-1000	200
10	Chan 10 RX Bldg Fuel pool pump room	RB	Q/30	749'	0.1-1000	100
11	Chan 11 RX Bldg Refueling control area	RB	Q/31	779'	0.01-100	5
12	Chan 12 Spare					
13	Chan 13 RX Bldg NW Fuel Pool	RB	Q/31	799'	0.01-100	5

TABLE 12.3-7 (Continued)

AREA RADIATION MONITORING SYSTEM UNIT 2

Channel No.	Monitor Description	Bldg.	Approx. Loc.	Elev.	Range (mR/hr)	Set Point (mR/hr)
14	Chan 14 RX Bldg Spent fuel pool	RB	S/31	818'	0.1-1000	50
15	Chan 15 RX Bldg Refueling floor area	RB	M/33	818'	0.01-100	5
16	Chan 16 RX Bldg Access to remote shutdown panel	RB	P/37	670'	0.01-100	5
17	Chan 17 TB Bldg Condensate pumps area	TB	J/31	656'	0.01-100	5
18	Chan 18 TB Bldg RPPT area	TB	L/36	676'	0.01-100	5
19	Chan 19 TB Bldg Air ejector room	TB	H/32	682'	.1-10,000	300
20	Chan 20 TB Bldg Feedwater heater area	TB	L/38	699'	0.1-1000	100
21	Chan 21 TB Bldg Reactor recirc. pump M.G. area	TB	M/38	729'	0.01-100	5
22	Chan 22 TB Bldg Generator bay area	TB	G/32	729'	0.01-100	5
23	Chan 23 TB Bldg Heat and vent. equipment room	TB	L/35	762'	0.01-100	5
24	Chan 24 TB Bldg Turbine front end	TB	G/42	729'	0.01-100	5
25	Chan 25 RX Bldg Residual heat removal area	RB	T/33	645'	0.1-1000	200
26	Chan 26 RX Bldg Tip drive area	RB	Q/31	719'	1-10,000	200

Note: All set points are estimated values. Actual set points may vary depending on operational considerations and will be determined by measured radiation levels.

CHAPTER 13.0

CONDUCT OF OPERATIONS

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TABLE 13.1-1

KEY TECHNICAL SUPPORT PERSONNEL RESUMES

Positions

Senior Vice President - Nuclear
Vice President - Nuclear Operations
Vice President - Engineering and Construction - Nuclear
Manager - Nuclear Plant Engineering
Manager - Nuclear Licensing
Manager - Nuclear Fuels
Manager - Nuclear Training
Manager - Nuclear Support
Manager - Nuclear Administration
Manager - Nuclear Safety Assessment
Manager - Nuclear Quality Assurance
Construction Manager
Manager - Procurement

Table 13.1-1 (Cont.)

RESUME: JACK R. CALHOUN

PRESENT POSITION:

Senior Vice President - Nuclear
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, Pa.

EDUCATION:

Tennessee Technological University
BS Electrical Engineering

ADDITIONAL MISC. COURSES:

Oak Ridge School of Reactor Technology (ORSORT)

GENERAL:

Member - American Nuclear Society (ANS)
Past National Chairman, Reactor Operation Division ANS

Consultant to Argonne Universities Association for reactor operations associated with the Experimental Breeder Reactor, Idaho Falls, Idaho

Past Advisor to Nuclear Engineering Department Pennsylvania State University

EXPERIENCE:

1980 - Present — Pennsylvania Power & Light Company
Senior Vice President - Nuclear

1949 - 1980 — Tennessee Valley Authority

1979-1980 Director of Nuclear Power - directed activities of operation and maintenance of all TVA nuclear facilities
1977-1979 Assistant Director of Power Production - responsible to Director of Power Production for activities of operating and maintenance for all TVA nuclear facilities
1971-1977 Chief, Nuclear Generation Branch - responsible for operation and maintenance of all TVA nuclear facilities
1968-1971 Power Plant Superintendent, Browns Ferry Nuclear Plant (3-1098 MW GE Reactors)
1964-1968 Assistant Chief, Power Plant Maintenance

Table 13.1-1 (Cont.)

		Branch, Division of Power Production - responsible for initial operation and maintenance planning for Browns Ferry Nuclear Plant
	1960-1964	Experimental Gas-Cooled Reactor Operations Superintendent and Assistant Project Manager, (6 months training on Berkeley Nuclear Power Station, Bristol, England)
	1958-1960	Assistant Power Plant Superintendent Shawnee Steam Plant (10-150 MW units)
	1954-1958	Electrical Maintenance Supervisor Johnsonville Steam Plant (6-135 MW units)
	1949-1954	Student Generating Plant Operator - Operator, Watts Bar Steam Plant (4-60 MW units)
1938 - 1945	U.S. Navy - 4 years enlisted service as an electrician mate on a Destroyer, 2 years as the Main Propulsion Electrical Officer on an Aircraft Carrier, and 2 years as the Electrical Officer on a Cruiser	

Table 13.1-1 (Cont.)

RESUME: BRUCE D. KENYON

PRESENT POSITION:

Vice President-Nuclear Operations
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, Pa.

EDUCATION:

Miami University
BA Mathematics

ADDITIONAL MISC. COURSES:

US Navy Submarine Officer Nuclear Training Program

Completed all training which resulted in a senior operator qualification on USS George Washington (SSBN 598)(W PWR).

Completed all training which resulted in a senior operator qualification at the Navy D1G Nuclear Prototype (GE PWR).

Completed all training which resulted in an NRC Senior Operators License on Millstone Unit #1 (GE BWR). This training included four months of classroom training and attendance at the GE BWR Simulator, Morris Illinois.

Completed all training which resulted in a NRC Senior Operators License on Millstone Unit #2 (CE PWR). This training included four months of classroom training, certification at the Westinghouse Simulator and attendance at the CE Simulator.

GENERAL:

Member EEI Nuclear Operations Committee

Past Member EEI Construction Committee

Member AIF Design, Construction, and Operations Committee

Member INPO Analysis Division - Industry Review Committee

Table 13.1-1 (Cont.)

Past Member EPRI Engineering and Operations Committee

EXPERIENCE:

1976 - Present	Pennsylvania Power & Light Company	
	1980-Present	Vice President-Nuclear Operations
	1979-1980	Assistant Vice President-Nuclear
	1978-1979	Construction Manager
	1976-1978	Manager-Nuclear Support
1970 - 1976	Northeast Utilities. Served successively as Startup Engineer, Startup Supervisor, Operations Supervisor, and Unit Superintendent on Millstone Unit #2. Responsibilities included preparation of test procedures, development and administration of Startup Test Program Development of plant operating procedures and management of Unit activities from early stages of constructon completion through commercial operation.	
1965 - 1970	U.S. Navy. Completed nuclear power training program. Served as Division Officer in various engineering and operations positions on the USS George Washington (SSBN 598). Also served as Leading Engineering Watch Officer at the DIG Prototype.	

Table 13.1-1 (Cont.)

RESUME: NORMAN W. CURTIS

PRESENT POSITION:

Vice President-Engineering and Construction-Nuclear
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA.

EDUCATION:

University of Maine
B.S. Engineering Physics

ADDITIONAL COURSES:

Executive Program - Columbia University -
Graduate School of Business Administration

GENERAL:

Registered Professional Engineer-Pennsylvania

Charter Member American Nuclear Society

Member-IEEE

Past Member - EEI
Construction Committee

EXPERIENCE:

1950-Present Pennsylvania Power & Light Company

1980-Present	V.P.-Engineering & Construction-Nuclear
1974-1980	V.P.-Engineering & Construction
1973-1974	Manager-Engineering & Construction
1972-1973	Construction Manager
1970-1972	Manager Power Supply
1965-1970	Superintendent System Operation
1954-1965	Project Engineer, Senior Project Engineer - Atomic Power Department
1950-1954	Helper, Wireman, Foreman - Construction Construction Department

Table 13.1-1 (Cont.)

RESUME: EARLE M. MEAD

PRESENT POSITION:

Manager-Nuclear Plant Engineering
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA

EDUCATION:

Bucknell University
BS Electrical Engineering

ADDITIONAL MISC. COURSES:

Muhlenberg College - Principals of Atomic Power
Courses in Business Management

Penn State Extension - Engineering Courses

University of Michigan - Management Courses

GENERAL:

Registered Professional Engineer - Commonwealth of Pennsylvania

Member - Institute of Electrical and Electronics Engineers

Member - American Society of Mechanical Engineers

Member - American Nuclear Society

EXPERIENCE:

1954 - Present Pennsylvania Power & Light Company

1978-Present	Manager-Nuclear Plant Engineering
1976 - 1978	Project Engineering Manager-Susquehanna
1974 - 1976	Project Manager - Susquehanna
1971 - 1974	Nuclear Plant Design Engineer
1966 - 1971	Senior Project Engineer - Atomic
1963 - 1966	Reactor Plant Engineer (Test Engineer at Peach Bottom Atomic Power Station
1954 - 1966	Various Positions including: Project Engineer, Engineer, and Assistant Salary Administrator

1952 - 1954 U.S. Army

Table 13.1-1 (Cont.)

RESUME: PHILIP H. HENRIKSON

PRESENT POSITION:

Manager - Nuclear Licensing
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA.

EDUCATION:

BS in Chemistry, University of Nevada
MS in Chemical Engineering (minor in Nuclear Engineering), University of Idaho
Juris Doctorate, Lincoln University

ADDITIONAL MISC. COURSES:

U.S. Navy Submarine Officer Nuclear Training Program

GENERAL:

Registered Professional Engineer - Nuclear

Member American Nuclear Society

Member American Bar Association

EXPERIENCE:

1980 - Present	Manager-Nuclear Licensing, Pennsylvania Power and Light Company
1973 - 1980	General Electric - Served in Safety and Licensing Operation of the Nuclear Energy Division. Was Program Manager for GE involvement in NRC's Systematic Evaluation Program. Prepared reload license submittals and technical specification revisions for eight nuclear power plants. Supported Taiwan Power for three years in obtaining two construction permits and two operating licenses.
1971 - 1973	Bechtel - Initial assignment was in Nuclear Systems generic group. Later was Assistant Nuclear Group Supervisor for Rancho Seco Nuclear Power Station.
1966 - 1970	U.S. Naval Officer - Qualified engineering officer on nuclear submarine propulsion plant. Served two years as engineering officer.

Table 13.1-1 (Cont.)

1964 - 1966	Laboratory Assistant, Desert Research Institute, Atmospheric Physics Division, performing radioactive cloud seeding experiments.

Table 13.1-1 (Cont.)

RESUME: JEROME S. STEFANKO

PRESENT POSITION:	Manager-Nuclear Fuels Pennsylvania Power & Light Company Two North Ninth Street Allentown, PA.
EDUCATION:	University of Dayton BS Physics
ADDITIONAL MISC. COURSES:	Graduate studies in Physics, Mathematics and Nuclear Engineering
GENERAL:	Member Edison Electric Institute (EEI) - Nuclear Fuel Cycle Committee Member Atomic Industrial Forum-Policy Committee on Uranium Mining and Milling Member American Nuclear Society Author of six papers to National Technical Societies Holder of two patents for Nuclear Reactor Startup and Controls Systems (under USAF sponsorship).
EXPERIENCE:	
1976 - Present	Pennsylvania Power & Light Company Manager - Nuclear Fuels
1971 - 1976	Westinghouse Electric Corporation- Applications Engineer-Nuclear Fuels Marking Division
1963 - 1971	Astronuclear Laboratory of Westinghouse Electric Corporation Senior Nuclear Design Engineer
1960 - 1963	Allis-Chalmers' Nuclear Division Core Physicist

Table 13.1-1 (Cont.)

MANAGER - NUCLEAR TRAINING

SEE TABLE 13.1-2

Table 13.1-1 (Cont.)

RESUME: STEVEN H. CANTONE

PRESENT POSITION:	Manager-Nuclear Support Pennsylvania Power & Light Company Two North Ninth Street Allentown, Pennsylvania
EDUCATION:	Stevens Institute of Technology Bachelor of Engineering
ADDITIONAL MISC. COURSES:	Alexander Hamilton Institute Modern Business Program
EXPERIENCE:	
1979-Present	Pennsylvania Power & Light Company Manager-Nuclear Support
1976-1979	Power Authority of the State of New York Superintendent of Power-IP3NPP
1963-1976	Consolidated Edison Co. of New York, Inc. 1974-1976 Chief Operations Engineer-Indian Point 1972-1974 Operations Engineer - Indian Point 3 1963-1972 Various Positions of Successively Increasing Responsibility

Table 13.1-1 (Cont.)

RESUME: WILLIAM J. HESKE

PRESENT POSITION:

Manager-Nuclear Administration
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, Pa.

EDUCATION:

U.S. Naval Academy
BS Engineering
Auburn University
MS Political Science

ADDITIONAL MISC COURSES:

Basic Personnel Officers Course - USAF
Minuteman Combat Crew Training - USAF
Air Command Staff College - USAF
Systems Analysis & Operations Research - U.S. Army
Industrial College of the Armed Forces

EXPERIENCE:

1979-Present — Pennsylvania Power & Light Company

1980-Present	Manager-Nuclear Administration
1979-1980	Ass't to V.P. Engineering & Construction

1957-1979 — United States Air Force

1976-1979	Deputy Commander for Maintenance and Director Logistics F.E. Warren AFB
1975-1976	Student-Industrial College of the Armed Forces, U.S. Army
1972-1975	Chief-Future Missile Systems Analysis, Strategic Air Command
1971-1972	Student-Air Command & Staff College
1968-1971	Chief Missile, Space, and Weapons Director Officer Manning
1965-1968	Chief Missile Manning Branch Strategic Air Command
1962-1965	Minuteman Crew Commander Malmstrom AFB
1957-1962	Personnel & Administrative Officer

Table 13.1-1 (Cont.)

MANAGER - NUCLEAR SAFETY ASSESSMENT

SEE TABLE 13.1-2

Table 13.1-1 (Cont.)

RESUME: ANDREW R. SABOL

PRESENT POSITION:

Manager-Nuclear Quality Assurance
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA

EDUCATION:

Purdue University
BS Mechanical Engineering

ADDITIONAL MISC. COURSES:

Undergraduate and graduate courses in Business Administration-Pennsylvania State University

GENERAL:

Professional Engineer Registration - Quality Engineering-California

Member American Society for Nondestructive Testing (ASNT)

Member American Society for Testing and Materials (ASTM)

EXPERIENCE:

1974 - Present	Pennsylvania Power & Light Company Manager - Nuclear Quality Assurance
1971 - 1974	Gilbert Associates, Inc., Quality Assurance Division Quality Assurance Program Manager for: Crystal River Unit #3 Project (Florida Power Corporation) and Erie County Nuclear Project (Ohio Edison Company)
1967 - 1971	The Pennsylvania State University Manager-Industrial Reference
1960 - 1967	U.S. Atomic Energy Commission 1963-1967 Reactor Production Engineer, Pittsburgh Naval Reactors Office 1961-1963 Nuclear Production Engineer, Pittsburgh Naval Reactors Office 1960-1961 General Engineer, Hartford Area Office Connecticut Aircraft Nuclear Engine

Table 13.1-1 (Cont.)

Laboratory

1956 - 1960	Curtis-Wright Corporation Nuclear Science and Engineering Department Research Engineer
1954 - 1956	Bethlehem Steel Corporation Technical Assistant to Superintendent of Hot Forge
1949 - 1950	Department of the Air Force FEAMCOM, Japan, Supervising-Engineering Drafting

Table 13.1-1 (Cont.)

RESUME: BYRON G. DIXON

PRESENT POSITION:

Construction Manager
Pennsylvania Power & Light Company
Two North Ninth Street
Allentown, PA

EDUCATION:

Bucknell University
B.S. Mechanical Engineering

GENERAL:

Member-American Society of Mechanical Engineers

EXPERIENCE:

1959 - Present. Pennsylvania Power & Light Company

1979-present	Construction Manager
1974-1979	Superintendent of Plant-Martins Creek SES
1970-1974	Asst. Superintendent of Plant Martins Creek SES Assigned to work with Martins Creek Unit 3 & 4 Project Team full time.
1970-1970	Plant Betterment Engineer-Generation Dept.
1968-1970	Supervisor of Operation-Martins Creek SES
1967-1968	Technical Sueprvisor-Martins Creek SES
1962-1967	Results Engineer-Brunner Island SES
1960-1962	Engineer-Martins Creek SES
1959-1960	Graduate Trainee-Various locations in Company

Table 13.1-1 (Cont.)

RESUME: ROGER A. INGLESE

PRESENT POSITION:

Manager - Procurement
Pennsylvnaia Power & Light Company
Two North Ninth Street
Allentown, PA.

EDUCATION:

Lehigh University
BS Mechanical Engineering

GENERAL:

Member National Association of Purchasing Management

Member Edison Electric Institute (EEI) Purchasing and Stores Committee

Registered Professional Engineer-Commonwealth of Pennsylvania

EXPERIENCE:

1956-Present — Pennsylvania Power & Light Company

1974-Present	Manager-Procurement
1973-1974	Assistant Manager - Purchasing and Stores
1964-1973	Supervisor - Rate Administration and Contracts Department
1961-1964	Supervisor -Rate Auditing, Rate and Contracts Department
1956-1961	Various Sales Positions Business Development Department

1953-1955 — Test Engineer-Aberdeen Proving Ground US Army

TABLE 13.1-2

SCHEDULE FOR FILLING OPEN KEY SUPERVISORY POSITIONS

Management is conducting interviews with candidates for the following positions. All other Plant Staff and Nuclear Department Key Supervisory Positions are filled.

Position	Duties Temporarily Performed by	Anticipated Schedule. For Filling
Manager-Nuclear Training	Vice President-Nuclear Operations	February, 1981
Manager-Nuclear Safety Assessment	Manager-Nuclear Support	90 days prior to fuel load
Health Physics Supervisor	Health Physics Engineer	90 days prior to fuel load

TABLE 13.1-3

QUALIFICATIONS OF KEY PLANT SUPERVISORS

Title
Superintendent of Plant
Assistant Superintendent of Plant
Supervisor of Operations
Shift Supervisors (6)
Technical Supervisor
Reactor Engineering Supervisor
Plant Engineering Supervisor
Instrumentation and Control/Computer Supervisor
Chemistry Supervisor
Maintenance Supervisor
Health Physics Supervisor

TABLE 13.1-3 (Continued)

SUPERINTENDENT OF PLANT

NAME: Harold William Keiser

EDUCATION AND TRAINING:

1972	University of Illinois, Urbana, Illinois Bachelor of Science Degree in Metallurgical Engineering
1973	University of Illinois, Urbana, Illinois Master of Science Degree in Nuclear Engineering

WORK EXPERIENCE:

1980-Present	Pennsylvania Power & Light Co. Superintendent of Plant - SSES
1979 - 1980	Consumers Power Company Operations/Maintenance Superintendent. Palisades Nuclear Plant.
Duties:	Responsible for operations and maintenance (Electrical, Mechanical, Instrument and Control) of the plant. Responsible for managing all outages activities including refueling outages. Responsible for nine months for managing the Chemistry and Health Physics Department.
1976 - 1979	Operations Superintendent, Palisades Nuclear Plant
Duties:	Responsible for safe, efficient operation of the plant, chemistry control and plant training. Managed the Operations Department, Chemistry Department, and Plant Training Department.
1976 - 1976	Senior Engineer, Nuclear Production Department
Duties:	Responsible as Staff Assistant to the Manager of Nuclear Power Plants for the coordination of nuclear power plant activities (Palisades, Big Rock, and

TABLE 13.1-3 (Continued)

	Midland) with corporate activities. Responsibilities included obtaining from the various corporate departments the support needed to maintain the plants in operation and to shorten their outages. Responsible for coordinating the plant's comments on corporate activities including representing the plants at corporate meetings and providing the necessary feedback. Responsible for coordination between all the nuclear plants ensuring that the administration functions uniformly.
1973 - 1976	Senior Engineer, Palisades Nuclear Plant
Duties:	Responsible as Project Engineer for overall planning, performance and scheduling of steam generator eddy current testing, tube plugging and tube removal. Responsible as Project Engineer for turnover acceptance of radioactive waste processing facility. This major modification included installation of systems necessary for eliminating radioactive waste releases including equipment for processing, recycling and solidification of all liquid and gaseous waste products. Responsibilities included design review, test witnessing, test acceptance and system acceptance after construction and start-up testing. Responsible also for coordination of construction and start-up activities with plant activities. Responsible as Project Engineer for plant design review of 300 gpm makeup water system and full flow condensate polishers. Responsibilities included design review, vendor selection, interfacing architect engineer's document control system with the plant's procurement of spare parts, and coordination of construction activities with plant activities.
1973 - 1975	General Engineer, Palisades Nuclear Plant

TABLE 13.1-3 (Continued)

Duties:	Responsible for the overall planning, performance and scheduling of steam generator eddy current testing, tube plugging and tube removal. Responsibilities included design, procurement, checkout and installation of equipment, procedure development, quality assurance, training of repairmen, and providing interface and coordination between the company and subcontractors.
	Responsible as Project Engineer for various engineering modifications.
1968 - 1973	University of Illinois
Duties:	Responsible for operation and maintenance of Triga Reactor. Performed design installation and pre-operational testing of reactor.
1961 - 1968	U.S. Navy
	Electrical Operator in nuclear field. Responsible for maintenance and operation of power generation/distribution equipment and electrical auxiliaries. Served as qualified reactor operator and electrical operator on USS Tecumseh (SSBN 628) for two years. Served as Staff Instructor at SIC prototype. Qualified in all phases of plant operation. Instructed and qualified trainees in electrical operations, basic nuclear and electrical theory.

LICENSES AND CERTIFICATES:

Senior Reactor Operator license,
University of Illinois, Triga Reactor

Senior Reactor Operator license,
Palisades Nuclear Plant

TABLE 13.1-3 (Continued)

ASSISTANT SUPERINTENDENT OF PLANT

NAME: Donald J. Thompson

EDUCATION AND TRAINING:

1968	University of Nevada BSEE

WORK EXPERIENCE:

1980-Present	Assistant Superintendent of Plant Susquehanna Steam Electric Station
1975 to 1980	Plant Planner/Scheduler/Maintenance Supervisor/ Outage Supervisor Portland General Electric Trojan Nuclear Plant
Duties:	Responsible for planning, supervising and scheduling outages. Coordination of all departments, maintain cost control system, assure schedules are met, prevent impacts between work groups, assure quality is maintained and work plans meet with the ALARA Program. Responsible for electrical and mechanical maintenance and warehousing. Responsibility required knowledge of the nuclear plant Quality Assurance requirements, technical specifications, various codes and standards and good maintenance practices. Scheduled activities with systems such as PERT, CPM, and Project 2.
1973 to 1975	Manager of Refueling Operations Morrison Kmudsen Inc. West Milton Naval Reactor Facility
Duties:	Supervised engineers, superintendents, and craft personnel; responsible for support of refueling operations for Naval Reactor prototype plants at the West Milton New York Naval Reactor Facility; responsible to assure all refueling related activities were completed in a safe and timely manner. Responsibility

TABLE 13.1-3 (Continued)

	required a knowledge of radiological work practices, Quality Assurance, NDE Procedures, Nuclear Fuel handling procedures, welding techniques and labor contracts and their application
1968 to 1972	Nuclear Plant Engineer/Shift Supervisor/Outage Manager Westinghouse Electric Corporation Idaho Falls, Idaho
Duties:	Qualification as Engineering Officer of the Watch; 5 months of classroom training followed by in-plant training. Nine months as Nuclear Plant Engineer, with subsequent promotion to Shift Supervisor. As Shift Supervisor, responsible for the operation of a two unit plant; knowledgeable of all plant systems and procedures; responsible for a crew of 30 men and 4 engineers. As the Outage Manager responsible for the planning/scheudling, preparation, co-ordination and timely completion of outages; assisted by a 3 man staff and was responsible for operation and maintenance during outages.
1966 to 1968	Radar Repairman/Student Desert Research Institute Reno, Nevada
Duties:	Engineering aide to scientists working on various research projects; designed small circuits, repaired and maintained radar and computer equipment.

LICENSES AND CERTIFICATES:

None

TABLE 13.1-3 (Continued)

SUPERVISOR OF OPERATIONS

NAME: George Leon Adams

EDUCATION AND TRAINING:

1974	Pennsylvania State University B.S. Degrees in Nuclear Engineering and Administrative Management.
8/64 to 1/65	U.S. Navy Nuclear Power Training Unit D1G Prototype
2/64 to 8/64	U.S. Navy Basic Nuclear Power School Bainbridge, Maryland
9/63 to 12/63	U.S. Navy Electrician Mate 'A' School

WORK EXPERIENCE:

2/78-Present	Supervisor of Operations SSES
Duties:	Responsible for management of plant staff Operations Section during Initial Energization and Preoperational Test Program activities at plant site. Responsibilities include establishment and implementation of the necessary administrative controls, development and implementation of the necessary operating procedures for nuclear and balance-of-plant systems, and supervision of shift supervisory personnel. Evaluated senior license and license candidate personnel by oral examination. Provided extensive review and evaluation of control room operating procedures for Advanced Control Room (ACR)/ACR Simulator. Responsible for coordination of all Operations Section activities with other site groups and station sections during test program, and for development of all section programs/procedures to prepare for fuel load, startup testing and commercial operation.
5/77 to 2/78	Startup Engineer, General Electric Co. Shoreham Site.

TABLE 13.1-3 (Continued)

Duties:	Provided technical direction during pre-operational test phase. Developed and reviewed test procedures. Primary responsibility for Nuclear Steam Supply Shutoff System, and Reactor Water Cleanup System. Provided interface between nuclear vendor and utility personnel.
10/76 to 5/77	Startup Engineer, General Electric Co. Brunswick Site.
Duties:	Provided Technical Direction during Power Test Phase - SRO Certified. Coordinated all test activities with/for the utility Shift Supervisor for duraton of the Power Test Program.
6/74 to 10/76	Training Engineer, General Electric Co.
Duties:	Developed and conducted training courses in all phases of BWR Power Plant Training at the GE BWR Training Center, Morris, Illinois. Included six (6) months at Millstone Unit 1 during refueling outage on assignment as Engineer.
6/70 to 6/74	Reactor Operator on "TRIGA" Research Reactor at Pennsylvania State University while earning degree.
5/63 to 6/70	Operator/Instructor U.S. Navy Nuclear Program.
Duties:	Naval experience includes three (3) years in Engineering Group aboard naval vessel in Electrical Division; participated in startup test program. Two (2) years assignment at naval prototype (D1G) as operator/staff instructor.

LICENSES AND CERTIFICATES:

1970	RO License, Pennsylvania State Univeristy TRIGA Reactor
1975	GE SRO Certified Dresden II and III
1975	SRO Licensed, Dresden II and III
1977	GE SRO Certified Brunswick Nuclear Power Station, Unit 1

TABLE 13.1-3 (Continued)

SHIFT SUPERVISOR

NAME: Phillip M. Bartel

EDUCATION AND TRAINING:

	High School Graduate
1964-1965	Navy Nuclear Power School Completed Enlisted Qualifications
1972	Westinghouse Nuclear Training Center PWR Cold License Training
1978	SRO Certification Training-Browns Ferry

WORK EXPERIENCE:

1/77-Present	Shift Supervisor - Susquehanna SES
Duties:	Responsible for supervision of shift personnel and directing all shift operating activities during nuclear power plant test program including the Initial Energization and Preoperational Testing Phases. Responsible for operation of all power plant equipment during test program, and for the implementation and enforcement of applicable administrative controls such as those relating to work authorization, personnel safety, protective permit and tag, and conduct of operations.
4/74 to 1/77	Startup and Test Coordinator United Engineers and Constructors
Duties:	Coordinated Startup and test efforts at two large fossil fueled units.
10/73 to 4/74	Field Engineer - Bechtel Power Corp. Davis-Besse Nuclear Power Station
Duties:	Verified proper system installation and initiated plant system installation checks. Compiled all system documentation required for release of plant systems to the utility.

TABLE 13.1-3 (Continued)

9/71 to 10/73	Operations Foreman - Duquesne Light Co. Beaver Valley Nuclear Power Station
Duties:	Supervised the startup and proper operation fo plant systems. Researched and wrote system descriptions and operating procedures. Completed cold license training for the NRC SRO License.
2/64 to 1/71	Navy Nuclear Power Program Qualified as Engineering Watch Supervisor
Duties:	Supervised all engineering watch standers in the operation of the S5W PWR Reactor Plant. Responsible for the training of electrical operators and supervised the maintenance and repair of all electrical equipment. Participated in all phases of a reactor fueling, approach to criticality, and power range testing.

LICENSES AND CERTIFICATES:

1972	SRO Certification for Westinghouse PWR Plant from the Westinghouse Nuclear Training Center
1978	SRO Certification for General Electric BWR Plant from General Physics

TABLE 13.1-3 (Continued)

SHIFT SUPERVISOR

NAME: Russell H. Halm

EDUCATION AND TRAINING:

1966	High School Graduate
1969 - 1970	Navy Nuclear Power School Completed Enlisted Qualifications
1974	G.E. BWR Operator Training SRO Certification

WORK EXPERIENCE:

7/78 - Present	Shift Supervisor - Susquehanna S.E.S.
Duties:	Responsible for supervision of shift personnel and directing all shift operating activities during nuclear power plant test program including the Initial Energization and Preoperational Testing Phases. Responsible for operation of all power plant equipment during test program, and for the implementation and enforcement of applicable administrative controls such as those relating to work authorization, personnel safety, protective permit and tag, and conduct of operations.
9/74 - 7/78	BWR Instructor G.E. BWR Training Center
Duties:	Instructor for all phases of BWR operator training including design orientation, plant specific technology, observation, hot license, retraining, and fundamentals. Responsible for setting up with Utilities, retraining programs and technology courses. Called upon to meet with NRC personnel and answer questions concerning generic BWR operational licensing issues.
7/68 - 7/74	Navy Nuclear Power Program
Duties:	Qualified as Engineering Watch Supervisor and Lead Engineering Laboratory Technician.

TABLE 13.1-3 (Continued)

Supervised all Engineering Watch Standers in the operation of the D1G prototype and the A2W PWR plant. Responsible for plant operations and test chemistry along with training of Mechanical Watch Standers at the D1G prototype. Involved with repair and replcement of reactor coolant pump and other steam and reactor components during refueling at D1G prototype and new construction of the CVAN Nimitz.

LICENSES AND CERTIFICATES:

1975	G.E. SRO Certification
1976	NRC SRO License Dresden Units 2 & 3

TABLE 13.1-3 (Continued)

SHIFT SUPERVISOR

NAME: Stanley J. Laskos

EDUCATION AND TRAINING:

1957	High School Graduate
1962	Stationary Engineers Course by PSE&G Co.
7/72 - 6/73	Reactor Operators Training Program by N.U.S. Corp. Nuclear Plant System Qualifications
8/73 - 11/74	Accelerated Westinghouse Training Program for Cold Senior Reactor Operators
3/75	Training Program at Westinghouse Pool Reactors at Zion, Illinois
4/75 - 9/75	Westinghouse Nuclear Training Center, Cold License Certification on PWR's
10/75 - 9/76	Westinghouse On Site Training Package for Cold SRO's Including: Health Physics Technician Course Pre-License Review Series Fuel Handling Qualification
1/77	General Electric Design Orientation
10/77	General Electric Advanced Control Room Course
4/78	General Physics BWR Cold License Certification Program

WORK EXPERIENCE:

5/77 - Present	Shift Supervisor - Susquehanna S.E.S.
Duties:	Responsible for supervision of shift personnel and directing all shift operating activities during nuclear power plant test program including the Initial Energization and Preoperational Testing Phases. Responsible for operation of all power plant equipment during test program, and for the implementation and enforcement of applicable administrative controls such as those relating to work

TABLE 13.1-3 (Continued)

	authorization, personnel safety, protective permit and tag, and conduct of operations.
8/74 - 5/77	Acting Shift Supervisor/Shift Foreman Public Service Electric & Gas Co. Salem Nuclear Generating Station
Duties:	Performed duties of Shift Supervisor/Shift Foreman at Salem under SRO License #SOP 2724 Docket #55-5438 during all phases of plant construction, including Cold Hydro, Hot Functional Testing, Pre-Criticality Testing, Power Testing, Initial Core Load and Criticality. Participated in SRO Requalification Program.
7/72 - 8/74	Control Room Operator
Duties:	Reactor Operator Trainee at Salem. Performed duties of Control Room Operator and assisted in the start-up of plant construction.
6/60 - 6/72	Linden Generating Station Public Service Electric & Gas Co.
	Twelve years responsible power plant experience as a Control Operator, Equipment Operator Special, and Equipment Operator at a large fossil generating station. Assisted in the start-up and operated modern high pressure oil fired boilers, condensing and extraction turbines, Pratt & Whitney and Westinghouse gas turbine peaking units. Performed major and minor system cutouts, tagging and electrical switching.

LICENSES AND CERTIFICATES:

1962	State of New Jersey Stationary Engineers License #A-40397
1975	Cold License Certification from Westinghouse PWR Training Center
1976	NRC Senior Reactor Operators License (Cold) #SOP 2724 Docket #55-5438

TABLE 13.1-3 (Continued)

1978	BWR Cold License Certification from General Physics Corporation at the Browns Ferry Nuclear Plant Simulator

TABLE 13.1-3 (Continued)

SHIFT SUPERVISOR

NAME: Thomas R. Markowski

EDUCATION AND TRAINING:

	High School Graduate
1965-1967	Pennsylvania State University Various Technical Courses
1976-1978	SRO License Training Program - SSES

WORK EXPERIENCE:

4/76-Present	Shift Supervisor - Susquehanna SES
Duties:	Responsible for supervision of shift personnel and directing all shift operating activities during nuclear power plant test program including the Initial Energization and Preoperational Testing Phases. Responsible for operation of all power plant equipment during test program, and for the implementation and enforcement of applicable administrative controls such as those relating to work authorization, personnel safety, protective permit and tag, and conduct of operations.
1967-4/76	Martins Creek Steam Electric Station Pennsylvania Power and Light Company
Duties:	Plant control operator of a 160 MW coal fired power plant. Plant control operator and assistant shift supervisor for the startup and operation of an 850 MW oil fired power plant. Responsible for writing operating guidelines for the 850 MW unit. During the initial eight months of operation of the 850 MW unit, supervised over 100 startups and shutdowns.

LICENSES AND CERTIFICATES:

G.E. BWR Training Center S.R.O. Certification

TABLE 13.1-3 (Continued)

SHIFT SUPERVISOR

NAME: Joseph H. Murray

EDUCATION AND TRAINING:

	High School Graduate
1969-1970	Navy Nuclear Power Program Completed Enlisted Qualifications
1976	GE BWR Technology
1976	GE BWR Operator Training Received SRO Certification
1977	GE BWR Observation Training

WORK EXPERIENCE:

6/76-Present	Shift Supervisor - Susquehanna SES
Duties:	Responsible for supervision of shift personnel and directing all shift operating activities during nuclear power plant test program including the Initial Energization and Preoperational Testing Phases. Responsible for operation of all power plant equipment during test program, and for the implementation and enforcement of applicable administrative controls such as those relating to work authorization, personnel safety, protective permit and tag, and conduct of operations.
6/74-1/76	Engineering Associate - Stone & Webster Corp.
Duties:	Assigned as responsible engineer for three NSSS systems and five BOP systems. Responsibilities included certification of conformance to NRC Regulatory Guides and CFR Requirements, equipment selections and specifications, line sizing and system resistance calculations.
6/68-6/74	Navy Nuclear Power Program

TABLE 13.1-3 (Continued)

	Qualified as:	Engine Room Supervisor Leading Engineering Laboratory Technician
Duties:		As Engine Room Supervisor, responsible for supervising the operation of reactor and steam plant support systems, including turbine-generators, main engines, etc. Also responsible for the supervision of seven men for maintenance and repairs on the above equipment. As leading ELT, responsible for supervising the analyzation and maintenance of reactor and steam plant chemistry; shipboard radiation protection, maintenance of all records and reports, and training of personnel for qualification as radiation health workers.

LICENSES AND CERTIFICATES:

1976	SRO Certification on a GE BWR Plant issued by the GE BWR Training Center.

TABLE 13.1-3 (Continued)

SHIFT SUPERVISOR

NAME: David M. Smith

EDUCATION AND TRAINING:

1964	High School Graduate
1964 - 1973	U.S. Navy Nuclear Power Program Completed Enlisted Qualifications
1975	U.S. NRC Senior Reactor Operator License Training, Carolina Power and Light Co.
1978	U.S. NRC Senior Reactor Operator License Training, Power Authority of the State of New York.

WORK EXPERIENCE:

4/80-present	Shift Supervisor - Susquehanna SES
Duties:	Responsible for supervision of shift personnel and directing all shift operating activities during nuclear power plant test program including the Initial Energization and Preoperational Testing Phases. Responsible for operation of all power plant equipment during test program, and for the implementation and enforcement of applicable administrative controls such as those relating to work authorization, personnel safety, protective permit and tag, and conduct of operations.
8/77 - 4/80	Shift Supervisor - James A. Fitzpatrick Nuclear Power Plant. Power Authority of the State of New York
Duties:	Supervises on a shift-to-shift basis the safe, continuous operation of the plant according to established procedures, NRC licenses, Operations' quality assurance program requirements and requests by the load dispatcher.
6/76 - 8/77	Shift Foreman - Brunswick Steam Electric Plant Carolina Power and Light Co.

TABLE 13.1-3 (Continued)

Duties:	Supervised on a shift-to-shift basis the plant operations during all phases of plant construction, including cold hydro, hot functional testing, initial core loading and criticality, and commercial power operations.
5/75 - 6/76	Control Room Operator - Brunswick Steam Electric Plant - Carolina Power and Light Co.
Duties:	Operated the plant controls and equipment during all phases of plant opreation from initial construction to power operations including refueling operations.
4/73 - 5/75	Auxiliary Operator - Brunswick Steam Electric Plant - Carolina Power and Light Co.
Duties:	Operated plant equipment outside the control under the direction of the Control Room Operator during all phases of plant construction, fuel loading and startup testing.
10/64 - 2/73	U.S. Navy Nuclear Power Program
Duties:	Qualified as Engineering Watch Supervisor, Reactor Operator, and Reactor Technician. Supervised the Engineering Watch Standers in the operation of the AlW prototype. Reactor Operator/Technican on the U.S.S. Enterprise CVA(N)-65.

LICENSES AND CERTIFICATES:

1975	NRC Senior Reactor Operator License SOP-2469
1978	NRC Senior Reactor Operator License SOP-3216

TABLE 13.1-3 (Continued)

TECHNICAL SUPERVISOR

NAME: Louis D. O'Neil

EDUCATION AND TRAINING:

1960-1964	Susquehanna University BA - Biology
1967-1969	Lycoming College Courses in Advanced Math and Physics
1969-1970	Pennsylvania State University BS - Mechanical Engineering
1974	GE BWR Design Orientation
1974	Radioactive Waste Management for Nuclear Power Reactors
1975	GE Process Instrumentation and Control
1975	Planning for Nuclear Emergencies
1975	Research Reactor Training at the Pennsylvania State University
1976	GE BWR Technology
1976	GE BWR Operator Training Received SRO Certification
1977	GE BWR Observation Training
1978	GE BWR Station Nuclear Engineering
1979	PP&L BOP Systems Course
1979	PP&L NSSS Systems Course
1980	PP&L SRO License Training Program

WORK EXPERIENCE:

1974-Present	Technical Supervisor - Susquehanna SES
Duties:	Manages the Technical Support Group to ensure safe, reliable and efficient operation of the station in compliance

TABLE 13.1-3 (Continued)

	with NRC and other regulatory requirements. The chemistry, radiochemistry, nuclear engineering, plant engineering and instrument and control maintenance services are provided by this group.
	In addition, have overall responsibility for the Startup Test Program covering Unit fuel load through power escalation testing. (NOTE: Effective 6/80 the Instrument & Controls/Computer Group was reorganized as an independent section).
1972-1974	Results Engineer - Montour Steam Electric Station Pennsylvania Power and Light Company
Duties:	Assist the Technical Supervisor in assigning and supervising routine and non-routine work of technical department personnel. Supervised all instrument and control activities at the station which consisted of one 800 MW coal fired unit in commercial operation and a second 800 MW coal fired unit which went through final construction and startup phases to commercial operation. During startup of Unit 2, all instrumentation and control checkout and calibration was performed by station personnel. Routine responsibilities included supervision of unit startups and shutdowns. In addition, assumed responsibility of "On-call Supervisor" in scheduled rotation with other plant management personnel to cover weekends and holidays. "On-call Supervisor" was the senior plant supervisor during such periods and responsible for the station operation.
1971-1972	Engineer-Results - Montour Steam Electric Station Pennsylvania Power and Light Company
Duties:	Involved in the final construction and startup phases of Unit 1, an 800 MW coal fired unit. Reviewed final construction of assigned systems for operating acceptability, participated as principle

TABLE 13.1-3 (Continued)

	station engineer in the startup of such systems, evaluated system performance, recommended design changes as required and provided engineering support and assistance to operating and maintenance personnel. Supervised and directed station chemistry activities during the startup phase and subsequent operation.
1970-1971	Engineer-Results - Brunner Island Steam Electric Station Pennsylvania Power and Light Company
Duties:	Assist the Results Engineer and Technical Supervisor in providing engineering support for the station. The station consisted of three coal fired units with a total output of 1450 MW. Responsibilities included performing engineering investigations and evaluations of plant and equipment operation, both normal and abnormal. Participated in eight week annual overhaul and inspection of a supercritical unit. Had primary technical responsibility for effective operation of station water treatment systems and chemistry activities.

LICENSES AND CERTIFICATES:

SRO Certification on a GE BWR Plant issued by the GE BWR Training Center

Registered Professional Engineer - Commonwealth of Pennsylvania

TABLE 13.1-3 (Continued)

REACTOR ENGINEERING SUPERVISOR

NAME: Adam H. Geesey Jr.

EDUCATION AND TRAINING:

1965-1969	Susquehanna University B.A. Physics
1969-1971	The Pennsylvania State University M.S. Nuclear Engineering
1972	GE BWR Design Orientation
1972	GE BWR Introduction to Operations (1 week Simulator Course)
1975	GE Process Instrumentation and Control (Lecture Portion)
1975	MIT Nuclear Fuel Management Seminar
1977	GE BWR Technology
1977	GE BWR Operator Training (Receive SRO Certification)
1978	GE BWR Observation Training
1978	GE BWR Station Nuclear Engineering Course
1979	SSES Balance of Plant Systems Course SSES SRO Licensing Training
1979	SSES HSS Systems Course SSES SRO Licensing Training
1980	SSES Systems and Simulator Training SSES SRO Licensing Training

WORK EXPERIENCE:

1977-Present	Reactor Engineering Supervisor, Susquehanna SES Plant Staff
Duties:	Develops and directs nuclear plant Reactor Engineering Programs that will provide the assurance that nuclear fuel performance is safe, reliable and efficient, the associated records and surveillances are generated, maintained

TABLE 13.1-3 (Continued)

	and performed in compliance with the requirements of the operating license and the fuel warranty; fuel and core components are inspected and suitable for service and provide technical direction and support to plant operations relating to the fuel.
1974-1977	Results Engineer, Susquehanna SES Plant Staff
Duties:	Responsible for reviewing and commenting on various plant specifications, drawings, etc., being reviewed at the time. Coordinated the plant staff responsibilities for the writing of various FSAR sections and wrote initial drafts of several sections. Responsible for plant staff activities in the fuel and fuel handling areas and performed many related reviews. Filled in as Acting Technical Supervisor in his absence.
1972-1974	Engineer, Susquehanna SES Plant Staff
Duties:	Performed reviews on many project documents being generated at that time, was responsible for all activities in the fuel and fuel handling areas and during this period was stationed at an operating BWR for the duration of its first two refueling outages where I helped the station nuclear engineer carry out his duties.
1971-1972	Engineer, Susquehanna Engineering Project Team
Duties:	Assembled NSSS documents in packets for review and send them for review to applicable company groups. Coordinated the resolution of comments on these documents. Performed cost and technical analysis for scope changes on NSSS equipment and systems.
1970-1971	AEC Trainee and Reactor Operator, Breazeale Nuclear Reactor, The Pennsylvania State University.

TABLE 13.1-3 (Continued)

Duties:	Operated a TRIGA reactor including pulses, power changes and manipulations necessary to conduct different experiments. Completed a thesis on a conceptual design of a reactor protection system that was calculated to be more reliable than those currently utilized by the industry.
1969-1970	AEC Trainee Reactor Operator Trainee, Breazeale Nuclear Reactor, The Pennsylvania State University.
Duties:	Trained to become an AEC licensed Reactor Operator on the Penn State TRIGA reactor.

LICENSES AND CERTIFICATES:

Formerly held an AEC Reactor Operator's License on the Penn State TRIGA reactor.

Registered Professional Engineer (in the Nuclear Engineering Discipline) by Commonwealth of Pennsylvania.

Senior Reactor Operator Certification on a GE BWR Plant issued by the GE BWR Training Center.

TABLE 13.1-3 (Continued)

PLANT ENGINEERING SUPERVISOR

NAME: Jerome A. Blakeslee, Jr.

EDUCATION & TRAINING:

1966-1970	Lafayette College BS - Mechanical Engineering
1972-1973	Pennsylvania State University MS - Nuclear Engineering Research Reactor Licensing Training Received Operator License
1974	GE BWR Design Orientation
1974	Radioactive Waste Management for Nuclear Power Reactors
1976	Operational Quality Assurance and Audits for the Nuclear Power Industry
1976	GE BWR Technology
1976	GE BWR Operator Training Received SRO Certification
1977	GE BWR Observation Training
1978	Station Nuclear Engineering

WORK EXPERIENCE:

1977-Present	Plant Engineering Supervisor - Susquehanan SES
Duties:	Manages the Plant Engineering Group and assists the Technical Supervisor in ensuring the safe, reliable and efficient operation of the station in compliance with NRC and other regulatory requirements. Responsible for technical review of preoperational and acceptance tests procedures and test results, and determination of operational readiness of systems for fuel load. Responsible for compliance with Surveillance Testing requirements, review and benefit analysis of station design changes, review and preparation of plant procedures, preparation and revision of the station

TABLE 13.1-3 (Continued)

	Systems Description Manual, conducting equipment performance and availability studies, and providing technical support to other plant groups as necessary.
1975-1977	Results Engineer - Susquehanna SES
Duties:	Responsible for continuing development of the plant procedures program and development of the station Systems Description Manual. Performed reviews of station design documents.
1973-1975	Engineer - Results - Susquehanna SES
Duties:	Performed reviews of station design documents. As a member of the Operational QA Task Force began development of the Susquehanna Operational Quality Assurance Program.
1972-1973	Operator Trainee/Licensed Operator - Pennsylvania State University - Breazeale Nuclear Reactor
Duties:	As a member of the staff, performed reactor operation and other operational support activities at the research reactor facility.
1970-1972	Engineer - Results - Martin's Creek SES
Duties:	Assisted the Technical Supervisor and Results Engineer in providing engineering support for the station. The station consisted of two 160 MW coal fired units. Responsibilities included performing engineering investigations and evaluations of plant and equipment operation, and participating in annual overhaul and inspection of the units. Had primary technical responsibility for effective operation of the water treatment systems and chemistry activities.
1970	Engineer - Mechanical Test
Duties:	Participated in acceptance testing of turbine generator unit at Brunner Island

TABLE 13.1-3 (Continued)

SES Unit Three, an 800 MW coal fired unit. Participated in electrostatic precipitator performance testing at Brunner Island SES and Martin's Creek SES.

LICENSES AND CERTIFICATES:

SRO Certification on a GE BWR Plant issued by the GE BWR Training Center

Operator License on 1 MWt Breazeale Nuclear Reactor at Pennsylvania State University

Registered Professional Engineer - Commonwealth of Pennsylvania

TABLE 13.1-3 (Continued)

INSTRUMENT & CONTROL/COMPUTER SUPERVISOR

NAME: Joseph B. Rimsky.

EDUCATION AND TRAINING:

1962-1967	Drexel Institute of Technology, BSEE (Instrumentation and Computer Options)
1967	Computer Hardware Repair Training
1975	Reactor Technology and Operations Training, Penn State
1975	G.E. BWR Fundamentals Training (Simulator Training)
1976	G.E. BWR Technology Training
1979	IRD Mechanalysis - Machine Balancing
1979	G.E. BWR Nuclear Instrumentation
1980	BWR Fundamentals Training (Simulator Training)

WORK EXPERIENCE:

4/80 to Present	Instrument and Controls/Computer Supervisor Susquehanna SES
Duties:	To develop, administer and supervise the instrumentation and controls/computer program for Susquehanna S.E.S. This involves long range planning of the plant's needs to meet NRC compliance and maintain safe, continuous and efficient operation. Areas of involvement include; outage planning, surveillance procedures and schedules, personnel training, operations review, instrument maintenance and calibration planning, computer updating and maintenance, records preparation and filing, instrumentation and control problem resolution and any other activities which involve the plant's instrumentation, controls and computer system. The major challenge of this task is on-going review required to maintain the I & C/Computer Program in compliance with the Technical

TABLE 13.1-3 (Continued)

	Specifications, Regulatory Guides, ANSI Standards and the Code of Federal Regulations.
4/79 to 4/80	Instrument and Control Engineer - Susquehanna SES
Duties:	Provided input to the I&C/Computer Supervisor for developing the I&C Programming. Responsible for instrument and control problem analysis for the plant systems. Researching these problems and submitting recommendations for change of procedure, equipment, or purchases and spare parts. Prepared reports to be used by the group for functions such as justification for budget items, NRC reporting, audit reporting, etc.
5/73 to 4/79	Field Engineer - Starting and Testing - PP&L
Duties:	Accountable for the checkout and initial operation of power plant systems. Included within the duties are pre-operational testing and supervising the initial operation of the systems to insure system reliability and design integrity.
2/77 to 4/77	Plant Betterment Engineer - PP&L
Duties:	Represent the power plant to management on all matters with an environmental impact; review, follow-up and resolve any out of compliance reports originating from the Chemical Lab or Environmental Engineer. Represent Power Production Department at all boiler inspections.
6/67 to 5/73	System Start-up Engineer - Leeds & Northrup Company
Duties:	Responsible for the commissioning of digital and analog control systems which automated large industrial steam supplies and associated process. As lead engineer the overall supervision and direction of the control installation was the major area of responsiblity. This area

TABLE 13.1-3 (Continued)

	included staffing, scheduling, training and work performance.
9/62 to 6/67	Co-op Student - Taylor Instrument Companies
Duties:	Formal training program provided varing duties in different departments. Assignments were in manufacturing, design, testing, and systems engineering.

LICENSES AND CERTIFICATES:

None

TABLE 13.1-3 (Continued)

CHEMISTRY SUPERVISOR

NAME: Robert G. Johnson

EDUCATION AND TRAINING:

1956-1958	Mason City Junior College Associate of Arts - Math
1958-1961	University of Northern Iowa BA - Math
1961-1974	University of Northern Iowa Science. No Degree
1970	GE BWR Technology
1971	GE BWR Chemistry
1972	General Electric Radiation Engineer Training for Health Physics

WORK EXPERIENCE:

4/77-Present	Chemistry Supervisor - Susquehanna SES
Duties:	To develop and direct nuclear station chemistry and radiochemistry programs that will provide appropriate monitoring of plant process in order to demonstrate system function is within specified limits and plant releases are as low as reasonably achievable.
1971-4/77	Plant Chemist - Duane Arnold Energy Center Iowa Electric Light and Power Company
Duties:	Installation of sampling equipment and field activities associated with the initiation of the DAEC Preoperational Environmental Radioactivity Monitoring Program including the training of the technicians. Purchasing of initial equipment and chemicals for the radiochemistry laboratory. Training chemistry technicians. Setting up and calibration of radiochemistry laboratory equipment. Assembling and review of the Plant Chemistry and Counting Room Manual. Design verification and review of

TABLE 13.1-3 (Continued)

	operating/testing procedures for plant chemistry treatment equipment. Taught plant chemistry to the DAEC operator trainees taking the Basic Nuclear course. Calibration and testing of plant radiation process monitors. Supervision of Chemistry Laboratory through startup, refueling and normal plant operations. Supervision of maintenance of condensate demineralizers including repair of valves, retubing and acid cleaning.
1965-1971	Chemistry Technician in System Laboratory Iowa Electric Light and Power Company
Duties:	Perform analysis of water, fossil fuel, lubricating oil, and insulating oil. Involved with the change over from zeolite softners to deminerilizers. Chemical cleaning of heat exchanges and boilers.

LICENSES AND CERTIFICATES:

None

TABLE 13.1-3 (Continued)

SUPERVISOR OF MAINTENANCE

NAME: Robert G. Byram

EDUCATION AND TRAINING:

1963-1968	Drexel University BS - Mechanical Engineering
1969-1974	Franklin and Marshall College MS - Physics

WORK EXPERIENCE:

1977-Present	Supervisor of Maintenance Susquehanna SES
Duties:	Plan, control, and direct the activities of the Maintenance Department, which consists of mechanical and electrical maintenance crews, to ensure safe, reliable and efficient operation of the station in compliance with the NRC and other regulatory agency requirements.
1976-1977	Starting and Testing Mechanical Group Supervisor Pennsylvania Power and Light Company
Duties:	Responsible for the inspection, testing and operation of various power plant systems prior to and during the startup and initial operating phases of new generation facilities. Provided expertise in power plant mechanical systems, supervised the preparation of tests and special procedures, and coordinated the participation in the startup programs of plant supervisory personnel and vendors.
1975-1976	Senior Engineer Consumers Power Company
Duties:	Total responsibility for complete engineering function performed by outside A/E consulting firms on generating plant modification projects. This included definition of project scope through

TABLE 13.1-3 (Continued)

	required studies, economic justification and budget submittals, cost and schedule control, preparation of material/installation specifications and procurement function, and final detail engineering.
1974-1975	Senior Maintenance Planning Engineer Tampa Electric Company
Duties:	Responsible for directly supervising the activities of five maintenance engineers and planning all maintenance activities at a six-unit, 1300 MW, coal-fired generating station. Under direction from the Assistant Plant Superintendent, supervised the daily activities of two general maintenance foremen and eight maintenance crews.
1973-1974	Results Engineer - Brunner Island Steam Electric Station Pennsylvania Power and Light Company
Duties:	Responsibilities associated with assuring that optimum operating performance was attained. This included direct supervision of three office clerks who tabulated plant operating data to compute boiler/turbine efficiences and three coal samplers who continuously sampled coal received for lab analysis and subsequent heat rate calculations. This position was also titled "Fuels Coordinator" which involved coordinating coal rail shipments with the company's central Coal Bureau and evaluating coal quality as to slagging characteristics and heating value. It also involved direct supervision of coal yard foremen and daily yard coal handling operating/maintenance procedures.
1971-1973	Engineer-Maintenance - Brunner Island Steam Electric Station Pennsylvania Power and light Company
Duties:	Responsible for maintenance planning and coordination of scheduled unit outages and forced outages on all three units at

TABLE 13.1-3 (Continued)

Brunner Island. Included close contact with equipment manufacturers on maintenance procedures, preparation of sub-contracts to the company's Construction Dept. and outside contractors, and preparation of a complete maintenance memorandum displaying all work to be done during a scheduled unit outage. Responsible for turbine generator spare parts inventory and the performance/testing of entire turbine hydraulic/mechanical control systems.

LICENSES AND CERTIFICATES:

None

TABLE 13.1-3 (Continued)

HEALTH PHYSICS SUPERVISOR

See Table 13.1-2

13.2 TRAINING PROGRAM

13.2.1 PLANT PERSONNEL TRAINING PROGRAM

The Training Program for the Susquehanna Steam Electric Station is formulated to develop and maintain an organization qualified to assume the responsibility for operation, maintenance, and technical considerations for the facility. In order to accomplish these objectives and to provide the necessary control of the overall plan, three separate training programs listed below are utilized:

a. Initial Plant Staff Training Program

b. Requalification Training Program, and

c. Replacement Training

The Initial Plant Staff Training Program is designed to produce competent, trained personnel at all levels of the plant organization. The programs are designed to allow placement of personnel into specific levels based on employee experience and intended position.

The Requalification Training Program provides continuing training for plant personnel commensurate with their area of responsibility.

The Replacement Training Program is designed to supply qualified personnel for the station organization.

The Superintendent of Plant may waive portions of the training program for individuals based on their previous experience and/or qualifications.

13.2.1.1 Program Description

13.2.1.1.1 Initial Plant Staff Training

Figure 13.2-1 shows the present schedule for the various Initial Plant Training Programs. Should significant differences or changes occur in those courses not yet conducted, the appropriate course outline and description will be revised by amendment.

13.2.1.1.2 Operations Section Training Program

This program is designed for individuals who are to assume responsibility for the licensed and non licensed operator positions and fulfills the general requirements and qualifications set forth in ANSI N18.1-1971. The program is structured to allow personnel of varying experience and education to enter the Cold Licensing Training Program at various levels and still fulfill the eligibility requirements for NRC cold licensing prior to fuel loading.

13.2.1.1.2.1 Initial Cold License Training

The program is designed for cold license candidates with no formal power plant experience or training. The program is divided into seven phases to ensure proper administration, documentation, and completeness of training.

- o. Phase I - Conventional Power Plant Operator Experience Program.
- o. Phase II - Academic Program for Nuclear Power Plant Personnel.
- o. Phase III - Basic BWR Technology
- o. Phase IV - BWR Simulator Training
- o. Phase V - BWR Observation Training
- o. Phase VI - Systems, Procedures and On-The-Job Training
- o. Phase VII - BWR Refresher Training

Those plant control operator license candidates with no power plant experience will participate and qualify in all seven phases, while those with only a conventional power plant background will participate and qualify in Phases II through VII. Operators, and other staff members, who will be cold licensed with a nuclear background and/or related academic or technical training will participate and qualify in selected portions of phases II through VI and all of Phase VII. The extent of their participation in Phases II through VI will be based on their background and documented in station training records.

o Phase I - Conventional Power Plant Experience Program

The Conventional Power Plant Experience Phase of the Susquehanna SES Training Program is designed to provide power plant experience to those license candidates who lack the minimum power plant experience requirements. This experience will be provided prior to the start of the formal License Training Program (Phases II-VII), so that by the time of the Nuclear Regulatory Commission Licensing Examination, the candidate will have had two years of power plant experience of which a minimum of one year will have been nuclear power plant experience. This program is approximately one year in duration and includes supervised on-the-job training in major operator positions (excluding fossil boiler related positions) at a PP&L conventional power plant. Also included in the one year experience program are approximately ten weeks of formal classroom training which includes but is not limited to the following areas:

Basic Power Plant Operation.

Steam Turbine Fundamentals

Power Plant Mathematics

Basic Thermodynamics and Fluid Mechanics

Plant Cycle and Plant Performance

Basic Electrical and Plant Instrumentation

Basic Print Reading

Basic Water Chemistry

Introduction to Nuclear Power and Nuclear Plant Systems

o Phase II - Academic Program for Nuclear Power Plant Personnel

This course is conducted by the General Physics Corporation of Columbia, Maryland. It is designed to refresh basic courses received in high school and to acquaint those personnel, with little or no nuclear background, with nuclear phenomenon and the BWR concept as they apply to practical reactor technology. The course material and the approximate number of classroom hours allotted to each major topic are as follows:

Subject	Classroom Hours
First Segment - Mathematics and Classical Physics	200
Review of Introductory Mathematics	16
Exponents and Logarithms	36
Algebra	64
Geometry and Trigonometry	24
Mathematics of Dynamic Systems	20
Classical Physics	40
Second Segment - Physics	200
Atomic Physics	24
Nuclear Physics	60
Reactor Core Physics	68
Reactor Operations	48
Third Segment - Related Technologies	200
Introduction to Nuclear Power Plant Systems	28
Chemistry	28
Health Physics	56
Fundamentals of Electricity and Electronics	48
Nuclear Instrumentation	40
Fourth Segment - Nuclear Power Plant Technology	200
Theory and Application of Nuclear Power Plant Systems	88
Physics Review	56
Overall Nuclear Power Plant Operations	56
	800

Cold license applicants, with no previous nuclear experience, will be assigned to a Research Reactor Training Course conducted by the Pennsylvania State University. This 2-week, course gives the student actual hands-on experience with an open pool nuclear reactor and allows the cold license applicant to obtain at least the minimum of 10 reactor startups necessary to establish cold license eligibility requirements. The course includes, but is not limited to, the following subject material:

- Reactor Operations
- Fuel Handling
- Flux Mapping
- Normal Reactor Operation

- o Instrumentation Effects
- o Control Rod Calibration
- o Laboratory Demonstrations, and
- o Control Transient Effects

o <u>Phase III - Basic BWR Technology</u>

The Basic BWR Technology course is designed to impart the details of the BWR nuclear steam supply system to the operator trainees.

The course consists of approximately 5 weeks of classroom lecture on BWR nuclear steam supply system components, fuel description, thermal-hydraulics, radiation monitoring and nuclear instrumentation system operations. Important interfaces with the balance of plant systems are also taught.

The lectures are presented by GE BWR Training personnel using conventional classroom techniques. Classes are scheduled for approxmately 7 hours per day and suggested study assignments are normally made daily. Progress is measured by weekly written and final comprehensive examinations.

It is anticipated that the course material covered will be as follows.

Schedule changes and adjustments to course content will be made as necessary to meet the particular needs of the students

Week 1

Introduction to Course
Plant Orientation
Reactor Principles Review
Reactor Vessel and Internals
BWR Thermal Hydraulics Review
Fuel Description
Nuclear Boiler Instrumentation

Week 2

Examination 1
Control Rod Drive Mechanism
Control Rod Drive Hydraulic System
Rod Control and Information System
Rod Pattern Control System
Recirculation System
Recirculation Flow Control System
Reactor Water Cleanup System

Source Range Monitoring System
Intermediate Range Monitoring System
Local Power Range Monitoring System
Average Power Range Monitoring System

Week 3

Examination 2
Traversing In-Core Probe System
Main Steam System
Reactor Pressure Control (Electro-Hydraulic Control)
Feedwater Control System
Reactor Protection
Containment and Related Systems
Introduction to Radwaste Systems (Off Gas, Liquid and Solid Radwaste)

Week 4

Examination 3
Introduction to Electrical Distribution
Reactor Core Isolation System
Introduction to Emergency Core Cooling System
High Pressure Core Spray System
Auto Depressurization System
Low Pressure Core Spray
Residual Heat Removal System and Hot Standby Operation
Emergency Core Cooling Systems Integrated Response
Standby Liquid Control System
Process Radiation Monitoring
Area Radiation Monitoring

Week 5

Examination 4
Performance Monitoring System
BWR Materials
BWR Chemistry
Fuel Pool Cooling System
Reactor Refueling
Plant Operations
Transient Analysis
Review

Final Examination

o Phase IV - BWR Simulator Training

The BWR simulator course is taught at the General Electric BWR Training Center, Morris, Illinois, and is designed to provide the operator trainee with the skills necessary to safely operate a large Boiling Water Reactor power plant.

The course consists of approximately 12 weeks of classroom lectures, simulator control room exercises, and in-plant oral seminars. This combination of instructional techniques affords the optimum mixture for successful skill training. The final examination consists of written, control room performance, and plant oral examinations.

Lectures and exercises are presented and guided by qualified, GE BWR Training Personnel. Classroom lectures are scheduled for approximately 8 teaching hours per day. Suggested reading and study assignments are made daily; written examinations are given weekly to monitor progress. In addition, at approximately the mid-point of the course, oral examinations are given to monitor the progress of each student's skill acquisition. The control room portion of the course is normally accomplished on night shifts of 8 hours . Four hours are spent in the simulator control room (total approximately 112 hours) with exercises and demonstrations guided by the licensed instructor. The other 4 hours are devoted to oral seminars . Each student rotates to appropriate control room operating positions, including shift supervisor, so that all personnel have equal opportunity to perform plant evolutions from each operating position.

The following is an anticipated week-by-week schedule of the course. Schedule changes and adjustments to course content will be made as necessary to meet the particular needs of the students.

Week 1

Introduction to the BWR Training Center
Reactor Vessel and Internals
Reactor Fuel
Nuclear Boiler Instrumentation
Control Rod Drive Mechanism
Control Rod Drive Hydraulics
Reactor Manual Control
Recirculation System
Recirculation Flow Control
Reactor Water Cleanup System
Shutdown Cooling and Head Spray
Source Range Monitoring (SRM)
Intermediate Range Monitoring (IRM)

Local Power Range Monitoring (LPRM)
Average Power Range Monitoring (APRM)
Rod Block Monitor

Week 2

Week 1 Examination
Traversing In-Core Probe (TIP)
Rod Worth Minimizer (RWM)
Main Steam
Turbine and Lube Oil System
Electro-Hydraulic Control System (EHC)
EHC Pressure Control and Logic
Condensate and Feedwater
Feedwater Control
Circulating Water
Generator and Auxiliaries
Generator Excitation
AC Electrical Distribution
Diesel Generators and DC Electrical Distribution
Reactor Protection System (RPS)
Primary and Secondary Containment

Week 3

Week 2 Examination
Fuel Pool Cooling and Cleanup
Off Gas System
Liquid Radwaste
Water Systems
Isolation Condenser
Introduction to Emergency Core Cooling System (ECCS)
High Pressure Coolant Injection (HPCI)
Automatic Depressurization System (ADS)
Low Pressure Coolant Injection (LPCI)
Core Spray
Emergency Core Cooling System Integrated Response
Standby Liquid Control
Process Radiation Monitoring
Area Radiation Monitoring
Reactor Physics Review

Week 4

Pre-Start and Functional Checks
Reactor Startups
Heatups
Manipulation of Auxiliary Systems
Power Changes in the Intermediate Range
Surveillance Testing
Transfer to Run Mode

Turbine Warmup and Roll

Week 5

Reactor Heatup and Transfer to Run Mode
Turbine Roll
Generator Synchronization and Loading
Surveillance Testing
Continued Loading to 100% Power
Operations at Full Power
Transient Analysis
Quiz 1
Maneuvering by Flow Control
Reactor Shutdown
Discussion on Decay Heat Operation and Removal
Plant Problems
Drills on Abnormal and Emergency Conditions

Week 6

Pre-Startup and Functional Checks
Reactor Startups and Heatups
Manipulation of Auxiliary Systems
Plant Problems
Drills on Abnormal and Emergency Conditions
Power Changes in the Intermediate Range
Surveillance Testing
Transfer to Run Mode
Turbine Warmup and Roll
Operator Synchronization and Loading
Quiz 2
Mid-Course Performance Examination

Week 7

Technical Specificatons Bases Review
Review Certification Exam Format and Content
Physics Problem Solving
Mid-Course Control Room Checks
Solid Radwaste
Health Physics Review
BWR Chemistry
Thermal-Hydraulics
Process Computer
Circuit Breaker Control
Fuel Handling and Fuel Loading Physics

Week 8

Steady-State Operation at 50% Load
Surveillance Testing
Increase to Full Load
Drills on Abnormal and Emergency Conditions
Operations at Full Power
Maneuvering by Flow Control
Begin Reactor Shutdown
Reactor Shutdown and Cooldown
Flooding of Reactor Vessel
Plant Problems
Reactor Startups and Heatups
Scram and Scram Recoveries

Week 9

Operation at Full Load
Drills in Abnormal and Emergency Conditions
Shutdown to Hot Standby
Quiz 3
Plant Startup from Hot Standby to Full Power
Reactor Heatup
Generator Synchronization and Loading

Week 10

Operation at 50% Load
Scrams and Scram Recoveries
Surveillance Testing
Operation at Full Power
Drills
Individual Student Operations
Quiz 4
Transient Analysis Review

Week 11

Review and Study
Reactor Operator Certification Examination

Week 12

Control Room and Dresden Plant Oral Examination
Control Room Performance Demonstration
Senior Reactor Operator Certification Examination

o Phase V - BWR Observation Training

BWR observation training is designed to acquaint the operator trainee with the day-to-day routine of an operating BWR. This will involve exposure to plant operating and maintenance evolutions, station record keeping, and procedures.

The course consists of approximately 4 weeks of guided observation of an operating BWR. All observation is conducted under the guidance of an experienced GE training personnel.

The course is structured to provide experience in various aspects of plant operation. The flexibility is achieved by allowing the course director to adjust the group schedule to fit important plant evolutions. Daily work and observational assignments are made at the beginning of each work day.

The following are weekly highlights of a typical BWR observation schedule:

Week 1

Plant Evacuation Procedures/Station Emergency Plan
Health Physics Procedures
Electrical Distribution
Reactor Instrumentation
Control Rods and Hydraulic Drive System
Recirc MG set, support systems, and controls
Main Steam System Controls and Instrumentation
Residual Heat Removal System - All Modes

Week 2

Turbine, EHC System, and Turbine Support Systems
Generator, Generator Excitation, and Generator Support Systems
Turbine and Reactor Building Closed Cooling Water System
Circulating and Service Water Systems
Fire Protection Systems
Core Spray System

Week 3

High Pressure Coolant Injection System
Reactor Core Isolation Cooling System
Reactor Protection MG sets
Automatic Depressurazation System
Traversing In-core Probe System
Neutron Monitoring and Associated Control Systems
Radioactive Waste Handling Equipment and Procedures
Performance of Routing Plant Equipment Checks

Week 4

Instrument and Service Air Systems
Process and Area Radiation Monitoring Systems
Fuel Pool Cooling System
Standby Liquid Control System
Plant Performance Logs
Observance of Routine Plant and/or Surveillance Procedures In Progress
Review
Final Exam and Walk-Through

o Phase VI - SYSTEMS, PROCEDURES, and ON-THE-JOB TRAINING

The systems, procedures, and on-the-job training phase will be approximately 20 weeks in length of which a minimum of 8 weeks will be class room instruction. However, the weeks may not be scheduled consecutively due to plant testing and work load considerations. This phase will provide cold license candidates with an in-depth study of Susquehanna SES systems and equipment; nuclear characteristics; and Normal, Abnormal, Emergency and Administrative Procedures and Technical Specifications. Further operational training is accomplished as components, systems, or parts of systems are checked, tested, and placed in routine operation to provide necessary auxiliary support for other systems.

Instructors for the various Phase VI lectures will be supplied by the Susquehanna staff, other PP&L organizations, vendors or consultants. Selections of the particular individual to conduct a specific training lecture will be based upon individual availability and knowledge of the subject matter involved.

o Phase VII - BWR Pre-License Refresher Training

By the time an applicant for the NRC's Senior Reactor Operator or Reactor Operator License completes their training, a time span of about 3 years will have elapsed. Because of the long lead time required for these training programs a Pre-licensing Refresher Course will be conducted. This course will be approximately 2-4 weeks in duration and will be scheduled to end about 3 months before initial fuel loading. The course will be presented by PP&L employees or an outside organization. An examination will be given at the end of the refresher training.

The course will consist of, but not be limited to:

a. Theory and principles of operation

b. General and specific plant operating characteristics

c. Plant instrumentation and control systems

d. Plant protection systems

e. Normal, abnormal, and emergency operating procedures

f. Radiation control and safety

g. Technical Specifications

h. Applicable portions of Title 10 Chapter 1 Code of Federal Regulations

13.2.1.1.2.2 Non-Licensed Operator Training Program

The program is designed for non-licensed operators and is divided into three phases which provide a logical progression from the entry level to final job qualification.

- o. Phase I - Academic Training
- o. Phase II - Susquehanna SES System Lectures
- o. Phase III - Susquehanna SES System Qualification

This training is progressive and candidates for non-licensed positions must successfully complete the training appropriate to their assigned job. Phase I may be exempted by passing a written exam.

- o Phase I: The course consists of basic training in Nuclear Power Plant Fundamentals. The program is about 160 hours long and consists of classroom training or equivalent self-study time. The areas to be covered will include such subjects as math, chemistry, atomic and nuclear physics, health physics, nuclear instrumentation and reactor operations. Progress is measured by periodic quizzes and examinations.

- o Phase II - The phase consists of basic lectures on Susquehanna SES systems and covers, as applicable, the following areas of each system:

  General System Description
  Major Components and Flow Paths
  Instruments and Controls
  Alarms and Trips
  Power Feeds
  Operating and Emergency Procedures

The phase will be approximately 4 weeks in length and during each week, approximately 80% of the time will be spent in class and the remaining 20% will be spent in the plant tracing systems. There will be weekly quizzes and a final exam at the end of the course.

o Phase III - This phase must be completed by operators on the systems for which they are responsible. This phase will take about 10 weeks to complete. However, the 10 weeks may not be consecutive due to work-load considerations Operators will be checked out on each system to assure they can operate these systems under normal, abnormal, and emergency situations. The check out will consist of an oral and/or written test on each system.

13.2.1.1.3 Maintenance Section Training Program

The Supervisor of Maintenance will receive training Level III Health Physics training as described in Subsection 12.5.3.7, selected training in plant systems operation and specialized vendor training on specific plant equipment.

Foremen will receive additional experience on-the-job during the preoperational test program through the supervision of maintenance activities.

Station Mechanics and Leaders for the initial plant staff will generally be selected from other PP&L facilities and will have practical experience in one or more crafts, and will through their previous experience and/or selection testing demonstrate a high degree of manual dexterity and capability of learning and applying the basic skills in maintenance operations.

Maintenance personnel will receive on-the-job training during the preoperational test program by performing maintenance activities. Selected personnel will receive specialized vendor training on specific equipment or skills such as control rod drive repair and welding.

Maintenance personnel requiring access to Radiation Work Permit Areas will receive Level II Health Physics training as described in Subsection 12.5.3.7.

13.2.1.1.4 Technical Section Training Program

The objective of the initial training program of the Technical Section is to provide competent personnel to support in the safe, efficient operation of the Susquehanna SES.

Selected supervisory and professional/technical personnel will attend GE's Design Orientation courses (or other formal instruction with a similar intent) to familiarize them with the design principles of a BWR. The major topics covered will include BWR components, core design, thermal-hydraulics, process and nuclear instrumentation design and operation and auxiliary systems.

13.2.1.1.4.1 Chemistry Personnel

By initial fuel loading, in addition to those courses described in Subsection 13.2.1.1.4, selected chemistry supervisory personnel will receive specialized training through a course such as "BWR Chemistry" offered by GE. The course enables students to complete both radiological and chemical analyses for process control, waste disposal, effluent monitoring, process and laboratory instrument calibration and evaluation. The course also covers compliance with and interpretation of chemical and radiochemical aspects of the technical specifications, licenses and plant warranties.

The Chemistry Leaders and Chemistry Analysts will receive in-house training as developed by supervisory chemistry or other appropriate personnel, covering topics similar to those in the "BWR Chemistry" course. As appropriate, they may also attend vendor-sponsored training sessions to assure understanding and proper operation of laboratory instruments. Progress will be measured through oral and/or written examinations.

13.2.1.1.4.2 Instrumentation & Control Personnel

By initial fuel loading, in addition to those courses described in Subsection 13.2.1.1.4, appropriate I&C supervisory personnel and selected I&C technicians will attend the GE "Nuclear Instrumentation" and "Process Instrumentation and Controls" courses or other formal instruction with similar intent.

The "Nuclear Instrumentation" course is broken into classroom and laboratory phases. The classroom phase covers the theory of operation and equipment demonstrations for the GE BWR nuclear, process and area radiation monitoring, control rod position information, reactor protection and traversing incore probe systems. The laboratory phase teaches detailed circuitry study, setup, calibration, testing, maintenance and repair for the various components of these systems and where possible for the overall system.

The "Process Instrumentation and Control" course teaches the theory of operation, setup, calibration, testing, maintenance and repair techniques for the basic instrumentation and control loop components for the GE BWR. Components to be covered include level, temperature, electrical properties, movement, chemical properties, sensing devices, transmitters, power supplies, signal conditioning modules and controllers. Primary instrument control loops will also be studied.

I&C technicians will also receive training covering topics such as AC/DC circuit fundamentals, transistor circuits, solid state devices and operational amplifiers and including "hands-on" experience with electrical and electronic circuits and components. As necessary, I&C personnel will attend courses offered by equipment vendors on various plant components.

Progress will be measured through oral and/or written examinations.

13.2.1.1.4.3 Reactor Engineering Personnel

By initial fuel loading, in addition to the courses described in Subsection 13.2.1.1.4, selected Reactor Engineering personnel will receive training through a course such as GE's "Station Nuclear Engineering". The course covers topics like reactor behavior, control rods, shutdown margins, technical specifications and Fuel Warranty Operation Provisions, core flow and thermal limit calculations, fuel failure and PCIOMR and water chemistry among others.

Progress will be measured through oral and/or written examinations.

13.2.1.1.5 Health Physics Training Program

Selected Health Physics supervisory personnel will receive specialiezed professional training in a course such as "Radiological Engineering" offered by GE or equivalent.

Health Physics Monitor Training Program is described in Subsection 12.5.3.7.

13.2.1.1.6 General Employee Training

All permanent plant personnel granted unescorted vital area access at the station will be trained in the following areas:

1. Appropriate plans and procedures, including applicable plant security and emergency procedures.

2. Radiological Health and Safety in accordance with Subsection 12.5.3.7

3. Industrial Safety.

4. Fire Protection Program.

5. Quality Assurance Program.

This training will be the responsibility of the Plant Training Supervisor and will be repeated on a two-year cycle. Personnel will be examined in the above areas to determine the effectiveness of general employee training.

Temporary Maintenance and Service personnel will be trained in the areas listed to the extent necessary to assure safe execution of their duties.

13.2.1.1.7 Fire Safety Training

The object of the fire safety training program is to provide training for the plant fire brigade, training for maintenance and inspection of fire protection equipment and training for the fire protection staff.

13.2.1.1.7.1 Fire Brigade Training

In addition to general employee training, individuals assigned to fire fighting duties will receive training in order that an effective fire fighting brigade will be available for fire emergencies. FIre brigade training sessions will be held a minimum of four times per year, with the basic program being repeated every two years. Training will be a blend of classroom sessions, practice sessions and fire fighting drills. Fire brigade members will receive instruction on fixed and portable fire fighhting equipment, fire protection measures of other plant features and will be trained in hands on experience with fire fighting equipment and techniques. The local fire departments will be invited and encouraged to participate in at least one training session per year.

13.2.1.1.7.2 Fire Protection Staff and Training

No training program is planned for the off site fire protection engineer. The position description requires that the incumbent be a qualified fire protection engineer with suitable background experience to meet the job requirements. A major part of the on site fire protection engineer's training will be on the job training and informal discussions with the off site fire protection engineer. This training will be augmented by vendor training schools and state fire fighting schools as necessary to carry out the job responsibilities. The on site fire protection engineer will have the responsibility of training or arranging for the training of fire brigade personnel, on site fire department training and training of personnel in charge of the maintenance of fire detection and fire suppression systems.

13.2.1.2 Coordination with Preoperational Tests and Fuel Loading

Figure 13.2-1 illustrates the relationship of the Plant Staff Training to preoperational testing and fuel loading.

13.2.2 REQUALIFICATION AND REPLACEMENT TRAINING

13.2.2.1 Licensed Operator Requalification Program

The Requalification Program for licensed and senior licensed individuals will be established and ready for implementation no later than 3 months following issuance of the station operating license.

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The Requalification Program is based on a two (2) year cycle and will be followed by successive two (2) year programs that will meet or exceed the requirements set forth in 10CFR55 Appendix A.

13.2.2.1.1 Preplanned Lecture Series

The requalification program will include preplanned lectures on a regular and continuing basis throughout the license period. Annual operator and senior operator written examination results will indicate the scope and depth needed in the following subjects:

a. Theory and principals of operations.

b. General and specific plant operating characteristics.

c. Plant instrumentation and control systems.

d. Plant Protection Systems.

e. Normal, abnormal, and emergency operating procedures.

f. Radiation control and safety.

g. Technical specifications.

h. Applicable portions of Title 10, Chapter 1, Code of Federal Regulations.

The lectures are presented or directed by a member of the station staff or a designated SRO/RO licensee with outside assistance scheduled as desired or as necessary. The selective use of films, videotapes, and individual study material are utilized to supplement the lectures. Videotapes or film presentations will not be used for more than 50% of the lecture series. A minimum of 60 hours. of lectures are scheduled annually, with the lectures spread reasonably throughout the year. Lectures may be deferred due to unanticipated station operating requirements; however, these lectures will be rescheduled for a later date within that requalification cycle.

13.2.2.1.2 On The Job Training

The on the job training portion of the requalification program will be covered under five segments.

13.2.2.1.2.1 Plant Evolutions

During the two year license period, each operator is required to manipulate facility controls through at least ten reactivity changes and each senior operator is required to manipulate or direct the manipulation of controls through a like number of reactivity manipulations from the applicable unit categories listed below. The evolutions will be a combination of manipulations distributed among various categories consistent with plant and personnel scheduling requirements, and will be selected from at least three categories. Staff members, including Shift Supervisors, will be encouraged to periodically take part in hands on control of plant systems.

The following are the unit categories:

1. Plant or reactor startup to include a range such that reactivity feedback from heat is noticeable

2. Plant shutdowns

3. Control rod sequence changes

4. Shutdown margin checks

5. Control rod scram insertion time tests

6. Any manual reactor power change of greater than or equal to 10%

7. Plant and reactor operation that involves emergency or transient procedures where reactivity is changing.

8. Refueling operations where fuel is moved in the core.

Completion of these items will be documented in the individual's training records. Should an individual fail to complete ten evolutions, completion may be met through simulator training described in Subsection 13.2.2.1.2.5.

13.2.2.1.2.2 Design, Procedure, and Facility License Change Review

This segment of the program will ensure that all changes or revisions to the license document, i.e., Operating Technical Specifications and Environmental Technical Specifications are reviewed. It further ensures that significant changes to procedures and completed facility design changes will also be reviewed.

When determined by the Supervisor of Operations and/or Plant Training Supervisor that a procedure revision or completed facility design change may effect the operations of the plant, it is then included as part of the review process.

The requirements of this program will be met by placing a copy of the change or revision, along with a Document Acknowledgement Sheet in the review notebook maintained in the control room. Periodically each license holder will review the contents of the notebook and upon completing the review will acknowledge, by signature and date, his understanding of the change. A file of completed Document Acknowledgement sheets will be maintained as a record of program compliance.

Changes that require immediate operator notification, i.e., interim changes will be covered by night orders to the shift supervisor or during an operations department meeting followed up by a document acknowledgement under this program.

13.2.2.1.2.3 Emergency Procedure Review

In order to ensure a continuing awareness of the action and responses necessary during abnormal and emergency situations each licensed individual will review the contents of all emergency procedures annually. To gain additional shift continuity in the overall response to these emergency situations an on-shift discussion is encouraged.

13.2.2.1.2.4 Operator Proficiency Evaluation

At least once per year during the term of an individual's license, he will be observed and formally evaluated while responding to actual or simulated casualties. In the case of simulated casualties, a hypothetical situation would be presented followed by a discussion of plant and operator response. Actual manipulation of plant controls is not required. A poor performance on two or more evaluations where "poor" is defined as not being able to competently and expeditiously perform the specified evolution will result in a Performance Review as specified in Subsection 13.2.2.1.4. All evaluations will be critiqued with the individual concerned and filed in the individual's training records.

13.2.2.1.2.5 Simulator Training

When necessary to fulfill the requirements of 10CFR55, App. A, and in particular to meet the retraining requirements specified in Subsections 13.2.2.1.2.1 and 13.2.2.1.2.4 each license holder may be required to attend an NRC approved simulator requalification program. The controls and characteristics of the simulator will reasonably approximate those of the facility for which the operator license is held. Approved simulator programs generally consist of intensive training emphasizing those evolutions, transients and casualties not normally experienced during routine power plant operation. Job responsibilities will be rotated such that each individual has an equal opportunity to manipulate plant controls as well as to direct responses to plant conditions. This training program will include conditions and parameters used in casualty evaluation as well as the actions to be followed for corrective action. A formal evaluation will be prepared concerning each individual's performance during the simulator training. The evaluation will be critiqued with the operator emphasizing any weak areas. Simulator evaluation sheets will be filed in the individual's training folder.

13.2.2.1.3 Annual Exam

Each licensed individual shall be administered a comprehensive examination which shall parallel in content and degree of difficulty an NRC licensing exam. The exam will be graded by category. Individuals scoring 80% or greater on a particular category will be exempt from attending retraining lectures covering that category. Any licensed operator with an average exam score of less than 70% shall be relieved of licensed responsibilities and undergo a performance review in accordance with Subsection 13.2.2.1.4. Each license holders' graded exam shall be retained as part of the training records. The results of the exam will be used to evaluate past, and to aid in determining future, retraining.

An individual who prepares, administers, or grades a written examination need not take the examination. A maximum of three licensed personnel may be exempted under this condition.

Retaining lectures may use training aids such as video tapes and films in lieu of an instructor. However, no more than 50% of the lectures may be substituted for by such aids.

13.2.2.1.4 Performance Review Program

The Performance Review Program will be implemented when an individual's performance falls below the minimum standards specified. Specifically whenever any one of the following situations occur the associated individual will undergo a performance review.

a. An average annual exam score of less than 70%

b. "Poor" rating on operator proficiency evaluations

c. Prolonged absence from license responsibilities

The Superintendent of Plant, Supervisor of Operations and Plant Training Supervisor will meet to determine a course of action necessary to upgrade the individual's performance to an acceptable level. This review shall determine the minimum requirements on an individual basis. The severity of the action taken will be dependent on such factors as examination performance, operating performance, observed operational and theoretical understanding and judged overall operator competence. If there is doubt concerning the individual's ability to safely operate the plant, he would be removed from licensed responsibilities pending satisfactory completion of the program specified by management review. When receiving less than 70% overall average on the annual written examination, an oral examination may be administered to determine whether or not an individual may resume licensed duties. However, the individual shall remain on the Performance Review Program until a score of 70% or better average is obtained on a written examination which specifically covers those areas of the exam where the licensee receives less than 70%. Management will complete a performance review summary which upon completion will be filed in the individual's training folder.

13.2.2.1.5 Prolonged Absence from License Responsibilities

In the event a licensed individual is absent from the site for a period of four months or longer, he will not be permitted to resume operational responsibilities until the following criteria has been met:

a. The satisfactory completion of an upgrading program determined under the provisions of SEction 13.2.2.1.4.

1. This program shall include as a minimum, the review of any facility design, operating procedure or

license changes which have taken place during the absentee period.

2. The individual must receive greater than 70% on a written or satisfactory on an oral exam covering the material in item 1 above. When an oral exam is used it shall be administered by the Supervisor of Operations or designated Senior Licensee.

b. This certification shall be documented in the individuals training record.

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13.2.2.1.6 Record Retention

The Plant Training Supervisor will be responsible for maintaining license holder requalification training records. Records of the requalification program shall be maintained to document each licensed operator's and senior operator's participation in the requalification program. The records shall contain copies of written examinations administered, the answers given by the licensee, results of evaluations and documentation of any additional training administered in areas in which an operator or senior operator has exhibited deficiencies. Records shall be maintained for the lifetime of the plant.

13.2.2.1.7 Licensed Staff Participation

Licensed staff personnel participate in the following areas of the requalification program:

a. Complete the annual written examination and participate in the lecture series based on the results.

b. Manipulate the controls or supervise the manipulation of controls through 10 reactivity changes.

c. Review day-to-day changes in the facility design procedures, and technical specifications.

d. Review abnormal and emergency procedures annually.

e. Are evaluated regarding actions to be taken during simulated abnormal and emergency conditions by a walk-through of the applicable procedure.

13.2.2.2 Refresher Training for Nonlicensed Personnel

As a minimum, all non-licensed personnel shall receive refresher instruction on administrative, radiation protection, emergency and security procedures once every two years.

13.2.2.2.1 Refresher Training for Nonlicensed Operators

Nonlicensed operators assigned on shift will participate in a requalification program and be trained, tested, and evaluated on a two year schedule.

13.2.2.2.2 Refresher Training for Maintenance Personnel

A retraining program is provided for maintenance personnel to ensure that they remain proficient in their particular job.

Retraining in specific areas is provided to the extent necessary for personnel to safely and efficiently carry out their assigned responsibilities in accordance with established policies and procedures.

Such training may consist of vendor presentations, technical training sessions, on-the-job work experience or programmed instruction. Maintenance personnel are evaluated on an annual basis where individual needs for retraining will be identified.

13.2.2.2.3 Refresher Training for Technical Section Personnel

Refresher courses will be provided to maintain an individual's level of expertise equal to or exceeding that required by his or her job responsibilities.

13.2.2.3 Replacement Training

Replacement training is designed to supply qualified personnel for all levels of the plant organization. It is the policy of PP&L to promote qualified personnel into job vacancies from candidates that are next in the line of promotion. Such individuals will receive training appropriate to the new position. Permanent replacement personnel procured from other sources will meet or exceed the requirements of the vacant position.

13.2.2.3.1 NRC Licensed Operator Replacement

Through a system of required operator qualification steps, personnel assigned as nonlicensed operators are provided training that prepares them for eventual NRC licensed operator positions. At such time as a need exists for replacement of licensed operators, individual preparation in theory, systems, operating procedures, emergency procedures, and health physics is conducted. Additionally, individual on-the-job training involving manipulation of the nuclear reactor plant controls during day-to-day operation, startups and shutdowns of the reactor or appropriate reactor simulator, is conducted.

Progress is reviewed as the replacement moves through the program. The review consists of periodic written and/or oral examinations.

13.2.2.3.2 Non-licensed Operator Replacement Training

Replacement training is designed to insure fully qualified personnel for all levels of plant operation. To the extent possible, persons who have already achieved the level of training required for a specific job will be advanced. In all cases the requirements of Subsection 13.2.1.1.2.2 or equivalent will be satisfied.

13.2.2.3.3 Maintenance Personnel Replacement Training

Replacement training is designed to supply qualified personnel for all levels of the Maintenance organization. It is the policy of PP&L to promote qualified personnel into job vacancies for candidates that are next in the line of promotion. Permanent replacement personnel will meet or exceed the requirements of the vacant position by virtue of previous education and experience. The Maintenance organization is specifically intended to provide the opportunity for personnel in lower level jobs to receive on-the-job-training that will prepare them for advancement.

The same quality of training provided for the original staff will be provided to personnel designated to fill vacancies in the maintenance organization.

13.2.2.3.4 Technical Section Personnel Replacement Training

Replacement training is designed to assure fully qualified personnel for all levels of the Technical Section. To the extent possible, persons who have already achieved the level of training required for a specific job will be advanced. In all cases the requirements of Subsection 13.2.1.1.4 or equivalent will be satisfied.

13.2.2.4 Records

Training records are established for each permanent plant employee. These records include, but are not limited to, lecture/annual examination questions and answers , lecture attendance records, performance evaluation records, and other records as may be required to adequately document all training received by station personnel.

Training records will be periodically reviewed in accordance with station procedures to assess the effectiveness of the training program.

13.2.2.5 Responsible Individual

The Plant Training Supervisor is responsible for the administration and conduct of the Susquehanna SES training program.

CHAPTER 14.0

INITIAL TESTS PROGRAM

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14.2 SPECIFIC INFORMATION TO BE INCLUDED IN FINAL SAFETY ANALYSIS REPORT

14.2.1 SUMMARY OF TEST PROGRAM AND OBJECTIVES

As construction of systems/components is completed, the construction organization relinquishes jurisdictional control of these systems/components through a formal turnover to PP&L. Eventually all plant systems/components are turned over to PP&L.

The Initial Test Program encompasses the scope of events that commence with system/component turnover and terminate with the completion of power ascension testing. The Initial Test Program is conducted in two separate and sequential subprograms, the Preoperational Test Program and the Startup Test Program. At the conclusion of these subprograms the plant is ready for normal power operation. Testing during the Initial Test Program is accomplished in five distinct and sequential phases:

a. Phase I - Component Inspection and Testing Phase

b. Phase II - Preoperational and Acceptance Testing Phase

c. Phase III - Initial Fuel Loading Phase

d. Phase IV - Initial Heatup and Low Power Testing Phase

e. Phase V - Power Ascension Test Phase

Phase I and Phase II are sequential on a system basis while Phases III, IV and V are sequential on a plant basis.

14.2.1.1 Preoperational Test Program

The Preoperational Test Program is defined as that part of the Initial Test Program that commences with system/component turnover and terminates with commencement of nuclear fuel loading. The program is subdivided into two phases in which plant equipment and systems are prepared for a higher degree of operability. The phases are:

1) Component Inspection and Testing Phase (Phase I)

2) Preoperational and Acceptance Test Phase (Phase II)

Component inspection and testing will insure that components and equipment are calibrated and checked, construction work on a particular system has been completed to the degree required and the system is initially operated and prepared for subsequent testing. After component inspection and testing is complete on a system, formal tests denoted as preoperational or acceptance tests are conducted during the Preoperational and Acceptance Test Phase. The Preoperational tests demonstrate, to the extent practicable, the capability of safety-related structures, systems, and components to meet their safety-related performance requirements. The completion of preoperational testing constitutes completion of Phase II of the Initial Test Program. Tests similar to preoperational tests denoted as acceptance tests (Table 14.2-2), may be conducted on additional non safety-related structures, systems, and components to demonstrate their capability to perform their nonsafety-related performance requirements.

To the extent practicable, the objectives of the Preoperational Test Program are to:

a. Verify the adequacy of plant design

b. Verify that plant construction is in accordance with design.

c. Demonstrate proper system/component response to anticipated transients and postulated accidents

d. Confirm the adequacy of plant operating and emergency procedures

e. Familiarize plant staff operating, technical, and maintenance personnel with plant systems

14.2.1.2 Startup Test Program

The Startup Test Program is defined as that part of the Initial Test Program that commences with the start of nuclear fuel loading and terminates with the completion of power ascension testing. Formal tests, denoted as startup tests, are conducted during this program. These tests confirm the design bases and demonstrate, to the extent practicable, that the plant will operate in accordance with design and is capable of responding as designed to anticipated transients and postulated accidents. Startup testing is sequenced such that the safety of the plant is never totally dependent upon the performance of untested structures, systems, or components. The completion of startup testing constitutes completion of Phases III, IV, and V of the Initial Test Program.

The objectives of the Startup Test Program are to:

a. Accomplish a controlled, orderly, and safe initial core loading

b. Accomplish a controlled, orderly, and safe initial criticality and heatup

c. Conduct low power testing sufficient to ensure that design parameters are satisfied and safety analysis assumptions are correct or conservative

d. Perform a controlled, orderly, and safe power ascension

14.2.2 ORGANIZATION AND STAFFING

The Superintendent of Plant - Susquehanna, has overall responsibility for the Initial Test Program. The Plant Staff and Integrated Startup Group (ISG) conduct the different phases of the test program. In addition to these basic organizational units the Superintendent of Plant - Susquehanna is assisted by two review organizations, the Plant Operations Review Committee (PORC) and the Test Review Board (TRB). The organization, authority, responsibility, and degree of participation of each of these organizational units during the Initial Test Program are described in the following sections.

14.2.2.1 Plant Staff

The Plant Staff consists of the permanent onsite PP&L personnel responsible for the safe operation and proper maintenance of the plant. Chapter 13 describes the Plant Staff organization. This section also establishes responsibilities, reporting relationships, and minimum qualification requirements for principal Plant Staff supervisory personnel.

The Plant Staff is utilized, to the fullest extent practicable, during the Initial Test Program. Specific responsibilities of the Plant Staff during the Initial Test Program are:

a. Performing selected preventive and corrective maintenance.

b. Operating plant equipment.

c. Calibrating instruments, meters.

d. Performing chemical and radiological inspections and tests

e. Providing required replacement and spare parts

f. Providing operator, technician, and maintenance support to the ISG

g. Ensuring that vendors, consultants, or other temporary personnel assisting the Plant Staff work in accordance with established project procedures

h. Confirming the adequacy of plant operating and emergency procedures to the extent practicable.

i. Authorizing and ensuring proper documentation, identification, and restoration of temporary modifications made during the Startup Test Program.

j. Authorizing and monitoring rework, modification, testing and maintenance during the Startup Test Program.

k. Coordinating preparation, review and approval of startup test procedures.

l. Coordinating performance of startup testing.

m. Coordinating review and approval of startup test results.

n. Planning and scheduling Startup Test Program activities.

14.2.2.2 Integrated Startup Group - Organization and Responsibilities

The Integrated Startup Group (ISG) is a temporary organizational unit established to augment the Plant Staff during the Initial Test Program. The ISG is comprised of individuals of various organizations (Bechtel, General Electric, PP&L, and others). Figure 14.2-1 shows the organizational structure of the ISG. The responsibility and qualification requirements of principal ISG supervisory personnel, the structure of the basic constituents comprising the ISG, and the responsibilities delegated to the ISG are described in the following sections.

14.2.2.2.1 ISG Supervisor

The ISG Supervisor has overall responsibility for supervising the conduct of the ISG. The ISG Supervisor reports to the Superintendent of Plant - Susquehanna on matters pertaining to the Initial Test Program. The minimum qualifications for the ISG Supervisor are one of the following:

a. Graduate of a four-year accredited engineering or science college or university, plus five years of experience in testing or operation (or both) of power plants, nuclear facilities, or similar industrial installations. At least two years of this experience should be associated with nuclear facilities; or if not, the individual shall have training sufficient to acquaint him thoroughly with the safety aspects of a nuclear facility.

b. High school graduate, plus ten years of experience in testing or operation (or both) of power plants, nuclear facilities, or similar industrial installations. At least two years of this experience should be associated with nuclear facilities; or if not, the individual shall have training sufficient to acquaint him thoroughly with the safety aspects of nuclear facilities.

14.2.2.2.2 Assistant ISG Supervisor

The Assistant ISG Supervisor performs a line function and reports to the ISG Supervisor. The Assistant ISG Supervisor is specifically responsible for supervision of Systems Group Leaders and assumes the responsibilities of the ISG Supervisor in his absence.

The minimum qualifications of the Assistant ISG Supervisor are the same as the ISG Supervisor and are as described in Subsection 14.2.2.2.1.

14.2.2.2.3 ISG Coordinator

The ISG Coordinator performs a staff function and reports to the ISG Supervisor. The ISG Coordinator is responsible for coordinating ISG interfacing activities with Plant Staff, Construction and various project support organizations involved in the Initial Test Program.

The ISG Coordinator is responsible for all ISG administrative and procurement activities, which includes tracking the development, review, approval and revision of all Preoperational and Acceptance Test Procedures. This also includes the development, review, approval and revision of all ISG Startup Administrative Manual and Startup Technical Manual Procedures.

14.2.2.2.4 ISG Administrative Assistant

The ISG Administrative Assistant performs a staff function and reports to the ISG Coordinator. The Administrative Assistant is responsible for all administrative activities for documents relating to ISG activities.

14.2.2.2.5 Group Leaders

Group Leaders perform line functions and report to the Assistant ISG Supervisor. Group Leaders are assigned a staff of System Startup Engineers. Group Leaders have overall responsibility for assigned systems.

14.2.2.2.6 NSSS Startup Site Representative

The NSSS Startup Site Representative performs a staff function reporting to the Assistant ISG Supervisor during the Preoperational Test Program and to the Superintendent of Plant during the Startup Test program. The NSSS Startup Site Representative is responsible for directing and coordinating activities of the GE field engineers assigned to him for the conduct of test or surveillance activities on NSSS systems.

14.2.2.2.7 LSTG Site Representative

The Large Steam Turbine Generator (LSTG) Startup Engineer performs a staff function and reports to the Assistant ISG Supervisor during the Preoperational Test Program and to the Superintendent of Plant during the Startup Test Program. He is the General Electric lead LSTG Startup Engineer and is responsible for directing and coordinating activities of the GE field engineers assigned to him for conduct of test or surveillance activities on GE supplied turbine-generator systems.

14.2.2.2.8 Responsibilities

Specific responsibilities of the ISG during the Initial Test Program are:

a. Recommending acceptance or rejection of system/component turnover to PP&L

b. Coordinating initial instrument, relay, and meter calibration

c. Coordinating initial digital and analog control loop checkout

d. Coordinating initial equipment operation

e. Coordinating system cleanliness verification after turnover

f. Ensuring that assigned vendors or other consultants perform work in accordance with approved procedures

g. Authorizing and ensuring proper identification, documentation, and restoration of temporary modifications made during the Preoperational Test Program (for selected systems/components this responsibility may be assumed by the Plant Staff prior to conclusion of the Preoperational Test Program).

h. Documenting and reporting design problems identified during the Initial Test Program until PP&L permanent plant procedures are implemented to perform this function, at which time this becomes a Plant Staff responsibility. Implementation of permanent plant procedures may be on a system, unit, or plant basis.

i. Documenting and reporting construction problems identified during the Initial Test Progran until PP&L permanent plant procedures are implemented to perform this function, at which time this becomes a Plant Staff responsibility. Implementation of permanent plant procedures may be on a system, unit, or plant basis.

j. Authorizing and monitoring rework, modification, and maintenance during the Preoperational Test Program (for selected systems/components this responsibility may be assumed by the Plant Staff prior to conclusion of the Preoperational Test Program).

k. Coordinating preparation, review, and approval of component and preoperational test procedures.

l. Coordinating performance of component and preoperational testing.

m. Coordinating review and approval of component and preoperational test results.

n. Planning and scheduling Preoperational Test Program activities.

14.2.2.3 Plant Operations Review Committee

The Plant Operations Review Committee (PORC) consists of the individuals assigned independent review responsibility in accordance with the requirements of Chapter 13. The responsibilities, reporting relationships, and qualification requirements of PORC members are also described in Chapter 13. During the Initial Test Program additional responsibilities of PORC include reviewing and recommending approval of startup test procedures prior to testing and reviewing and recommending approval of startup test results following testing.

14.2.2.4 Test Review Board

The Test Review Board (TRB) is a temporary review organization established specifically for the Preoperational Test Program. Test Review Board members may consist of individuals of various organizations (Bechtel, General Electric, PP&L, or others). The Test Review Board is responsible for review of preoperational
20 test procedures prior to testing and for review of preoperational test results after testing. The TRB recommends approval to the Superintendent of Plant.

The Superintendent of Plant is responsible for the assignment of individuals to the Test Review Board. These assignments may be on a permanent or temporary basis. The TRB Chairman is responsible for the conduct of the TRB and reports directly to the Superintendent of Plant. The minimum qualifications of the TRB Chairman are the same as identified in Subsection 14.2.2.2.1.

14.2.3 TEST PROCEDURES

The Initial Test Program is conducted in accordance with detailed component, preoperational, and startup test procedures. PP&L maintains overall responsibility for test procedure preparation, review, and approval. These activities are completed in a timely fashion to ensure that these procedures are suitable for NRC review at least 60 days prior to their intended use. 20

14.2.3.1 Procedure Preparation

Component test procedures are initially prepared by designated organizatons (Bechtel, General Electric, PP&L or others). The completed drafts are reviewed by other cognizant organizations and approved by the ISG Supervisor.

Preoperational and Startup test procedure drafts are initially prepared by designated organizations (Bechtel, General Electric, PP&L, or others) in accordance with the standard format of Figures 14.2-2A & B. The completed drafts are then reviewed by cognizant design organization representatives to ensure that test procedure objectives and acceptance criteria are consistent with current design document requirements. Review comments are resolved between the writing organization and the cognizant design organization representative.

The following items are the responsibility of the ISG for component and preoperational test procedures and the Plant Staff for Startup test procedures:

a. Updating procedure references to latest revisions.
b. Verifying the procedure has been revised to incorporate design changes.
c. Verifying procedure compatibility with field installation of equipment.
d. Resolving comments on procedures received from TRB, PORC or the Superintendent of Plant.
e. Evaluating reactor operating and testing experiences as supplied by the Manager-Nuclear Support in the development of the procedures.

14.2.3.2 Procedure Review and Approval

Following initial preparation the component tests are reviewed by cognizant organizations and sent back to the ISG for inclusion of comments. The ISG Supervisor then approves the component test procedures.

Following initial preparation, the Preoperational and Startup test procedures are processed through a formal review and approval cycle. The responsibility for coordinating this process and for resolving review comments lies with the ISG Supervisor or his designee for preoperational tests and with the Technical Supervisor or his designee for startup tests.

Specific review responsibilities are as follows:

a. For preoperational and acceptance test procedures the Test Review Board, under the direction of the TRB Chairman, is responsible for:

   1. Verifying procedure conformance with the FSAR, environmental technical specifications, and plant operating technical specifications.

   2. Ensuring technical adequacy of procedures.

   3. Recommending approval of test procedures.

   4. The Test Review Board is responsible for review of preoperational test procedures prior to testing and for review of preoperational test results after testing.

b. For the Startup Test Program test procedures the Plant Operations Review Committee, as described in Chapter 13 is responsible for:

   1. Verifying procedure conformance with the FSAR, environmental technical specifications, and plant operating technical specifications.

   2. Performing a nuclear safety review as required by the plant technical specifications.

   3. Ensuring technical adequacy of the procedures.

   4. Recommending approval of test procedures.

Upon completion of review and inclusion of required changes preoperational and startup test procedures are submitted for approval by the Superintendent of Plant.

14.2.4 Conduct of Test Program

The administrative controls that govern conduct of the Plant Staff and of the Integrated Startup Group during the Initial Test Program are specified by administrative procedures. These administrative procedures are PP&L controlled and approved documents. Administrative procedures define tasks to be performed, prescribe methods, and assign responsibilities for performing them.

The administrative procedures governing conduct of the Integrated Startup Group are contained in the Startup Administrative Manual which is approved by the Superintendent of Plant. These procedures do not establish the administrative controls of other project groups or organizations except as they interface with the Integrated Startup Group. The Startup Administrative Manual will be approved for use prior to start of the Initial Test Program. The administrative procedures governing conduct of the Plant Staff are as specified in Chapter 13. The schedule for preparation, review, and approval of these procedures is also described in Chapter 13. This schedule provides sufficient time for procedures to be available for use prior to the time they are required to be implemented.

14.2.4.1 Test Performance

Preoperational and Startup testing performed during the Initial Test Program is in accordance with approved test procedures. The method for preparing, reviewing, and approving these test procedures is detailed in Subsection 14.2.3. Prior to start of testing, a test director(s) is assigned to each procedure. The test director(s) is the individual designated as being responsible for coordinating test performance. Test directors for preoperational tests are assigned from the ISG or the Plant Staff by the ISG Supervisor or his designee. Test directors for startup tests are assigned by the Technical Supervisor or his designee.

Specific responsibilities of the test director include but are not limited to:

a. Verifying test prerequisites are complete and properly documented, except as provided by Subsection 14.2.4.2

b. Ensuring that required test apparatus/equipment is available and calibrated.

c. Documenting test performance on a single copy of the procedure, denoted as the official test copy

d. Ensuring that test precautions are observed during testing

e. Adhering to the detailed instructions of the approved procedure, except as provided by Subsection 14.2.4.3

f. Ensuring test personnel have been properly briefed

g. Documenting and reporting test exceptions

The plant operating staff is responsible for the safe and proper operation of equipment during testing. Should an unsafe condition arise, the plant operating staff shall take whatever action is necessary including, but not limited to, stopping the test in order to restore safe plant conditions. During startup testing, the plant operating staff is specifically responsible for compliance with operating technical specifications, and compliance with the provisions of the operating license.

14.2.4.2 Test Prerequisites

Specific test prerequisites are identified in each preoperational test procedure. The test director verifies that each prerequisite is completed and properly documented prior to signoff in the official test copy of the procedure. If a prerequisite in a preoperational test cannot be satisfied, the test director will institute a procedure modification to the Preoperational Test.

As a prerequisite to preoperational testing, proper operation of each alarm loop is verified and listed in an appendix to the test. During the preoperational test, system parameters are varied and interlocks are tested which cause alarms to actuate. Those alarms which are actuated during the course of the test will be documented in the body of the preoperational test.

14.2.4.3 Procedure Modifications

Tests are conducted in accordance with approved procedures. If necessary, these procedures may be modified to complete testing. Such procedure modifications are documented on a test change notice form. In addition to generation of a test change notice form, the test director amrks up the official test copy of the procedure and initials/dates the change.

Review and approval for test change notices on preoperational test procedures is provided by the TRB.

Test change notices for startup test procedures shall be initialed/dated by an on-shift licensed senior operator in addition to the test director. Review and approval for test change notices on startup test procedures is provided by the PORC.

Preparation, review and approval activities are accomplished before or after performance of associated testing based on the following criteria:

a) Non-Intent Changes

For procedure modifications that do not change acceptance criteria and do preserve the intent of the test, the test change notice may be approved after performance of associated testing.

b) Intent Changes

For procedure modifications that alter the acceptance criteria or the intent of the test, the test change notice is approved before performance of associated testing.

14.2.4.4 Design Problems

In the process of checkout, initial operation, and preoperational or startup testing design problems may be encountered. Such design problems are formally documented and reported to appropriate design organization representatives for resolution. Typical design problems include:

a. Errors or discrepancies in approved project design documents

b. Items that represent a potential hazard to personnel safety

c. Proposed facility modifications to meet design objectives

d. Failure of a tested system or component to satisfy design requirements or acceptance criteria

e. Operating problems where operation is in accordance with design requirements

Design response for all such reported items is mandatory. Should the response require a facility modification, the appropriate design documents are revised and issued to the field. Subsequent control of these modifications is described in Subsection 14.2.4.5.

14.2.4.5 Control of Rework, Modifications, and Repairs

A comprehensive listing of outstanding work items is maintained for each system during the Initial Test Program. This listing is maintained to ensure that identified work is performed. Typical listed work items include:

a. Incomplete or incorrect equipment installation

b. Equipment repairs (corrective maintenance)

c. Approved facility modifications

d. New or additional construction

This work is performed by the construction organization, the plant maintenance staff or a contract organization in accordance with approved procedures. In any event, in order to maintain the required controls, formal authorization is required to perform the work. During the Preoperational Test Program, this written authorization is obtained from the ISG through implementation of the appropriate ISG or Plant Staff administrative procedure. During the Startup Test Program, this written authorization is obtained from the Plant Staff through implementation of the appropriate Plant Staff administrative procedure. These administrative procedures, in addition to authorizing performance of the work, specify any retesting required as a result of the work and document completion of both the work and associated retesting. Closure of the work list item requires completion of both the specified work and the specified retesting, if required.

14.2.4.6 Test Phase Prerequisites

Completion of Phase I is a prerequisite of Phase II for each system. The completion of Phase II on safety-related systems is a prerequisite for commencement of the Startup Test Program.

Completion of each major phase of the Startup Test Program is a prerequisite to starting the succeeding phase. Subsection 14.2.11 identifies the specific testing scheduled to be conducted during each of these phases. A phase is considered complete only after the results of and exceptions to required testing are evaluated, reviewed, and approved per the requirements of Subsection 14.2.5.

14.2.5 REVIEW, EVALUATION, AND APPROVAL OF TEST RESULTS

PP&L has overall responsibility for review, evaluation, and approval of test results. The following sections establish the requirements for review, evaluation, and approval of individual test results, major test phase test results, and test plateau test results.

14.2.5.1 Individual Test Results

Upon completion of a component test, the System Engineer assembles the test results and submits them to the Group Leader for approval.

Upon completion of a preoperational or a startup test, the test director assembles a test package that includes the official test copy of the procedure and all related documentation. The preoperational test package is submitted to the Test Review Board Chairman who disseminates copies of the test package to TRB members responsible for performing an in-depth review and evaluation of test results. For startup test results the package is submitted to the chairman of PORC.

Test discrepancies, deficiencies, and omissions identified during testing or during review of test results are documented as test exceptions. Test exceptions occurring because of design problems are reported to appropriate design organization representatives for resolution per Subsection 14.2.4.4. Following TRB or PORC review and resolution of TRB or PORC comments, the chairmen have three options:

a. Recommend that the entire test be repeated.

b. Recommend that test results are unacceptable until all or part of the outstanding exceptions are resolved, in which case the test package is returned to the test director for further action.

c. Recommend acceptance of test results with or without exceptions, in which case the test package is submitted to the appropriate approval authority for final review and approval.

Final review and approval of preoperational test and startup test results is by the Superintendent of Plant. Final review and recommendation for approval of startup test results is by the Plant Operations Review Committee. Approval is by the Superintendent of Plant.

For test results approved with exceptions, each exception will be evaluated and assigned a required completion date relative to the different phases of the Initial Test Program. Test exceptions are resolved by processing them through the same review and approval cycle as associated test results.

14.2.5.2 Major Test Phase - Test Results

Commencement of each major test phase of the Startup Test Program, requires that outstanding work items be reviewed and the following commitments be satisfied:

a. Commencement of Initial Fuel Loading requires that the preoperational test results of Figure 14.2-4 be reviewed and approved.

b. Commencement of Initial Heatup and Low Power Testing requires that the Phase III startup test results be reviewed and approved.

c. Commencement of Power Ascension Testing requires that the Phase IV startup test results be reviewed and approved.

14.2.5.3 Power Ascension Testing - Test Results

Testing during the Power Ascension Test Phase is sequenced in distinct test plateaus. Prior to proceeding from one plateau to the next, the startup test results of the preceeding plateau are required to be reviewed and approved.

14.2.6 TEST RECORDS

A single copy of each approved procedure, denoted as the official test copy, is used as the official record of the test. Because of the format of startup test procedures, there will be one official test copy of a subtest for each Test Condition or plant operating condition in which the subtest is implemented. The completed official test records are assembled into a test package at the end of testing. This test package is retained in accordance with PP&L requirements for record retention.

14.2.7 CONFORMANCE OF TEST PROGRAMS WITH REGULATORY GUIDES

The safety-related performance requirements of the safety-related structures, systems, and components identified in Chapter 3 are tested in conformance with the regulatory positions established in the following regulatory guides or justification for exceptions is provided.

Number	Title
1.20	Vibration Measurements on Reactor Internals (Revision 2, May 1976).
1.41	Preoperational Testing of Redundant On-site Electric Power Systems to Verify Proper Load Group Assignments (March 16, 1973).
1.52	Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants (Revision 1, July 1976). Testing will be performed on the Control Structure Emergency Outside Air Supply System in accordance with the exceptions taken on Regulatory Guide 1.52 in Section 3.13.
1.56	Maintenance of Water Purity in Boiling Water Reactors (June 1973).
1.68	Initial Test Programs for Water-Cooled Reactors Power Plants (Revision 1, January 1977). (1) Reference: Section C.1 of the Regulatory Guide. Testing will be conducted on safety-related structures, systems, and components identified in Table 14.2-1 as required by 10CFR50. (2) Reference: Section C.9 of the Regulatory Guide. The requirements of Preoperational Test results documentation and reporting are satisfied by the format and content of the completed test procedures; generation of additional reports is not contemplated.

(3) Reference: Appendix A, Section 1.h(10) of the Regulatory Guide.

Not applicable because SSES does not use containment recirculation fan for post accident containment heat removal.

(4) Reference: Appendix A, Section 5.1.1 of the Regulatory Guide. The two pump trip is done at Test Condition 3 (approximately 100% core flow and 75% power).

(5) Reference: Appendix A, Section 5.c.c of the Regulatory Guide.

Demonstration of the operability of liquid radioactive waste system is provided in the preoperational test program. No additional testing is necessary during the power-ascension test phase.

1.68.1 Preoperational and Initial Startup Testing of Feedwater and Condensate systems for Boiling Water Reactor Power Plants (Revision 1, January 1977).

Testing may be limited by the availability of auxiliary steam.

1.68.2 Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants (January 1977).

1.70 Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (September 1975).

1.80 Preoperational Testing of Instrument Air Systems (June 1974).

The Instrument Air System is not safety related. However, the various components in the Instrument Gas System will be tested to verify that they fail as designed per the statement in Section 3.13. The movement of affected valves will be verified as part of the test associated with each respective valve's corresponding system test.

The action and flow of decay air is not an essential criteria of operation in relation to the affected valves. The valves are to fail with loss of gas to a safe position. Whether decaying pressure will hold some or all of the valves (except for those on the affected line) in normal operating positions is not of critical importance.

1.104 Overhead Crane Handling Systems for Nuclear Power Plants (February, 1976).

Exceptions for testing of the cranes are outlined in Section 3.13.

1.108 Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems at Nuclear Power Plants (August 1977).

The testing of diesel generators will conform to Regulatory Guide 1.108 per regulatory position 2.a.

Since sequence of events capability was not part of the design, testing will also take the same exceptions as outlined in Section 3.13.

1.140 Design, Testing and maintenance criteria for normal ventilation exhaust system air filtration and absorption units of light-water-cooled nuclear power plants (Revision 1).

Preoperational testing will comply with regulatory position C.5.

14.2.8 UTILIZATION OF REACTOR OPERATING AND TESTING EXPERIENCE IN THE DEVELOPMENT OF THE TEST PROGRAM

The Manager-Nuclear Support is responsible for ensuring that reactor operating and testing experiences of similar power plants are made known to the ISG and the Plant Staff during the Initial Test Program. The primary sources of experience information are NRC License Events and experiences of industry contacts. This information will be sorted and reported for a period of two years prior to fuel load on the first unit. The Manager-Nuclear Support is addressed in Subsection 17.2.1.

14.2.9 TRIAL USE OF PLANT OPERATING AND EMERGENCY PROCEDURES

The adequacy of Plant Operating and Emergency Procedures will be confirmed by trial-use during the Initial Test Program. Those procedures that do not require nuclear fuel are confirmed adequate to the extent practicable during the Preoperational

Test Program. Those procedures that require nuclear fuel are confirmed adequate to the extent practicable during the Startup Test Program.

The plant operating staff is responsible for confirmation of operating and emergency procedures. The Superintendent of Plant is responsible for ensuring that comments/changes identified during confirmation are incorporated in finalized procedures.

It is not intended that preoperational test procedures explicitly incorporate or reference plant operating and emergency procedures. These tests are intended to stand on their own since they are not necessarily compatible with configurations and conditions required for confirmation of facility operating and emergency procedures. Startup test procedures will incorporate and reference plant operating and emergency procedures to the extent practical.

14.2.10 INITIAL FUEL LOADING AND INITIAL CRITICALITY

Initial fuel loading is accomplished in accordance with startup test procedure, ST-3 Fuel Loading. Initial criticality is accomplished in accordance with startup test procedure ST-4, Full Core Shutdown Margin. These procedures comply with the general guidelines and regulatory positions contained in Regulatory Guide 1.68 (Revision 1, January 1977). Test abstracts establishing the objectives, prerequisites, test method, and acceptance criteria for these procedures are presented in Subsection 14.2.12.

14.2.11 TEST PROGRAM SCHEDULE

The Preoperational Test Program is scheduled for 15 months duration on the Unit 1 and Common components and for 12 months duration on the remaining Unit 2 components. (See Figure 14.2-4a and 14.2-4b). The subsequent Startup Test Programs are scheduled for six months on each unit.

The Preoperational Test Program sequential test schedules presented on Figures 14.2-4a and 14.2-4b offer one possible plan for an orderly and efficient progression of the program. While these sequences may be preferred, numerous alternatives exist. The schedule will be updated periodically at the jobsite to reflect construction status, manpower availability, and the required test prerequisites.

The safety-related structures, systems, and components will be preoperationally tested. The Preoperational Test Procedures are scheduled to be developed from September 1977 to January 1979.

The schedule of Unit 1 and Unit 2 Startup Tests is presented in Figure 14.2-5. This schedule establishes the required testing as a function of test condition. The test conditions are described on Figure 14.2-6. Startup testing will be divided into three Major Test Phases, and, within the Power Ascension Test Phase, into three distinct test plateaus. The testing included in each Major Test Phase and test plateau is described in Table 14.2-4. Even though this basic order of testing is required, there is still considerable flexibility in sequencing the startup testing specified to be conducted at each plateau. Detailed startup testing schedules, commensurate with the requirements of this schedule, will be developed at the job site.

14.2.12 INDIVIDUAL TEST DESCRIPTIONS

The individual preoperational tests to be conducted on safety-related structures, systems, and components are listed in Table 14.2-1. The abstracts of these preoperational tests are contained in Subsection 14.2.12.1 in numerical order. The Startup Test Program procedures are listed in Table 14.2-3. The abstracts of Startup Test procedures are contained in Subsection 14.2.12.2 in numerical order. The abstracts identify each test by title and number, describe the test objectives, specify the test prerequisites, provide a summary description of the test method, and establish the test acceptance criteria.

14.2.12.1 Preoperational Test Procedure Abstracts

(P2.1) 125 Volt DC System Preoperational Test

Test Objective - To demonstrate the ability of the 125 Volt dc system to perform the following:

A. The batteries can endure a complete discharge, based on their ampere hour rating, without exceeding the battery bank minimum voltage limit. (Performance Test)

B. The batteries can provide reliable stored energy to selected loads, indicated in Table 8.3-6, in the event of a design base accident. (Service Test)

C. The battery chargers can deliver their rated output.

D. The battery chargers can fully charge their associated batteries from design minimum charged state (i.e. after the service test) simultaneously providing power to the distribution panels for normal station loads.

E. That the alarms operate and annunciate at their specified abnormal condition.

F. The reliable 125V DC power is delivered to the ESF DC distribution panels.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required calibration and operation of instruments, protective devices, and breakers is verified. 480V AC Power, Resistor Load Bank, Battery Room Ventilation and Emergency Eyewash is available and/or in service.

Test Method - The Battery Performance Test is manually initiated by connecting the battery bank to the resistor load bank and discharging the batteries at a constant current for a specified period of time. The Battery Service Test is manually initiated by connecting the battery bank to the resistor load bank and simulating, as closely as possible, the load the batteries will supply during a design base accident. Then the battery charger is connected to the batteries and the distribution panels to verify that they can charge the batteries while simultaneously providing power to the normal plant loads. The battery charger is also connected to the resistor load bank and current is increased to its maximum rating with the charger isolated from its associated battery bank. Alarms are simulated and verified to be operated properly.

Acceptance Criteria - The batteries can satisfactorily deliver stored energy for the specified amount of time as required for the Performance and Service Test. The battery chargers can deliver rated output and can charge their associated battery bank from minimum voltage to a fully charged state in a specified amount of time while simultaneously supplying normal plant loads. The alarms operate at their engineered setpoints and annunciate in the Control Room.

(P4.1) 4.16 kV System Preoperational Test

Test Objective - To demonstrate the proper operation and load - carrying capability of breakers, switchgear, transformers, and cables. Also to demonstrate proper operation of protective devices, relaying and logic, transfer and trip devices, permissive and prohibit interlocks, and instrumentation and alarms.

Prerequisites - Construction is completed to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems including 125 volt dc systems are operable.

Test Method - The 4.16 KV system is energized. Required controls are operated or simulated signals are applied to verify proper operation of protective devices, relaying and logic, transfer and trip devices, permissive and prohibit interlocks, instrumentation and alarms, breakers, switchgear, transformers and cables.

Acceptance Criteria - The system performance parameters are in accordance with applicable design documents.

(P5.1) 480 Volt System Preoperational Test

Test Objective - To demonstrate the capability of the 480 Volt Load Centers and 480 Volt Motor Control Centers systems to provide electrical power to connected 480 Volt Load Centers and Motor Control Centers by demonstrating the proper operation of breakers, transfer and trip devices, relaying and logic, permissive and prohibit interlocks, instrumentation and alarms, motor-generator sets, and automatic transfer switches.

Prerequisites - Construction is completed to the extent necessary to perform this test and the system is turned over to the ISG. Required electrical power supply systems are available to energize the 480 Volt system. Required instruments and protective relays are calibrated and controls are operable.

Test Method - Feeder breakers are opened and closed by operating or simulating controls. Voltages on the bus being fed are measured to verify breaker operations, relaying and logic, permissive and prohibit interlocks and alarms. Signals are applied to verify alarms and instrumentation. Buses are de-energized and energized to verify automatic transfer, switch transfer, and re-transfer and motor-generator set operation.

Acceptance Criteria - The system performance parameters are in accordance with applicable design documents.

(P13.1) Fire Protection Water Systems

Test Objective - To demonstrate the proper operation of the Fire Protection Water System. The test will specifically demonstrate the following:

For Unit #1 testing:

1) Automatic and manual operation and reliability of the fire pumps OP511 and OP512.

2) Yard Loop Integrity and ability to provide water through any flow path to yard fire hydrants.

3) Hose Stations in Unit 1 and common are operational and water is available to the stations.

4) Automatic and manual operation of the Unit one and common sprinkler systems.

For Unit #2 testing:

1) Hose stations in Unit 2 are operational and water is available to the stations.

2) Automatic and manual operation of the Unit 2 sprinkler systems.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to ISG. Required instruments are calibrated and controls are operational. The river water makeup system, instrument air system, and the required electrical power supplies are available.

Test Method - The operating modes are initiated manually and, where applicable, automatically. Fire pump performance is determined for OP511 and OP512. Automatic and manual initiation of the individual sprinkler systems are conducted. Flow tests are conducted on end of line fire hydrants. Flow verification is established at the hose stations. Required controls are operated or simulated signals are applied to verify proper operation and proper alarm annunciation locally and remotely.

Acceptance Criteria - The system performance parameters are in accordance with applicable codes and design documents.

(P13.2) Carbon Dioxide Fire Protection System

Test Objective - To demonstrate the proper operation of the CO_2 fire extinguishing system. The test will specifically demonstrate the following:

1) The CO_2 storage tank and refrigeration system operate automatically to maintain the concentration of CO_2 in the tank.

2) The proper operation of the CO_2 automatic flooding systems.

3) The proper operation of the manual spurt CO_2 systems.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to ISG. Required instruments are calibrated and controls are operational. The required electrical power supplies are available.

Test Method - The operating modes are initiated manually and, where applicable, automatically. Required dampers and ducts close off the hazard area. The timers for CO_2 discharge agree with design criteria. The required controls are operated or simulated signals are applied to verify system interlocks and alarms.

Acceptance Criteria - System performance parameters are in accordance with applicable codes and design documents.

(P13.3) Fire and Smoke Detection Systems

Test Objective - To demonstrate the proper operation of the Fire and Smoke Detection System and related alarms.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to ISG. The required instruments are calibrated and controls are operational. The required electrical power supplies are available.

Test Method - The fire and smoke detector system required controls and instruments are operated or simulated signals are applied to ensure proper operation of interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with applicable codes and design documents.

(P13.4) Halon 1301 Extinguishing Systems

Test Objective - To demonstrate proper operation of the Halon Fire Protection system and related alarms.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to ISG. Required instruments are calibrated and controls are operable. Required electrical power supplies are available.

Test Method - The operating modes are initiated manually and automatically. The required controls are operated or simulated signals are applied to verify system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable codes and design documents.

(P14.1) Reactor Building Closed Cooling Water System Preoperational Test

Test Objective - To demonstrate the Reactor Building Closed Cooling Water System functions as designed.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available. The Service Water System, Instrument Air System, a makeup water source for the RBCCW System and the Emergency Service Water System are available.

Test Method - The system operation is initiated manually and the performance of the pumps is determined. Required controls are operated or simulated signals are applied to verify; automatic transfer to Emergency Service Water on RBCCW System low flow; and system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P16.1) RHR Service Water System Preoperational Test

Test Objective - To demonstrate the capability of RHR Service Water System to provide cooling water to connected components/systems and the ability of the system controls to alarm when abnormalities occur in the system and to operate in accordance with design intent.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available. The spray pond and a make-up water source to it are available. RHR Emergency Service Water is required to conduct the flow balancing test.

Test Method - System operation is initiated manually and where applicable automatically. The system is operated in the system design modes and RHR service water pump performance is determined. Required controls are operated or simulated signals are applied to verify automatic loop/valve alignments, system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with applicable design documents.

(P17.1) Instrument ac Power System Preoperational Test

Test Objectives - To demonstrate the ability of the 120V Instrument AC Power System to perform the following:

A. That full load power is delivered to the four class 1E electrically independent ESF load groups.

B. That full load power is delivered to the two non-class 1E distribution panels and that their automatic transfer switches shift load to their emergency sources upon loss of their normal sources, and back to normal power when it is restored.

C. That the alarms operate and annunciate upon loss of power.

D. That the four class 1E ESF distribution systems are electrically isolated from each other.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. The alarms operate properly, and 480V AC power and resistor load bank are available.

Test Method - The four class 1E ESF distribution panels are energized by manually closing their respective feeder breakers. A resistor load bank is connected to each distribution panel and current is increased to full load while maintaining required voltage of the three other distribution panels still energized. The remaining panel is de-energized to show that it does not affect the operation of the other three distribution panels. (This is performed for all four distribution panels.) Also, the undervoltage alarms are checked when each panel is de-energized. The two non-class 1E distribution panels are also energized by manually closing their respective feeder breakers. A resistor load bank is connected to each distribution panel and current is increased to full load. The automatic transfer switch normal supply breaker is manually opened to simulate a loss of normal power and the output voltage of the distribution panel is monitored to verify that the supply voltage switched from normal to emergency in a specified time period. The emergency supply breaker is opened and the output voltage of the distribution panel is monitored to verify that output voltage is not present. The emergency supply breaker is closed and the normal supply breaker is closed to restore normal power. Output voltage is monitored to verify that supply voltage switched from emergency to normal in the specified period of time. The non-class 1E distribution panel undervoltage alarms are verified when both normal and emergency supply breakers in the automatic transfer switches are opened.

Acceptance Criteria - That reliable 120V AC Power, at design load, is supplied to all instrument buses. That loss of normal supply to the automatic transfer switches causes a shift, in a specified time period, to the emergency supply and vice-versa when normal supply voltage is restored. That the four class 1E distribution panels are electrically isolated from each other and that loss of power alarms operate and annunciate in the Control Room.

(P23.1) Diesel Fuel Oil System Preoperational Test

Test Objective - To demonstrate that the diesel fuel oil system is capable of supplying fuel oil to connected plant equipment.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instrumentation is calibrated and controls are operable. Required electrical power supply systems are available. The diesel oil storage tank is at its normal operating level.

Test Method - System operation is initiated manually. The performance of the diesel transfer pumps is determined and the diesel day tank capacity is verified. Simulated signals are applied to verify system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P24.1) Diesel Generator System Preoperational Test

Test Objective - To demonstrate system reliability, proper voltage and frequency regulation under transient and steady-state conditions, proper logic correct setpoints for trip devices, and proper operation of initiating devices and permissive and prohibit interlocks. Starting, cooling, heating, ventilating, lubricating and fueling auxiliary systems will also be tested to demonstrate that their performance is in accordance with design.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Emergency service water, Diesel Building H&V, 125 Volt dc Power, and Instrument Air are available. The diesel oil day tank is filled and a make-up source is available.

Test Method - System operation is initiated manually and diesel generator capability to start and attain rated voltage within the specified time are verified. Diesel generators are loaded to the rated load and the performance is determined. Required controls are operated or simulated signals are applied to verify automatic start, sequential loading, D-G protection, load rejection capability and other system interlocks and alarms. Reliability is demonstrated through 69 consecutive valid start tests of station diesel generators, with a minimum of 23 valid start tests per individual diesel generator.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P25.1) Primary Containment Instrument Gas System Preoperational Test

Test Objectives - To demonstrate that the Containment Instrument Gas system functions as designed.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems, the Reactor Building Closed Cooling Water System and Instrument Air System are available.

Test Method - System operation is initiated manually to determine the performance of compressors, moisture separators, dryers and filters. Required controls are operated or simulated signals are applied to verify; instrument air system backup, isolation on primary containment isolation signal, and other system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P28.1) ESSW Pumphouse H&V System Preoperational Test

Test Objective - To demonstrate the capability of ESSW Pumphouse Heating and Ventilating System to maintain the required ambient temperature inside the ESSW Pumphouse.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems and the Instrument Air System are available.

Test Method - System operation is initiated manually and the fan air flow, damper operation, heater operation and ambient conditions inside the pumphouse are determined. Required controls are operated or simulated signals are applied to verify fan(s) automatic starts with associated pump starts and system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P28.3) Diesel Generator Building Heating and Ventilation System Preoperational Test

Test Objective - To demonstrate the capability of the system to maintain the required ambient temperatures inside the diesel generator building.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems, the Instrument Air System and Control Structure Chilled Water System are available.

Test Method - System operation is initiated manually and fan air flow, damper operation, heater operation and ambient temperatures inside the diesel generator building are determined. Required controls are operated or simulated signals are applied to verify fan automatic starts with associated D-G starts and system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with applicable design documents.

(P30.1) Control Structure H&V System Preoperational Test

Test Objective - To demonstrate the operability of the Control Structure H&V System and its interlocks inside the control structure building to demonstrate this system's ability to maintain a positive pressure above atmospheric during normal operation and high radiation signal when the emergency outside air supply mode is running. To demonstrate the ability of the Control Structure H&V to isolate before chlorine reaches the isolation dampers when chlorine is detected in the outside air intake.

Prerequisite - Construction is complete and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. The Control Structure Chilled Water

System, Instrument Air System and turbine building vent are available. Required electrical power supply systems are available.

Test Method - The system operation is initiated manually and fan performance, damper operations and heating element operation are determined. The differential pressures with respect to outside atmosphere are measured. Required controls are operated or simulated signals are applied to verify the emergency filter operation on high radiation signal, automatic recirculation on high chlorine signal, system manual isolation and other system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P30.2) Control Structure Chilled Water System Preoperational Test

Test Objective - To demonstrate the ability of the Control Structure Chilled Water System to provide chilled water flow to Control Structure Heating/Ventilating Units and Control room floor and computer room floor cooling units.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. The Service Water System, Emergency Service Water System, and Instrument Air System are available. Required electrical power supply systems are available.

Test Method - The system is operated to demonstrate chiller operation and chilled water pump performance. Required controls are operated or simulated signals are applied to verify automatic alignment of the system under emergency conditions (start of emergency condenser water recirculation pump) and other system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P34.1) Reactor Building H&V System Preoperational Test

Test Objective - To demonstrate the capability of the Reactor Building H&V System to maintain the required thermal environment inside the reactor building.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG.

Required instruments and controls are operable. The Instrument Air System is available. Required electrical power supply systems and Reactor Building Vent are available. The Reactor Building ventilation flow balancing, High Efficiency Particulate Air (HEPA) filter and charcoal absorber efficiency, and in-place leak tests are completed.

Test Method - The system is operated to measure the fan performance and determine the capability to maintain the Reactor Building at negative pressure within the required thermal environment and areas of greater potential contamination at a lower pressure than the rest of the building.

Required controls are operated or simulated signals are applied to verify the system isolation on LOCA and/or high radiation signal, and other system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P34.2) Reactor Building Chilled Water System Preoperational Test

Test Objective - To demonstrate that the Reactor Building Chilled Water System provides the required cooling water to connected coolers under normal and emergency conditions.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. The Reactor Building Closed Cooling Water System, Service Water System, Instrument Air System, Make-up Demineralizer Water System and required electrical power supply systems are available.

Test Method - The system is operated to demonstrate the chiller and chilled water pump operation. Required controls are operated or simulated signals are applied to verify system isolation, automatic valve alignment, equipment operation under emergency condition and system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P45.1) Feedwater System Preoperational Test

Test Objectives - The general objective of this test is to demonstrate proper operation of the Feedwater System. This will be accomplished to the extent possible utilizing the Auxiliary

Boilers as a steam supply. The test will specifically demonstrate:

1) All RFP and RFPT instruments have been calibrated in accordance with the vendor's instruction manuals and instrument data sheets.

2) All RFP and RFPT alarm and trip points have been set properly.

3) All recorders, indicators, annunciators, and computer inputs function correctly.

Prerequisites

1) Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG.

2) The Service Water System is operational.

3) The Main Turbine Lube-Oil System is filled and operational.

4) The Instrument Air System is operational.

5) The Computer is operational to the extent necessary to verify inputs from the feedwater system.

6) The 480 volt motor control centers necessary for this test are operational.

7) The 250 volt DC control centers necessary for this test are operational.

8) RFPT A, B, and C Lube-Oil reservoirs are filled.

Test Method - Normal and emergency responses of the lube oil and turbine trip systems are verified following simulation or process manipulation of the controlling variable.

Acceptance Criteria

1) Interlocks of the reactor feed pump turbine (RFPT) and of the alternate and emergency lube oil pumps and their corresponding alarms function as designed.

2) All abnormal conditions providing trip signals to the RFPTs function as designed.

(P45.2) Feedwater Control System Preoperational Test

Test Objectives - The general objective of this test is to demonstrate proper operation of the Feedwater Control System. This will be accomplished to the extent possible without actually pumping water with the feed pump turbines. The test will specifically demonstrate:

1) All feedwater control instruments have been calibrated over their full range in accordance with the vendor's instruction manuals and instrument data sheets.

2) All feedwater alarm and trip points have been set properly.

3) All recorders, indicators, annunciators, and computer inputs function correctly.

4) Interlocks to the main turbine, recirculation system, and feed pumps function correctly.

5) Feedwater control signals to the start-up regulating valve and turbine-driven feed pumps function correctly with simulated inputs and step commands originating from their respective control stations.

Prerequisites - The prerequisites for this test are as follows:

1) Construction of the system is complete to the extent required to conduct this test and the system is turned over to the ISG.

2) The 125 Volt DC system is operational.

3) The Instrument AC system is operational.

4) The 24 Volt DC system is operational.

5) Panel 1C651 annunciator is energized.

Test Method - Various level, flow, pressure, and speed signals will be simulated and the proper responses will be verified.

Acceptance Criteria

1) The reactor, main steam, and feedwater pressure and flow indicators, recorders, computer inputs, and trip points respond within designed tolerances.

2) Speed regulation response of each RFP Turbine is within design limits.

3) The response of the startup regulating valve is within design tolerances.

4) Changes in the control mode, selection of control channels, or integrity of incoming signal do not produce adverse changes in the controlled variables.

(P49.1) Residual Heat Removal System Preoperational Test

Test Objective - To demonstrate that the Residual Heat Removal System (RHRS) delivers cooling water as designed for each of the following system modes of operation: shutdown cooling, suppression pool spray, low pressure coolant injection (LPCI), suppression pool cooling, and fuel pool cooling.

Demonstrate operability of interlocks and isolation valves provided for overpressure protection from the reactor coolant system.

Testing will include demonstratons of proper operation of initiating devices, correct logic, proper operation of bypasses, proper operation of prohibit and permissive interlocks, and proper operation of equipment protective devices that could shut down or defeat the operation or functioning of such features.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems and the Instrument Air Systems are available. Reactor pressure vessel, suppression pool, fuel pool, and fuel pool skimmer surge tank are filled up to required level to provide enough suction head to the RHR pumps. Makeup water sources are available.

Test Method - The operating modes of the system are initiated manually and where applicable, automatically. RHR pump performance is determined for each operating mode. Control devices are operated or simulated signals are applied to verify valve alignment, LPCI mode operation for low reactor water level and high drywell pressure, and other system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with applicable engineering design documents.

(P50.1) Reactor Core Isolation Cooling System Preoperational Test

Test Objective - To demonstrate the capability of the Reactor Core Isolation Cooling (RCIC) System to deliver water to the reactor pressure vessel.

Prerequisites - Construction is complete to the extent necessary to perform these tests and the system is turned over to ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems and the Instrument Air System are available. Suppression pool and condensate storage tank are filled to provide enough suction head to RCIC pump and reactor pressure vessel is available to receive water. Auxiliary steam is available for RCIC turbine operation. Part of the RHR system will also be available to provide a suction flow path for RCIC pump.

Test Method - The system operation is initiated manually and automatically. The system is operated to determine the performance parameters for the RCIC turbine and pump and the barometric condensate pump. Control devices are operated or simulated signals are applied to verify automatic valve alignment (system isolation), turbine trip and start modes, and other system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with applicable engineering design documents.

(P51.1) Core Spray System Preoperational Test

Test Objectives - To demonstrate the ability of the Core Spray System to accept water from both the suppression pool (normal) and the condensate storage tank (backup) and deliver flow at adequate pressure to the reactor pressure vessel in an acceptable spray pattern.

Prerequisites - Construction is complete to the extent necessary to perform these tests and the system is turned over to the ISG. Power and control voltage is available for the motors, valves and instruments associated with this system. Required instruments are calibrated and controls are operable. The suppression pool and condensate storage tanks are filled to the required level. The reactor pressure vessel head is removed and the vessel can accept water. The condensate transfer system is available.

Test Method - The normal system operation is initiated automatically by simulating a Design Base Accident. The pumps are started and the appropriate valves and instruments are operated to ensure that water flow is established to the reactor pressure vessel. System logic, interlocks, and alarms are verified to be in accordance with design intent and system flows and pressures are verified to ensure that they are adequate to inject water into the reactor pressure vessel via the core spray spargers. The system is operated manually through the test line back to the suppression pool. Also, the system is manually lined up to accept water from the condensate storage tank and deliver core cooling water to the reactor pressure vessel.

Acceptance Criteria - That the core spray system can deliver cooling water at design flow and pressure to the reactor pressure vessel within a specified period of time for various simulated operating conditions.

(P51.1A) Core Spray System Pattern Preoperational Test

Test Objective - To demonstrate the ability of the Core Spray System to deliver a proper spray pattern at rated and runout conditions. This procedure shall also verify satisfactory physical response of system components within the reactor pressure vessel. The system discharge line restriction flow orifices shall be verified as being properly sized such that runout flow does not exceed system design values.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Power and control voltage is available for the motors, valves and instruments associated with this system. Required instruments are calibrated and controls are operable. The suppression pool is filled to the required level. The reactor pressure vessel head is removed and the vessel can accept water. The condensate transfer system is available.

Test Method - System operation shall be manually initiated, monitored and controlled such that vessel injection is achieved in accordance with test objectives.

Acceptance Criteria - The Core Spray System can deliver cooling water at design flow with an acceptable spray pattern to the reactor pressure vessel. During this test photographic records shall be made, no system abnormalities shall be observed, restriction flow orifices shall be properly sized, and free route from the core spray junction box vent holes shall be verified.

(P52.1) High Pressure Coolant Injection System Preoperational Test

Test Objective - To demonstrate that the High Pressure Coolant Injection System (HPCIS) delivers coolant water to the reactor.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to ISG. Required instruments are calibrated and controls are operable. The suppression pool and condensate storage tank are filled to provide the required suction head to the HPCI pump. The reactor pressure vessel head is off and the vessel is ready to receive water from the HPCI system. Required electrical power supply systems, Standby Gas Treatment, required ventilation systems and Instrument Air System are available. The Auxiliary Boiler or another source of steam supply is available to run the HPCI turbine.

Test Method - System operation is initiated manually and where applicable automatically. Reactor water low level and drywell high pressure signals are simulated to verify HPCI turbine automatic functions. System isolation is verified by operating required controls and or simulated signals. Steamline high differential pressure signals are simulated to verify automatic functions. Limited turbine and pump operation (depending upon auxiliary steam conditions) and automatic valve alignment are demonstrated. Containment isolation valves are functionally tested. Required controls are operated or simulated signals are applied to verify interlocks, trips and alarms.

Acceptance Criteria - The system performance characteristics are in accordance with applicable design documents.

(P53.1) Standby Liquid Control System Preoperational Test

Test Objective - To demonstrate the operation of the system with demineralized water. Demonstrate operability of instrumentation, controls, interlocks, and alarms. Verify operability of heaters, air spargers, and heat tracing. Conduct test firings of squib-actuated valves, and demonstrate design injection capability. Tests should be conducted as appropriate to verify redundancy and electrical independence.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. The reactor vessel is available to receive water injected from the Standby Liquid Control System. Required electrical power supply systems and a source of demineralized makeup water are available.

Test Method - System operation is initiated manually. Demineralized water is used for testing the system. The pumps are run taking suction from the standby liquid storage tank and the test tank. Squib valves are fired and the rate of demineralized water injection into the reactor vessel from each pump is measured. Required controls are operated or simulated signals are applied to verify interlocks and alarms.

Acceptance Criteria - The system performance characteristics are in accordance with the applicable design documents.

(P54.1) Emergency Service Water System Preoperational Test

Test Objective - To demonstrate that the Emergency Service Water System provides a supply of cooling water to the plant emergency equipment, to demonstrate the ability to start the ESW pumps from the remote shutdown panel, to demonstrate the ability of an ESW pump to start automatically when the associated diesel-generator unit starts, to demonstrate the proper operation of system automatic valve transfer schemes, and to demonstrate the proper operation of spray pond components.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available. The spray pond is filled to provide enough suction head for the ESW pumps, and a makeup source to the spray pond is available. The RHR service water system is in operation.

Test Method - The system is started manually and automatically through the associated diesel generator start signal. Pump flow paths are established and pump flows are measured for each loop. Flow balancing of the RHR Service Water System and Emergency Service Water System is performed. Proper operation of the line break detection system is verified. Required controls are operated and simulated signals are applied to verify interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P55.1) Control Rod Drive System Preoperational Test

Test Objective - To demonstrate the operation of the Control Rod Drive System including control rod drive hydraulic system and CRD mechanisms.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available. The condensate storage tank is filled to provide enough suction head to the CRD pump. The TBCCW System and Instrument Air System are available. The Reactor Manual Control System is operational to the point required for continuing with this test.

Test Method - System operation is initiated manually and the system flow and pressure control stations are adjusted. CRD pump performance parameters are measured. Control rod drives are exercised to verify latching, position indication and insert/withdraw speeds. Scram tests are conducted and scram times are measured for each control rod drive. Required controls are operated or simulated signals are applied to verify system interlocks and alarms. Rod buffer performance is also tested.

Acceptance Criteria - System performance parameters are in accordance with the applicable design documents.

(P56.1) Reactor Manual Control System Preoperational Test

Test Objectives - To verify the operation of the Reactor Manual Control System, including relays, control circuitry, switches, rod blocks, indicating lights and control valves and to verify operation of the Rod Worth Minimizer and Rod Sequence Control System.

Prerequisites - Construction is complete to the extent necessary to perform this test and system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available.

Test Method - System integrated operation is initiated manually. Controls are operated and simulated signals are applied to verify: rod blocks, alarms and interlocks of the reactor mode switch; proper operation of the rod position information system; and rod drift alarm circuit directional control valve time sequence for insert and withdraw commands. Proper operation of the Rod Worth Minimizer and Rod Sequence Control System are verified utilizing actual control rod manipulations.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P57.1) Uninterruptable AC Power System Preoperational Test

Test Objective - To demonstrate the ability of the Uninterruptable AC Power System to perform the following:

1) That full load power is supplied to the distribution panel

2) That the static transfer switch will automatically shift load from the preferred to the alternate source upon loss of the preferred source.

3) That the static transfer switch will automatically shift load from the preferred source to the alternate source when the preferred source becomes overloaded and shift back to the preferred source when the overload condition is cleared.

4) That loads can manually be switched from preferred to alternate source and vice-versa.

5) That alarms operate and annunciate at their specified abnormal condition.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required calibration and operation of instrument, protective devices and breakers is verified. 480V AC Power, 250 V DC Power, and Resistor Load Bank are available.

Test Method - The Uninterruptable Power Supply is energized by manually closing the 250 V DC preferred breaker (inverter) and the 480 V AC Alternate Breaker (Voltage Regulating Transformer). With the static transfer switch in normal mode, the load is increased by use of the Resistor Load Bank while the voltage and current is monitored. The current is gradually increased above normal rating until the automatic transfer switch shifts the overload to the alternate source. Then the load is slowly decreased to clear the overload and to verify that the automatic transfer switch shifts the load back to the preferred source. A loss of the preferred source is simulated to verify that the automatic transfer switch will shift the load to the alternate source. Then with both sources available the transfer switch is manually switched from the preferred to alternate source and vice versa by means of the bypass mode and normal mode pushbuttons. Alarms are either simulated or functionally checked throughout the above procedure.

Acceptance Criteria - That reliable 120 V AC Power, at design load is supplied to the distribution panel. That the automatic transfer switch will shift loads from the preferred to the alternate source with negligable power interruption upon loss of preferred source. That the automatic transfer switch will shift load from the preferred to the alternate source in an overloaded condition and back to the preferred source when the overload condition is cleared and, that the load can manually be shifted from the preferred to the alternate source and vice-versa that alarms operate at their engineered set points and annunciate in the control room.

(P58.1) Reactor Protection System Preoperational Test

Test Objective - To demonstrate the proper operation of the Reactor Protection System (RPS) in all combinations of logic and to demonstrate redundancy, electrical independence, mode switch operation, and safe failure on loss of power.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available. The Control Rod Drive System preoperational test is completed to the extent necessary to perform this test.

Test Method - Integrated system operation is initiated manually to verify M-G set performance and electrical independence. Required controls are operated or simulated signals are applied to verify: sensor relay-to-scram trip actuator response time, the ability to scram CRDs in conjunction with the CRD hydraulic system, scram reset delay time, mode switch operation, and system interlocks and alarms.

Acceptance Criteria - System performance is in accordance with the applicable design documents.

(P59.1) Primary Containment System Preoperational Test

Test Objective - To demonstrate the operability and isolation capability of the Primary Containment System. Containment isolation valve functional tests will be performed.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. The suppression pool is filled with demineralized water to the required level and the hotwell is available. The Containment

Instrument Gas System, Instrument Air System and required electrical power supply systems are available. All primary containment isolation valves are operable.

Test Method - The suppression pool cleanup system will be tested for proper operation; the primary containment isolation system will have signals simulated with the valves in the non-isolation position, to verify the primary containment isolates when an isolation signal is received. Valve closure times are verified for those valves specified in the FSAR in the various system preoperational tests. The test method is described in the General Test Statement.

Acceptance Criteria - The system performance is in accordance with the applicable design documents.

TP 2.14 Nuclear Boiler System Level Instrumentation Verification Test

Test Objective - To demonstrate that the nuclear boiler level instruments function as desired.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required Electrical Power Supply Systems are available. A method to raise and lower the reactor vessel water level is available.

Test Method - The actual reactor vessel water level will be changed to verify level switch trip points, indicating functions and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P59.2) Containment Integrated Leak Rate Test

Test Objective - To demonstrate that the total leakage from the containment does not exceed the maximum allowable leakage rate (La) at the calculated peak containment internal pressure (Pa), as defined in 10 CFR50, Appendix J.

Prerequisities - Construction of the primary containment, including installation of all portions of mechanical, fluid, electrical, and instrumentation systems penetrating containment is complete. Type B and Type C local leakage rate is satisfactorily complete. Required test equipment instruments and data acquisition systems are operable. Systems required to support the ILRT are operational.

Test Method - The test shall be conducted in accordance with the requirements of Subsection 6.2.6 of the FSAR.

Acceptance Criteria - Acceptance criteria for this test are in accordance with the requirements of Chapter 16 of the FSAR.

(P60.1) Containment Atmosphere Circulation System Preoperational Test

Test Objective - To demonstrate the capability of the Containment Atmosphere Circulation System to cool and circulate air inside the Containment.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available. The Reactor Building Chilled Water System or an alternate cooling water supply is available.

Test Method - The system operation is initiated manually, and flow for each fan is determined. Required controls are operated or simulated signals are applied to verify; automatic start of standby units and other system interlocks and alarms. No heat loads are simulated during the test.

Acceptance Criteria - The system performance is in accordance with the applicable design documents.

(P61.1) Reactor Water Cleanup System Preoperational Test

Test Objectives - To demonstrate the operability of the Reactor Water Cleanup and Filter Demineralizer System. In particular the following items are to be demonstrated:

1) The ability of individual components, instrumentations, alarms and interlocks to function properly.

2) Verify proper system performance by verifying all flow paths, flow rates and component performances to be in accordance with design specifications.

3) The ability of the system and filter to isolate by simulating each sensor to its trip point.

4) Verify the RWCU system containment isolation valves will respond properly to all control signals and closing times are within required specifications.

5) The ability of the filter/demineralizer valve and pump operating sequence to operate properly.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. The Reactor vessel is filled to provide enough suction head to the Reactor Water Cleanup Recirculation Pumps. The Reactor Building Closed Cooling Water System, Instrument Air System, condenser hotwell or Liquid Radwaste Collection System, and the RWCU Precoat System are available. Required electrical power supply systems are available.

Test Method - System operation is initiated manually. Pump flow and filter and demineralizer differential pressures are determined. Precoat and backwash cycles are tried. Controls are operated or simulated signals are applied to verify system isolation upon initiation of the respective NSSS isolation relay, other system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P64.1) Reactor Recirculation System Preoperational Test

Test Objectives - To demonstrate the operability of the Reactor Recirculation components and the system.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available. The Reactor Building Closed Cooling Water System, is available. The reactor vessel is filled with demineralized water to the required level.

Test Method - System operation is initiated manually. The system is tested by individual and integrated operation of M-G sets, pumps, and valves. Performance of the M-G sets, recirculation pumps, and jet pumps are determined to the extent possible during this test. Required controls are operated or simulated signals are applied to verify interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

TP 2.16 Reactor Internals Vibration and Inspection

Test Objective - The test objective is to detect damage, excessive wear, loose parts, or other evidence of unacceptable vibration which could result from assembly errors or undesirable deviations from the previously qualified prototype plant construction.

This test is a quality assurance measure which experimentally confirms the absence of excessive vibration of core support structures, jet pumps, lower plenum components, and other major internal structures. The test is conducted without fuel and is not intended to be a test of fuel or incore instrument vibration. However, the specified test conditions, without fuel present, provide a level of vibration excitation of major internal structures which is at least as high as that measured in normal power operation.

Prerequisites - To the extent necessary to perform this test all reactor internals components are installed except as follows.

1. The core matrix is empty; there are no fuel assemblies, incore instrumentation tubes, or neutron source rods. Control blades are withdrawn or not installed. Fuel support castings are installed.

2. The dryer assembly need not be installed.

3. One or both of the access hole covers on the shroud support plate must remain unwelded until after the test to provide access for inspection. Temporary closures must be provided.

The reactor vessel is closed, filled, and ready for pressurization. The recirculation pumps are operable. The RHR system pumps are operable to provide necessary temperature rise. Clean-up system heat exchangers are operable for temperature control.

Test Method - A visual inspection is made before and after the required maximum allowable speed pump runs. These flow runs include 35 hours of two-loop operation and 14 hours each for loops A and B. These hours may not be sequential, but they must be between the initial and final inspections.

Acceptance Criteria - Initial and final inspection results are acceptable.

(P69.1) Liquid Radwaste Collection System Preoperational Test

Test Objective - To demonstrate the capability of the Liquid Radwaste Collection System to collect liquid waste.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable.

Required electrical power supply systems are available. Liquid Radwaste Collection System and storage tanks are available.

Test Method - Sump pumps are operated and performance characteristics are determined. Level controls are operated to verify pump starts and alarms. Liquid radwaste discharge valves from primary containment are verified to close upon containment isolation signal .

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P70.1) Standby Gas Treatment System And Secondary Containment Isolation Preoperational Test

Test Objective - To demonstrate the capability of the Standby Gas Treatment System (SGTS) to function as designed.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available. The Reactor Building Heating and Ventilation System, SGTS vent, and Instrument Air System are available.

Test Method - System operation is initiated manually and where applicable automatically. Required controls are operated or simulated signals are applied to verify secondary containment isolation and start of SGTS. SGTS performance is determined by measuring secondary containment pressures, system pressures and fan flow rates. System interlocks and alarms are verified.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P73.1) Containment Atmospheric Control System Preoperational Test

Test Objective - To demonstrate the capability of the Containment Atmospheric Control System to analyze containment oxygen and hydrogen content and to demonstrate the operability of the hydrogen recombiner (actual process is not demonstrated at this time).

To demonstrate the operability of the purge supply and exhaust systems, and to show the valves work according to the designed permissives and interlocks. To test the vacuum breakers and show proper operation of the controls and actuators, which will demonstrate the ability to limit the drywell and suppression pool internal and differential pressures.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instrumentation are calibrated and controls are operable. Required electrical power supply system are available.

Test Method - The oxygen and hydrogen analyzers are utilized to determine the containment atmospheric analysis. The pump and blower performances are determined. The Hydrogen Recombiner System will be operated to the extent practicable. Simulated signals are applied to verify interlocks and alarms. The system valves will be operated and a purge flow will be established to demonstrate proper operation. Vacuum breakers will be actuated to show proper directional movement when permissives are available to control circuitry.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P75.1) 24 Volt DC System Preoperational Test

Test Objective - To demonstrate the ability of the ± 24 Volt DC System to perform the following:

1) That the batteries can ensure a complete discharge, based on their ampere-hour rating, without exceeding the battery bank minimum voltage limit. (Performance Test)

2) That the batteries can provide reliable stored energy to their design loads as indicated in Table 8.3-8 in the event of a Design Base Accident.

3) That the battery chargers can deliver their rated output.

4) That the battery chargers can fully charge their associated batteries from design minimum discharge (i.e. after the service test) while simultaneously providing power to the distributed panel for normal station loads.

5) That alarms operate and annunciate at their specified abnormal condition.

6) That reliable ± 24 Volt DC is delivered to the distribution panels.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required calibration and operation of instrument protective devices and breakers is verified. 120 V AC, Resistor Load Bank, Battery Room Ventilation and Emergency Eyewash is available and/or in service.

Test Method - The battery performance test is manually initiated by connecting the battery bank to the Resistor Load Bank and discharging the batteries at a constant current for a specified period of time.

The Battery Service Test is manually initiated by connecting the battery bank to the Resistor Load Bank and simulating, as closely as possible, the load the batteries will supply during a Design Base Accident.

Then the battery charger is connected to the batteries and the distribution panels to verify that they can equalize charge the batteries while simultaniously providing power to the normal plant loads. The battery charger is also connected to the Resistor Load Bank and current is increased to its maximum rating with the charger isolated from its associated battery bank.

Alarms are simulated and verified to operate properly.

Acceptance Criteria - The batteries can satisfactorily deliver stored energy for the specified amount of time as required for the performance and service tests. The battery chargers can deliver rated output, and can charge their associated battery bank from minimum voltage to a fully charged state in a specified amount of time while simultaneously supplying normal plant loads. The alarms operate at their engineered setpoints and annunciate in the control room.

(P76.1) Plant Leak Detection System Preoperational Test

Test Objective - To demonstrate the operability, of the Plant Leak Detection System.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available.

Test Method - Sump levels will be varied (if practicable) or simulated signals are applied to level sensors to verify the leak detection system alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P78.1) Source Range Monitoring System Preoperational Test

Test Objective - To demonstrate the operability of the Source Range Monitoring (SRM) System.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required reactor internals are installed, instruments are calibrated and controls are operable. Required electrical power supply systems are available.

Test Method - Source Range Monitor Detector insert/retract drive mechanisms are operated to verify proper operation. Required simulated signals are applied to verify SRM channel trips, indicating lights and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P78.2) Intermediate Range Monitoring System Preoperational Test

Test Objective - To demonstrate the operability of the Intermediate Range Monitoring (IRM) System.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required reactor internals are installed, instruments are calibrated and controls are operable. Required electrical power supply systems are available.

Test Method - Intermediate Range Monitors detector insert/retract drive mechanisms are operated. Required simulated signals are applied to verify IRM channel trips, rod blocks, indicating lights and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P78.3) Average Power Range Neutron Monitoring System Preoperational Test

Test Objective - To demonstrate the operability of the Average Power Range Neutron Monitoring (APRM System) including LPRM's, Recirc. flow bias signals and Rod Block Monitor.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required reactor internals are installed. Instruments are calibrated and controls are operable. Required electrical power supply systems are available.

Test Method - Each LPRM is checked from detector to its end function. Required input signals are simulated to verify LPRM channel trip lamps, remote meters and alarms. Required signals from the LPRM System are simulated to each APRM channel to verify trip functions, indicating meters, lights and alarms. Each flow transmitter is checked from flow element to its end function. Signals are simulated to verify flow inducted trips, remote meters and alarms. Required signals from the LPRM and flow bias systems are simulated to each RBM channel to verify trip functions, indicating lights, and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P78.4) Traversing Incore Probe System Preoperational Test

Test Objective - To demonstrate the proper operation of the Traversing In-Core Probe System. Specific objectives are to demonstrate the following:

1) Manual and automatic Operation.

2) Proper operation of all interlocks, overrides and automatic functions.

3) Proper operation of all indications and alarms.

4) Simulated operation of the shear valves.

5) Proper interface between the TIP system and process computer.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. LPRMs are installed inside the reactor vessel and required instruments are calibrated and controls are operable. TIP tracing X-Y recorder and purge system are available.

Test Method - System operation is initiated manually. The indexer interlock, shear valve control and monitoring, ball valve control and monitoring, squib circuits and purging operations are verified. Required controls are operated or simulated signals are applied to verify interlocks external to the system and system alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P79.1) Area Radiation Monitoring System Preoperational Test

Test Objective - To demonstrate the operability of the Area Radiation Monitoring System.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and required electrical power supply systems are available. The required radioactivity sources with known strengths are available.

Test Method - The radioactive sources are used or simulated signals are applied to verify area radiation monitor channel trips, indicating lights, and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

20 (P79.2) Process Radiation Monitoring System Preoperational Test

Test Objective - To demonstarte the operability of the Process Radiation Monitoring System.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and required electrical power supply systems are available. The required radioactivity sources with known strengths are available.

Test Method - The radioactive sources are used or simulated signals are applied to verify process radiation monitor channel trips, locating lights, interlocks, and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P80.1) Reactor Non-nuclear Instrumentation System Preoperational Test

Test Objective - To demonstrate that the Reactor Non-nuclear Instrumentation System functions as designed.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and the controls are operable. All relays that are initiated from reactor vessel level and pressure sensors are placed in the untripped condition.

Test Method - Simulated signals are applied to instrument loops and trip functions, indicating functions and alarms are verified.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P81.1) - Fuel Handling System-Preoperational Test

Test Objective - To demonstrate that the refueling platform, refueling grapple and the reactor servicing tools function as designed.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available. The fuel pool or reactor cavity are available to test the fuel grapple. The Reactor Manual Control System is available to test the refueling platform interlocks.

Test Method - The refueling platform travel speed and interlocks with the Reactor Manual Control System are verified. All servicing tools are tried for proper operation. Load tests for the fuel grapple are performed and the fuel grapple is operated at designated speeds. System alarms are verified by operating the controls or simulating the required signals.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P83.1) - Main Steam System Preoperational Test

Test Objectives - To demonstrate the proper operation of Main Steam Isolation Valves, the ability of the Safety Relief valves to operate correctly in safety and automatic depressurization modes, the ability of the Main steam Isolation Valve Leakage Control System to collect steam from main steam lines by demonstrating the operational air blowers, heaters, and motor operated valves when the system is initiated, the ability of the main steam drip leg drains to function properly.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems, Instrument Air System and the Containment Instrument Gas System are available.

Test Method - The Main Steam Isolation Valves are exercised and functionally checked for closure on required logic circuit trips, loss of control power and loss of normal supply of air (or gas) by using charged accumulator. The Automatic Depressurization System is functionally checked for proper operation in automatic and manual modes. The Remote Shutdown Panel operation is demonstrated. Each Safety/Relief valve is verified operational when any one of its control solenoids is energyzed. Valves are also checked for the following: tail closed on loss of air/loss of power, and full stroke operation. The Main Steam Isolation Valve Leakage Control System is initiated manually and checked for proper operation. The Main Steam Line Drip Legs and Main Steam Line Branch Valves are functionally checked for proper operation. Alarms are checked out during the above operations.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P88.1) 250 Volt DC System Preoperational Test

Test Objective - To demonstrate the capability of the 250 volt dc system to provide dc power to connected buses.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems and a load resistor bank are available. The Battery Room Ventilation system is also available.

Test Method - The system is operated and a load capacity test is conducted for the battery with the battery charger disconnected. Required controls are operated or simulated signals are applied to verify battery charger performance, system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

(P99.1) Reactor Building Crane Preoperational Test

Test Objective - The general objectives of this test are to demonstrate the following:

1) The performance of the reactor building crane's components.

2) Establishment of baseline data for all functional components.

3) That all warning signals are working per design intent.

4) The capability of the crane to operate in a designated area in accordance with design requirements.

Prerequisites - Construction is complete and the system is turned over to the ISG. Required electrical power supply systems are available and controls are operable. Required loads are available to perform load testing of this crane. Construction phase static load testing (125% of rated load) is completed.

Test Method - The lighting system for the crane is energized and observed for proper operation. The bridge and the trolley are speed-tested in both directions. Current and voltage readings are taken in both directions. The proximity switches are tested for both the bridge and the trolley including trolley movement restriction switches in zones A, B, and C.

The main hoist and the auxiliary hoist are speed-tested traveling up and traveling down. Current and voltage readings are taken in both directions. All limit switches are tested. A loss of power situation is created for both hoists to check the brakes ability to hold without power. An overspeed test is simulated for each hoist. The main hoist load limit switch is also tested.

The above listed tests are run from the pendant pushbutton control system. Operability of the crane is also demonstrated from the cab and by radio control. The anticollision system is tested and the crane power source is verified.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

TP2.23 REACTOR BUILDING CRANE TESTING

OBJECTIVE:

To supplement load testing of the reactor building overhead crane.

PREREQUISITES:

Construction is complete to the extent required to perform the test, and the crane is available for service.

TEST METHOD:

1. Braking capability of the main and auxiliary hoist under rated load is verified (all brakes operational).

2. The ability of each individual main and auxiliary hoist brake to stop and hold rated load while lowering at rated speed is tested.

3. The capability of limiting movement of the main hook to 1/32" and the auxiliary hook to 1/16" in both raise and lower direction at rated load is tested from a complete standstill over an average of ten successive movements.

4. Voltage and current of all crane motors is recorded while running at rated load and rated speed.

5. The capability of the main hoist to limit an uncontrolled drop at rated load and rated speed to less than 1/2" hook movement is verified.

6. Simultaneous bridge and trolley movement at rated load and the ability of the zone proximity switches to restrict crane movement within safe limits is also verified.

ACCEPTANCE CRITERIA:

All crane parameters are within design limits.

(P100.1) Cold Functional Test

Test Objective - To demonstrate that the plant systems are capable of operating on an integrated basis in normal and emergency modes, to demonstrate that adequate power supplies for the class IE equipment will exist, and to assure that optimum tap settings have been selected for transformers supplying power from offsite sources to class IE busses.

Prerequisites - Required system preoperational tests have been completed and plant systems are ready for operation on an integrated basis.

Test Method - Emergency Core Cooling Systems (RHR & Core Spray) are lined up in their normal standby mode. The plant electrical system is lined up per normal electrical system lineup (For Unit 1 this lineup may be different than the lineup for two unit operation). Loss of coolant accident signals are initiated with and without a loss of offsite power. Voltages and loads are adjusted, as practical, to simulate the anticipated ranges of variations. Proper response of the electrical distribution system, diesel generators, and ECCS pumps will be verified.

Acceptance Criteria - Systems performance parameters are in accordance with the applicable design documents.

14.2.12.2 Startup Test Program Procedure Abstracts

All those tests comprising the Startup Test Program (Table 14.2-3) are discussed in this section. For each test a description is provided for test purpose, test prerequisites, test description and statement of test acceptance criteria, where applicable. Additions, deletions, and changes to these discussions are expected to occur as the test program progresses. Such modification to these discussions will be reflected in amendments to the FSAR.

In describing the purpose of a test, an attempt is made to identify those operating and safety-oriented characteristics of the plant which are being explored.

Where applicable, a definition of the relevant acceptance criteria for the test is given and is designated either Level 1 or Level 2. A Level 1 criterion normally relates to the value of a process variable assigned in the design of the plant, component systems or associated equipment. If a Level 1 criterion is not satisfied, the plant will be placed in a suitable hold-condition until resolution is obtained. Tests compatible with this hold-condition may be continued. Following resolution, applicable tests must be repeated to verify that the requirements of the Level 1 criterion are now satisfied.

A Level 2 criterion is associated with expectations relating to the performance of systems. If a Level 2 criterion is not satisfied, operating and testing plans would not necessarily be altered. Investigations of the measurements and of the analytical techniques used for the predictions would be started.

For transients involving oscillatory response, the criteria are specified in terms of decay ratio (defined as the ratio of successive maximum amplitudes of the same polarity). The decay ratio must be less than unity to meet a Level 1 criterion and less than 0.25 to meet Level 2.

(ST-1) Chemical and Radiochemical

Test Objectives - The principal objectives of this test are a) to secure information on the chemistry and radiochemistry of the reactor coolant, and b) to determine that the sampling equipment, procedures and analytic techniques are adequate to supply the data required to demonstrate that the chemistry of all parts of the entire reactor system meet specifications and process requirements.

Specific objectives of the test program include evaluation of fuel performance, evaluations of demineralizer operations by direct and indirect methods, measurements of filter performance, confirmation of condenser integrity, demonstration of proper steam separator-dryer operation, and calibration of certain process instrumentation. Data for these purposes is secured from a variety of sources: plant operating records, regular routine coolant analysis, radiochemical measurements of specific nuclides, and special chemical tests.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - Prior to fuel loading, a complete set of chemical and radiochemical samples is taken to ensure that all sample stations are functioning properly and to determine initial concentrations. Subsequent to fuel loading, during reactor heatup, and at each major power level change, samples are taken and measurements made to determine the chemical and radiochemical quality of reactor water and reactor feedwater, and performance of filters and demineralizers. Calibrations will be made of monitors in the vent, liquid waste system and liquid process lines.

Acceptance Criteria - Level 1 - Chemical factors defined in the Technical Specifications and Fuel Warranty must be maintained within the limits specified. The activity of gaseous and liquid effluents must conform to license limitations. Water quality must be known at all times and should remain within the guidelines of the Water Quality Specifications.

Level 2 - Not applicable.

(ST-2) Radiation Measurements

Test Objectives The objectives of this test are (a) to determine the background radiation levels in the plant environs prior to operation for base data on activity buildup and (b) to monitor radiation at selected power levels to assure the protection of personnel during plant operation.

Prerequisites - The required preoperational tests have been completed; the Superintendent of Plant has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

Test Method - A survey of natural background radiation at selected locations throughout the plant will be made prior to fuel loading. Subsequent to fuel loading, during reactor heatup and at power levels of approximately 25%, 60% and 100% of rated power, gamma radiation level measurements and, where appropriate, thermal and fast neutron measurements will be made at selected locations throughout the plant.

Acceptance Criteria - Level 1 - The radiation doses of plant origin and the occupancy times of personnel in radiation zones shall be controlled consistent with the guidelines of the standards for protection against radiation outlined in 10CFR20.

Level 2 - The radiation doses of plant origin shall meet the following limits depending upon which Radiations Zone the radiation base survey point is located:

Radiation Zone	Limit
I	≤ 0.5 mRem/hr.
II	≤ 2.5 mRem/hr.
III	≤ 15 mRem/hr.
IV	≤ 100 mRem/hr.

Note: All areas designated Radiation Zone V have potential radiation doses of ≥ 100 mRem/hr. Readings taken in Zone V during the Startup Test Program may be less than 100 mRem/hr; however, since Zone V is defined in terms of potential levels, there are no Acceptance Criteria for Zone V base survey points.

(ST-3) Fuel Loading

Test Objective - The objective of this test is to achieve the full and proper core complement of nuclear fuel assemblies through a safe and efficient fuel loading evolution.

Prerequisites - The required Preoperational Tests have been completed. In addition, prior to starting this test procedure, the following prerequisites will be met:

a. Fuel and Control Rod inspections will be complete.

b. Control Rods will be installed and tested.

c. Reactor vessel water level will be established and minimum level prescribed.

d. The standby liquid control system will be operable and in readiness.

e. Fuel handling equipment will have been checked and dry runs completed.

f. The status of protection systems, interlocks, mode switches, alarms, and radiation protection equipment will be prescribed and verified.

g. Water quality must meet required specifications.

The following prerequisites will be met prior to commencing actual fuel loading to assure that this operation is performed in a safe manner:

a. The status of all systems required for fuel loading will be specified and will be in the status required.

b. At least three movable neutron detectors will be calibrated and operable. At least three neutron detectors will be connected to the high flux scram trips. They will be located so as to provide acceptable signals during fuel loading.

c. Source range monitoring Nuclear instruments will be checked with a neutron source prior to fuel loading or resumption of fuel loading if sufficient delays are incurred.

d. The status of secondary containment will be specified and established.

e. Reactor vessel status will be specified relative to internal component placement and this placement established to make the vessel ready to receive fuel.

f. The high flux trip points will be set for a relatively low power level.

g. Neutron sources will be installed near the center of the core and at other specified locations.

Test Method - Before the first fuel assembly is taken from the fuel pool and inserted into the reactor, core components (fuel support castings, blade guides, control rod drives, etc.) will be installed, tested and/or verified. This procedure begins with the steps required to assemble and load neutron sources, includes the activities necessary to monitor neutron population using specially constructed fuel loading chambers (FLCs), and culminates with the insertion of fuel assemblies into the reactor core. Fuel loading continues until the core is fully loaded, verified and ready to perform subsequent Startup Tests.

Control rod functional tests, subcriticality checks, and shutdown margin demonstrations will be performed periodically during the loading.

Acceptance Criteria - Level 1 - The partially loaded core must be subcritical by at least 0.38% $\Delta k/k$ with the analytically determined, highest worth rod fully withdrawn.

(ST-4) Full Core Shutdown Margin

Test Objective - The purpose of this test is to demonstrate that the reactor will be subcritical throughout the first fuel cycle with any single control rod fully withdrawn.

Prerequisites - The following prerequisites will be complete prior to performing the full core shutdown margin test:

a) The predicted critical rod position is available

b) The Standby Liquid Control System is available

c) Nuclear instrumentation is available with neutron count rate of at least three counts per second and signal to noise ratio greater than two to one

d) High-flux scram trips are set conservatively low

e) Instrumentation has been checked or calibrated as appropriate

Test Method - This test will be performed in the fully loaded core in the xenon-free condition. The shutdown margin test will be performed by withdrawing the control rods from the all-rods-in configuration until criticality is reached. If the highest worth rod will not be withdrawn in sequence, other rods may be withdrawn providing that the reactivity worth is equivalent. The difference between the measured K_{eff} and the calculated K_{eff} for the in-sequence critical will be applied to the calculated value to obtain the true shutdown margin.

Acceptance Criteria - Level 1 - The shutdown margin of the fully loaded, cold (68°F or 20°C), xenon-free core occuring at the most reactive time during the cycle must be at least 0.38% $\Delta k/k$ with the analytically strongest rod (or its reactivity equivalent) withdrawn. If the shutdown margin is measured at some time during the cycle other than the most reactive time, compliance with the above criterion is shown by demonstrating that the shutdown margin is 0.38% $\Delta k/k$ plus an exposure dependent correction factor which corrects the shutdown margin at that time to the minimum shutdown margin.

Level 2 - Criticality should occur within $\pm$1.0% $\Delta k/k$ of the predicted critical.

(ST-5) Control Rod Drive System

Test Objective - The objectives of the Control Rod Drive System test are; a) to demonstrate that the Control Rod Drive (CRD)

System operates properly over the full range of primary coolant temperatures and pressures from ambient to operating, and b) to determine the initial operating characteristics of the entire CRD System.

Prerequisites - The required preoperational tests have been completed.

Test Method - The CRD tests performed during the startup test program are designed as an extension of the tests performed during the preoperational CRD system tests. Thus, after it is verified that all control rod drives operate properly when installed, they are tested periodically during heatup to assure that there is no significant binding caused by thermal expansion of the core components. A list of all control rod drive tests to be performed during startup testing is given in Table 14.2-5.

Acceptance Criteria - Level 1 - Each CRD must have a normal withdraw time greater than or equal to 40 seconds.

The mean scram time of all operable CRDs must not exceed the values specified in the plant technical specifications. (Scram time is measured from the time the pilot scram valve solenoids are deenergized.)

The mean scram time of the three fastest CRDs in a two by two array must not exceed the values specified in the plant technical specifications. (Scram time is measured from the time the pilot scram solenoids are deenergized)

Level 2 - Each CRD must have a normal insert or withdraw speed of 3.0 ± 0.6 inches per second, indicated by a full 12-foot stroke in 40 to 60 seconds. With respect to the control rod drive friction tests, if the differential pressure variation exceeds 15 psid for a continuous drive in, a settling test must be performed, in which case, the differential settling pressure should not be less than 30 psid nor should it vary by more than 10 psid over a full stroke.

(ST-6) SRM Performance and Control Rod Sequence

Test Objectives - The objective of this test is to demonstrate that the operational sources, SRM instrumentation, and rod withdrawal sequences provide adequate information to achieve criticality and increase power in a safe and efficient manner for each of the specified rod withdrawal sequences.

Prerequisites - The required preoperational tests have been completed.

Test Method - The operational neutron sources will be installed and source range monitor count-rate data will be taken during rod withdrawals to critical and compared with stated criteria on signal and signal count-to-noise count ratio.

A withdrawal sequence has been calculated which completely specifies control rod withdrawals from the all-rods-in condition to the rated power configuration. Each sequence will be used to attain cold criticality.

Movement of rods in a prescribed sequence is monitored by the Rod Worth Minimizer and rod sequence control system, which will prevent out of sequence withdrawal.

Acceptance Criteria - Level 1 - There must be a neutron signal count-to-noise count ratio of at least 2 to 1 on the required operable SRMs or Fuel Loading Chambers. There must be a minimum count rate of 3 counts/second on the required operable SRMs or Fuel Loading Chambers.

The IRMs must be on scale before the SRMs exceed the rod block set point.

(ST-7) Reactor Water Cleanup System

Test Objectives - The objective of this test is to demonstrate specific aspects of the mechanical operability of the Reactor Water Cleanup System. (This test, performed at rated reactor pressure and temperature, is actually the completion of the preoperational testing that could not be done without nuclear heating).

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - With the reactor at rated temperature and pressure, process variables will be recorded during steady state operation in three modes as defined by the System Process Diagram: Blowdown, Hot Standby, and Normal. Additional system configurations will also be aligned to verify proper performance of the bottom head flow and temperature indicators.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - The temperature at the tube side outlet of the non-regenerative heat exchangers shall not exceed 130°F in the blowdown mode and 120°F in the normal mode.

The pump available NPSH will be 13 feet or greater during the hot standby mode defined in the process diagrams.

The cooling water supplied to the non-regenerative heat exchangers shall be within the flow and outlet temperature limits indicated in the process diagrams.

During two pump operations at rated core flow, the bottom head temperature as measured by the bottom drain line thermocouple should be within 30°F of the recirculation loop temperatures.

Bottom head flow indicator FI-1R610 shall indicate within 25 gpm of RWCU flow indicator FI-R609 when total system flow is thru the botton head drain.

(ST-8) Residual Heat Removal System

Test Objectives - The objectives of this test are to demonstrate the ability of the Residual Heat Removal (RHR) System to: 1) remove heat from the reactor system so that the refueling and nuclear system servicing can be performed and 2) condense steam while the reactor is isolated from the main condenser.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - The suppression pool cooling mode will be used to measure the RHR heat exchanger capacity. Data will be obtained to determine the heat transfer rate with rated flow on both sides of the heat exchanger with a large temperature differential between the service and suppression pool water. Ideally this temperature differential can be established during extended RCIC or relief valve operations. This heat transfer data can then be used as a benchmark by the appropriate design personnel to check the heat exchanger capacity during other modes of operation. (An ideal demonstration of the RHR heat exchanger capacity would consist of measuring the heat transfer rate in the shutdown cooling mode with the reactor at 50 psig or less. Unfortunately, the decay heat load is insignificant during the startup test period. Use of this mode with low core exposure results in exceeding the 100°F/hr cooldown rate of the vessel. The shutdown cooling mode will be demonstrated after a trip or a cooldown from Test Condition 6.

The RHR system steam condensing mode is used to condense steam while the reactor is isolated from the main condenser and reactor vessel water level is being maintained by RCIC. This test will demonstrate system operability and stability.

Acceptance Criteria - Level 1 - The transient response of any system-related variable to any test input must not diverge.

Level 2 - The RHR system shall be capable of operating in the steam condensing, suppression pool cooling and shutdown cooling modes (with both one and two heat exchangers) at the flow rates and temperature differentials indicated on the process diagrams. System-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

The time to place the RHR heat exchangers in the steam condensing mode with the RCIC using the heat exchanger condensate flow for suction shall be one half hour or less.

(ST-9) Water Level Measurement

Test Objectives - The objective of this test is to verify consistent response of the narrow range and wide range level instrumentation.

Prerequisites - The required preoperational tests have been completed. All system instrumentation is installed and calibrated.

Test Method - In the test, the response of the Narrow Range and Wide Range Level Channels plus the Upset Range Channel are recorded at steady state conditions. In addition, the Shutdown and Fuel Zone Level channels values, Reactor Recirculation temperatures and drywell temperatures are recorded to provide supporting information.

The data is analyzed by calculating the Average Narrow Range Level and Wide Range Level value which includes the Upset Range and then determining the differences from the average value for each of the level channels. These differences are compared to the acceptance criteria.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - The indicator readings on the Narrow Range Level System should agree within ±1.5 inches of the average reading.

The Wide and Upset Range Level System indicators should agree within ±6 inches of the average reading.

(ST-10) IRM Performance

Test Objectives - The objective of this test is to adjust the Intermediate Range Monitor System to obtain an optimum overlap with the SRM and APRM systems.

Prerequisites - The required preoperational tests have been completed.

Test Method - Initially the IRM system is set during the Preoperational Test Program. SRM-IRM and IRM-APRM overlap is verified the first time sufficient neutron flux conditions arise. After the APRM calibration, the IRM gains will be adjusted as necessary to optimize the IRM overlap with the SRMs and APRMs.

Acceptance Criteria - Level 1 - Each IRM channel must be adjusted so that overlap with the SRMs and APRMs is assured.

(ST-11) LPRM Calibration

Test Objectives - The objective of this test is to calibrate the Local Power Range Monitoring System.

Prerequisites - The required preoperational tests have been completed. Instrumentation for calibration has been checked.

Test Method - Prior to power operation, LPRM response to control rod movement is verified. Later in the program, the LPRM channels will be calibrated to make the LPRM readings proportional to the neutron flux in the water gap at the chamber elevation. Calibration factors will be obtained through the use of either an off-line or a process computer calculation that relates the LPRM reading to average fuel assembly power at the chamber height.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - Each LPRM will be within 10% of its calculated value.

(ST-12) APRM Calibration

Test Objective - The objective of this test is to calibrate the Average Power Range Monitoring (APRM) system.

Prerequisites - The required preoperational tests have been completed. Instrumentation for calibration has been checked.

Test Method - A heat balance will be made after each major power level change. Each APRM channel reading will be adjusted to be consistent with the core thermal power as determined from the heat balance. During heatup a preliminary calibration will be made by adjusting the APRM amplifier gains so that the APRM readings agree with the results of a constant heatup rate heat balance. The APRMs should be recalibrated in the power range by a heat balance as soon as adequate feedwater indication is available.

Acceptance Criteria - Level 1 - The APRM channels must be calibrated to read equal to or greater than the actual core thermal power.

(ST-13) NSSS Process Computer

Test Objective - The objective of this test is to verify the NSSS performance of the process computer under plant operating conditions.

Prerequisites - The required preoperational tests have been completed.

Test Method - During Test Condition 1, the TIP system will be aligned such that the TIP position plotted by the computer corresponds to the actual TIP position at steady power conditions between Test Conditions 1 and 3. The Dynamic System Test Case will be run to verify that the rsults of NSSS performance calculations are correct.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - Programs OD-1, P-1 and OD-6 will be considered operational when:

(1) The MCPR calculated by an independent method and the process computer either:

a. Are in the same fuel assembly and do not differ in value by more than 2% or,

b. For the case in which the MCPR calculated by the process computer is in a different assembly than that calculated by the independent method, for each assembly, the MCPR and CPR calculated by the two methods shall agree within 2%.

(2) The maximum LHGR calculated by the independent method and the process computer either:

a. Are in the same fuel assembly and do not differ in value by more than 2%, or

b. For the case in which the maximum LHGR calculated by the process computer is in a different assembly than that calculated by the independent method, for each assembly, the maximum LHGR and LHGR calculated by the two methods shall agree within 2%.

(3) The MAPLHGR calculated by the independent method and the process computer either:

a. Are in the same fuel assembly and do not differ in value by more than 2%, or

b. For the case in which the MAPLHGR calculated by the process computer is in a different assembly than that calculated by the independent method for each assembly, the MAPLHGR and APLHGR calculated by the two methods shall agree within 2%.

(4) The LPRM calibration factors calculated by the independent method and the process computer agree to within 2%.

(5) The remaining programs will be considered operational upon successful completion of the static and dynamic testing.

(ST-14) RCIC System

Test Objective - The objectives of this test are to verify the proper operation of the Reactor Core Isolation Cooling (RCIC) system over its expected operating pressure and flow ranges, and to demonstrate reliability in automatic mode starting from cold standby when the reactor is at power conditions.

Prerequisites - The required preoperational tests have been completed. Initial turbine operation (uncoupled) must have been performed to verify satisfactory operation and over-speed trip. Instrumentation has been installed and calibrated.

Test Method - The RCIC System is designed to be tested in two ways: (1) by flow injection into a test line leading to the Condensate Storage Tank (CST), and (2) by flow injection directly into the reactor vessel.

The earlier set of CST injection tests consist of manual and automatic mode starts at approximately 150 psig and near rated reactor pressure conditions. The pump discharge pressure during these tests is throttled to be approximately 100 psi above the reactor pressure to simulate the largest expected pipeline pressure drop. This CST testing is done to demonstrate general system operability and stability.

Reactor vessel injection tests follow which consist of manual and automatic mode starts near rated reactor pressure and automatic mode start at approximately 150 psig reactor pressure conditions to demonstrate operability and stability.

After all final controller and system adjustments have been determined, a defined set of demonstration tests must be performed with that one set of adjustments. Two consecutive reactor vessel injections starting from cold conditions in the automatic mode must satisfactorily be performed to demonstrate system reliability. Following these tests, a set of CST injections starting from cold conditions in the automatic mode are done to provide a benchmark for comparison with future surveillance tests. ("Cold" is defined as a minimum three days without any kind of RCIC operation.)

After the manual start portion of certain of the above tests is completed, and while the system is still operating, small step distrubances in speed and flow command are input (in manual and automatic mode respectively) in order to demonstrate satisfactory stability. This is to be done at both low (above minimum turbine speed) and near rated flow initial conditions to span the RCIC operating range. During testing at 150 psig, this is done only near rated flow initial conditions.

A demonstration of extended operation of up to 2 hours (or until pump and turbine oil temperature is stabilized) of continuous running at rated flow conditions is to be scheduled at a convenient time during the Startup test program.

<u>Acceptance Criteria - Level 1</u> - The average pump discharge flow must be equal to or greater than the 100% rated value after 30 seconds have elapsed from automatic initiation at any reactor pressure between 150 psig (10.5 kg/cm^2) and rated.

The RCIC turbine shall not trip or isolate during auto or manual start tests.

Note: If any Level 1 criteria are not met, the reactor will only be allowed to operate up to a restricted power level defined by Figure 14.2-7 until the problem is resolved. Also consult the plant Technical Specifications for actions to be taken.

<u>Level 2</u> - In order to provide an overspeed and isolation trip avoidance margin, the transient start first and subsequent speed peaks shall not exceed the rated RCIC turbine speed.

The speed and flow control loops shall be adjusted so that the decay ratio of any RCIC system related variable is not greater than 0.25.

The turbine gland seal condenser system shall be capable of preventing steam leakage to the atmosphere.

The delta P switch for the RCIC steam supply line high flow isolation trip shall be calibrated to actuate at 300% of the maximum required steady state flow, with the reactor assumed to be near the pressure for main relief valve actuation.

(ST-15) HPCI System

Test Objective - The objective of this test is to verify the proper operation of the High Pressure Coolant Injection (HPCI) system over its expected operating pressure and flow ranges, and to demonstrate reliability in automatic starting from cold standby when the reactor is at rated pressure conditions.

Prerequisites - The required preoperational tests have been completed. Initial turbine operation (uncoupled) must have been performed to verify satisfactory operation and over-speed trip. Instrumentation has been installed and calibrated.

Test Method - The HPCI system is designed to be tested in two ways: (1) by flow injection into a test line leading to the Condensate Storage Tank (CST), and (2) by flow injection directly into the reactor vessel.

The earlier set of CST injection tests consist of manual and automatic mode starts at approximately 150 psig and near rated reactor pressure conditions. The pump discharge pressure during these tests is throttled to be approximately 100 psi above the reactor pressure to simulate the largest expected pipeline pressure drop. This CST testing is done to demonstrate general system operability and stability.

Reactor vessel injection tests which consist of manual and automatic mode start near rated reactor pressure to demonstrate operability and stability.

After all final controller and system adjustments have been determined, a defined set of demonstration tests must be performed with that one set of adjustments. Two consecutive reactor vessel injections starting from cold condidtions in the automatic mode must satisfactorily be performed to demonstrate system reliability. Following these tests, a set of CST injections starting from cold conditions in the automatic mode ("cold" is defined to a minimum three days without any kind of HPCI operation) are done to provide a benchmark for comparison with future surveillance tests.

After the manual start portion of certain of the above tests is completed, and while the system is still operating, small step disturbances in speed and flow command are input (in manual and automatic mode respectively) in order to demonstrate satisfactory stability. This is to be done at both low (above minimum turbine speed) and near rated flow initial conditions to span the HPCI operating range.

A continuous running test is to be scheduled at a convenient time during the Startup Test Program. This demonstration of extended operation should be for up to 2 hours or until steady turbine and pump conditions are reached or until limits on plant operation are encountered.

Pump flow testing will also be verified since auxiliary boiler supply is insufficient to fully test the sytem during the Preoperational Test Program.

Acceptance Criteria - Level 1 - The average pump discharge flow must be equal to or greater than the 100% rated value after 25 seconds have elapsed from automatic initiation at any reactor pressure between 150 psig (10.5 kg/cm^2) and rated.

The HPCI turbine shall not trip or isolate during auto or manual start tests.

Level 2 - In order to provide an overspeed and isolation trip avoidance margin, the transient start first peak shall not come closer than 15% (of rated speed) to the overspeed trip, and subsequent speed peaks shall not be greater than rated speed.

The speed and flow control loops shall be adjusted so that the decay ratio of any HPCI system related variable is not greater than 0.25.

The turbine gland seal condenser system shall be capable of preventing steam leakage to the atmosphere.

The delta-P switch for the HPCI steam supply line high flow isolation trip shall be calibrated to actuate at 300% of the maximum required steady state flow, with the reactor assumed to be near the pressure for main relief valve actuation.

(ST-16) Selected Process Temperatures

Test Objectives - The objective of this procedure is to establish the proper setting of the low speed limiter for the recirculation pumps to avoid coolant temperature stratification in the reactor pressure vessel bottom head region.

Prerequisites - The required preoperational tests have been completed. System instrumentation has been calibrated.

Test Method - During initial heatup while at hot standby conditions, the bottom drain line temperature, recirculation loop suction temperature and applicable reactor parameters are monitored as the recirculation flow is slowly lowered to minimum stable flow. Utilizing this data it can be determined whether coolant temperature stratification occurs when the recirculation pumps are on and if so, what minimum recirculation flow will prevent it.

Monitoring the preceeding information during planned pump trips will determine if temperature stratification occurs in the idle recirculation loops or in the lower plenum when one or more loops are inactive.

Acceptance Criteria - Level 1 - The reactor recirculation pumps shall not be started nor flow increased unless the coolant temperatures between the steam dome and bottom head drain are within 145°F.

The recirculation pump in an idle loop must not be started unless the loop suction temperature is within 50°F of the active loop or within 70°F of steam dome temperature.

Level 2 - Not Applicable.

(ST-17) System Expansion

Test Objectives - The purposes of this test are to demonstrate that (1) the reactor drywell piping system during system heatup and cool-down is free to expand and move without unplanned obstruction or restraint, (2) the piping does "shakedown" after a few thermal expansion cycles, (3) the piping system is working in a manner consistent with the assumptions of the stress analysis, and (4) there is sufficient agreement between the calculated values of displacements and the measured values of displacement.

Prerequisities - Instrumentation has been installed and calibrated.

Test Method - Hanger positions of major equipment and piping in the Nuclear Steam Supply System and auxiliary systems in the reactor drywell are recorded after each major thermal cycle until a shakedown has taken place (normally about two cycles). During intitial heatup, a visual inspection is made at an intermediate reactor water temperature to assure components are free to move as designed. Adjustments are made as necessary. Devices for measuring continuous pipe deflections are mounted on main steam and recirculation lines and motion during heatup is compared with calculated values.

Acceptance Criteria - Level 1 - There shall be no obstructions which will interfere with the thermal expansion of the main steam and recirculation piping systems.

The displacements at the established transducer locations shall not exceed the allowable values.

Level 2 - The displacements at the established transducer locations shall not exceed the expected values.

(ST-18) TIP Uncertainty

Test Objectives - The objective of this test is to determine the uncertainty of the TIP system readings.

Prerequisites - System installation completed and required preoperational tests completed and verified. Instrumentation has been calibrated and installed.

Test Mehtod - The TIP uncertainty consists of a random noise component and a geometric component, the geometric component being due to variation in the water gap geometry and TIP tube orientation from TIP location to location. Measurement of these components is obtained by taking repetitive TIP readings at a single TIP location, and by analyzing pairs of TIP readings taken at TIP locations which are symmetrical about the core diagonal of fuel loading and control rod symmetry.

The random noise uncertainty is determined from successive TIP runs made at the common location (32-33) with each of the TIP machines making six runs at index position 10. The TIP data will be obtained by simultaneous operation of the Process computer OD-2 program which provides 24 nodal TIP values for each TIP traverse. The standard deviation of the random noise is derived by taking the square root of the average of the variances at nodal levels 5 through 22, where the nodal variance is obtained from the fractional deviations of the successive TIP values about their nodal mean value.

The total TIP uncertainty is determined by performing a complete set of TIP traverses as required by Process Computer program OD-1. The total TIP uncertainty is obtained by dividing the standard deviation of the symmetric TIP pair nodal ratios by the square root of 2. The nodal TIP ratio is defined as the nodal BASE value of the TIP in the lower right half of the core divided by its symmetric counterpart in the upper left half.

The geometric component of TIP uncertainty is obtained by statistically subtracting the random noise component from the total TIP uncertainty.

The TIP data will be taken with the reactor operating with an octant symmetric rod pattern and at steady state conditions. One set of TIP data will be taken at approximately 50% power and at least one other set at 75% power or above. The acceptance criteria for this subtest uses the "average uncertainties" for all data sets. Therefore additional performance of the subtest may be scheduled and the previous values of uncertainty will be used in the averaging to determine the acceptability of the results.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - The total TIP uncertainty (including random noise and geometrical uncertainties) obtained by averaging the uncertainties for all data sets must be less than 6.0%.

NOTE: A minimum of two and up to six data sets may be used to meet the above criteria.

(ST-19) Core Performance

Test Objectives - The objectives of this test are a) to evaluate the core thermal power and b) to evaluate the following core performance parameters: 1) maximum linear heat generation rate (MLHGR), 2) minimum critical power ratio (MCPR) and 3) maximum average planar linear heat generation rate (MAPLHGR).

Prerequisites - The required preoperational tests have been completed.

Test Method - The core performance evaluation is employed to determine the principal thermal and hydraulic parameters associated with core behavior. These parameters are:

Core flow rate

Core thermal power level

MLHGR

MCPR

MAPLHGR

Prior to the verification of the Process Computer in ST-13, an independent method will be used to calculate there parameters. After the successful completion of ST-13, the process computer will be used.

Acceptance Criteria - Level 1 - The Maximum Linear Heat Generation Rate (MLHGR) of any rod during steady-state conditions shall not exceed the limit specified by the Plant Technical Specifications.

The steady-state Minimum Critical Power Ratio (MCPR) shall not exceed the limits specified by the Plant Technical Specifications.

The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) shall not exceed the limits specified by the Plant Technical Specifications.

Steady-state reactor power shall be limited to the rated MWT and values on or below the licensed analytically determined power-flow line.

Level 2 - Not applicable.

(ST-20) Steam Production Verification

Test Objective - The objective of this test is to demonstrate that the NSSS is providing steam sufficient to satisfy all appropriate warranties.

Prerequisites - Required preoperational tests have been completed. All required instrumentation is installed and calibrated.

Test Method - A NSSS steam output performance test of 100 hours of continuous operation at the warranted steam output will be performed.

Acceptance Criteria - Level 1 - The NSSS parameters as determined by using normal operating procedures shall be within the appropriate license restrictions. Each NSSS shall be capable of supplying 13,432,000 pounds per hour of steam of not less than 99.7% quality at a pressure of 985 psia at the outlet of the second main steam line isolation valve, as based upon a final feedwater temperature of 380°F, measured as near the reactor pressure vessel as practicable, and a control rod drive feed flow of 39,000 pounds per hour at 80°F. The reactor feedwater flow must equal the steam flow less the rod drive feed flow. Thermal-dynamic parameters are consistent with 1967 ASME Steam tables. Correction techniques for conditions that differ from the preceeding will be mutually agreed to prior to the performance of the test.

Level 2 - Not applicable.

(ST-21) Core Power-Void Mode Response

Test Objectives - The objective of this test is to measure the stability of the core power-void dynamic response and to demonstrate that its behavior is within specified limits.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been calibrated.

Test Method - The core power void loop mode that results from a combination of the neutron kinetics and core thermal hydraulic dynamics is least stable near the natural circulation end of the rated 100 percent power rod line. A fast change in the reactivity balance is obtained by moving a very high worth rod only 1 or 2 notches and by simulating a failure of the pressure regulator.

Acceptance Criteria - Level 1 - The transient response of any system related variable to any test input must not diverge.

Level 2 - Not applicable.

(ST-22) Pressure Regulator

Test Objectives - The objectives of this test are to demonstrate the takeover capability of the backup pressure regulator upon failure of the controlling pressure regulator and to demonstrate smooth pressure control transition between the control valves and bypass valves when reactor steam generation exceeds steam flow used by the turbine.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - The pressure set point will be decreased rapidly and then increased rapidly by about 10 psi and the response of the system will be measured in each case. It is desirable to accomplish the set point change in less than 1 second. At specified test conditions the load limit setpoint will be set so that the transient is handled by control valves, bypass valves and both. The backup regulator will be tested by simulating a failure of the operating pressure regulator so that the backup regulator takes over control. The response of the system will be measured and evaluated.

Acceptance Criteria - Level 1 - The transient response of any pressure control system related variable to any test input must not diverge.

Level 2

a) Pressure control system related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25 when operating above lower limit of the automatic load following range.

b) When in the recirculation manual mode, the pressure response time from initiation of pressure setpoint step change to the turbine inlet pressure peak shall be ≤ 10 seconds.

c) Pressure control system deadband, delay, etc., shall be small enough that steady state limit cycles (if any) shall produce steam flow variations no larger than ±0.5 percent of rated steam flow.

d) The normal difference between regulator set points must be small enough that the peak neutron flux and/or peak vessel pressure remain below the scram settings by 7.5 percent and 10 psi respectively, for the Regulator Failure Test performed at Test Condition 6.

(ST-23) Feedwater System

Test Objectives - The objectives of this test are a) to demonstrate acceptable response to the feedwater control system for reactor water level control, b) to demonstrate stable reactor response to subcooling changes,i.e. loss of feedwater heating,c) to demonstrate the capability of the automatic core flow runback feature to prevent low water level scram following the trip of one feedwater pump, and d) to demonstrate the maximum feedpump runout capability is compatable with licensing assumptions.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Condition 1 - At Test Condition (TC) with the water level being automatically controlled using the low load valve and the recirculation system in Manual, ±5 inch step changes in the water level setpoint will be made to demonstrate proper response and operability of the feedwater system at low reactor power.

At Test Conditions 2, 3 and 6, with one feedwater pump in manual and the others in auto, a ±5% change in the manually controlled feed pump will be made. The response of the feedwater system to these steps will be analyzed and compared to the applicable acceptance criteria. The recirculation system will be in manual for these tests. At Test Conditions 1, 2, 3, 4, 5 & 6, with the recirculation system in manual, ±5 inch changes in the water level setpoint will be made to demonstrate proper response and stability of the feedwater system. At TC 6, this test will also be done with the recirculation system in auto.

At approximately 80% power, a simulated loss of power to the extraction steam bleeder-trip valves will be initiated which will result in the most severe restriction of extraction steam to one feedwater heater string. Recordings of the transient will be as analyzed and compared to the predicted response and acceptance criteria.

At Test Condtion 6, one feedwater pump will be tripped to demonstrate the capability to avoid a scram and prevent a low reactor water level trip due to the loss of one feedwater pump.

A maximum feedwater runout capability test will be done to demonstrate that the actual capability is compatible with licensing assumptions.

Acceptance Criteria - Level 1 - The transient response of any level control system-related variable to any test input must not diverge.

For the feedwater heater loss test, the maximum feedwater temperature decrease due to a single failure case must be less than or equal to 100°F. The resultant MCPR must be greater than the fuel thermal safety limit.

The increase in heat flux cannot exceed the predicted Level 2 value by more than 2%. The predicted value will be based on the actual test values of feedwater temperature change and power level.

The feedwater flow runout capability must not exceed the assumed value in the FSAR.

Level 2 - Level control system-related variables may contain oscillatroy modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

The open loop dynamic flow response of each feedwater actuator (turbine or valve) to small (§10%) step disturbances shall be:

(1)	Maximum time to 10% of a step disturbance	1.1 sec.
(2)	Maximum time from 10% to 90% of a step disturbance	1.9 sec.
(3)	Peak overshoot (% of step disturbance)	15%

The average rate of response of the feedwater actuator to large (20% of pump flow) step disturbances shall be between 10 percent and 25 percent rated feedwater flow/second. This average response rate will be assessed by determining the time required to pass linearly through the 10 percent and 90 percent response points.

The increase in heat flux cannot exceed the predicted value referenced to the actual Feedwater temperature change and the initial power level.

A scram must be avoided from low water level with a 3 inch margin following a trip of one of the operating feedwater pumps.

(ST-24) Turbine Valve Surveillance

Test Objectives - The objective of this test is to demonstrate acceptable procedures and maximum power levels for periodic surveillance testing of the main turbine control, stop, intercept and bypass valves without producing a reactor scram.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - Starting at 45 to 65% power, and continuing at progressively higher power levels, each turbine control, main stop and intermediate stop valve will be closed individually and the response of the reactor will be observed. The margin to scram for reactor pressure and neutron flux and the margin to main steam line isolation will be plotted for each tested power level. These plots will be used to determine the maximum power level at which turbine valve surveillance testing can be performed. The test of the control, main stop, intermediate stop and bypass valves are performed at the predicted highest power level to demonstrate that the Acceptance Criteria are satisfied. Rate of valve stroking and timing of the close-open sequence will be such that minimum practical disturbance is introduced and that PCIOMR limits are not exceeded.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - Peak neutron flux must be at least 7.5% below the scram trip setting. Peak vessel pressure must remain at least 10 psi below the high pressure scram setting. Peak steam flow in each line must remain 10% below the high flow isolation trip setting.

(ST-25) Main Steam Isolation Valves

Test Objectives - The objectives of this test are (a) to functionally check the main steam isolation valves (MSIVs) for proper operation at selected power levels, (b) to determine reactor transient behavior during and following simultaneous full closure of all MSIVs, (c) to determine isolation valve closure time and (d) to determine the maximum power at which a single valve closure can be made without a scram.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - The Main Steam Isolation Valves (MSIVs) are operated during this test to verify their functional performance and to determine closure times. While functionally testing the operation of the MSIVs, the time necessary for closing each individual valve will be noted. The fastest MSIV will then be tested to determine what power level an MSIV can experience fast closure without causing a scram. All MSIVs will later be used to demonstrate a full isolation subsequently leading to a scram. (The Nuclear Steam Supply Shutoff System (NSSSS) logic will be used to initiate the full isolation). The acceptability of the fast criteria (3 seconds) is determined by utilizing the full stroke time without dealy extrapolated from measured stroke times between 10% closed and 90% closed. The acceptability of the slow criteria (5 seconds) is determined by utilizing the full stroke time with delay extrapolated for the final 10% of stroke.

Acceptance Criteria - Level 1

The positive change in vessel dome pressure occurring within 30 seconds after closure of all MSIVs must not exceed predicted values by more than 25 psi.

The positive change in heat flux following closure of all MSIVs shall not exceed predicted values by more than 2% of rated value. Following the closure of all MSIV's, the reactor must scram to limit the severity of neutron flux and heat flux transients.

The average of the closure times for the fastest MSIV in each steam line, exclusive of electrical delay, shall not be less than 3.0 seconds.

Closure time for any MSIV, exclusive of electrical delay, shall not be greater than 5.0 seconds.

Closure time for the fastest MSIV shall be greater than or equal to 2.5 seconds.

Feedwater control settings must prevent flooding the main steam lines during the full isolation test.

The time delay between the close initiation signal and the extrapolated initial valve movement from 100% open for any MSIV shall be less than or equal to 0.5 seconds.

Level 2 - The positive change in vessel dome pressure occurring within the first 30 seconds after the closure of all MSIVs must not exceed the predicted values. Predicted values will be referenced to actual test conditions of initial power level and dome pressure and will use beginning of life nuclear data.

The change in heat flux occurring within the first 30 seconds after the closure of all MSIVs must not exceed the predicted values. Predicted values will be referenced to actual Test Conditions of initial power level and dome pressure and will use beginning of life nuclear data.

If water level reaches Level 2 setpoint during the MSIV fuel closer test, RCIC shall automatically initiate and reach rated flow.

During the MSIV full closure test, the relief valves must reclose properly (without leakage) following the pressure transient).

During full closure of individual MSIVs, peak vessel dome pressure must remain at least 10 psi below scram setting value.

During full closure of individual MSIVs, peak neutron flux must remain at least 7.5% below scram value.

During full closure of individual MSIVs, steam flow in individual lines must remain at least 10% below the high flow isolation trip setting.

During full closure of individual MSIVs, the peak heat flux must remain at least 5% less than its scram value.

If water level reaches the reactor vessel low water level (Level 2) setpoint, RCIC shall automatically initiate and reach rated flow.

(ST-26) Relief Valves

Test Objectives - The objectives of this test are to verify that the relief valves function properly, reseat properly after operation and contain no major blockages in the relief valve discharge piping.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate. Factory test results on SRV flow and operating times have been reviewed.

Test Method - Testing done at 250 psig reactor pressure consists of cycling each relief valve to verify proper operation. The transient monitoring system will be used to record the results of this test. The data collected will compare the operation of individual relief valves against the operation of all relief valves. During relief valve operation, core power - and therefore steam generation rate - is maintained constant. The pressure control system will close the bypass valves an amount proportional to the relief valve steam flow to maintain constant reactor pressure. This bypass valve motion will be monitored and a comparison of the response for each relief valve operation will be made. If differences exist, it could suggest a partial obstruction of the relief valve or its tailpipe. Tailpipe temperature will be recorded to verify the relief valve has properly reseated. Reactor variables will also be recorded to verify system stability during opening and closing each relief valve.

Testing done at rated reactor pressure consists of manually operating each relief valve at rated reactor pressure. The decrease in Main Generator output will be monitoried during the operation of each relief valve to provide an indication of relief valve flow. By comparison of the generator output response for each relief valve operation, any flow obstruction in the valve or its tailpipe can be identified. Each valve will be opened for approximately 10 seconds to allow for variables to stabilize. Reactor variables will also be recorded to verify system stability during opening and closing each relief valve.

Acceptance Criteria - Level 1

There should be a positive indication of steam discharge during the manual actuation of each valve.

Level 2 - Pressure control system-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

The temperature measured by thermocouples on the discharge side of the valves shall return to within 10°F of the temperature recorded before the valve was opened.

During the 250 psig functional test, the steam flow through each relief valve, as measured by the initial and final bypass valve position, shall not be less than 10% of bypass valve position under the average of all bypass valve responses.

During the rated pressure test, the steam flow through each relief valve, as measured by MWe shall not be less than 0.5% of rated MWe less than the average of all the valve responses.

(ST-27) Turbine Trip and Generator Load Rejection

Test Objectives - The objective of this test is to demonstrate the response of the reactor and its control systems to protective trips in the turbine and generator.

Prerequisites - The required preoperational tests have been completed. All instrumentation has been calibrated.

Test Method - At Test Condition 3, a turbine trip will be manually initiated by depressing the Turbine Trip pushbutton in the main control room. At Test Condition 6, a generator load rejection will be manually initiated by remotely opening the generator synchronizing breaker from the control room. During both transients, reactor water level, pressure, neutron flux and simulated heat flux will be recorded and compared to predicted results and acceptance criteria.

At approximately 24% power, a generator load rejection within bypass capacity will be manually initiated as described above. This will demonstrate the ability to ride through a load rejection within bypass capacity without a scram.

During all 3 transients, main turbine stop, control and bypass valve positions and reactor water level will be recorded and compared to the acceptance criteria.

Acceptance Criteria - Level 1

a) For Turbine and Generator trips there should be a delay of no more than 0.1 seconds following the beginning of control or stop valve closure before the beginning of bypass valve opening. The bypass valves should be opened to a point corresponding to greater than or equal to 80 percent of full open within 0.3 seconds from the beginning of control or stop valve closure motion.

b) Feedwater system settings must prevent flooding of the steam line following these transients.

c) The positive change in vessel dome pressure occurring within 30 seconds after either generator or turbine trip must not exceed the Level 2 criteria by more than 25 psi.

The positive change in simulated heat flux shall not exceed the Level 2 criteria by more than 2% of rated value.

Level 2

a) There shall be no MSIV closure in the first 3 minutes of the transient and operator action shall not be required in that period to avoid the MSIV trip.

b) The positive change in vessel dome pressure and in simulated heat flux which occur within the first 30 seconds after the initiation of either generator or turbine trip must not exceed the predicted values.

(Predicted values will be referenced to actual test conditions of initial power level and dome pressure and will use beginning of life nuclear data.

c) For the Generator trip within the bypass valves capacity, the reactor shall not scram for initial thermal power values less than or equal to 25 percent of rated.

(ST-28) Shutdown from Outside the Main Control Room

Test Objective - The objective of this test is to demonstrate that the reactor can be shutdown, maintained in a hot shutdown condition, and cooled from hot shutdown to cold shutdown, all from outside the main control room.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - While operating at approximately 30% power synchronized to the grid with normal electrical system alignment, the control room will be evacuated and the reactor scrammed from outside the control room. Reactor level and pressure will then be controlled from outside the main control room. The Shutdown Cooling mode of RHR will be placed into service with water supply from the ultimate heat sink. During this demonstration, some supervisory and operating personnel will remain in the control room to protect non-safety-related equipment from unnecessary damage if conditions arise and to assume control of the plant if conditions warrant.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - During a simulated control room evacuation, the reactor must be brought to the point where cooldown is initiated and under control, and the reactor vessel pressure and water level are controlled using equipment and controls outside the control room. The test is deemed successful when reactor pressure is less than 135 psig and the RHR shutdown cooling system has been put in operation.

(ST-29) Recirculation Flow Control System

The objectives of this test are:

a) To demonstrate the flow control capability of the plant over the entire pump speed range, including individual local manual, combined Master Manual Operation and Automatic Load Following.

b) To determine that all electrical compensators and controllers are set for desired system performance and stability.

Prerequisites - The required preoperational tests have been completed.

All instrumentation has been calibrated.

Test Method - At Test Conditions 1, 2, 3, 5 and 6, the stability of the recirculation flow control system is demonstrated by performing step changes in recirculation pump speed while in the manual mode. This testing is also done in the auto mode at Test Conditions 3, 5 and 6. Ramp changes in recirculation pump speed are done along the 100% rod line to demonstrate operability and stability in the auto mode and to set the lower limit of auto mode operation.

Acceptance Criteria - Level 1 - The transient response of any system-related variable to any test input must not diverge.

Level 2 - A scram shall not occur due to recirculation flow control maneuvers.

The APRM neutron flux trip avoidance margin shall be greater than or equal to 7.5% when the power maneuver effects are extrapolated to those that would occur along the 100% rated rod line.

The Automatic Load Following (ALF) range along the 100% rated rod line shall be as specified in the plant contract. The power changes in the ALF mode for small, medium or maximum load change commands shall meet the time requirements specified in the plant contract.

The decay ratio of any oscillatory controlled variable must be less than or equal to 0.25.

Steady state limit cycles (if any) shall not produce turbine steam flow variations greater than $\pm$.5% of rated steam flow.

In the scoop tube reset functions, the speed demand meter must agree with the speed meter within 6% of rated generator speed, along the 100% rated rod line.

(ST-30) Recirculation System

Test Objectives - The objectives of this test are:

a. Obtain recirculation system performance data during pump trip, flow coastdown, and pump restart.

b. Verify that the feedwater control system can satisfactorily control water level without a resulting turbine trip and associated scram.

c. Record and verify acceptable performance of the recirculation two pump circuit trip sytem.

d. Verify the adequacy of the recirculation runback to mitigate a scram upon loss of one feedwater pump.

e. Verify that no recirculation system cavitation will occur in the operable region of the power-flow map.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - Single recirculation pump trips will be made at Test Condition (TC) 3 and TC-6. These trips will be initiated by tripping the M-G Set Drive Motor Breaker from the control room. Reactor parameters will be recorded during the transient and analyzed to verify non-divergence of oscillatory responses, adequate margins to RPS scram set points, and capability of the feedwater system to prevent a high level trip. The capability to restart the recirc. pump at a high power level will also be demonstrated. At TC-3, both recirculation pumps RPT breakers will be simultaneously tripped using a temporarily installed test switch. The data gathered will be used to demonstrate acceptable pump coastdown performance prior to high power turbine trips and generator load rejects.

A loss of a reactor feed pump will be simulated at TC-3 to demonstrate the proper operation of the recirculation pump runback circuits. This is done prior to an actual planned feed pump trip.

Both the jet pumps and the recirculation pumps will cavitate at conditions of high flow and low power where NPSH demands are high and little feedwater subcooling occurs. However, the recirculation flow will automatically runback upon sensing a decrease in feedwater flow. The maximum recirculation flow is limited by appropriate stops which will run back the recirculation flow from the possible cavitation region. At TC-3, it will be verified that these limits are sufficient to prevent operation where recirculation pump or jet pump cavitation occurs.

Acceptance Criteria - Level 1 - The response of any level related variables during a pump trip must not diverge.

The two pump drive flow coastdown transient, during the first 3 seconds of an RPT trip, must be equal to or faster than specified.

Level 2 - The reactor shall not scram during the one pump trip.

The APRM margin to avoid a scram shall be at least 7.5% during the one pump trip recovery.

The reactor water level margin to avoid a high level trip shall be at least 3.0 inches during the one pump trip.

The data from the two pump drive flow coastdown transient shall be analyzed by G.E. - San Jose to ensure compatability with the safety analysis.

Runback logic shall have settings adequate to prevent recirculation pump operation in areas of potential cavitation.

The recirculation pumps shall runback upon a trip of the runback circuit.

(ST-31) Loss of Turbine-Generator and Offsite Power

Test Objectives - The objectives of this test are to demonstrate that the required safety systems will initiate and function properly without manual assistance, the electrical distribution and diesel generator systems will function properly, and the HPCI and/or RCIC systems will maintain water level if necessary during a simultaneous loss of the main turbine-generator and offsite power.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - With the unit synchronized to the grid at approximately 30% power, the main turbine-generator will be manually tripped immediately followed by a manual trip of the unit's offsite power source breaker, both trips initiated from the control room. To ensure a full simulation of the loss of all offsite power to Unit 1 during Unit 1 testing, all Unit 1 and Common loads will be transferred to Unit 1 Auxiliary and Startup Busses and appropriate breakers racked out to prevent automatic transfer of the loads to Unit 2 sources. During Unit 2 testing, to ensure a full simulation of the loss of all offsite power to Unit 2 while minimizing the impact on Unit 1 operations, all Unit 2 loads will be transferred to Unit 2 Auxiliary and Startup busses, all Unit 1 and common loads will be transferred to Unit 1 Auxiliary and Startup Busses, and appropriate breakers will be racked out to prevent automatic transfer of Unit 2 loads to Unit 1 sources.

Reactor water level and the operation of safety systems will be monitored to verify that the acceptance criteria are satisfied. The proper response of the electrical distribution system will be checked.

The loss of offsite power condition will be maintained for at least 30 minutes to demonstrate that necessary equipment, controls, and indication are available following station blackout to remove decay heat from the core using only emergency power supplies and distribution system.

Acceptance Criteria - Level 1 - All safety systems, such as the Reactor Protection System, the diesel-generator, RCIC and HPCI must function properly without manual assistance, and HPCI and/or RCIC system action, if necessary, shall keep the reactor water level above the initiation level of Core Spray, LPCI and ADS.

Level 2 - Not applicable.

(ST-32) Containment Atmosphere and Main Steam Tunnel Cooling

Test Objective - The objective of this test is to verify the ability of the drywell coolers and circulation fans and the Zone I reactor building HVAC system and the main steam tunnel coolers to maintain design conditions in the drywell and reactor building portion of the mainsteam tunnel, respectively, during operating conditions and post scram conditions.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - During heatup, at test conditions 2 and 6, and following a planned scram from 100% power, data will be taken to ascertain that the containment atmospheric conditions are within design limits.

Acceptance Criteria - Level 1 - not applicable

Level 2 - The general drywell area is maintained at an average temperature less than or equal to 135°F, with maximum local temperature not to exceed 150°F except for the areas beneath the reactor pressure vessel and around the recirculation pump motors which are specified in succeeding criteria.

The area beneath the reactor pressure vessel is maintained at an average temperature less than or equal to 135°F, maximum local temperature not to exceed 165°F, with minimum local temperature above 100°F.

The area around the recirculation pump motors is maintained at an average temperature less than or equal to 128°F, with maximum local temperature not to exceed 135°F.

The drywell head area is maintained at temperatures between 135°F and 150°F.

The reactor building portion of the mainsteam pipeway is maintained at or below 120°F.

(ST-33) Drywell Piping Vibration

Test Objectives - The purpose of this test is to 1) assess that vibration levels from the various dynamic loadings during transients and steady-state conditions, 2) assure that long term fatique failure will not occur due to underestimating dynamics effects caused by cyclic loading during plant transient operations, and 3) assure that vibration levels at the recirculation and main steam piping systems meet acceptable limits.

Prerequisites - Instrumentation has been installed and calibrated.

Test Method - Devices for measuring continuous vibration are mounted on main steam and recirculation lines and vibration during steady state and transient operation is compared with calculated values.

Acceptance Criteria - Level 1 - The measured amplitude (peak to peak) of main steam and recirculation line vibration shall not exceed the allowable values.

Level 2 - The measured amplitude (peak to peak) of main steam and recirculation line vibration shall not exceed the expected values.

(ST-34) Control Rod Sequence Exchange

(This test number was previously assigned to the RPV Internals Vibration test which is now performed during the Preoperational Test Program. The test description for the RPV Internals Vibration test is now in TP2.16 which follows the abstract for P64.1.)

Test Objective - The objective of this test is to perform a representative sequence exchange of control rod patterns at the power level at which such exchanges will be done during plant operation and demonstrate that core limits and PCIOMR threshold limits will not be exceeded.

Prerequisites - Instrumentation has been checked or calibrated as appropriate.

Test Method - The control rod sequence exchange begins on the 100 percent load line by reducing core flow to minimum and reducing thermal power to between the low power set point of the rod worth minimizer (or the rod sequence control system) and the thermal power necessary to keep nodal powers below the PCIOMR threshold. Also, in reducing thermal power, care is taken to avoid exceeding the design limits of the core total peaking factor. The ensuing steps involve utilizing the system process computer followed by APRM data and extensive utilization of the TIP machines. The exchange is performed a row or column at a time. Acceptance Criteria are verified by taking and analyzing TIP traces of the affected TIP locations taken at the completion of every two rows (or columns).

Acceptance Criteria - Level 1 - Completion of the exchange of one rod pattern for the complimentary pattern with continual satisfaction of all licensed core limits constitutes satisfaction of the requirements of this procedure.

Level 2 - All nodal powers shall remain below their PCIOMR threshold limit during this test.

(ST-35) Recirculation System Flow Calibration

Test Objectives - The objective of this test is to perform a complete calibration of the installed recirculation system flow instrumentation.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - During the testing program at operating conditions which allow the recirculation system to be operated at speeds required for rated flow at rated power, the jet pump flow instrumentation will be adjusted to provide correct flow indication based on the jet pump flow.

After the relationship between drive flow and core flow is established, the flow biased APRM/RBM system will be adjusted to match this relationship.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - Jet pump flow instrumentation shall be adjusted such that the jet pump total flow recorder will provide a correct core flow indication at rated conditions.

The APRM/RBM flow-bias instrumentation shall be adjusted to function properly at rated conditions.

(ST-36) Cooling Water Systems

Test Objectives - The objective of this test is to verify that the performance of the Reactor Building Closed Cooling Water (RBCCW), the Turbine Building Closed Cooling Water (TBCCW), and Service Water Systems are adequate with the reactor at rated temperature.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate.

Test Method - With the reactor at rated pressure, following initial heatup, data will be obtained to verify that the heat exchanger outlet temperatures are within design values. Flow rate adjustments will be made as necessary to achieve satisfactory system performance. The test will be repeated at selected power levels to verify continued satisfactory performance with higher plant heat loads.

Acceptance Criteria - Level 1 - Not applicable.

Level 2 - All controls operate properly and heat exchanger capacities meet the requirements of design specifications.

(ST-37) Gaseous Radwaste System

Test Objectives - The objective of this test is to demonstrate the proper operation of the Gaseous Radwaste System and the containment nitrogen inerting system during plant operation.

Prerequisites - The required preoperational tests have been completed. Instrumentation has been checked or calibrated as appropriate. In addition, the 100% power trip testing shall have been completed or 120 effective full power days shall not have elapsed prior to performing the nitrogen inerting test.

Test Method - The pressure, temperature, relative humidity, system flow and percentage of radiolytic hydrogen in the offgas are periodically monitored during startup and at steady-state conditions. Provided that measurable and sufficient fission gases and fission gas daughter products are present in the offgas, decontamination factors across several charcoal beds are determined. For the nitrogen inerting system, the proper nitrogen concentration will be verified by the as installed plant oxygen detectors/instruments in the two major volumes of the primary containment.

Acceptance Criteria - Level 1 - The release of radioactive gaseous and particulate effluents must not exceed the limits specified in the site technical specifications. There shall be no loss of flow of dilution steam to the noncondensing stage when the steam jet air ejectors are pumping.

Level 2-The system flow, pressure, temperature, and relative humidity shall comply with design specifications. The catalytic recombiner, the hydrogen analyzer, the activated carbon beds and the filters shall be performing their required function.

The containment nitrogen inerting system shall be capable of inerting the primary containment free volume within 24 hours from the start of the test and the resulting oxygen concentration shall be less than or equal to 4%.

(ST-38) BOP Piping System Expansion

Test Objectives - The purpose of this test is to demonstrate that the piping systems identified in Table 3.9-33 respond to thermal expansion consistent with stress analysis results.

Prerequisites - Engineering review of the piping systems is completed. Instrumentation has been installed and calibrated.

Test Method - Devices for measuring continuous pipe deflections are mounted on piping systems which are inaccessable during power operations and motion during operation is compared with calculated values. Accessable piping systems will be walked down and inspected visually.

Acceptance Criteria - Level 1 - There shall be no obstructions which will interfere with the designed thermal expansion of the tested piping systems.

The displacements at the established transducer locations shall not exceed the allowable values.

Level 2 - The displacements at the established transducer locations shall not exceed the expected values.

(ST-39) BOP Piping Dynamic Transients

Test Objective - The objective of this test is to demonstrate that system piping identified in Table 3.9-33 is adequately designed and restrained to withstand the transient loading condition listed in Table 3.9-33.

Prerequisites: Engineering review of the piping system after construction is completed. Instrumentation has been installed and calibrated.

Test Method - Devices for measuring continuous loads, displacements, accelerations and pressures are mounted on piping systems and responses during transients are compared with calculated values. Those portions of the systems which are non-safety related are visually inspected prior to, during and subsequent to the transient loading condition.

Acceptance Criteria - Level 1 - The maximum measured loads, displacements, accelerations and pressures shall not exceed the allowable values.

Level 2 - The maximum measured loads, displacements, accelerations and pressures shall not exceed the expected values. On visually inspected systems, no excessive piping response was observed and that post transient walkdown shows no signs of excessive piping response (such as damaged insulation, markings on piping, structural or hanger steel, or walls, damaged pipe supports, etc.).

(ST 40) BOP Piping Steady State Vibration

Test Objective - The objective of this test is to demonstrate that the steady state vibratory levels of the piping systems listed in Table 3.9-33 are within acceptable limits.

Prerequisities - Engineering review of the piping system is complete. Instrumentation has been installed and calibrated.

Test Method - Devices for measuring continuous vibration are mounted on piping systems which are inaccessable during power operations and vibrations during steady state operations are compared with calculated values. Accessable piping systems will be walked down and inspected visually.

Acceptance Criteria - Level 1 - The maximum measured amplitude (peak to peak) shall not exceed the allowable values.

Level 2 - The maximum measured amplitude (peak to peak) shall not exceed the expected values. The piping may be qualified as acceptable, if upon visual examination, a qualified test engineer judges the vibratory response to be within acceptable limits.

14.2.12.3 Requested Acceptance Test Procedure Abstracts

Tests comprising the Acceptance Test procedures are listed in Table 14.2-2. For each test a description is provided for objective, prerequisites, method and acceptance criteria, where applicable. Modifications to these descriptions will be reflected in amendments to the FSAR.

A3.1 13.8 KV SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate the capability of the 13.8 kV system to provide electrical power to the Startup and Unit Auxiliary 13.8 kV Busses by demonstrating the proper operation of breakers, relaying and logic, permissive and prohibit interlocks, and instrumentation and alarms.

Prerequisites - Construction is completed to the extent necessary to perform this test and the systems are turned over to the ISG. Required 230 kV transmission lines are available to energize the 13.8 kV system. Required instruments and protective relays are calibrated and controls are operable.

Test Method - Breakers are opened and closed by operating or simulating controls to verify breaker operation, relaying and logic, permissive and prohibit interlocks, instrumentation and alarms, and automatic transfers.

Acceptance Criteria - The system performance parameters are in accordance with applicable design documents.

A7.1 LIGHTING SYSTEM AND MISCELLANEOUS 120V DISTRIBUTION ACCEPTANCE TEST

Test Objectives - To demonstrate the ability of the Plant Lighting System to provide normal lighting in all areas of the plant and emergency lighting in designated areas of the plant.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Normal and essential 480 volt AC and 125 volt DC power is available. Required test instruments are calibrated and controls are operable.

Test Method - The systems operation is tested by measuring voltage regulation and by verifying proper switchover from normal to emergency power. The emergency DC lighting is tested to provide lighting in designated areas upon loss of all AC power.

Acceptance Criteria - The system performance parameters are in acccrdance with the applicable design document.

A11.1 STATION SERVICE WATER SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate the capability of Station Service Water System to provide cooling water to connected components/systems.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are cperable. Required electrical pcwer supply systems are available. Water supply frcm the cooling tower is available.

Test Method - System cperation is initiated normally. The system is cperated in the different design modes and Service Water Pump performance is determined. Required controls are operated or simulated signals are applied tc verify automatic features, system interlocks and alarms.

Acceptable Criteria - The system performance parameters are in accordance with applicable design documents.

A15.1 TBCCW SYSTEM ACCEPTANCE TEST

Test Objective - To demcnstrate proper operation of the TBCCW system, specifically to furnish cooling water to miscellaneous turbine plant heat exchangers, coclers, and chillers, and to demcnstrate the abiltiy of a standby pump to automatically replace the operating pump in case of pressure loss in the header.

Prerequisites - Construction is completed to the extent necessary tc perform this test and the system is turned over to the ISG. Required electrical pcwer supply systems are available to energize the necessary 480 volt mctcr control centers. Required insruments are calibrated and contrcls are available. The service water system is available. The instrument air system is available.

Test Method - The system operation is initiated manually, and where applicable automatically. The system is operated in the system design modes and TBCCW pumps performance is determined. Required ccntrols are operated cr simulated to verify automatic system functions and alarms.

Acceptance Criteria

1) Each of the two TBCCW pumps is capable of delivering a minimum flow of 292.5 gpm.

2) With one pump in operation, the standby pump starts automatically at a low header pressure of less than or equal to 70 psig.

3) The TBCCW system provides cooling water to the following:

   a. Control rod drive pump bearing and oil coolers
   b. Condensate pump motor bearing coolers
   c. Instrument air compressor coolers
   d. Service air compressor coolers
   e. EHC fluid coolers
   f. Turbine Building sample station chillers
   g. Auxiliary Boiler sample station chillers

A18.1 INSTRUMENT AIR SYSTEM ACCEPTANCE TEST

Test Objective - The general objective of this test is to demonstrate proper operation of the Instrument Air System. Specific objectives are to demonstrate the following:

1) The ability of the Instrument Air System to provide air to outlets located throughout the plant.

2) System controls function in accordance with design intent.

3) Alarms function properly to provide alert of an abnormality in the Instrument Air System.

4) Instrument air dryers reduce instrument air moisture in accordance with design requirements.

5) Standby Instrument Air Unit, under AUTO Mode, starts automatically when the system pressure is down.

Prerequisites - Construction turnover of the system is complete to the extent required to conduct the test. The system has been walked through, verified complete and air blowing has been completed. The required Technical Tests have been completed and the required instruments are calibrated.

Test Method - Both compressors are fully tested in both Manual and Auto mode of operation. The Dryer packages are tested for effectiveness and all automatic trips and alarms are verified.

Acceptance Criteria - The system performance parameters are in accordance with applicable engineering design documents.

A19.1 SERVICE AIR SYSTEM ACCEPTANCE TEST

Test Objectives - The objectives of this test are as follows:

1) To demonstrate that the compressors can provide pressurized air (115-130 psig) to outlets located throughout the plant.

2) To demonstrate that system controls and alarms function in accordance with the design intent.

3) To demonstrate that the standby compressor will start automatically if the system pressure is low.

Prerequisites - The prerequisites of this test are as follows:

1) Construction is complete to the extent necessary to conduct this test and system is turned over to ISG.

2) All component inspections, tests and calibrations have been completed satisfactorily.

Test Method - The system will be pressurized by starting the compressors. Compressor modes and functions will be checked for proper operation. Alarms will be verified as they are induced during normal operation or simulation.

Acceptance Criteria

1) The service air compressors have the capacity to deliver 440 scfm of air each and provide air to outlets located throughout plant.

2) The compresors will automatically trip when an abnormal condition exists and alarms perform their design function.

3) The standby compressor will automatically start if the lead compressor fails or if its operation cannot meet service air system demand.

4) The Service Air System is capable of providing backup supply to the Instrument Air System

A20.1 BUILDING DRAINS - NON RADIOACTIVE ACCEPTANCE TEST

Test Objectives - The objectives of this test are as follows:

1) To demonstrate that system controls and alarms function in accordance with the design intent.

2) To demonstrate the waste filter is capable of automatically dewatering sludge.

3) To demonstrate the diesel generator floor drain sump pumps operate automatically.

Prerequisites - Construction is complete to the necessary extent and the system is turned over to ISG. Required instrumentation is calibrated and controls are operable. Required electrical power supply systems are available. Instrument air is available.

Test Method - Low, High and High-High sump levels are simulated to verify pumps start and stop as required.

Acceptance Criteria - The system performs in accordance with design documents.

A22.1 MAKEUP DEMINERALIZER SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate the capability of the Makeup Demineralizer System to provide quality water consistant with the requirements of the Final Safety Analysis Report.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to ISG. All instrumentation contained in this system is calibrated and the controls are operational. The Water Pretreatment System and the Neutralization Basins are available.

Test Method - A normal, automatic regeneration of makeup demineralizers shall be performed verifying all regeneration sequence interlocks and verifying that the Makeup Demineralizer conforms to FSAR requirements.

All interlocks shall be verified that will remove the Makeup Demineralizer from service upon its effluent water quality not meeting specifications.

Acceptance Criteria - The Makeup Demineralizer shall be capable of making water in accordance with FSAR requirements at a flow rate between 20 and 120 gpm. It shall also be capable of performing automatic shutdowns, startups and regenerations per its design requirements.

A30.3 CONTROL STRUCTURE MISCELLANEOUS H&V SYSTEM ACCEPTANCE TEST

Test Objectives - To demonstrate that the Control Structure Miscellaneous H&V maintains temperature and delivers an adequate air supply and exhaust to various areas in accordance with design requirements.

Prerequisites - Construction is complete and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. The Turbine Building Vent and Instrument Air Systems are in service. Required electrical power supply systems are available.

Test Method - The system operation is initiated manually and fan performance, damper operations and heating or cooling operation (where applicable) are determined. Required controls are operated or simulated signals are applied to verify fan interlocks, high-high temperature from charcoal filters (where applicable), electric duct heater operation and associated alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A31.1 COMPUTER UNINTERRUPTIBLE POWER SUPPLY ACCEPTANCE TEST

Test Objective - The general objective of this test is to demonstrate proper operation of the Computer Uninterruptible Power Supply. Specific objectives are to demonstrate the following:

1) The ability of the static transfer switch to provide automatic transfer of the 120 VAC distribution panel loads from the preferred to the alternate supply on loss of the preferred supply or overcurrent or in case of load side fault.

2) The ability of the manual transfer switch and manual operation of the static transfer switch to transfer distribution panel loads between the preferred and the alternate source.

Prerequisites - Construction turnover of the system is complete to the extent required to conduct this test. The system has been walked through and verified complete. The required Technical Tests have been completed and the required instruments are calibrated.

Test Method - The power supply is operated at full load, the static transfer switch is tested, the manual transfer is tested and all alarms and computer inputs associated with the system are verified.

Acceptance Criteria - The system performance parameters are in accordance with applicable engineering design documents.

A31.2 PROCESS COMPUTER ACCEPTANCE TEST

Test Objective - The objective of this test is to demonstrate proper operation of the computer. Specific objectives are to demonstrate the ability of the DCS to monitor unit operation and generate video displays for operator use; the PMS to perform BOP calculations, log data, make historical records, generate video displays and generate alarm status summary displays; the NSS subsystem program to provide an accurate determination of the core thermal performance and data loading, and to supplement procedural requirements for control and manipulation during reactor startup and shutdown.

Prerequisites - Construction turnover of the system is complete to the extent required to conduct this test. The system has been walked through and verified complete. The required Technical Tests have been completed and the required instruments are calibrated.

Test Method - Computer inputs are verified, the software programs are tested and computer self-protection and alarm functions are verified.

Acceptance Criteria - The system performance parameters are in accordance with applicable engineering design documents.

A33.1 TURBINE BUILDING HEATING & VENTILATING SYSTEM ACCEPTANCE TEST

Test Objectives - The objectives of this test are as follows:

1) To provide filtered and tempered air to all areas of the Turbine Building.

2) To maintain air flow from areas of lesser potential contamination to areas of greater potential contamination.

3) To exhaust air from potentially contaminated spaces to particulate and charcoal filters.

4) To maintain the Turbine Building at a slightly negative pressure with respect to atmosphere to minimize exfiltration to outside atmosphere.

5) To recirculate and cool Turbine Building air to reduce exhaust volume.

6) To discharge all exhaust air through the Turbine Building Exhaust Vent.

7) To supply cool air to the Reactor Recirculation Motor - Generator sets.

Prerequisites

1) Flow balancing is completed

2) Instrument Air System is operational.

3) Fire Protection System is operational.

Test Method - The system will be tested with manual controls and automatically where applicable. All interlocks, start and trip schemes will also be verified.

Acceptance Criteria

1) Maintain building temperature above 40°F.

2) Maintain building spaces below the following maximum temperatures:

   a) General areas 104° F

   b) Electrical rooms 104° F

   c) Mechanical areas 120° F

A33.2 TURBINE BUILDING CHILLED WATER SYSTEM ACCEPTANCE TEST

Test Objectives - The objectives of this test are as follows:

1) To demonstrate the ability of the Turbine Building Chilled Water System to maintain design temperature.

2) To demonstrate the ability of the Service Water System to remove the chiller condenser heat.

Prerequisites

1) Construction is complete to the extent required to complete this test.

2) The following systems are operational:

a) Instrument Air System

b) Turbine Building H&V is functionally checked

c) Service Water System

d) Makeup Demineralizers

e) Expansion tank IT-123 is filled halfway and pressurized to 20 psi

Test Method - The system will be initiated manually and automatically with all automatic functions verified. All interlocks will be verified and alarms checked as they occur during normal process variation.

Acceptance Criteria - Turbine Building Chilled Water System will supply water at 50°F.

A35.1 FUEL POOL COOLING AND CLEANUP SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate that the Fuel Pool Cooling and Cleanup System filters, demineralizes and cools the fuel pool water. The system is able to maintain a minimum differential pressure in the heat exchangers and will prevent siphoning of water from the fuel pool to any cooling water supply line.

Prerequisite - Construction is complete to the extent necessary to perform this test and the system is turned over to ISG. Required instruments are calibrated and controls are operable. The Demineralized Water Transfer System, Service Water System, Sample System, Condensate System, Instrument Air System, Residual Heat Removal System, Liquid Radwaste Drain System, Emergency Service Water System, Solid Radwaste System and required electrical power supply systems are available.

Test Method - The system is operated to demonstrate the demineralizer heat exchangers and fuel pool cooling pumps operation. Required controls are operated or simulated signals are applied to verify system operation, automatic valve alignment and system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design requirements.

A37.1 DEMINERALIZED WATER TRANSFER SYSTEM ACCEPTANCE TEST

Test Objectives - To demonstrate proper operation of the Demineralized Water Transfer system by verifying the following: The ability to supply condensate for various plant systems, including the condenser hotwells. The ability to supply condensate to the suction of the high pressure coolant injection(HPCI), reactor core isolation cooling (RCIC), core spray, and control rod drive (CRD) pumps. The ability to supply demineralized water as makeup to the reactor, radwaste, and closed coolant systems. The ability to supply demineralized water to the condensate storage tank & refueling water storage tank.

Prerequisites - Construction is complete to the extent necessary to peform this test and the system is turned over to ISG. Hydrostatic testing, velocity flushing and air blowing have been complete to the extent required to perform this test. Required instruments are calibrated and controls are operable. Required electrical power supply systems, makeup demineralizers, and instrument air are available. The associated plant systems which are capable of receiving water from the Demineralized Water System are available to the extent required to perform this test.

Test Method - The operating modes of this system are initiated manually and, where applicable, automatically. The system is operated to determine performance of all pumps. Control devices are operated or simulated signals are applied to verify system automatic functions and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents. All automatic trips and alarms actuate within their allowable limits.

A38.1 LOW PRESSURE AIR SYSTEM ACCEPTANCE TEST

Test Objectives - The objective of this test is to demonstrate proper operation of the Low Pressure Air System; specifically to demonstrate the ability to provide air for the liquid radwaste filters, and the liquid radwaste demineralizers, as these processes require. The ability to provide backup air to the cement silo and to operate intermittently on demand is demonstrated. The protection of the compressor against low oil pressure, high oil temperature, high air discharge temperature, high cooling water temperature and low cooling water pressure is demonstrated.

Prerequisites - Construction is complete to the extent necessary to perform the test and the system is turned over to the ISG.

Required instruments are calibrated and Technical Tests are complete.

Test Method - The system is operated in the Manual and Automatic modes of operation. The Flow Rate is verified and all trips and alarms are tested.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A39.1 CONDENSATE DEMINERALIZER SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate the ability of the Condensate Demineralizer System to process full condensate flow producing effluent of acceptable quality thereby providing reasonable assurance that contaminants which may be introduced to the condenser during normal and abnormal plant operation will be removed. Also demonstrate that resin transfer, cleaning and regeneraion are pushbutton initiated, fully automatic processes that clean and regenerate for reuse. Demonstrate valving and controls are such that a ready standby unit can be placed in service, or any operating unit can be taken out of service from the local control panels.

Prerequisites - Consruction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Component technical procedures, component calibrations have been completed satisfactorily.

Test Method - The system will be tested while processing water at 100% rated flow and at 120% rated flow, verifying that monitored influent and effluent parameters do not exceed design values. Resin capacity will be tested (one bed minimum) by processing the design quantity of water and verifying that monitored effluent paramenters do not exceed design values prior to achieving the design output. Control functions related to all modes of operation shall be demonstrated. Flow paths will be verified under actual operation as will all valve operations, motor-driven equipment performance, demonstration of all monitoring control and support equipment while processing dirty, exhausted resin charges exposed to condensate flow, through the regeneration modes, returning the resin charge to inservice processing condensate to design quality effluent. Simulation of functions will be used where off-normal conditions cannot be established or redundant testing of the same function under actual conditions serves no purpose.

Acceptance Criteria - Each vessel passing rated flow will produce water quality at design spec or better. Each vessel is capable of passing 120% rated flow for a short period of time. The condensate demineralizer and regeneration systems are pushbutton initiated, automatically controlled from a local control panel for all modes of operation. An automatically controlled isolation valve protects the resin transfer system from condensate system pressure. A proper concentration of acid solution is supplied to regenerate the cation resins and the proper concentrations of caustic solution at the proper temperature is supplied to regenerate the anion resins.

A40.1 LUBE OIL TRANSFER, STORAGE, & PURIFICATION SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate the ability of the system to transfer lube oil from one lube oil reservoir to another at rated flowrates and to demonstrate proper operation of the controls and the alarms of the lube oil centrifuge.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to ISG. Required instruments are calibrated and the controls are operable. Demineralized water transfer system and Instrument air system are operational. Required electrical supply systems are available and lube oil is available in sufficient quantity.

Test Method - The lube oil transfer pump performance parameters are measured and recorded. The batch oil tank pump performance parameters are measured and recorded. The centrifuge and oil heaters control and alarm circuits are tested and the operating parameters are measured and recorded. All flowpaths are then verified.

Acceptance Criteria - The system performance is in accordance with the applicable design documents.

A41.1 MAIN COOLING TOWER AND AUXILIARIES ACCEPTANCE TEST

Test Objectives - To demonstrate the proper operation of the cooling tower, cooling tower makeup and level control, chlorination system, sulfuric acid addition system, and the blowdown treatment system.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG.

Required instruments are calibrated and controls are operable. Required electrical power supply systems, instrument air system, plant makeup water system, and chlorination building H&V are available.

Test Method - Sliding gate valves and bypass valve operation is verified. Makeup system is verified to keep basin water level at the proper level. Chlorination addition capabilities are verified, and the acid system is verified to control pH at the proper value. The blowdown treatment system will remove enough chlorine to allow the plant to meet the requirements of its environmental discharge permit.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A42.1 CIRCULATING WATER SYSTEM ACCEPTANCE TEST

Test Objectives - To demonstrate proper operation of the Circulating Water System.

Prerequisites - Construction is complete to the extent necessary to run this test and the system is turned over to the ISG.

Required instruments are calibrated and controls are operable. Required electrical power supply systems and the cooling tower system are available.

Test Method - Pump protective interlocks and system design pressures and flows are verified.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A43.1 MAIN CONDENSER AIR REMOVAL SYSTEM ACCEPTANCE TEST

Test Objectives - The objectives of this test are as follows:

1) To demonstrate the ability of the mechanical vacuum pump to pull a vacuum on the condenser.

2) To demonstrate the ability of the SJAE's to maintain condenser vacuum when pump is tripped.

3) To demonstrate system ability to remove noncondensible gases from the main condenser and discharge them to the off-gas system.

4) To condense any steam removed from the condenser with the noncondensible gases and return the condensate to the condenser.

Prerequisites - The prerequisites for this test are as follows:

1) Construction is complete to the extent necessary to perform this test and system is turned over to ISG.

2) The main turbine is on turning gear.

3) The aux. boiler is operational and the main turbine seals are established.

4) Instrument Air System is operational.

5) Turbine Bldg. H&V is operational.

6) The Condensate System is operational.

7) The Off-Gas System is operational.

8) The separator-silencer 1T-107 is filled to the proper level.

9) All steam lines are properly drained of condensate.

Test Method - A vacuum will be pulled on the condenser using the mechanical vacuum pump and it will be maintained using the SJAE's. Valve interlocks will be checked as will all automatic functions. Alarms will be verified as they are induced during normal system change or simulation.

Acceptance Criteria

1) The mechanical vacuum pump can pull a vacuum of 5 in. Hga in 95 min. on the main condenser.

2) The SJAE's can maintain the vacuum after the mechanical vacuum pump is shutdown.

3) Valve sequencing operates per design.

A44.1 CONDENSATE SYSTEM ACCEPTANCE TEST

Test Objectives - To demonstrate the following:

(1) The ability of the condensate pumps and their associated valves to function properly.

(2) The ability of the system to maintain minimum recirculation flow through each condensate pump.

(3) The ability of the Turbine Building Closed Cooling Water System to provide sufficient cooling flow for the condensate pump bearings.

(4) The ability of the Hotwell Load Control to maintain condenser at normal operating level.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Power and control voltage is available for the associated motors, valves and instruments. Required calibration and operation of instruments, protective devices and controls is verified. Motor bearing cooling and pump seal water and instrument air is available. Main condensers are cleaned and filled with water.

Test Method - The system operation is manually initiated by starting the condensate pumps and establishing flow through various paths. System logic, interlocks and alarms are verified to be in accordance with design intent and system flows, pressures are within engineering specifications under various simulated operating conditions.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents for the conditions simulated during the test.

A46.1 EXTRACTION STEAM SYSTEM ACCEPTANCE TEST

Test Objectives - The general objective of this test is to demonstrate proper operation of the Extraction Steam System and Feedwater Heaters - Drains and Vents System. Specific objectives are to demonstrate the following:

1) The isolation valves in the Extraction Steam Sytem, the Feedwater Heater Drain System, and the Feedwater Heater Vents operate as required by their design.

2) All associated systems that drain to the feedwater heater systems isolate when required by the Feedwater Heater System design.

3) The alarms function to provide indication of an abnormality in the system.

Prerequisites - Construction is completed to the necessary extent and the system is turned over to ISG. Required instrumentation is calibrated and controls are operable. Required electrical power supply systems are available. Plant demineralized water and instrument air is available.

Test Method - Extraction Steam and Feedwater Heater System tests are simualted and performed with no steam present to the turbine. All system interlocks are tested.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A65.1 RADWASTE BUILDING AIR FLOW SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate the ability of the Radwaste Building Air Flow System to provide an adequate amount of filtered air to the Radwaste Building and to exhaust an adequate amount of air from the Radwaste Building.

Prerequisites - Construction is complete to the extent necessary for the test and the system is turned over to the ISG. 480 V power and instrument air are available. The required instruments are calibrated and controls are operable.

Test Method - The system is put into operation manually. Proper operation of all interlocks between system components is verified. The system air balance report and filter test reports are reviewed to ensure conformance with design specifications.

Acceptance Criteria - The system performs in accordance with design documents.

A65.2 RADWASTE BUILDING CHILLED WATER SYSTEM ACCEPTANCE TEST

Test Objectives - To demonstrate the ability of the Radwaste Building Chilled Water System to provide an adequate amount of chilled water to the Radwaste Building Air Supply System cooling coils.

Prerequisites - Construction is complete to the extent necessary for performing this test and the system is turned over to the ISG. 480V power and instrument air are available. The Makeup Demineralized Water System is available to provide makeup water as needed. The required instruments are calibrated and controls are operable.

Test Method - The system is placed in operation. Proper operation of all interlocks between system components is verified. All safety switches on both chillers are tested to ensure that they will shut down the associated unit when necessary. The system flow balance report is reviewed to verify that flowrates are within design specifications.

Acceptance Criteria - The system performs in accordance with design documents.

A68.1 RADWASTE SOLIDS HANDLING SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate the capability of the Radwaste Solids Handling System to control, collect, handle, process, package, solidify and temporarily store the wet waste sludges, spent resins and evaporator concentrates.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available.

Test Method - System operation is initiated manually. Required controls are operated and process is varied to verify interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A68.2 SPENT RESIN HANDLING SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate the capability of the spent resin collection system to control, collect, handle and discharge spent resin to the liquid radwaste filters.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available.

Test Method - System operation is initiated manually. Required controls are operated and process is varied to verify interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A69.2 LIQUID RADWASTE SUBSYSTEMS ACCEPTANCE TEST

Test Objective - To demonstrate the capability of the subsystems to collect, process, store and monitor for reuse or disposal all potentially radioactive liquid waste.

Prerequisites - Construction is complete to the extent necessary to perform this test and the subsystems are turned over to the ISG. Required instruments are calibrated and controls are operable. Required Electrical Power Supply Systems are available. Liquid radwaste subsystem storage tanks and sample tanks are available to be filled with water.

Test Method - Subsystem pumps are operated and performance characteristics are determined. Level controls are operated to verify alarms, pump starts and pump shutoffs. Performance of the liquid radwaste filtration, demineralization, chemical waste neutralization, chemical radwaste evaporation system, laundry radwaste filtration and effluent isolation is determined to the extent possible during this test.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A71.1 GASEOUS RADWASTE RECOMBINER CCW ACCEPTANCE TEST

Test Objective - To demonstrate the proper operation of the GRRCCW system, specifically, that the cooling pumps supply the rated flow to the system, the cooling water is temperature controlled, and the chemical addition tank has flow capabilities for adding chemicals to the system.

Prerequisites - Construction is completed to the extent necessary to perform this test and the system is turned over to the ISG. Required electrical power supply systems are available. Required instruments are calibrated and controls are available. The instrument air system is available. The service water system is operational and lined up to the GRRCCW heat exchangers.

Test Method - The system operation is initiated manually, and where applicable automatically. The system is operated in the system design modes and GRRCCW pumps performance is determined. Required controls are operated or simulated to verify automatic system functions and alarms.

Acceptance Criteria - The Unit One (1) and Common cooling water flow through the heat exchangers is temperature controlled through a range of 90° to 120°F. The Unit One (1) and common cooling water pumps deliver 1124 gpm to the respective system. Chemicals can be added to the system when flow is established through the Unit One (1) and common chemical addition tanks.

A72.1 OFF GAS RECOMBINER SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate the operation of the Off-Gas Recombiner System, specifically, that the system will operate in the standby, pre-start and process modes and that the standby recombiner can be brought on line within 10 minutes.

Prerequisites - Construction is completed to the extent necessary to perform this test and the system is turned over to the ISG. Required electrical power supply systems are available. Required instruments are calibrated and controls are avialable. The instrument air system is operational. The following systems are operational as needed: Condensate system, GRRCCW System, RBCCW, Auxiliary Boiler, and Main Condenser.

Test Method - The system operation is initiated manually, and, where applicable, automatically. The system is operated in the system design modes, required controls are operated or simulated to verify automatic system functions and alarms.

Acceptance Criteria - The Unit I and common Off-Gas Recombiner Systems perform the following:

1) The Off Gas Recombiner System will operate in the Standby, Prestart and Process modes.

2) A standby recombiner can maintain recombiner temperature close to 300°F and can be brought on line in 10 minutes.

3) The Off Gas Recombiners can be transfered and shut down locally and from the main control room.

4) The Charcoal Absorber subtrains are capable of being transfered and isolated locally and from the main control room.

A74.1 NITROGEN STORAGE AND SUPPLY SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate the capability of the Nitrogen Storage and Supply to provide and control the supply of nitrogen gas for primary containment purging and to maintain an inert atmosphere in containment.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical power supply systems are available.

Test Method - System operation is initiated manually. The system is operated in the different design modes and system performance is determined. Required controls are operated or simulated signals are applied to verify automatic features, system interlocks and alarms.

Acceptance Criteria - The system performance parameters are in accordance with applicable design documents.

A76.2 PROCESS SAMPLING SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate proper operation of the Process Sampling System. This is performed by proving:

1) The operability of the reactor and turbine building thermal baths.

2) The ability of the chemical fume hood to control out-leakage when drawing grab samples.

3) The ability of the system to provide required monitoring of sample fluids.

4) Capability of obtaining grab samples.

Prerequisites - Construction is complete to the extent necessary to peform this test and the system is turned over to ISG. Required instrumentation is calibrated and controls are operable. Required electrical power supply systems are available. Plant demineralized water is available. Turbine Bldg. and Reactor Bldg. closed cooling water is available.

Test Method - Tests whenever feasible will be performed when the process being sampled is in operation. Other tests, such as main steam samples, will be simulated. All sampling devices will be calibrated and alarm conditions set.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A84.1 MOISTURE SEPARATORS ACCEPTANCE TEST

Test Objective - To demonstrate the ability of the moisture separator drain tank level controls to maintain level and provide a main turbine trip signal as a result of high level.

PREREQUISITES - Construction is complete to the extent necessary to perform this test and the system is turned over to ISG. Hydrostatic testing, velocity flushing and air blowing have been completed. Required instruments are calibrated and controls are operable. Required electrical power supplies, water supplies and instrument air are available. The associated plant systems which are capable of receiving water are available to the extent necessary to peform this test.

TEST METHOD - The water level in the drain tank will actually be varied and the proper operation of the level controls, level alarms and level trips will be verified.

ACCEPTANCE CRITERIA - The system performance parameters are in accordance with the applicable design documents. All automatic trips and alarms actuate within their allowable limits.

A85.2 FREEZE PROTECTION SYSTEM ACCEPTANCE TEST

Test Objective - To demonstrate the ability of the system to supply and interrupt power to the individual heater circuits at the correct voltage and current in both the AUTO and MANUAL modes of operation and to demonstrate the sytem's ability to detect a loss of source supply voltage on a faulty heater circuit.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to ISG. The required instruments are calibrated and the controls are operable.

Test Method - Each control panel is energized and proper source supply voltage verified. The required controls will be operated and signals simulated as necessary to verify the individual heater circuits function per design in the AUTO, OFF, and Manual modes, and are providing the design specified heat requirements for the applications.

ACCEPTANCE CRITERIA - The system performance parameters are in accordance with the applicable design documents, technical spec's. and vendor prints.

A91.1 ANNUNCIATOR SYSTEM ACCEPTANCE TEST

Test Objective - The objective of this test is to demonstrate the ability of the main control room annunciators to provide audible and visual indication of an alarm condition.

Prerequisites - Construction turnover of the system is complete to the extent required to conduct this test. The system has been walked through, verified complete and the component technical tests have been completed.

Test Method - Simulated alarms are applied and the audible and visual indication verified. Annunciator loss of power and ground detection feature are also tested, where applicable.

Acceptance Criteria - The system performance parameters are in accordance with applicable engineering design documents.

A92.1 TURBINE STEAM SEALS & DRAINS ACCEPTANCE TEST

Test Objective - The objective of this test is to demonstrate the proper operation of the turbine steam seal system and drains using the auxiliary boiler steam supply to the turbine steam seal header. Also, the test will demonstrate the ability of the steam packing exhauster to maintain a proper vacuum on the steam seal exhaust header.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. Required electrical supply systems are available. The instrument air system is operational. The auxiliary boilers are available and in the standby mode. The condensate system is operational. The main turbine and feedwater turbines are available to be placed on turning gear. The main condensers are lined up to receive drains and to provide support to seal the main and reactor feed pump turbines.

Test Method - The auxiliary boilers will provide a continuous and regulated supply of steam to the steam seal evaporator header. The performance of the steam packing exhauster to maintain a proper vacuum on the exhaust header is verified. Simulated and automatic signals are applied to verify system interlocks and alarms for the seal steam evaporator drain tank, seal steam system and steam packing exhauster.

Acceptance Criteria - The steam packing exhauster will maintain an approximate vacuum of 5.0 inches H O on the seal steam evaporator exhaust header during normal operating conditions. The auxiliary steam system can provide a continuous amount of clean steam to the seal steam evaporator header at approximately 4 psig to supply the following with sealing steam: the main turbine shaft seals, the stem packings of the main steam stop valves, control valves, and bypass valves, the combined intermediate valves, the shaft seals of the reactor feed pump turbines, and the stem packings of the reactor feed pump turbine stop and control valves.

A93.1 TURBINE LUBE OIL SYSTEM ACCEPTANCE TEST

Test Objectives - To demonstrate the proper operation of the Turbine Lube Oil System.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls operable. Required electrical power supply systems are available. The Service Water System and the Main Turbine-Generator Assembly is available.

Test Method - System operation is initiated manually and automatically testing all trips and interlocks. The main reservoir vapor extractor is tested manually and automatically to verify proper vacuum in the main reservoir and isolation on detection of fire. All main lube oil pumps are tested for proper manual and automatic start to verify proper bearing oil supply pressures during all conditions including loss of AC power. Bearing lift pumps are tested manually and automatically to verify proper bearing lift for turning gear operation. The main turbine turning gear is tested for both manual and auto engaging and starting to ensure proper rotation during shaft cooldown.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A93.2 TURBINE VALVES, VALVE TEST, EHC AND SUPERVISORY SYSTEMS ACCEPTANCE TEST

Test Objectives - To demonstrate the proper operation of the turbine EHC and supervisory system.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls operable. Required electrical power supply systems are available. The Main Condenser, Stator Cooling and Instrument Air Systems are available.

Test Method - Hydraulic System Manual and Automatic Modes are tested. All turbine trip paths are verified. All system stop, control and bypass valves are tested for EHC operation. Turbine warm-up, speed select, and load ramp functions are verified. Turbine steam lead drain valves are tested for proper operation.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A98.1 MAIN GENERATOR AND EXCITATION SYSTEM ACCEPTANCE TEST

Test Objectives - To demonstrate the ability of the protective relays and their associated interlocks to shutdown the generator.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Component calibrations and alarm verifications are complete to the extent necessary to perform this test.

Test Method - Through the use of jumpers, lifted leads, pulled fuses, and manual manipulation of relay contacts conditions are simulated to initiate automatic responses of the generator protection circuitry. Proper operation of the generator protection circuitry is verified.

Acceptance Criteria - The following is verified:

(1) The ability of the voltage regulator to transfer from auto to manual upon initiation of design events.

(2) The ability of the exciter field breaker to function according to design basis events.

(3) The ability of the primary and backup lockout relays to trip the generator upon initiation of design basis events.

A99.2 Communications System Acceptance Test

Test Objective - To demonstrate the ability of the three part communications system (PA, Plant Maint./Test Jack, and Plant Evacuation and Alarm Systems) components to function as an integrated system. The PA system to provide communications and a medium for transmitting plant alarms in conjunction with the Plant Evacuation Alarm System. The Plant Evacuation Systems ability to generate the necessary tones and frequencies and the Plant Maint./Test Jack Systems ability to provide an additional independent means of communication.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to ISG. The required instruments are calibrated and the controls are operable.

Test Method - By operating the required controls each Public Address station will be tested in the transmit and receive modes on all channels. The associated speakers will be tested for functional audibility. The systems loop separation and muting features will be operationally verified.

The Plant Maint./Test Jack System will be tested by operating the required controls and verifying each Jack Stations transmit/receive capability on all of the systems 23 channels. An integrated test with several remote Jack Stations attached will also be performed.

The Plant Evacuation and Alarm System will be used in conjunction with the PA system to broadcast all 5 of the possible tones and frequencies generated by the system. The siren will be operationally tested. Also the systems isolation and silencing features will be operationally verified.

Acceptance Criteria - The systems performance is in accordance with the applicable design documents.

A99.6 Seismographical Monitoring System Acceptance Test

Test Objective - To verify the operability of the seismic monitoring instrumentation (digital cassette accelerographs, playback unit, response spectrum analyzer and triaxial accelerometers) and to demonstrate proper integrated response of the system to activate upon occurance of a seismic event as designed.

Prerequisites - Construction is complete to the extent necessary to perform this test and the system is turned over to the ISG. Required instruments are calibrated and controls are operable. The required electrical power supply system is available. All recorders have ample paper and all accelerographs are loaded with the proper magnetic tape cassettes.

Test Method - Both an internal calibration feature on the SMR-102 (seismic monitoring recorder) and a simulated seismic event at each triaxial accelerometer are used as "trigger input" to the seismic monitoring system to verify automatic initiation and alarm actuations. Playback (production of time-history seismic graphs) is demonstrated by manual transfer of cassette tapes from the digital cassette accelerographs to the seismic monitoring recorder.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

control and bypass valves are tested for EHC operation. Turbine warm-up, speed select, and load ramp functions are verified. Turbine steam lead drain valves are tested for proper operation.

Acceptance Criteria - The system performance parameters are in accordance with the applicable design documents.

A98.1 MAIN GENERATOR AND EXCITATION SYSTEM ACCEPTANCE TEST

Test Objectives - To demonstrate the ability of the protective relays and their associated interlocks to shutdown the generator.

Prerequisites - Contstruction is complete to the extent necessary
20 to perform this test and the system is turned over to the ISG. Component calibrations and alarm verifications are complete to the extent necessary to perform this test.

Test Method - Through the use of jumpers, lifted leads, pulled fuses, and manual manipulation of relay contacts conditions are simulated to initiate automatic responses of the generator protection circuitry. Proper operation of the generator protection circuitry is verified.

Acceptance Criteria - The following is verified:

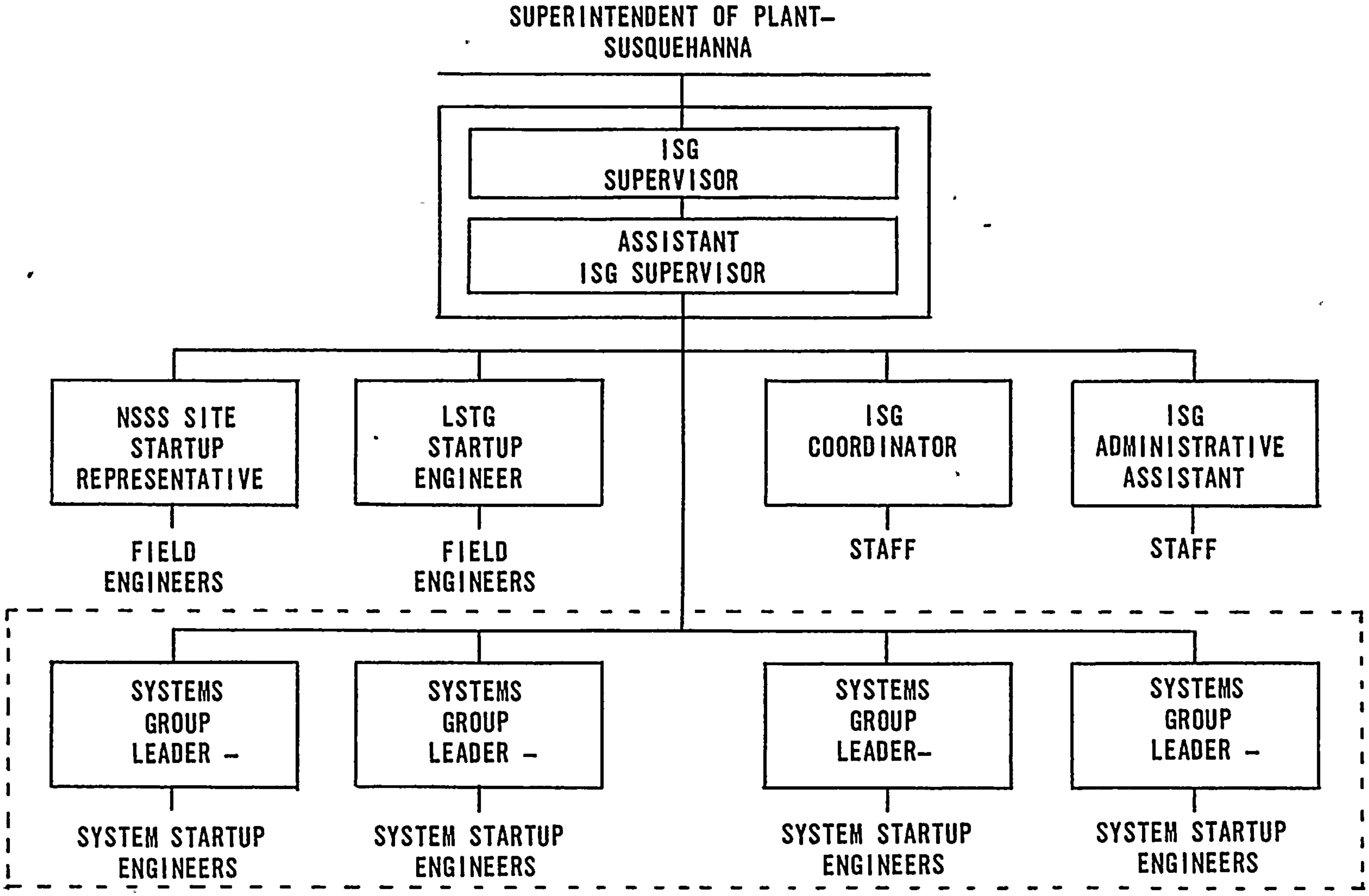
(1) The ability of the voltage regulator to transfer from auto to manual upon initiation of design events.

(2) The ability of the exciter field breaker to function according to design basis events.

(3) The ability of the primary and backup lockout relays to trip the generator upon initiation of design basis events.

A99.2 Communications System Acceptance Test

Test Objective - To demonstrate the ability of the three part communications system (PA, Plant Maint./Test Jack, and Plant Evacuation and Alarm Systems) components to function as an integrated system. The PA system to provide communications and a
20 medium for transmitting plant alarms in conjunction with the Plant Evacuation Alarm System. The Plant Evacuation Systems ability to generate the necessary tones and frequencies and the Plant Maint./Test Jack Systems ability to provide an additional independent means of communication.



SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

INTEGRATED STARTUP GROUP
ORGANIZATION

FIGURE 14.2–1

4. The credit taken for the functioning of normally operating plant systems.

5. The operation of engineered safety systems that is required.

6. The effect of a single failure or an operator error on the event.

15.0.3.2.1 Single Failures or Operator Errors

15.0.3.2.1.1 General

This paragraph discusses a very important concept pertaining to the application of single failures and operator errors to analyses of the postulated events. Single active component failure (SACF) criteria have been required and successfully applied on past NRC approved docket applications to design basis accident categories only. Reference 15.0-1 infers that a "single failures and Operator errors" requirement should be applied to transient events (both high, moderate, and low probability occurrences) as well as accident (very low probability) situations.

Transient evaluations have been judged against a criteria of one single equipment failure "or" one single operator error as the initiating event with no additional single failure assumptions to the protective sequences although a great majority of these protective sequences utilized safety systems which can accommodate SACF aspects. Even under these postulated events, the plant damage allowances or limits were very much the same as those for normal operation.

Reference 15.0-1 suggests that the transient and accident scenarios should now include "and" (multi-failure) event sequences. The format request follows:

For initiating occurrence	1)	an equipment failure or an operator error, and
For single equipment failure or operator error analysis	2)	another equipment failure or failures and/or another operator error or errors.

This is considered a new requirement and the impact will need to be completely evaluated. While this is under consideration GE has evaluated and presented the transients and accidents in this chapter in the above new requirement manner.

Event categorization relative to transient and accident analysis is discussed here. If the evaluation is done per the new multi-failure methods, the event frequency categories should be modified.

The original categorization of events was based on frequency of the initiating event alone and thus the allowance or limit was accordingly established based on that high frequency level. With the introduction of additional assumptions and conditions (initial event and SCF and/or SOE), the total event would not fall into a lower frequency/probability category. Thus, less restrictive limits or allowances should be applied in the analysis of transients and accidents. This needs to be considered and evaluated.

GE has evaluated and presented the transients and accidents in this chapter by the more restrictive old allowances and limits of the event categorization presently in effect.

Most events postulated for consideration are already the results of single equipment failures or single operator errors that have been postulated during any normal or planned mode of plant operations. The types of operational single failures and operators errors considered as initiating events and subsequent protective sequence challenges are identified in the following paragraphs:

15.0.3.2.1.2 Initiating Event Analysis

1. The undesired opening or closing of any single valve (a check valve is not assumed to close against normal flow)

   or

2. The undesired starting or stopping of any single component

   or

3. The malfunction or maloperation of any single control device

   or

4. Any single electrical component failure

   or

5. Any single operator error.

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18.1.3.3 Statement of Response

The facility staffing requirements are presented in Subsection 6.2.2 of the Technical Specifications. These requirements are consistent with those given in Table 18.1-1.

The plant policy on operations personnel working hours is discussed in administrative procedure AD-QA-300, "Conduct of Operations," and is specifically defined in procedure AD-QA-302, "Operations Administrative Controlls."

18.1.4 IMMEDIATE UPGRADING OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS (I.A.2.1)

18.1.4.1 Statement of Requirement

Applicants* for senior operator licenses shall have 4 years of responsible power plant experience. Responsible power plant experience should be that obtained as a control room operator (fossil or nuclear) or as a power plant staff engineer involved in the day-to-day activities of the facility, commencing with the final year of construction. A maximum of 2 years power plant experience may be fulfilled by academic or related technical training, on a one-for-one time basis. Two years shall be nuclear power plant experience. At least 6 months of the nuclear power plant experience shall be at the plant for which he seeks a license. Effective date: Applications received on or after May 1, 1980.

Applicants for senior operator licenses shall have held an operator's license for 1 year. Effective Date: Applications received after December 1, 1980. The NRC has not imposed the 1-year experience requirement on cold applicants for SRO licenses. Cold applicants are to work on a facility not yet in operation; their training programs are designed to supply the equivalent of the experience not available to them.

Senior operator*: Applicants shall have 3 months of shift training as an extra man on shift.

Control room operator*: Applicants shall have 3 months training on shift as an extra person in the control room. Effective date: Applications received after August 1, 1980.

*Precritical applicants will be required to meet unique qualifications designed to accommodate the fact that their facility has not yet been in operation.

Training programs shall be modified, as necessary, to provide:

1) Training in heat transfer, fluid flow and thermodynamics.

2) Training in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged.

3) Increased emphasis on reactor and plant transients. Effective date: Present programs have been modified in response to Bulletins and Orders. Revised programs should be submitted for OLB review by August 1, 1980.

Content of the licensed operator requalification programs shall be modified to include instruction in heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core. Effective date: May 1, 1980.

The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license; 80% overall and 70% each category. Effective date: Concurrent with the next facility administered annual requalification examination after the issue date of this requirement.

Programs should be modified to require the control manipulations listed in Enclosure 4 of NUREG 0737, item I.A.2.1. Normal control manipulations, such as plant reactor startups, must be performed. Control manipulations during abnormal or emergency operations must be walked through with, and evaluated by, a member of the training staff at a minimum. An appropriate simulator may be used to satisfy the requirements for control manipulations. Effective date: Programs modified by August 1, 1980. Renewal applications received after November 1, 1980 must reflect compliance with the program.

Certifications completed pursuant to Sections 55.10(a)(6) and 55.33a(4) and (5) of 10 CFR Part 55 shall be signed by the highest level of corporate management for plant operation (for example, Vice President for Operations). Effective date: Applications received on or after May 1, 1980.

18.1.4.2 Interpretation

None required.

18.1.4.3 Statement of Response

A program is established to assure that all reactor operator and senior reactor operator license candidates (beyond the initial compliment required to startup Units 1 & 2) have the prescribed experience, qualifications, and training. Candidates will be prepared and certified in accordance with Nuclear Department Instruction NDI-QA-4.2.1. Administrative procedure AD-00-103, "Selection Process for Operations Personnel," details the process by which the qualifications of candidates for operations positions will be evaluated in the future.

The initial startup crews will have completed extensive training devised in part to recognize the non-operational status of the units. This program includes real time training on the Susquehanna SES simulator which duplicates the actual unit and thus in many respects equates to the experience requirements. Subsection 13.1.3 describes the qualifications commitments for the existing plant staff.

18.1.5 ADMINISTRATION OF TRAINING PROGRAMS (I.A.2.3)

18.1.5.1 Statement of Requirement

Pending accreditation of training institutions, licensees and applicants for operating licenses will assure that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses demonstrate senior reactor operator (SRO) qualifications and be enrolled in appropriate requalification programs.

Training center and facility instructors who teach systems, integrated responses, transient and simulator courses shall demonstrate their competence to NRC by successful completion of a senior operator examination. Effective date: Applications should be submitted no later than August 1, 1980 for individuals who do not already hold a senior operator license.

Instructors shall be enrolled in appropriate requalification programs to assure they are cognizant of current operating history, problems, and changes to procedures and administrative limitations. Effective date: Programs should be initiated May 1, 1980. Programs should be submitted to OLB for review by August 1, 1980.

18.1.5.2 Interpretation

The "instructors" referenced in this requirement are those individuals who teach systems specific to BWRs, integrated responses, transients, and simulator courses to licensed operators or license candidates.

18.1.5.3 Statement of Response

Certification of instructors is described in Nuclear Department Instruction NDI-QA-4.1.4. This procedure delineates which instructors are required to pass the examination for senior reactor operators (SRO). All instructors who teach materials identified in Subsection 18.1.5.2 are certified as SROs.

18.1.6 REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS (I.A.3.1)

18.1.6.1 Statement of Requirement

A new category shall be added to the operator written examination entitled, "Principles of Heat Transfer and Fluid Mechanics."

A new category shall be added to the senior operator written examination entitled, "Theory of Fluids and Thermodynamics."

Time limits shall be imposed for completion of the written examinations:

1. Operator: 9 hours.

2. Senior Operator: 7 hours.

The passing grade for the written examination shall be 80% overall and 70% in each category.

All applicants for senior operator licenses shall be required to be administered an operating test as well as the written examination. Effective date: Examinations administered on or after May 1, 1980.

Applicants will grant permission to NRC to inform their facility management regarding the results of the examinations for purposes of enrollment in requalification programs. Applications received on or after May 1, 1980.

not mitigated, could result in conditions of inadequate core cooling.

(f) Description of indications available to the BWR operator for the detection of adequate core cooling

(g) Description and justification of analysis methods for extremely degraded cases.

(2) NEDO 24934, "BWR Emergency Procedure Guidelines BWR 1-6," Revision 1, January 1981.

Guidelines for BWR Emergency Procedures based on identification and response to plant symptoms; including a range of equipment failures and operator errors; including severe multiple equipment failures and operator errors which, if not mitigated, would result in conditions of inadequate core cooling; including conditions when core cooling status is uncertain or unknown.

B. Adequacy of Submittals

The submittals described in paragraph A have been discussed and reviewed extensively among the BWR Owners' Group, the General Electric Company, and the NRC staff. The NRC staff has found (NUREG-0737, page I.C.1-3) that "the analysis and guidelines submitted by the General Electric Company (GE) Owners' Group...comply with the requirements (of the NUREG-0737 clarification)." In Reference 1, the Director of the Division of Licensing states, "we find the Emergency Procedure Guidelines acceptable for trial implementation (on six plants with applications for operating licenses pending)."

PP&L believes that in view of these findings, no further detailed justification of the analyses or guidelines is necessary at this time. Reference 1 further states, "(during the course of implementation we may identify areas that require modification or further analysis and justification." The enclosure to Reference 1 identifies several such areas. PP&L will work with the BWR Owners' Group in responding to such requests.

By our commitment to work with the Owners' Group on such requests, on schedules mutually agreed to by the NRC and the Owners' Group, and by reference to the BWR Owners' Group analyses and guidelines already submitted, our response to the NUREG-0737 requirement "for reanalyses of transients and accidents and inadequate core cooling and preparation of guidelines for development of emergency procedures" by January 1, 1981, is complete.

Emergency procedures based on those guidelines have been developed and are currently in trial use on the Susquehanna SES Simulator. These procedures have been reviewed by the NRC. Final versions which incorporated NRC comments were submitted in a letter from N. W. Curtis to B. J. Youngblood on May 15, 1981(PLA-791).

18.1.9 SHIFT RELIEF AND TURNOVER PROCEDURES (I.C.2)

No requirement stated in NUREG 0737. Refer to Subsection 18.2.8 which contains the response to the requirement in NUREG 0694.

18.1.10 SHIFT SUPERVISOR RESPONSIBILITY (I.C.3)

No requirement stated in NUREG 0737. Refer to Subsection 18.2.9 which contains the response to the requirement in NUREG 0694.

18.1.11 CONTROL ROOM ACCESS (I.C.4)

No requirement stated in NUREG 0737. Refer to Subsection 18.2.10 which contains the response to the requirement in NUREG 0694.

18.1.12 FEEDBACK OF OPERATING EXPERIENCE (I.C.5)

18.1.12.1 Statement of Requirement

Applicants for an operating license shall prepare procedures to assure that information pertinent to plant safety originating inside or outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

(1) Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

(2) Identify the administrative and technical review steps necessary in translating recommendations by the

18.1.21.3.2.1 Sample Points

a) Wetwell and Drywell Atmosphere

Provision will be made to obtain gas samples from two separate areas in both the drywell and wetwell. The sample lines will tap into the containment air monitoring system (CAMS) sample lines outside of primary containment and after the second containment isolation valve. The two drywell sample taps are on the highpoint line, sampling at elevation 790', and the midpoint line, sampling at elevation 750'.

b) Secondary Containment Atmosphere

A sample line will be installed to allow sampling of the secondary containment atmosphere. The location of this point has yet to be determined. This sample point would be useful in determining the post accident accessibility of the reactor building.

c) Reactor Coolant and Suppression Pool Liquid Samples.

When the reactor is pressurized reactor coolant samples will be obtained from a tap off the jet pump pressure instrument system. The sample point will be on a non-calibrated jet pump instrument line outside of primary containment and after the excess flow check valve. This sample point location is preferred over the normal reactor sample points on the reactor water clean up system inlet line and recirculation line since the reactor clean-up system is expected to remain isolated under accident conditions, and it is possible that the recirculation line containing the sample line may be secured. The jet pump instrument line has been determined to be the optimum sample point for accident conditions since: 1) the pressure taps are well protected from damage and debris, 2) if the recirculation pumps are secured, there is normally excellent circulation of the bulk of the coolant past these taps (natural circulation), and 3) the taps are located sufficiently low to permit sampling at a reactor water level which is even below the lower core support plate.

A single sample line is also connected to both loops in the RHR system. The sample lines will tap off the high pressure switch instrument lines coming off the common section of the RHR system return line. This sample point provides a means of obtaining a reactor coolant sample when the reactor is not pressurized and at least one of the RHR loops is operated in the shutdown cooling mode. Similarly, a suppression pool sample can be obtained from an RHR loop lined up in the suppression pool cooling mode.

18.1.21.3.2.2 Isolation Valves and Sample Lines

Containment isolation for the drywell and wetwell gas sample lines is provided by the existing CAMS sample line isolation valves. The jet pump instrument sample line containment isolation is provided by an existing isolation valve and excess flow check valve upstream of the sample tap. All gas sample lines from the sample taps to and including the first flow control valves are seismic category 1 except for the secondary containment sample line which has no control valve before it enters the sample panel. The sample lines from the RHR system are seismic category 1 through both system isolation valves and a flow restricting orifice. The sample line from the jet pump instrument system is seismic category 1 to the flow control/isolation valve. All containment isolation valves upstream of the sample taps can be overridden from the control room. All isolation and control valves shown in Figure 18.1-11 which are within the Q boundary are controlled by a single permissive switch in the control room and individually controlled at the sampling control panel located adjacent to the sample station.

The solenoid isolation and control valves which are part of the post accident sample system to the Q boundary will be environmentally qualified. The gas sample lines are heat traced to prevent precipitation of moisture and the resultant loss of iodine in the sample lines.

18.1.21.3.2.3 Piping Station

The piping station, which is to be installed within the reactor building, includes sample coolers and control valves which determine the liquid sample flow path to the sample station. The location for the piping station is shown in Figure 18.1-12. Cooling water will come from the Reactor Building Closed Cooling Water System.

18.1.21.3.2.4 Sample Station and Control Panels

The location of the sample station, control panels and associated equipment is shown in Figure 18.1-13. The sample station consists of a wall mounted frame and enclosures. Included within the sample station are equipment trays which contain modularized liquid and gas samplers. The lower liquid sample portion of the sample station is shielded with 6 inches of lead brick, whereas the upper gas sampler has 2 inches of lead shielding. The control instrumentation is installed in two control panels. One

18.1.23 RELIEF AND SAFETY VALVE TEST REQUIREMENTS (II.D.1)

18.1.23.1 Statement of Requirement

Boiling-water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents.

Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualificaiton shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.

Preimplementation review will be based on EPRI, BWR, and applicant submittals with regard to the various test programs. These submittals should be made on a timely basis as noted below, to allow for adequate review and to ensure that the following valve qualification date can be met:

Final BWR Test Program--October 1, 1980

Postimplementation review will be based on the applicants' plant-specific submittals for qualification of safety relief valves. To properly evaluate these plant-specific applications, the test data and results of the various programs will also be required by the following dates:

BWR Generic Test Program Results--July 1, 1981
Plant-specific submittals confirming adequacy of safety and relief valves based on licensee/applicant preliminary review of generic test program results--July 1, 1981
Plant-specific reports for safety and relief valve qualification--October 1, 1981
Plant-specific submittals for piping and support evaluations--January 1, 1982

18.1.23.2 Interpretation

None required.

18.1.23.3 Statement of Response

PP&L is participating in the BWR Owner's Group (BWROG) program to test safety/relief valves (SRVs). Wyle Laboratories in Huntsville, Alabama has been contracted to design and build a test facility. The design is complete and construction is well underway. The facility will be capable of high and low pressure valve tests.

Documentation of the BWROG testing program was sent to the NRC on September 17, 1980 by a letter from D.B. Waters to R.N. Vollmer. A summary of this document is provided below.

An engineering evaluation was done to identify the expected operating conditions for SRVs during design basis transients and accidents. This evaluation indicates the SRVs may be required to pass low pressure liquid as a result of the Alternate Shutdown Mode (described in Subsection 15.2.9). No other significantly probable event, even combined with a single active failure or single operator error, produces expected operating conditions that justify qualification of SRVs for extreme operating conditions. Therefore a test program was developed to demonstrate the SRVs' capabilities as may be necessary during the Alternate Shutdown Mode.

The test results were submitted by a letter to A. Schwencer from N. W. Curtis on July 1, 1981 (PLA-865). A plant specific SRV qualification report was submitted to the NRC on October, 1981 (PLA-940). This report includes all necessary evaluations of piping and supports.

18.1.24 SAFETY/RELIEF VALVE POSITION INDICATION (II.D.3)

18.1.24.1 Statement of Requirement

Reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

The basic requirement is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

The valve position should be indicated in the control room. An alarm should be provided in conjunction with this indication.

18.1.29.2 Interpretations

From item 4, the opening of containment isolation valves must require a deliberate operator action.

From item 5, the containment isolation setpoint pressure should be optimized to prevent unnecessary isolations during normal operations. However, containment isolation must not be prevented or delayed during an accident.

18.1.29.3 Statement of Response

(1) Containment isolation signals are actuated by several sensed parameters (refer to Table 3.3.2-1 in the Technical Specifications). This complies with SRP Subsection 6.2.4, Paragraph II-6.

(2) Each process line penetrating containment was reviewed to determine whether it is an essential or non-essential line for purposes of isolation requirements. The classification for each line is given in Table 18.1-10.

Justification for the classification as an essential or non-essential line was also developed and is provided in Table 18.1-11. Systems identified as essential are those which may be required to perform an indispensable safety function in the event of an accident. Non-essential systems are those not required during or after an accident. Since instrument lines are not governed by isolation signals but are equipped with a manual isolation valve followed by an excess flow check valve outside the containment, the review of these lines was limited to ensure compatibility with the penetration listing in Table 6.2-12a.

(3) All lines to non-essential systems are provided with isolation capability. All isolation valves in these lines, except the reactor water clean-up system (RWCU) discharge valves (G33-1FC042 and 1F104 receive auto-isolation signals (refer to Table 18.1-10). The isolation function for the RWCU discharge lines is provided by three series check valves (141-1F010A,B, HV-14107A,B and G33-1F039A,B) which prevents back flow from the reactor vessel. The RWCU discharge isolation valves are not closed to prevent the loss of the filter cake in the RWCU filter demineralizer system and injection of resin into the vessel on restart of the RWCU system.

(4) All containment isolation valves, except those listed below, will not automatically open on logic reset. Some valves

require corrective action to comply with this requirement. All such actions will be completed prior to fuel load. Refer to a letter from N. W. Curtis to B. J. Youngblood on April 14, 1981 (PLA-715) for this commitment. An override of any isolation signal will not cause automatic opening of any isolation valve.

a) The following valves in the Liquid Radwaste, Reactor Water Sample, and Reactor Building Chilled Water systems are normally open valves and will close upon a containment isolation signal.

HV-16116	A1 & A2
HV-16108	A1 & A2
HV-18781	A1 & A2 & B1 & B2
HV-18782	A1 & A2 & B1 & B2
HV-18791	A1 & A2 & B1 & B2
HV-18792	A1 & A2 & B1 & B2
B31-1F019	
B31-1F020	

When the containment isolation logic is reset the above valves would have reopened. The logic for these valves will be modified by fuel load to ensure that they will not reopen on logic reset. Table 18.1-10 reflects the modified configuration of these valves.

b) The RCIC and HPCI turbine steam supply line isolation valves (HV-1F007, HV-1F008, HV-1F002 and HV-1F003) are normally open valves and will close upon a steam line break isolation signal. These valves are essential valves and do not receive a containment isolation signal. Reopening of these valves will occur if the hand switches are not placed in the closed position by the operator prior to actuation of the reset switch and the isolation parameters have cleared.

These valves are equipped with key-locked maintained contact switches to insure that these valves are open during ECCS initiation. If a pipe break condition were detected, then these valves will be automatically closed. After the pipe break problems are cleared these valves can be reopened to their normal emergency positions by deliberate operator action using the key-locked reset switches for each system. The operator is required to ensure that the valve switches are in the correct position prior to operating the keylock reset switch.

c) The inboard HPCI and RCIC isolation valves each have a pressure equalization valve (HV-1F100 and HV-1F088) around them. The equalization valves are normally

The system is powered from non-class IE instrument AC power. An independent battery backup is provided which is capable of providing power for 8 hours.

This equipment is installed and will be operational by fuel load.

18.1.30.3.2 Sampling and Analysis of Plant Effluents

Each of the five plant vents has a continuous isokinetic sample drawn from it in accordance with ANSI-N13.1. Each sample is then taken through short runs of heat traced tubing to a Eberline Model FAAM (Fixed Airborne Activity Monitor). In the FAAM the sample stream then passes through a HEPA filter which removes particulates. Upon leaving the HEPA filter the sample stream passes through a charcoal filter which removes iodines. When required this filter can be replaced with a silver zeolite filter. Capabilities for purging the sample line with compressed air are provided under manual control. The sample stream is next measured for noble gas activity and then returned to the plant vent. During normal operation the HEPA and charcoal filters are monitored by radiation detectors and this information is presented to the operator in the control room. Under accident conditions these detectors will saturate and the filters must be removed, placed in a shielded container, and analyzed in a laboratory. The FAAM also has provisions for obtaining a grab samples.

The isokinetic sample is in compliance with ANSI-N13.1-1969. To accomplish this, each vent has an air profile (final gas treatment) station to eliminate turbulent and rotating gas flow. The average stack velocity and volume are then measured by means of a multipoint, self-averaging Pitot transverse station. An air flow controller then simultaneously withdraws a multipoint sample under isokinetic flow conditions by means of an isokinetic sample rack. This isokinetic sample is then directed to the Final Airborne Activity Monitor.

The system is designed such that plant personnel can remove samples, replace sample media and transport the samples in shielded containers to an analysis facility. Radiation exposures for this process are not in excess of 3 rem whole-body exposure and 18.5 rem to the extremities during the duration of the accident.

Procedures for analyzing samples both normal and accident conditions are described in Subsection 12.5.3.5.5. The equipment used to analyze these samples is described in Subsection 1.2.5.2.7.1. Additional instrumentation and procedures for sampling and analyzing implant iodine are described in Subsection 18.1.70.

The installation plant vent sampling and monitoring system is complete.

18.1.30.3.3 Containment High-Range Radiation Monitor

Redundant Class 1E in-containment radiation monitors will be provided. The monitors will be Victoreen Model 875 high range radiation monitors. These monitors are capable of measuring radiation levels of 1R/hr to 1 x 10^7 R/hr (Gamma) for photon energies of between 60 KeV to 3 MeV. An accuracy of ± 20% is obtained on lower decades.

The detectors will be unshielded and physically separated on opposite sides of the reactor pressure vessel.

Logarithmic indicating recorders will be provided for Channels A and B on front row panel 1C601.

A common red high radiation annunciator for both channels will be provided on control room front row panel 1C601. A common white system trouble light will also be provided for both channels on control room front row panel 1C601.

The containment radiation monitoring system is designed to be safety grade. This equipment will be qualified to IEEE-344-1975, IEEE-323-1974 and NUREG-1588 in accordance with the Commission order on May 23rd, 1980 (CLI-20-81).

The installation of the containment radiation monitoring system will be complete by January 1982.

18.1.30.3.4 Containment Pressure Monitor

Two Class 1E redundant drywell chamber pressure measurements will be provided as follows:

SERVICE	RANGE
LOCA Range	0 to 65 psia
HI Range	0 to 250 psig

The LOCA and HI ranges are divided into two divisions. Continuous, individual indication of all four Division I and II pressure measurements will be provided by indicating recorders for the operation on front row panels 1C601.

Normal operating pressures in the drywell and wetwell are monitored by a -1 to +3 psig instrument installed in each chamber. An indicator on control panel 1C601 will display these pressures. A selector switch is provided to allow the operator to monitor either drywell or wetwell pressure. These instruments are non-safety grade with the exception of the transmitters, which are designed to meet containment pressure boundary service.

The accuracy of these instruments is ± 2% of full scale.

The containment accident range pressure monitors are designed to be safety grade. This equipment will be qualified to IEEE-344-1975, IEEE-323-1974 and NUREG 0588 in accordance with the Commission order on May 23rd, 1980 (CLI-20-81).

The containment pressure instrumentation will be installed by January 1982.

18.1.30.3.5 Containment Water Level Monitor

Redundant wide and narrow range safety grade instruments will be installed to continuously monitor suppression pool water level. The channel A measurements will be displayed on control room front row panel 1C601. The channel B measurements will be recorded on front row panel 1C601.

The narrow range instruments measure between 18 and 26 feet. The wide range instruments measure between 4.5 and 49 feet. This covers the required range of from the lowest ECCS suction to 5 feet above normal water level. Normal water level is approximately 23 feet.

The accuracy of these instruments is ±2% of full scale.

Installation of the suppression pool water level instrumentation will be complete by January 1982.

18.1.30.3.6 Containment Hydrogen Monitor

Continuous and redundant indication and recording of hydrogen will be provided on control room front row panel 1C601. These instruments will have a range of 0 to 30%.

The containment hydrogen monitoring system is designed to be safety grade. The equipment will be qualified to IEEE-344-1975, IEEE-323-1974 and NUREG-0588 in accordance with the Commission order on May 23rd, 1980 (CLI-20-81).

The accuracy of these instruments is ±2% of full scale.

Installation of the hydrogen monitoring instrumentation will be complete by January 1982.

18.1.31 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (II.F.2)

18.1.31.1 Statement of Requirement

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

18.1.31.2 Interpretation

None required.

18.1.31.3 Statement of Response

The Susquehanna SES reactor vessel water level instruments utilize redundant cold reference legs. The reference legs are connected to redundant and diverse level instruments by parallel instrument lines. The level instruments provide a range of measurement from below the active fuel to above the main steam lines. The fuel zone instruments are calibrated to LOCA conditions. This configuration provides optimum performance for all operating conditions and credible transients. The reactor vessel water level instruments are standard for BWR/5's and later BWR/4's and were evaluated by the BWR Owners' Group and found to be adequate to detect inadequate core cooling (ICC). This instrumentation is described and documented in NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors". Since the present design uses the optimum reference leg configuration, no additional instrumentation or modifications to instrumentation are needed for detection of ICC.

Symptom based procedures are being developed (in response to requirement I.C.1) for proper identification of ICC. These possible scheme that should be considered is ADS actuation on low reactor-vessel water level provided no high-pressure coolant injection or high-pressure coolant system flow exists and a low-pressure emergency core cooling system is running. This logic would complement, not replace, the existing ADS actuation logic.

Applicants for operating license shall provide results of feasibility study 1 year prior to issuance of operating license. A description of the proposed modification for staff approval is required four months prior to issuance of an operating license.

18.1.54.2 Interpretation

The ADS actuation logic may not be automatically actuated for steam line breaks (SLB) outside containment. The operator must manually actuate the ADS after diagnosing that an SLB has occurred. The ADS actuation logic should be modified to provide automatic actuation for all Design Basis Accidents.

18.1.54.3 Statement of Response

The BWR Owners' Group (BWROG) has completed a generic feasibility study in response to this requirement. The results were transmitted by a letter from D. B. Waters to D. G. Eisenhut on March 31, 1981.

PP&L has reviewed the BWROG report and has committed to modify the ADS logic such that it will be initiated after an appropriate time delay following a low reactor water level condition (level 1). This commitment is discussed in a letter from N. W. Curtis to A. Schwencer on June 17, 1981 (PLA-851).

18.1.55 RESTART OF CORE SPRAY AND LOW PRESSURE COOLANT INJECTION SYSTEMS (II.K.3.21)

18.1.55.1 Statement of Requirement

The core-spray and low-pressure, coolant-injection (LPCI) system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart, if required, to assure adequate core cooling. Because this design modification affects several core-cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification.

All applicants for operating license should submit documentation four months prior to the expected issuance of an operating license or four months prior to the listed implementation date, whichever is later.

18.1.55.2 Interpretation

None required.

18.1.55.3 Statement of Response

PP&L concurs with the BWR Owners' Group position which was forwarded to the NRC by letter from D. B. Waters (BWROG) to D. G. Eisenhut (NRC), December 29, 1980.

The BWROG report states that the current ECCS design represents the optimum approach to BWR safety. No modifications to existing LPCI and core spray systems are necessary in response to this requirement.

18.1.56 AUTOMATIC SWITCHOVER OF REACTOR CORE ISOLATION COOLING SYSTEM SUCTION (II.K.3.22)

18.1.56.1 Statement of Requirement

The reactor core isolation cooling (RCIC) system takes suction from the condensate storage tank with manual switchover to the suppression pool when the condensate storage tank level is low. This switchover should be made automatically. Until the automatic switchover is implemented, licensees should verify that clear and cogent procedures exist for the manual switchover of the RCIC system suction from the condensate storage tank to the suppression pool.

Documentation must be submitted four months prior to issuance of the staff safety evaluation report or four months prior to the implementation date, whichever is later. Modifications shall be completed by January 1, 1982.

18.1.56.2 Interpretation

None required.

18.1.56.3 Statement of Response

1Manual switchover of the RCIC suction from the condensate storage tank (CST) to the suppression pool on low CST level is covered in the Emergency Operating Procedures.

Specifically, this item is addressed in the following Emergency Operating Procedures:

EO-00-022 (Cooldown) Section 2.C.
EO-00-023 (Containment Control) Section 2.D.

This procedural guidance is an interim measure and will be revised to discuss automatic switchover of the RCIC suction when the design change is implemented.

The design changes for automatic switchover are being developed. All modifications will be completed by January 1982.

18.1.57 CONFIRM ADEQUACY OF SPACE COOLING FOR HIGH PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING SYSTEMS (II.K.3.24)

18.1.57.1 Statement of Requirement

Long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems may require space cooling to maintain the pump-room temperatures within allowable limits. Licensees should verify the acceptability of the consequences of a complete loss of alternating-current (AC) power. The RCIC and HPCI systems should be designed to withstand a complete loss of offsite AC power to their support systems, including coolers, for at least 2 hours.

All applicants for operating license should submit documentation four months prior to the expected issuance of the staff safety evaluation report for an operating license or four months prior to the listed implementation date, whichever is later.

18.1.57.2 Interpretation

Confirm that HPCI and RCIC room cooling can be maintained to enable continuous operation during a loss of offsite AC power for 2 hours.

18.1.57.3 Statement of Response

The HPCI and RCIC room unit coolers and their support systems are designed to withstand the consequences of a complete loss of offsite AC power since these are powered from onsite diesel generators. Each HPCI and RCIC room is provided with a 100% capacity redundant unit cooler. Refer to Subsection 9.4.2.2.

18.1.58 EFFECT OF LOSS OF ALTERNATING-CURRENT POWER ON RECIRCULATION PUMP SEALS (II.K.3.25)

18.1.58.1 Statement of Requirement

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (AC) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

Applicants for operating licenses shall submit the evaluation and proposed modifications no later than 6 months prior to expected issuance of the staff safety evaluation report in support of license issuance, whichever is later. Modifications must be completed by January 1, 1982.

18.1.58.2 Interpretation

Evaluate the effect of a loss of offsite AC power for 2 hours on the recirculation pump seals.

18.1.58.3 Statement of Response

The system(s) providing cooling water to the recirculation pump seals will be modified to automatically receive emergency power following a loss of offsite power. These modifications will be completed prior to the first refueling outage.

18.1.59 PROVIDE A COMMON REFERENCE LEVEL FOR VESSEL LEVEL INSTRUMENTATION (II.K.3.27)

18.1.59.1 Statement of Requirement

Different reference points of the various reactor vessel water level instruments may cause operator confusion. Therefore, all level instruments should be referenced to the same point. Either the bottom of the vessel or the top of the active fuel are reasonable reference points.

All applicants for operating license should submit documentation four months prior to the expected issuance of the staff safety evaluation report for an operating license or four months prior to the listed implementation date, whichever is later.

18.1.59.2 Interpretation

None required.

18.1.59.3 Statement of Response

Susquehanna SES will be modified so that all reactor water level indications use the same reference point, the bottom of the steam dryer skirt. This commitment was previously stated in letters from N. W. Curtis to A. Schwencer on July 21 and August 4, 1981 (PLA-888, -987).

18.1.60 VERIFY QUALIFICATION OF ACCUMULATORS ON AUTOMATIC DEPRESSURIZATION SYSTEM VALVES (II.K.3.28)

18.1.60.1 Statement of Requirement

Safety analysis reports claim that air or nitrogen accumulators for the automatic depressurization system (ADS) valves are provided with sufficient capacity to cycle the valves open five times at design pressures. GE has also stated that the emergency core cooling (ECC) systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. Licensee should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the licensee must show that the accumulator design is still acceptable.

The ADS valves, accumulators, and associated equipment and instrumentation must be capable of performing their functions during and following exposure to hostile environments and taking no credit for nonsafety-related equipment or instrumentation. Additionally, air (or nitrogen) leakage through valves must be accounted for in order to assure that enough inventory of compressed air is available to cycle the ADS valves.

All applicants for operating license shall submit documentation four months before the expected issuance of the staff safety evaluation report for an operating license or four months before the listed implementation date, whichever is later.

18.1.60.2 Interpretation

None required.

18.1.60.3 Statement of Response

The BWR Owners' Group (BWROG) is performing a generic evaluation in response to this requirement. PP&L will prepare a response following review of the BWROG report.

18.1.61 REVISED SMALL-BREAK LOSS OF COOLANT ACCIDENT METHODS (II.K.3.30)

18.1.61.1 Statement of Requirement

The analysis methods used by nuclear steam supply system vendors and/or fuel suppliers for small-break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT test and Semiscale Test facilities.

The Bulletins and Orders Task Force identified a number of concerns regarding the adequacy of certain features of small-break LOCA models, particularly the need to confirm specific model features (e.g., condensation heat transfer rates) against applicable experimental data. These concerns, as they applied to each light-water reactor (LWR) vendor's models, were documented in the task force also concluded that, in light of the TMI-2 accident, additional systems verification of the small-break LOCA model as required by II.4 of Appendix K to 10 CFR 50 was needed. This included providing experimental verification of the various modes of single-phase and two-phase natural circulation predicted to occur in each vendor's reactor during small-break LOCAs.

Based on the cumulative staff requirements for additional small-break LOCA model verification, including both integral system and separate effects verification, the staff considered model revision as the appropriate method for reflecting any potential upgrading of the analysis methods.

The purpose of the verification was to provide the necessary assurance that the small-break LOCA models were acceptable to calculate the behavior and consequences of small primary system breaks. The staff believes that this assurance can alternatively be provided, as appropriate, by additional justification of the acceptability of present small-break LOCA models with regard to specific staff concerns and recent test data. Such justification could supplement or supersede the need for model revision.

The specific staff concerns regarding small-break LOCA models are provided in the analysis sections of the B&O Task Force reports for each LWR vendor, (NUREG-0635, -0565, -0626, -0611, and -0623). These concerns should be reviewed in total by each holder of an approved emergency core cooling system model and addressed in the evaluation as appropriate.

The recent tests include the entire Semiscale small-break test series and LOFT Tests (L3-1) and L3-2). The staff believes that the present small-break LOCA models can be both qualitatively and quantitatively assessed against these tests. Other separate effects tests (e.g., ORNL core uncovery tests) and future tests, as appropriate, should also be factored into this assessment.

Based on the preceding information, a detailed outline of the proposed program to address this issue should be submitted. In particular, this submittal should identify (1) which areas of the models, if any, the licensee intends to upgrade, (2) which areas the licensee intends to address by further justification of acceptability, (3) test data to be used as part of the overall verification/upgrade effort, and (4) the estimated schedule for performing the necessary work and submitting this information for staff review and approval.

Licensees shall submit an outline of a program for model justification/revision by November 15, 1980. Licensees shall submit additional information for model justification and/or revised analysis model for staff approval by January 1, 1982. Licensees shall submit their plant-specific analyses using the revised models by January 1, 1983 or one year after any model revisions are approved. Applicants shall submit appropriate information in accordance with the licensing review schedule.

18.1.61.2 Interpretation

None required.

18.1.61.3 Statement Of Response

PP&L considers that the reactor vendor, General Electric, is the most appropriate party to work with the staff in resolving staff concerns with small break LOCA models for BWRs. Accordingly, the staff should direct their questions regarding the scope and schedule for this requirement to General Electric (attn. R. H. Buchholz, Manager, BWR Systems Licensing). Copies of correspondence on this item should be sent to PP&L so that we may remain cognizant of the progress of the program to resolve the staff's concerns on this requirement.

18.1.62 PLANT-SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10CFR PART 50.46 (II.K.3.31)

18.1.62.1 Statement of Requirement

Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents (LOCAs) as described in item II.K.3.30 to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees.

18.1.62.2 Interpretation

None required.

18.1.62.3 Statement of Response

Plant specific calculations will be performed, if required, following NRC approval of LOCA model revisions required by item II.K.3.30 (see Subsection 18.1.61).

18.1.63 EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO FUEL CLADDING FAILURE (II.K.3.44)

18.1.63.1 Statement of Requirement

For anticipated transients combined with the worst single failure an assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result in a stuck-open relief valve should be included in this category.

All applicants for operating license should submit documentation four months prior to the expected issuance of the staff safety evalaution report for an operating license or four months prior to the listed implementation date, whichever is later.

18.1.63.2 Interpretation

None required.

18.1.63.3 Statement of Response

The BWR Owners' Group has prepared a generic response to this requirement. The report was transmitted to D. G. Eisenhut by a letter from D. B. Waters on December 29, 1980. This response contains an evaluation of analyses performed to demonstrate the core remains covered or no significant fuel damage occurs from an anticipated transient with a single failure. PP&L has reviewed this response and finds it is applicable to Susquehanna SES. The report concludes that the core remains covered for all evaluated combinations of anticipated transients and single failures.

18.1.64 EVALUATION OF DEPRESSURIZATION WITH OTHER THAN THE AUTOMATIC DEPRESSURIZATION SYSTEM (II.K.3.45)

18.1.64.1 Statement of Requirement

Analyses to support depressurization modes other than full actuation of the automatic depressurization system (ADS) (e.g., early blowdown with one or two safety relief valves) should be provided. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapid cooldown.

All applicants for operating license should submit documentation four months prior to the expected issuance of the staff safety evaluation report for an operating license or four months prior to the listed implementation date, whichever is later.

18.1.64.2 Interpretation

None required.

18.1.64.3 Statement of Response

The BWR Owners' Group submitted a generic response to this requirement. This response was transmitted by letter to D. G. Eisenhut from D. B. Waters on December 29, 1980. PP&L has reviewed this response and find it applicable to Susquehanna SES. The report concludes that no improvement can be gained by a slower depressurization and actually could be detrimental to core cooling. Therefore no additional action is necessary in response to this requirement.

18.1.65 MICHELSON CONCERNS (II.K.3.46)

18.1.65.1 Statement of Requirement

A number of concerns related to decay heat removal following a very small break LOCA and other related items were questioned by Mr. C. Michelson of the Tennessee Valley Authority. These concerns were identified for PWRs. GE was requested to evaluate these concerns as they apply to BWRs and to assess the importance of natural circulation during a small-break LOCA in BWRs.

18.1.65.2 Interpretation

None required.

18.1.65.3 Statement of Response

The General Electric Company has responded to the questions posed by Mr. Michelson. This response was sent by letter from R. H. Buchholz to D. F. Ross on February 21, 1980. These responses are applicable to Susquehanna SES and no further response is necessary.

18.1.66 EMERGENCY PREPAREDNESS-SHORT TERM (III.A.1.1)

No requirement stated in NUREG 0737. Refer to Subsection 18.2.38 which contains the response to the requirement in NUREG 0694.

18.1.67 UPGRADE EMERGENCY SUPPORT FACILITIES (III.A.1.2)

18.1.67.1 Statement of Requirement

A detailed statement of the requirement can be found in NUREG-0696. The implementation schedule was announced in Generic Letter 81-10 on February 18, 1981. This schedule is as follows: Design information for emergency response facilities should be provided in connection with the operating license review process. These facilities shall be operational by October 1, 1982 or prior to fuel load, whichever is later. Interim facilities, as described in NUREG-0694 shall be provided by fuel load.

18.1.67.2 Interpretation

None required.

18.1.67.3 Statement of Response

The proposed method of responding to this requirement was submitted by a letter to B. J. Youngblood from N. W. Curtis on April 2, 1981 (PLA-704). Details on the emergency response facilities are presented in Appendix I of the Emergency Plan.

18.1.68 EMERGENCY PREPAREDNESS-LONG TERM (III.A.2)

18.1.68.1 Statement of Requirement

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in

NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

NUREG-0654, Revision 1; NUREG-0696, "Functional Criteria for Emergency Response Facilities;" and the amendments to 10 CFR Part 50 and Appendix E to 10 CFR Part 50 regarding emergency preparedness, provide more detailed criteria for emergency plans, design, and functional criteria for emergency response facilities and establishes firm dates for submission of upgraded emergency plans for installation of prompt notification systems. These revised criteria and rules supersede previous Commission guidance for the upgrading of emergency preparedness at nuclear power facilities.

Requirements of the new emergency-preparedness rules under paragraphs 50.47 and 50.54 and the revised Appendix E to Part 50 taken together with NUREG-0654 Revision 1 and NUREG-0696, when approved for issuance, go beyond the previous requirements for meteorological programs. To provide a realistic time frame for implementation, a staged schedule has been established with compensating actions provided for interim measures.

Specific milestones have been developed and are presented below.

Milestones are numbered and tagged with the following code; a-date, b-activity, c-minimum acceptance criteria. They are as follows:

(1) a. Fuel load.

b. Submittal of radiological emergency response plans.

c. A description of the plan to include elements of NUREG-0654, Revision 1, Appendix 2.

(2) a. Fuel load.

b. Submittal of implementing procedures.

c. Methods, systems, and equipment to assess and monitor actual or potential offsite consequences of a radiological emergency condition shall be provided.

(3) a. Fuel load.

b. Implementation of radiological emergency response plans.

c. Four elements of Appendix 2 to NUREG-0654 with the exception of the Class B model of element 3, or

Alternative to item (3) requiring compensating actions:

A meteorological measurements program which is consistent with the existing technical specifications as the the baseline or an element 1 program and/or element 2 system of Appendix 2 to NUREG-0654, or two independent element 2 systems shall provide the basic meteorological parameters (wind direction and speed and an indicator or atmospheric stability) on display in the control room. An operable dose calculational methodology (DCM) shall be in use in the control room and at appropriate emergency response facilities.

The following compensating actions shall be taken by the licensee for this alternative:

(i) If only element 1 or element 2 is in use:

- The licensee (the person who will be responsible for making offsite dose projections) shall check communications with the cognizant National Weather Service (NWS) first order station and NWS forecasting station on a monthly basis to ensure that routine meteorological observations and forecasts can be accessed.

- The licensee shall calibrate the meteorological measurements program at a frequency no less than quarterly and identify a readily available source of meteorological data (characteristic of site conditions) to which they can gain access during calibration periods.

- During conditions of measurements system unavailability, an alternate source of meteorological data which is characteristic of site conditions shall be identified to which the licensee can gain access.

- The licensee shall maintain a site inspection schedule for evaluation of the meteorological measurements program at a frequency no less than weekly.

- It shall be a reportable occurence if the meteorological data unavailability exceeds the goals outline in Proposed Revision 1 to Regulatory Guide 1.23 on a quarterly basis.

(ii) The portion of the DCM relating to the transport and diffusion of gaseous effluents shall be consistent with the characteristics of the Class A model outlined in element 3 of Appendix 2 to NUREG-0654.

(iii) Direct telephone access to the individual responsible for making offsite dose projections (Appendix E to 10 CFR Part 50(IV)(A)(4)) shall be available to the NRC in the event of a radiological emergency. Procedures for establishing contact and identification of contact individuals shall be provided as part of the implementing procedures.

This alternative shall not be exercised after July 1, 1982. Further, by July 1, 1981, a functional description of the upgraded programs (four elements) and schedule for installation and full operational capability shall be provided (see milestones 4 and 5).

(4) a. March 1, 1982.

b. Installation of Emergency Response Facility hardware and software.

c. Four elements of Appendix 2 to NUREG-0654, with exception of the Class B model of element 3.

(5) a. July 1, 1982.

b. Full operational capability of milestone 4.

c. The Class A model (designed to be used out to the plume exposure EPZ) may be used in lieu of Class B model out to the ingestion EPZ. Compensating actions to be taken for extending the application of the Class A model out to the ingestion EPZ include access to supplemental information (meso and synoptic scale) to apply judgment regarding intermediate and long-range transport estimates. The distribution of meteorological information by the licensee should be as described in Table 18.1-13 by July 1, 1982.

(6) a. July 1, 1982 or at the time of the completion of milestone 5, whichever is sooner.

b. Mandatory review of the DCM by the licensee.

c. Any DCM in use should be reviewed to ensure consistency with the operational Class A model. Thus, actions recommended during the initial phases of a radiological emergency would be consistent with those after the TSC and EOF are activated.

(7) a. September 1, 1982.

b. Description of the Class B model provided to the NRC.

c. Documentation of the technical bases and justification for selection of the type Class B model by the licensee with a discussion of the site-specific attributes.

(8) a. June 1, 1983.

b. Full operational capability of the Class B model.

c. Class B model of element 3 of Appendix 2 to NUREG-0654, Revision 1

Applicants for an operating license shall meet at least milestones 1, 2, and 3 prior to the issuance of an operating license. Subsequent milestones shall be met by the same dates indicated for operating reactors. For the alternative to milestone 3, the meteorological measurements program shall be consistent with the NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Secton 2.3.3 program as the baseline or element 1 and/or element 2 systems.

18.1.68.2 Interpretation

None required.

18.1.68.3 Statement of Response

Milestones 1, 2 and 3 are being addressed as a part of the short term emergency preparedness requirement III.A.1.1. Refer to Subsection 18.2.38 for response. Responses to these and other milestones will be incorporated into Appendix I of the Emergency Plan.

18.1.69 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIAL (III.D.1.1)

18.1.69.1 Statement of Requirement

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction.

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction--Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

This requirement shall be implemented prior to issuance of a full-power license.

Applicants shall provide a summary description, together with initial leak-test results, of their program to reduce leakage from systems outside containment that would or could contain primary coolant or other highly radioactive fluids or gases during or following a serious transient or accident. Applicants shall submit this information at least four months prior to fuel load.

18.1.69.2 Interpretation

None required.

18.1.69.3 Statement of Response

1. Program summary description:

1.1 The following systems will be leak tested (the frequency is indicated in ( ) after each each item):

A.	Residual Heat Removal	(18 months)
B.	Reactor Core Isolation Cooling	"
C.	Core Spray	"
D.	High Pressure Core Injection	"
E.	Scram Discharge	"
F.	Reactor Water Clean-up*	"
G.	Standby Gas Treatment	"
H.	Containment Air Monitors	"
I.	Post Accident Sampling	"

Initial leak-test results will be available when the first measurements are made, prior to completion of the startup test program.

* NOTE: The RWCU system will not have significant post-accident radioactivity because the suction is isolated by containment isolation signals (refer to Table 18.1-10). However, this system may conceivably be used in some post-accident scenarios, and will therefore be leak tested.

1.2 The following systems contain radioactive material but are excluded from our program (justification for exclusion follows each item):

A. Main Steam - identified by NEDO-24782 as not to be regarded as containing highly radioactive fluid following an accident.

B. Feed water - same justification as A.

C. Main Steam Line Drain - this system is isolated following a LOCA.

D. Reactor Water Sample - this system will not be used following an accident, a separate post-accident sampling station is being developed in response to item II.B.3.

E. Recirculation Pump Seal Water (from CRD pumps) - lines are protected by check valves and an excess flow check valves.

F. Floor & Equipment Drains - this sytem isolated following a LOCA and will not be used following an accident.

G. Suppression Pool Clean-up & Drain - same justification as F.

1.3 Method for obtaining actual leak rates

A. Water - leakage will be collected in a graduated measuring device and timed to determine GPM leak rate. Implementing procedures will establish criteria for initiation of leak rate quantification.

B. Steam - an estimate of the size of the leak will be made (i.e. equivalent pipe diameter steam flow). Flowrate will be determined using standard Handbook data. This will be converted to a GPM flowrate using the specific volume of the steam at the given conditions.

2. The two gaseous systems are tested as follows:

A. Standby Gas Treatment System - This system is subject to filter efficiency testing in accordance with the Technical Specifications which includes "DOP" and refrigerant injection.

B. Containment Air Monitors - These are tested while the system is under normal running conditions by checking each mechanical joint with liquid soap.

3. Consideration was given to the Standby Gas system regarding the incident at North Anna Unit 1 in 1979. The standby gas piping and duct work from the containment to the filters are gas tight and do not include any pressure relief devices which would allow gases to escape to the Reactor Building. The piping is rated at 150 psig and the duct work is HVM-GS-G (High Velocity Medium Pressure - Galvanized Steel - Gas tight).

In light of the above, the actions stated in 1.1.G and 2.A have resulted.

4. Technical Specifications will incorporate an acceptance criteria of 5 GPM total leakage rate for the systems listed in 1.1 with the exception of:

A. Standby Gas Treatment - which is limited to the acceptance criteria stated in Technical Specifications Subsection 4.6.5.3 and

B. The containment air monitors - which has an acceptance criteria of zero leakage as determined by a liquid soap test.

The program will implemented prior to fuel load.

18.1.70 INPLANT IODINE RADIATION MONITORING (III.D.3.3)

18.1.70.1 Statement of Requirement

Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments using sample media that will collect iodine selectively over xenon (e.g., silver zeolite) for the following reasons:

(1) The physical size of the auxiliary and/or fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

(2) Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

(3) Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

(4) The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high-dose-rate areas.

After January 1, 1981, each applicant and licensee shall have the capability to remove the sampling cartridge to a low-background, low-contamination area for further analysis. Normally, counting rooms in auxiliary buildings will not have sufficiently low backgrounds for such analyses following an accident. In the low background area, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples under accident conditions. There should be sufficient samplers to sample all vital areas.

18.1.70.2 Interpretation

PP&L is in basic agreement with the technical discussion as outlined in this requirement. It should be noted that Susquehanna SES is a BWR and does not possess an auxiliary building. Consequently, it is premature to suggest that our counting facilities within the control structure will be inadequate to effectively count air samples. Additionally, purging of the air sample cartridges may not be necessary if an effective collection media is used for radioiodine air sampling.

18.1.70.3 Statement of Response

PP&L intends to meet the requirements defined in this item. To summarize the program being implemented, three (3) Eberline Instrument Corporation PING - 2A Particulate, Iodine and Noble Gas Air Monitoring Systems are provided for air sampling plant areas where personnel may be present during accident conditions. The systems are cart mounted with battery back-ups.

Grap samples are obtained using the equipment specified in Subsection 12.5.2.6.3. During accident conditions silve zeolite cartirdges will be used for radioiodine analysis in conjunction with two (2) Eberline stabilized assay meters (SAM-2).

Air samples are evaluated as specified in Subsection 12.5.3.5.5. In addition to initial training provided for Health Physics personnel, periodic drills are conducted in accordance with the Susquehanna Emergency Plan Section 8.1.2 (See Amendment 25 of Operating License Application).

Analysis of iodine cartridges will be performed in a low background, low contamination area. During accident conditions, preliminary analysis will be performed by onsite radiation monitoring teams in the counting room, if accessible using a SAM-2. Final analysis will be performed in the health physics facilities in the training center (onsite) or in another laboratory facility (offsite) where appropriate sensitivity can be achieved. Prior to analysis, cartridges will be purged using station service air or bottled nitrogen which is stored on site.

All equipment and procedures will be available for use by fuel load.

18.1.71 CONTROL ROOM HABITABILITY REQUIREMENTS (III.D.3.4)

18.1.71.1 Statement of Requirement

Licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

All licensees must make a submittal to the NRC regardless of whether or not they met the criteria of the Standard Review Plans (SRP) sections listed below. The new clarification specifies that licensees that meet the criteria of the SRPs should provide the basis by referencing past submittals to the NRC and/or providing new or additional information to supplement past submittals.

18.1.71.1.1 Requirements for Licensees that Meet Criteria

All licensees with control rooms that meet the criteria of the following sections of the Standard Review Plan:

2.2.1-2.2.2	Identification of Potential Hazards in Site Vicinity
2.2.3	Evaluation of Potential Accidents;
6.4	Habitability Systems

shall report their findings regarding the specific SRP sections as explained below. The following documents should be used for guidance:

(a) Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of Regulatory Power Plant Control Room During a Postulated Hazardous Chemical Release";

(b) Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accident Chlorine Release"; and,

(c) K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

Licensees shall submit the results of their findings as well as the basis for those findings by January 1, 1981. In providing the basis for the habitability finding, licensees may reference their past submittals. Licensees should, however, ensure that these submittals reflect the current facility design and that the information requested in Attachment 1 of NUREG 0737 is provided.

18.1.71.1.2 Requirements for Licensees that Do Not Meet Criteria

All licensees with control rooms that do not meet the criteria of the above-listed references, Standard Review Plans, Regulatory Guides, and other references shall perform the evaluations and identify appropriate modifications, as discussed below.

Each licensee submittal shall include the results of the analyses of control room concentrations from postulated accidental release of toxic gases and control room operator radiation exposures from airborne radioactive material and direct radiation resulting from design-basis accidents. The toxic gas accident analysis should be performed for all potential hazardous chemical releases occurring either on the site or within 5 miles of the plant-site boundary. Regulatory Guide 1.78 lists the chemicals most commonly encountered in the evaluation of control room habitability but is not all inclusive.

The design-basis-accident (DBA) radiation source term should be for the loss-of-coolant accident LOCA containment leakage and engineered safety feature (ESF) leakage contribution outside containment as described in Appendix A and B of Standard Review Plan Chapter 15.6.5. In addition, boiling-water reactor (BWR) facility evaluations should add any leakage from the main steam isoaltion valves (MSIV) (i.e., valve-stem leakage, valve seat leakage, main steam isolation valve leakage control system release) to the containment leakage and ESF leakage following a LOCA. This should not be construed as altering the staff recommendations in Section D of Regulatory Guide 1.96 (Rev. 2) regarding MSIV leakage-control systems. Other DBAs should be reviewed to determine whether they might constitute a more-severe control-room hazard than the LOCA.

In addition to the accident-analysis results, which should either identify the possible need for control-room modifications or provide assurance that the habitability systems will operate under all postulated conditions to permit the control-room operators to remain in the control room to take appropriate actions required by General Design Criterion 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems. Attachment 1 of NUREG 0737, item III.D.3.4 lists the information that should be provided along with the licensee's evaluation.

18.1.71.1.3 Documentation and Implementation

Applicants for operating licenses shall submit their responses prior to issuance of a full-power license. Modifications needed for compliance with the control-room habitability requirements specified in this letter should be identified, and a schedule for completion of the modifications should be provided. Implementation of such modifications should be started without awaiting the results of the staff review. Additional needed modifications, if any, identified by the staff during its review will be specified to licensees.

18.1.71.2 Interpretation

None required.

18.1.71.3 Statement of Response

The control room HVAC system layout and functional design includes protection of the control room from radioactive and toxic gases. Subsection 6.4 provides a complete description of this system and compliance to habitability requirements. Refer to Subsection 6.4 for the response to this requirement.

REFERENCES

18.1-1 Letter, D. G. Eisenhut (NRC) to S. T. Rogers (BWR Owners' Group), regarding Emergency Procedure Guidelines, October 21, 1980

18.1-2 U.S. Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" USNRC Report NUREG-0578, July 1979, Recommendation 2.1.6b.

18.1-3 U.S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," USNRC-0660, Vols. 1 and 2, May 1980, Section II.B.2.

18.1-4 Letter from D. G. Eisenhut (NRC) to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, Subject: Preliminary Clarification of TMI Action Plan Requirements, dated September 5, 1980.

18.1-5) U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," USNRC Report NUREG-0737, November, 1980, Item II.B.2.

18.1-6) U.S. Nuclear Regulatory Commission, IE Bulletin No. 79-01B, "Environmental Qualification of Class IE Equipment", January 14, 1980.

18.1-7 U.S. Nuclear Regulatory Commission, "Interim Staff Position on Environmental Qualification Report NUREG-0588, December 1979.

18.1-8 USNRC Standard Review Plan 6.4, "Habitability Systems", Revision 1.

18.1-9 USNRC Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors", Revision 2, June 1974.

18.1-10 USNRC Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, November 1978.

18.1-11 USNRC Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants," November 1974.

18.1-12 Code of Federal Regulations, 10CFR Part 50, Appendix A, GDC 19, Revised as of January 1, 1980.

18.1-13 C. Michael Lederer, et al., <u>Table of Isotopes</u>, Lawrence Radiation Laboratory, University of California, March 1968.

18.1-14 D. S. Duncan and A. B. Spear, <u>GRACE I - An IBM 704-709 Program Design for Computing Gamma Ray Attenuation and Heating in Reactor Shields</u>, Atomics International, (June 1959).

18.1-15 D. S. Duncan and A. B. Spear, <u>GRACE II - An IBM 709 Program for Computing Gamma Ray Attenuation and Heating in Cylindrical and Spherical Geometries</u>, Atomics International, November 1959.

18.1-16 Memorandum of Telephone Conversation, S. Ford of LIS to N. Anderson of NRC's Lessons Learned Task Force, Subject: TMI Requirements at SHNPP, April 9, 1980.

18.1-17 USNRC Regional Meeting Minutes, Region I, Subject: TMI Review Requirements at SHNPP, April 9, 1980.

18.1-18 USNRC Regional Meeting Minutes, Region IV and V, Subject: TMI Review Requirements, 9/26/79.

TABLE 18.1-10 (CONTINUED)

P&ID. SYSTEM	E OR NE	BASIS (1)	PENETR. NO.	VALVE NO.	VALVE ACTUA-TION	AUTOMATIC ACTUATION SIGNALS (2)	ELECTRICAL SCHEMATIC DIAGRAM	GE ELEM & ACTUATING RELAY	AUTO OPEN ON ISO RESET	VALVE STATUS LOCATION INDICATION SOURCE	LOCAL PNL	CONTROL RM	HAND SWITCH NO.	TYPE (14)	POWER SOURCES (3) VALVE MOTOR	CONTROL	LOGIC	OTHER REMARKS
M-155 HPCI (CONT.)	E	21.	X-210	-1F066	RM	-	15	-		ZS	-	C601	15566	CR2940 (KLRO) MAINT	1D27.4		-	
				-1F049	CK	-	-	-	-	-	-	-	-	-	-	-	-	
	E	23.	X-209	-1F042	AI	L	E-152 SH 14	E41-69 (K34)	NO	..	-	..	15542	CR2940 SRA	1D264		BUS B	
	E	5.	X-9B	-1F006	AC	G -	9	(K2)	(16)	..	-	..	15506	CR2940 SRA	1D264		BUS B	
M-157 CONTMT ATMOS CONTROL	NE	24.	X-26	HV 15711	AI	Y (11)	E-171 SH 19	B21-131 (K83)	NO	ZS	-	C601	15711	E-30AB MOM	1Y226		RPS B	(17)
				HV 15713	..	Y ..	8	(K84)	..	..	-	..	15713	..	1Y216		RPS A	..
				HV 15714	..	Y -	3	(K83)	..	..	-	..	15714	..	1Y226		RPS B	..
	E	25.	X-60A	SV 15740B	..	Y (12)	4	(K84)	..	..	CB220B	..	15740B	..	1Y216	1D614	RPS A	(13)
				SV 15742B	..	Y ..	..	(K84)	..	..	..	..	15742B	..	..	..	..	..
	E	25.	X-60A	SV 15776B	..	Y ..	..	(K84)	..	..	..	..	15740B	..	..	..	..	..
				SV 15774B	..	Y ..	..	(K84)	..	..	..	..	15742B	..	..	..	..	..
	E	25.	X-60A	SV 15750B	..	Y ..	..	(K84)	..	..	..	..	15740B	..	..	..	..	..
				SV 15752B	..	Y ..	..	(K84)	..	..	..	..	15742B	..	..	..	..	..
	NE	24.	X-202	HV 15703	..	Y (11)	8	(K84)	..	..	-	..	15703	..	..		..	(17)
				HV 15704	..	Y -	3	(K83)	..	..	-	..	15704	..	1Y226		RPS B	..
				HV 15705	..	Y (11)	19	(K83)	..	..	-	..	15705	..	1Y226		..	..
	E	25.	X-221A	SV 15780B	..	Y (12)	4	(K84)	..	..	CB220B	..	15740B	..	1Y216	1D614	RPS A	(13)
				SV 15782B	..	Y ..	..	(K84)	..	..	..	..	15742B	..	..	..	..	..
	E	25.	X-238A	SV 15736B	..	Y ..	..	(K84)	..	..	..	..	15740B	..	..	..	..	..
				SV 15734B	..	Y ..	..	(K84)	..	..	..	..	15742B	..	..	..	..	..
	E	25.	X-80C	SV 15740A	..	Y ..	..	(K83)	..	..	CB220A	..	15740A	..	1Y226	1D624	RPS B	..
				SV 15742A	..	Y ..	..	(K83)	..	..	..	..	15742A	..	..	..	..	..
	E	25.	X-80C	SV 15750A	..	Y ..	..	(K83)	..	..	..	..	15740A	..	..	..	..	..
				SV 15752A	..	Y ..	..	(K83)	..	..	..	..	15742A	..	..	..	..	..
	E	25.	X-80C	SV 15776A	..	Y ..	..	(K83)	..	..	..	..	15740A	..	..	..	..	..
				SV 15774A	..	Y ..	..	(K83)	..	..	..	..	15742A	..	..	..	..	..
				SV 15767	..	Y -	9	(K83)	..	..	-	..	15767	..	..		RPS A	-
	NE	24.	X-25	HV 15722	..	Y -	3	(K84)	..	..	-	..	15722	..	1Y216		..	(17)
				HV 15723	..	Y -	..	(K83)	..	..	-	..	15723	..	1Y226		RPS B	..
				HV 15721	..	Y (11)	..	(K83)	..	..	-	..	15721	..	1Y226		..	..
	NE	24.	X-201A	HV 15725	..	Y (12)	..	(K84)	..	..	-	..	15725	..	1Y216		RPS A	..
				HV 15724	..	Y -	..	(K83)	..	..	-	..	15724	..	1Y226		RPS B	..
	E	25.	X-238B	SV 15734A	..	Y (12)	4	(K83)	..	..	CB220A	..	15742A	..	1Y226	1D624	..	(13)
				SV 15736A	..	Y ..	4	(K83)	..	..	..	..	15740A	..	1Y226	..	..	..
				SV 15737	..	Y ..	9	(K83)	..	..	-	..	15737	..	1Y226		..	..
	E	25.	X-221B	SV 15780A	..	Y ..	4	(K83)	..	..	CB220A	..	15740A	..	1Y226	1D624	..	..
				SV 15782A	..	Y ..	4	(K83)	..	..	..	..	15742A	..	1Y226	..	..	..

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FIGURE TO BE PROVIDED AT A LATER DATE

Rev. 27, 10/81

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

PASS P&ID

FIGURE 8.1-11

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FIGURE TO BE PROVIDED AT A LATER DATE

Rev. 27, 10/81

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

LOCATION OF PASS - EL. 719'-1"

FIGURE 18.1-12

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FIGURE TO BE PROVIDED AT A LATER DATE

Rev. 27, 10/81

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

LOCATION OF PASS - EL. 729'

FIGURE 18.1-13

18.2.10.2 Interpretation

None required.

18.2.10.3 Statement of Response

Administrative procedure AD-QA-300, "Conduct of Operations," provides the authority and instructions for control room access control.

18.2.11 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF (I.C.5)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.12 for response.

18.2.12 NSSS VENDOR REVIEW OF PROCEDURES (I.C.7)

18.2.12.1 Statement of Requirement

Obtain nuclear steam supply system vendor review of low-power testing procedures to further verify their adequacy. This requirement shall be met prior to fuel load.

Obtain NSSS vendor review of power-ascension test and emergency procedures to further verify their adequacy. This requirement must be met before issuance of a full-power license.

18.2.12.2 Interpretation

None required.

18.2.12.3 Statement of Response

The General Electric Company, through its site startup organization, will review all startup tests associated with NSSS systems and will review all Emergency Operating procedures. The startup tests encompass the low power testing and the power ascension testing phases. These reviews will be completed prior to fuel load.

18.2.13 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NEAR-TERM OPERATING LICENSE APPLICANTS (I.C.8)

18.2.13.1 Statement of Requirement

Correct emergency procedures, as necessary, based on the NRC audit of selected plant emergency operating procedures (e.g., small-break LOCA, loss of feedwater, restart of engineered safety features following a loss of AC power or, steam-line break).

18.2.13.2 Interpretation

None required.

18.2.13.3 Statement of Response

The emergency procedures have been transmitted to the NRC by a letter from N. W. Curtis to B. J. Youngblood on March 4, 1981. No further response is necessary until the NRC completes the audit and issues specific requirements.

18.2.14 CONTROL ROOM DESIGN (I.D.1)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.16 for response.

18.2.15 TRAINING DURING LOW POWER TESTING (I.G.I)

18.2.15.1 Statement of Requirement

Define and commit to a special low-power testing program approved by NRC to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training. This requirement shall be met before fuel load.

Supplement operator training by completing the special low-power test program. Tests may be observed by other shifts or repeated on other shifts to provide training to the operators. This requirement shall be met before issuance of a full-power license.

18.2.15.2 Interpretation

None required.

18.2.15.3 Statement of Response

The BWR Owners' Group has prepared a generic response to this requirement. This was transmitted to D. G. Eisenhut by a letter from D. B. Waters on February 4, 1981. PP&L concurs with this response. This generic approach outlines an extensive testing program designed to contribute to and provide for extensive training opportunities during the start-up program. The objectives of this program are to provide:

1. A plant that has been thoroughly tested.

2. An operating staff that has received the maximum experience and in-plant training to safely operate it.

3. Plant procedures that have been reviewed and revised to provide the staff with proven directions and controls.

Susquehanna's Operator Training Program has been in progress since 1977 and is completing the final phases of training at this time. This program utilizes the Susquehanna Simulator located at the plant site and provides the operators with extensive training prior to actual operations in the plant itself. The Simulator is also used for procedure development and check out.

The Operator Training Program that is being developed for the Preoperational and Low Power Testing Program incorporates and builds on the extensive training already completed by the operations section. It will include the recommendation presented in the BWR Owners' Group position but goes beyond those recommendations by maximizing the use of the Susquehanna Simulator in preparing the operators for the start-up tests to be performed.

The objective of the Operator Training Program is to provide each operator with the maximum learning experience during the start-up phase. In order to achieve this objective, a comprehensive training program is being developed that utilizes the many training opportunities that are available during this period and ensures actual testing. This program covers the period from Preoperational/Acceptance Testing through the Power Test Program on Unit I. To support this amount of training the operations section which is staffed for six sections has reorganized into four sections. This reorganization provided the benefit of allowing more operators off shift to attend formal training as well as provide more operating experience for each shift team. Every effort is being made to keep the shifts intact and provide training that promotes the "Shift Team" concept.

The training program being developed covers the areas of activities listed below but recognizes the overlap that exists between some of the areas.

I. Preoperational/Acceptance Testing
II. Cold Functional Testing
III. Hot Functional Testing
IV. Start-up Tests
V. Additional Testing

Each area of testing has activities that lend itself to operator training. The major ones are outlined in Table 18.2-1. The training program provides a vehicle to identify activities that have a significant benefit for training, documents this training, and ensures that all shift crews receive equal experience opportunities. The program also attempts to schedule repeats of certain evolutions that are considered critical and cannot be routinely performed at a later time. The training program will identify areas of testing/training that while not required by start-up program would have additional training benefit. This testing/training could then be scheduled into the testing program as additional testing.

Finally this program will develop the basis for the In-Plant Drill Program. This comprehensive approach to testing/training more than adequately satisfies the requirements of NUREG 0737.

Susquehanna SES will conduct a test which simulates a loss of AC power condition for the reactor and containment systems. The purpose of this test will be to obtain data relative to the performance of these systems under the imposed condition of no AC power available for mitigation of transient effects. Several key factors associated with performance of this test include:

(a) No blocking of low pressure ECCS functions will be provided. Transient conditions imposed by the test are not expected to initiate these systems on low water level. Adequate data will be obtained prior to injection of low pressure ECCS flow on a high drywell pressure signal in conjunction with a confirmatory low reactor pressure signal. If initiation points are reached earlier in the test, this would represent a criteria for test termination and the ECCS would be allowed to function as designed.

(b) In most if not all cases, Limiting Conditions for Operation will not be violated. It is possible that as test plans are developed, certain LCO's will be identified as inhibiting test performance. However, for LCO's that are being approached as a result of transient effects during the test, this will represent criteria for test termination to insure plant safety.

(c) Loss of power to plant instrumentation will not be simulated. To accomplish the major goal of this test, instrumentatin will be maintained functional for proper data collection. This constraint is also considered necessary to insure plant sfaety. Training of licensed operators in response to the loss of all station AC condition will be provided at PP&L's plant specific simulator. This will provide the opportunity to experience the instrumentation blackout condition without placing the plant in jeopardy.

(d) Plant AC busses will remain energized, the emergency diesel generators will be available or operating, and breakers to safeguards equipment will not be racked out. These precautions are considered necessary for plant safety.

(e) Test termination criteria will be established for parameters such as reactor vessel temperature limits, containment pressure and temperature, reactor vessel level, suppression pool temperature, HPCI and RCIC room temperatures, and CRD temperatures.

The test will be conducted at a convenient point during the first fuel cycle, with the constraint that adequate decay heat exists to provide a valid test. Several options are being explored for test initiation (e.g., turbine trip from 5% power following load reduction, turbine trip from 85% power, etc.) It is our position that the method finally selected will have little bearing on the test, since plant performance during the initial state of such transients is adequately tested in startup tests.

The commitments previously made in response to Item I.G.1 for additional testing and training will be fulfilled, as it is felt that valuable input will be obtained for the loss of AC powertest.

The test may cause certain conditions which are inconsistent with NRC requirements. We therefore will provide the NRC with the test plan and pertinent documents for review and approval. Performance of this test is contingent upon a favorable safety evaluation and obtaining appropriate commission approvals.

Since the NRC has indicated that they may not require a test at Susquehanna SES if results from a test at another plant resolve their concerns, PP&L's commitment is contingent on a continuing NRC requirement for this test.

18.2.16 REACTOR COOLANT SYSTEM VENTS (II.B.1)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.19 for response.

18.2.17 PLANT SHIELDING (II.B.2)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.20 for response.

18.2.18 POSTACCIDENT SAMPLING (II.B.3)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.21 for response.

18.2.19 TRAINING FOR MITIGATING CORE DAMAGE (II.B.4)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.22 for response.

18.2.20 RELIEF AND SAFETY VALVE TEST REQUIREMENTS (II.D.1)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.23 for response.

18.2.21 RELIEF AND SAFETY VALVE POSITION INDICATION (II.D.3)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.24 for response.

18.2.22 CONTAINMENT ISOLATION DEPENDABILITY (II.E.4.2)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.29 for response.

18.2.23 ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION (II.F.1)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.30 for response.

18.2.24 INADEQUATE CORE COOLING INSTRUMENTS (II.F.2)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.31 for response.

18.2.25 ASSURANCE OF PROPER ESF FUNCTIONING (II.K.1.5)

18.2.25.1 Statement of Requirement

Review all valve positions, positioning requirements, positive controls and related test and maintenance procedures to assure proper ESF functioning. This requirement shall be met by fuel load.

18.2.25.2 Interpretation

None required.

18.2.25.3 Statement of Response

Operating and surveillance procedures are currently being developed. Writing the procedures to reflect ESF requirement is a key objective of procedure originators. Additionally, these procedures will receive a review (independent of the originator) to provide further assurance that the procedure is technically correct and provides for accomplishment of procedural objectives (including maintenance of proper safety function).

18.2.26 SAFETY RELATED SYSTEM OPERABILITY STATUS (II.K.1.10)

18.2.26.1 Statement of Requirement

Review and modify, as required, procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known. This requirement shall be met by fuel load.

18.2.26.2 Interpretation

None required.

18.2.26.3 Statement of Response

Surveillance testing will be controlled by administrative procedure AD-QA-422. This procedure, which is currently being drafted, will require that surveillance implementing procedures contain a review of redundant component operability prior to removing the system to be tested from service, (if such removal is required by the test), a review of proper system status prior to return of the tested system to service, and provide for notification to Operations of the need for system status changes.

Administrative Procedure AD-QA-306, "System Status and Equipment Control," (see Subsection 18.1.13.3) establishes control of system status as an operations responsibility and will provide the same reviews described above during normal operations and maintenance activities. Maintenance procedures will only cover activities while systems and components are removed from service, the Operations section will actually accomplish changes in system status as controlled by the described Instruction.

18.2.27 TRIP PRESSURIZER LOW-LEVEL COINCIDENT SIGNAL BISTABLES (II.K.1.17)

This requirement is not applicable to Susquehanna SES.

18.2.28 OPERATOR TRAINING FOR PROMPT MANUAL REACTOR TRIP (II.K.1.20)

This requirement is not applicable to Susquehanna SES.

18.2.29 AUTOMATIC SAFETY GRADE ANTICIPATORY TRIP (II.K.1.21)

This requirement is not applicable to Susquehanna SES.

18.2.30 AUXILIARY HEAT REMOVAL SYSTEMS OPERATING PROCEDURES (II.K.1.22)

18.2.30.1 Statement of Requirement

Describe the automatic and manual actions necessary for proper functioning of the auxiliary heat removal systems that are used when the main feedwater system is not operable. This requirement shall be met by fuel load.

18.2.30.2 Interpretation

None required.

18.2.30.3 Statement of Response

A generic response to this requirement was provided by General Electric in NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," (August, 1979) and supplement I. A plant specific description is provided below.

If the main feedwater system is not operable, a reactor scram will be automatically initiated when reactor water level falls to Level 3 (540.5 inches above vessel bottom or 178.2 inches above the top of the active fuel). The operator can then remote manually initiate the reactor core isolation cooling system from the main control room, or the system will be automatically initiated when reactor water level decreases to Level 2 (489.5 inches above vessel bottom or 127.2 inches above the top of the active fuel) due to boil-off. At this point, the high pressure coolant injection system will also automatically start supplying makeup water to the vessel. These systems will continue automatic injection until the reactor water level reaches Level 8 (581.5 inches above vessel bottom or 219.2 inches above top of the active fuel), at which time the high pressure coolant injection turbine and the reactor core isolation cooling turbine are automatically tripped.

In the nonaccident case, the reactor core isolation cooling system is utilized to furnish subsequent makeup water two the reactor pressure vessel. The Reactor core isolation cooling system and the high pressure coolant injection system will restart automatically when the level falls to Level 2 (The reactor core isolation cooling system is being modified to automatically restart, see subsection 18.1.50). No manual actions are required for these systems to restart. Reactor vessel pressure is regulated by the automatic or remote manual operation of the main steam relief valves which blow down to the suppression pool.

To remove decay heat, assuming that the main condenser is not available, the steam condensing mode of the residual heat removal system is initiated by the operator. This involves remote manual alignment of the residual heat removal system valves. If the steam condensing mode is unavailable for any reason, the main steam relief valves can be manually actuated from the control room. Remote manual alignment of the residual heat removal system into the suppression pool cooling mode is then required for suppression pool heat removal. Makeup water to the vessel is still supplied by the reactor core isolation cooling system under manual control.

For the accident case with the reactor pressure vessel at high pressure, the high pressure coolant injection system is utilized to automatically provide the required makeup flow. No manual operations are required since the high pressure coolant injection system will cycle on and off automatically as water level reaches Level 2 and Level 8, respectively. If the high pressure coolant injection system fails under these conditions, the operator can manually depressurize the reactor vessel using the automatic depressurization system to permit the low pressure emergency core cooling systems to provide makeup coolant. Automatic depressurzation will occur if all of the following signals are present: high drywell pressure 1.69 psig, Level 3 water Level permissive, Level 1 water level (398.5 inches above vessel bottom or 36.2 inches above the top of the active fuel), pressure in at least one low pressure injection system and the run out of a 120 second timer (set at 105 seconds) which starts with the coincidence of the other four signals.

18.2.31 REACTOR LEVEL INSTRUMENTATION (II.K.1.23)

18.2.31.1 Statement of Requirement

For boiling water reactors, describe all uses and types of reactor vessel level indication for both automatic and manual initiation of safety systems. Describe other instrumentation that might give the operator the same information on plant status. This requirement shall be met before fuel load.

18.2.31.2 Interpretation

None required.

18.2.31.3 Statement of Response

The response to this requirement was provided by General Electric in NEDO-24708, Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," (August 1979) and Supplement I.

18.2.32 COMMISSION ORDERS ON BABCOCK AND WILCOX PLANTS (II.K.2)

These requirements are not applicable to Susquehanna SES.

18.2.33 REPORTING REQUIREMENTS FOR SAFETY/RELIEF VALVE FAILURES OR CHALLENGES (II.K.3.3)

18.2.33.1 Statement of Requirement

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report. This requirement shall be met before issuance of a full-power license.

18.2.33.2 Interpretation

Prompt reporting to the NRC consists of notification within 24 hours by telephone with confirmation by telegraph, mailgram or facsimile transmission, followed by a written report within 14 days.

The annual operating report has been supplanted by more detailed Monthly Operating Reports. Documentation required to be included in the annual report will be supplied in Monthly Operating Reports.

18.2.33.3 Statement of Response

Subsection 6.9.1.8 of the Technical Specifications will be changed to require prompt reporting with written followup for failures of main steamline Safety/Relief Valves to reclose after actuation. Procedure(s) for reporting of Reportable Occurences are being written addressing Technical Specification requirements.

Subsection 6.9.1.9 of the Technical Specifications currently requires documentation of all challenges to main steamline Safety/Relief Valves to be included in the Monthly Reactor Operating Report. Procedure(s) for preparation and submittal of these monthly reports are being written incorporating this reporting requirement.

18.2.34 PROPORTIONAL INTEGRAL DERIVATIVE CONTROLLER (II.K.3.9)

This requirement is not applicable to Susquehanna SES.

18.2.35 ANTICIPATORY REACTOR TRIP MODIFICATION (II.K.3.10)

This requirement is not applicable to Susquehanna SES.

18.2.36 POWER OPERATED RELIEF VALVE FAILURE RATE (II.K.3.11)

This requirement is not applicable to Susquehanna SES.

18.2.37 ANTICIPATORY REACTOR TRIP ON TURBINE TRIP (II.K.3.12)

This requirement is not applicable to Susquehanna SES.

18.2.38 EMERGENCY PREPAREDNESS-SHORT TERM (III.A.1.1)

18.2.38.1 Statement of Requirement

Comply with Appendix E, "Emergency Facilities," to 10 CFR Part 50, Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," and for the offsite plans, meet essential elements of NUREG-75/111 (Ref. 28) or have a favorable finding from FEMA. This requirement shall be met prior to fuel load.

Provide an emergency response plan in substantial compliance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (which may be modified as a result of public comments solicited in early 1980) except that only a description of and completion schedule for the means for providing prompt notification to the population (App. 3), the staffing for emergencies in addition to that already required

(Table B.1), and an upgraded meteorological program (App. 2) need be provided (Ref. 10). NRC will give substantial weight findings on offsite plans in judging the adequacy against NUREG-0654. Perform an emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations. This requirement shall be met before issuance of a full-power license.

18.2.38.2 Interpretation

PP&L is interpreting Emergency Facilities as encompassing those requirements for TSC, Interim TSC, EOF, Interim EOF, SPDS, OSC as outlined in draft NUREG 0696 and TMI Action Items in 0737. Complete Site, State, County, Township and Municipality Emergency Plans using the Guidelines of NUREG-0654 Rev. 1. Exercise the plans to ensure they are integrated and workable. Comply with meteorological requirements of NUREG 0654 Appendix 2, Rev. 1

18.2.38.3 Statement of Response

Emergency facility design criteria was submitted to NRC for review and approval in February 1981. Interim facility use is defined in the Susquehanna SES Emergency Plan and complies with NUREG 0737 requirements.

Susquehanna SES Emergency Plan Rev. 2 was submitted to the NRC 10/30/80 complying with 10 CFR 50 Appendix E and General Criteria of draft NUREG 0654. Susquehanna SES Emergency Plan Implementing Procedures were submitted to the NRC for review in January 1981. NRC comments have not been received on either submittal. Susquehanna SES Emergency Plan Rev. 3 complying with NRC comment letter and NUREG 0654 Rev. 1 will be submitted subsequent to receipt of NRC comments. The Commonwealth of Pennsylvania Emergency Plan has been submitted to FEMA complying with NUREG 0654 Rev. 1. The County, Township and Municipality Plans are scheduled for submittal to FEMA by 3/16/81.

Susquehanna SES Integrated Emergency Exercise will be held prior to fuel load.

Susquehanna SES Emergency Plan, Rev. 3 and the Susquehanna SES Emergency Plan Implementing Procedures will meet the requirements of NUREG 0654, Appendix 2 and related documents. The computerized dose projection calculations will be based on a straight-line Gaussian model with weather and building wake correction factors included in the methodology. The Emergency Operations Facility will be provided with the computerized dose projection system results in accordance with functional requirements for this facility as specified in draft NUREG 0696.

18.2.39 UPGRADE EMERGENCY SUPPORT FACILITIES (III.A.1.2)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.67 for response.

18.2.40 PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT (III.D.1.1)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.69 for response.

18.2.41 INPLANT RADIATION MONITORING (III.D.3.3)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.70 for response.

18.2.42 CONTROL ROOM HABITABILITY (III.D.3.4)

Requirement superseded by NUREG 0737. Refer to Subsection 18.1.71 for response.

QUESTION 32.27:

The staff's position with regards to post accident monitoring and safe shutdown display instrumentation is stated in Branch Technical Position ICSB 23 in Appendix 7A of the NRC Standard Review Plan. State your conformance or justify your alternatives.

The description contained in Section 7.5.1a.4.2 of the FSAR refers to Post Accident Tracking in only general statements. Discuss the extent to which the monitoring devices and safe shutdown equipment will conform to the position set forth above, and revise the FSAR accordingly. Your response should identify the channels that are recorded and their qualification.

RESPONSE:

Post accident monitoring is discussed in PP&L's updated response to TMI related requirements (PLA-659, N.W.Curtis to B.J. Youngblood dated 3/16/81). PP&L has committed (PLA 722, Curtis to Youngblood dated 4/15/81) to performing a comparison of Susquehanna SES instrumentation with the requirements of Regulatory Guide 1.97, Rev. 2, and will provide justification for exceptions during the third quarter of 1981. 23

QUESTION 032.28:

Identify the equipment which has been environmentally qualified by inference from tests done on similar equipment or previous operating experience and, for each item, provide the basis for the extrapolation in accordance with the requirements of IEEE Std 323-1971.

RESPONSE:

Table 032.28-1 identifies all non-NSSS Class 1E electrical equipment which has been environmentally qualified to IEEE 323 by interence from tests done on similar equipment.

Table 032.28-2 identifies all non-NSSS Class 1E equipment which has been environmentally qualified to IEEE 323 by previous operating experience.

The qualification documentation for this equipment is presently located in the Bechtel home office in San Francisco and is available for NRC audit.

The NSSS equipment information will be provided later.

QUESTION 032.44:

The seismic qualification of indicators and recorders for post accident monitoring which is described in FSAR Section 7.5.1a.4.2.3.3 is unacceptable. It is the staff's position that post accident indicators and recorders must satisfy their minimum performance requirements before and after a seismic event without adjustments or repair. (The staff acknowledges that these electro-mechanical devices may not accurately indicate during severe vibrational excitement). Provide a modified design to conform to the above position or justify the exception taken.

RESPONSE:

Post accident monitoring is discussed in PP&L's updated response to TMI related requirements (PLA-659, N.W. Curtis to B.J. Youngblood dated 3/16/81). PP&L has committed (PLA-722, Curtis to Youngblood dated 4/15/81) to performing a comparison of Susquehanna SES instrumentation with the requirements of Regulatory Guide 1.97, Rev. 2, and will provide justification for exceptions during the third quarter of 1981. 23

QUESTION 032.84

Revise your discussion of safety/relief valve discharge line temperature monitoring to correctly identify the monitoring scheme. If the thermocouples are actually all connected in parallel, provide a discussion of how the open relief valve is identified.

RESPONSE:

Subsection 7.6.1a.4.3.7.2.1 has been amended to show the monitoring scheme for the safety/relief valve discharge line temperature.

Positive value position indication monitors will be addressed in our response to item 2.1.3.a of NUREG-0578.

QUESTION 112.10:

To aid us in our licensing review of Susquehanna, you are requested to provide the following information to us:

1. Describe those actions being taken by you to preclude the occurrence of cracking such as described in IE Bulletin 80-07.

2. Provide a commitment to adopt whatever long-term solution is approved.

3. If you anticipate receiving an Operating License before a long-term solution is agreed upon, describe any short-term actions which you will take to prevent or detect excessive cracking. Provide a rationale as to why these actions are sufficient to justify plant operation until a long-term solution is found.

RESPONSE:

1. General Electric (NED) has undertaken evaluation and testing of the BWR4 beam design. When the results of this study are completed, we will enumerate those step(s) taken to preclude the occurrence of cracking of the jet pump hold-down beams such as described in IE Bulletin 80-07.

2. It is anticipated that there will be alternative long term solutions developed and approved. These approved solutions will be evaluated at the conclusion of current investigative activities and a commitment made at that time.

3. If a long term solution is not agreed upon prior to receipt of an operating license, the following short term actions will be taken to prevent or detect excessive cracking:

   a. A procedure will be instituted to monitor jet pump loop flow/recirculation pump speed ratios daily. A deviation from the normal range may indicate a problem wherein individual jet pump performance data will be used to determine if a problem exists.

   b. A performance monitoring program will be established to obtain and periodically update a "normal" operation data base. Operating data will then be compared to the data base at least weekly to provide an early indication of potential problems. Calibration checks of the instrumentation used in this program will be performed at least every 18 months or more frequently should there be a tendency for significant drift over the 18 month period.

   c. A review of the jet pump operability technical specifications will be performed with the objective of making them more responsive to operating experience. The recommended Technical Specification includes daily monitoring of Recirculation Pumps Flow/Speed Ratio and Jet Pump Loop Flow/Pump Speed Ratio to detect potential problems.

QUESTION 211.8

The SRP 5.4.7 states the residual heat removal system (RHRS) should meet the requirements of General Design Criterion (GDC) 34 of Appendix A to 10 CFR Part 50. The RHR by itself cannot accomplish th eheat removal functions as required by GDC 34. To comply with the single failure criterion the FSAR describes an alternate method of achieving cold shutdown in Section 15.2.9. Insufficient information is provided to allow an adequate evaluation of this alternate method. In particular, we have recently approved Revision 2 to SRP 5.4.7 (containing Branch Technical Position RSB 5-1) which delineates accceptable methods for meeting the single failure criterion. This Branch Technical Position requires testing to demonstrate the expected performance of the alternate method for achieving cold shutdown. You should describe plans to meet this requirement. In addition, we require that all components of the alternate system be safety grade (seismic Category I).

As a result of this requirement, the air supply to the automatic depressurization system (ADS) valves, including the system upstream of the accumulators, must be safety grade. This air supply must be sufficient to account for air consumption necessary for valve operation plus air loss due to system leakage over a prolonged period with loss of offsite power.

RESPONSE

As discussed in Subsection 9.3.1.5.1, the gas supply to the ADS values and the backup gas supply to the ADS accumulators is safety grade. Codes covering the design and construction of these components are discussed in Subsection 9.3.1.5.1.

All components that are a part of the alternate shutdown loop (ECCS pumps and valves, ADS valves, MSIV's, etc.) are routinely tested as required by technical specifications. Testing of the total alternate shutdown system would not provide any additional pertinent information and would result in introducing lower quality (suppression pool) water into the vessel. Bsed on the above, we do not feel that testing of the total loop is necessary or desirable.

This issue was tentatively resolved with the NRC on the Shoreham docket (BWR/4) by an agreement to test one safety relief valve in San Jose simulating the alternate shutdown condition. The rationale for acceptance of this plan was that the SRV is the only component in the loop which has not been demonstrated to be suitable for alternate shutdown conditions. This test was successfully completed in December 1979.

General Electric in conjunction with the Three Mile Island Owners Group is planning further SRV testing in response to TMI related issues. This testing will include conditions similar to the alternate shutdown conditions and will include a valve of Crosby Manufacture as is used in the Susquehanna plant. It is expected that these tests will further confirm that an in-plant test is not required to demonstrate alternate shutdown conditions capability.

QUESTION 211.68:

The analyses presented to show conformance to the ASME Boiler and Pressure Vessel Code for overpressure protection references NEDO-10802 as the analytical model for plant transient evaluation. General Electric has submitted to the staff an updated analytical model (ODYN) to evaluate plant transients. Reanalyze the overpressure sizing transient using the ODYN code unless assurance can be provided that the NEDO-10802 analysis is bounding with regard to predicting peak pressure. The analysis must include the effects of the high pressure recirculation pump trip (RPT) and the turbine stop valve/control valve closure recirculation pump trip where applicable. Provide analysis to justify that the closure of all main steam isolation valves (MSIV) is the most severe overpressure transient when considering the new code, the second safety-grade scram and the effects of RPT.

RESPONSE:

The review of the ODYN code has yet to be completed by the staff. Currently, the staff review has not concluded review of the adequacy of the margin inherent in the proposed ODYN licensing basis. Additionally, it appears further review may be necessary in the area of input parameters. In light of the incomplete nature of the review it is not prudent to reanalyze with the ODYN code at this time.

The ODYN/REDY (NEDO-10802) comparisons performed to date have supported the conservatism of the REDY analysis for this category of events. Additionally, the ODYN code has not been shown to result in any modification of the relative severity of the pressurization events such that the MSIV closure with flux scram is expected to remain the limiting event.

Consideration of the high pressure trip of the recirculation pumps has been considered on a generic basis previously. This is covered in the response to Question 211.4.

Additional discussion of the analytical basis for overpressure protection analyses is provided in the response to Question 211.4.

QUESTION 211.110:

Correct Figure 15A.6-31, "Protection Sequences Main Turbine Trip--Without Bypass:"

(1) For the event to occur at <30% power, protection sequences should be the same as for generator trip without bypass as shown in Figure 15A.6-30.

(2) Delete HPCI that is connected with incident detection circuitry.

Also, confirm that subsequent to initial core cooling the sequence of operations to extended core cooling would be the same as shown in Figure 15A.6-26, "Protection Sequences for Loss of Main Condenser Vacuum."

RESPONSE:

The above corrections to Figure 15A.6-31 will be incorporated in an amendment to the FSAR.

The protection sequence subsequent to initial core cooling to achieve extended core cooling would be the same as indicated on Figure 15A.6-26.

QUESTION 211.118:

For the loss of feedwater heating transient in the manual flow control mode the thermal power monitor (TPM) is used to scram the reactor. Explain the need for the TPM and provide specific transients for which this trip singla initiates caram. Discuss how surveillance testing of the TPM is incorporated in the station technical specifications.

RESPONSE:

The Susquehanna SES plant does not have the thermal power monitor, and hence, was not included in the analysis. See Subsection 15.1.1.3.3 of the FSAR.

QUESTION 211.119:

For the recirc flow control failure with increasing flow transient (Section 15.4.5) provide the initial operating MCPR determined at 65% NB rated power and 60% core flow. In addition, provide the K_f factors as a function of core flow for the automatic and manual flow control modes of operation. Furthermore, provide the maximum flow control set point calibration limit (e.g., 100% or 105% of rated flow) for the recirc loop flow control valves used in the transient analysis.

Provide recirculation pump M-G set points for the manual flow control mode assumed in the analysis. Also, you reference the GE topical report NEDO-10802 as the dynamic model to simulate this event. Since NEDO-10802 does not describe the complete event, discuss in greater detail the overall method used to calculate the CPR.

RESPONSE:

The initial operating MCPR at 65% NB rated power and 60% core flow is 1.23, assuming slow runout to 102.5% of rated flow.

A plot of K_f factors versus core flow is shown on Figure 211.119-1. Note that the M-G set points for the manual flow control mode are shown on the figure; because flow control is provided by M-G sets, there are no flow control values and thus no flow control set points.

The overall method used to calculate the Δ CPR for recirculation pump runout is as follows:

1) The hot channel is set on the MCPR safety limit at the pump runout value (e.g. 102.5%) on the 105% steam flow power-flow line by appropriately changing the radial power distribution.

2) Using the same power distribution, MCPR's are evaluated all along the 105% power-flow line. These MCPR's represent the limits for each particular off-rated condition. Slow runout (i.e. steady-state analysis) is assumed in this calculation since it is conservative with respect to fast runout.

TABLE 211.125

TRANSIENT ANALYSIS RESULTS*

Reactor Cycle	Transient	Exposure Point	Peak Vessel Pressure	CPR
BWR4 251-764 Equil. cycle	Load rejection w/o bypass	Rated EOC (104.2% power)	1235	.17
	(Reduced Feedwater) Heating	Extended EOC (100% power)	1219	0.16
	Feedwater Controller failure	Rated EOC (104.2% power)	1202	0.12
	(Reduced Feedwater) Heating	Extended EOC (100% power)	1060	0.05

* ODYN ANALYSIS RESULTS

QUESTION 211.139:

The response to question 211.113 does not provide sufficient detail on non-safety grade equipment and components which mitigate transients and accidents. Provide a table of the non-safety grade equipment and components assumed to mitigate consequences for each transient and accident in Chapter 15. For those events where non-safety grade systems are used; provide the change in consequences or results when taking credit for safety grade equipment only.

RESPONSE:

The use of non-safety grade equipment for transient analysis is to be an issue which will be addressed in detail by the Licensing Review Group. To enhance interim evaluation a description of the role of non-safety grade equipment is included here. Table 211.139-1 highlights transients which utilize non-safety grade equipment.

It is important to note that the analysis for each of the transients in Table 211.139-1 is based on the single-failure criterion associated with moderate frequency events (i.e., abnormal transients are defined as events which occur as a result of equipment malfunctions as a result of a single active component failure or operator error). Following this single failure, the resulting transient is simulated in a conservative fashion to show the response of primary sytem variables and how the various plant systems would interact and function. In these transients, the consideration of any additional failures is not considered appropriate within the realm of the abnormal transient definition, but shifts them to infrequent events. Although certain transient events assume the operation of specific non-safety grade equipment to provide a realistic transient signature, failures of such equipment would not make these events more thermally or pressure limiting than the limiting accidents already addressed in the FSAR Chapters 5 and 15. In fact, many of the events which have a level 8 turbine trip (a non-safety grade trip) would be less severe if the level 8 trip were assumed not to function.

Failure of the relief valve function of the safety-relief system for any event will not result in a transient which exceeds the peak pressure response of the limiting event presented in Chapter 5.0. Failure of the level 8 turbine trip of failure of the bypass to open when the level 8 trip does occur were studied for a BWR similar to the Susquehanna design. The increase in ΔCPR was about 0.02 for a delay in the turbine trip and 0.08 for failure of bypass. Although thermal margins are reduced, no significant (if any) fuel damage is expected. The offsite doses (if any) would be negligible, and therefore no impact from a health and safety viewpoint. The loss of feedwater event is analytically about the same with or without the recirculation runback ahead of the level 2 trip. In summary, the thermal and pressure safety limits are not compromised by inclusion of the simulated response of non-safety grade systems.

Table 211.139-1 shows thich non-safety grade systems or components were assumed to actuate in the FSAR analysis.

QUESTION 211.140:

The analysis of transients and accidents in Chapter 15.0 does not state which of the RPS time response delays in Table 7.2-5 is used in the REDY computer code model (NEDO-10802). For each transient and accident in Chapter 15.0, specify whether the sensor or overall delay time is used in the analysis and why the specified delay time is conservative.

RESPONSE:

In all Chapter 15 events, the "Maximum Overall Time Delay" of Table 7.2-5 is utilized for each scram encountered and reported in each event scenario. This allows for maximum specified sensor and logic delays.

QUESTION 211.175:

For the "failure of RHR shutdown cooling" transient, the FSAR considers alternate shutdown cooling methods in the event the residual heat removal (RHR) system in the suction line may not be used because of valve failure. In the analysis, valves in the automatic depressurization system (ADS) were used to transfer fluid (steam, water or a combination of these) from the reactor vessel to the suppression pool. The RHR system removes the added heat by removing cooling water from the suppression pool and injecting it into the reactor vessel. We require that you perform a test or cite previous test results to demonstrate that the ADS valves can discharge the fluid under the most limiting conditions when the fluid is all water. Show that the alternate method is a viable means of shutdown cooling by comparing the system hydraulic losses with the available pump head. Hydraulic losses should be provided for each system component and, wherever possible, should be derived from experimental results.

RESPONSE:

In response to NUREG 0578, requirement 2.1.2, the BWR Owners Group has initiated tests to demonstrate the ADS valves are qualified for a spectrum of conditions. These tests will include a low pressure liquid test to simulate alternate shutdown viability of the ADS valves. Valve hydraulic losses will be determined as a result of those tests. Therefore, system hydraulic losses can be readily compared to available pump head. The BWR Owners Group has committed to the NRC to have the tests completed by July of 1981.

QUESTION 211.192:

Since the initial discovery of cracking in boiling water reactor (BWR) control rod drive return line (CRDRL) nozzles, General Electric (GE) has proposed a number of solutions to the problem. One solution GE has proposed is a system modification that involves total removal of the CRDRL and cutting and capping of the CRDRL nozzle. It appears from your response to 211.7 that SSES plans this modification.

The staff asked for more information on the impact of this modification on your plant and also required a SSES commitment to preoperational testing to verify performance of the modified CRD system in Question 211.43. When you respond to 211.43, you should address the applicable items and staff concerns specified in the letter from D. Eisenhut, NRC, to R. Gridley, GE, dated January 28, 1980, on the subject of control rod drive return line (CRDRL) removal and capping CRDRL nozzles.

RESPONSE:

The referenced letter from D. Eisenhut, NRC, to R. Gridley, GE, dated January 28, 1980, essentially documents the NRC position on the CRD return line deletion. Pages 3 and 4 of that letter provide a summary of the NRC conclusions on this subject. In their final conclusion, 251" BWR/4 plants (such as Susquehanna) are accepted for return deletion - contingent upon the Utility performing some demonstration tests (These tests will be performed as part of normal performance and preoperational testing). The second and third conclusions do not pertain to the Susquehanna design. The fourth conclusion places the requirements for the installation of the GE recommended pressure equalizing valves between the cooling water and exhaust water headers, the installation of flush ports on carbon steel exhaust water headers, and the replacement of any carbon steel pipe in the flow stablizer loop. Referring to the CRD system P&ID (Figures 4.6-5a and 4.6-5b) for Susquehanna, all these requirements are met in the GE designed system - redundant pressure equalizing valves are installed between the cooling water and exhaust water headers; the exhaust water header is constructed of stainless steel and therefore does not require flush ports; and there is no carbon steel pipe in the CRD system downstream of the main drivewater filters. The fifth NRC conclusion requires the Utility to develop procedures for optimizing the CRD system flow to the reactor pressure vessel. The sixth conclusion is not applicable to the CRD system design for Susquehanna.

QUESTION 211.228:

The response to Questions 211.13 and 211.105 require additional clarification. Reference was made to another BWR/4 with LPCI modification (Shoreham) and the results of an analysis for LPCI diversion at Shoreham was identified as applicable to Susquehanna. Does Susquehanna have an interlock similar to that at Shoreham which would prevent LPCI diversion prior to reflooding the reactor core to the 2/3 level? If not, justify the use of the Shoreham analysis for LPCI diversion at Susquehanna.

Describe operator requirements to activate LPCI diversion. Can the diverted LPCI loop be returned to provide additional core flooding, if required? What instructions, if any, are provided to the operator to ensure that the operable LPCI loop is not prematurely diverted to containment cooling in the event that one LPCI loop is disabled?

RESPONSE:

The Susquehanna Plant, unlike the Shoreham Plant has no level interlock on the LPCI diversion logic. However, for the Shoreham LPCI diversion analysis no credit was taken for the level interlock device. In that analysis LPCI diversion was always assumed to occur at 10 minutes subsequent to the LOCA initiation signals. Both Susquehanna and Shoreham are BWR/4 plants with LPCI modification and thus have the same complement of ECC systems. Therefore, the Shoreham LPCI diversion analysis results are representative of the expected results for Susquehanna.

Before the LPCI flow can be diverted to either the pool cooling mode or containment spray (wetwell/drywell) mode, the operator has to close the LPCI throttling valve (F017) and then initiate the "manual" switch of the desired diversion mode and open the appropriate valve.

In order to return the diverted LPCI loop to provide additional core flooding, the operator merely needs to close the diversion valve, and manually open the LPCI throttling valve.

Instructions to the operator ensuring that the LPCI flow is not prematurely diverted to the other modes are contained in the "Emergency Procedures Guidelines" currently being written by General Electric for the utilities. Extracts of the guidelines pertinent to LPCI diversion are given below:

- do not secure an ECCS unless there are at least two independent indications that adequate core cooling is assured.

- do not divert RHR pumps from the LPCI mode unless adequate core cooling is assured.

In addition, the Operation and Maintenance Instructions for the RHR system specifies that when the water level in the reactor has been restored to the two-thirds level, and if the drywell pressure has increased to at least 2 psig, only then can the operator make use of the containment spray/cooling operation to depressurize the drywell and/or cool the suppression pool water.

There are no interlocks on the LPCI other than those described above that would prevent the operator from diverting LPCI to drywell or containment sprays. However, in the short term following a loss-of-coolant accident, the operators primary concern will be assuring adequate inventory in the core. In addition, from FSAR Tables 6.2-1 and 6.2-5 and Figure 6.2-2, it is seen that the peak containment pressure for the recirculation line break is well below the design limit with no credit for containment sprays. Consequently, there would be no reason for the operator to divert LPCI to containment spray, since no violation of containment design pressure limits would occur anyway.

QUESTION 211.272

It is indicated that the "pressure regulator-closed" transient with failure of the backup pressure regulator is less severe than the "turbine trip with bypass" transient in Section 15.2.3. This agrees with GESSAR 238. As a result, only a qualitative evaluation of the transient was provided. However, quantitative results from the Grand Gulf FSAR indicate the opposite. The staff's concern is that quantitative results for this transient may be similar to those for Grand Gulf. Provide a quantitative analysis of the "pressure regulator-closed" transient with the ODYN mode assuming failure of the backup pressure regulator and revise Section 15.2.1.2.3 accordingly. This request should be coordinated with the ODYN request via the 211.160 question.

RESPONSE:

This transient will be included in the ODYN reanalysis for Susquehanna.

QUESTION 331.2:

Section 12.3.4 of Regulatory Guide 1.70, Revision 2 calls for information on the sensitivity of airborne radioactivity monitoring systems.

Describe how your continuous airborne readioactivity systems will provide adequate coverage of general areas, rooms, and corridors which have a possibility of containing airborne radioactivity and which may be occupied by personnel. In order to provide adequate coverage, the systems must be capable of detecting ten MPC -hours of airborne particulate and iodine radioactivity.

RESPONSE:

The combination of the airborne radioactivity monitoring system in conjunction with administrative controls restricting and limiting personnel access, standard health physics practices, ventilation flow patterns throughout the plant, plant equipment layout, lack of significant radioactive airborne sources in normally occupied areas (radiation zones I and II), and administrative control of access into applicable radiation and high radiation areas is sufficient to ensure that airborne radioactivity levels are safe in terms of the required duration of personnel access. A general review of these concepts follows:

a. Significant releases of airborne radioactive materials within the plant will be detected by effluent monitors as described in Table 11.5-1 in Section 11.5 of the FSAR. Air flow patterns are normally from occupied areas to non-occupied areas and from low airborne radioactive material areas to high airborne radioactive material areas. Readouts and alarms are located locally and in the control room.

b. Additional Continuous Air Monitors (CAMS) with local readout and alarm will normally be located in the following areas:

   818' elevation - Reactor Building/central

   749' elevation - Reactor Building/outside RWCU pump room

   719' elevation - Reactor Building/outside personnel access hatch to drywell

   676' elevation - Rad Waste Building/outside control room

   676' elevation - Turbine Building/near reactor feed pumps

   729' elevation - Reactor Building Control Room

   729' elevation - Turbine Deck/outside control structure

The particulate and iodine channels shall be capable of detecting at least 4×10^{-10} μci/cc and 5×10^{09} μci/cc respectively.

c. Administrative controls for limiting exposure to airborne radioactivity concentrations greater than 10 MPC-HRS. are as follows:

1. Routine airborne radioactivity surveys of various accessible radiation zones within the plant. The routine monitoring schedule and frequency will be delineated in Station Health Physics Procedures. These locations may be modified with consideration of plant operating status.

2. Access to airborne radioactivity areas with concentrations greater than 25% of the applicable MPC and/or MPC mixture will normally be controlled via a Radiation Work Permit (RWP). Entry into and/or work in the area will be preceded by a survey sufficient to determine the radiological conditions present and protection for these conditions will be specified on the RWP.

3. Access to areas where the potential for high radioactive airborne concentration exist due to work conditions shall be controlled via the RWP process.

QUESTION 331.7

Provide a breakdown of the activities which are included in the total 237.7 man-rem/unit for routine maintenance. Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light Water Reactor Plants Design Stage Man-Rem Estimates," which has been published for comment will provide further guidance.

RESPONSE

The method used to estimate the routine maintenance dose for Susquehanna is discussed in revised Subsection 12.4.1.3.2.

QUESTION 331.11:

You have not described an in-plant accident radiation and airborne radioactivity monitoring system as required in our standard review plan.

It is our position that the in-plant accident radiation and airborne radioactivity monitoring systems should provide personnel with the capability to assess the radiation hazard in areas which may need to be accessed during the course of an accident. The accident monitoring systems may include the normal area radiation monitors, airborne radioactivity monitors, and portable radiation monitoring equipment. The accident monitoring systems should have a usable range which includes the maximum calculated accident levels, and they should be designed to operate properly in the environment caused by the accident. Describe your accident monitoring systems, and describe how your systems will meet this position.

RESPONSE:

The response to this question will be included in Pennsylvania Power & Light Company's response to NUREG-0578.

QUESTION 331.14:

Provide results of design review for additional shielding required to provide access to vital areas and protect safety equipment after a core degradation accident.

RESPONSE:

At the present time, a study on the requirements for additional shielding to provide access to vital areas and protect safety equipment after a core degradation accident is being performed. The results of this study will be part of Pennsylvaia Power and Light Company's response to NUREG-0578.

QUESTION 331.15:

Provide a description of the radiation protection features incorporated in the system for sampling and analyzing reactor coolant and containment atmosphere after a core degradation accident.

RESPONSE:

The system for sampling and analyzing reactor coolant and containment atmosphere after a core degradation accident is being designed. This system will be discussed in Pennsylvania Power & Light Company's response to NUREG-0578.

Question 331.16

Table 13.1-2 of your FSAR states that you will fill the position of Health Physics Supervisor 90 days prior to fuel loading. In Section 12.5.1.4 your have committed that the individual filling the Health Physics Supervisor will meet the criteria for Radiation Protection Manager in Regulatory Guide 1.8. Your technical specifications will also require such qualifications. You should provide a resume of the education, training, and experience of the individual selected to fill this position as soon as it is available.

Response:

The resume of the acting Health Physics Supervisor is provided in Table 331.16-1. The qualifications of this individual satisfy the criteria for Radiation Protection Manager in Regulatory Guide 1.8. The position of Health Physics Supervisor will be filled 90 days prior to fuel load.

QUESTION 371.6

Discuss the implications of the September 30, 1976 amendment to 18 CFR Part 803, that requires compensation for water withdrawn from the Susquehanna River during periods of low flow.

(a) How do you intend to comply with the regulation?

(b) Will the plant have to be shut down during low river flow? How often will the UHS pond have to be used to comply with the regulation?

RESPONSE

The applicant intends to comply with the 9/30/76 amendment to 18 CFR Part 803 by either supplying replacement water to the river from an augmentation reservoir or purchasing augmentation water from another source during low flow conditions. The applicant has pursued both alternatives since 1974, however the additional water may not be available in either case in 1981 for Unit 1 operation.

The applicant filed a formal request with the Baltimore District, Corps of Engineers, to purchase water supply storage in the Cowaneaque Reservoir.

The Corps of Engineers has subsequently received a Susquehanna River Basin Commission (SRBC) concurrence that it would be appropriate for the Corps to make a study to determine the acceptability of the request.

The applicant, as a backup, is preparing an environmental report to be submitted to the lead review agency detailing the environmental analysis for the Pond Hill site augmentation Reservoir. This is expected to be completed and ready for submittal by year's end.

If neither of the options becomes available and there are periods of low flow and we are not able to provide augmentation water by another acceptaBle means and are required to shut down then the results of this shutdown can be seen in Table 371.6-1. Table 371.6-1 Impact of Low Flow on Susquehanan Operations details the three cases cited above and the subsequent reduction of capacity factor should the water not be available.

The USGS has estimated the Q7-10 at Wilkes-Barre (the closest upstream location where river flows have been measured) to be 770CFS. Average consumptive use of the Susquehanna SES is 50CFS which makes the Susquehanna SES low flow criterion 820CFS. Analysis of historical river (1905-1975) flow data based on a TAMS *report of May 26, 1978 shows that the river will be below this (820CFS) rate an average of four (4) days per year.

Further statistical analysis shows the maximum duration recorded to be ninety-six (96) days with a frequency of occurrence of once in 158 years. The particular low flow duration that has a frequency of once in 30 years (approximate plant life) is twenty-eight (28) days.

The Ultimate Heat Sink (UHS) will not be used to comply with this regulation.

*Reference: Availability of Susquehanna River Flow, TAMS May 26, 1978.

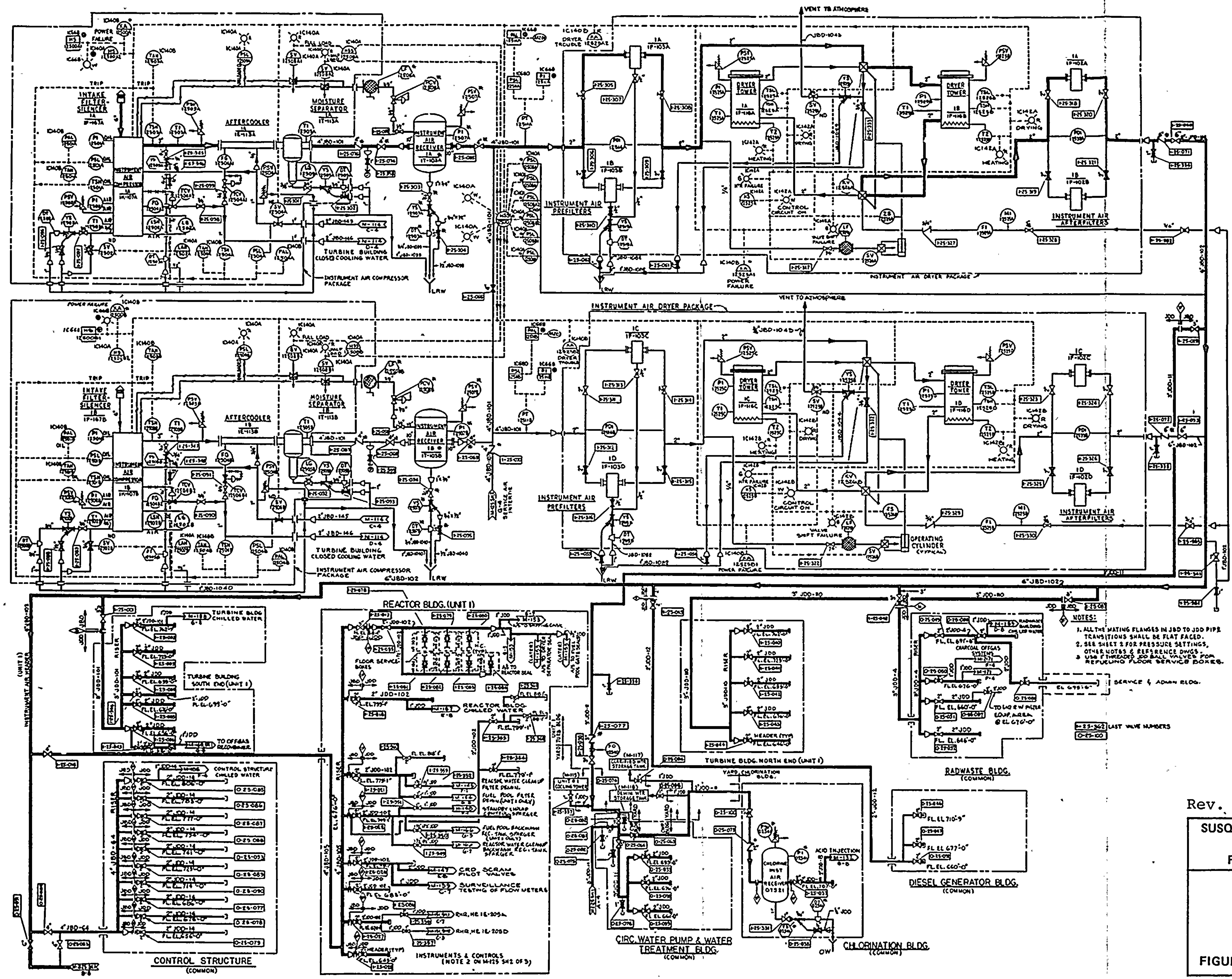
QUESTION 422.3:

Please provide the qualifications of the person filling the position of Quality Supervisor.

RESPONSE:

As described in Subsection 17.2.1.1.1.4.1.1, the Quality Supervisor meets the requirements set forth in Section 4.4.5 of ANSI/ANS 3.1-1978.

The resume of the Susquehanna Quality Supervisor has been provided in Table 13.1-4.



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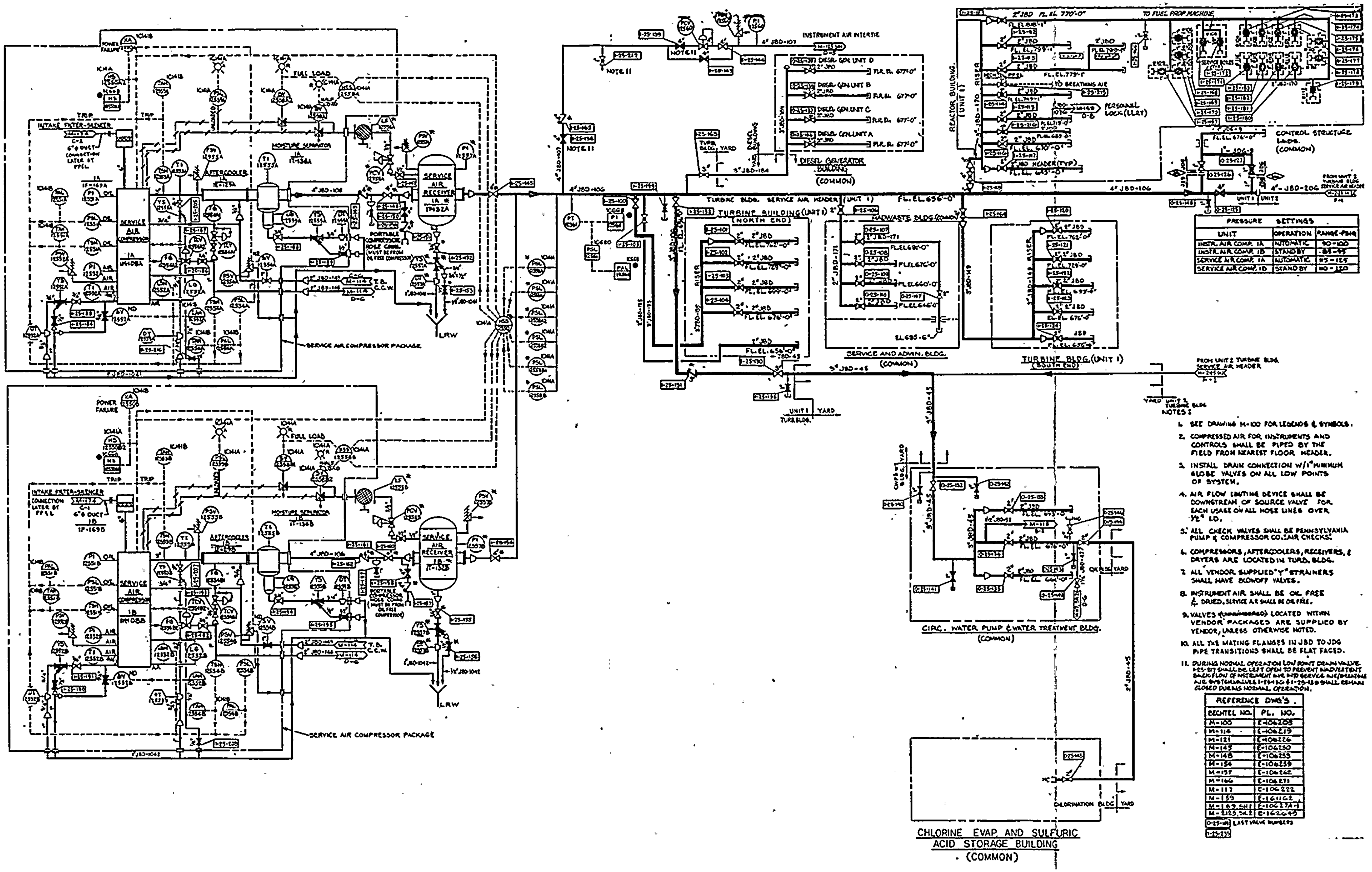
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

P&ID
COMPRESSED AIR SYSTEM
(INSTRUMENT AIR)

FIGURE 9.3–1



PRESSURE SETTINGS		
UNIT	OPERATION	RANGE-PSIG
INSTR. AIR COMP. 1A	AUTOMATIC	90-100
INSTR. AIR COMP. 1B	STANDBY	85-95
SERVICE AIR COMP. 1A	AUTOMATIC	115-125
SERVICE AIR COMP. 1B	STANDBY	110-120

NOTES:

1. SEE DRAWING M-100 FOR LEGENDS & SYMBOLS.
2. COMPRESSED AIR FOR INSTRUMENTS AND CONTROLS SHALL BE PIPED BY THE FIELD FROM NEAREST FLOOR HEADER.
3. INSTALL DRAIN CONNECTION W/1" MINIMUM GLOBE VALVES ON ALL LOW POINTS OF SYSTEM.
4. AIR FLOW LIMITING DEVICE SHALL BE DOWNSTREAM OF SOURCE VALVE FOR EACH USAGE ON ALL HOSE LINES OVER 1/2" I.D.
5. ALL CHECK VALVES SHALL BE PENNSYLVANIA PUMP & COMPRESSOR CO. AIR CHECKS.
6. COMPRESSORS, AFTERCOOLERS, RECEIVERS, & DRYERS ARE LOCATED IN TURB. BLDG.
7. ALL VENDOR SUPPLIED "Y" STRAINERS SHALL HAVE BLOWOFF VALVES.
8. INSTRUMENT AIR SHALL BE OIL FREE & DRIED. SERVICE AIR SHALL BE OIL FREE.
9. VALVES (UNNUMBERED) LOCATED WITHIN VENDOR PACKAGES ARE SUPPLIED BY VENDOR, UNLESS OTHERWISE NOTED.
10. ALL THE MATING FLANGES IN JBD TO JDG PIPE TRANSITIONS SHALL BE FLAT FACED.
11. DURING NORMAL OPERATION LOW POINT DRAIN VALVE 1-25-217 SHALL BE LEFT OPEN TO PREVENT INADVERTENT BACK FLOW OF INSTRUMENT AIR INTO SERVICE AIR/BREATHING AIR SYSTEM. VALVES 1-25-136 & 1-25-139 SHALL REMAIN CLOSED DURING NORMAL OPERATION.

REFERENCE DWG'S.	
BECHTEL NO.	PL. NO.
M-100	E-106205
M-114	E-106219
M-121	E-106226
M-145	E-106250
M-148	E-106253
M-154	E-106259
M-157	E-106262
M-166	E-106271
M-117	E-106222
M-159	E-161162
M-169, SH.1	E-106274-1
M-2125, SH.2	E-162645

0-25-181 LAST VALVE NUMBERS
1-25-232

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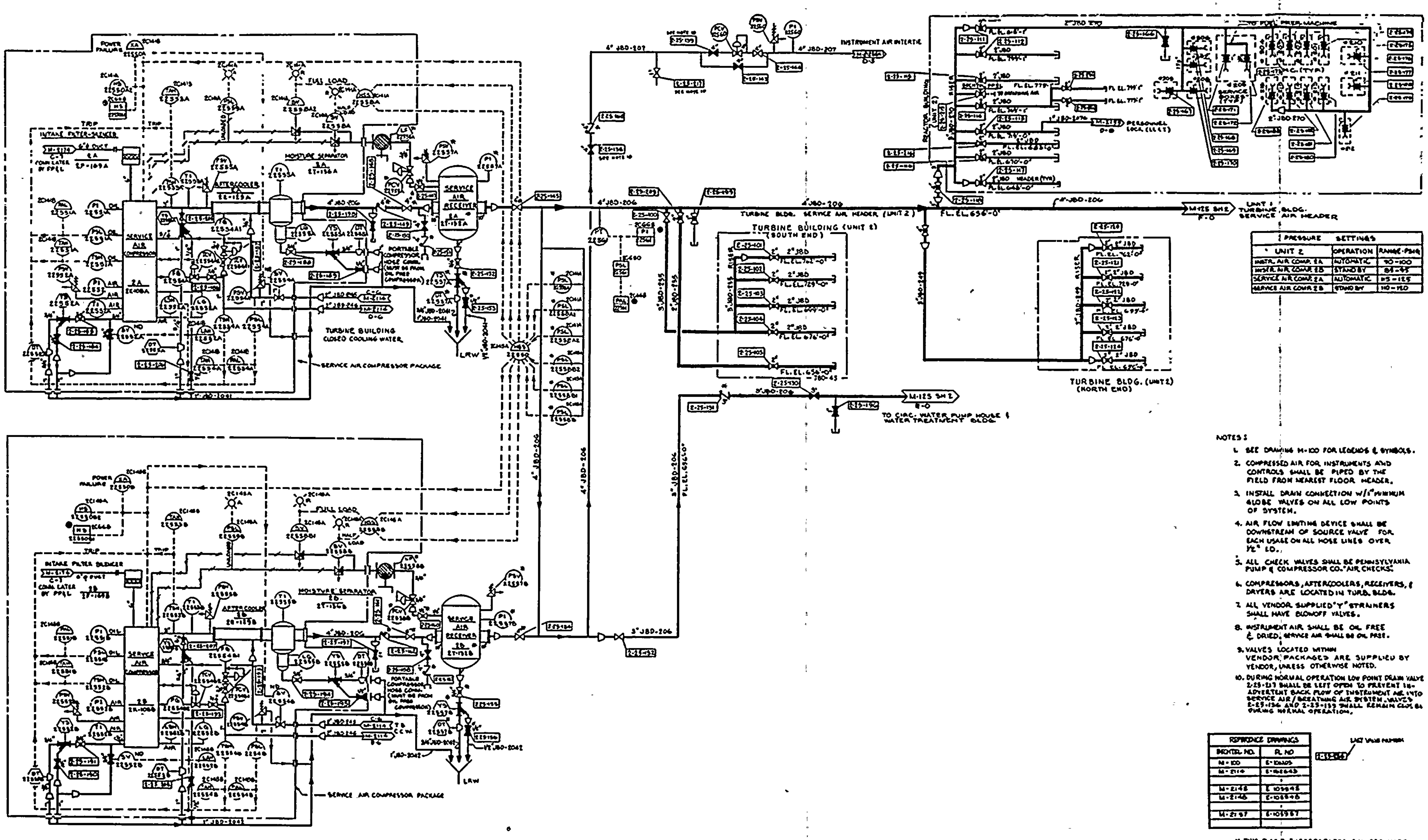
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

P&ID
COMPRESSED AIR SYSTEM

FIGURE 9.3-2



PRESSURE SETTINGS		
UNIT 2	OPERATION	RANGE-PSIG
INSTR. AIR COMP. 2A	AUTOMATIC	90-100
INSTR. AIR COMP. 2B	STANDBY	85-95
SERVICE AIR COMP. 2A	AUTOMATIC	115-125
SERVICE AIR COMP. 2B	STANDBY	110-120

NOTES:

1. SEE DRAWING M-100 FOR LEGENDS & SYMBOLS.
2. COMPRESSED AIR FOR INSTRUMENTS AND CONTROLS SHALL BE PIPED BY THE FIELD FROM NEAREST FLOOR HEADER.
3. INSTALL DRAIN CONNECTION W/1" MINIMUM GLOBE VALVES ON ALL LOW POINTS OF SYSTEM.
4. AIR FLOW LIMITING DEVICE SHALL BE DOWNSTREAM OF SOURCE VALVE FOR EACH USAGE ON ALL HOSE LINES OVER 1/2" I.D.
5. ALL CHECK VALVES SHALL BE PENNSYLVANIA PUMP & COMPRESSOR CO. "AIR CHECKS."
6. COMPRESSORS, AFTERCOOLERS, RECEIVERS, & DRYERS ARE LOCATED IN TURB. BLDG.
7. ALL VENDOR SUPPLIED "Y" STRAINERS SHALL HAVE BLOWOFF VALVES.
8. INSTRUMENT AIR SHALL BE OIL FREE & DRIED. SERVICE AIR SHALL BE OIL FREE.
9. VALVES LOCATED WITHIN VENDOR PACKAGES ARE SUPPLIED BY VENDOR, UNLESS OTHERWISE NOTED.
10. DURING NORMAL OPERATION LOW POINT DRAIN VALVE 2-25-127 SHALL BE LEFT OPEN TO PREVENT INADVERTENT BACK FLOW OF INSTRUMENT AIR INTO SERVICE AIR/BREATHING AIR SYSTEM. VALVES 2-25-136 AND 2-25-139 SHALL REMAIN CLOSED DURING NORMAL OPERATION.

REFERENCE DRAWINGS	
BECHTEL NO.	PL NO
M-100	E-10405
M-2114	E-162645
M-2148	E-105945
M-2146	E-105946
M-2157	E-105967

LAST VALVE NUMBER [illegible]

11. THIS P&I.D. INCORPORATES CHANGES MADE ON THE CORRESPONDING UNIT 1 P&ID UP TO REVISION 12.

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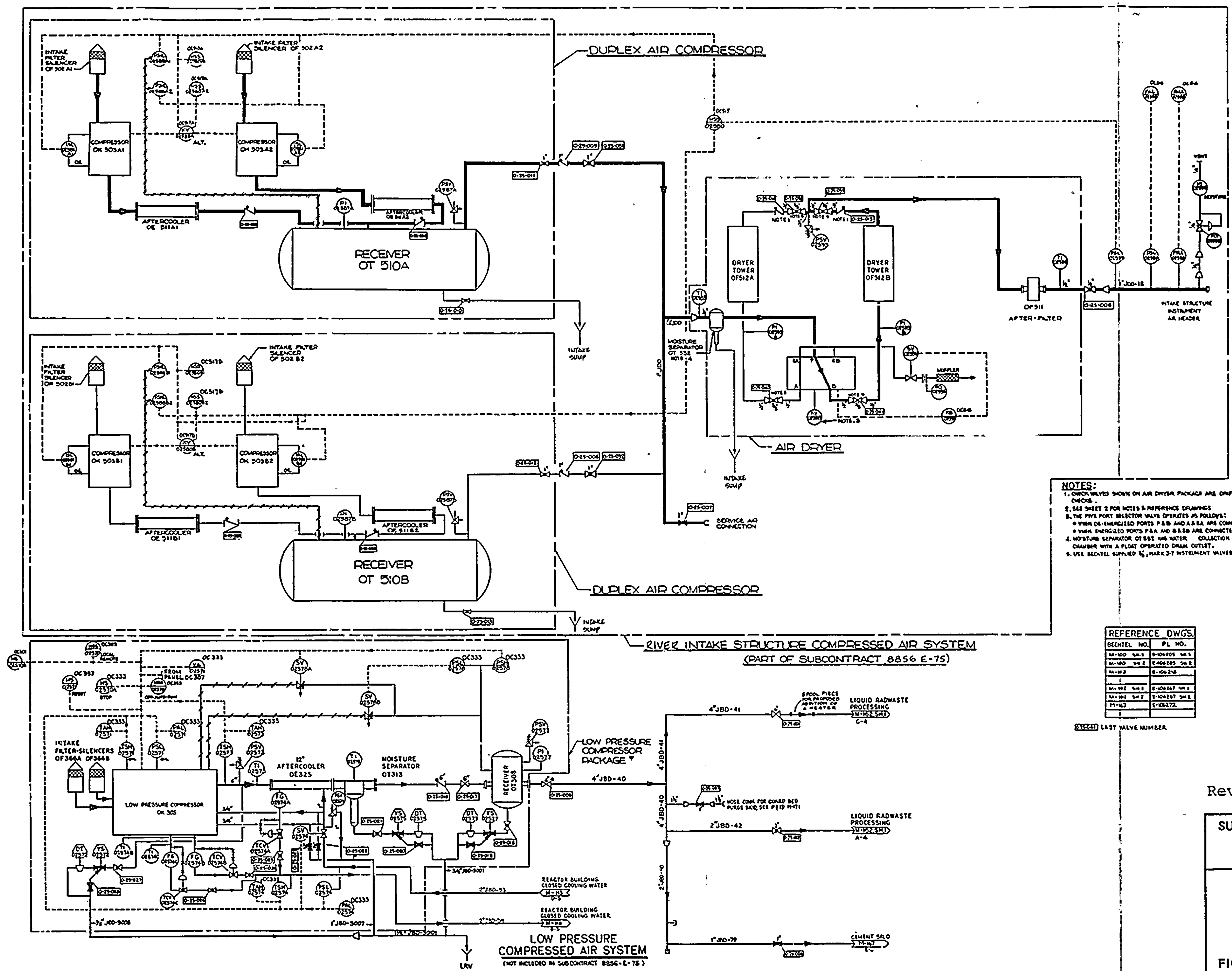
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

P&ID
UNIT 2
COMPRESSED AIR SYSTEM
(SERVICE AIR)

FIGURE 9.3–3b



NOTES:

1. CHECK VALVES SHOWN ON AIR DRYER PACKAGE ARE ORIFICE CHECKS.
2. SEE SHEET 2 FOR NOTES & REFERENCE DRAWINGS
3. THE FIVE PORT SELECTOR VALVE OPERATES AS FOLLOWS:
   - WHEN DE-ENERGIZED PORTS P & B AND A & EA ARE CONNECTED.
   - WHEN ENERGIZED PORTS P & A AND B & EB ARE CONNECTED
4. MOISTURE SEPARATOR OT 552 HAS WATER COLLECTION CHAMBER WITH A FLOAT OPERATED DRAIN OUTLET.
5. USE BECHTEL SUPPLIED ½", MARK 3-7 INSTRUMENT VALVES.

REFERENCE DWG'S.

BECHTEL NO.	PL NO.
M-100 SH.1	E-106205 SH.1
M-100 SH.2	E-106205 SH.2
M-113	E-106218
M-162 SH.1	E-106267 SH.1
M-162 SH.2	E-106267 SH.2
M-167	E-106272

0-25-041 LAST VALVE NUMBER

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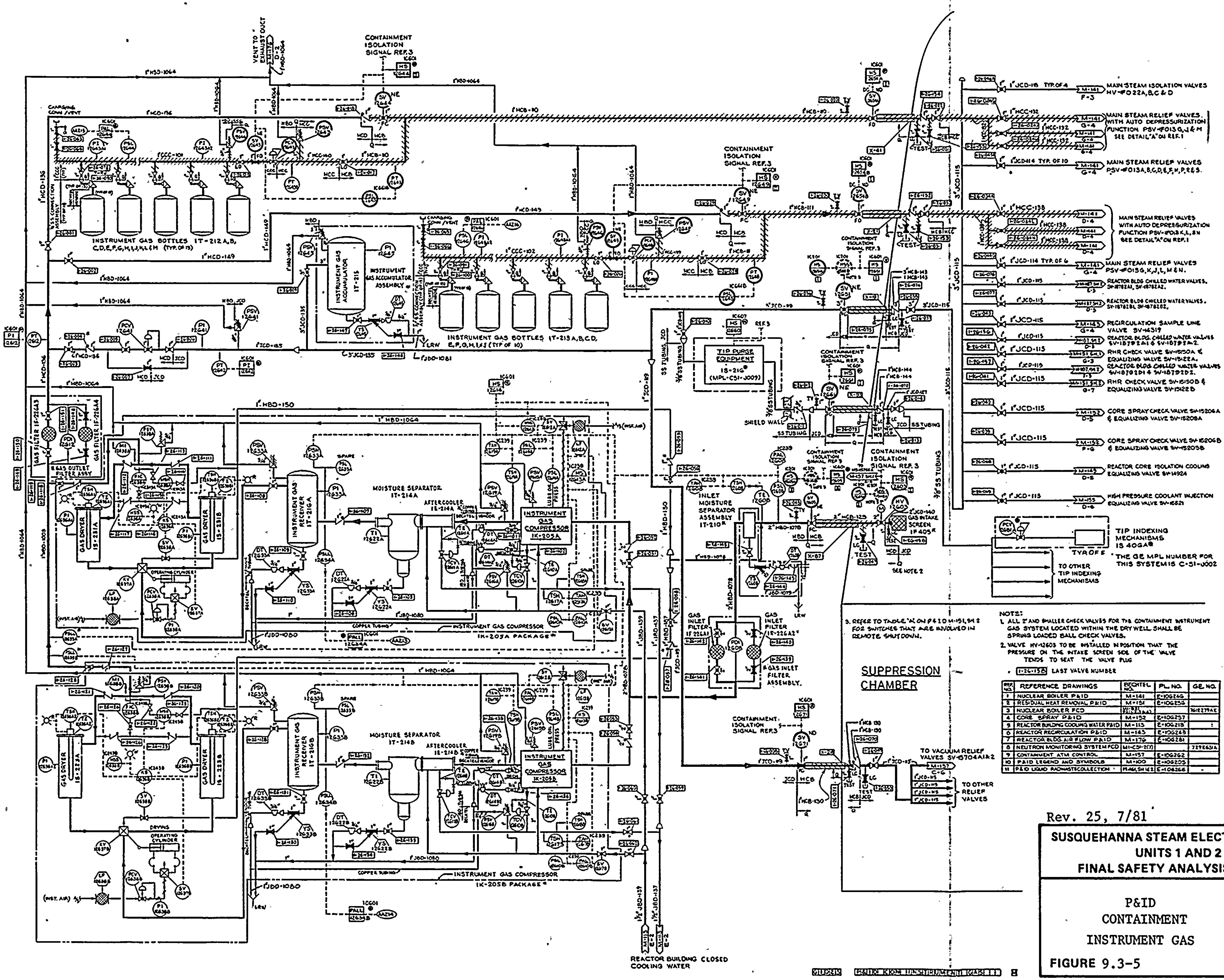
P&ID
COMPRESSED AIR
SYSTEM

FIGURE 9.3–4



3. REFER TO TABLE 'A' ON P&ID M-151, SH 2 FOR SWITCHES THAT ARE INVOLVED IN REMOTE SHUTDOWN.

NOTE:

1. ALL 2" AND SMALLER CHECK VALVES FOR THE CONTAINMENT INSTRUMENT GAS SYSTEM LOCATED WITHIN THE DRY WELL SHALL BE SPRING LOADED BALL CHECK VALVES.
2. VALVE HV-12603 TO BE INSTALLED IN POSITION THAT THE PRESSURE ON THE INTAKE SCREEN SIDE OF THE VALVE TENDS TO SEAT THE VALVE PLUG

• 1-26-158 LAST VALVE NUMBER

REF. NO.	REFERENCE DRAWINGS	BECHTEL NO.	PL. NO.	GE. NO.
1	NUCLEAR BOILER P&ID	M-141	E-106246	
2	RESIDUAL HEAT REMOVAL P&ID	M-151	E-106255	
3	NUCLEAR BOILER FCD	[illegible]		761E279AE
4	CORE SPRAY P&ID	M-152	E-106257	
5	REACTOR BUILDING COOLING WATER P&ID	M-113	E-106218	
6	REACTOR RECIRCULATION P&ID	M-143	E-106248	
7	REACTOR BLDG. AIR FLOW P&ID	M-176	E-106281	
8	NEUTRON MONITORING SYSTEM FCD	M1-C51-2(7)		729E651A
9	CONTAINMENT ATM CONTROL	M-157	E-106262	
10	P&ID LEGEND AND SYMBOLS	M-100	E-106205	
11	P&ID LIQUID RADWASTE COLLECTION	M-161, SH 1&2	E-106266	

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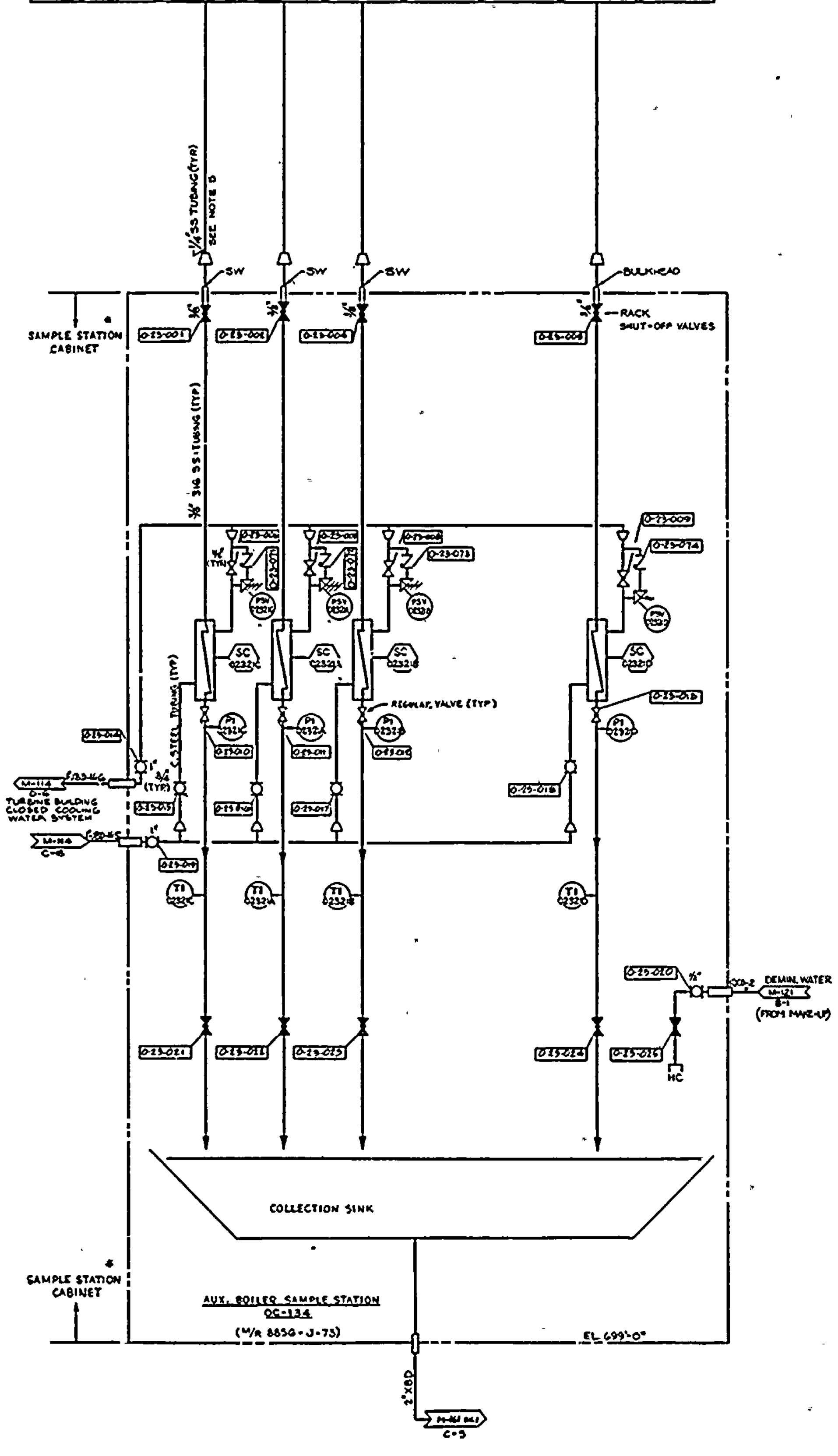
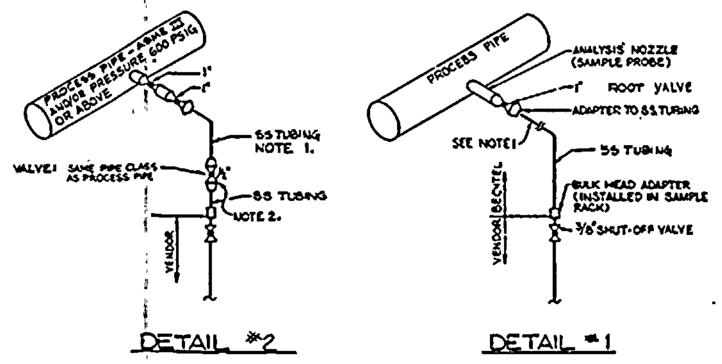
P&ID
CONTAINMENT
INSTRUMENT GAS

FIGURE 9.3-5

SERVICE	AUX. BOILER STEAM	AUX. BOILER "A" BLOWDOWN	AUX. BOILER "B" BLOWDOWN			AUX. BOILER FEED PUMP	
P&ID LOCATION	M-121 G-3	M-121 D-4	M-121 D-4			M-121 B-7	
ANALYSIS NOZZLE NO.	AN-02102A,B	—	—			AN-02107	
PIPE CLASS AND LINE NO. DATA	6"-GBD-2	1"-GBD	1"-GBD			3"-GBD-1	
PRESS. PSIG MAX. / NORM.	250 / 250	270 / 250	270 / 250			280 / 280	
TEMP. °F MAX. / NORM.	406 / 406	408 / 406	408 / 406			212 / 212	





NOTES:

1. TUBING FROM ROOT VALVE TO THE SAMPLE STATION CABINET SHALL BE IN ACCORDANCE WITH PIPING CLASS SPECIFICATION JG-100
2. ALL LINES WITH A PRIMARY PRESSURE RATING OF 600 PSIG AND HIGHER OR CLASSIFIED ASME III, CLASS 3 SHALL HAVE AN ADDITIONAL ISOLATION VALVE (SEE DETAIL #2 ABOVE) LOCATED ABOVE SAMPLE STATION.
3. FOR INSTALLATION OF ANALYSIS NOZZLES REFERENCE DWG. 8856 M-199 SHEETS 606 & 605
4. DEVICES SHOWN WITH A LINE IN THE BALLOON ARE LOCATED IN THE INSTRUMENT SECTION OF THE SAMPLE STATIONS.
5. ALL 1/2" SS TUBING BETWEEN PROCESS LINES AND SAMPLE STATIONS SHALL HAVE .065" WALL THICKNESS.
6. LAST VALVE NO. 0-23-074
   1-23-234

REFERENCE DWGS	
BECHTEL NO.	P.L. NO.
M-121	E-106226
M-114	E-106219
M-161	E 106266-1

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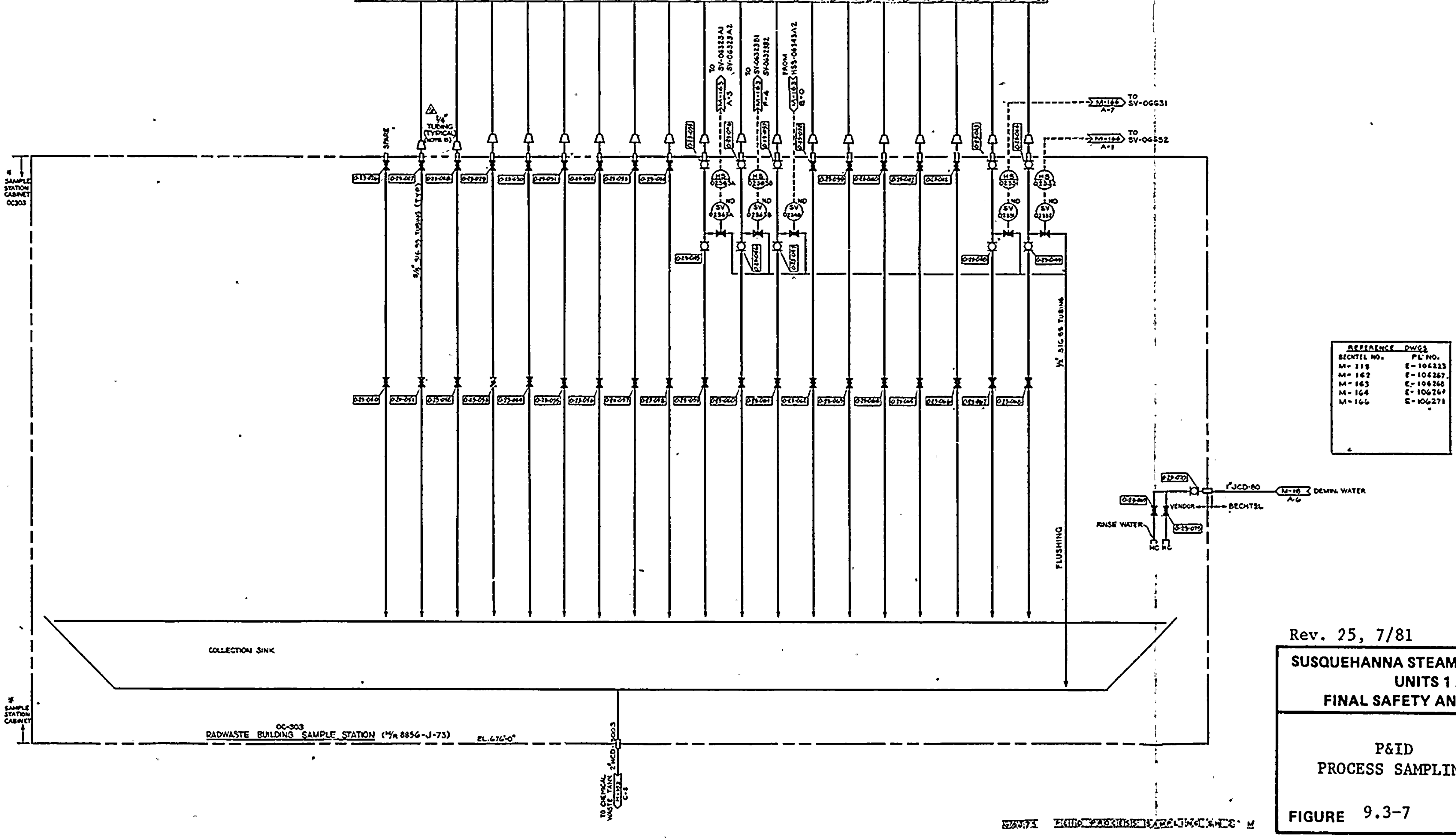
P&ID
PROCESS SAMPLING

FIGURE 9.3-6



SERVICE	LIQUID RADWASTE COLLECTION TANKS A & B	LIQUID RADWASTE COLLECTION TANKS C & D	LIQUID RADWASTE COLLECTION TANKS E & F	LIQUID RADWASTE SAMPLE TANKS A & B	LIQUID RADWASTE SAMPLE TANKS C & D	LIQUID RADWASTE SAMPLE TANKS E & F	LIQUID RADWASTE SURGE TANKS	CHEMICAL WASTE TANK	EVAP. CONC. DISCHARGE "A"	EVAP. CONC. DISCHARGE "B"
P & ID LOCATION	M-162 SHT 1 E-6	M-162 SHT 1 D-6	M-162 SHT 1 B-6	M-162 SHT 1 F-1	M-162 SHT 1 D-1	M-162 SHT 1 B-1	M-162 SHT 2 E-4	M-163 B-5	M-163 A-3	M-163 G-4
ANALYSIS NOZZLE NO.	AN 06206A	AN 06206B	AN 06206C	AN 06285A	AN 06285B	AN 06285C	AN 06215	AP 06308	AP06323A	AP06323B
PIPE CLASS & LAR NO DATA	4"HBD-35	4"HBD-35	4"HBD-35	6"HCD-45	4"HCD-45	4"HCD-45	4"HBD-40	1½"HCD-69	1½"HCD-74	1½"HCD-74
PRESS MAX / NORM PSIG	140 / 60	140 / 60	140 / 60	110 / 60	110 / 60	110 / 60	140 / 60	60 / 30	125 / 60	125 / 60
TEMP MAX / NORM °F	155 / 120	155 / 120	155 / 120	140 / 120	140 / 120	140 / 120	155 / 120	155 / 120	160 / 140	160 / 140

SERVICE	EVAP. CONC. STORAGE TANK	EVAP. DISTIL SAMPLE TANK	LAUNDRY DRAIN TANK "A" DISCHARGE	LAUNDRY/ DRAIN TANK "B" DISCHARGE	LAUNDRY DRAIN FILTER EFFLUENT	LAUNDRY DRAIN SAMPLE TANK	REACTOR WATER CLEAN-UP DECANT PUMP DISCHARGE	WASTE SLUDGE DECANT PUMP DISCHARGE
P & ID LOCATION	M-163 F-2	M-163 D-2	M-164 C-5	M-164 C-5	M-164 D-3	M-164 D-0	M-166 B-8	M-166 B-1
ANALYSIS NOZZLE NO.	AP 06348	AP 06358	AP06408A	AP06408B	AP 06415	AP06420	AP 06631	AP 06632
PIPE CLASS & LAR NO DATA	2"HCD-79	1½"HCD-68	1½"HBD-46	1½"HBD-46	1½"HBD-48	1½"HBD-20	2"HCD-36	2"HCD-92
PRESS MAX / NORM PSIG	100 / 70	100 / 70	110 / 100	110 / 100	110 / 25	70 / 50	90 / 80	90 / 80
TEMP MAX / NORM °F	180 / 140	140 / 120	155 / 105	155 / 105	155 / 105	155 / 105	120 / 90	120 / 90

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P&ID
PROCESS SAMPLING

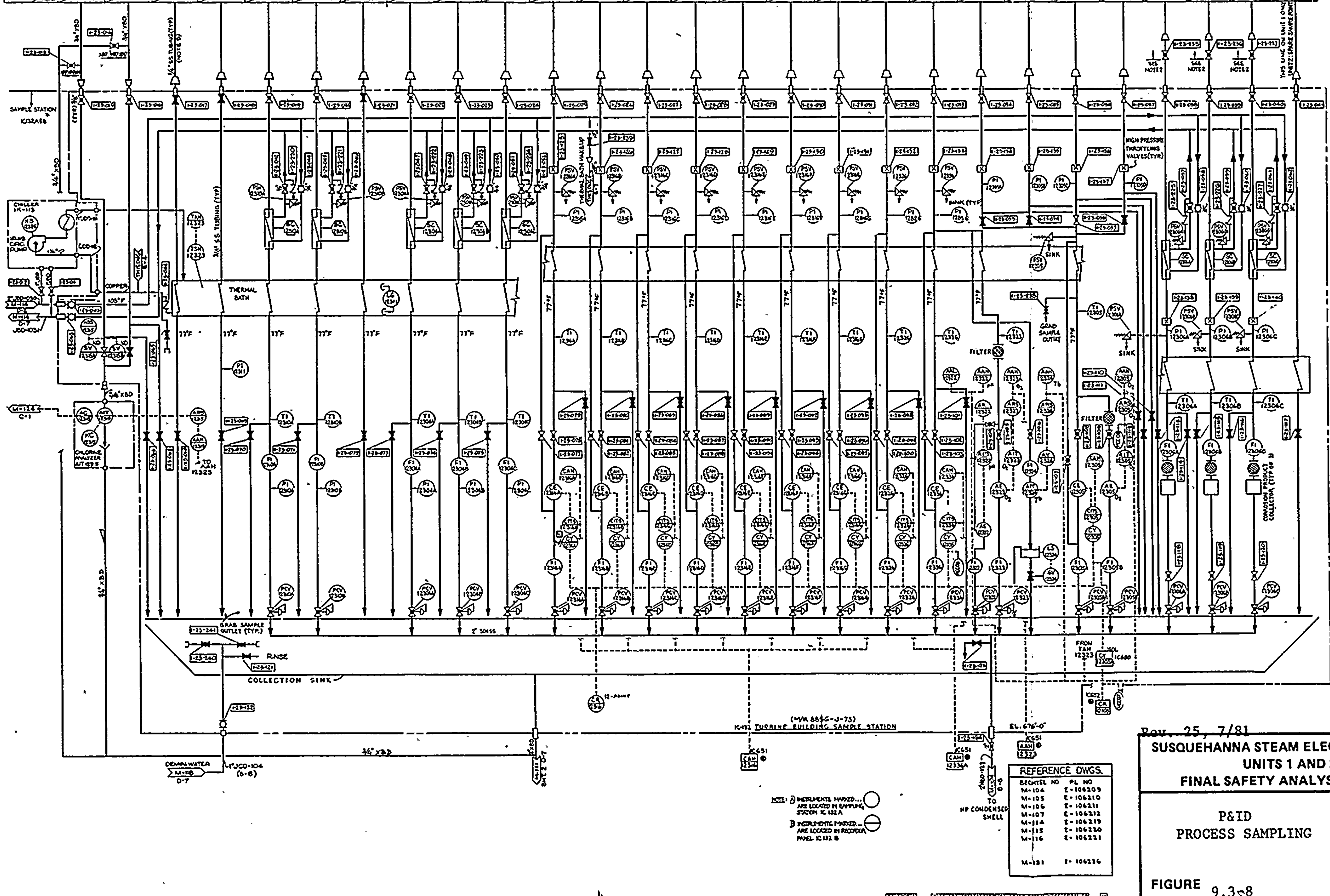
FIGURE 9.3-7

SERVICE	HP COND CIRCULATING WATER OUTLET	HP COND CIRCULATING WATER OUTLET	RECOMBINER CLOSED LOOP COOLING WATER UNIT 1	TURBINE BLDG. COOLING WATER PUMP DISCH.	STEAM SEAL EVAPORATOR CHLORIDE SAMPLE	STEAM SEAL EVAPORATOR SEPARATOR DRAIN SAMPLE	START-UP CLEANUP RETURN LINE
P&ID LOCATION	M-115 A-2	M-115 A-1	M-131 F-4	M-114 F-5	M-107 A-5	M-107 A-5	M-105 F-0
ANALYSIS NOZZLE NO	AN11551A	AN11551B	AN13114	AN11405	AP10760	AP10750	AN10507
PIPE CLASS AND LINE NO. DATA	3/4"	3/4"	8" JBD-110	4" JBD-144	2" HBD-169	2" HBD-169	16" GBD-113
PRESS. PSIG MAX / NORM	30 / 30	30 / 30	75 / 50	80 / 71	180 / 100	180 / 100	60 / VAC
TEMP. °F MAX / NORM	127 / 85	127 / 85	135 / 135	122 / 122	299 / 255	299 / 255	130 / 130

SERVICE	DRAIN FROM HEATER "3A"	DRAIN FROM HEATER "3B"	DRAIN FROM HEATER "3C"	COND. FILTER DEMIN."A" DISCH.	COND. FILTER DEMIN."B" DISCH.	COND. FILTER DEMIN."C" DISCH.	COND. FILTER DEMIN."D" DISCH.
P&ID LOCATION	M-104 A-7	M-104 A-4	M-104 A-2	M-116-1 D-0	M-116-1 D-1	M-116-1 D-2	M-116-1 D-3
ANALYSIS NOZZLE NO	AN10407A	AN10407B	AN10407C	AN11610A	AN11610B	AN11610C	AN11610D
PIPE CLASS AND LINE NO. DATA	14" HBD-116	14" HBD-116	14" HBD-116	16" GBD-115	16" GBD-115	16" GBD-115	16" GBD-115
PRESS. PSIG MAX / NORM	45 / 40	45 / 40	45 / 40	700 / 577	700 / 577	700 / 577	700 / 577
TEMP. °F MAX / NORM	293 / 227	293 / 227	293 / 227	150 / 133	150 / 133	150 / 133	150 / 133

SERVICE	COND. FILTER DEMIN."E" DISCH.	COND. FILTER DEMIN."F" DISCH.	COND. FILTER DEMIN."G" DISCH.	COND. FILT. DEMIN. INFLUENT HEADER	COND. FILT. DEMIN. EFFLUENT HEADER	COND. PUMP "A" DISCH.	COND. PUMP "B" DISCH
P&ID LOCATION	M-116-1 D-4	M-116-1 D-6	M-116-1 D-7	M-116-1 F-7	M-116 G-2	M-105 D-0	M-105 D-2
ANALYSIS NOZZLE NO	AN11610E	AN11610F	AN11610G	AN11600	AN11621	AN10507A	AN10507B
PIPE CLASS AND LINE NO. DATA	16" GBD-115	16" GBD-115	16" GBD-115	30" GBD-107	30" GBD-115	18" GBD-105	18" GBD-105
PRESS. PSIG MAX / NORM	700 / 577	700 / 577	700 / 577	700 / 577	700 / 577	700 / 577	700 / 577
TEMP. °F MAX / NORM	150 / 133	150 / 133	150 / 133	150 / 133	150 / 133	150 / 133	150 / 133

SERVICE	COND. PUMP "C" DISCH	COND. PUMP "D" DISCH	REACTOR FEED PUMP "A" DISCH.	REACTOR FEED PUMP "B" DISCH.	REACTOR FEED PUMP "C" DISCH.	COMMON RECOMBINER CLOSED LOOP COOLING WATER
P&ID LOCATION	M-105 D-3	M-105 D-4	M-106 F-7	M-106 F-5	M-106 F-2	M-131 F-4
ANALYSIS NOZZLE NO	AN10507C	AN10507D	AN10606A	AN10606B	AN10606C	AN-05114
PIPE CLASS AND LINE NO. DATA	18" GBD-105	18" GBD-105	18" DBD-101	18" DBD-101	18" DBD-101	6" JBD-7
PRESS. PSIG MAX / NORM	700 / 577	700 / 577	2000 / 1080	2000 / 1080	2000 / 1080	75 / 50
TEMP. °F MAX / NORM	150 / 133	150 / 133	387 / 387	387 / 387	387 / 387	135 / 135



NOTE: 1) INSTRUMENTS MARKED ... ARE LOCATED IN SAMPLING STATION 1C 132 A

2) INSTRUMENTS MARKED ... ARE LOCATED IN RECORDER PANEL 1C 132 B

REFERENCE DWGS.	
BECHTEL NO	PL NO
M-104	E-106209
M-105	E-106210
M-106	E-106211
M-107	E-106212
M-114	E-106219
M-115	E-106220
M-116	E-106221
M-131	E-106236

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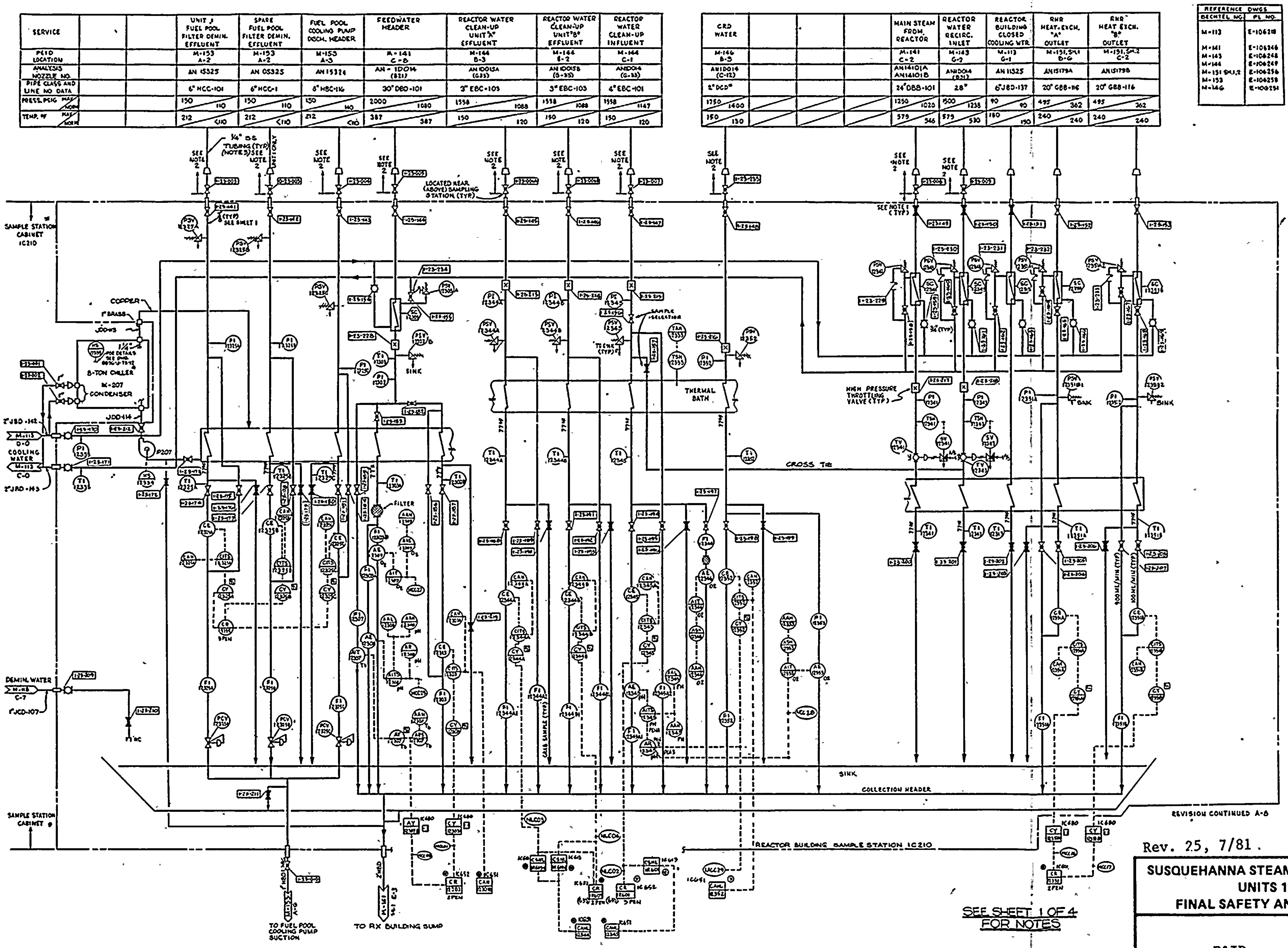
P&ID
PROCESS SAMPLING

FIGURE 9.3-8



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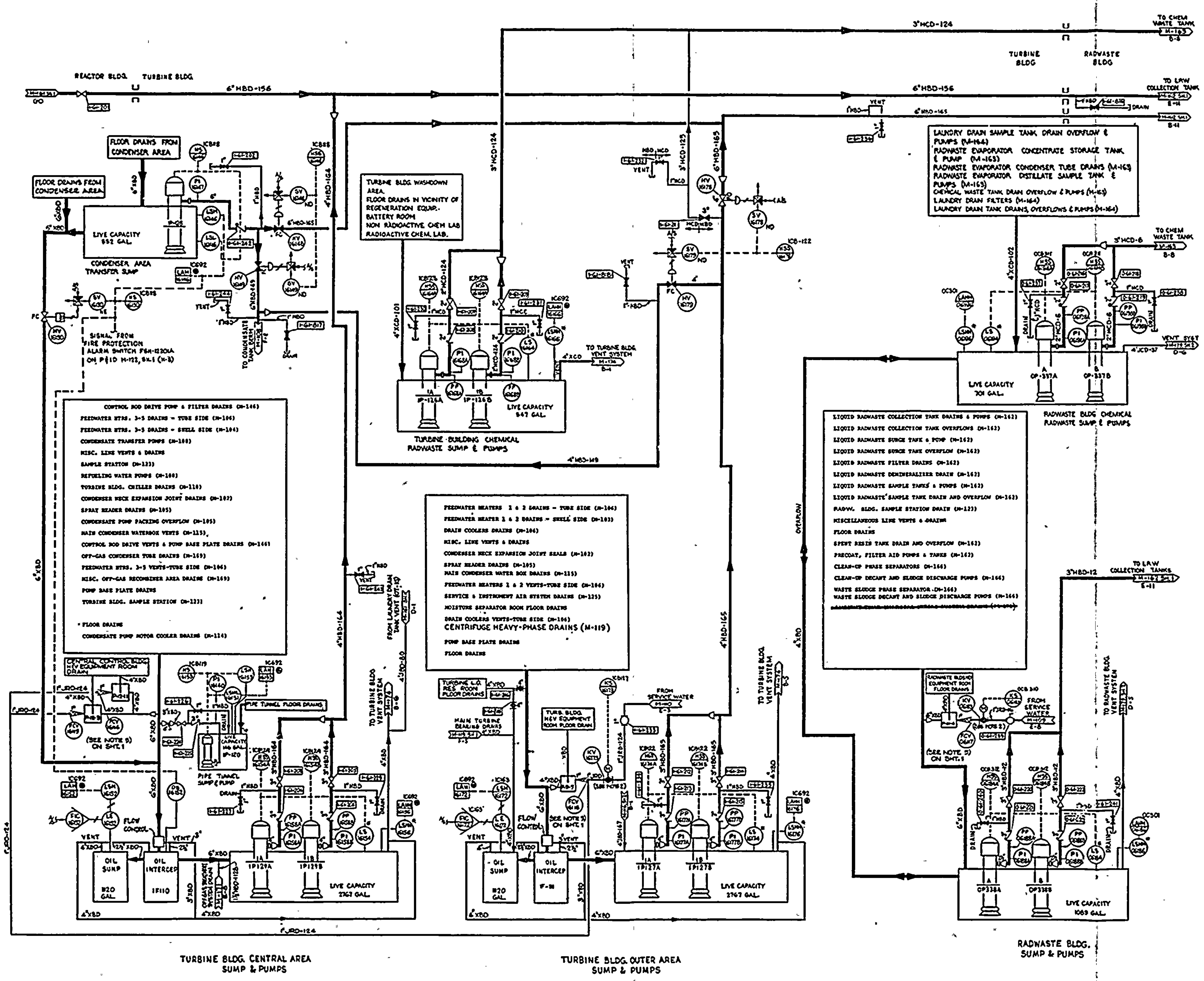
P&ID
PROCESS SAMPLING

FIGURE 9.3-9



REFERENCE DRAWINGS	
BECHTEL NO.	PL. NO
M-102	E-106207
M-103	E-106208
M-104	E-106209
M-105	E-106210
M-106	E-106211
M-108	E-106213
M-110	E-106215
M-113	E-106218
M-114	E-106219
M-115	E-106220
M-116, SH.2	E-106221
M-123	E-106228
M-125	E-106230
M-141	E-106246
M-143	E-106248
M-144	E-106249
M-145	E-106250
M-146	E-106251
M-147	E-106252
M-152	E-106257
M-153	E-106258
M-161 SH.1&2	E-106266
M-162 SH.1	E-106267
M-163	E-106268
M-164	E-106269
M-166	E-106271
M-169	E-106274
M-171	E-106276
M-174	E-106279
M-179, SH.2	E-106284
J-161	B-103451
P-3-5	E-106356 SH.5
P-28-4	E-106375 SH.4
P-41-4	E-106388 SH.4

NOTES:

1. LAST MANUAL VALVE NUMBER ON THIS SHEET IS X-61-246
2. VALVES KV-16173 & KV-06183 TO HAVE RAMCON OPERATORS. SEE NOTE 5 ON M-161 SHEET 1.

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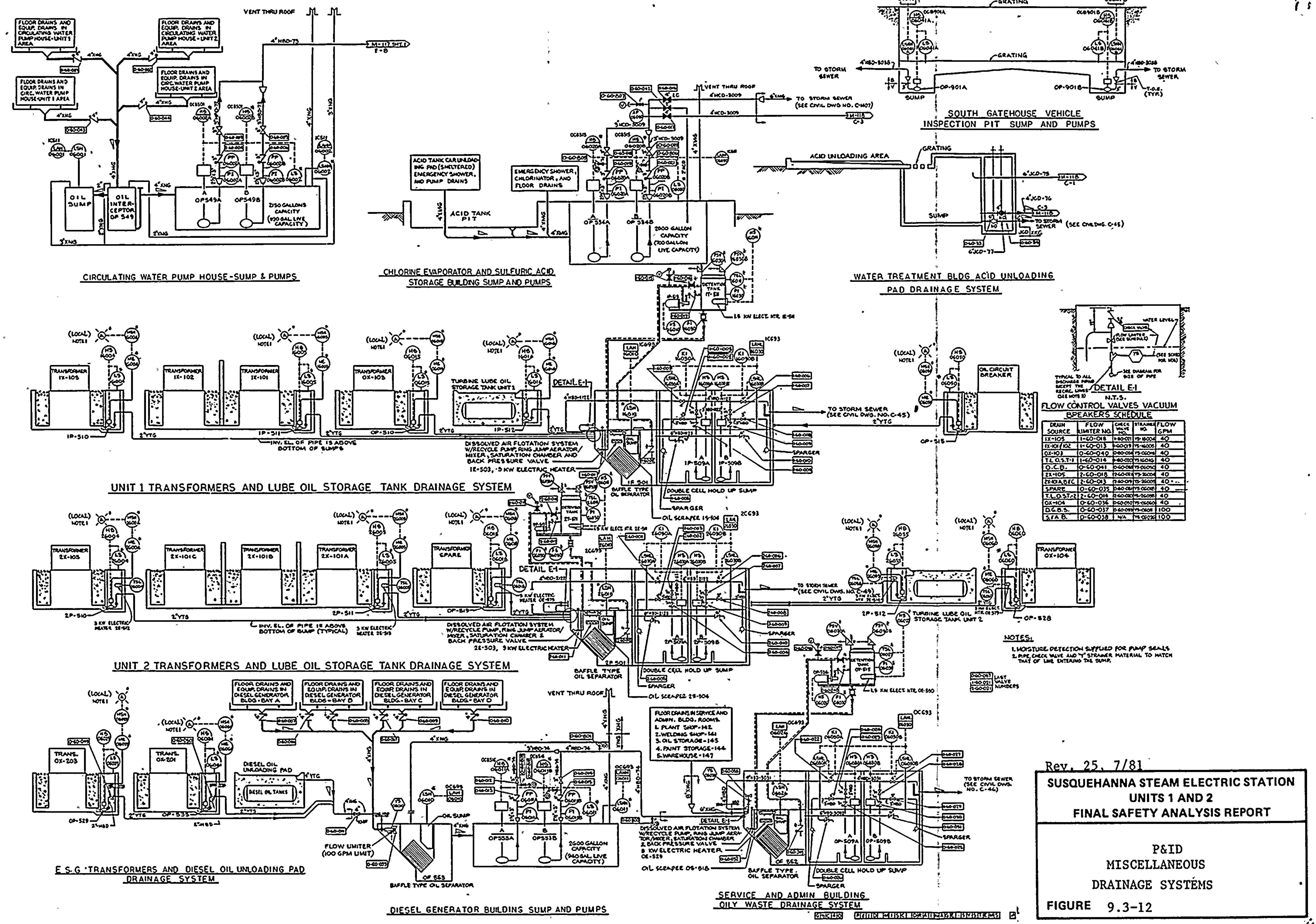
P&ID
LIQUID RADWASTE
COLLECTION

FIGURE 9.3-11



FLOW CONTROL VALVES VACUUM BREAKERS SCHEDULE

DRAIN SOURCE	FLOW LIMITER NO.	CHECK VALVE NO.	STRAINER NO.	FLOW GPM
1X-105	1-60-018	1-60-021	YS-16004	40
1X-101/102	1-60-013	1-60-019	YS-16006	40
0X-103	0-60-040	0-60-056	YS-06015	40
T.L.O.S.T.-1	1-60-014	1-60-020	YS-16016	40
O.C.B.	0-60-041	0-60-058	YS-06050	40
2X-105	2-60-018	2-60-021	YS-26004	40
2X-101A,B/C	2-60-013	2-60-019	YS-26005	40
SPARE	0-60-035	0-60-081	YS-06002	40
T.L.O.S.T.-2	2-60-014	2-60-020	YS-26095	40
0X-104	0-60-036	0-60-052	YS-06060	40
D.G.B.S.	0-60-037	0-60-053	YS-06011	100
S/A B.	0-60-038	N/A	YS-06026	100

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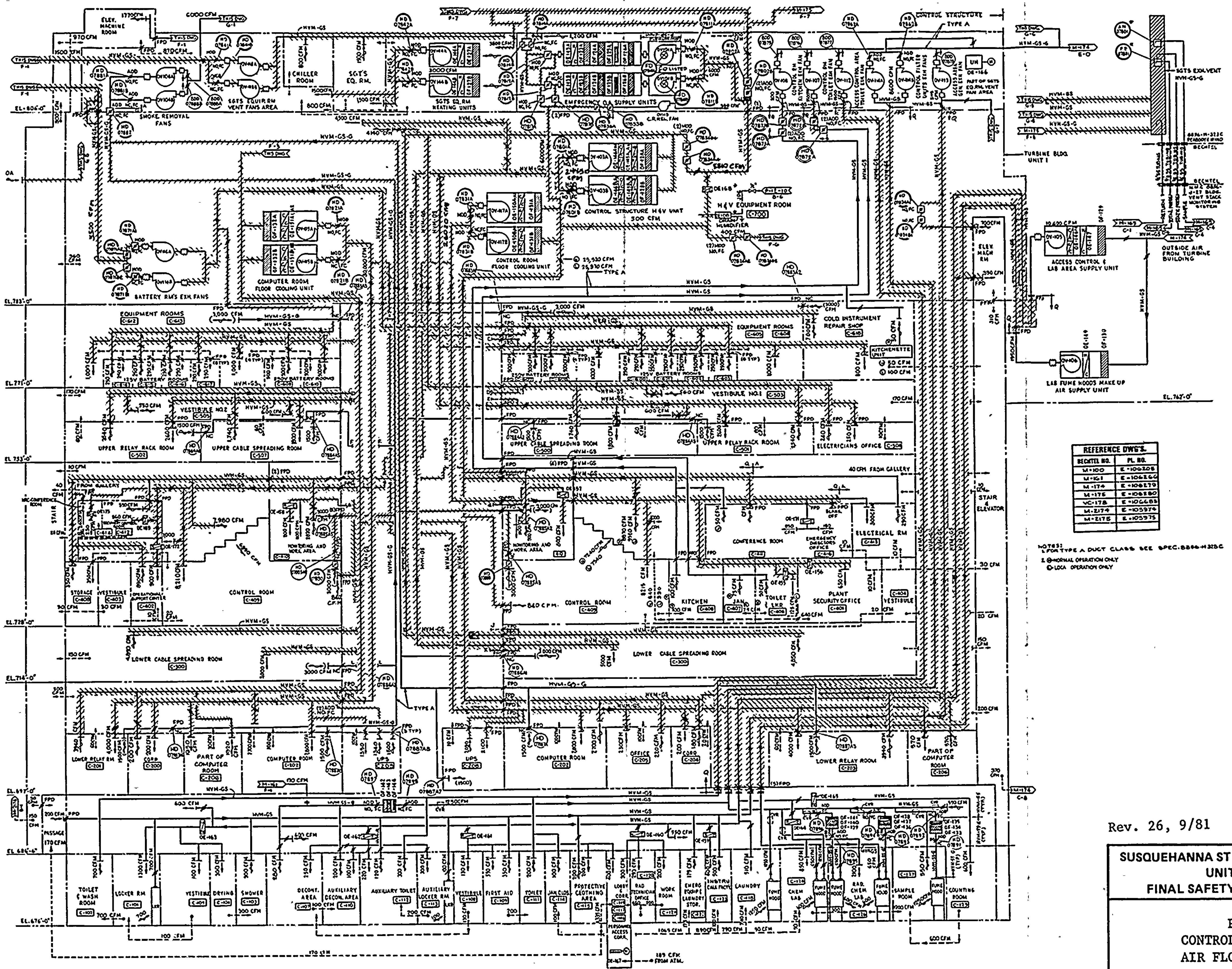
P&ID
MISCELLANEOUS
DRAINAGE SYSTEMS

FIGURE 9.3-12



REFERENCE DWG'S.	
BECHTEL NO.	PL NO.
M-100	E-106205
M-161	E-106266
M-174	E-106279
M-175	E-106280
VC-178	E-106683
M-2174	E-105974
M-2175	E-105975

NOTES:
1. FOR TYPE A DUCT CLASS SEE SPEC. 8856-M-32DC
2. ⊗ NORMAL OPERATION ONLY
○ LOCA OPERATION ONLY

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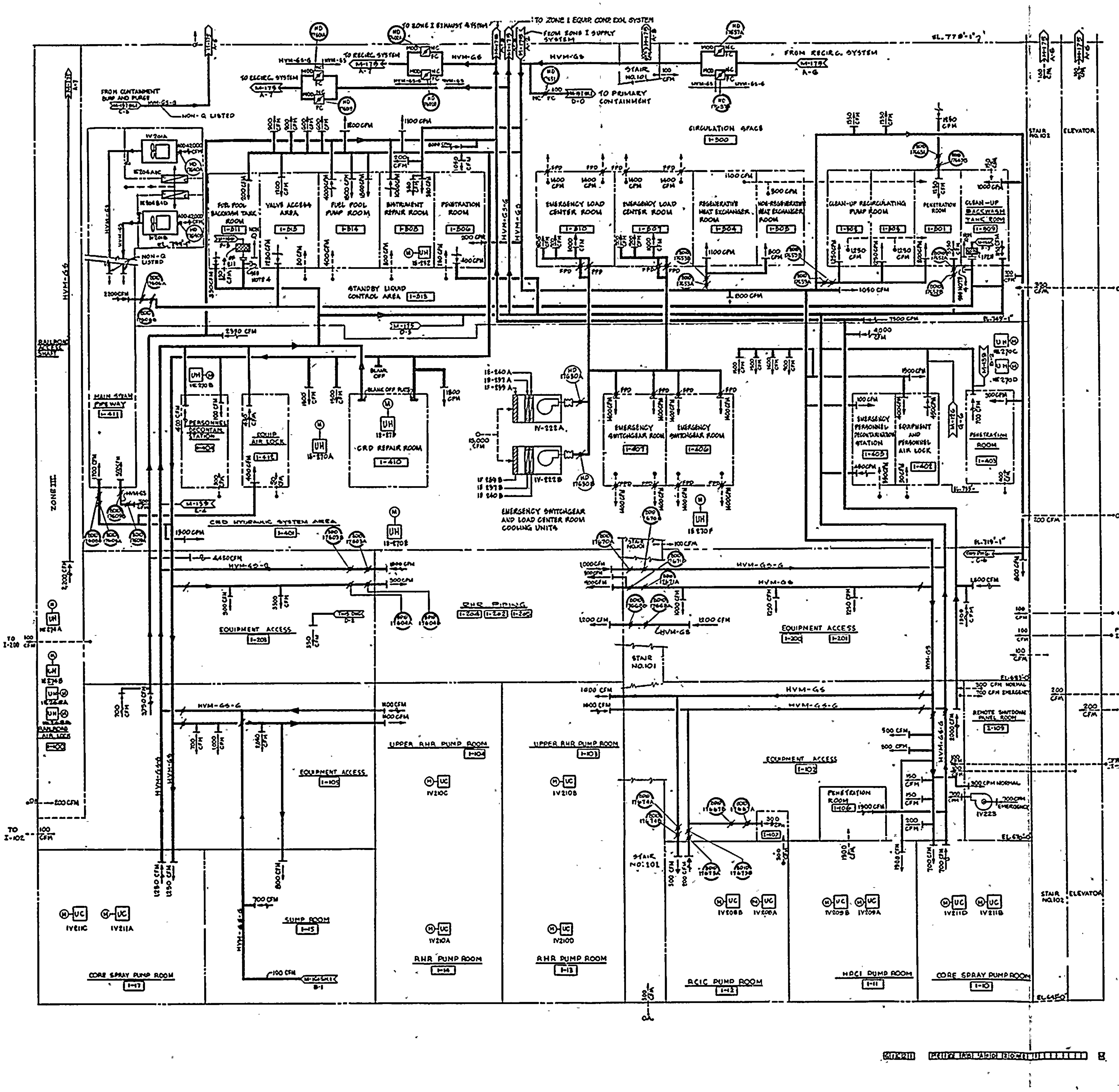
P&ID
CONTROL STRUCTURE
AIR FLOW DIAGRAM

FIGURE 9.4-1



REFERENCE DRAWINGS

BECHTEL NO.	PL. NO.
M-126	E-106231
M-139	E-106244
M-175	E-106280
M-100 SH1&2	E-106205&2
M-166	E-106271

NOTES

1. THE BRANCH DUCTS SERVING EACH INDIVIDUAL SPACE, OR ROOM SHALL BE PROVIDED WITH A BALANCING DAMPER.
2. DRAWING SHOWS VENTILATING ZONE I FOR UNIT I.
3. ALL DUCTWORK PENETRATIONS OF BUILDING FIRE PARTITIONS OR FLOORS SHALL BE PROVIDED WITH FIRE DAMPERS.
4. PROVIDE PLUGGED 1/2" HALF COUPLINGS WELDED TO SHEET METAL DUCT.

ALL DUCTWORK ON THIS P&ID IS Q-LISTED, EXCEPT AS INDICATED AND NOTED.

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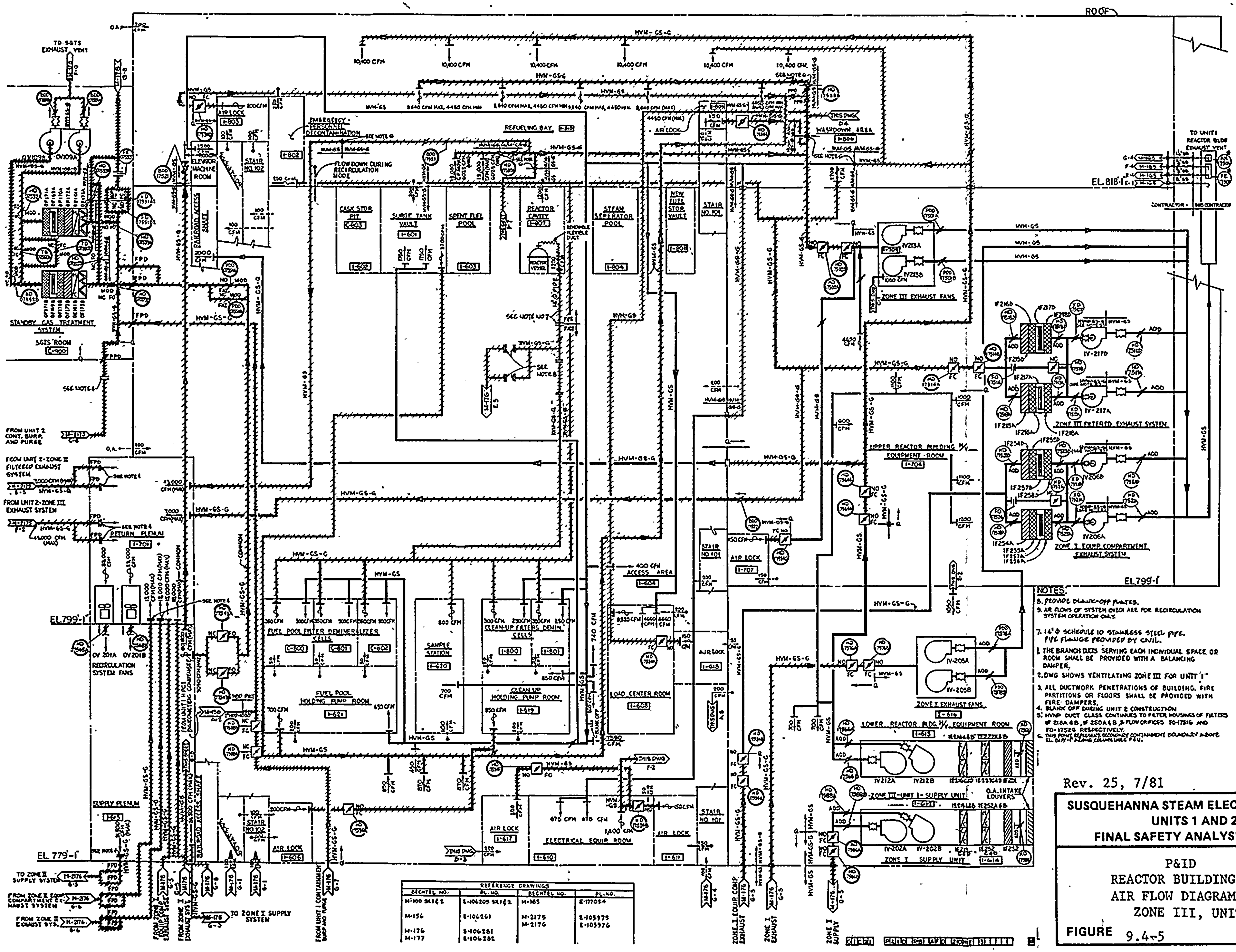
P&ID
REACTOR BUILDING
AIR FLOW DIAGRAM
ZONE 1

FIGURE 9.4-4



NOTES:

8. PROVIDE BLANK-OFF PLATES.
9. AIR FLOWS OF SYSTEM OV201 ARE FOR RECIRCULATION SYSTEM OPERATION ONLY.
7. 14" ∅ SCHEDULE 10 STAINLESS STEEL PIPE. PIPE FLANGE PROVIDED BY CIVIL.
1. THE BRANCH DUCTS SERVING EACH INDIVIDUAL SPACE OR ROOM SHALL BE PROVIDED WITH A BALANCING DAMPER.
2. DWG SHOWS VENTILATING ZONE III FOR UNIT "1"
3. ALL DUCTWORK PENETRATIONS OF BUILDING. FIRE PARTITIONS OR FLOORS SHALL BE PROVIDED WITH FIRE DAMPERS.
4. BLANK OFF DURING UNIT 2 CONSTRUCTION
5. HVNP DUCT CLASS CONTINUES TO FILTER HOUSINGS OF FILTERS IF 216A & B, IF 250A & B & FLOW ORIFICES FO-17516 AND FO-17526 RESPECTIVELY.
6. THIS POINT REPRESENTS SECONDARY CONTAINMENT BOUNDARY ABOVE EL. 818'-1" ALONG COLUMN LINES P & U.

REFERENCE DRAWINGS

BECHTEL NO.	PL. NO.	BECHTEL NO.	PL. NO.
M-100 SH 1 & 2	E-106205 SH 1 & 2	M-165	E-177054
M-156	E-106261	M-2175	E-105975
		M-2176	E-105976
M-176	E-106281		
M-177	E-106282		

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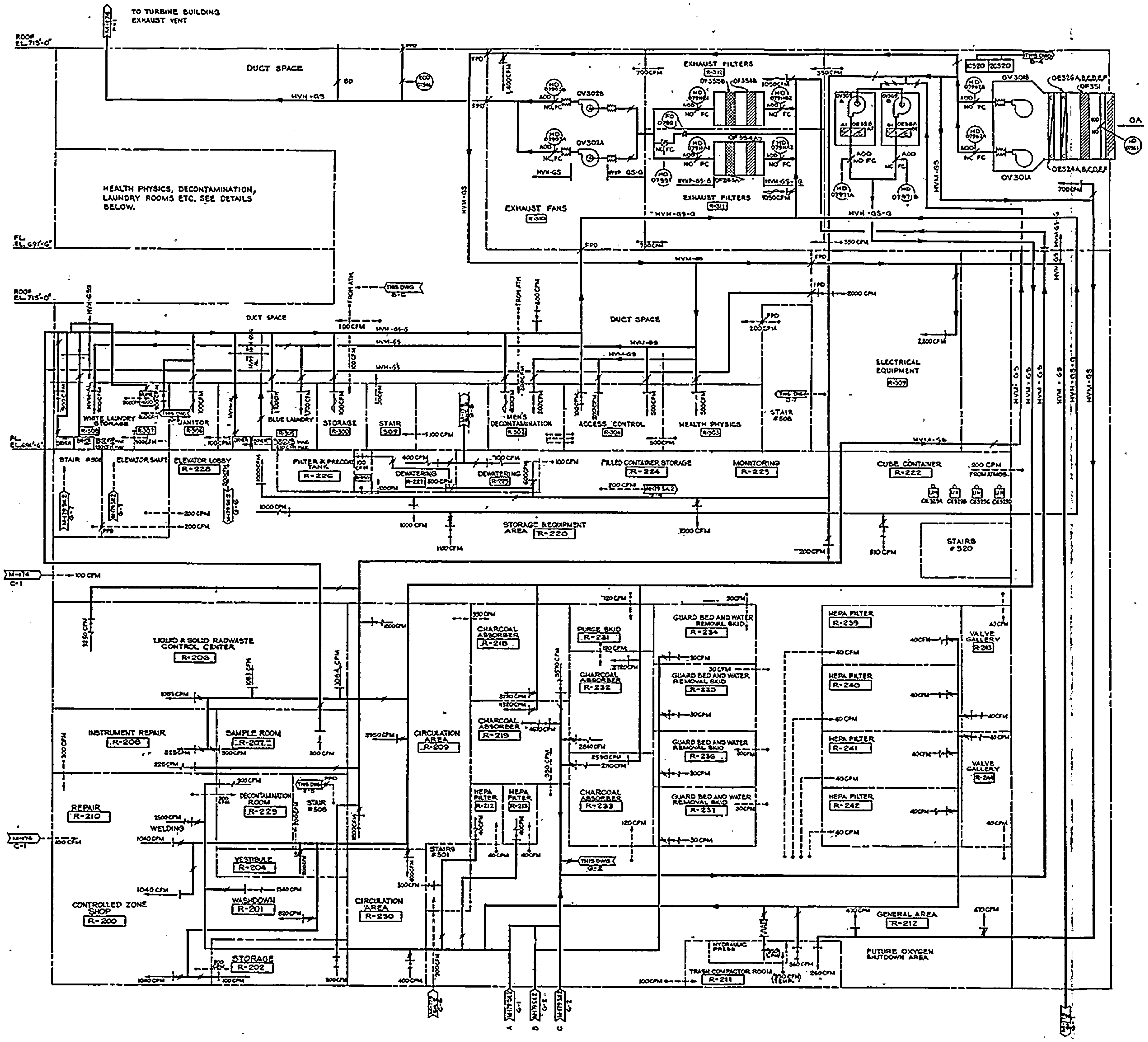
P&ID
REACTOR BUILDING
AIR FLOW DIAGRAM
ZONE III, UNIT 1

FIGURE 9.4-5



REFERENCE - DWGS	
BECHTEL NO.	PL. NO.
M-100	E-106205
M-161	E-106266
M-162	E-106267
M-170	E-106275
M-174	E-106279
M-179, SH.2	E-106284,SH.2
VC-179	E-106684
V-179	B-105519

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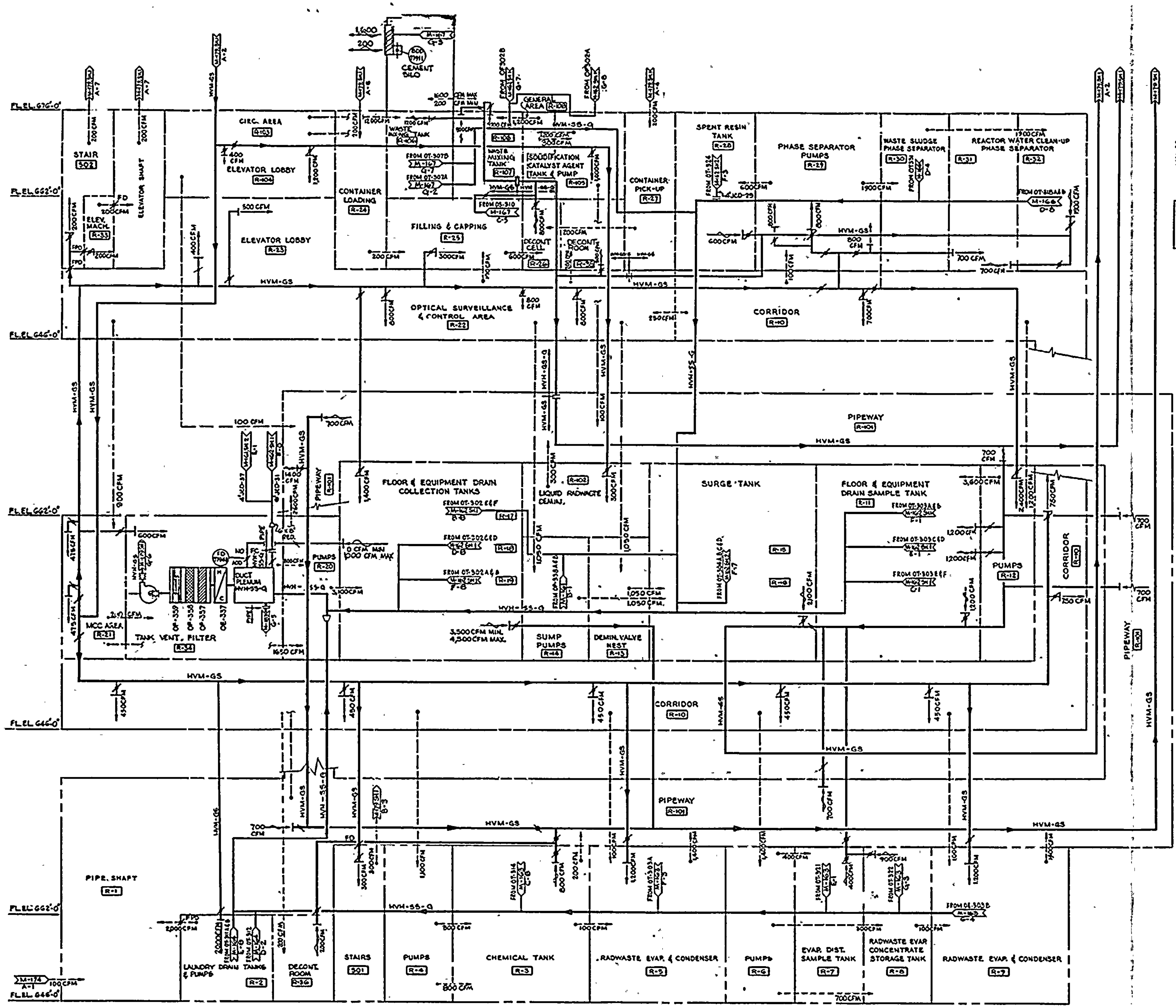
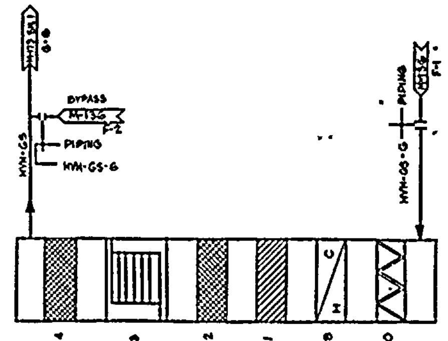
P&ID
RADWASTE BUILDING
AIR FLOW DIAGRAM

FIGURE 9.4-10





PRIMARY COOLANT DEGASIFIER EXHAUST FILTER SYSTEM

REFERENCE DWGS	
BECHTEL NO.	PLAN NO.
M-100	E-106205
M-161	E-106266
M-162	E-106267
M-163	E-106268
M-164	E-106269
M-166	E-106271
M-167	E-106272
M-174	E-106279
M-179 SH.1	E-106284, SH.1
VC-179	E-106681
V-179	B-106519

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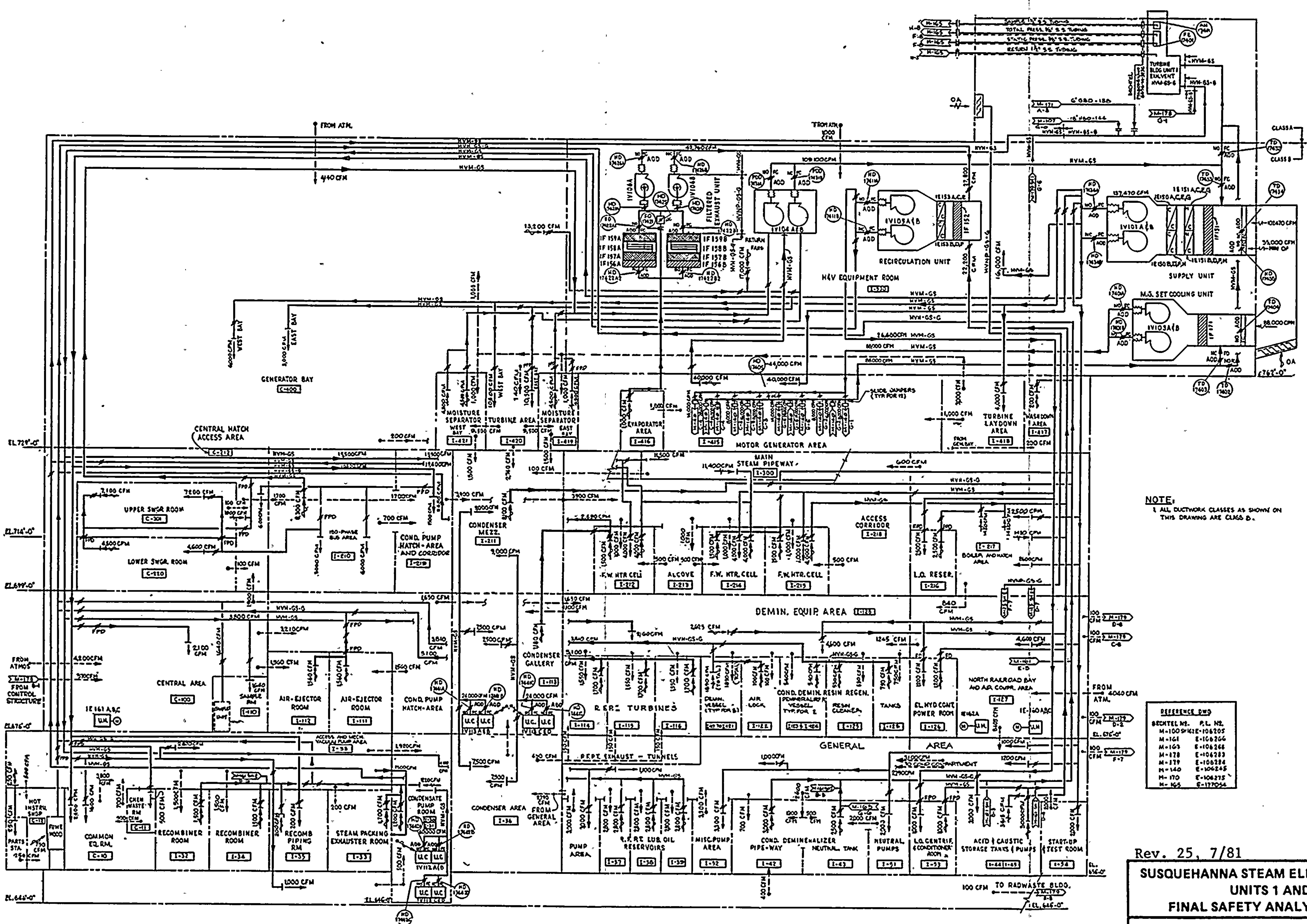
P&ID
RADWASTE BUILDING
AIRFLOW DIAGRAM

FIGURE 9.4-11



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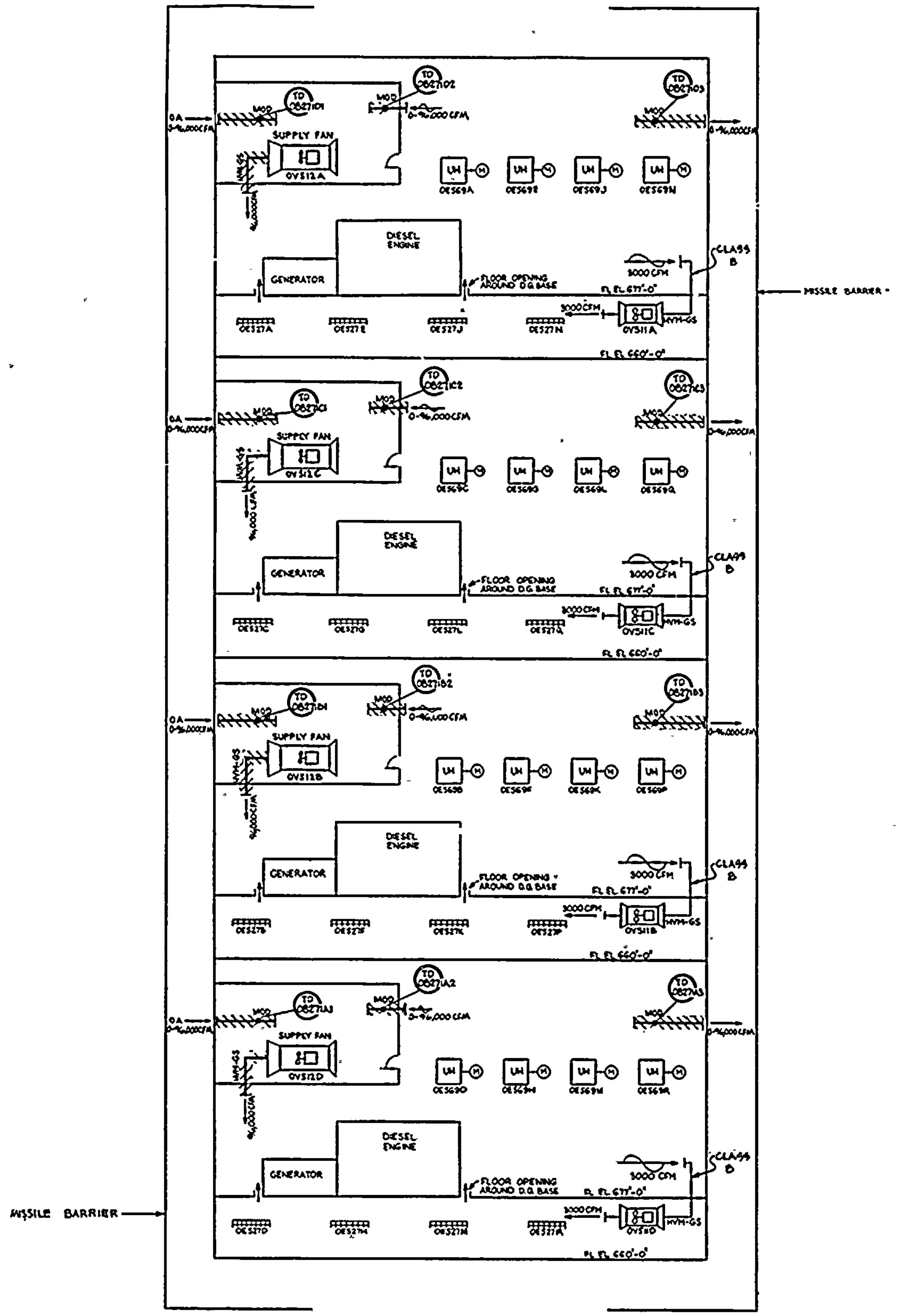
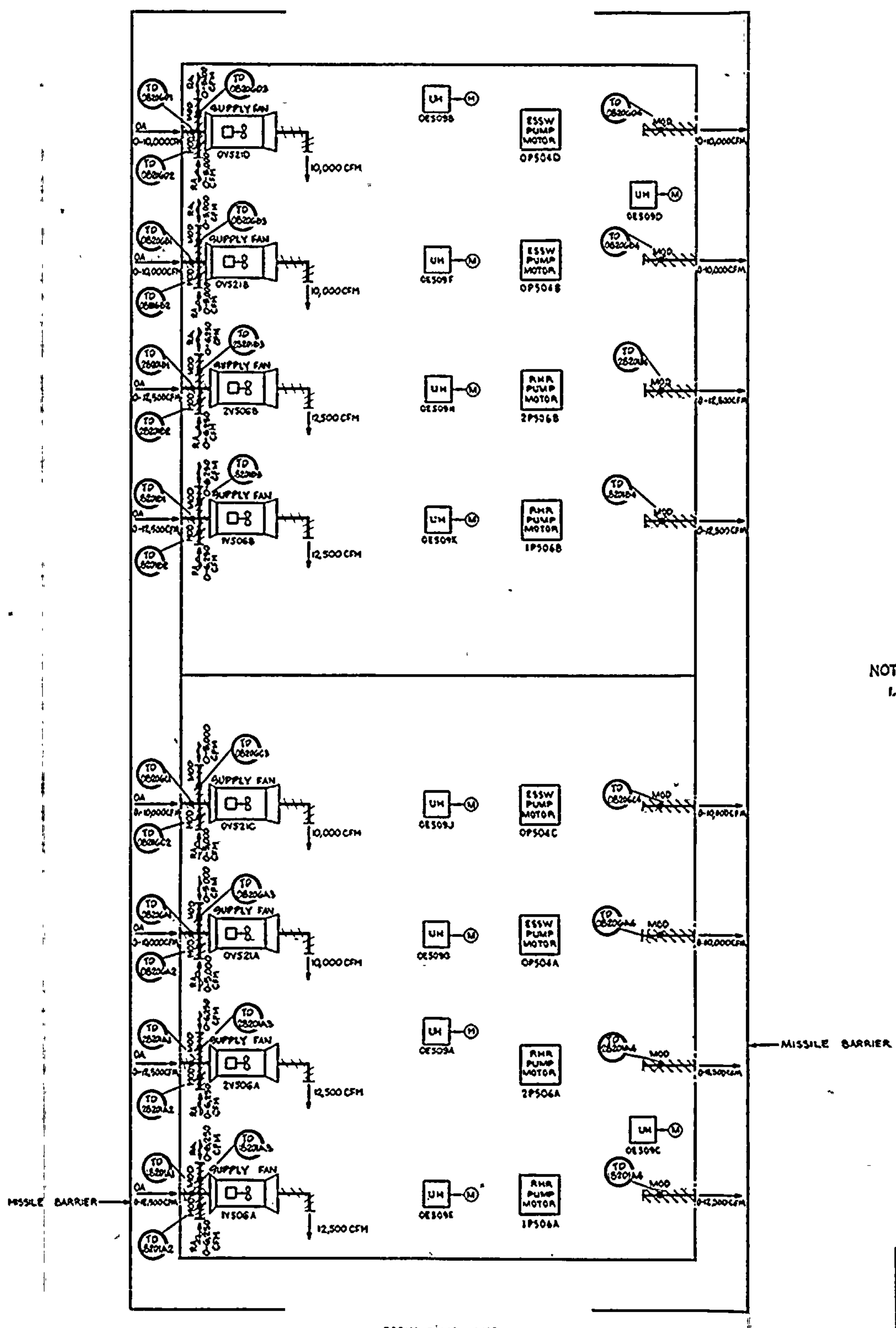
P&ID
TURBINE BUILDING
AIR FLOW DIAGRAM

FIGURE 9.4-13





NOTE:

1. ALL DUCTWORK CONSTRUCTION CLASS Q AND HVM-GS EXCEPT AS NOTED ON THIS DRAWING.

REFERENCE DWGS.	
BECHTEL NO.	PL. NO.
M-100, SH.1,2	E-106205-1&2
VC-182	E-106687
V-182	B-106522

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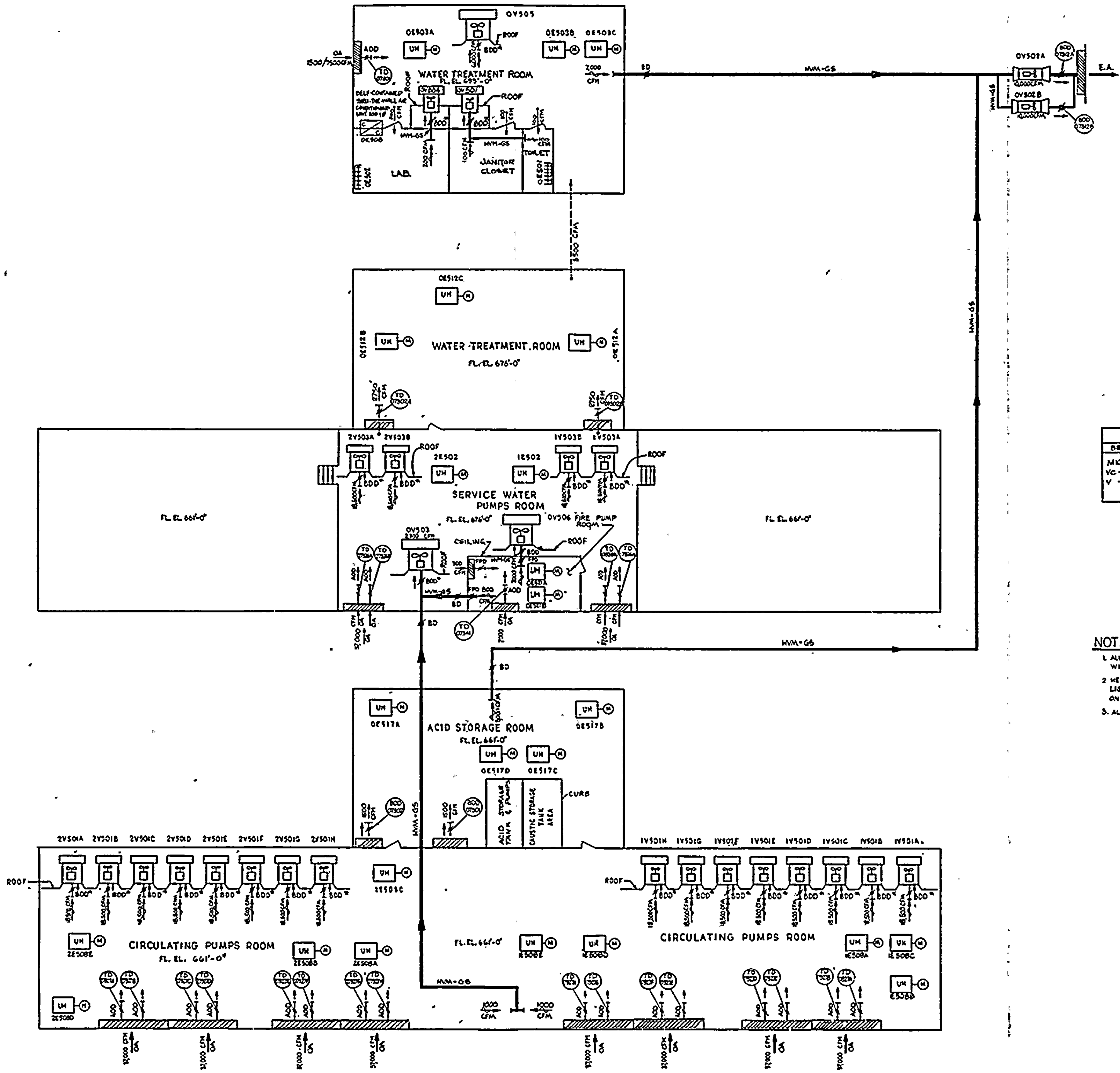
P&ID
DIESEL GENERATOR
AND ESSW PUMPHOUSE
AIRFLOW DIAGRAM

FIGURE 9.4-19



REFERENCE DRAWINGS	
BECHTEL NO.	PL. NO.
M100 SH 1 & 2	E-106205 SH 1 & 2
VC-173	E-106678
V -173	B-106513

NOTE:

1. ALL FANS EXCEPT OV502A & OV502B ARE PROVIDED WITH BACK DRAFT DAMPERS.
2. HEAVY LINES DESIGNATE DUCT RUNS, LIGHT LINES INDICATE THAT A DAMPER IS MOUNTED ON EQUIPMENT
3. ALL DUCTWORK CLASSES ON THIS DWG. ARE CLASS B

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P&ID
CIRCULATING WATER
PUMPHOUSE
AIRFLOW DIAGRAM

FIGURE 9.4-20

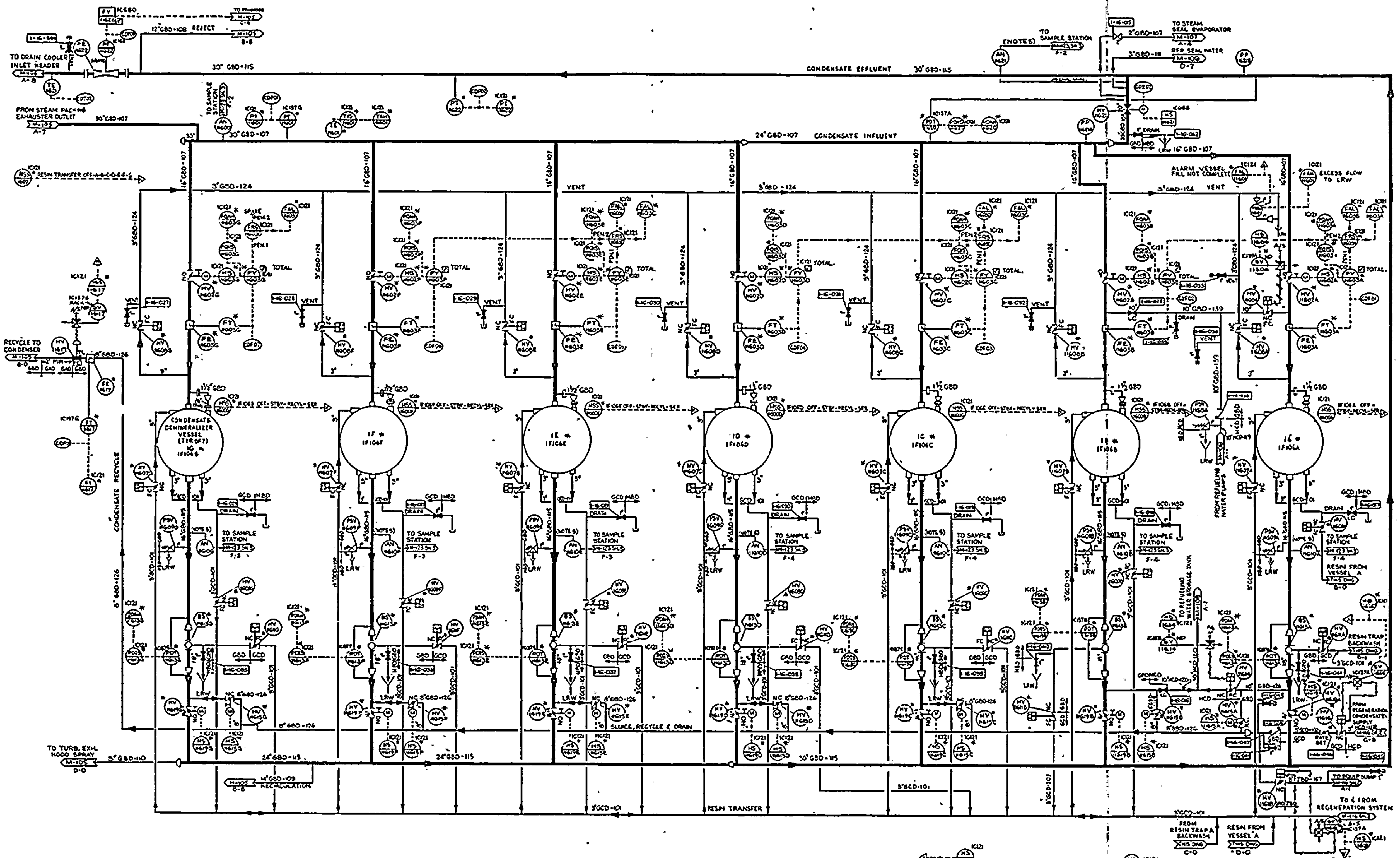
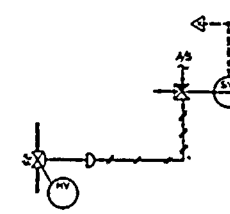
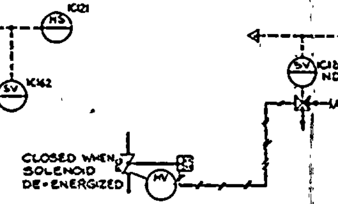


TABLE 8-4 SOLENOID VALVES IN CABINET			
SV	CABINET	SV	CABINET
SV 11604	1C137A	SV 11607 B THRU G	1C137B TO 1C137G
SV 11607A	✓	SV 11608 B THRU G	✓
SV 11608A	✓	SV 11609 B THRU G	✓
SV 11609A	✓	SV 11618 B THRU G	✓
SV 11611A	✓		
SV 11614	✓		
SV 11616	✓		
SV 11618	✓		



TYPICAL DETAIL OF AIR OPERATED VALVES WITH DIAPHRAM ACTUATOR UNLESS SHOWN OTHERWISE



TYPICAL DETAIL OF AIR OPERATED VALVES WITH PISTON ACTUATOR UNLESS SHOWN OTHERWISE

NOTES CONT:

4. ALL HIGH POINT VENTS AND LOW POINT DRAINS TO LRW ARE 1" AND ARE LINE CLASS HBD DOWNSTREAM OF LAST ISOLATION VALVE UNLESS OTHERWISE SHOWN. EXAMPLE:

5. INSTALL STAINLESS STEEL ROOT VALVES (ONE 1" ECD-GB PER ANALYSIS NOZZLE) ON AN-11610A THRU G AND ON AN-11621 TO COMPLY WITH GE/NED SPEC.# 22A2708.

NOTES:

1. VENDOR SYSTEM PANEL NO. 1C121.
2. CYCLE CONTROLS ARE NOT SHOWN ON P&ID.
3. MISC. VALVE NO'S ON SHEET 1 USED: 1-16-015 THRU 1-16-049 SPARE: 1-16-010 THRU 1-16-100 AND 1-16-001 THRU 1-16-014

REFERENCE DWG'S.	
BECHTEL NO.	PL. NO.
M-100	E-106205
M-105	E-106210
M-108	E-106213
M-86 Sh. 2	E-106221 Sh. 2
M-123 Sh. 3	E-106228 Sh. 3
J-116	B-103406
M-106	E-106211
M-107	E-106213

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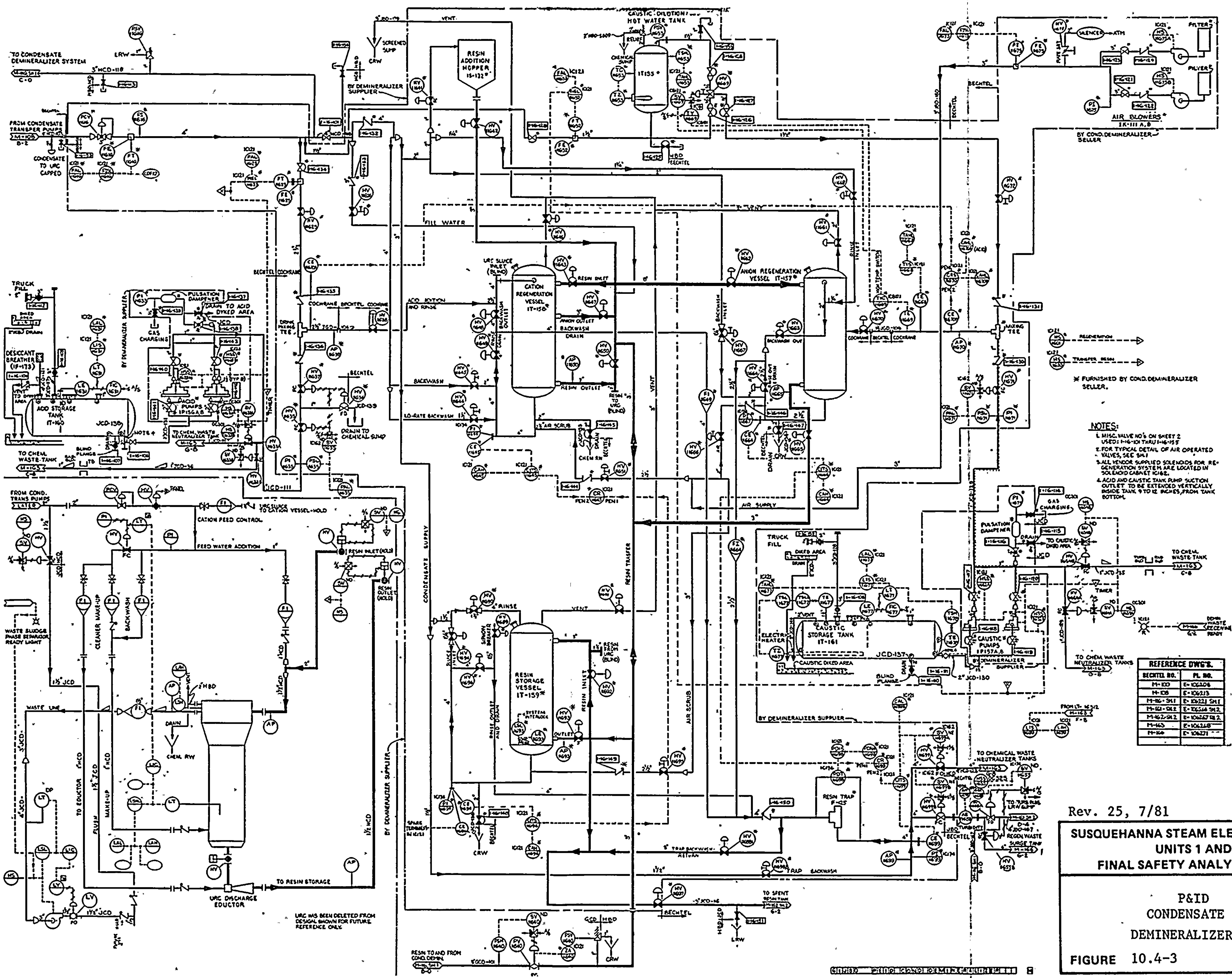
P&ID
CONDENSATE
DEMINERALIZER

FIGURE 10.4.2



REFERENCE DWG'S.	
BECHTEL NO.	PL. NO.
M-100	E-106206
M-108	E-106213
M-116-SH.1	E-106221 SH.1
M-161-SH.2	E-106246 SH.2
M-162-SH.2	E-106247 SH.2
M-163	E-106248
M-164	E-106271

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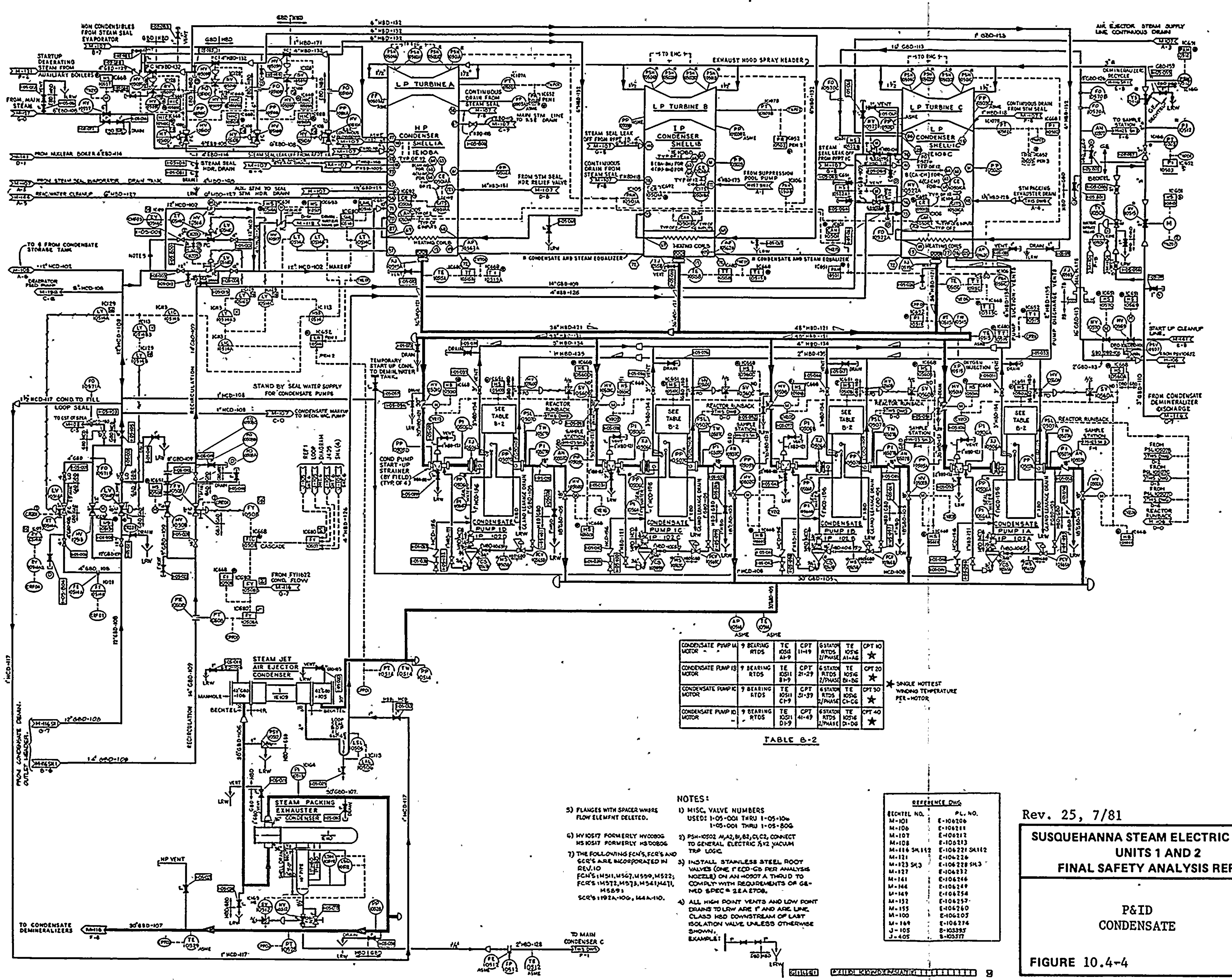
P&ID
CONDENSATE
DEMINERALIZER

FIGURE 10.4-3



CONDENSATE PUMP 1A MOTOR	9 BEARING RTDS	TE 10511 A1-9	CPT 11-19	6 STATOR RTDS 2/PHASE	TE 10516 A1-A6	CPT 10 ★
CONDENSATE PUMP 1B MOTOR	9 BEARING RTDS	TE 10511 B1-9	CPT 21-29	6 STATOR RTDS 2/PHASE	TE 10516 B1-B6	CPT 20 ★
CONDENSATE PUMP 1C MOTOR	9 BEARING RTDS	TE 10511 C1-9	CPT 31-39	6 STATOR RTDS 2/PHASE	TE 10516 C1-C6	CPT 30 ★
CONDENSATE PUMP 1D MOTOR	9 BEARING RTDS	TE 10511 D1-9	CPT 41-49	6 STATOR RTDS 2/PHASE	TE 10516 D1-D6	CPT 40 ★

★ SINGLE HOTTEST WINDING TEMPERATURE PER-MOTOR

TABLE B-2

NOTES:

1) MISC. VALVE NUMBERS USED: 1-05-001 THRU 1-05-106
1-05-001 THRU 1-05-806

2) PSH-10502 A1,A2,B1,B2,C1,C2, CONNECT TO GENERAL ELECTRIC ½x2 VACUUM TRIP LOGIC

3) INSTALL STAINLESS STEEL ROOT VALVES (ONE 1" ECD-GB PER ANALYSIS NOZZLE) ON AN-10507 A THRU D TO COMPLY WITH REQUIREMENTS OF GE-NED SPEC # 22A2708.

4) ALL HIGH POINT VENTS AND LOW POINT DRAINS TO LRW ARE 1" AND ARE LINE CLASS HBD DOWNSTREAM OF LAST ISOLATION VALVE UNLESS OTHERWISE SHOWN.
EXAMPLE:

5) FLANGES WITH SPACER WHERE FLOW ELEMENT DELETED.

6) HV10517 FORMERLY HV00806
HS10517 FORMERLY HS00806

7) THE FOLLOWING FCN'S, FCR'S AND SCR'S ARE INCORPORATED IN REV.10
FCN'S: M511, M567, M559, M522;
FCR'S: M572, M573, M561, M471, M589;
SCR'S: 192A-106, 144A-110.

REFERENCE DWG.

BECHTEL NO.	PL. NO.
M-101	E-106206
M-106	E-106211
M-107	E-106212
M-108	E-106213
M-116 SH.1&2	E-106221 SH.1&2
M-121	E-106226
M-123 SH.3	E-106228 SH.3
M-127	E-106232
M-141	E-106246
M-144	E-106249
M-149	E-106254
M-152	E-106257
M-155	E-106260
M-100	E-106205
M-169	E-106274
J-105	8-105395
J-405	8-105517

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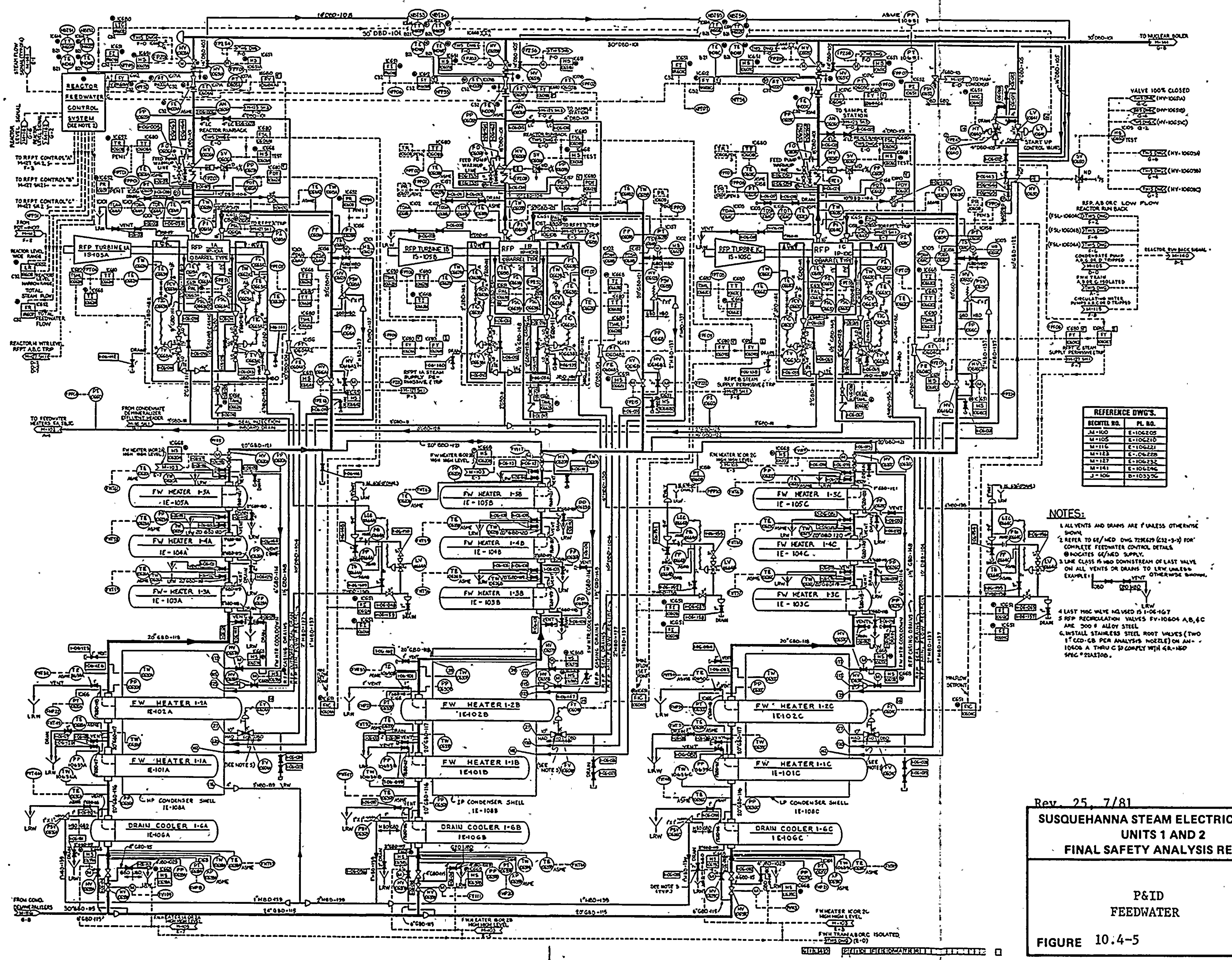
P&ID
CONDENSATE

FIGURE 10.4-4



REFERENCE DWG'S.	
BECHTEL NO.	PL NO.
M-100	E-106205
M-105	E-106210
M-116	E-106221
M-123	E-106228
M-127	E-106232
M-141	E-106246
J-106	B-103356

NOTES:

1. ALL VENTS AND DRAINS ARE 1" UNLESS OTHERWISE SHOWN.
2. REFER TO GE/NED DWG 729E629 (C32-3-X) FOR COMPLETE FEEDWATER CONTROL DETAILS. Ⓖ INDICATES GE/NED SUPPLY.
3. LINE CLASS IS HBD DOWNSTREAM OF LAST VALVE ON ALL VENTS OR DRAINS TO LRW UNLESS OTHERWISE SHOWN. EXAMPLE:

   DBD VENT
   LRW
4. LAST HBC VALVE NO. USED IS 1-06-167
5. RFP RECIRCULATION VALVES FV-10604 A,B,&C ARE 300 # ALLOY STEEL
6. INSTALL STAINLESS STEEL ROOT VALVES (TWO 1" CCD-GB PER ANALYSIS NOZZLE) ON AN-10606 A THRU C TO COMPLY WITH GE-NED SPEC #22A2708.

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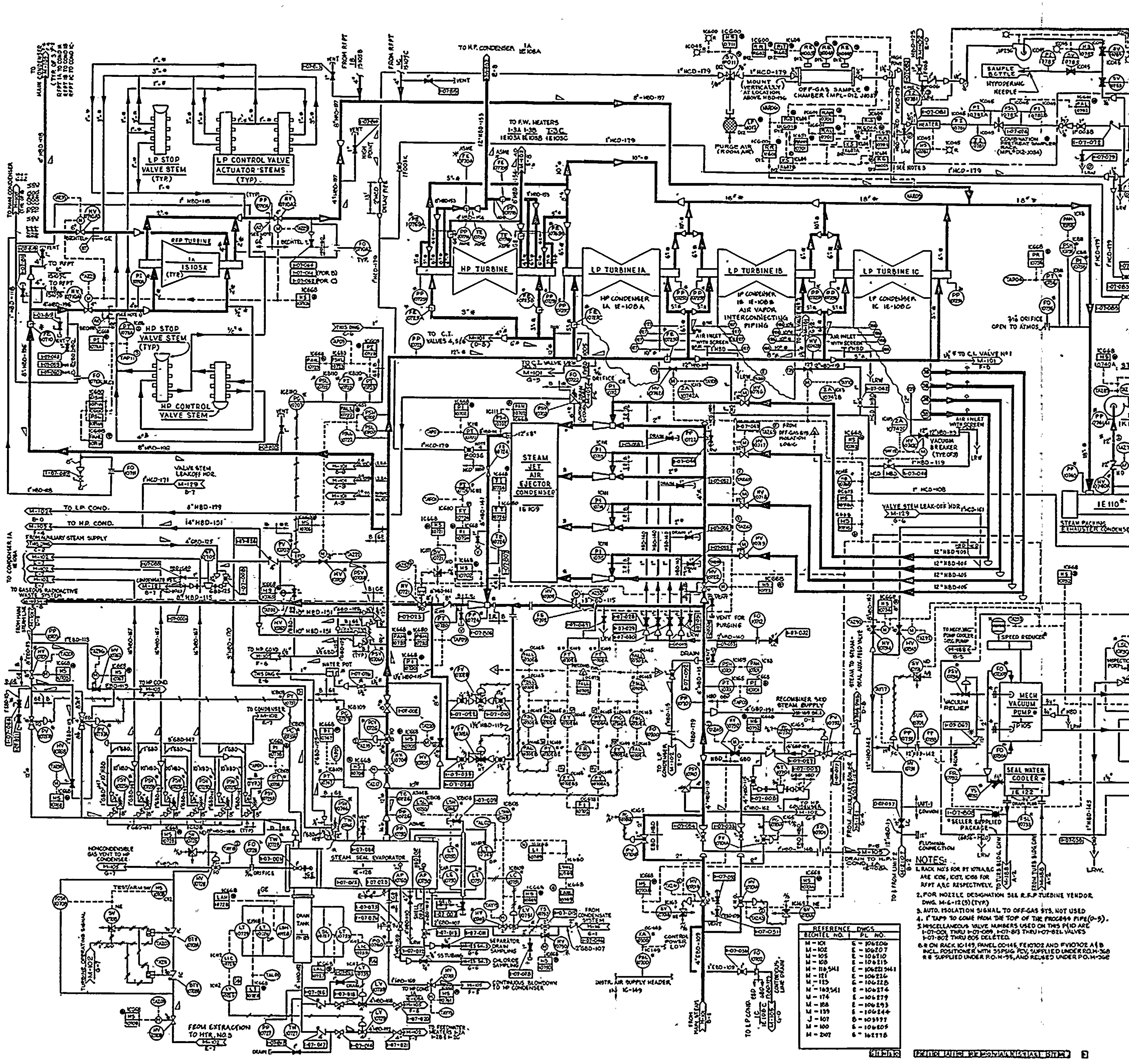
P&ID
FEEDWATER

FIGURE 10.4-5



NOTES:

1. RACK NO'S FOR PT 1071A,B,C ARE 1C06, 1C57, 1C58 FOR RFPT A,B,C RESPECTIVELY.
2. FOR NOZZLE DESIGNATION SEE R.F.P TURBINE VENDOR DWG. M-6-12(5)(TYP.)
3. AUTO. ISOLATION SIGNAL TO OFF-GAS SYS. NOT USED
4. 1" TAPS TO COME FROM THE TOP OF THE PROCESS PIPE(0-5).
5. MISCELLANEOUS VALVE NUMBERS USED ON THIS P&ID ARE 1-07-001 THRU 1-07-069, 1-07-813 THRU 1-07-826. VALVES 1-07-802 THRU 805 DELETED.
6. * ON RACK 1C-149, PANEL 0C-145, FE10702 AND FV10702 A & B INCL. POSITIONER WITH 35PSIG PCV, SUPPLIED UNDER P.O.M-368
** SUPPLIED UNDER P.O.-M-95, AND REUSED UNDER P.O.M-368

REFERENCE DWGS	
BECHTEL NO.	PL NO.
M-101	E-106206
M-102	E-106207
M-105	E-106210
M-108	E-106213
M-116,SH.1	E-106221 SH.1
M-121	E-106226
M-123	E-106228
M-163,SH.1	E-106274
M-174	E-106279
M-128	E-106293
M-139	E-106244
J-107	B-103997
M-100	E-106205
M-2107	E-162778

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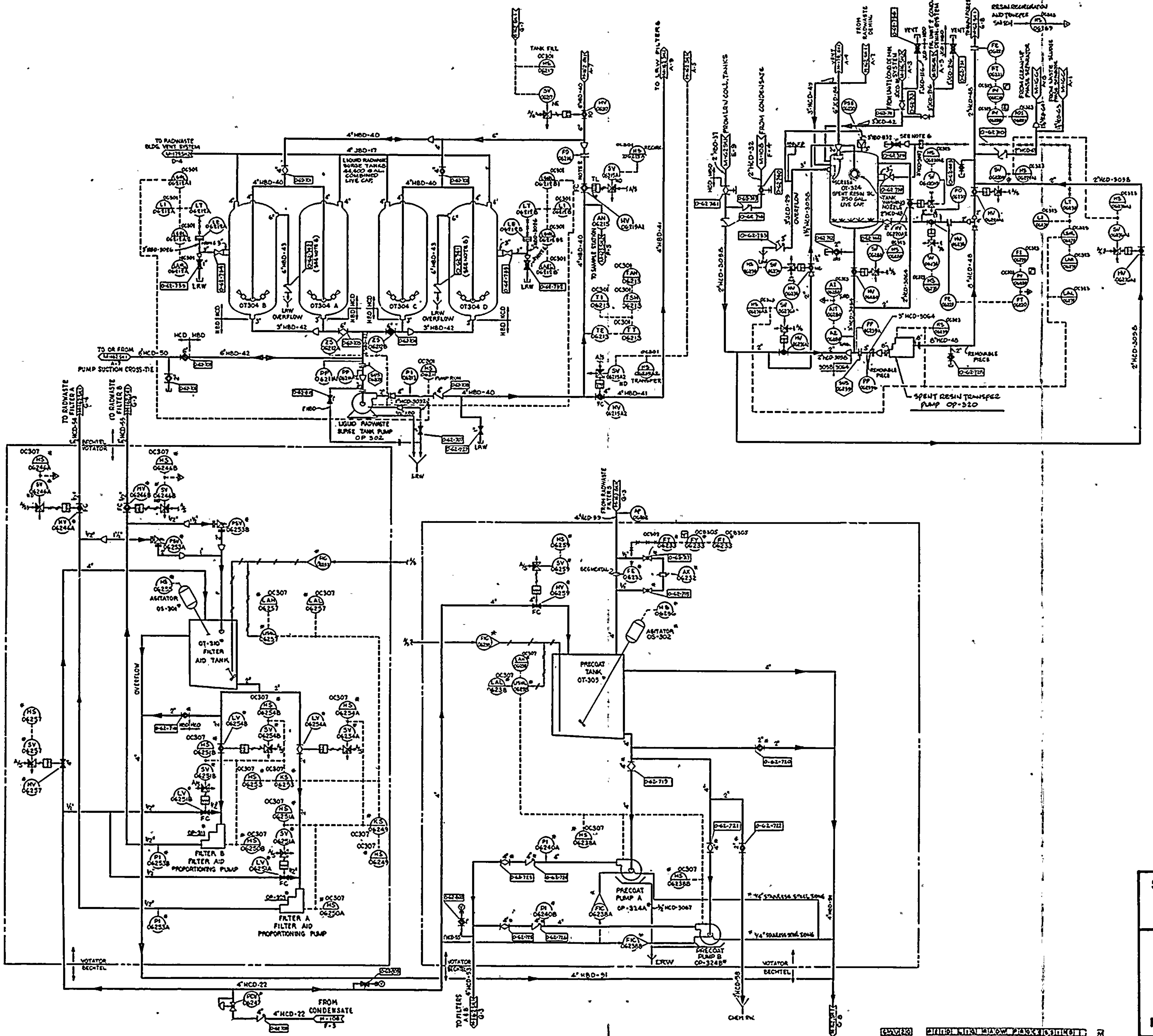
P&ID
AIR REMOVAL AND
SEALING SYSTEM

FIGURE 10.4-9



FOR NOTES SEE SHEET 1

REFERENCE DRAWINGS	
BECHTEL NO.	PL NO.
M-108	E-106213
M-116 SH.2	E-106221 SH.2
M-123 SH.2	E-106228 SH.2
M-125 SH.3	E-106230 SH.
M-162 SH.1	E-106287 SH.1
M-166	E-106271
M-179 SH.2	E-106284 SH.2
J-162	B-103462
J-462	D-103749

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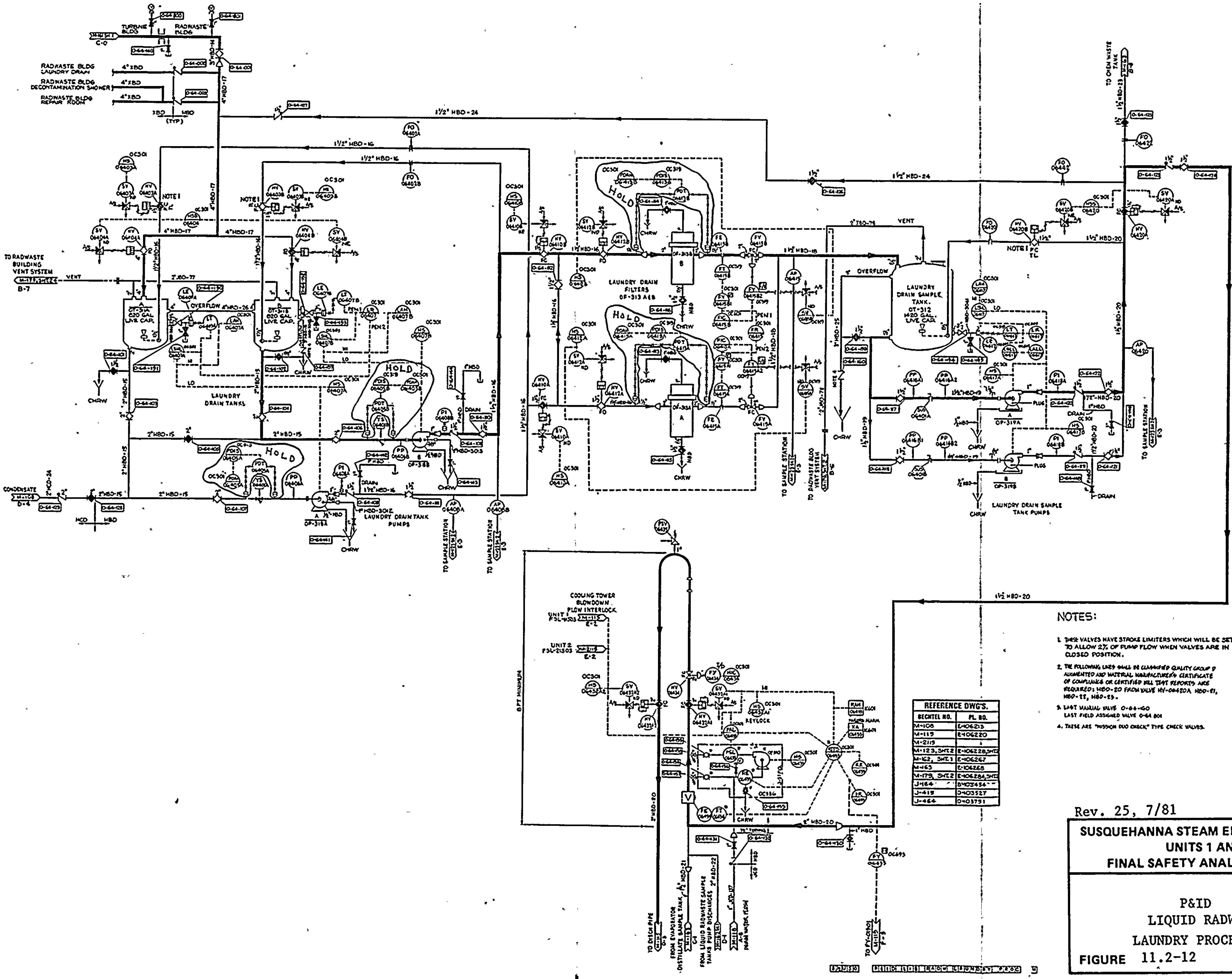
P&ID
LIQUID RADWASTE
PROCESSING

FIGURE 11.2-10



NOTES:

1. THESE VALVES HAVE STROKE LIMITERS WHICH WILL BE SET TO ALLOW 2% OF PUMP FLOW WHEN VALVES ARE IN CLOSED POSITION.
2. THE FOLLOWING LINES SHALL BE CLASSIFIED QUALITY GROUP D AUGMENTED AND MATERIAL MANUFACTURER'S CERTIFICATE OF COMPLIANCE OR CERTIFIED MILL TEST REPORTS ARE REQUIRED: HBD-20 FROM VALVE HV-06420A HBD-21, HBD-22, HBD-23.
3. LAST MANUAL VALVE 0-64-160
   LAST FIELD ASSIGNED VALVE 0-64 801
4. THESE ARE "MISSION DUO CHECK" TYPE CHECK VALVES.

REFERENCE DWG'S.	
BECHTEL NO.	PL. NO.
M-108	E-106215
M-119	E-106220
M-2119	
M-123, SHT.2	E-106228, SHT.2
M-162, SHT.1	E-106267
M-163	E-106268
M-179, SHT.2	E-106284, SHT.2
J-164	D-105434
J-415	D-103527
J-464	D-103751

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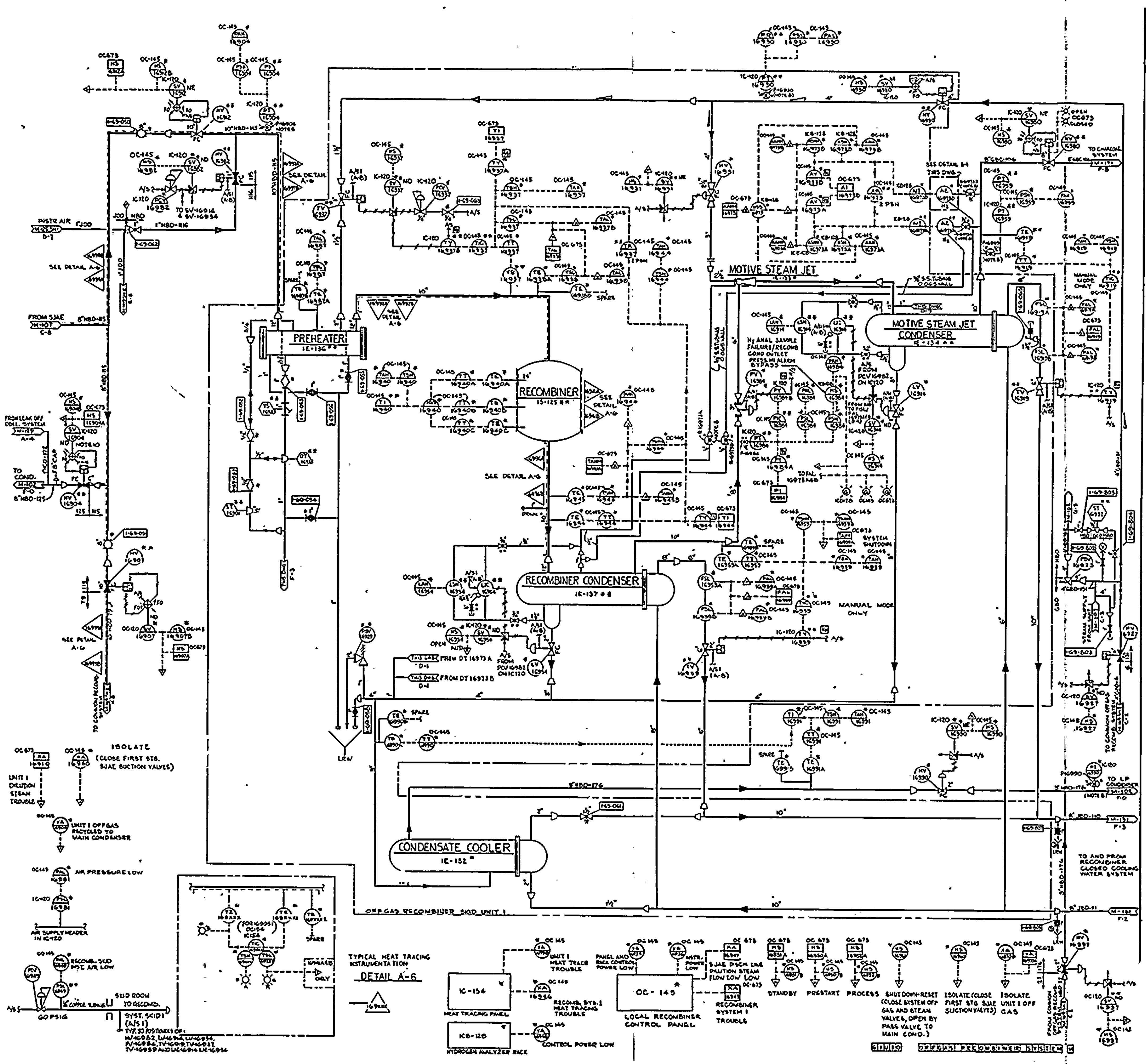
P&ID
LIQUID RADWASTE
LAUNDRY PROCESSING

FIGURE 11.2-12



NOTES

1) * DENOTES COMPONENTS SUPPLIED BY HELIX PROCESS SYSTEMS UNDER P.O. M-368

** DENOTES COMPONENTS SUPPLIED BY AIRCO UNDER P.O. M-95 AND REUSED UNDER P.O. M-368.

2) VALVE STEM LEAK-OFF CONNECTIONS SHALL BE CONNECTED TO THE LEAK-OFF COLLECTION SYSTEM, P&ID M-129.

3) BACKING RINGS SHALL NOT BE USED IN ANY WELD. SHEET 306 OF DRAWING 8856-M-199, PIPING CLASS SHEET, SHALL BE USED FOR WELD END PREPARATIONS.

4) THE FOLLOWING LINES SHALL BE CLASSIFIED ANSI B31.1 (QUALITY GROUP D AUGMENTED) AND MATERIAL MANUFACTURER'S CERTIFICATES OF COMPLIANCE OR MILL TEST REPORTS ARE REQUIRED: HBD-115 FROM AND INCLUDING VALVE 1-69-050 TO PREHEATER INLET AND TO VALVE HV-16982, AND HBD-115 FROM VALVE 1-69-051 TO HV-16907; HBD-116 FROM AND INCLUDING VALVE 1-69-062 TO VALVE HV-16982; HBD-176 FROM 1E-152 TO VALVE HV-16990; HBD-79; ALL ON-SKID PIPING.

5) ALL TE'S ARE SINGLE ELEMENT UNLESS OTHERWISE NOTED.

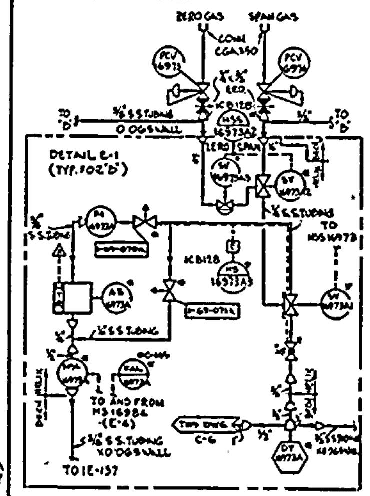
6) LINE GBC-106 WAS INITIALLY SPECIFIED AS ASME SECTION III, CLASS 3 (QUALITY GROUP C). HOWEVER, THIS LINE ONLY NEEDS TO BE INSTALLED PER ANSI B31.1 (QUALITY GROUP D AUGMENTED) REQUIREMENTS

7) ON-SKID PIPING IS DESIGNED BY HELIX PROCESS SYSTEMS UNDER P.O. M-368 AND IS PURCHASED AND INSTALLED BY BECHTEL.

8) 1/2" KEROTEST ROOT VALVES WITH DIAPHRAGM STEM SEALS ARE PROVIDED BY HELIX UNDER P.O. M-368 DO NOT USE STANDARD ROOT VALVES.

9) HELIX SUPPLIED VALVES WITH POSITIONERS AND REUSED LIC'S ARE PROVIDED WITH PCV'S BY HELIX TO BE SET AT 35 PSIG.

10) PROVIDE SEPARATE AIR SUPPLY FOR SV-16906 TO RACK 1C-120



REFERENCE DRAWINGS	
BECHTEL NO.	PL NO.
M-100	E-106205
M-102	E-106207
M-107	E-106212
M-129	E-106234
M-125, SH. 2	E-106230-2
M-131	E-106236
M-171	E-106276
M-169 SH 2	E-106274-2

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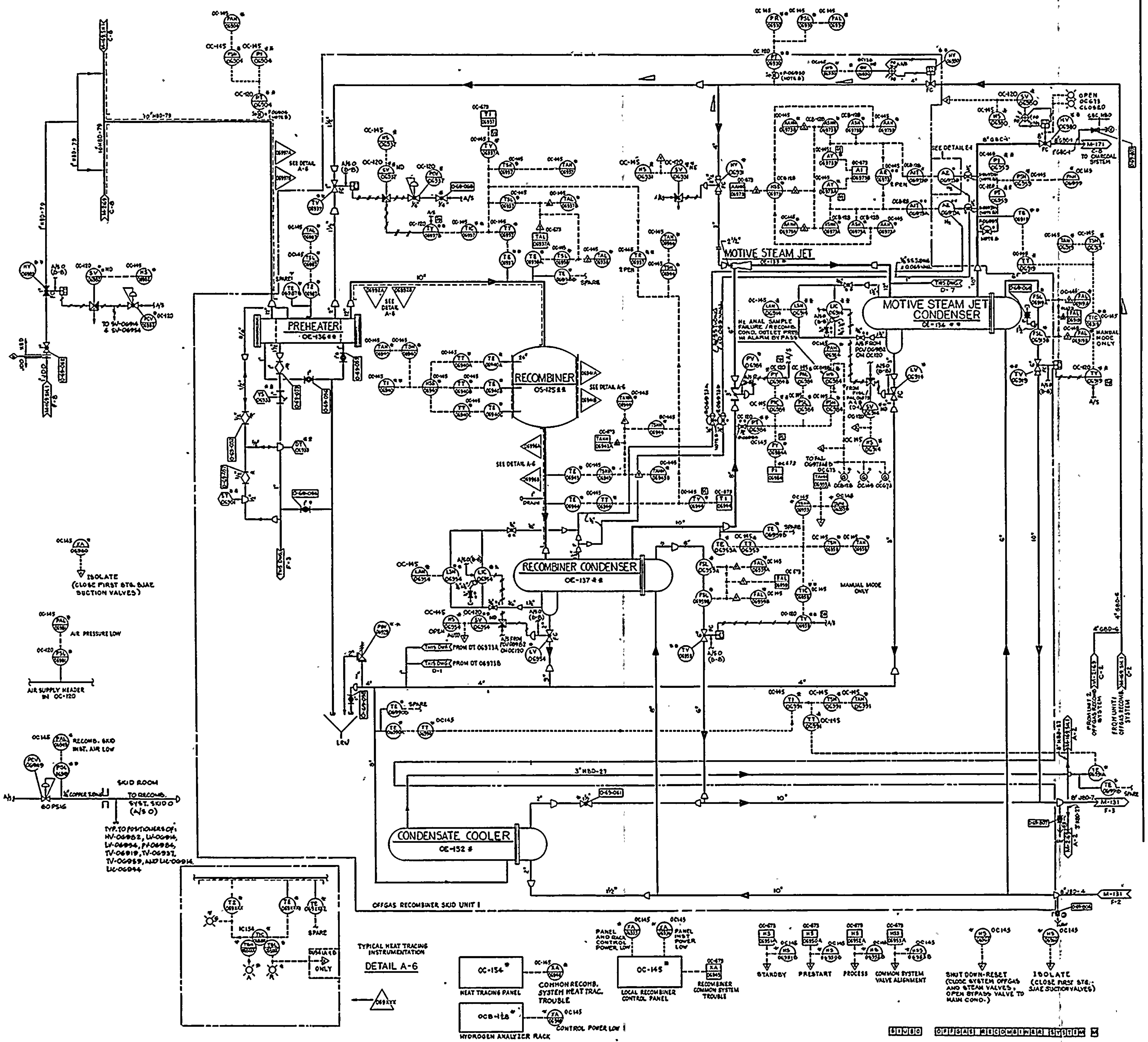
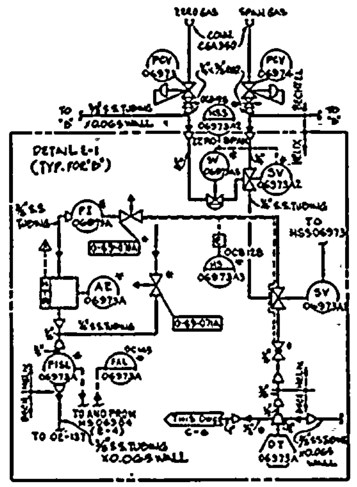
P&ID
UNIT 1 OFFGAS
RECOMBINER SYSTEM

FIGURE 11.3-3a



NOTES

1) # DENOTES COMPONENTS SUPPLIED BY HELIX PROCESS SYSTEM UNDER P.O. M-368
   # # DENOTES COMPONENTS SUPPLIED BY AIRCO UNDER P.O. M-95. AND REUSED UNDER P.O. M-368
2) VALVE STEM LEAK-OFF CONNECTIONS SHALL BE CONNECTED TO THE LEAK OFF COLLECTION SYSTEM P&ID M-49.
3) BACKING RINGS SHALL NOT BE USED IN ANY WELD. SHEET 306 OF DRAWING 8856-M-199, PIPING CLASS SHEET, SHALL BE USED FOR WELD END PREPARATIONS.
4) THE FOLLOWING LINES SHALL BE CLASSIFIED ANSI B31.1 (QUALITY GROUP D AUGMENTED) AND MATERIAL MANUFACTURER'S CERTIFICATES OF COMPLIANCE OR MILL TEST REPORTS ARE REQUIRED: HBD-79; HBD-27, ALL ON-SKID PIPING.
5) ALL TE'S ARE SINGLE ELEMENTS UNLESS OTHERWISE NOTED.
6) LINE GBC-1 WAS INITIALLY SPECIFIED AS ASME SECTION III, CLASS 3 (QUALITY GROUP C), HOWEVER THIS LINE ONLY NEEDS TO BE INSTALLED PER ANSI B31.1 (QUALITY GROUP D AUGMENTED) REQUIREMENTS.
7.) ON-SKID PIPING IS DESIGNED BY HELIX PROCESS SYSTEMS UNDER P.O. M-368, AND PURCHASED AND INSTALLED BY BECHTEL.
8.) ½" KEROTEST ROOT VALVES WITH DIAPHRAGM STEM SEALS ARE PROVIDED BY HELIX UNDER P.O. M-368. DO NOT USE STANDARD ROOT VALVES.
9) HELIX SUPPLIED VALVES WITH POSITIONERS ARE REUSED LIC'S ARE PROVIDED WITH PCV'S BY HELIX TO BE SET AT 25 PSIG.
10) LAST SEQUENTIAL VALVE NUMBER USED IS 0-69-071.



REFERENCE DRAWINGS	
BECHTEL NO.	PL NO.
M-100	E-106205
M-121	E-106226
M-125,SH-2	E-106230-2
M-131	E-106236
M-169,SH.1	E-106274-1
M-2169	E-176970
M-129	E-106234
M-2129	E-162652

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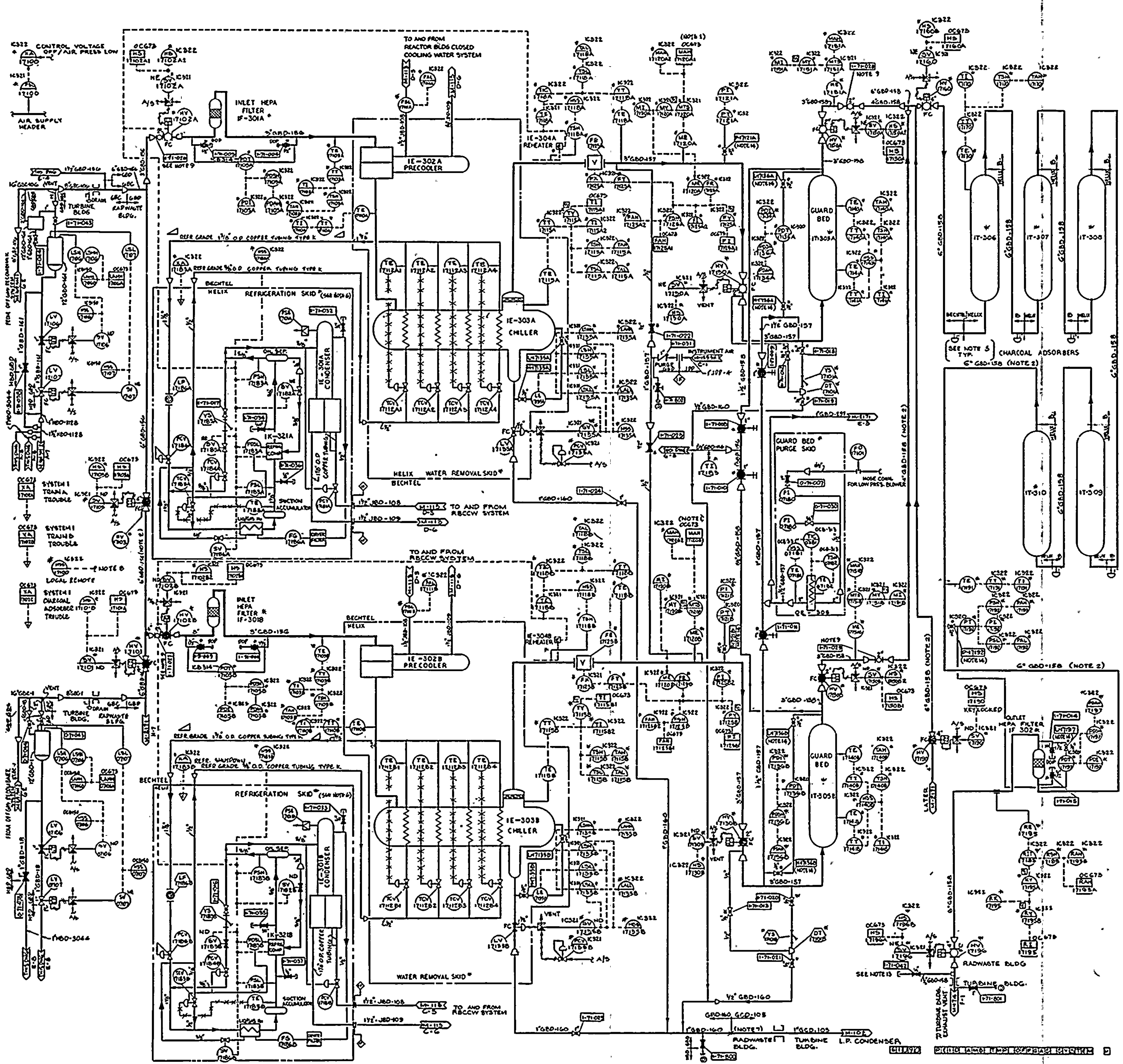
P&ID
UNIT 1 OFFGAS
RECOMBINER SYSTEM

FIGURE 11.3-3b



NOTES.

1) CONTROLS AND INSTRUMENTATION LOCATED IN THE MAIN CONTROL ROOM WILL NOT BE FURNISHED BY HELIX PROCESS SYSTEMS.
2) ALL PROCESS PIPING 4"GBD AND LARGER SHALL BE SCHEDULE 80.
3) BACKING RINGS SHALL NOT BE USED IN ANY WELD. SHEET 306 OF DRAWING 8856-M-189, PIPING CLASS SHEET, SHALL BE USED FOR WELD-END PREPARATIONS.
4) LAST SEQUENTIAL VALVE NUMBER USED IS 1-71-044.
5) 1" DRAIN BY BECHTEL WELD PLUG AFTER SYSTEM PRESSURE TEST.
6) THE REFRIGERATION UNITS FOR SUBTRAINS A & B ARE MOUNTED ON A COMMON SKID.
7) PIPE GCD-105 IS ALREADY PARTIALLY INSTALLED AND THEREFORE REUSED FOR THIS SYSTEM.
8) HS 17100 SWITCHES CONTROL FROM LOCAL TO CONTROL ROOM FOR ALL ISOLATION VALVES WITH LOCAL AND CONTROL ROOM SWITCHES.
9) MANUAL VALVES 1-71-026 THRU 029 SHALL BE 600# RATED WITH DOUBLE STEM PACKING AND SEAL AIR CONNECTION.
10) ALL BALL VALVES EXCEPT 1-71-003 THRU -006 AND 1-71-014 AND -015 SHALL BE SUPPLIED WITH INSTRUMENT AIR TO THE STEM SEAL LEAK-OFF CONNECTION. A SEPARATE SUPPLY FOR EACH GROUP OF VALVES IN A SUBTRAIN AND CHARCOAL TRAIN SHALL BE PROVIDED WITH A SHUT-OFF ROOT VALVE AND AIRSET. AIRSETS TO CONTROL OUTPUT PRESSURE AT 10 PSIG.
11) EXTENDED VALVE STEMS BY BECHTEL.
12) THE FOLLOWING LINES SHALL BE CLASSIFIED QUALITY GROUP D AUGMENTED AND MATERIAL MANUFACTURER'S CERTIFICATE OF COMPLIANCE OR CERTIFIED MILL TEST REPORTS ARE REQUIRED: GBD-18; GBD-156, GBD-157; GBD-158; GBD-160 FROM IE-303 A & B TO VALVES 1-71-024 & 1-71-025 AND FROM LINES 3"GBD-157 TO VALVES 1-71-019 & 1-71-021, GBD-161, GBD-18.
13) GRAB SAMPLE CONNECTION TO BE USED WITH HYPODERMIC NEEDLE AND SAMPLE BOTTLE EVACUATED AT RACK 1C049 (STEAM JET AIR EJECTOR CONDENSER DISCHARGE) KEEP SAMPLE LINE AS SHORT AS POSSIBLE. LOCATE AT COLUMN LINES N4, 11.9 AT EL 694'-6".
14) 1/2" KEROTEST ROOT VALVES WITH DIAPHRAGM STEM SEALS ARE PROVIDED BY HELIX UNDER P.O. M-366. DO NOT USE STANDARD ROOT VALVES.

REFERENCE DRAWINGS	
BECHTEL NO.	PL NO.
M-100	E-106205
M-102	E-106207
M-113	E-106216
M-169	E-106274
M-141 SH 2	E-106266 SH 2
M-174	E-106279
M-2171	E-178979
M-125 SH 1	E-106230 SH 1

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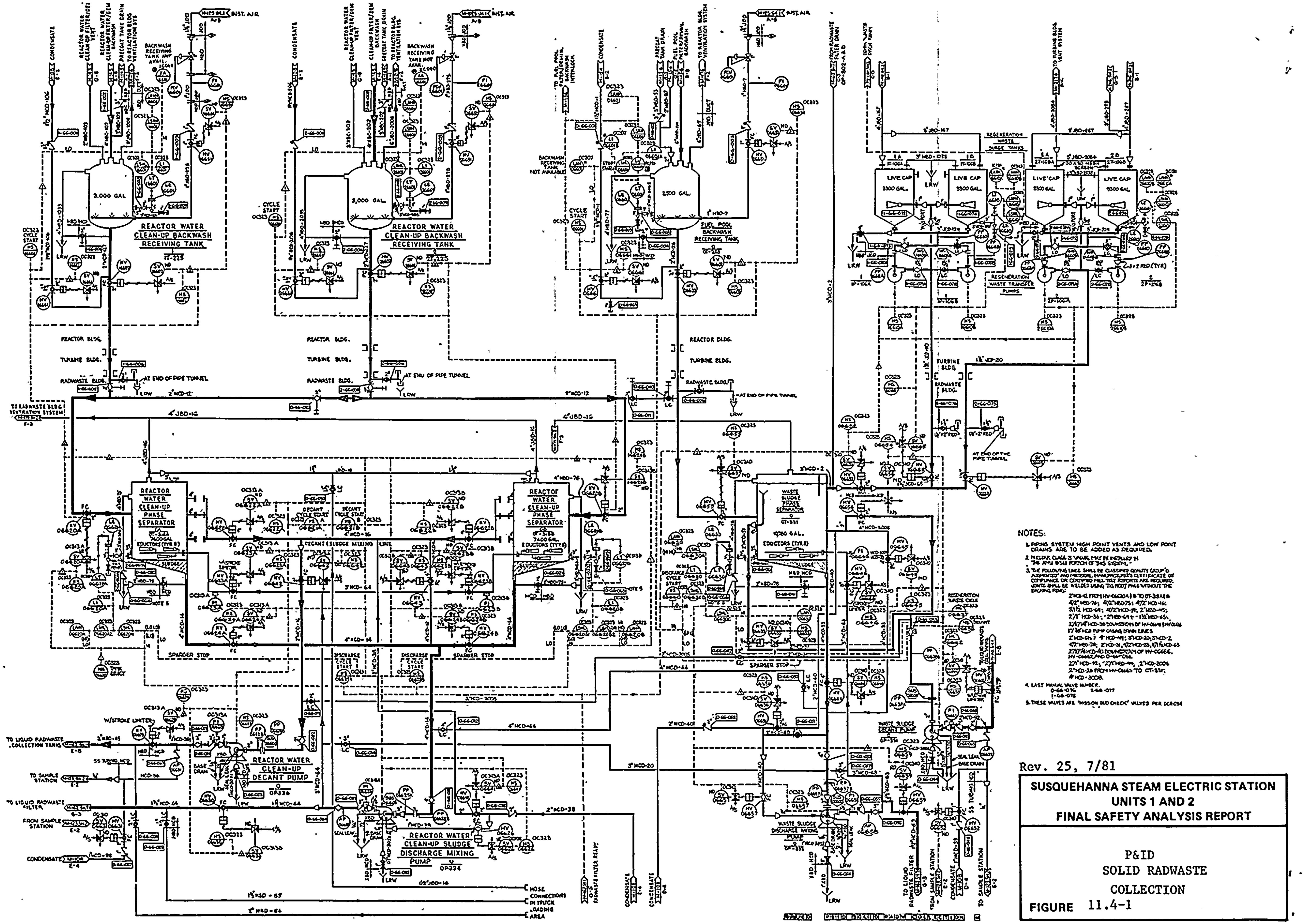
P&ID
AMBIENT TEMPERATURE
CHARCOAL OFFGAS
TREATMENT SYSTEM

FIGURE 11.3-4



NOTES:

1. PIPING SYSTEM HIGH POINT VENTS AND LOW POINT DRAINS ARE TO BE ADDED AS REQUIRED.
2. NUCLEAR CLASS 3 VALVES MAY BE INSTALLED IN THE ANSI B31.1 PORTION OF THIS SYSTEM.
3. THE FOLLOWING LINES SHALL BE CLASSIFIED QUALITY GROUP "D AUGMENTED" AND MATERIAL MANUFACTURER'S CERTIFICATE OF COMPLIANCE OR CERTIFIED MILL TEST REPORTS ARE REQUIRED. JOINTS SHALL BE WELDED USING TIG ROOT PASS WITHOUT THE BACKING RING:
   2"HCD-12 FROM HV-06620A & B TO OT-338A & B
   4/2" HCD-76; 4/2" HBD-75; 4/2" HCD-46;
   3/1½ HCD-64; 4/2" HCD-19; 2" HBD-45;
   2"/1" HCD-36; 2" HCD-64; 1½" HBD-65;
   2"/1"/¾" HCD-38 DOWNSTREAM OF HV-06626 & HV-06626
   1"/¾" HCD PUMP CASING DRAIN LINES
   2" HCD-65; 4" HCD-44; 3" HCD-20; 3" HCD-2
   4/2" HBD-78; 2" HCD-31, 4/2" HCD-23, 3/1½ HCD-63
   2"/1"/¾ HCD-40 DOWNSTREAM OF HV-06666, HV-06667 AND 0-66-051.
   2/1" HCD-92; 2/1" HBD-44; 2" HCD-3005
   2" HCD-26 FROM HV-06665 TO CT-331;
   4" HCD-3008.
4. LAST MANUAL VALVE NUMBER.
   0-66-076 2-66-077
   1-66-078
5. THESE VALVES ARE "HIBSON DUO CHECK" VALVES PER DCR-034

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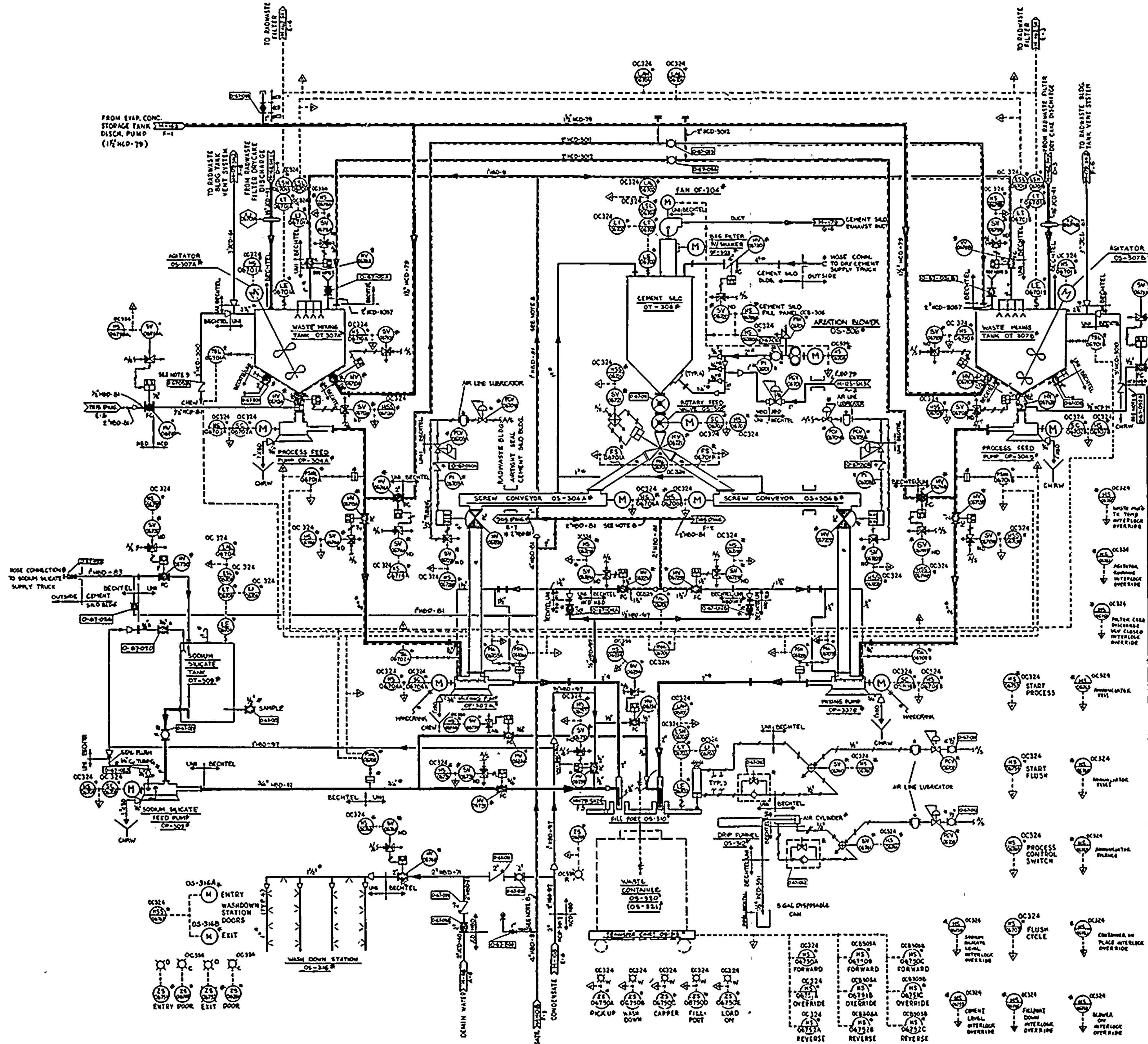
P&ID
SOLID RADWASTE
COLLECTION

FIGURE 11.4-1



NOTES

1. ALL EQUIPMENT, PIPING AND INSTRUMENTATION WITHIN THE SUPPLY BOUNDARIES OR MARKED BY AN ASTERISK IS PROVIDED BY UNITED NUCLEAR INDUSTRIES UNDER PURCHASE ORDER NO. 8856-M-76. VALVE EXTENSION STEMS BY BECHTEL.
2. HEAT TRACING ON TANKS, PUMPS AND VENDOR SUPPLIED PIPING BY UNI, THERMAL INSULATION BY BECHTEL.
3. ALL STATUS LIGHTS ARE PROVIDED WITH UNI SUPPLIED CONTROL PANEL OC-324.
4. PIPING SYSTEM HIGH POINT VENTS AND LOW "X" DRAINS LATER
5. DRILL 2" HOLE IN BLIND FLANGE ON TANK TOP CENTER AND WELD A 2" SCH 40 S PIPE WITH FUNNEL FOR CHEM. FLUSHING
6. LAST MANUAL VALVE 0-67-062A & B {0-67-064 PER D.S.
7. REFERENCE 8856-J-36-1 AND 36-2 DRAWINGS FOR SOLENOID VALVE LOCATION; ALSO REFER TO 8856-M74 VENDOR DWGS.
8. HBD-81 IS HEAT TRACED TO FURNISH HOT WATER FLUSH FOR SOLIDIFICATION EQUIPMENT, EXCEPT FOR SODIUM SILICATE FLUSH.
9. THESE VALVES ARE "MISSION DUO-CHEK" CHECK VALVES.

REFERENCE DRAWINGS

BECHTEL NO.	PL NO.
M-108	E-106225
M-118	E-106223
M-125, SH.1	E-106230, SH.1
M-162, SH.1	E-106267, SH.1
M-163	E-106268
M-179, SH.2	E-106284, SH.2

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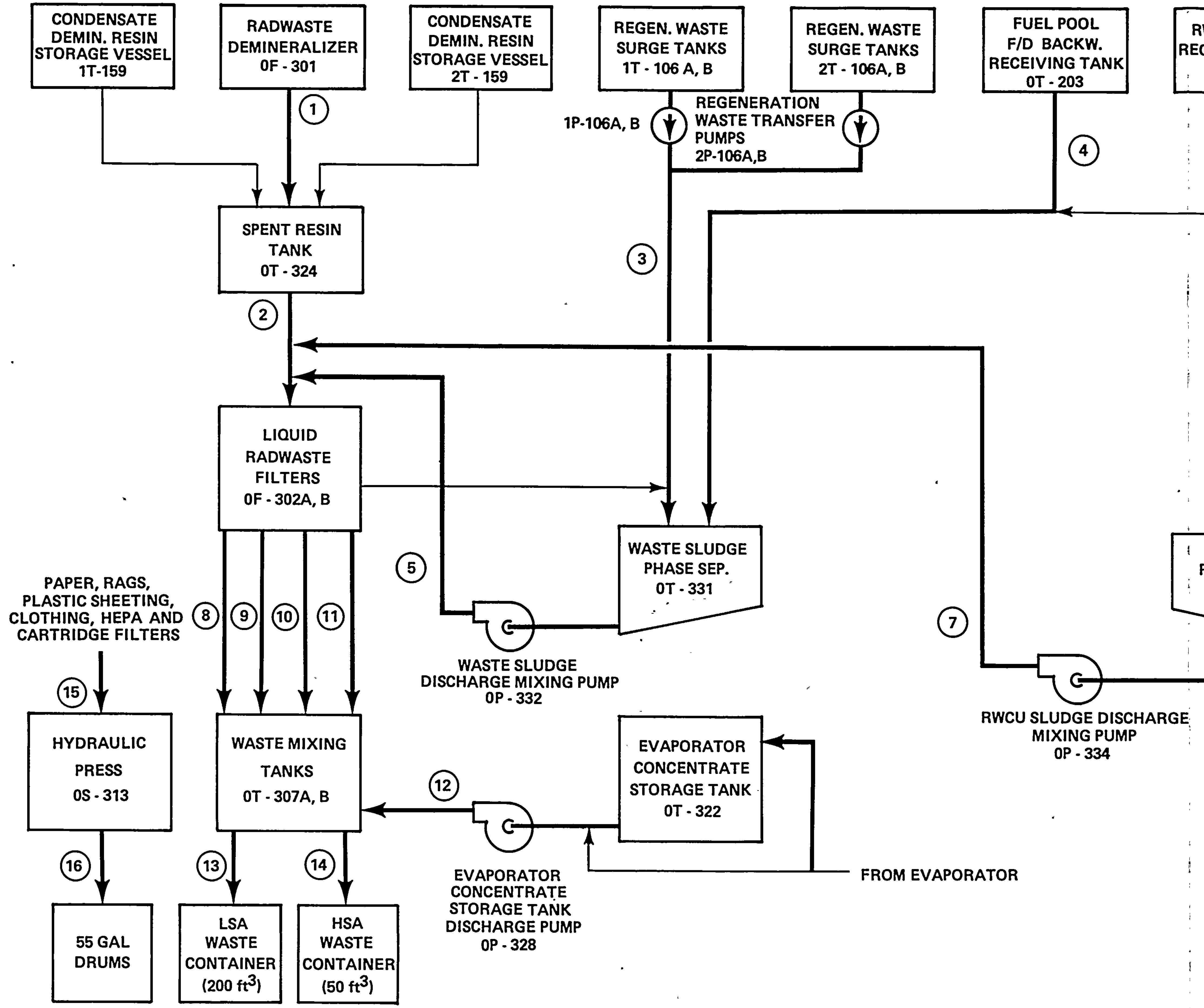
P&ID
UNITS 1 & 2
RADWASTE SOLIDIFICATION

FIGURE 11.4–2



SUSQUEHANNA STEAM ELECTRIC STATION
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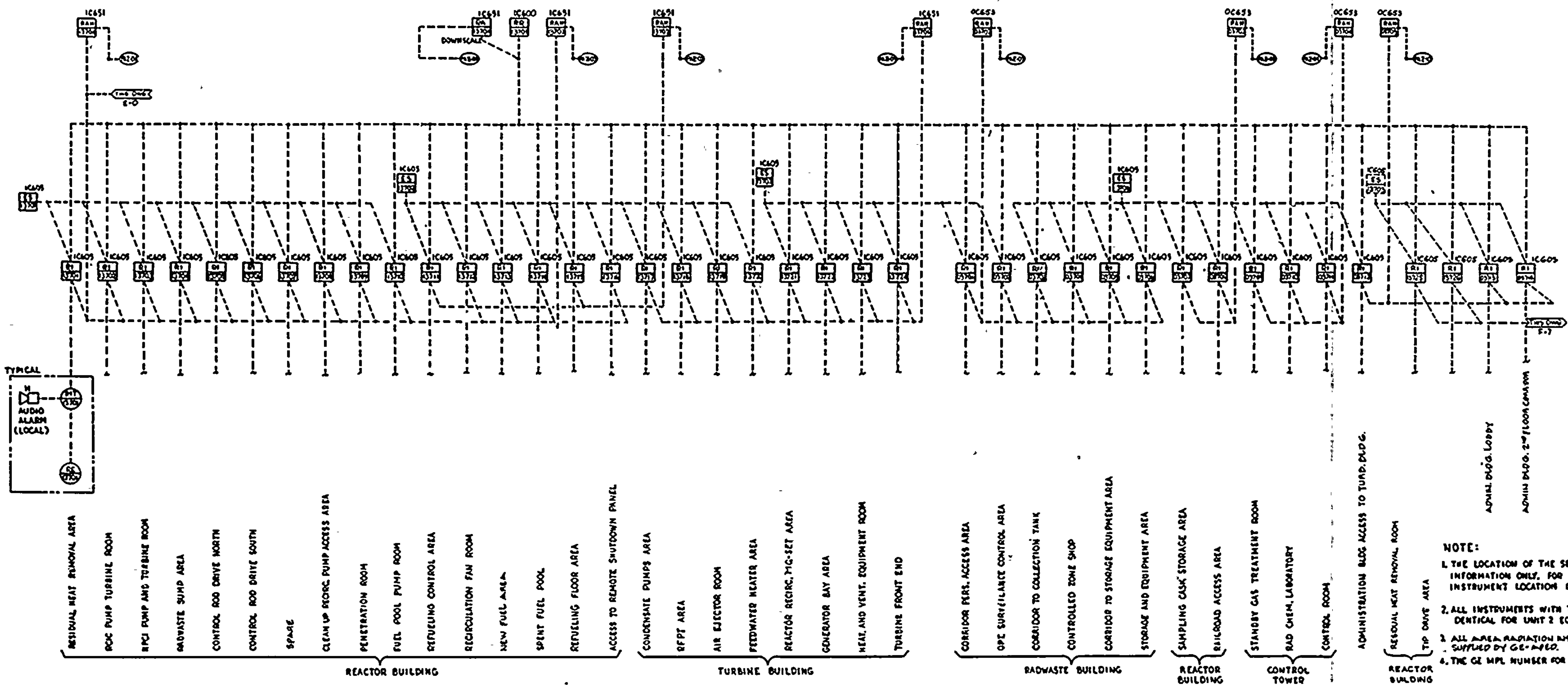
SOLID WASTE MANAGEMENT SYSTEM FLOW DIAGRAM

FIGURE 11.4-3



NOTE:

1. THE LOCATION OF THE SENSOR ELEMENT IS FOR INFORMATION ONLY. FOR EXACT LOCATION SEE INSTRUMENT LOCATION DRAWINGS.
2. ALL INSTRUMENTS WITH THE PREFIX 1 ARE IDENTICAL FOR UNIT 2 EQUIPMENT.
3. ALL AREA RADIATION INSTRUMENTS ARE SUPPLIED BY GE-APED.
4. THE GE MPL NUMBER FOR AREA RADIATION IS D21

CHANNELS PER UNIT

CHANNEL DESIGNATION	INSTR IDENT. NO.	SERVICE	BLDG	UNIT 1 LOC.*	ELEV.	SENSITIVITY	UNIT 2 LOC.*
CHAN. 1	RE-13701	RESIDUAL HEAT REMOVAL AREA	RX	T/22	645'	M	T/31
CHAN. 2	RE-13702	RCIC PUMP TURBINE ROOM	RX	T/21	645'	H	T/30
CHAN. 3	RE-13703	HPCI PUMP AND TURBINE ROOM	RX	S/21	645'	H	S/30
CHAN. 4	RE-13704	RADWASTE SUMP AREA	RX	S/20	645'	M	S/36
CHAN. 5	RE-13705	CONTROL ROD DRIVE NORTH	RX	R/21	719'	M	R/30
CHAN. 6	RE-13706	CONTROL ROD DRIVE HYDR. UNIT SOUTH	RX	R/29	719'	M	R/37
CHAN. 7	RE-13707	SPARE					
CHAN. 8	RE-13708	CLEAN UP RECIRC. PUMP ACCESS AREA	RX	R/21	749'	H	R/37
CHAN. 9	RE-13709	PENETRATION ROOM	RX	T/25	749'	M	T/33
CHAN. 10	RE-13710	FUEL POOL PUMP ROOM	RX	Q/28	749'	M	Q/30
CHAN. 11	RE-13711	REFUELING CONTROL AREA	RX	Q/27	779'	H	Q/31
CHAN. 12	RE-13712	RECIRCULATION FAN ROOM	RX	U/29	779'	H	SPARE
CHAN. 13	RE-13713	NEW FUEL AREA	RX	Q/27	779'	H	Q/31
CHAN. 14	RE-13714	SPENT FUEL POOL	RX	S/27	818'	M	S/31
CHAN. 15	RE-13715	REFUELING FLOOR AREA	RX	M/25	818'	H	M/33
CHAN. 16	RE-13716	ACCESS TO REMOTE SHUTDOWN PANEL	RX	P/21	670'	H	P/37
CHAN. 17	RE-13717	CONDENSATE PUMPS AREA	TB	J/26	676'	H	J/31
CHAN. 18	RE-13718	RFPT AREA	TB	L/21	676'	H	L/31
CHAN. 19	RE-13719	AIR EJECTOR ROOM	TB	H/25	682'	M	H/32
CHAN. 20	RE-13720	FEEDWATER HEATER AREA	TB	L/20	699'	H	L/38
CHAN. 21	RE-13721	REACTOR RECIRC. MG-SET AREA	TB	M/20	729'	H	M/38
CHAN. 22	RE-13722	GENERATOR BAY AREA	TB	G/26	729'	H	G/32
CHAN. 23	RE-13723	HEAT. AND VENT. EQUIPMENT ROOM	TB	L/23	762'	H	L/35
CHAN. 24	RE-13724	TURBINE FRONT END	TB	G/16	729'	H	G/42
CHAN. 25	RE-13725	RESIDUAL HEAT REMOVAL RM.	RX	T/24	645'	M	T/33
CHAN. 26	RE-13726	TIP DRIVE AREA	RX	Q/22	719'	M	Q/31
CHAN. 27	RE-03713	ADMIN. BLDG. LOBBY (GRD. LEVEL)	ADMIN BLDG	P/11	676'	H	SPARE
CHAN. 28	RE-03714	ADMIN. BLDG. 2ND FLOOR CORRIDOR	ADMIN BLDG	P/11	691'	H	SPARE

* THIS IS THE APPROXIMATE LOCATION. FOR FINAL LOCATION SEE SPECIFIC INSTRUMENT LOCATION DRAWINGS.

SENSOR SENSITIVITY RANGES

H	HIGH SENSITIVITY	.01 TO 100 mr/h
M	MEDIUM SENSITIVITY	0.1 TO 1000 mr/h
L	LOW SENSITIVITY	1 TO 10,000 mr/h
LL	VERY LOW SENSITIVITY	100 TO 1000,000 mr/h

METEOROLOGICAL TOWER INSTRUMENTS

SENSOR	SIGNAL CONDITIONER	TELEMETER TRANSMITTER	SERVICE DESCRIPTION	TELEMETER RECEIVER	CONVERTER	PANEL	RECORDER	PANEL
XSE-03701	XST-03701	XTT-03701	WIND SPEED 300'	XTR-03701	XY-03701	OC-693	XR-03701	C693
XXE-03702	XXT-03702	XTT-03702	WIND DIRECTION 300'	XTR-03702	XY-03702	OC-693	XR-03701	C693
XSE-03703	XST-03703	XTT-03703	WIND SPEED 30'	XTR-03703	XY-03703	C-680	XR-03703	C653
XXE-03704	XXT-03704	XTT-03704	WIND DIRECTION 30'	XTR-03704	XY-03704	C-680	XR-03703	C693
XTE-03705	XTT-03705A	XTT-03705B	AMB. TEMP 30'	XTR-03705	XY-03705	OC-693	XR-03704	C693
XME-03706	XMT-03706	XTT-03706	DEWPOINT 30'	XTR-03706	XY-03706	OC-693	XR-03701	C693
NOTE 1	XDT-03707	XTT-03707	Δ TEMP. 30'-300'	XTR-03707	XY-03707	C-680	XR-03703	C653
	XDT-03708	XTT-03708	Δ TEMP 30'-100'	XTR-03708	XY-03708	OC-693	XR-03704	C693
	XXT-03709	XTT-03709	SIGMA 300'	XTR-03709	XY-03709	OC-693	XR-03702	C693
	XXT-03710	XTT-03710	SIGMA 30'	XTR-03710	XY-03710	OC-693	XR-03702	C693
XLE-03711	XLT-03711	XTT-03711	RAIN FALL (GAL.)	XTR-03711	XY-03711	OC-693	XR-03702	C693

NOTE

1. SENSORS FOR THESE SIGNALS ARE OUTPUTS FROM SIGNAL CONDITIONERS CONTAINED IN THE "CLIMET TRANSLATOR", CONNECTIONS ARE INTERNAL.

COMMON PLANT CHANNELS

CHANNEL DESIGNATION	INSTR. IDENT NO.	SERVICE	BLDG	LOC.	ELEV.	SENSITIVITY
CHAN. 29	RE-03701	CORR. PERS. ACCESS AREA	RW	K/3	646'	M
CHAN. 30	RE-03702	OPE. SURVEILANCE CONTROL AREA	RW	G/8	646'	M
CHAN. 31	RE-03703	CORRIDOR TO COLLECTION TANK	RW	J/12	646'	M
CHAN. 32	RE-03704	CONTROLLED ZONE SHOP	RW	K/12	676'	M
CHAN. 33	RE-03705	CORR. TO STORAGE EQUIPMENT AREA	RW	J/9	676'	M
CHAN. 34	RE-03706	STORAGE AND EQUIPMENT AREA	RW	G/6	676'	M
CHAN. 35	RE-03707	SAMPLING CASK STORAGE AREA	RX	S/29	818'	H
CHAN. 36	RE-03708	RAILROAD ACCESS AREA	RX	U/29	670'	H
CHAN. 37	RE-03709	STANDBY GAS TREATMENT ROOM	CTR TWR	K/27	806'	H
CHAN. 38	RE-03710	RAD CHEM. LABORATORY	CTR TWR	M/27	676'	H
CHAN. 39	RE-03711	CONTROL ROOM	CTR TWR	L/29	728'	H
CHAN. 40	RE-03712	ADMIN. BLDG. ACCESS (SPARE)	ADM BLDG	N/12	676'	H

INTAKE STRUCTURE AND ENVIRONMENTAL INSTRUMENTATION

SIGNAL FROM	SERVICE DESCRIPTION	TELEMETRY RECEIVER	PANEL	COMPUTOR POINT
XIT-02805A	PUMP OP505A MOTOR CURRENT	XIR-03720	OC671	WAI-10
XIT-02805B	PUMP OP505B MOTOR CURRENT	XIR-03721	OC671	WAI-20
XIT-02805C	PUMP OP505C MOTOR CURRENT	XIR-03722	OC671	WAI-30
XIT-02805D	PUMP OP505D MOTOR CURRENT	XIR-03723	OC671	WAI-40
XTT-01506	DISCHARGE TEMP	XTR-01506	OC671	VDT-01
XLT-02808	INTAKE PUMP CHAMBER A LEVEL	XLR-03715	OC671	VIL-02
XTT-02868	RIVER TEMPERATURE	XTR-03724	OC671	VIT-01
XLT-02868	RIVER LEVEL	XLR-03725	OC671	VIL-01
XLT-02809	INTAKE PUMP CHAMBER B LEVEL	XLR-03718	OC671	VIL-03

TURBINE BUILDING BAROMETRIC PRESSURE

PRESSURE TRANSMITTER	COMPUTER CABINET	COMPUTER POINT
PT13701	1C616	TAP13

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P&ID
AREA RADIATION
MONITORING AND
METEOROLOGICAL INSTRUMENTS

FIGURE 12.3-29