TABLE OF CONTENTS

CHAPTER 11

RADIOACTIVE WASTE MANAGEMENT SYSTEM

Section	Title	Page
11.1	SOURCE TERMS	11.1-1
11.1.1	FISSION PRODUCTS	11.1-1
11.1.2	CORROSION PRODUCTS	11.1-6
11.1.3	TRITIUM PRODUCTION	11.1-7
11.1.4	NITROGEN-16 PRODUCTION	11.1-7
11.1.5	FUEL EXPERIENCE	11.1-12
11.1.6	LEAKAGE SOURCES	11.1-12
11.2	LIQUID WASTE SYSTEMS	11.2-1
11.2.1	DESIGN BASES	11.2-1
11.2.2	SYSTEM DESCRIPTION	11.2-1
11.2.2.1	Boron Recovery System	11.2-1
11.2.2.2	Liquid Waste System	11.2-5
11.2.3	OPERATING PROCEDURES	11.2-30
11.2.3.1	Boron Recovery System	11.2-30
11.2.3.2	Liquid Waste Processing	11.2-31
11.2.4	PERFORMANCE TESTS	11.2-31
11.2.5	ESTIMATED RELEASES	11.2-32
11.2.6	RELEASE POINTS	11.2-32
11.2.7	DILUTION FACTORS	11.2-32
11.2.8	ESTIMATED DOSES	11.2-32
11.3	GASEOUS WASTE SYSTEM	11.3-1
11.3.1	DESIGN BASES	11.3-1
11.3.2	SYSTEM DESCRIPTION	11.3-1

TABLE OF CONTENTS

Section	Title	Page
11.3.2.1	Waste Gas Processing Systems	11.2-1
11.3.3	OPERATING PROCEDURE	11.3-10
11.3.4	PERFORMANCE TESTS	11.3-11
11.3.5	ESTIMATED RELEASES	11.3-11
11.3.6	RELEASE POINTS	11.3-14
11.3.7	DILUTION FACTORS	11.3-14
11.3.8	ESTIMATED DOSES	11.3-14
11.4	PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS	11.4-1
11.4.1	DESIGN BASES	11.4-1
11.4.2	CONTINUOUS MONITORING	11.4-2
11.4.2.1	Process Radiation Monitor	11.4-2
11.4.2.2	Gaseous Discharge Monitor	11.4-4
11.4.2.3	Liquid Waste Discharge Monitor	11.4-5
11.4.2.4	Steam Generator Blowdown Monitor	11.4-7
11.4.2.5	Condenser Air Ejector Monitor	11.4-8
11.4.2.6	Component Cooling Water Monitor	11.4-9
11.4.3	SAMPLING	11.4-9
11.4.3.1	Preconcentrator and Waste Filters	11.4-14
11.4.3.2	Flash Tank	11.4-15
11.4.3.3	Preconcentrator Ion Exchanger	11.4-15
11.4.3.4	Boric Acid Concentrator	11.4-17
11.4.3.5	Boric Acid Condensate Ion Exchanger	11.4-18
11.4.3.6	Boric Acid Condensate Tank	11.4-18
11.4.3.7	Circulating Water Discharge	11.4-18
11.4.3.8	Boric Acid Holding Tank	11.4-19

TABLE OF CONTENTS

Section	Title	Page
11.4.3.9	Equipment Drain Tank	11.4-19
11.4.3.10	Chemical Drain Tank	11.4-19
11.4.3.11	Laundry Drain Tank	11.4-19
11.4.3.12	Waste Concentrator	11.4-19
11.4.3.13	Waste Ion Exchanger	11.4-20
11.4.3.14	Waste Condensate Tank	11.4-20
11.4.3.15	Gas Analyzer	11.4-20
11.4.4	CALIBRATION AND MAINTENANCE	11.4-21
11.5	SOLID WASTE SYSTEM	11.5-1
11.5.1	DESIGN BASES	11.5-1
11.5.2	SYSTEM DESCRIPTION	11.5-2
11.5.3	EQUIPMENT DESCRIPTION	11.5-11
11.5.4	EXPECTED VOLUMES	11.5-14
11.5.5	PACKAGING	11.5-14
11.5.6	STORAGE FACILITIES	11.5-14
11.5.7	SHIPMENT	11.5-15
11.5.8	INSTRUMENT APPLICATION	11.5-15
11.6	OFF-SITE RADIOLOGICAL MONITORING PROGRAM	11.6-1
11.6.1	EXPECTED BACKGROUND	11.6-1
11.6.2	CRITICAL PATHWAYS	11.6-1
11.6.3	SAMPLING MEDIA, LOCATIONS AND FREQUENCY	11.6-2
11.6.4	ANALYTICAL SENSITIVITY	11.6-2
11.6.5	DATA ANALYSIS AND PRESENTATION	11.6-4
11.6.6	PROGRAM STATISTICAL SENSITIVITY	11.6-4

LIST OF TABLES

Table	Title	Page
11.1-1	Reactor Coolant Specific Activity	11.1-4
11.1-2	Bases for Reactor Coolant Radioactivity	11.1-5
11.1-3	Crud Specific Activity-Operating Reactors	11.1-8
11.1-4	Long-Lived Iostopes in Crud	11.1-10
11.1-5	Crud Core Residence Time and Specific Activity	11.1-10
11.1-6	Sources of Tritium Production	11.1-11
11.1-7	Nitrogen-16 Production Parameters	11.1-11
11.2-1	Sources and Volumes of Liquid Waste	11.2-2
11.2-2	Design Data for Boron Recovery System Components	11.2-2
11.2-3	Boron Recovery System Process Flow Data	11.2-8
11.2-4	Expected Filter and Ion Exchanger Performance	11.2-10
11.2-5	Boron Recovery System Performance Data	11.2-11
11.2-6	Boron Recovery System Maximum Nuclide Concentrations During Normal Operations	11.2-12
11.2-7	Boron Recovery System Maximum Nuclide Concentrations During Anticipated Operations	11.2-14
11.2-8	Design Data for Liquid Waste System Components	11.2-19
11.2-9	Liquid Waste System-Pressure, Temperature and Flow Data	11.2-21
11.2-10	Liquid Waste System Expected Performance	11.2-23
11.2-11	Liquid Waste System Normal Activity Distribution	11.2-24

LIST OF TABLES (CONT'D)

Table	Title	Page
11.2-12	Liquid Waste System Anticipated Operational Occurrence Activity Distribution	11.2-26
11.2-13	Assumptions Used in Calculating Estimated Normal and Anticipated Operational Occurrence Releases	11.2-28
11.2-14	Estimated Normal and Anticipated Operational Occurrence Liquid Releases	11.2-34
11.2-15	Offsite Doses Due to Plant Liquid Releases	11.2-36
11.3.1	Waste Gas System Flow Data Points	11.2-2
11.3-2	Component Data	11.3-4
11.3-3	Gas Collection Header Source Points	11.3-8
11.3-4	Estimated Normal and Anticipated Operational Occurrence Gaseous Releases	11.3-12
11.3-5	Offsite Doses Due to Plant Gaseous Releases	11.3-15
11.4-1	Process Radiation Monitors	11.4-6
11.4-2	Sample Locations and Analysis	11.4-12
11.5-1	Inputs to Solid Waste System Per Core Cycle Based on Continuous Operation with 1 Percent Failed Fuel	11.5-1
11.5-2	Concentrator Bottoms Input Activity to Solid Waste System Per Core Cycle	11.5-3
11.5-3	Concentrator Bottoms Activity After 90 Day Decay	11.5-4
11.5-4	Activity Buildup on Ion Exchange Resins Per Core Cycle with 1 Percent Failed Fuel	11.5-7
11.5-5	Activity on Ion Exchanger Resins After Decay	11.5-8
11.5-6	Activity Buildup on Filter Cartridge Per Core Cycle	11.5-9

10/4

LIST OF TABLES (CONT'D)

Table	Title	Page
11.5-7	Activity on Filter Cartridges After Decay	11.5-10
11.5-8	Design Data for Solid Waste System Components	11.5-13
11.5-9	Instrument Application	11.5-16

LIST OF FIGURES

Figure	Title
11.1-1	Escape Rate Coefficients
11.2-1	Waste Management System P&I Diagram
11.2-2	Boric Acid Concentrator Package
11.2-3	Waste Management System P&I Diagram
11.2-4	Waste Management System P&I Diagram
11.2-5	Site Plot Plan
11.3-1	Waste Management System P&I Diagram

CHAPTER 11

RADIOACTIVE WASTE MANAGEMENT SYSTEM

11.1 SOURCE TERMS

11.1.1 FISSION PRODUCTS

The mathematic model used for determining the specific concentration of nuclides in the reactor coolant involves a group of time dependent simultaneous equations. The fuel pellet region and reactor coolant region are analyzed by applying a mass balance of production and removal for each nuclide thereby establishing a set of first order, differential equations. In the fuel pellet region the mass balance includes the fission product production by direct fission yield, by parent fission product decay, and by neutron activation, and removal by decay, by neutron activation, and by escape to the reactor coolant. In the coolant region the analysis includes the fission product production by escape from the fuel through defective fuel rod cladding, by parent decay in the coolant, and by neutron activation of coolant fission products. Removal is by decay, by coolant purification, by feed (makeup) and bleed (letdown to waste management system) for fuel burnup, and by leakage or other feed and bleed due to such operations as cold or hot startups and shutdowns or load follow operations.

The expression derived to determine the fission product inventory in the fuel pellet region is:

$$\frac{dN_{p,i}}{dt} = FY_iP + (f_{i-1}\lambda_{i-1} + \sigma_{i-1}\phi)\ N_{p,i-1} - (\lambda_{i-1} + \nu_i + \sigma_i\phi)\ N_{p,i}$$

and in the reactor coolant region is:

$$\frac{dN_{c,i}}{dt} = D\nu_iN_{p,i} + (f_{i-1}\lambda_{i-1} + \sigma_{i-1}\phi FCS)N_{c,i-1}$$

$$-(\lambda_i + \frac{R}{W}\ \eta_i + \frac{(1-\eta_i)C}{C_o-Ct} + \frac{Q}{W})\ N_{c,i}$$

where:

N = population, atoms

F = average fission rate, fissions/MWt-sec

Y = U235 fission yield of nuclide fraction

P = core power, MWt

λ = decay constant, sec^{-1}

σ = microscopic cross section, cm^2

ϕ = thermal neutron flux, neutrons/cm^2-sec

ν = escape rate coefficient, sec^{-1}

f = branching fraction

t = time, sec

D = defective fuel rod cladding, fraction

FCS = core coolant volume to reactor coolant volume ratio

R = purification flow rate during power cycle, lbs/sec

W = reactor coolant mass during power cycle, lbs

η = resin efficiency of chemical and volume control system ion exchanger for a given nuclide, fraction

C_o = initial boron concentration, ppm

C = boron reduction rate by feed and bleed, ppm/sec compensating for fuel burnup

Q = leakage or other feed and bleed from reactor coolant lbs/sec

Subscripts

p, pellet region

c, core region

i, designates the nuclide parameters (i.e., $_{54}Xe^{133}$)

i-1 designates the parent nuclide parameters (i.e., $_{53}I^{133}$)

The model does not involve the fuel plenum and gap region of the core because the escape rate coefficients represent the overall release from the fuel pellets to the reactor coolant. Plenum and gap region activities are calculated using diffusion theory as described in Section 15.4. The productions terms involving the microscopic cross section are used only to produce Cs^{134} from CS^{133}, the stable end product of the 133 chain. The removal term involving the microscopic cross section is used only with Xe^{135} and only in the pellet region because of insignificant effects on other nuclides and in the coolant region.

The fission product activity concentrations used as basic source terms are given in Table 11.1-1. The data used for these calculations are given in Table 11.1-2. The effects of expected plant operation as a result of start-ups and shutdowns are simulated by using a constant liquid waste rate of 1.2 gpm or 617,000 gallons per year. The tritium concentration is based on no recycle of concentrator distillate.

The primary factor in determining the fission product inventories is the escape rate coefficient. This is an empirical coefficient which was derived from experiments initiated by Bettis and performed in the NRX and MTR reactors.[1] The escape rate coefficients derived from these data are given in Table 11.1-2. The escape rate coefficients were obtained from test rods which were operated at high linear heat rates. The linear heat rates were uniform over the test sections which were 10.25 inches in length. The exact linear heat rates were not precisely known but post-irradiation showed that some tests rods had experienced centerline melting. Later tests were conducted in the NRX reactor to determine the effect of rod length on the release of fission gases and iodines from defective fuel.[2] A by-product of these experiments was the effect of linear heat rate on the escape rate coefficient. The escape rate coefficient for several nuclides as a function of the linear heat rate is shown in Figure 11.1-1. Also shown in Figure 11.1-1 are the escape rate coefficients used for noble gases and halogens. Since the average heat rate for a fuel rod will be well below the crossover points in Figure 11.1-1, which is above 17 kilowatts per foot in each case, the presently used escape rate coefficients are conservative.

TABLE 11.1-1

REACTOR COOLANT SPECIFIC ACTIVITY (μCi/cc), 577F

Nuclide	Half Life	Abnormal Operation (1.0% Failed Fuel)	Normal Operation (0.1% Failed Fuel)	Nuclide	Half Life	Abnormal Operation (1.0% Failed Fuel)	Normal Operation (0.1% Failed Fuel)
H-3	12.3y	0.132	0.107	I-133	21h	5.66	0.566
Br-84	32m	4.66(-2)*	4.66(-3)	Xe-133	5.3d	181.0	18.1
Kr-85m	4.4h	1.49	0.149	Te-134	42m	2.62(-2)	2.62(-3)
Kr-85	10.8y	0.385	3.85(-2)	I-134	52m	0.62	6.2(-2)
Kr-87	76m	0.81	8.1(-2)	Cs-134	2.1y	0.10	1.0(-2)
Kr-88	2.8h	2.6	0.26	I-135	6.7h	2.7	0.27
Rb-88	18m	2.55	0.255	Xe-135	9.2h	7.53	0.753
Rb-89	15m	6.4(-2)	6.4(-3)	Cs-136	13d	2.55(-2)	2.55(-3)
Sr-89	51d	5.07(-3)	5.07(-4)	Cs-137	30y	0.32	3.2(-2)
Sr-90	28.8y	2.61(-4)	2.61(-5)	Xe-138	17m	0.36	2.6(-2)
Y-90	64h	1.02(-3)	1.02(-4)	Cs-138	32m	0.69	6.9(-2)
Sr-91	9.7h	3.56(-3)	3.56(-4)	Ba-140	12.8d	6.11(-3)	6.11(-4)
Y-91	59d	0.111	1.11(-2)	La-140	40.2h	5.85(-3)	5.85(-4)
Mo-99	67h	2.03	0.203	Pr-143	13.6d	5.84(-3)	5.84(-4)
Ru-103	39.6d	4.13(-3)	4.13(-4)	Ce-144	285d	4.13(-3)	4.13(-4)
Ru-106	367d	2.48(-4)	2.48(-5)	(Corrosion Products)			
Te-129	67m	2.51(-2)	2.51(-3)	Co-60	5.2y	5.19(-4)	5.19(-4)
I-129	1.7(7)y	7.21(-8)	7.21(-9)	Fe-59	45d	2.13(-5)	2.13(-5)
I-131	8.0d	3.97	0.397	Co-58	71d	4.66(-3)	4.66(-3)
Xe-131m	12d	1.48	0.148	Mn-54	312d	2.75(-5)	2.75(-5)
Te-132	78h	0.33	3.3(-2)	Cr-51	27d	3.8(-3)	3.8(-3)
I-132	2.3h	1.02	0.102	Zr-95	65d	9.35(-7)	9.35(-7)

*( )Denotes Power of ten

TABLE 11.1-2

BASES FOR REACTOR COOLANT RADIOACTIVITY

Core power level, Mwt	2700
Fuel cycle full power days	357
Percent failed fuel	1.0
CVCS purification ion exchanger decontamination factor	10
Purification flow rate (CVCS purification ion exchanger), gpm	40
Effective purification flow rate for lithium and cesium remova, gpm	8
Fission product escape rate coefficeints, sec^{-1} (Based on centerline melting of fuel)	
Noble gases	6.5×10^{-8}
Halogens, Cs	2.3×10^{-8}
Te, Mo.	1.4×10^{-9}
All others	1.4×10^{-11}
Feed and bleed liquid waste for fuel burnup, gal/yr	216,000
Other feed and bleed liquid waste, gal/yr	617,000
Thermal neutron flux, $n/cm^3 sec$	$\simeq 4.3 \times 10^{13}$
Reactor coolant volume, ft^3	9662.

11.1.2 CORROSION PRODUCTS

The activity of radioactive crud and its thickness on primary system surfaces has been evaluated using measured data from six pressurized water reactors. These reactors are Connecticut-Yankee, Indian Point 1, Yankee Rowe, Saxton, Shippingport, and SM-1.

Even though these reactors have different water chemistries and different materials in contact with the reactor coolant, their crud activity (dpm/mg-crud) is markedly similar. The average and maximum activities of the long-lived isotopes and the corrosion product concentration (crud) for each of the reactors are shown in Table 11.1-3. The average activity for the six reactors and the maximum activity occurring in any one reactor are given at the bottom of the table. As can be seen from these numbers, there is less than a factor of ten difference between the average and maximum values.

The half-lives, reactions and gamma decay energies for each of the long-lived isotopes (significant isotopes remaining after 48 hours decay) are as shown in Table 11.1-4.

The radioactive crud originates on in-core and out-of-core surfaces. The crud plates out on the core surfaces and re-erodes after a short irradiation period. The irradiation period or core residence time in seconds is determined by the following expressions.

$$t_{res} = \frac{1}{\lambda} \left[1 - \frac{AS\ (16.67)}{\Sigma_c\ \phi C}\right]$$

$$\Sigma_c \phi = \frac{f\ N}{AMU} \left(\sigma_{th}\ \phi_{th} + \sigma_f\ \phi_f\right)$$

where: λ = is the decay constant, sec^{-1}

A = is the circulating crud activity, dpm/mg

S = is the total reactor coolant system surface area, cm^2

C = is the core surface area, cm^2

AMU = is atomic mass unit of nuclide, gm/mole

N = is Avogadro's Number, atoms/mole

f = fractional abundance of the parent nuclide in the in-core film

σ_{th} = thermal neutron cross section, cm^2

σ_f = fast neutron cross section, cm^2

ϕ_{th} = thermal neutron flux, neutrons/cm^2-sec

ϕ_f = fast neutron flux, neutrons/cm^2-sec

Residence time for each nuclide for each of the six reactors was evaluated using core and system parameters for each plant along with the maximum activity given in Table 11.1-3 for any plant. The longest resulting residence time for each nuclide is listed in Table 11.1-5. The maximum crud specific activities (dpm/mg) for the various long-lived isotopes in the crud are as shown in Table 11.1-5 based on maximum core residence times and parameters for the nuclear steam supply system (NSSS).

The activity concentrations of the crud in the reactor coolant are determined by utilizing the maximum specific activities given in Table 11.1-5 and the average crud concentration (56 ppb) identified in Table 11.1-3. The crud activity is shown in Table 11.1-1 along with the fission production activity concentrations.

11.1.3 TRITIUM PRODUCTION

Tritium is produced in the coolant or enters the coolant from a number of sources. One source is the fissioning of uranium within the fuel which yields tritium as a ternary fission product. Since Zircaloy fuel cladding reacts with tritium to form zirconium hydride, no tritium diffuses through the cladding (3) (4). Therefore, the tritium released to the coolant from the fuel is only from defective fuel.

Tritium is also produced by the reaction of neutrons with boron in the control element assemblies (CEA's). Data from operating plants using B_4C control rods indicates that no tritium is released from the control rods. The tritium may combine with carbon to form hydrocarbons and/or with lithium to form lithium hydride thereby preventing diffusion through the NiCiFe cladding. The low internal temperature of the B_4C control rods may also prohibit tritium diffusion. To account for possible control rod cladding defects, it is assumed that one percent of the tritium produced in the CEA's is released to the coolant.

Major sources of tritium are the activation of boron, lithium, deuterium, and nitrogen within the reactor coolant. Boron in the form of boric acid is used in the coolant for reactivity control and is the major source of tritium. Lithium is produced in the coolant as a result of neutron-boron reaction. Lithium may be added as a pH control agent. The deuterium is a natural constituent of water. Nitrogen may be present due to aeration of the coolant during shutdown and due to aerated makeup water.

Table 11.1-6 identifies the contribution of each source to the total tritium concentration.

11.1.4 NITROGEN-16 PRODUCTION

Nitrogen-16 is produced from the reaction of fast neutrons with oxygen forming Oxygen-17 which is very unstable and decays by emitting a proton thus forming Nitrogen-16. Nitrogen-16 has a half life of 7.35 seconds and emits a gamma ray of high energy 82 percent of the time. The gamma energies are 6.13 and 7.10 Mev in a ratio of 12.5 to 1. The nitrogen activity at the reactor vessel coolant outlet nozzles is 3.02×10^8 dpm/cm^3. The basis for this result is tabulated in Table 11.1-7.

TABLE 11.1-3

CRUD SPECIFIC ACTIVITY (dpm/mg) - OPERATING REACTORS

Reactor		Nuclide							
		^{60}Co	^{58}Co	^{54}Mn	^{51}Cr	^{59}Fe	^{95}Zr	ppb	Reference
Conn. Yankee	avg.	3.64+4[a]	3.00+7	4.63+5	3.32+6	3.99+5	-	114[b]	1
	max.	1.80+5	1.03+8	1.13+6	5.30+6	1.20+6		344	
Indian Point 1	avg.	1.46+6	5.14+6	6.61+5	1.34+6	3.04+5	3.04+5	14	2
	max.	2.00+6	9.10+6	2.00+6	8.21+6	3.33+6	4.22+5	30	
Yankee Rowe	avg.	9.10+6	1.02+7	8.56+5	3.19+6	9.73+5	-	179	3, 4, 5
	max.	3.23+7	2.15+7	3.63+6	7.68+6	1.44+6	-	902	
Saxton	avg.	5.35+6	4.56+7	7.71+6	9.00+7	2.70+6	-	27	3,6,7,8
	max.	2.20+7	1.51+8	1.37+7	1.10+8	6.00+6	-	250[c]	
Shipping-port	avg.	2.26+7	2.83+6	1.26+6	2.19+6	1.80+6	7.00+5	4	9,10,11
	max.	4.75+7	3.16+6	1.72+6	2.20+6	1.80+6	9.70+5	65	
SM-1	avg.	3.90+5	2.10+6	1.40+5	4.90+5	5.00+5	-	-	10
	max.	-	-	-	-	-	-	-	
	avg.	6.54+6	1.59+7	1.77+6	1.77+7	1.17+6	4.86+5	56	
	max.	4.75+7	1.51+8	1.37+7	1.10+8	6.00+6	9.70+5	902	

a) Denotes power of ten.

b) Amount of circulating crud in primary coolant ppb.

c) Shutdown value.

LIST OF REFERENCES TO TABLE 11.1-3

1) Connecticut Yankee Monthly Operation Reports No. 68-2 through No. 69-1.

2) Indian Point 1 Semi-Annual Operation Reports, 9/66 through 3/68, Docket No. 50-3.

3) Corrosion Product Behavior in Stainless-Steel-Clad Water Reactor Systems, Nuclear Applications, Vol. 1, October 1965.

4) Chemical Evaluation of Yankee Core I Fuel Cladding Corrosion Products, Picone L. F., Taylor G. R., August 1966, WCAP-6072.

5) Large Closed-Cycle Water Reactor Research and Development Program, Progress Report 4/1/63 - 6/30/63, WCAP-3739.

6) Large Closed-Cycle Water Reactor Research and Development Program, Progress Report 1/1/65 - 3/31/65, WCAP-3269-12.

7) The Saxton Chemical Shim Experiment, Weisman J., Bartnoff S., July 1965, WCAP-3269-24.

8) Large Closed-Cycle Water Reactor Research and Development Program, Progress Report 4/1/65 - 6/30/65, WCAP-3269-13.

9) Decontamination of the Shippingport Atomic Power Station, Abrams C.S., Salterelli E. A., January 1966, WAPD-299.

10) Boiling Water Reactor Technology, Status of the Art Report Volume II, Water Chemistry and Corrosion, C. R. Breden, February 1963, ANL-6562.

11) Radiation Buildup on Mechanisms and Thermal Barriers, E. Weingart, June 1963, WAPD-PWR-TE-145.

TABLE 11.1-4

LONG-LIVED ISOTOPES IN CRUD

Nuclide	$T_{1/2}$	Parent	Reaction	α/dis	E(Mev)
^{60}Co	5.28y	^{59}Co	N, γ	2.00	1.25
^{58}Co	71d	^{58}Ni	N, P	1.00	0.81
^{54}Mn	314d	^{54}Fe	N, P	1.00	0.84
^{51}Cr	27.8d	^{50}Cr	N, γ	0.10	0.32
^{59}Fe	45d	^{58}Fe	N, γ	1.00	1.18
^{95}Zr	65d	^{94}Zr	N, γ	2.00	0.75

TABLE 11.1-5

CRUD CORE RESIDENCE TIME AND SPECIFIC ACTIVITY

Isotope	t_{res} (b)	Activity (c) (dpm/mg)
^{60}Co	300d	2.85 (+7) (a)
^{58}Co	62d	2.56 (+8)
^{54}Mn	74d	1.52 (+6)
^{59}Fe	160d	1.18 (+6)
^{51}Cr	44d	2.09 (+8)
^{95}Zr	0.5d	5.11 (+4)

(a) Denotes power of ten.

(b) Based on maximum crud activities from operating plants.

(c) Based on maximum crud activity levels and parameters for for this plant.

TABLE 11.1-6

SOURCES OF TRITIUM PRODUCTION

Source	Basis	Concentration* (μCi/cc at 577F)
Fission	1.0% failed fuel	0.028
CEA	1.0% cladding defects	0.0021
Boric Acid	Base loaded boron concentration	0.1008
Deuterium	150 ppb	0.0009
Lithium	0.5 ppm	0.00036
Nitrogen	10 cc(N_2)/kg(H_2O)	0.0000112
Total		0.132

* Calculated on same basis as Table 11.1-1.

TABLE 11.1-7

NITROGEN-16 PRODUCTION PARAMETERS

Parameter	Value
Fast Neutron Flux, n/cm^2-sec	6.61×10^{13}
Coolant Density, g/cm^3	0.73
Core Transit Time, sec	0.853
Coolant Recirculation Time, sec	11.29
Reaction Cross Section, barns	2×10^{-5}
Reactor Power, Mwt	2700

11.1.5 FUEL EXPERIENCE

Current operation of stainless steel clad fuel rods in the Connecticut Yankee reactor shows fuel failure rates on the order of 0.01 percent. Zircaloy-clad UO_2 fuel in the Obrigheim reactor in Germany sustained a fuel failure rate just over 0.1 percent in its first cycle; this has fallen in the second cycle to essentially zero (<0.001 percent). The fuel failure rate in the Dresden 1 reactor over a nine-year period has averaged <0.1 percent with the rate more recently being even lower. Fuel in the Mihama reactor in Japan and the Point Beach reactor has exceeded the burnup at which failures in fuel of similar design were observed in Ginna, without exhibiting increases in coolant activity (indicative of fuel defects).

The fact that widespread defects in some reactors, associated with fuel clad contamination have now been recognized and corrective measures taken, provides further assurance that failures at this frequency from this cause will not occur in the future. Existing licensing regulations limit coolant activity to that associated with 1 percent failed fuel. Over the lifetime of an operating reactor, it is expected that coolant activity levels corresponding to 0.1 percent failed fuel will predominate.

11.1.6 LEAKAGE SOURCES

There are several potential sources of leakage from the plant systems that can contribute to the total release to the environs. If leakage occurs from systems containing reactor coolant, gaseous radioactivity could be released via several pathways.

Normal leakage from the reactor coolant system exposed to the containment atmosphere is expected to be 40 gallons per day or less. Nuclide activities given in Table 11.1-1 are used as source terms. Under equilibrium conditions, 10 percent or less of the iodine and particulates leaking into the containment remains in the atmosphere and is available for release. The other 90 percent of the iodine and particulate is either plated out in the containment or remains in the liquids and is collected in the containment sump. The annual average exposed leakage into the reactor auxiliary building is expected to be 10 gallons per day or less. The specific activity of the leakage is indicated in Table 11.1-1 with 1 percent failed fuel. The reactor auxiliary building, turbine area and condenser air ejector equipment airborne and particulate concentrations are evaluated on the basis of 20 gallons per day primary-to-secondary estimated normal leakage and 120 gpd anticipated operational occurrence leakage with nuclide specific activities as identified in Table 11.1-1.

Means of detecting leakages are discussed in Section 5.2.4.

Estimated liquid and gaseous released due to leakage from various systems containing radioactivity are discussed in Sections 11.2.5 and 11.3.5, respectively.

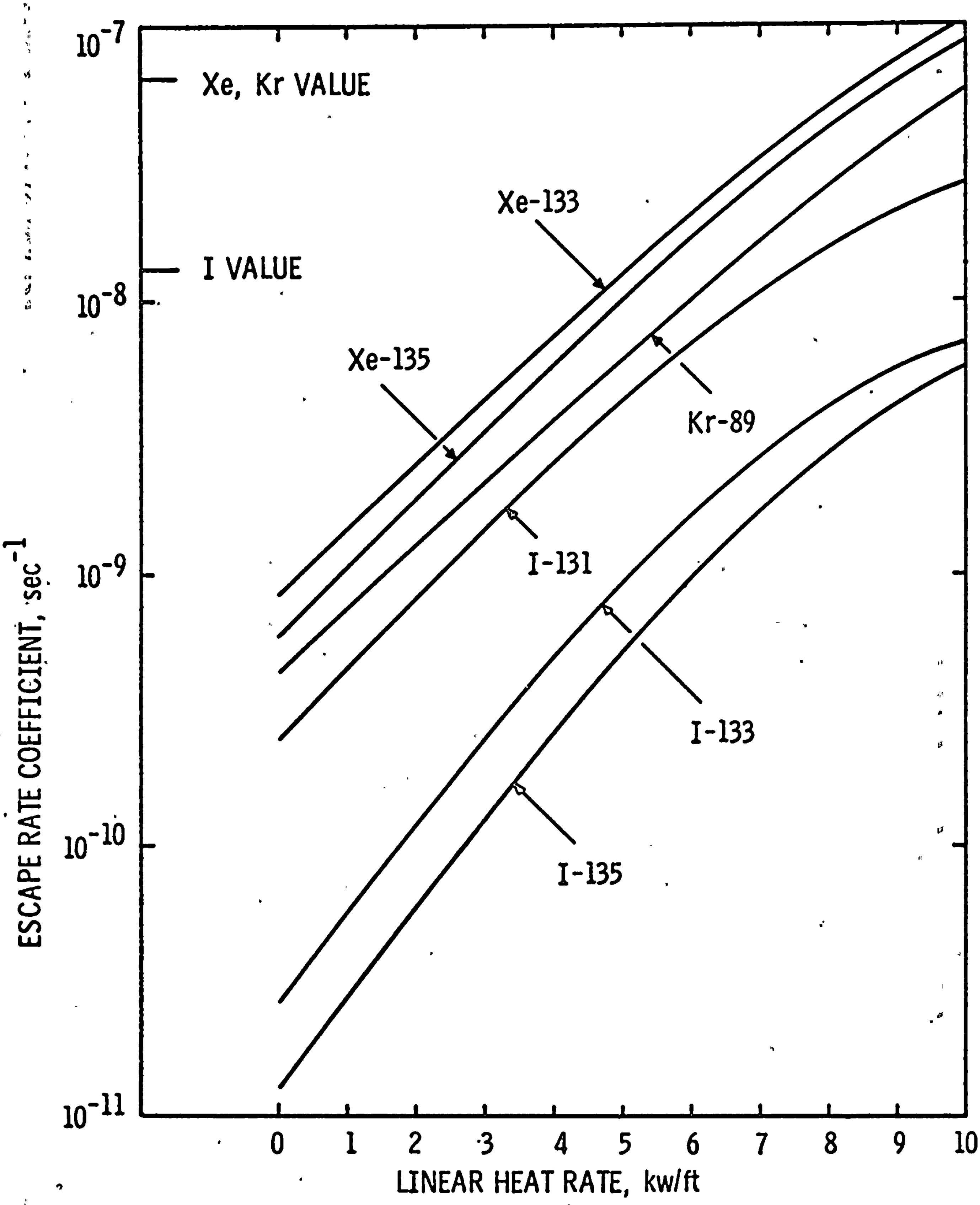
REFERENCES FOR SECTION 11.1

1. J. D. Eichenberg, "Effects of Irradiation on Bulk UO_2", WAPD-183, October 1957.

2. G. M. Allison and H. K. Rae, "The Release of Fission Gases and Iodines from Defective UO_2 Fuel Elements of Different Lengths", AECL-2206, June 1965.

3. James M. Smith, Jr. "The Significance of Tritium in Water Reactors", GE, APED, 9/19/67.

4. Joseph W. Ray et. al., "Investigation of Tritium Generation and Release in PM Nuclear Power Plants", BMI-1787, 10/31/66.



13-160

FLORIDA POWER & LIGHT CO. Hutchinson Island Plant	Escape Rate Coefficients	Figure 11.1-1

11.2 LIQUID WASTE SYSTEMS

11.2.1 DESIGN BASES

The boron recovery system and the liquid waste system (integral parts of the waste management system) are designed to:

a) process the various potentially radioactive liquid wastes such that the radioactivity release to the environs during normal operation will be as low as practicable. The numerical design objectives for releases during normal operation are to limit average annual liquid activity release quantity to 5 Ci and average annual activity release concentration to 2×10^{-8} μCi/cc excluding tritium and dissolved fission product gases.

b) Limit the annual average tritium discharge concentration to 5×10^{-6} μCi/cc in accordance with the proposed Appendix I to 10 CFR 50.

c) Limit releases due to anticipated operational occurrences within 10 CFR 20 limits.

11.2.2 SYSTEM DESCRIPTION

Liquid waste influent to the waste management system, shown in Table 11.2-1 is segregated by chemistry and/or probable source activity for more efficient processing. Tritiated, hydrogenated, borated reactor coolant quality wastes of potentially high activity are mainly processed in the boron recovery system. Aerated, chemically contaminated, and low activity liquid wastes are received and processed separately in the liquid waste system. Table 9.3-6 lists the drains routed to the drain tanks.

11.2.2.1 Boron Recovery System

The major influent to the boron recovery system is reactor coolant from the chemical and volume control system letdown due to feed and bleed operations for shutdown, startups, and boron dilution over core life. Reactor coolant quality water from valve and equipment leakoffs, drains and relief valve within the containment are collected in the reactor drain tank and subsequently processed by this system. Reactor coolant from leakoffs and drains in the reactor auxiliary building are collected in the equipment drain tank of the liquid waste system.

The boron recovery P&I diagrams are shown on Figures 11.2-1, 11.2-2, and 11.2-3. The borated and hydrogenated water discharged by the reactor drain pumps or diverted by the chemical and volume control system volume control tank diversion valve (V-2500) is sent to the flash tank where dissolved hydrogen and fission gases are stripped by a counter current flow of nitrogen gas from the liquid and discharged to the gas decay tanks. Hydrogen is stripped from the water so that an explosive gas mixture does not occur in subsequent process equipment. Use of nitrogen cover gas in the holdup tanks provides additional protection. The nitrogen stripping medium maintains a slight overpressure in the flash tank to prevent air in-leakage, thus precluding potential formation of an

TABLE 11.2-1

SOURCES AND VOLUMES OF LIQUID WASTE

1. Boron Recovery System

Source	Waste Generating Operation	Volume (gallons/year)
Chemical and volume control system	Boron reduction for fuel burnup	216,000
	Cold shutdown and startups	332,000
	Hot shutdowns and startups	161,000
	Refueling shutdown and startup	68,000
Resin dewatering	Sluice and dewater 256 ft^3 resin per year at 2 ft^3 water/ft^3 resin	3,800
Reactor coolant leakage	200 gpd for Four Reactor Coolant Pumps	62,600
		843,400

TABLE 11.2-1

2. Liquid Waste System

Source	Waste Generating Operation	Volume (gallons/year)
Equipment drains and leakage	75 gal/day	28,000
Sample and laboratory sink drains	20 gal/day	7,000
Equipment decontamination	10 gpm for 20 minutes per day	73,000
Floor drains	5 gpm for 10 minutes per day	18,000
Fuel cask washdown	400 gal/cask per refueling	30,000
Sub-total		156,000
Laundry	200 gal/day	73,000
Showers	4 showers per day at 30 gal per shower	44,000
Steam Generator Blowdown	50 gph blowdown for 1 core cycle (anticipated operational occurance)	435,000
Steam Generator Blowdown	200 gpd for 1 core cycle (estimated normal operation)	72,500
Total	Anticipated Operational Occurrence	708,000
Total	Estimated Normal Operation	345,500

explosive gas mixture in the flash tank. In the event of a high liquid level in the flash tank, the influent is automatically diverted to the holdup tanks. A low level in the flash tank stops the flash tank pumps automatically. The flash tank high and low pressure and level alarms are annunciated in the control room. A flow switch in the inlet line to the flash tank automatically opens the nitrogen supply valve and starts the flash tank pumps when water enters the tank. The degassed liquid is automatically pumped from the flash tank to the holdup tanks. The holdup tanks provide sufficient storage capacity to accumulate discharges until a sufficient volume is available for further processing on a batch basis. The radioactivity of the liquid is significantly reduced during storage by natural decay of the short half-lived radionuclides. During this period any degassification and radioactive decay can be monitored by liquid sample analysis or periodic sampling of the tank gas space with the gas analyzer. Air in-leakage to the holdup tanks is precluded by a nitrogen overpressure maintained in the tanks. As the holdup tanks fill, nitrogen is displaced to other interconnected holdup tanks or to the gas collection header. The holdup tanks have high and low level and pressure alarms which are annunciated in the control room. In the event process fluid has to bypass the flash tank due to malfunction of the flash tank controls or pumps, the stored liquid can be recirculated by the holdup drain or recirculation pumps to the flash tank to remove any hydrogen once the flash tank returns to service. The holdup recirculation pump supplies flushing water to the preconcentrator ion exchangers and spent resin tank during resin sluice operations.

Normally, the contents of the holdup tanks are transferred to the boric acid concentrator through the preconcentrator filter and preconcentrator ion exchanger. If necessary, the contents of one holdup tank may be recirculated through a preconcentrator filter and ion exchanger while a second holdup tank contents are processed through the other preconcentrator filter and ion exchanger prior to discharge to the boric acid concentrators. The holdup pumps and holdup recirculation pump are stopped on low holdup tank level. The boric acid concentrators have both automatic sampling and local grab sample provisions to ensure control of the effluent chemistry. The boric acid concentrators provide a level signal that, after the initial manual pump start, automatically starts and stops the holdup or holdup recirculation pump aligned to either boric acid concentrator. The two boric acid concentrators have a very low usage factor (approximately 4 percent) and thus provide a high system reliability and availability.

The bottoms from each boric acid concentrator are pumped via the boric acid discharge strainer to the boric acid holding tank for temporary storage and sampling. The recovered boric acid may then be returned to the makeup tanks for recycle or discharged to the drumming station for ultimate offsite disposal, if recycle is not desired.

The concentrator distillate passes through one of the two boric acid condensate ion exchangers to remove boron carryover and into one of the two boric acid condensate tanks. While one boric acid condensate tank is filling, the other is sampled and discharged. In the unlikely event that the contents of the tank do not meet the chemical or radioactivity limitations, the contents can be recycled to the holdup tanks for further processing or recycled through the boric acid condensate ion exchangers.

A local high and low level alarm is provided on the boric acid condensate tanks. The boric acid condensate pumps automatically stop on low water level in the tanks.

Prior to controlled discharge of the treated liquid waste, the fluid must be analyzed and its activity verified as acceptably low. Discharge is accomplished through an effluent radiation monitor which records the release activity level and automatically terminates discharge on high radiation. If reuse in the plant is desired, the fluid in analyzed for acceptability of both chemistry and activity.

Design data for the major components are listed in Table 11.2-2. Flow, temperature, and pressure data are given in Table 11.2-3 with the locations corresponding to process points on Figures 11.2-1, 11.2-2, 11.2-3.

The nuclide concentrations for normal operation and for anticipated operational occurrences adjusted to 70F are indicated for selected locations in the chemical and volume control system in Tables 11.2-6 and 11.2-7, respectively. The selected locations are indicated by the process points on Figures 11.2-1, 11.2-2, and 11.2-3. Normal operation is defined as operating with 0.1 percent failed fuel, and anticipated operation all occurrences is defined as operation with 1 percent failed fuel.

Analysis is made assuming that the activities in Table 11.1-1 exist in the coolant upstream of the purification equipment in conjunction with the expected equipment performance given in Table 11.2-4 and in Table 11.2-5.

Systems similar in function and design to the boron recovery system described herein have been used successfully at plants such as Connecticut Yankee and Ginna. Even with significant coolant radioactivity at Ginna, releases have been controlled well within the 10 CFR 20 limits.

All process components have been used extensively in the nuclear industry to remove radioactive contaminants from liquids. The performance of process units used in the analysis is in agreement with general industry experience.

11.2.2.2 Liquid Waste System

The liquid waste system is shown on Figure 11.2-4. Liquid waste include those from the laboratory sink drains, decontamination area drains, floor drains, building sumps, laundry effluent, and contaminated showers. In the event of a primary-to-secondary leak, the steam generator blowdown is processed by this system. The wastes are segregated for batch processing by collection in the equipment drain tank, chemical drain tank, and laundry drain tanks.

Low activity, aerated, and potentially dirty liquid drains and building sumps discharge to the equipment drain tank. Low activity chemical drains from the sampling system, decontamination drains, and chemical laboratory drains

TABLE 11.2-2

DESIGN DATA FOR BORON RECOVERY SYSTEM COMPONENTS

1. Ion Exchangers	Preconcentrator	Boric Acid Condensate
Quantity	2	2
Type	Deep Resin Bed	Deep Resin Bed
Design Pressure, psig	150	150
Design Temperature, F	200	200
Normal Operating Pressure, psig	40	25
Normal Operating Temperature, F	100	120
Resin Volume, ft^3	32	32
Materials	SS	SS
ASME Code, Section	VIII	VIII

2. Tanks	Reactor Drain	Flash	Holdup	Boric Acid Condensate	Boric Acid Holding
Quantity	1	1	4	2	1
Internal Volume, gal	1600	400	40,000	7,300	2,400
Design Pressure, psig	25	15	10	Atmos.	Atmos.
Design Temperature, F	250	250	240	250	200
Normal Operating Pressure, psig	0.5	0.5	0.5(psia)	Atmos.	Atmos.
Normal Operating Temperature, F	120	120	120	120	150
Blanket Gas	Nitrogen	Nitrogen	Nitrogen		
Material	SS	SS	SS	SS	SS
ASME Code, Class or Division	III,C	III,C	VIII,1	None	None

3. Pumps	Flash Tank	Reactor Drain	Holdup Drain, Holdup Recir. Boric Acid Cond.	Boric Acid Holding
Quantity-Full Capacity	2	2	5	2
Type	Centrifugal	Centrifugal	Centrifugal	Centrifugal
Design Pressure, psig	150	150	150	150
Design Temperature, F	200	200	200	200

TABLE 11.2-2 (Cont.)

3. Pumps	Flash Tank	Reactor Drain	Holdup Drain, Holdup Recir. Boric Acid Cond.	Boric Acid Holding
Design Conditions				
Flow, gpm	150	50	50	50
Head, ft	51	140	140	96
Wetted Materials	SS	SS	SS	SS
Seal Type	Mechanical	Mechanical	Mechanical	Mechanical
Motor Horsepower	7.5	5	5	5
Motor Voltage, volt	460	460	460	460
ASME Pump and Valve Code, Section	III,C	III,C	None	None

4. Filters	Preconcentrator
Quantity	2
Type of Elements	Replaceable Cartridge
Retention of 2 micron particles, %	98
Design Pressure, psig	150
Design Temperature, F	200
Design Flow, gpm	100
Material	Stainless Steel
ASME Code, Section	VIII

5. Boric Acid Concentrator

Quantity	2
Design Pressure, psig	80
Design Temperature, F	250
Design Flow, gpm	20
Cooling Water Flow Rate, gpm.	650
Steam Required at 15 psig, lb/hr	13,000
ASME Code, Section	VIII

TABLE 11.2-3

BOROEN RECOVERY SYSTEM PROCESS FLOW DATA

Mode #1(1)

Location:	1	2	3	4	5	6	7	8
Flow, gpm	200 gpd	50	50	50	50	60	50	50
Pressure, psig	1	4	12	7.5	6.5	4	45	2
Temperature, F	120	120	120	120	120	120	120	120

Mode #2(1)

Location:	1	2	3	4	5	6	7	8
Flow, gpm	200 gpd	-	-	40	40	50	40	40
Pressure, psig	1	-	-	5	4.5	4	44	2
Temperature, F	120	-	-	120	120	120	120	120

Mode #3(1)

Location:	1	2	3	4	5	6	7	8
Flow, gpm	200 gpd	-	-	84	84	94	84	84
Pressure, psig	1	-	-	21	18.5	4	34	2
Temperature, F	120	-	-	120	120	120	120	120

Mode #4(1)

Location:	1	2	3	4	5	6	7	8
Flow, gpm	200 gpd	-	-	128	128	138	128	128
Pressure, psig	1	-	-	51	45.5	4	18	2
Temperature, F	120	-	-	120	120	120	120	120

(Continued)

TABLE 11.2-3 (Cont.)

Mode #5[(1)]

Location:	12	13	14	15	16	17	18	19	20	21
Flow, gpm	20	20	20	20	0-10	16-19	16-19	50	50	50
Pressure, psig	3	65	62	60	5	5	2	4	60	10
Temperature, F	120	120	120	120	120	120	120	120	120	120

Mode #6[(1)]

Location:	9	10	11	15a	16a
Flow, gpm	50	50	50	50	50
Pressure, psig	4	32	27	24	20
Temperature, F	120	120	120	120	120

Mode #7

Location:	22	23
Flow, gpm	50	50
Pressure, psig	4	40
Temperature, F	150	150

NOTES:

(1) The modes of operation are defined as follows:

Mode No.	Description
1	Processing RDT Contents
2	CVCS Normal purification VCT Diversion Processing
3	CVCS Intermediate Purification VCT Diversion Processing
4	CVCS Maximum Purification VCT Diversion Processing
5	Processing holdup tank contents via the boric acid Concentrator
6	Holdup Tank Contents Recirculation
7	Pumping BAHT Contents to the BAMT in the CVCS.

(2) The pressure drop across the filters and ion exchangers will vary with loading. The pressure drops are typical.

TABLE 11.2-4

EXPECTED FILTER AND ION EXCHANGER PERFORMANCE

Nuclide	Chemical and Volume Control System		Boron Recovery Waste Processing System	
	Filter DF	Ion Exchanger DF	Filter DF	Ion Exchanger DF
Br-84	1	10^3	1	10^2
Rb-88	1	1	1	10^2
Rb-89	1	1	1	10^2
Sr-89	1	10^2	1	10
Sr-90	1	10^2	1	10
Y-90	1	1	1	10
Sr-91	1	10^2	1	10
Y-91	1	1	1	10
Mo-99	1	1	1	10
Ru-103	1	10	1	10
Ru-106	1	10	1	10
Te-129	1	10	1	10
I-129	1	10^3	1	10^3
I-131	1	10^3	1	10^3
Te-132	1	10	1	10
I-132	1	10^3	1	10^3
I-133	1	10^3	1	10^3
Te-134	1	10	1	10
I-134	1	10^3	1	10^3
Cs-134	1	1	1	10^2
I-135	1	10^3	1	10^3
Cs-136	1	1	1	10^2
Cs-137	1	1	1	10^2
Cs-138	1	1	1	10^2
Ba-140	1	10^2	1	10
La-140	1	10	1	10
Pr-140	1	10	1	10
Ce-144	1	10	1	10
Co-60	10	1	1	1
Fe-59	10	1	1	1
Co-58	10	1	1	1
Mn-54	10	1	1	1
Cr-51	10	1	1	1
Zr-95	10	1	1	1

TABLE 11.2-5

BORON RECOVERY SYSTEM PERFORMANCE DATA

Flash Tank DF for Fission Gases	2
Boric Acid Concentrator	
DF for Liquid (Influent to Distillate)	200
DF for Fission Gases	5
Holdup Tank Delay Factor, days	9
Annual System Condensate Discharged to CWD[(1)]	
Volume, gal.	843,000
Activity, Curies H-3	339
Dissolved Fission Product Gases	24,500
All Others	0.89

Note: (1) The volume and activity values obtained are based on 843,000 gals of waste to BMS with 1% failed fuel.

TABLE 11.2-6

BORON RECOVERY SYSTEM (BRS) MAXIMUM NUCLIDE CONCENTRATIONS (70F) DURING NORMAL OPERATIONS (μCi/cc)

Nuclide	CVCS 6 BRS 1	CVCS 7	CVCS 10 BRS 2,3,4,5	BRS 6,7,8	BRS 12,13,14	BRS 15	BRS 16	BRS 18,19,20,21
H-3	1.48(-1)	1.48(-1)	1.48(-1)	1.48(-1)	1.48(-1)	1.48(-1)	1.48(-1)	1.48(-1)
Br-84	6.43(-3)	6.43(-3)	6.43(-6)	6.43(-6)	0	0	0	0
Kr-85m	2.05(-1)	2.05(-1)	2.05(-1)	1.03(-1)	0	0	0	0
Kr-85	1.22(-1)	1.22(-1)	1.22(-1)	6.10(-2)	1.08(-2)	1.08(-2)	0	5.40(-3)
Kr-87	1.12(-1)	1.12(-1)	1.12(-1)	5.60(-2)	0	0	0	0
Kr-88	3.59(-1)	3.59(-1)	3.59(-1)	1.80(-1)	0	0	0	0
Rb-88	3.52(-1)	3.52(-1)	3.52(-1)	3.52(-1)	0	0	0	0
Rb-89	8.85(-3)	8.85(-1)	8.85(-1)	8.85(-3)	0	0	0	0
Sr-89	7.00(-4)	7.00(-4)	7.00(-6)	7.00(-6)	6.00(-6)	6.00(-7)	3.08(-5)	3.08(-9)
SR-90	3.60(-2)	3.60(-2)	3.60(-4)	3.60(-4)	3.60(-4)	3.60(-5)	1.80(-3)	1.80(-7)
Y-90	1.41(-4)	1.41(-4)	1.41(-4)	1.41(-4)	1.36(-5)	1.36(-6)	6.94(-5)	6.94(-9)
Sr-91	4.91(-4)	4.91(-4)	4.91(-6)	4.91(-6)	9.78(-13)	9.78(-13)	4.88(-11)	4.88(-15)
Y-91	1.53(-2)	1.53(-2)	1.53(-2)	1.53(-2)	1.38(-2)	1.38(-3)	6.92(-2)	6.92(-6)
Mo-99	2.81(-1)	2.81(-1)	2.81(-1)	2.81(-1)	2.90(-2)	2.90(-3)	1.45(-1)	1.45(-5)
Ru-103	5.71(-4)	5.71(-4)	5.71(-5)	5.71(-5)	4.88(-5)	4.88(-6)	2.44(-4)	2.44(-8)
Ru-106	3.42(-5)	3.42(-5)	3.42(-6)	3.42(-6)	3.36(-6)	3.36(-6)	1.68(-5)	1.68(-9)
Te-129	3.46(-3)	3.46(-3)	3.46(-3)	3.46(-4)	0	0	0	0
I-129	9.95(-9)	9.95(-9)	9.95(-12)	9.95(-12)	9.95(-11)	9.95(-15)	4.98(-13)	4.98(-17)
I-131	5.48(-1)	5.48(-1)	5.48(-1)	5.48(-4)	2.53(-4)	2.53(-7)	1.26(-5)	1.26(-9)
Xe-131m	2.05(-1)	2.05(-1)	2.05(-1)	1.03(-1)	5.02(-2)	5.02(-2)	0	1.01(-2)
Te-132	4.55(-2)	4.55(-2)	4.55(-4)	4.55(-4)	6.68(-5)	6.68(-6)	3.34(-4)	3.34(-8)
I-132	1.41(-1)	1.41(-1)	1.41(-1)	1.41(-4)	0	0	0	0

*Numbers in (   ) are powers of ten.

TABLE 11.2-6

Nuclide	CVCS 6 BRS 1	CVCS 7	CVCS 10 BRS 2,3,4,5	BRS 6,7,8	BRS 12,13,14	BRS 15	BRS 16	BRS 18,19,20,21
I-133	7.83(-1)	7.83(-1)	7.83(-4)	7.83(-4)	6.29(-7)	6.29(-10)	3.14(-8)	3.14(-12)
Xr-133	2.50(+1)	2.50(+1)	2.50(+1)	1.25(+1)	3.16(0)	3.16(0)	0	6.37(-
Te-134	3.35(-3)	3.35(-3)	3.35(-4)	3.35(-4)	0	0	0	0
I-134	8.55(-2)	8.55(-2)	8.55(-5)	8.55(-5)	0	0	0	0
Cs-34	3.38(-2)	3.38(-2)	3.38(-2)	3.38(-2)	3.38(-2)	3.38(-4)	1.69(-2)	1.69(-6)
I-135	3.73(-1)	3.73(-1)	3.73(-4)	3.73(-4)	0	0	0	0
Xe-135	1.04(0)	1.04(0)	1.04(0)	5.20(-1)	0	0	0	0
Cs-136	4.56(-5)	4.56(-3)	4.56(-3)	4.56(-3)	2.84(-3)	2.84(-3)	1.42(-3)	1.42(-7)
Cs-137	1.38(-1)	1.38(-1)	1.38(-1)	1.38(-1)	1.38(-1)	1.38(-3)	6.93(-2)	6.93(-6)
Xe-138	4.98(-2)	4.98(-2)	4.98(-2)	2.48(-2)	0	0	0	0
Cs-138	9.53(-2)	9.53(-2)	9.53(-2)	9.53(-2)	0	0	0	0
Ba-140	8.44(-4)	8.44(-4)	8.44(-6)	8.44(-6)	5.18(-6)	5.18(-7)	2.60(-6)	2.60(-10)
La-140	8.06(-4)	8.06(-4)	8.06(-5)	8.06(-5)	1.94(-5)	1.94(-6)	9.68(-5)	9.68(-9)
Pr-143	8.06(-4)	8.06(-4)	8.06(-5)	8.06(-5)	5.05(-5)	5.05(-6)	2.56(-5)	2.56(-9)
Ce-144	5.70(-4)	5.70(-4)	5.70(-5)	5.70(-5)	5.45(-5)	5.45(-6)	2.72(-5)	2.72(-9)
Co-60	7.16(-4)	7.16(-5)	7.16(-5)	7.16(-5)	7.16(-5)	7.16(-5)	3.58(-3)	3.58(-7)
Fe-59	2.94(-5)	2.94(-6)	2.94(-6)	2.94(-6)	2.55(-6)	2.55(-6)	1.28(-4)	1.28(-8)
Co-58	6.41(-3)	6.41(-4)	6.41(-4)	6.41(-4)	5.92(-4)	5.92(-4)	2.96(-2)	2.96(-6)
Mn-54	3.79(-5)	3.79(-6)	3.79(-6)	3.79(-6)	3.71(-6)	3.71(-6)	1.86(-4)	1.86(-8)
Cr-51	5.25(-3)	5.25(-4)	5.25(-4)	5.25(-4)	4.17(-4)	4.17(-4)	2.10(-2)	2.10(-6)
Zr-95	1.30(-6)	1.30(-7)	1.30(-7)	1.30(-7)	1.18(-7)	1.18(-7)	5.90(-6)	5.90(-10)

*Numbers in ( ) are powers of ten.

TABLE 11.2-7

BORON RECOVERY SYSTEM (BRS) MAXIMUM NUCLIDE CONCENTRATIONS (70F) DURING ANTICIPATED OPERATIONS (μCi/cc)

Nuclide	CVCS 6 1	CVCS 7	CVCS 10 2, 3, 4, 5	6, 7, 8	12, 13, 14	15	16	18, 19, 20, 21
H-3	1.82(-1)	1.82(-1)	1.82(-1)	1.82(-1)	1.82(-1)	1.82(-1)	1.82(-1)	1.82(-1)
Br-84	6.43(-2)	6.43(-2)	6.43(-5)	6.43(-5)	0	0	0	0
Kr-85m	2.05(0)	2.05(0)	2.05(0)	1.02(0)	0	0	0	0
Kr-85	1.22(0)	1.22(0)	1.22(0)	6.10(-1)	1.08(-1)	1.08(-1)	0	5.40(-2)
Kr-87	1.12(0)	1.12(0)	1.12(0)	5.60(-1)	0	0	0	0
Kr-88	3.59(-0)	3.59(0)	3.59(0)	1.80(0)	0	0	0	0
Rb-88	3.52(0)	3.52(0)	3.52(0)	3.52(0)	0	0	0	0
Rb-89	8.85(-2)	8.85(-2)	8.85(-2)	8.85(-2)	0	0	0	0
Sr-89	7.00(-3)	7.00(-3)	7.00(-5)	7.00(-5)	6.00(-5)	6.00(-6)	3.08(-4)	3.08(-4)
Sr-90	3.60(-1)	3.60(-1)	3.60(-3)	3.60(-3)	3.60(-3)	3.60(-4)	1.80(-2)	1.80(-6)
Y-90	1.41(-3)	1.41(-3)	1.41(-3)	1.41(-3)	1.36(-4)	1.36(-5)	6.94(-4)	6.94(-8)
Sr-91	4.91(-3)	4.91(-3)	4.91(-5)	4.91(-5)	9.78(-12)	9.78(-12)	4.88(-11)	4.88(-15)
Y-91	1.53(-1)	1.53(-1)	1.53(-1)	1.53(-1)	1.38(-1)	1.38(-2)	6.92(-1)	6.92(-5)
Mo-99	2.81(0)	2.81(0)	2.81(0)	2.81(0)	2.90(-1)	2.90(-2)	1.45(0)	1.45(-4)
Ru-103	5.71(-3)	5.71(-3)	5.71(-4)	5.71(-4)	4.88(-4)	4.88(-5)	2.44(-3)	2.44(-3)
Ru-106	3.42(-4)	3.42(-4)	3.42(-5)	3.42(-5)	3.36(-5)	3.36(-6)	1.68(-4)	1.68(-8)
Te-129	3.46(-2)	3.46(-2)	3.46(-3)	3.46(-3)	0	0	0	0
I-129	9.95(-8)	9.95(-8)	9.95(-11)	9.95(-11)	9.95(-11)	9.95(-14)	4.98(-12)	4.98(-12)
I-131	5.48(0)	5.48(0)	5.48(-3)	5.48(-3)	2.53(-3)	2.53(-6)	1.26(-4)	1.26(-8)
Xe-131m	2.05(0)	2.05(0)	2.05(0)	1.03(0)	5.02(-1)	5.02(-1)	0	1.01(-1)
Te-132	4.55(-1)	4.55(-1)	4.55(-3)	4.55(-3)	6.68(-4)	6.68(-5)	3.34(-3)	3.34(-7)
I-132	1.41(0)	1.41(0)	1.41(-3)	1.41(-3)	0	0	0	0
I-133	7.83(0)	7.83(0)	7.83(-3)	7.83(-3)	6.29(-6)	6.29(-9)	3.14(-7)	3.14(-11)
Xe-133	2.50(+2)	2.50(+2)	2.50(+2)	1.25(+2)	3.16(+1)	3.16(+1)	0	6.33(0)

TABLE 11.2-7 (Cont'd)

Nuclide	CVCS 6 1	CVCS 7	CVCS 10 2, 3, 4, 5	6, 7, 8	12, 13, 14	15	16	18, 19, 20, 21
Te-134	3.35(-2)	3.35(-2)	3.35(-3)	3.35(-3)	0	0	0	0
I-134	8.55(-1)	8.55(-1)	8.55(-4)	8.55(-4)	0	0	0	0
Cs-134	3.38(-1)	3.38(-1)	3.38(-1)	3.38(-1)	3.38(-1)	3.38(-3)	1.69(-1)	1.69(-5)
I-135	3.73(0)	3.73(0)	3.73(-3)	3.73(-3)	0	0	0	0
Xe-135	1.04(+1)	1.04(+1)	1.04(+1)	5.20(0)	0	0	0	0
Cs-136	4.56(-2)	4.56(-2)	4.56(-2)	4.56(-2)	2.84(-2)	2.84(-4)	1.42(-2)	1.42(-6)
CS-137	1.38(0)	1.38(0)	1.38(0)	1.38(0)	1.38(0)	1.38(-2)	6.93(-1)	6.93(-5)
Xe-138	4.98(-1)	4.98(-1)	4.98(-1)	2.48(-1)	0	0	0	0
Cs-138	9.53(-1)	9.53(-1)	9.53(-1)	9.53(-1)	0	0	0	0
Ba-140	8.44(-3)	8.44(-3)	8.44(-5)	8.44(-5)	5.18(-5)	5.18(-6)	2.60(-5)	2.60(-9)
La-140	8.06(-3)	8.06(-3)	8.06(-4)	8.06(-4)	1.94(-4)	1.94(-5)	9.68(-4)	9.68(-8)
Pr-143	8.06(-3)	8.06(-3)	8.06(-4)	8.06(-4)	5.05(-4)	5.05(-5)	2.56(-4)	2.56(-8)
Ce-144	5.70(-3)	5.70(-3)	5.70(-4)	5.70(-4)	5.45(-4)	5.45(-5)	2.72(-4)	2.72(-8)
Co-60	7.16(-4)	7.16(-5)	7.16(-5)	7.16(-5)	7.16(-5)	7.16(-5)	3.58(-3)	3.58(-7)
Fe-59	2.94(-5)	2.94(-6)	2.94(-6)	2.94(-6)	2.55(-6)	2.55(-6)	1.28(-4)	1.28(-8)
Co-58	6.41(-3)	6.41(-4)	6.41(-4)	6.41(-4)	5.92(-4)	5.92(-4)	2.96(-2)	2.96(-6)
Mn-54	3.79(-5)	3.79(-6)	3.79(-6)	3.79(-6)	3.71(-6)	3.71(-6)	1.86(-4)	1.86(-8)
Cr-51	5.25(-3)	5.25(-4)	5.25(-4)	5.25(-4)	4.17(-4)	4.17(-4)	2.10(-2)	2.10(-6)
Zr-95	1.30(-6)	1.30(-7)	1.30(-7)	1.30(-7)	1.18(-7)	1.18(-7)	5.90(-6)	5.90(-10)

*Numbers in ( ) are powers of ten.

flow to the chemical drain tank. When a sufficient volume is collected in the drain tanks, the contents are sampled and neutralized, if required, and then pumped through a filter to the waste concentrator. The boric acid concentrators are available for processing liquid wastes if the waste concentration is not available for service.

Concentrator bottoms are pumped to the drumming station which is described in Section 1.5. The condensate (distillate) passes through the waste ion exchanger and is collected in the waste condensate tanks and monitoring for radioactivity. In the unlikely event that discharge limitations can not be met, the condensate can be recirculated through the waste ion exchanger, returned to the waste concentrator, or the holdup tanks in the boron recovery system for further treatment. Normally at least one holdup tank will be available for storage of waste condensate for reprocessing through a boric acid concentrator if necessary.

After the condensate activity has been determined to be sufficiently low by sample analysis, the tanks are pumped out at a controlled rate to the circulating water discharge. The activity of the discharge line effluent is monitored and recorded. Should the activity exceed the high set point value, the discharge is automatically terminated.

The alundry wastes are collected in the laundry drain tanks and analyzed for activity. Becasue of negligible activity levels, the laundry waste is normally pumped from the tanks through a filter to the circulating water discharge via the radiation monitor mentioned above. The tank contents can be processed in the waste concentrator prior to discharge if significant activity is detected.

All tanks are equipped with level instrumentation with alarm and their respective pumps are tripped on low level signals.

All piping 2 1/2 inch and smaller is field run. Line sizes are shown on Figure 11.2-4.

Designed data for the major components is given in Table 11.2-8. Flow, temperature, and pressure are given in Table 11.2-9 with the locations corresponding to data points on Figure 11.2-4. Expected performance of components are given in Table 11.2-10.

The nuclide concentrations adjusted to 70F for normal operation and anticipated operational occurrences are given in Tables 11.2-11 and 11.2-12, respectively. The selected locations are indicated by the process data points on Figure 11.2-4.

As indicated in Table 11.2-1, an estimated 156,000 gallons per year of liquid wastes, exclusive of laundry wastes and steam generator blowdown effluent, are processed in the liquid waste system. The activity of the aerated liquid wastes collected in the equipment drain tank and chemical train tank, in the absence of steam generator blowdown, is approximately 1 percent of the reactor coolant activity, owing to dilution from washdown and decontamination procedures.

Only processed when high level reached otherwise blown out.

Radioactive steam generator blowdown is processed in the system and a 20 gpd primary-to-secondary leakage rate is analyzed for expected normal operation with an average blowdown rate of approximately 200 gpd. A 120 gpd continuous leak with a 1200 gpd blowdown is analyzed for the anticipated operational occurrence. The specified blowdown is required to maintain the steam generator total dissolved solids (TDS) below a normal limit of 500 ppm (400 ppm boric acid plus 100 ppm of normal boiler water additives or impurities). A maximum of 1000 ppm TDS is allowed for operation which would reduce the required blowdown by a factor of 2.

Major radioactivity removal is accomplished by the concentrators. Experience indicates that a concentration DF (bottoms to distillate) of 10^4 can be obtained. The concentrators are specified at this rating and testing by the vendor will confirm the performance.

Table 11.2-13 lists the leakage sources & the assumptions used in calculating estimated normal and anticipated operational occurrence releases due to leakage sources.

The time dependent equations used for calculating the activities in the steam generator, condenser, and turbine system are:

$$\frac{dA^i_{sg}}{dt} = Q_i + F_c Ac^i - (\lambda^i + B)A_{sg}) - (F_{sg}) \cdot (PF^i_1)\ A^i_{sg}$$

$$\frac{dA^i_T}{dt} = (F_{sg})\ (PF^i_1) - (\lambda^i + L_T)\ A^i_T - F_T A^i_T$$

$$\frac{dA^i_c}{dt} = F_T\ (1-PF^i_2)\ A^i_T - \lambda^i\ A^i_c - F_c\ A^i_c$$

where:

A^i_{sg} = the total activity of isotope i in the steam generator liquid at time t , μCi.

Q_i = the steam generator leak rate for isotope i, μCi/sec

F_c = condensate flow rate, condenser volume/sec

A^i_c = condenser condensate activity of isotope i at time t, μCi

λ^i = decay constant for the i^{th} isotope, sec^{-1}

B = steam generator blowdown rate, vol/sec

F_{sg} = steam generator flow rate, vol/sec

PF_1^i = steam generator partition factor for the i^{th} isotope.

L_T = the turbine system leakage rate, vol/sec

A_T^i = activity in turbine steam for isotope i at time t, μCi

F_T = turbine flow rate, vol/sec

PF_2^i = condenser partition factor for isotope i

The above equations were solved for the equilibrium values of A_c^i, A_T^i and A_{sg}^i. The turbine system volume for this analysis includes the steam generator steam space, piping between the steam generator and turbine steam space, and condenser steam space. Condenser activity is based on the condensate volume in the hotwell and the piping between the hotwell and steam generator. Steam generator activity is based on the liquid inventory of steam generator.

TABLE 11.2-8
DESIGN DATA FOR LIQUID WASTE SYSTEM COMPONENTS

1. Ion Exchanger

	Waste
Quantity	1
Type	Deep Resin Bed
Design Pressure, psig	150
Design Temperature, F	200
Normal Operating Pressure, psig	40
Normal Operating Temperature, F	120
Resin Volume, ft^3	32
Materials	SS
ASME Code	VIII

2. Tanks

	Equipment Drain	Chemical Drain	Laundry Drain	Waste Condensate
Quantity	1	1	2	2
Internal Volume, gal.	1000	1000	1000	1725
Design Pressure, psig	Atmos.	Atmos.	Atmos.	Atmos.
Design Temperature, F	200	200	200	250
Normal Operating Pressure, psig	Atmos.	Atmos.	Atmos.	Atmos.
Normal Operating Temperature, F	120	120	120	120
Material	SS	SS	SS	SS
ASME Code, Division	VIII, I (1)	VIII, I (1)	VIII, I (1)	N.A.

3. Pumps

	Equipment Drain, Chemical Drain Laundry Drain, Waste Condensate
Quantity	6
Type	Centrifugal
Design Pressure, psig	150
Design Temperature, F	200
Design Conditions	
Flow, gpm	50
Head, ft	140
Wetted Materials	SS
Seal Type	Mechanical
Motor Horsepower	7.5
Motor Voltage	460
Code	N.A.

TABLE 11.2-8

4. Filters	Waste	Laundry
Quantity	1	1
Type of Elements	Replaceable Cartridge	Replaceable Cartridge
Particle Retention	25 micron, absolute	150 micron, nominal
Design Pressure, psig	150	150
Design Temperature, F	200	200
Design Flow, gpm	50	50
Material	Stainless Steel	Stainless Steel
ASME Code	VIII	VIII

5. Waste Concentrator

Quantity	1
Design Pressure, psig	80
Design Temperature, F	250
Design Flow, gpm	2
Cooling Water Flow Rate, gpm	130
Steam Required at 15 psig, lb/hr	1300
ASME Code	VIII

TABLE 11.2-9

LIQUID WASTE SYSTEM
PRESSURE, TEMPERATURE AND FLOW DATA

Mode 1[1]

Location	1	2	3	4	5	6	7	8	9
Pressure, psig	0.5	0.5	2.0	67	0.5	2.0	2.0	0.5	2.0
Temperature, F	120	120	120	120	120	120	120	120	120
Flow, gpm	0.825	.912	7	2	.209	0	0	0.223	0

Location	10	11	12	13	14	15	16
Pressure, psig	2.0	2.0	67	5	4	4	62
Temperature, F	120	120	120	120	120	120	120
Flow, gpm	0	0	2	2	2	55	50

Mode 2[2]

Location	1	2	3	4	5	6	7	8	9
Pressure, psig	0.5	0.5	2.0	2.0	0.5	2.0	67	0.5	2.0
Temperature, F	120	120	120	120	120	120	120	120	120
Flow, gpm	0.825	0.912	0	0	0.209	7	2	.223	0

Location	10	11	12	13	14	15	16	
Pressure, psig	2.0	2.0	67	5	4	62	62	
Temperature, F	120	120	120	120	120	120	120	12
Flow, gpm	0	0	2	2	2	55	50	

TABLE 11.2-9

Mode 3(3)

Location	1	2	3	4	5	6	7	8	9
Pressure (psig)	0.5	0.5	2.0	2.0	0.5	2.0	2.0	0.5	2.0
Temperature F	120	120	120	120	120	120	120	120	120
Flow gpm	0.825	0.912	0	0	0.209	0	0	0.223	55

Location	10	11	12	13	14	15	16
Pressure psig	62	47	2.0	0	0	0	0
Temperature, F	120	120	120	120	120	120	120
Flow, gpm	50	50	0	0	0	0	0

(1) Mode 1 Processing equipment drain tank contents via the waste concentrator; discharging a waste condensate tank.

(2) Mode 2 Chemical drain tank contents via the waste concentrator; discharging a waste condensate tank.

(3) Mode 3 Discharging a laundry drain tank.

TABLE 11.2-10

LIQUID WASTE SYSTEM EXPECTED PERFORMANCE

Ion Exchanger Decontamination Factor	
Tritium, Corrosion products	1
All Others	10
Filter Delay Factor	
All Nuclides	1
Waste Concentrator Delay Factor	200
Holdup Time for Liquid Waste System, days	
Anticipated	1
Normal	4.5

TABLE 11.2-11

LIQUID WASTE SYSTEM NORMAL ACTIVITY DISTRIBUTION (μCi/cc at 70F)

Location	1	2	3 & 12	13	14	15	21
H-3	2.47(-3)	1.48(-3)	2.21(-3)	2.21(-3)	2.21(-3)	2.21(-3)	2.21(-3)
Br-84	2.67(-7)	6.45(-5)	0	0	0	0	0
Kr 85m	0	0	0	0	0	0	0
Kr 85	0	0	0	0	0	0	0
Kr 87	0	0	0	0	0	0	0
Kr 88	0	0	0	0	0	0	0
Rb 88	6.37(-6)	3.53(-3)	0	0	0	0	0
Rb 89	1.33(-7)	8.86(-5)	0	0	0	0	0
Sr 89	9.53(-6)	7.02(-6)	8.34(-6)	1.67(-8)	1.67(-9)	1.57(-9)	1.67(-4)
Sr 90	6.02(-7)	3.61(-7)	5.39(-7)	1.07(-9)	1.07(-10)	1.07(-10)	1.07(-5)
Y 90	4.43(-7)	1.41(-6)	2.17(-7)	4.34(-10)	4.34(-11)	1.35(-11)	4.34(-6)
Sr 91	2.78(-7)	4.93(-6)	6.73(-10)	1.35(-12)	1.35(-13)	0	1.35(-8)
Y 91	2.15(-4)	1.54(-4)	1.89(-4)	3.78(-7)	3.78(-8)	3.59(-8)	3.78(-3)
Mo 99	9.13(-4)	2.81(-3)	4.65(-4)	9.30(-7)	9.30(-8)	3.05(-8)	9.30(-3)
Ru 103	7.38(-6)	5.72(-6)	6.41(-6)	1.28(-8)	1.28(-9)	1.18(-9)	1.29(-4)
Ru 106	5.73(-7)	3.44(-7)	5.08(-7)	1.02(-9)	1.02(-10)	1.01(-10)	1.02(-5)
Te 129	2.33(-7)	3.47(-5)	0	0	0	0	0
I 129	1.67(-10)	1.00(-10)	1.49(-10)	2.98(-13)	2.98(-14)	2.98(-14)	2.98(-9)
I 131	3.77(-3)	5.50(-3)	2.86(-3)	5.72(-6)	5.72(-7)	3.88(-7)	5.72(-2)
Xe 131m	0	0	0	0	0	0	0
Te 132	1.66(-4)	4.57(-4)	9.33(-5)	1.87(-7)	1.87(-8)	7.18(-9)	1.87(-3)
I 132	1.93(-5)	1.51(-3)	0	0	0	0	0
I 133	9.20(-4)	7.84(-3)	7.80(-5)	1.56(-7)	1.56(-8)	4.43(-10)	1.56(-3)
Xe 133	0	0	0	0	0	0	0
Te 134	1.56(-7)	3.62(-5)	0	0	0	0	0
I 134	4.92(-6)	8.59(-4)	0	0	0	0	0
Cs 134	5.64(-4)	3.38(-4)	5.02(-4)	1.01(-6)	1.01(-7)	1.01(-7)	1.01(-2)
I 135	1.48(-4)	3.74(-3)	1.54(-8)	3.08(-11)	3.08(-12)	0	3.08(-7)
Xe 135	0	0	0	0	0	0	0
Cs 136	4.05(-5)	4.56(-5)	3.29(-5)	6.58(-8)	6.58(-9)	5.16(-9)	6.58(-4)
Cs 137	2.31(-3)	1.39(-3)	1.75(-3)	3.53(-6)	3.53(-7)	3.53(-7)	3.53(-2
Xe 138	0	0	0	0	0	0	0
Cs 138	3.08(-6)	9.55(-4)	0	0	0	0	0
Ba 140	7.40(-6)	8.45(-6)	6.01(-6)	1.20(-8)	1.20(-9)	9.40(-10)	1.20(-4)
La 140	1.72(-6)	8.10(-6)	5.30(-7)	1.06(-9)	1.06(-10)	1.65(-11)	1.06(-5)
Pr 143	7.28(-6)	8.09(-6)	5.96(-6)	1.o9(-8)	1.19(-9)	9.48(-10)	1.19(-4)
Ce 144	9.53(-6)	5.72(-6)	8.43(-6)	1.69(-8)	1.69(-9)	1.67(-9)	1.69(-4)

TABLE 11.2-11 (Cont.)

Location	1	2	3 & 12	13	14	15	21
Co 60	1.20(-5)	7.20(-6)	1.07(-5)	2.14(-8)	2.14(-8)	2.14(-8)	2.14(-4)
Fe-59	1.38(-7)	2.95(-7)	1.68(-7)	3.36(-10)	3.36(-10)	3.13(-10)	3.36(-6)
Co 58	9.25(-5)	6.45(-5)	8.14(-5)	1.63(-7)	1.63(-7)	1.56(-7)	1.63(-3)
Mn 54	6.35(-7)	3.81(-7)	5.62(-7)	1.12(-9)	1.12(-9)	1.11(-9)	1.12(-5)
Cr 51	6.15(-5)	5.26(-5)	5.27(-5)	1.05(-7)	1.05(-7)	9.36(-8)	1.05(-3)
Zr 95	1.83(-8)	1.29(-8)	1.61(-8)	3.22(-11)	3.22(-11)	3.07(-11)	3.22(-7)

TABLE 11.2-12

LIQUID WASTE SYSTEM ANTICIPATED OPERATIONAL OCCURENCE ACTIVITY DISTRIBUTION (μCi/cc at 70F)

WMS Location	1	2	3 & 12	13	14	15	21
H-3	1.83(-2)	1.83(-3)	1.39(-2)	1.39(-2)	1.39(-2)	1.39(-2)	1.39(-2)
Br 84	1.24(-5)	6.45(-4)	0	0	0	0	0
Kr 85m	0	0	0	0	0	0	0
Kr 85	0	0	0	0	0	0	0
Kr 87	0	0	0	0	0	0	0
Kr 88	0	0	0	0	0	0	0
Rb 88	3.82(-4)	3.53(-2)	0	0	0	0	0
Rb 89	3.82(-4)	3.53(-2)	0	0	0	0	0
Rb 89	8.0(-6)	8.86(-4)	0	0	0	0	0
Sr 89	5.72(-4)	7.02(-5)	4.34(-4)	8.68(-7)	8.68(-8)	8.56(-8)	8.68(-3)
Sr 90	3.61(-5)	3.61(-6)	2.75(-5)	5.50(-8)	5.50(-9)	5.50(-9)	5.50(-4)
Y 90	2.66(-5)	1.41(-5)	1.80(-5)	3.60(-8)	3.60(-9)	2.78(-9)	3.60(-4)
Sr 91	1.67(-5)	4.93(-5)	4.56(-6)	9.12(-9)	9.12(-10)	1.64(-10)	9.12(-5)
Y 91	1.29(-2)	1.54(-3)	9.85(-3)	1.97(-5)	1.97(-6)	1.95(-6)	1.97(-1)
Mo 99	5.48(-2)	2.81(-2)	3.72(-2)	7.44(-5)	7.44(-6)	5.80(-6)	7.44(-1)
Ru 103	4.43(-4)	5.72(-5)	3.35(-4)	6.70(-6)	6.70(-7)	6.58(-7)	6.70(-2)
Ru 106	3.44(-5)	3.44(-6)	2.62(-5)	5.24(-7)	5.24(-8)	5.24(-8)	5.24(-3)
Te 129	1.40(-5)	3.47(-4)	0	0	0	0	0
I 129	1.00(-8)	1.00(-9)	7.62(-9)	1.52(-11)	1.52(-12)	1.52(-12)	1.52(-7)
I 131	2.26(-1)	5.50(-2)	1.66(-1)	3.32(-4)	3.32(-5)	3.04(-5)	3.32(0)
Xe 131m	0	0	0	0	0	0	0
Te 132	9.96(-3)	4.57(-3)	6.91(-3)	1.38(-5)	1.38(-6)	1.12(-6)	1.38(-1)
I 132	1.16(-3)	1.51(-2)	6.49(-7)	1.30(-9)	1.30(-10)	9.44(-14)	1.30(-5)
I 133	5.52(-2)	7.84(-2)	2.77(-2)	5.54(-5)	5.54(-6)	2.51(-6)	5.54(-1)
Xe 133	0	0	0	0	0	0	0
Te 134	9.33(-6)	3.62(-4)	0	0	0	0	0
I 134	2.95(-4)	8.59(-3)	0	0	0	0	0
Cs 134	3.38(-2)	3.38(-3)	2.58(-2)	5.15(-5)	5.15(-6)	5.15(-6)	5.15(-1)
I 135	8.85(-3)	3.74(-2)	1.37(-4)	2.74(-7)	2.74(-8)	2.28(-10)	2.74(-3)
Xe 135	0	0	0	0	0	0	0
Cs 136	2.43(-3)	4.56(-4)	1.81(-3)	3.62(-6)	3.62(-7)	3.46(-7)	3.62(-2)
Cs 137	1.38(-1)	1.38(-2)	1.02(-1)	2.04(-4)	2.04(-5)	2.04(-5)	2.04(0)
Xe 138	0	0	0	0	0	0	0
Cs 138	1.85(-4)	9.55(-3)	0	0	0	0	0
Ba 140	4.44(-4)	8.45(-5)	3.30(-4)	6.60(-7)	6.60(-8)	6.25(-8)	6.60(-3)
La 140	1.03(-4)	8.10(-5)	6.44(-5)	1.29(-7)	1.29(-8)	8.53(-9)	1.29(-3)
Pr 143	4.37(-4)	8.09(-5)	3.26(-4)	6.52(-7)	6.52(-8)	6.19(-8)	6.52(-3)

TABLE 11.2-12 (Cont'd)

WMS Location	1	2	3 & 12	13	14	15	21
Ce 144	5.72(-4)	5.72(-5)	4.36(-4)	8.72(-7)	8.72(-8)	8.72(-8)	8.72(-3)
Co 60	7.20(-5)	7.20(-6)	5.49(-5)	1.10(-7)	1.10(-7)	1.10(-7)	1.10(-3)
Fe 59	8.26(-7)	2.95(-7)	6.76(-7)	1.35(-9)	1.35(-9)	1.33(-9)	1.35(-5)
Co 58	5.55(-4)	6.45(-5)	4.21(-4)	8.42(-7)	8.42(-7)	8.34(-7)	8.42(-3)
Mn 54	3.81(-6)	3.81(-7)	2.90(-6)	5.80(-9)	5.80(-9)	5.80(-9)	5.80(-5)
Cr 51	3.69(-4)	5.26(-5)	2.78(-4)	5.56(-7)	5.56(-7)	5.41(-7)	5.56(-3)
Zr 95	1.10(-7)	1.29(-8)	8.35(-8)	1.67(-10)	1.65(-10)	1.65(-10)	1.67(-6)

TABLE 11.2-13

ASSUMPTIONS USED IN CALCULATING ESTIMATED NORMAL AND ANTICIPATED OPERATIONAL OCCURRENCE RELEASES

	Estimated Normal Releases	Anticipated Operational Occurrence
1. **Steam Generator Blowdown Releases**		
Main Steam flow rate, lb/hr	11,206,000	11,206,000 (1)
Steam generator liquid inventory per unit lb	130,500	130,500
Blowdown rate, gal/day	200	1,200
Blowdown rate, gal/yr.	72,500	435,000
Fuel failure, %	0.1	1.0
Iodine partition coefficient	5×10^{-2}	5×10^{-2}
2. **Releases to Reactor Auxiliary Building (2)**		
Fuel failure, %	0.1	1.0
Leakage to reactor auxiliary building, gal/day	10	10
Iodine partition coefficient	1×10^{-4}	1×10^{-4}
3. **Secondary System Releases**		
Steam generator leakage rate, gal/day	20	120
Failed fuel, %	0.1	1.0
Condenser iodine partition coefficient	5×10^{-4}	5×10^{-4}
Flow rate into condenser, lb/hr	7,840,000	7,840,000
Steam weight in main steam piping from steam generator to turbine stop valves (both loops), lb	7940	7940
Condenser Steam Space (3) lb	327	327
Turbine Steam (3), lb	87,000	87,000

(1) Released for the anticipated operational occurrence are assumed processed thru the waste management system.

(2) Assumed reactor coolant activities corrected to 120F.

(3) Including hotwell

Table 11.2-13 (Continued)

Steam Generator Steam Space,[4] lb	9,500	9,500
Total Steam Space, lb	104,767	104,767
Condenser hotwell water weight, lb	680,000	680,000
Liquid inventory between hotwell and steam generator, lb	587,000	587,000
Steam leakage to turbine building, lb/hr	20	20
Gland Steam Condenser flow rate, lb/hr	4602	4602

4. Containment Purge System

Releases independent of % failed fuel	
Reactor coolant system leak rate, lb/hr	50
Airborne activity cleanup flow rate	20,000 cfm = .48 volume/hr.
HEPA Filter Efficiency, % (5)	99.9
Charcoal Filter Efficiency, % (5)	
Organic Iodine	70
Inorganic Iodine	90
Operating time of removal system after shutdown, hr	10
Purge initiation after shutdown, hr	10
Purge flow rate	20,000 cfm = .48 volume/hr.
Purge frequency, times/yr.	4
Purge filter efficiencies, %	0
Iodine partition coefficient	0.1

5. Liquid Waste System Releases

Miscellaneous waste, gal	156,000
Laundry and showers	117,000

(3) Including hotwell

(4) Including piping

(5) These valves were also used for the auxiliary building gaseous releases.

11.2.3 OPERATING PROCEDURES

11.2.3.1 Boron Recovery System

Operating procedures for the boron recovery system are written to logically reflect the different monitoring and operating functions required of the plant operators. Procedures include the following.

a) Monitoring requirements of the flash tank pumps and control pumps and procedures for valve lineup to the holdup tanks.

b) Monitoring requirements of reactor drain tank operations and procedures for processing the tank contents through the flash tank to a holdup tank.

c) Monitoring and actions to be taken in the event of flash tank bypass operation to avoid explosive mixtures of hydrogen and air in the holdup tanks.

d) Monitoring requirements for the conditions of the four holdup tanks and processing requirements for chemical and radioactivity sample analysis.

e) Procedures and valve lineups for processing a holdup tank contents through a preconcentrator filter and ion exchanger back to the same or different tank and processing sampling requirements.

f) Procedures and valve lineups for processing a holdup tank contents through a preholdup filter and ion exchanger to either or both of the boric acid concentrators. Operation of pump and concentrator interlocks. Process sampling requirements.

g) Procedures for concentrator startup, steady state, shutdown, and layup operations. Sampling requirements to confirm low carryover and to confirm low carryover and to confirm proper operation of boron concentration controls. Automatic and manual batch bottoms discharge control procedures and heat tracing system operating procedures.

h) Monitoring requirements for the condensate tanks conditions and procedures for valve lineups. Sampling and reprocessing procedures.

i) Operating procedures for the environmental discharge instrumentation and valves. The method for determining the discharge flow rate in terms of radioactivity discharge limits.

j) Valve lineup procedures from the concentrators to the holding tank and sampling and transfer to the Chemical and Volume Control System or drumming station. Monitoring and operating requirements for the heat tracing system.

11.2.3.2 Liquid Waste System

Operating procedures for the liquid waste system include the following:

a) Procedures for equipment drain tank influent valve lineups, monitoring tank conditions, sampling, and valve lineups to the waste concentrator or a boric acid concentrator.

b) Procedures for chemical drain tank influent valve lineups, monitoring tank conditions, sampling and valve lineup to the waste or a boric acid concentrator.

c) Procedures for laundry tank influent valve lineups, monitoring tank conditions, sampling, and valve lineup for discharge or to the waste concentrator.

d) Procedures for waste concentrator startup, steady state, shutdown and layup operations. Sampling procedures to assure low carry-over and to monitor bottoms concentration. Heat tracing system operating procedures.

e) Monitoring requirements for waste condensate tank conditions, valve lineups, sampling, and reprocessing. Discharge is done under the same procedure used for the boron recovery system.

11.2.4 PERFORMANCE TESTS

The boron recovery and liquid waste processing systems will be tested prior to initial power plant start-up to verify satisfactory flow characteristics through the equipment, to demonstrate satisfactory performance of pumps and instruments, to check for leak tightness of piping and equipment, and to verify proper operation controls. All piping and components are to be checked to ensure that they are properly installed. All manual and automatic valves are operated and checked for functionability. All alarms are checked for operability and verification of locations. The concentrators are tested for operation before installation at the site and after installation to assure proper integration with the system. The boric acid and waste concentrators are shop tested prior to shipment to demonstrate compliance with performance objectives. During hot functional testing, the boron recovery system operation is integrated with the chemical and volume control system. The purpose of this test is to check the procedures and system components used for receiving and processing waste water. Boric acid transfer operations and waste liquid disposal procedures are tested.

During normal plant operation, periodic testing verifies that the system components are operating as designed. Filter performance is determined by comparing the quantity of particulate from inlet and outlet samples. Filters are monitored for differential pressure and radiation level on a regular basis. Ion exchanger performance is determined by comparing the level of radioactivity from inlet and outlet samples. Ion exchangers are monitored for differential pressure and radiation level on a regular

basis. To ensure that the flash tank is performing adequately, a liquid sample from the influent and effluent can be obtained to verify the hydrogen stripping function of the tank. When processing waste reactor coolant, boric acid concentrator performance is determined by comparing the concentration of boric acid from inlet, bottoms, and condensate samples. All liquid discharges to the environs are sampled for radioactivity before discharge. The discharge radiation monitor is calibrated on a regular basis to assure accuracy.

11.2.5 ESTIMATED RELEASES

The expected liquid releases from the plant are summarized by nuclide in Table 11.2-14. The reactor coolant concentrations utilized from this evaluation are indicated in Table 11.1-1. The curies released from the boron recovery and liquid waste systems are determined by multiplying the nuclide concentrations in the boric acid and waste condensate tanks as shown in Tables 11.2-6, 11.2-7, 11.2-11 and 11.2-12 by the waste processed through each system. The waste schedule is shown in Table 11.2-1. Technical Specifications for plant liquid releases are discussed in Section 16.3.9.

11.2.6 RELEASE POINTS

The only liquid release point from the waste management system to the environs is via the boric acid condensate or the waste condensate pumps discharge to the circulating water discharge. Prior to discharge the contents of the tank to be discharged are sampled for chemical and radioactivity concentrations. If the contents of the tank are found acceptable in terms of environmental discharge limits, the tank contents are discharged. A radiation monitor is provided in the discharge line to verify that the fluid discharge is below the applicable radioactivity limits. In the event that the discharged activity is unacceptable, the discharge monitor automatically terminates discharge operations. The discharge is located on the P & I diagram Figure 11.2-3 and 11.2-4. Release points from the plant are shown on Figure 11.2-5.

11.2.7 DILUTION FACTORS

An average annual dilution factor of 1.73×10^5 is obtained in the circulating water discharge based on 513,000 gpm flow in the discharge canal and an average annual liquid effluent flow rate of 2.95 gpm from the boron recovery system and the liquid waste system. The discharge and intake pipes are separated by 2300 feet of ocean shoreline which results in a neglible recirculation of discharged water.

11.2.8 ESTIMATED DOSES

None of the liquid releases from the plant result in doses via drinking water. The surrounding municipalities do not depend on the sea as a source of potable water. All local groundwater runoff is towards the ocean which further diminishes the possibility contamination of drinking water.

The only applicable exposure pathway comes from the ingestion of foods grown in the seawater containing plant releases. The doses calculated from this source follow the guidelines from "Statement on the Selection of as Low as Practicable Design Objectives and Technical Specifications for the Operation of Light Water Cooled Nuclear Power Reactors," by Carl C. Gamertsfelder. The population density information used for the calculations is from Section 2.1.3. The average daily consumption statistics used are taken from "Regional and Other Related Aspects of Shellfish Consumption - Some Preliminary Findings from the 1969 Consumer Panel Survey," by Morton M. Miller and Darrel A. Nash.

Total fish consumption by the population was calculated on the basis of the total edible fish harvest in St. Lucie County for 1970. It was assumed that 10 percent of the fish harvest came from waters containing plant effluent releases diluted to 5 percent of the concentration in the discharge canal. No decay time was assumed.

The maximum dose to an individual and total dose to the general public resulting from plant liquid releases are shown in Table 11.2-15. Doses to the whole body and gastro-intestinal tract were calculated with the largest dose occurring to the gastro-intestinal tract.

10/4

TABLE 11.2-14

EXPECTED LIQUID RELEASES FROM THE LIQUID WASTE SYSTEM

	Curies Released per Year		Activity in Circulating Water Discharge (µCi/cc)	
Nuclide	Normal Operation (0.1% Failed Fuel)	Anticipated Operation (1.0% Failed Fuel)	Normal Operation (0.1% Failed Fuel)	Anticipated Operation (1.0% Failed Fuel)
H-3	2.26 (+2)*	3.39 (+2)	2.2. (-7)	3.32 (-7)
Br-84	0	0	0	0
Kr-85m	0	0	0	0
Kr-85	3.74 (+1)	3.74 (+2)	3.67 (-8)	3-67 (-7)
Kr-87	0	0	0	0
Kr-88	0	0	0	0
Rb-88	0	0	0	0
Rb-89	0	0	0	0
Sr-89	1.13 (-5)	2.80 (-4)	1.12 (-14)	2.75 (-13)
Sr-90	6.90 (-7)	1.79 (-5)	6.80 (-16)	1.75 (-14)
Y-90	2.09 (-5)	2.70 (-4)	2.04 (-14)	2.64 (-13)
Sr-91	0	0	0	0
Y-91	2.11 (-2)	2.16 (-1)	2.07 (-11)	2.11 (-10)
Mo-99	4.60 (-2)	4.74 (-1)	4.50 (-11)	4.64 (-10)
Ru-103	7.55 (-5)	4.85 (-4)	7.44 (-14)	8.67 (-13)
Ru-106	5.60 (-6)	6.67 (-5)	5.50 (-15)	6.53 (-14)
Te-129	0	0	0	0
I-129	0	0	0	0
I-131	5.95 (-4)	6.74 (-2)	5.84 (-13)	6.60 (-11)
Xe-131m	3.73 (+1)	3.72 (+2)	3.65 (-8)	3.65 (-7)
Te-132	1.01 (-3)	1.27 (-2)	1.00 (-12)	1.24 (-11)
I-132	0	0	0	0
I-133	8.90 (-7)	5.76 (-3)	8.50 (-16)	5.65 (-12)
Xe-133	2.37 (+3)	2.37 (+4)	2.32 (-6)	2.32 (-5)
Te-134	0	0	0	0
I-134	0	0	0	0
Cs-134	1.75 (-3)	2.17 (-2)	1.70 (-12)	2.12 (-11)
I-135	0	0	0	0
Xe-135	0	0	0	0
Cs-136	3.40 (-4)	3.96 (-3)	3.33 (-13)	3.86 (-12)
Cs-137	5.28 (-3)	6.61 (-2)	5.16 (-12)	6.47 (-11)

Numbers in ( ) are powers of ten

TABLE 11.2-14

EXPECTED LIQUID RELEASES FROM THE LIQUID WASTE SYSTEM

Nuclide	Curies Released per Year: Normal Operating (0.1% Failed Fuel)	Curies Released per Year: Anticipated Operation (1.0% Failed Fuel)	Activity in Circulating Water Discharge (µCi/cc): Normal Operation (0.1% Failed Fuel)	Activity in Circulating Water Discharge (µCi/cc): Anticipated Operation (1.0% Failed Fuel)
Xe-138	0	0	0	0
Cs-138	0	0	0	0
Ba-140	9.39 (-6)	2.30 (-4)	9.08 (-15)	2.26 (-13)
La-140	2.99 (-6)	4.90 (-5)	2.95 (-15)	4.79 (-14)
Pr-143	7.89 (-5)	9.24 (-4)	7.78 (-14)	9.03 (-13)
Ce-144	8.72 (-5)	1.04 (-3)	8.54 (-14)	1.02 (-12)
Co-60	1.10 (-3)	1.12 (-3)	1.08 (-12)	1.10 (-12)
Fe-59	1.13 (-5)	1.14 (-5)	1.11 (-14)	1.12 (-14)
Co-58	9.08 (-3)	9.25 (-3)	8.89 (-12)	9.05 (-12)
Mn-54	5.71 (-5)	5.83 (-5)	5.59 (-14)	5.70 (-14)
Cr-51	5.91 (-3)	6.00 (-3)	5.79 (-12)	5.87 (-12)
Zr-95	1.79 (-6)	1.83 (-6)	1.75 (-15)	1.78 (-15)
Total: H-3	2.26 (+2)	3.39 (+2)	2.21 (-7)	3.32 (-7)
Total: Noble Gas	2.54 (+3)	2.45 (+4)	2.39 (-6)	3.32 (-7)
Total: All Others	9.13 (-2)	8.9 (-1)	9.03 (-11)	8.66 (-10)

* Numbers in ( ) are powers of ten.

TABLE 11.2-14 (Continued)

EXPECTED LIQUID RELEASES FROM THE LIQUID WASTE SYSTEM

Turbine Building Floor Drainage

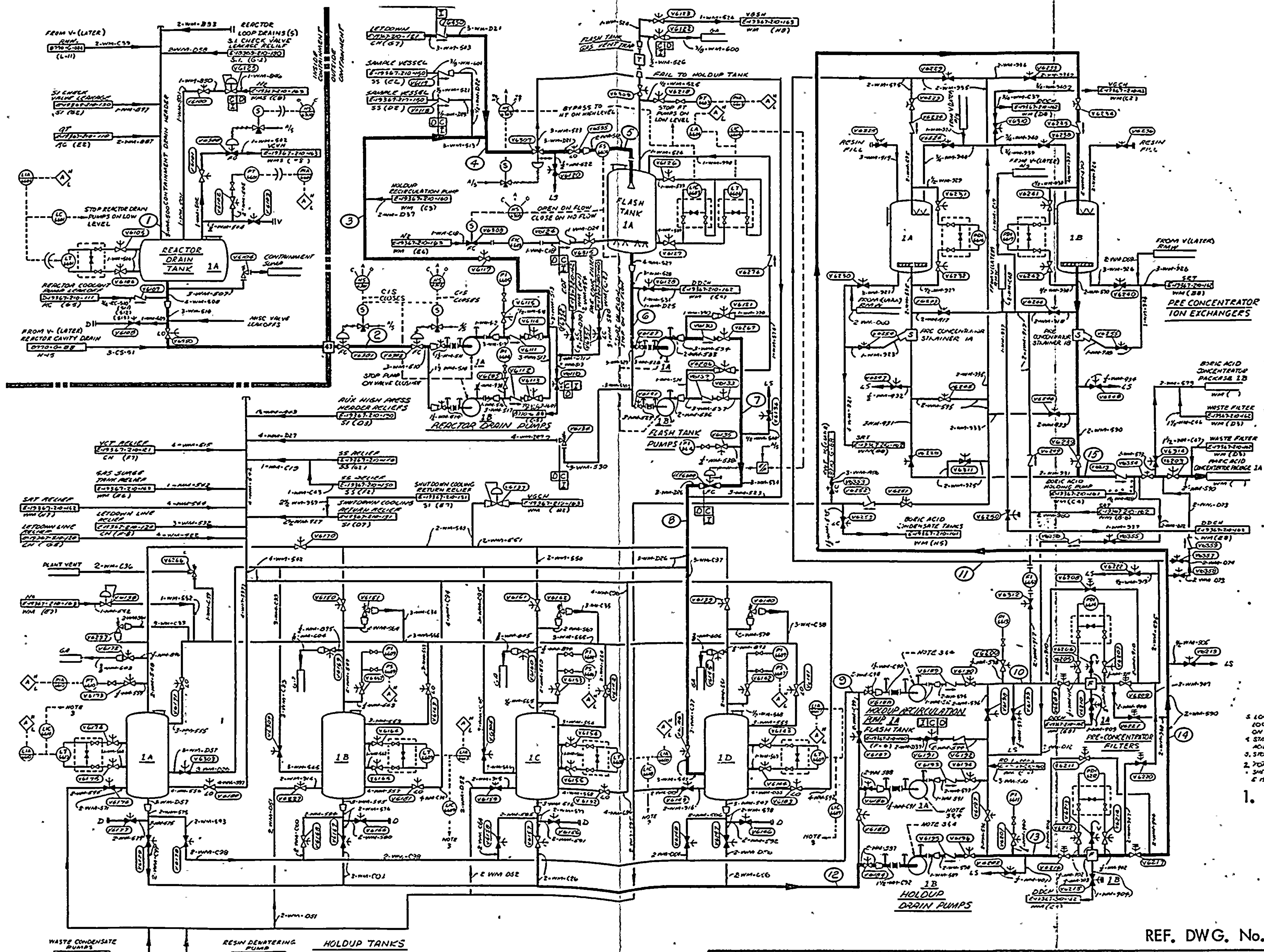
Nuclide	Curies Released Per Year Normal Operating (0.1% Failed Fuel)	Anticipated Operation (1.0% Failed Fuel)
H-3	1.29 (-1)	5.67 (-1)
Br-84	3.33 (-7)	2.11 (-5)
Rb-88	1.03 (-5)	6.18 (-4)
Rb-89	2.25 (-7)	1.36 (-5)
Sr-89	4.32 (-5)	1.32 (-4)
Sr-90	1.84 (-6)	6.14 (-5)
Y-90	1.99 (-11)	1.20 (-10)
Sr-91	2.67 (-6)	1.48 (-4)
Y-91	6.65 (-4)	2.38 (-2)
Mo-99	1.59 (-3)	8.76 (-2)
Ru-103	2.13 (-5)	8.08 (-4)
Ru-106	2.07 (-6)	6.36 (-5)
Te-129	1.21 (-4)	4.70 (-3)
I-129	6.49 (-10)	1.93 (-8)
I-131	7.66 (-3)	3.78 (-1)
Te-132	2.98 (-4)	1.62 (-1)
I-132	9.77 (-4)	5.33 (-2)
I-133	1.46 (-3)	8.55 (-2)
Te-134	2.65 (-7)	1.59 (-5)
I-134	1.13 (-6)	4.23 (-4)
Cs-134	4.33 (-4)	2.62 (-2)
I-135	6.41 (-4)	1.37 (-2)
Cs-136	8.94 (-4)	8.63 (-6)
Cs-137	3.23 (-3)	8.55 (-2)
Cs-138	4.42 (-6)	2.99 (-4)
Ba-140	1.64 (-5)	7.80 (-4)
La-140	2.85 (-6)	1.62 (-4)
Pr-143	1.66 (-5)	7.54 (-4)
Ce-144	3.38 (-5)	1.05 (-3)
Co-60	2.30 (-5)	1.38 (-4)
Fe-59	7.28 (-7)	4.37 (-6)
Co-58	1.20 (-4)	1.04 (-3)
Mn-54	1.12 (-6)	7.02 (-6)
Cr-51	1.11 (-4)	6.66 (-4)
Zr-95	2.02 (-11)	1.22 (-10)

TABLE 11.2-15

OFFSITE DOSES DUE TO PLANT

LIQUID RELEASES

	Normal Operation	Anticipated Operational Occurrence
Whole body dose due to ingestion of seafood:		
a) Maximum to individual	0.26 mrem/yr	13.4 mrem/yr
b) Total to population	0.056 man-rem/yr	2.9 man-rem/yr
Gastro-intestinal tract dose due to ingestion of seafood:		
a) Maximum to individual	1.8 mrem/yr	4.44 mrem/yr
b) Total to population	0.39 man-rem/yr	0.96 man-rem/yr



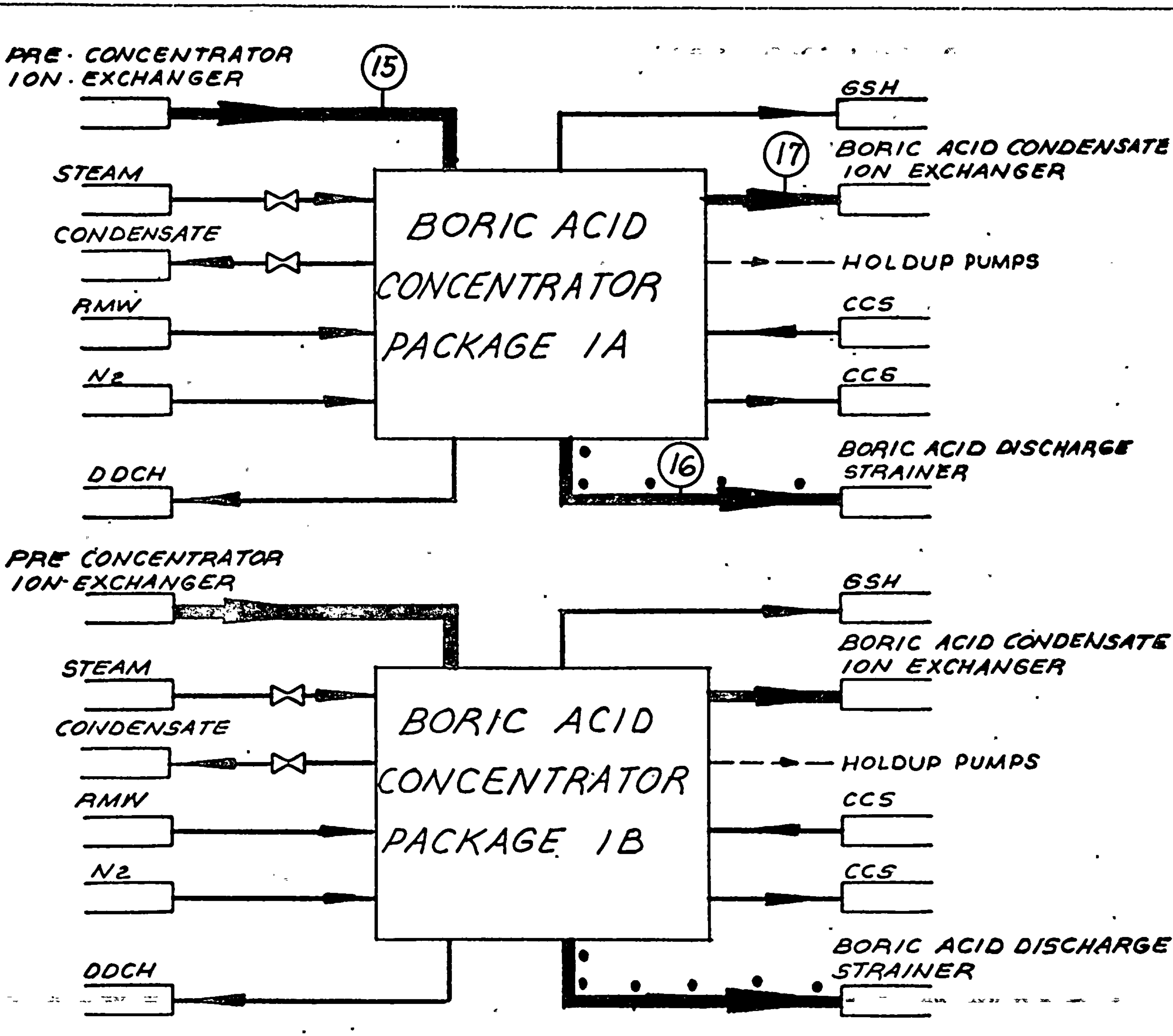
1. For Legend, Abbreviations, & Symbols See Fig. 1.2-20

13-161

REF. DWG. No. E-19367-210-160, REV. 05

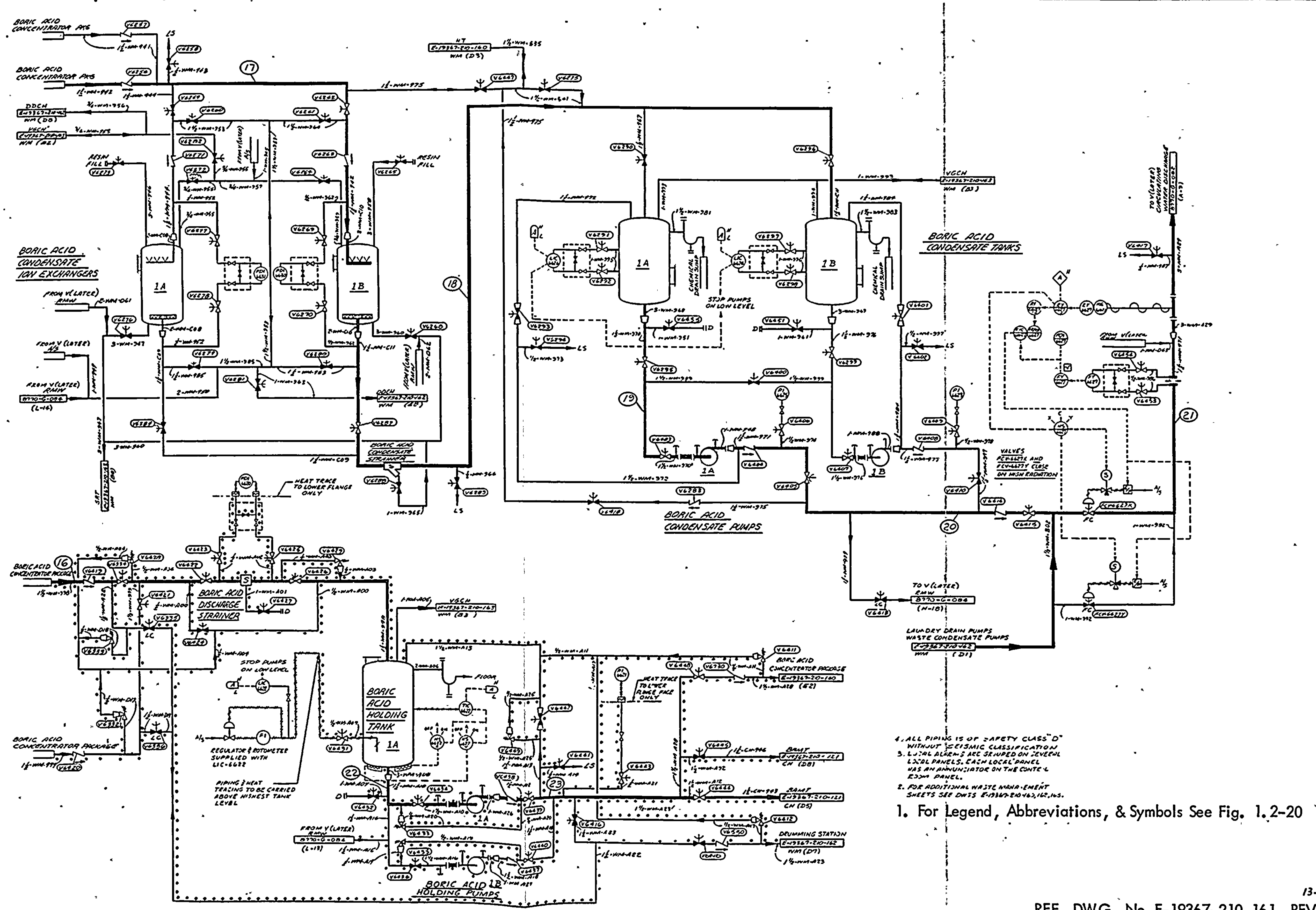
FLORIDA POWER & LIGHT CO. Hutchinson Island Plant	Waste Management System - P & I Diagram	Figure 11.2-1

FLORIDA POWER & LIGHT CO.
Hutchinson Island Plant



Boric Acid Concentrator Package

Figure 11.2-2

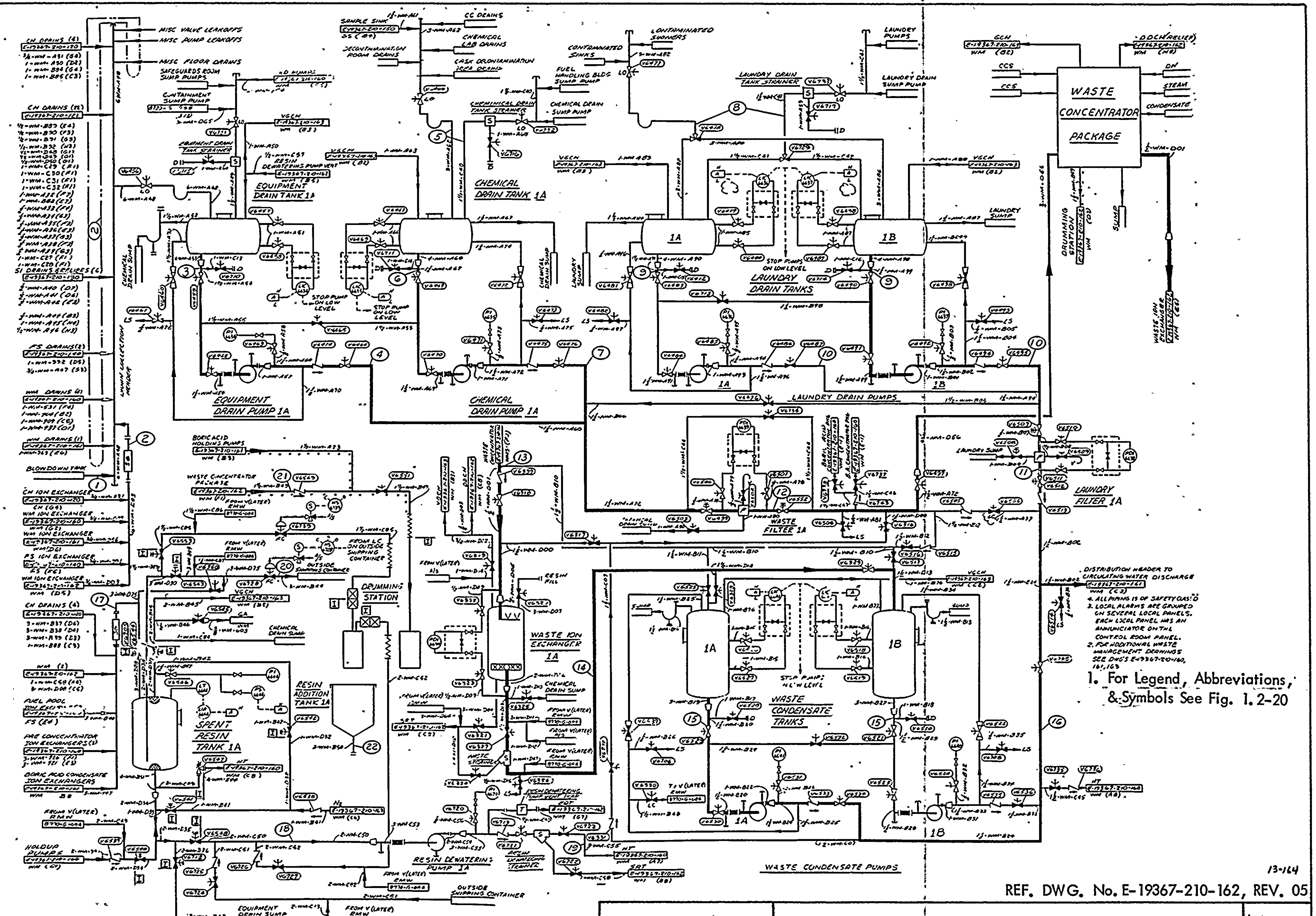


4. ALL PIPING IS OF SAFETY CLASS "D" WITHOUT SEISMIC CLASSIFICATION
3. LOCAL ALARMS ARE GROUPED ON SEVERAL LOCAL PANELS. EACH LOCAL PANEL HAS AN ANNUNCIATOR ON THE CONTROL ROOM PANEL.
2. FOR ADDITIONAL WASTE MANAGEMENT SHEETS SEE DWGS E-19367-210-162, 163, 165.
1. For Legend, Abbreviations, & Symbols See Fig. 1.2-20

13-163

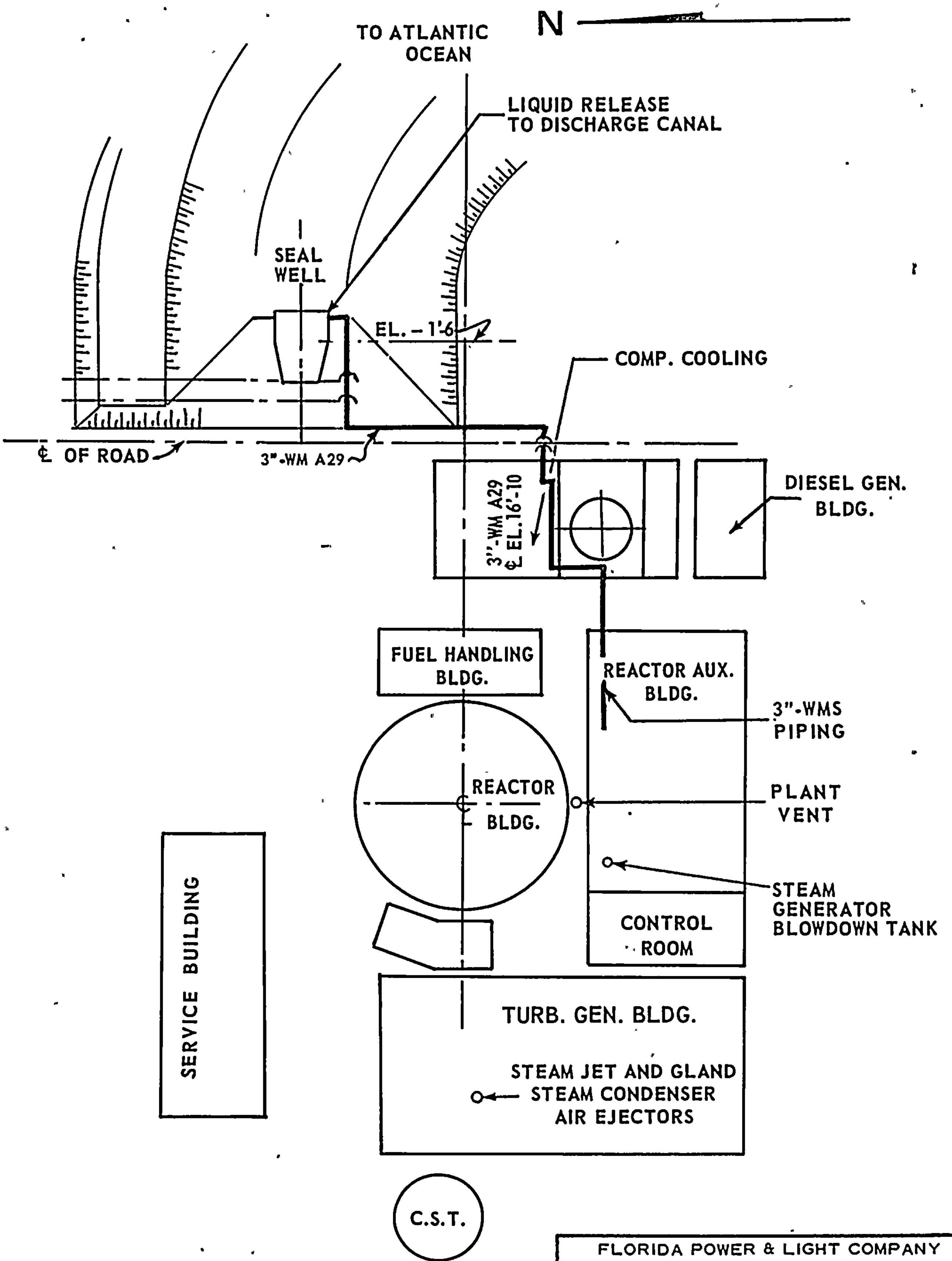
REF. DWG. No. E-19367-210-161, REV. 05

FLORIDA POWER & LIGHT CO. Hutchinson Island Plant	Waste Management System - P & I Diagram	Figure 11.2-3



REF. DWG. No. E-19367-210-162, REV. 05

FLORIDA POWER & LIGHT CO. Hutchinson Island Plant	Waste Management System - P & I Diagram	Figure 11.2-4



FLORIDA POWER & LIGHT COMPANY
HUTCHINSON ISLAND PLANT
UNITS 1 & 2

PLANT LIQUID AND GASEOUS RELEASE POINTS

FIGURE 11.2-5

11.3 GASEOUS WASTE SYSTEM

11.3.1 DESIGN BASES

The gaseous waste system processes the vent gases from equipment located in the chemical and volume control system, waste management system and fuel pool system, such that the radioactive gaseous release to the environs will be as low as practicable. The numerical design objective for releases during normal operation is to limit the site boundary noble gas dose to less than 10 mrem/year and iodine-131 and particulate site boundary concentrations to 10^{-5} times 10 CFR 20 limits. Releases due to anticipated operational occurrences will be within 10 CFR 20 limits.

11.3.2 SYSTEM DESCRIPTION

11.3.2.1 Waste Gas Processing Systems

The principal flow paths of the waste gas system are shown on Figure 11.3-1. Process flow and activity data are given in Table 11.3-1 and component data in Table 11.3-2.

Plant gaseous releases come from the steam generator blowdown, reactor auxiliary building ventilation, turbine system leakage, steam jet air ejector operation, gland steam condenser operation and containment purging in addition to releases from the gas collection header and gas surge header. Releases resulting from blowdown, turbine leakage, the steam jet air ejector and gland steam condenser are based on equations (1) (2) and (3) in Section 11.2.2.2. Gland steam has the same activity as turbine steam. Releases from the reactor auxiliary building ventilation system are based on leakage of unprocessed reactor coolant at 120 F. Containment purging results in a constant release regardless of percent failed fuel since the airborne radioactivity removal system is operated for as long as required to achieve the required activity level referred to in Section 12.2.

TABLE 11.3-1

WASTE GAS SYSTEM FLOW DATA POINTS

ANTICIPATED OPERATIONAL OCCURRENCE FOR 1% FAILED FUEL *

Position No.	1	2	3	4	5	6
Flow, ft^3/yr	12,944	100-500	1820	10,310	25,650	50,724
Pressure, psig	0.5-7	0.5-7	0.5-7	0.5-7	0.5-7	0.5-7
Temperature/F	70-120	70-120	70-120	70-120	70-120	70-120
Kr-85m $\mu\frac{Ci}{cc}$	0		11.98	8.02	2.54	2.92
Kr-85 "	0.522		0	5.86	1.41	2.47
Kr-87 "	0		0	4.36	1.318	1.55
Kr-88 "	0		0	13.9	4.48	5.10
Xe-131m "	0.735		13.1	8.5	2.58	3.7
Xe-133 "	71.8		850.0	980.0	325.0	412.0
Xe-135 "	0.0062		0	25.0	16.0	13.2
Xe-138 "	0		0	2.08	0.576	0.721
I-129 "	8.04 (-12)		5.82 (-14)	9.95 (-14)	4.08 (-12)	4.14 (-12)
I-131 "	3.42 (-3)		1.143 (-5)	5.47 (-5)	1.74 (-3)	1.762 (-3)
I-132 "	0		0	1.5 (-5)	5.9 (-6)	6.05 (-6)
I-133 "	5.87 (-4)		3.68 (-9)	7.8 (-5)	3.26 (-4)	3.33 (-4)
I-134 "	0		0	8.53 (-6)	3.44 (-6)	3.48 (-6)
I-135 "	0		0	1.036 (-4)	4.19 (-5)	4.25 (-5)

*Concentrations of nuclides are reduced by a factor of 10 for the expected operation of 0.1% failed fuel.

TABLE 11.3-1

WASTE GAS SYSTEM FLOW DATA POINTS

ANTICIPATED OPERATIONAL OCCURRENCE FOR 1% FAILED FUEL *

Position No.	7	8	9	10	11	12
Flow, ft^3/yr	50,724	50.724	50,724	50,724	50,724	395,660
Pressure, psig	0.5-7	10-165	10-165	10	<0.1	<0.1
Temperature/F	70-120	70-110	80	80	80	70-120
Kr-85m $\mu\frac{Ci}{cc}$	2.92	9.92	0	0	0	3.72 (-3)
Kr-85 "	2.47	8.40	8.36	2.46	2.46	2.52 (-2)
Kr-87 "	1.55	5.27	0	0	0	2.02 (-3)
Kr-88 "	5.10	17.32	0	0	0	6.47 (-3)
Xe-131m "	3.7	12.58	2.23	0.655	0.655	2.86 (-2)
Xe-133 "	412.0	1400.0	27.0	7.94	7.94	2.09
Xe-135 "	13.2	44.8	0	0	0	1.88 (-2)
Xe-138 "	0.721	2.45	0	0	0	9.02 (-4)
I-129 "	4.14 (-12)	1.41 (-11)	1.41 (-11)	4.14 (-12)	4.14 (-12)	2.95 (-13)
I-131 "	1.762 (-3)	6.0 (-3)	4.54 (-4)	1.335 (-4)	1.335 (-4)	6.32 (-5)
I-132 "	6.05 (-6)	2.06 (-5)	0	0	0	2.72 (-6)
I-133 "	3.33 (-4)	1.14 (-3)	0	0	0	1.41 (-5)
I-134 "	3.48 (-6)	1.182 (-5)	0	0	0	1.592 (-6)
I-135 "	4.25 (-5)	1.442 (-4)	0	0	0	6.72 (-6)

*Concentrations of nuclides are reduced by a factor of 10 for the expected operation of 0.1% failed fuel.

TABLE 11.3-2

COMPONENT DATA

1. Waste Gas Compressor

Type	Diaphragm Positive Displacement
Quantity	2
Capacity, SCFM	2
Discharge Pressure, psig	0-165
Codes	ASME P&V III for Nuclear Power Nov. 1968 ASME Power Test Code PTC-9 , Displacement Compressors, Vacuum Pumps and Blowers.
Materials	Carbon Steel
Design Temperature, F	150 - Inlet; 350 - Outlet
Design Pressure, psig	200

2. Compressor Aftercooler

Type	Shell and Tube
Quantity	2
Codes: Gas Side	ASME Boiler and Pressure Vessel Code 1968 Section III Unfired Pressure Vessels, Class C
Shell Side	Section VIII of above code
Materials	Carbon Steel
Discharge Temperature, F	110

3. Compressor Inlet Filter

Type	Stainless Steel Screen
Quantity	2
Rating	5 micron
Clean Pressure Drop, psi @ 2 CFM	0.3
Code	ASME Boiler and Pressure Vessel Code 1968 Section III Unfired Vessels, Class C

TABLE 11.3-2

4. Gas Surge Tank

Type	Vertical
Quantity	1
Volume, ft^3	10
Design Pressure, psig	40
Design Temperature, F	200
Code	ASME Boiler and Pressure Vessel Code 1968, Section III, Class C
Material	Carbon Steel

5. Gas Decay Tank

Type	Vertical
Quantity	3
Volume, each, ft^3	144
Design Pressure, psig	190
Design Temperature, F	250
Codes	ASME Boiler and Pressure Vessel Code 1968, Section III, Class C
Material	Carbon Steel

Waste gas is collected from the various source components by three headers; containment vent header, gas surge header, and gas collection header. The containment vent header receives hydrogenated potentially radioactive gas mixtures vented from the reactor drain tank, quench tank, refueling failed fuel detector vent, and reactor vessel head vent within the containment and directs the gases to the gas surge header. Hydrogenated and potentially radioactive gases vented from the volume control tank, flash tank, and boric acid concentrators in the reactor auxiliary building are also directed to the gas surge header along with the discharge gas from the gas analyzer. The vented gases flow to the surge tank where they are collected prior to being compressed. The gases remain in the surge tank until removed by the waste gas compressors which are automatically controlled by pressure instrumentation located on the surge tank. The surge tank is equipped with a drain line to remove any water that accumulates in the tank due to condensation. A level switch with a local alarm is on the surge tank indicates to the operator when the tank should be emptied.

Since the contents of the surge tank are expected to contain significant amounts of hydrogen, the gas is sampled frequently by the gas analyzer to determine the oxygen-hydrogen composition. If oxygen should increase above 2 percent by volume (2 percent by volume below allowable limit) an alarm is annunciated, and provisions are made to purge with nitrogen. A nitrogen line connected to the surge tank, and a pressure regulating valve in the line opens when pressure in the tank falls below 1.5 psig thereby maintaining a positive pressure above atmospheric and preventing air ingress.

The gases flow from the gas surge tank to a compressor where they are compressed to 165 psig and cooled by an aftercooler prior to entering the gas decay tanks where the gases are held up for radioactive decay. Two compressors are available for transferring the gas to the gas decay tanks controlled by pressure instrumentation on the surge tank. One compressor starts when pressure in surge tank increases to 3 psig and stops when pressure falls to 1.5 psig. The second compressor starts at 7 psig and stops at 5 psig.

Aftercoolers supplied with each gas compressor cool the compressed gas prior to entering the gas decay tanks. There are three gas decay tanks (each provided with a pressure indicator including local alarm and temperature indicator) which receive the compressed gas from the waste gas compressors. The decay tanks have sufficient storage capacity for an average 30 day holdup, and after radioactivity has decayed to an acceptable level which is consistent with the design objective and has been verified by laboratory sample analysis, the gas is released to the environment via the plant vent at a controlled rate.

The fill procedure for the decay tanks is to have only one tank lined up to the compressor discharge. When the pressure in the tank increases to 165 psig the tank is isolated and manually switched over to an empty tank. The gaseous radioactivity in the filled isolated tank is allowed to decay for an average decay time of approximately 30 days. During this decay period the gas is periodically sampled and activity level determined. The sampling technique used prior to release of gas and the continuous monitoring systems are further discussed in Section 11.4.3.

The following components are located in the discharge line from the gas decay tanks to the plant vent; a pressure reducing valve, pressure indicator, needle valve, pneumatic operated fail closed on-off valve, an in-line radiation monitor, and a gas flow meter.

Prior to release, the required flow rate is determined, and the set point on the radiation monitor is established. Initially the discharge valve from the gas decay tank, needle valve and pneumatic operated valve are closed. The on-off pneumatic operated valve is opened and placed in automatic, and the discharge valve on the desired gas decay tanks is opened. The pressure reducing valve will automatically close when pressure from the decay tank in excess of 10 psig is sensed at the pressure reducing valve outlet. The needle valve is then opened as required to establish the desired flow rate to the plant vent, and the pressure reducing valve will open to maintain constant downstream pressure of 10 psig. The on-off pneumatic operated valve automatically closes on high radioactivity level thus terminating discharge flow. An alarm will annunciate this event in the control room. When discharge flow decreases and the decay tank pressure decreases to approximately 10 psig, as noted by observing the gas flow meter and pressure indicator on the gas decay tank, the pressure reducing valve set point must be reduced to vent the tank down to atmospheric pressure.

The system flow paths and release points of the gases from the gas decay tanks and gas collection header are indicated on Figure 11.3-1. The diagram also shows the only flow bypass line in the waste gas system. This line from the gas surge tank to the plant vent bypasses the waste gas compressor and gas decay tanks. This path is used to bypass compressors and gas decay tanks when the air or nitrogen is purged from process equipment after initial plant startup or maintenance operations. During these periods essentially no activity will be present in the gas streams and it is unnecessary to route these gases to the gas decay tanks. A locked closed valve facilitates administrative control of this bypass line.

The waste gas system has connections for sampling the gas in the containment vent header gas surge tank, and each gas decay tank. The gas to the gas surge header is primarily hydrogen, and a gas analyzer located in the sampling system is used to monitor hydrogen and oxygen concentration. Connections are available in the sampling system as outlined in Section 11.4.3 for taking a grab sample via gas analyzer to determine activity level prior to release.

The gas collection header collects the gases from primarily aerated vents of process equipment in the waste management system, chemical and volume control system, and fuel pool system. A listing of sources is given in Table 11.3-3. Because of the large volume of gas and the low activity level from the sources, the gases are routed directly to the plant vent. The gases and expected activities to the plant vent from the gas collection header are given in Table 11.3-1 at process data point No. 12. As a further check on activity from this source, the plant vent contains radioactivity monitors with alarms to indicate unexpected activity release.

TABLE 11.3-3

GAS COLLECTION HEADER SOURCE POINTS

1. Preconcentrator ion exchanger vent
2. Holdup tank vent
3. Boric acid condensate ion exchanger vent
4. Boric acid holding tank vent
5. Boric acid condensate tank vents
6. Waste ion exchanger
7. Equipment drain tank vent
8. Chemical drain tank vent
9. Laundry drain tank vents
10. Waste condensate tank vents
11. Spent resin tank vent
12. Waste concentrator vent
13. CVCS ion exchanger vents
14. Fuel pool system ion exchanger vent
15. Boric acid makeup tank vents
16. Charging pump vents
17. Charging pump seal lubrication tank vents
18. Boric acid makeup pump vents

The hydrogen and nitrogen required for plant operations are also a part of this system and redundant supply headers for each gas are provided. Hydrogen gas is supplied to the volume control tank gas space to maintain the desired concentration of reactor coolant dissolved hydrogen to suppress the net decomposition of water in the reactor flux. Nitrogen cover and/or purge gas is provided to the holdup tanks, quench tank, reactor drain tank, safety injection tanks, spent resin tank, and gas surge tank. A nitrogen stream is supplied to the flash tank for degassing liquid waste, and periodic purges with nitrogen are provided as required for various waste management system and chemical and volume control system components. The two gas supply systems include relief valves, regulators, and instrumentation with alarms and valving to allow flexible operation. A low pressure alarm indicates when the backup source should be placed in service.

11.3.3 OPERATING PROCEDURE

The waste gas system operating procedures include the following.

a) Tank purging and venting procedures for all tanks venting to the waste gas system and for the tanks in the waste gas system.

b) Gas surge tank to gas decay tank valve and compressor lineup procedures, requirements for monitoring proper automatic operation of the gas compressors, procedures for servicing the compressors, filters, and after coolers.

c) Procedures for isolating a gas decay tank after filling and for monitoring the contents during the decay period.

d) Procedures for setting the high trip set point on the discharge radiation monitor, for lining up the valves for discharge, and for determining the discharge flow rate and establishing it, monitoring requirements during the discharge period, procedure to reduce gas decay tank pressure to atmospheric at the end of the discharge period.

e) Procedures for draining water from the gas surge tank and the gas decay tanks.

f) Procedures for valve lineup, and establishing regulator set points and criteria for changing hydrogen gas cylinders.

g) Procedures for valve lineup and establishing regulator set points and criteria for changing nitrogen gas cylinder.

h) Procedures for gas analyser manual and automatic control of sample point selection, procedures for obtaining a gas sample in the sampling cylinder for laboratory analysis and for setting the sequential timer and for calibrating the gas analyzers to assure representative automatic sampling.

11.3.4 PERFORMANCE TESTS

The gas compressors including suction filter and discharge aftercooler are shop tested to assure proper operation. The gas analyzer system is shop tested.

The system preoperational tests are as follows:

a) Instrumentation is checked for accuracy of readout and control and alarm set points;

b) Control valves are checked for proper operation and relief and pressure regulation valve set points are confirmed;

c) Operation of the entire system is checked by supplying nitrogen from various points such as the reactor drain tank, and flash tank. Proper automatic operation of the compressors and controllability of discharge flow rate is verified;

d) Leak testing is done to assure negligible leakage.

During plant operation, periodic testing is done as follows:

a) The discharge radiation monitor is periodically calibrated with standard radiation source;

b) The hydrogen and oxygen gas analyzers are periodically calibrated with zero and upscale gases;

c) Process instrumentation is periodically calibrated.

11.3.5 ESTIMATED RELEASES

The estimated releases from the waste gas system in curies per year are listed in Table 11.3-4. The source terms and assumptions used to determine these releases are given in Section 11.3.2. All the noncondensable gases in the condenser are released to the environment. The releases from the auxiliary building are calculated by multiplying the leak rate for auxiliary building equipment by the ventilation system filter efficiencies. The assumed leak rates and filter efficiencies are given in Table 11.2-13.

During normal operation the expected activity releases from the plant is based on a 0.1 percent failed fuel, while the anticipated operational occurrences are assumed at 1.0 percent failed fuel. The radioactivity that is released from the failed fuel is reduced in concentration by the process equipment in the chemical and volume control system and waste managment system. For iodine, the purification and preconcentrator ion exchanger decontamination factor (DF) is 10^3, and a liquid to gas partition factor of 10^4 is used for the reactor drain tank, flash tank, holdup tank, equipment drain tank, and boric acid concentrator. Noble gases are stripped

10/4

TABLE 11.3-4

ESTIMATED NORMAL AND ANTICIPATED OPERATIONAL OCCURRENCE GASEOUS RELEASES (Curies per Year)

Isotope	Gas Surge Header		Gas Collection Header		Steam Generator Blowdown Tank		Reactor Auxiliary Building Ventilation	
	Estimated Normal Releases	Anticipated Operational Occurrence Releases	Estimated Normal Releases	Anticipated Operational Occurrence Releases	Estimated Normal Releases	Anticipated Operational Occurrence Releases	Estimated Normal Releases	Anticipated Operational Occurrence Releases
Kr^{85m}	0.0	0.0	4.15(0)	4.15(1)	2.06(5)	7.52(-3)	1.65(0)	1.65(1)
Kr-85	3.52(2)	3.52(3)	2.82(1)	2.8191(2)	1.21(-5)	4.47(-3)	9.78(-1)	4.26(0)
Kr-87	0.0	0.0	2.26(0)	2.26(1)	1.05(-5)	3.82(-3)	8.95(-1)	8.96(0)
Kr-88	0.0	0.0	7.23(0)	7.23(1)	3.59(-5)	1.31(-2)	2.87(0)	2.87(1)
I-129	5.94(-10)	5.94(-9)	3.2(-10)	3.2(-9)	7.9(-10)	2.88(-7)	7.97(-12)	7.97(-11)
I-131			1.25(-3)	1.25(-2)	1.55(-2)	5.65(0)	4.39(-4)	4.39(-3)
Xe^{131}	9.38(1)	9.38(2)	3.198(1)	3.1984(2)	2.03(-5)	7.49(-3)	1.64(0)	1.64(1)
I-132	4.0(-3)	4.0(-4)	3.03(-3)	3.03(-2)	3.53(-3)	1.28(0)	1.13(-4)	1.13(-3)
I-133	0.0	0.0	1.571(-2)	1.571(-1)	3.53(-3)	1.28(0)	6.26(-4)	6.26(-3)
Xe-133	1.138(3)	1.138(4)	2.3304(3)	2.33042(4)	2.49(-3)	9.12(-1)	2.00(+2)	2.00(+3)
I-134	0.0	0.0	1.78(-3)	1.78(-2)	1.75(-5)	6.38(-3)	1.10(-5)	1.10(-4)
I-135	0.0	0.0	7.51(-3)	7.51(-2)	5.62(-4)	2.05(-1)	8.32(-4)	8.32(-3)
Xe-135	0.0	0.0	2.10(1)	2.10(2)	4.22(-2)	1.54(+1)	8.32(0)	8.32(+1)
Xe-138	0.0	0.0	1.0(0)	1.002(1)	4.82(-6)	1.76(-3)	2.87(-1)	2.87(0)

10/4

TABLE 11.3-4

ESTIMATED NORMAL AND ANTICIPATED OPERATIONAL OCCURRENCE GASEOUS RELEASES (Curies per year)

Isotope	Turbine Building Leakage from Secondary System		Steam Jet Air Ejector		Containment Purge	
	Estimated Normal Releases	Anticipated Operational Occurrence Releases	Estimated Normal Releases	Anticipated Operational Occurrence Releases	Estimated Normal Releases	Anticipated Operational Occurrence Releases
Kr^{-85m}	8.36(-6)	5.01(-4)	3.28(0)	1.97(+2)	7.42(-1)	7.42(-1)
Kr^{-85}	2.17(-6)	1.30(-4)	8.52(0)	5.11(+1)	2.03(+1)	2.03(+1)
Kr^{-87}	4.51(-6)	2.70(-4)	1.77(0)	1.06(+2)	2.99(-1)	2.99(-1)
Kr^{-88}	1.46(-5)	8.74(-4)	5.71(0)	3.43(+2)	1.17(0)	1.17(0)
I^{-129}	3.21(-10)	1.93(-8)	1.27(-7)	3.78(-6)	2.84(-9)	2.84(0)
I^{-131}	6.30(-3)	3.78(-1)	1.23(0)	7.41(+1)	2.84(-9)	2.84(-2)
Xe^{-131}	3.30(-6)	1.98(-4)	3.27(0)	1.96(+2)	1.56(-1)	1.56(-1)
I^{-132}	8.88(-4)	5.33(-2)	2.46(-1)	1.48(+1)	3.40(-2)	3.40(-2)
I^{-133}	1.42(-3)	8.55(-2)	2.60(-1)	1.68(+1)	2.18(-1)	2.18(-1)
Xe^{-133}	6.50(-4)	3.91(-2)	4.00(+2)	2.40(+4)	2.18(-1)	2.18(-1)
I^{-134}	7.03(-6)	4.23(-4)	2.20(-3)	1.32(-1)	1.67(-2)	1.67(-2)
I^{-135}	2.28(-4)	1.37(-2)	4.50(-2)	2.68(0)	1.00(-1)	1.00(-1)
Xe^{-135}	8.64(-4)	5.13(-2)	2.6(-1)	1.55(+1)	4.46(0)	4.46(0)
Xe^{-138}	1.80(-6)	1.10(-4)	7.4(-1)	4.45(+1)	5.82(-2)	5.82(-2)

by a factor of approximately 2 in the flash tank and 5 in the boric acid concentrator. Activity will also be reduced by natural decay due to residence time in the reactor drain tank, holdup tank, and gas decay tanks.

11.3.6 RELEASE POINTS

Release points to the plant vent from the gas decay tanks are indicated on Figure 11.3-1. The gaseous effluents from the reactor auxiliary building ventilation system and containment purge system are also released from the vent. The plant vent is shown on reactor auxiliary building general arrangement Figure 1.2-14, and on the ventilation systems flow diagram, Figure 9.4-1. Steam generator blowdown is released to the blowdown tank in the reactor auxiliary building. The blowdown tank is shown on general arrangement, Figures 1.2-15, and Figure 10.4-1. The air ejector releases are from the turbine area. The air ejectors are shown on Figures 1.2-5 and 10.1-3 flow diagram. All the release points are shown on the site plot plan (Figure 11.2-5).

11.3.7 DILUTION FACTORS

The value of X/Q used for the plant gaseous releases at the site boundary is 1.73×10^{-6} sec/m^3 which is based on the worst average annual atmospheric dispersion condition as discussed in Sections 2.3.4 and 2.3.5. Doses to the population at greater distances up to 50 miles are calculated using the worst average meteorological data given in Table 2.3-69.

11.3.8 ESTIMATED DOSES

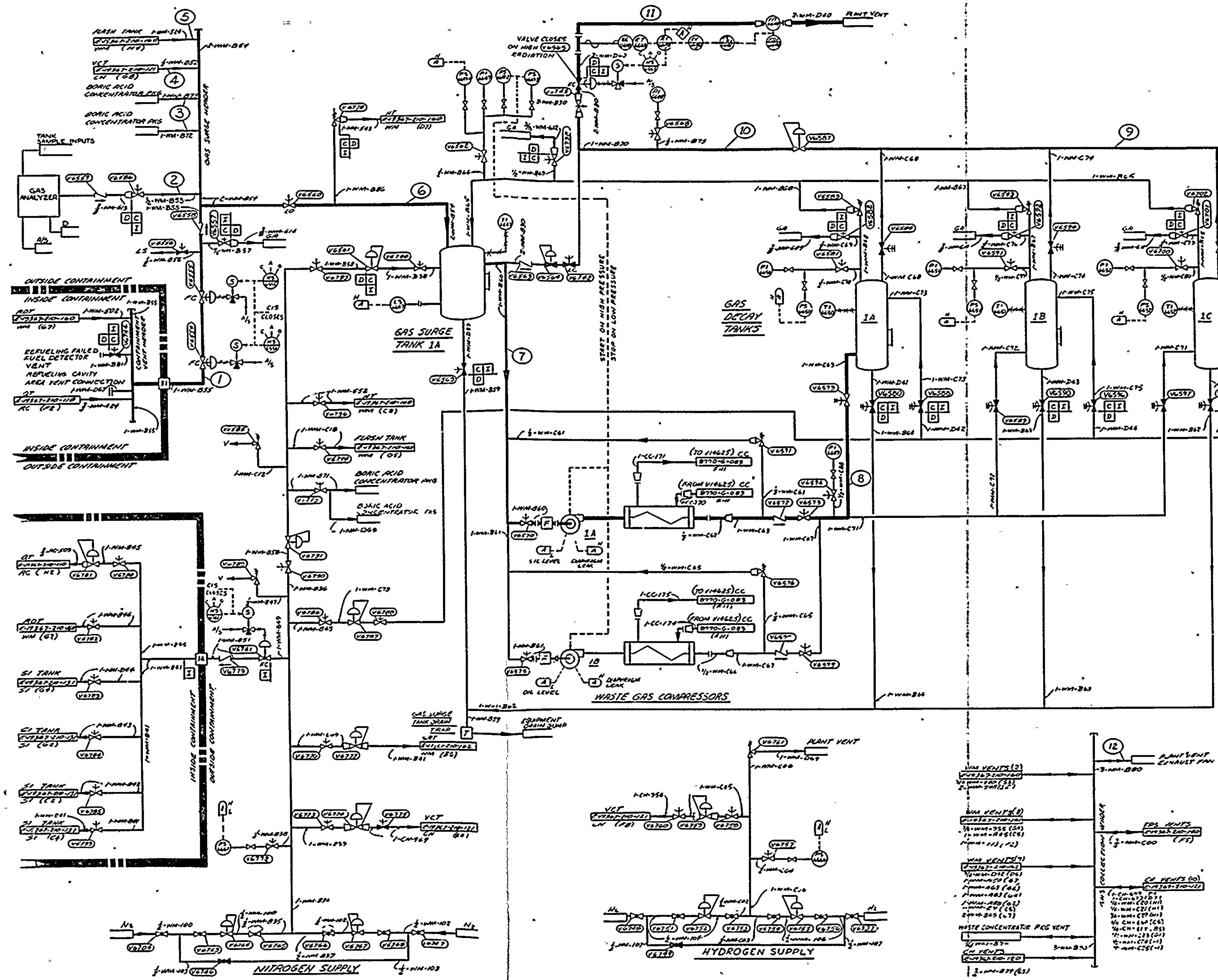
Doses received by the general public resulting from the gaseous releases discussed above are listed in Table 11.3-5. The results are presented for the maximum whole dose to an individual, the maximum thyroid dose to an individual, and the whole body and thyroid doses to the population within a 50 mile radius of the plant. The man-rem doses are calculated from population density data in Section 2.1.3.

10/4

TABLE 11.3-5

OFFSITE DOSES DUE TO PLANT GASEOUS RELEASES

	Normal Operation	Anticipated Operational Occurrence
Whole body dose to:		
a) Individual at site boundary	0.144 mrem/yr	2.44 mrem/yr
b) Population within 50 mile radius	0.52 man-rem/yr	9.58 man-rem/yr
Thyroid dose to:		
a) Individual at site boundary	1.49 mrem/yr	50 mrem/yr
b) Population within 50 mile radius	5.0 man-rem/yr	181 man-rem/yr



3. LOCAL ALARMS ARE GROUPED ON SEVERAL LOCAL PANELS. EACH LOCAL PANEL HAS AN ANNUNCIATOR ON THE CONTROL ROOM PANEL.

2. FOR ADDITIONAL WASTE MANAGEMENT DRAWING SEE E-19367-210-160, [illegible]

1. For Legend, Abbreviations, & Symbols See Fig. 1.2-20

13-165

REF. DWG. No. E-19367-210-163, REV. 05

FLORIDA POWER & LIGHT CO.
Hutchinson Island Plant

Waste Management System - P & I Diagram

Figure 11.3-1

11.4 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS

11.4.1 DESIGN BASES

The process, efficient and area radiation monitoring systems are designed to warn of any radiation health violation which might develop, give early warnings of a malfunction which might lead to an unsafe health condition or unit damages and provide continual indication and recordings of radiation levels for normal operation, for anticipated operational occurrences and for a reasonable range of accident conditions. The system operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff to provide timely, adequate information for continued safe operation and for assurance that personnel exposure does not exceed 10CFR20 guidelines.

Sampling systems, detectors, power supplies, ratemeters and recorders are designed to operate within the environmental conditions for the area in which they are located.

11.4.2 CONTINUOUS MONITORING

Table 11.4.1 lists the process radiation monitors.

11.4.2.1 Process Radiation Monitor

The primary purpose of the process radiation monitor is to alert plant operators to an increase in coolant radioactivity as quickly as possible. Such an increase in radioactivity would usually be caused by crud released in the reactor coolant system or chemical and volume control system letdown line. However, an increase in specific fission product nuclide activity along with an increase in gross gamma activity would be indicative of failed fuel cladding.

The monitor is located in the chemical and volume control system letdown line in parallel with the purification filter but upstream of the ion exchangers. This location was selected because a continuous sample at relatively low temperature and pressure can be conveniently obtained and the sample effluent can be returned to the purification system without difficulty. This location also provides an optimum compromise between a minimum sample lag time and the required delay time for sample background radioactivity decay. The time lag from the reactor coolant system to the monitor is sifficient for all operating conditions to permit N^{16} to decay to a low level that will not interfere with monitor readings.

Gross gamma activity concentration and the activity concentration of a specific nuclide are monitored simultaneously. Iodine-135 is the specific nuclide chosen for activity monitoring for several reasons.

First, it is a fission product that is found in relative abundance in the reactor coolant in the event of fuel cladding failure. It is released from defective fuel with relative ease and does not plate out on the system surfaces. Finally, Iodine-135 is chosen because its 1.28 Mev gamma can be readily monitored using discrimination techniques, and its 6.7 hour half-life is long relative to sample lag time but short enough to provide indication of current fission product escape from the fuel. Some fission products that would contribute to gross gamma level plate out on the walls of the piping but since the gross gamma activity after minimum lag time is dominated by gaseous fission product activity, the gross gamma reading will be representative.

The reactor coolant specific activities as measured by the process radiation monitor are listed in Table 11.1-1 at a coolant temperature of 577F. The normal operating temperature of the process radiation monitor is 120F. Therefore, the data in Table 11.1-1 is increased by the density ratio of 1.39 to obtain the acting seen by the monitor.

The monitor consists of shielding, sampler, detector holder, gamma scintillation detector and preamplifier, logarithmic ratemeter, and single-channel linear ratemeter/analyser. The gamma scintillation detector is a sodium iodide crystal, 1-1/2" diameter by 1" long, with photomultiplier tube and integral preamplifier which monitors gamma radiation in the energy range of 100 Kev to 3 Mev. The minimum and maximum detectable coolant activities are 10^{-4} and 10^{2} in μCi/cc of I-135 respectively. A remotely operated Cs-137 check source is used to test the overall operation of the system.

The sampler is of the in-line type with a sample volume 2-1/2 inches in diameter and 1-3/4 inches deep. The sampler and tubing are fabricated from stainless steel designed for 200 psig at 250F. The detector assembly is completely shielded in 4 μ geometry by approximately 4-1/2 inches of lead to reduce the background signal contribution. The shielding assembly also includes a removable collimating plug that must be placed between the detector and sample cell to increase the operating range of the monitor above 10^{-1} μCi/cc of I-135.

After amplification the detector output signal is fed to a linear ratemeter/analyzer and a logarithmic ratemeter located in the control room. The linear ratemeter/analyzer can monitor a specific fission product gamma activity (I^{135}) with a total range of 0 to 10^{6} cpm. The logarithmic ratemeter measures gross gamma activity with a range of 10 to 10^{6} cpm. Alarms are provided on both the linear ratemeter/analyzer and the logarithmic ratemeter with adjustable set points over the complete range.

The process radiation monitor is a trend monitor and its primary purpose is to indicate the possibility of the fuel clad filaure. Alarm set points are normally set at a value slightly above the current reading. It is expected that gross activity and perhaps Iodine-135 activity will periodically increase above the alarm set points due to normal plant transients. Consequently the alarm will periodically activate and the operator must determine the cause of the alarm. If an alarm is received and the Iodine-135 activity has increased and remains significantly above the prior steady state level, additional fuel failure can be assumed to have occurred. However, if an alarm is received due to high Iodine-135 activity concurrent with an increase in gross activity and a plant load increase, curd burst release can be suspected. In time, the coolant activity should return to the prior, lower, steady state value. If an alarm is received, the set points are immediately raised above the higher levels and the alarm circuits are reset. This assures that any successive increases will be brought to the operators attention. If coolant activity decreases, the set points are also lowered.

11.4.2.2 Gaseous Discharge Monitor

The primary purpose of the gaseous waste discharge monitor is to continuously monitor and record all gaseous radioactivity released from the gas decay tanks and to prevent radioactivity in excess of applicable limits from being released to the environment.

This monitor is located in the waste management system in the gaseous discharge line downstream of the gas decay tanks but upstream of the plant stack. Therefore monitoring is accomplished before the gases are diluted in the plant stack. The concentrations of radioactivity expected to be released from the gas decay tanks are given in Table 11.3-1 under position number 10.

The monitor measures gross gamma radioactivity. Even though some of the partially volatile fission products (iodines) will adhere to the walls of the piping and tanks this will not cause false low readings since the activity actually being released is the only concern. Shielding is provided with this monitor to minimize the effects of background radiation.

The monitor consists of an in-line sample detector assembly, shielding, check source, and logarithmic rate meter and recorder. Flow through the sampler is 1 to 10 scfm and all parts in contact with the process gas are stainless steel.

The detector is a thin-wall Geiger-Muller tube which is shielded against background gamma radiation for a minimum sensitivity of 2×10^{-5} µCi/cc of Xe^{133}. The total detector range is 10^{-5} to 10^{-3} µCi/cc based on discharged activity of unknown identity. A collimating plug is used in conjunction with the detector and is used for range of 10^{-1} to 10^{3} µCi/cc. Without the collimating plug detector range is 10^{-5} to 1.0 µCi/cc. The logarithmic rate meter receives the output signal from the detector. The range of rate meter is 10 to 10^{6} counts per minute.

There are two separate calibration check sources for the monitor. A remotely operated radiation check source can be exposed to the detector to calibrate the activity range of the detector. An internally generated count rate signal is used to calibrate the logarithmic rate meter.

The two pen horizontal strip chart recorder receives an electrical signal from the logarithmic rate meter and continuously records the gaseous radioactivity that is released from the plant. The gas discharge flow rate is measured by a recorder in the control room.

The monitor has an alarm and trip system that is operated by the logarithmic rate meter. The set point of the alarm and trip is adjustable over the full range of the rate meter. Prior to the release of gaseous activity, a sample is taken for radioactive analysis as described in Section 11.4.3. Both gross and isotopic radioactivity analyses are performed to verify that the activity in the tank can be released to the environment without exceeding the applicable limits. The alarm set point of the monitor is set slightly greater than the level determined by the gross gamma radioactivity analysis but below the applicable release limits.

If the activity should increase above the set point, an alarm is activated in the control room and valve V-6565 on Figure 11.3-1 is tripped closed stopping discharge flow. The operator must then determine the cause of the alarm and trip. Another sample would be taken for radioactivity analysis to determine if the alarm and trip were due to high activity or instrument mis-calibration or malfunction. If high activity is the source of alarm, discharging from that particular gas decay tank must either be justified at a higher activity level or the contents must be allowed to decay for an additional time period. However, if radioactivity analysis is identical or slightly less than that found prior to initiating the gaseous discharge, the operator must recalibrate the detector and logarithmic rate meter. The logarithmic rate meter is calibrated by activating the internal count rate signal which is an internal component of the rate meter. If necessary, adjustments are made to the rate meter to coincide with the count rate signal. The rate meter is then turned to the detector reading and the standard check source is positioned in front of the detector by activating the pushbutton on the rate meter control panel. Adjustments are made as required. If it has been established that the alarm was caused by instrument malfunction, discharging can continue only after the malfunction has been rectified.

11.4.2.3 Liquid Waste Discharge Monitor

The primary purpose of the liquid waste discharge monitor is to continuously monitor and record the radioactivity (gross gamma) that is being discharged in the liquid waste being released to the circulating water canal. The monitor will also terminate the liquid discharge from the plant if the radioactivity being released exceeds the monitor set point which is set below the applicable activity release limits.

The monitor is located in the Waste Management System in the discharge line to the circulating water canal. Monitoring before dilution with circulating water allows greater accuracy of measurement.

The monitor for this service measures gross gamma radioactivity. The expected concentrations of each nuclide for 0.1 percent and 1.0 percent failed fuel conditions are given in Tables 11.2-6 and 11.2-7, position 21, for boron recovery system releases and Tables 11.2-11 and 11.2-12, position 15, for liquid waste system releases.

Since all the liquid discharge flow passes through the monitor, representative sampling is assured. If excessive radioactivity should be deposited in the sampler, the sampler may be removed for decontamination. Shielding is provided to reduce the effect of background radiation.

The monitor consists of an in-line sampler, detector assembly, shielding, check source, logarithmic ratemeter and recorder. Flow through the sampler is 10 to 50 gpm and all wetted parts are stainless steel.

TABLE 11.4-1

PROCESS RADIATION MONITORS

Channel	Monitor	Type Detector	Minimum Sensitivity	Isotope	Range	Set Point	Maximum Area Radiation Level at Detector	Remarks
R-202	Chemical and Volume Control System Process Radiation Monitor	Gamma Scintillation crystal with photomultiplier tube and integral preamplifier	10^{-4}μCi/cc	I-135	10^{-4} to 10^{2}μCi/cc	Full Range	5 mr/hr gamma Co-60	The set point can be adjusted as operations dictate.
R-627	Waste Management System Liquid Effluent	Gamma Scintillation crystal with photomultiplier tube and integral preamplifier	5 x 10^{-6}μCi/cc	Cs-137	10^{-6} to 10^{-1}μ Ci/cc	Full Range	0.1 mr/hr gamma Co-60	The monitor is adjusted to a set point following an isotopic analysis of the liquid waste to be discharged. The monitor will verify the analysis by alarming should the discharge activity exceed the predetermined value.
R-628	Waste Management System Gas Effluent	Thin wall Geiger-Muller tube	2 x 10^{-5}μCi/cc	Xe-133	10^{-5} to 10^{+3}μCi/cc	Full Range	0.1 mr/hr gamma Co-60	The monitor is adjusted to a set point following an isotopic analysis of the gas to be discharged. The monitor will verify the analysis by alarming should the discharge activity exceed the predetermined value.
	Steam Generator Blowdown	Gamma Scintillation Crystal with Photomultiplier Tube and Integral Preamplifier	1 x 10^{-6}μCi/cc	Cs-137	10^{-6} to 10^{-1}μCi/cc	Full Range	0.1 mr/hr gamma Co-60	(Remarks by EBASCO)

The monitor detector is a gamma scintillation crystal with photo-multiplier tube and integral preamplifier shielded against back-ground gamma radiation for a minimum detector sensitivity of 5 x $10^{-6}\mu Ci/cc$ of Cs-137. The operating range is 10^{-6} to $10^{-1}\mu Ci/cc$ based on discharged activity of unknown identity.

The logarithmic ratemeter receives the output signal from the detector and has a range of 10 to 10^{6} cpm. Ratemeter sensitivity is adjustable over the range of preamplifier output. Gammas with energies in the range of 100 Kev to 3 Mev can be detected.

The monitor contains two separate calibration check sources, remotely operated radiation check source for calibrating the detector and an internally generated count rate signal for calibrating the logarithmic rate meter. A two pen strip chart recorder receives an electrical signal from the logarithmic rate meter and records the liquid radioactivity that is released from the plant. The liquid discharge flow rate is measured and is recorded in the control room.

If the activity reaches the set point, an alarm is activated in the control room and valves FCV-6627X and Y shown on Figure 11.2-2 are tripped closed stopping discharge flow. The operator must then determine the cause of the alarm and trip. Operator actions would be the same as those described for the gaseous discharge monitor.

11.4.2.4 Steam Generator Blowdown Monitor

The primary purpose of the steam generator blowdown radioactivity monitors is to continuously monitor and record the gross gamma activity discharged during blowdown of the steam generators.

One monitor is located in each blowdown sample line upstream of the blowdown line isolation valve.

The concentrations of the nuclides present in the stream generator blowdown line are listed under WMS, location 1, of Tables 11.2-11 and 11.2-12 for normal and anticipated operational occurrences, respectively.

One blow down monitoring system is provided for each steam generator. Each consists of a sampler, gamma scintillation detector assembly with an integral photomultiplier tube and preamplifier, shielding, check source, logarithmic rate meter. One recorder serves both monitors.

The detector is a gamma scintillation crystal with photomultiplier tube and integral preamplifier shielded against background radiation for a minimum detector sensivity of 1.0 x 10^{-6} $\mu Ci/cc$ of Cs-137. The operating range is 10^{-6} to 10^{-1} $\mu Ci/cc$ based on discharged activity of unknown identity.

The logarithmic ratemeter receives the output signal from the detector and has a range of 10 to 10^6 cpm. The ratemeter sensitivity is adjustable over the range of preamplifier output. Gammas with energies in the range of 100 Kev to 3 Mev can be detected.

The monitor contains two separate calibration check sources, remotely operated radiation check source for calibrating the detector, and internally generated count rate signal for calibrating the logarithmic ratemeter.

A two pen strip chart recorder in the control room receives electrical signals from the two logarithmic ratemeters and records the radioactivity in the blowdown from each steam generator.

Each ratemeter activates a high radiation alarm in the control room and provides a signal to trip closed the blowdown valves, FCV-27, 28, 29, and 30, on high radiation. The high radioactivity alarm set point is adjustable over the full range of the ratemeter.

Since blowdown water normally contains no radioactivity and very low chemical concentration, it is normally discharged directly to the circulation water canal. The blowdown radioactivity monitor alarm/trip set point is selected to assure that applicable discharge limits are not exceeded.

The monitor has an alarm and trip system controlled by the logarithmic ratemeter. The high radioactivity set point is adjustable over the full range of the ratemeter. The set point is determined as a result of sample analysis prior to the discharge of any fluid from the blowdown tank. The analysis must verify that the concentration of radioactivity to be discharged is less than applicable release limits. The alarm and trip set point of the logarithmic ratemeter is set slightly above the activity determined from the gross radioactivity analysis but below the applicable release limits.

The detector assembly and the logarithmic ratemeter may need to be recalibrated. The ratemeter is calibrated first by activating the count rate signal which is an integral part of the ratemeter. If necessary, adjustments are made to the ratemeter reading to coincide with the count rate signal. The ratemeter is then switched to the detector and the standard check source is exposed to the detector by activating the pushbutton on the ratemeter control panel. Adjustments are then made as required.

11.4.2.5 Condenser Air Ejector Monitor

Activity levels are recorded in the control room and alarms annunciated when the activity level exceeds predetermined limits. The channel consists of an off-line effluent sampler, a beta scintillation detector, a Cs-137 check source, a logarithmic ratemeter and power supply. The detector assembly has an integral preamplifier with a dynamic range of 25 and a voltage gain of 50. This dynamic

range allows resolution of pulses from 80 Kev to 2 Mev. The sampler is shielded with 3" of lead. The detector has a plastic scintillator, 2" diameter and approximately 10 mil thick, with a 10 stage photomultiplier. The operation limit for pressure is 50 psig and temperature limit 125 F. The ratemeter covers a range of $10-10^6$ cpm with a meter accuracy of ± 2 percent full scale. The channel activity level is recorded in the control room.

The ratemeter has lights to indicate the local indicator failure to indicate a high radiation level warning and to indicate a high-high radiation alarm. A positive displacement, direct drive, dry vane 4 cfm pumping system is provided with this channel.

11.4.2.6 Component Cooling Water Monitor

The component cooling water monitors are designed to provide an indication to operation personnel whenever the activity in the component cooling system reaches or exceeds a preestablished level above the normal radiation level.

Each monitor consists of a shielded sampler, detector holder, gamma scintillation detector, a Cs-137 check source, preamplifier and logarithmic ratemeter. The gamma scintillation detector utilizes a 1-1/2" diameter, 1" thick sodium iodide crystal with a 10 stage photomultiplier tube and integral preamplifier. The preamplifier has a dynamic range which allows resolution of pulses from ranges of 80 Kev to 2 Mev. The minimum and maximum detectable coolant activities to Cs-137 in 2 mr/hr background of 1.25 Mev gamma radiation incident on the detector shield is 2.8×10^{-6} µCi/cc to 1.6×10^{-2} µCi/cc. The sampler is of the off-line type with a sample volume of 191 in ^{3}. The sampler is provided with 3" of lead shielding.

The detectors are located downstream of each of the component cooling system heat exchangers so that any possible release of radioactivity via a tube leak is monitored. The ratemeter associated with this channel is similar to the one described in 11.4.2.5. This channel is annunciated both locally and in the control room whenever an abnormal level of activity is detected. The effected heat exchanger may then be isolated and removed from service by closing isolation valves.

The activity levels are recorded in control room. The alarm set points are set at a slightly higher radiation level than normally would be expected.

11.4.3 SAMPLING

Periodic sampling is required for a performance check on some of the process equipment, and to alert the operator to any abnormal condition that may be developing. The samples required are local liquid samples and gaseous samples taken directly by the gas analyzer. The location for these samples are as follows:

a) <u>Local Liquid Samples</u>

Flush tank influent and effluent

Preconcentrator filter influent and effluent

Preconcentrator ion exchanger influent and effluent

Boric acid concentrator distillate and bottoms

Boric acid condensate ion exchanger influent and effluent

Boric acid condensate tank recirculation

Boric acid holding tank recirculation

Equipment drain tank recirculation

Chemical drain tank recirculation

Laundry drain tank recirculation

Waste filter influent and effluent

Waste condensate tank recirculation

Waste ion exchanger discharge

Waste concentrator distillate and bottoms

Liquid effluent discharge to the circulating water canal

b) <u>Gas Analyzer Samples</u>

Volume control tank

Flash tank

Holdup tank

Spent resin tank

Containment vent header

Gas surge tank

Gas decay tank

Additional samples are also taken at various strategic locations in the plant but these sample stations are connected directly to the sampling system and are discussed in Section 9.3.2.

The basis for selecting the above locations for sampling stations is to check the performance of process equipment thus maintaining

the required chemistry throughout the operating plant. For liquid samples the expected composition of the nuclides taken at a specific point should be consistent with the specific activities and decontamination factors as listed in Section 11.2.

For all the liquid samples the analysis performed will determine isotopic concentrations. The sample can also be used for a gross elemental analysis. The gaseous samples, however, are analyzed for gross concentration, as well as isotopic concentrations. Gas samples to the gas analyzer are analyzed for gross oxygen and hydrogen concentrations and alarms warn the operator if a potential explosive mixture exists at the sample point. A connection of the discharge side of the gas analyzer is available for taking a grab sample for isotopic analysis.

The frequency of taking samples from the various locations along with the analysis to be performed is listed in Table 11.4-2. The frequency of sampling for the gaseous content is used as a preliminary guide and may be increased or decreased as operating personnel become more familiar with the plant.

TABLE 11.4-2

SAMPLE LOCATIONS AND ANALYSIS

Sample Location	Sample Analysis	Frequency
Flash Tank		
Influent and Effluent	Dissolved hydrogen and gaseous activity	2/year
Pre-Concentrator Filter		
Influent and Effluent	Suspended solids and suspended solids activity	1/week
Pre-Concentrator Ion Exchanger		
Influent and Effluent	Gross beta-gamma and lithium content	1/week
Boric Acid Concentrator Bottoms and Distillate	Boron content and gross beta-gamma activity	2/batch*
Boric Acid Condensate Ion Exchanger		
Influent and Effluent	Boron content and gross beta-gamma activity	3/year
Boric Acid Condensate Recirculation	Isotopic	Prior to discharge content of tank
Discharge Line to the Circulating Water Discharge	Gross beta-gamma activity	During each boric acid tank discharge operation
Boric Acid Holding Tank	Boron content and gross beta-gamma	Each time tank is emptied
Equipment Drain Tank	Gross beta-gamma and boron content	Each time contents of tank are processed or discharged to circulating water canal
Chemical Drain Tank	Gross beta-gamma	Each time contents of tank are processed or discharged to circulating water canal

* This frequency may be reduced as experience is gained on the units operation.

TABLE 11.4-2 (Cont.)

Sample Location	Sample Analysis	Frequency
Laundry Drain Tank	Gross beta-gamma	Each time contents of tank is discharged to circulating water canal
Waste Filter Influent and Effluent	Suspended solids and suspended solids activity	1/week
Waste Condensate Tank	Gross beta-gamma	Each time contents of tank is discharged to circulating water canal
Waste Ion Exchanger	Boron content and Gross beta-gamma	3/year
Waste Concentrator Distillate and Bottoms	Boron content and gross beta-gamma	5/batch
	GAS SAMPLES	
Volume Control Tank	Hydrogen, oxygen and isotopic analysis	1/week
Flash Tank	Hydrogen, oxygen and isotopic analysis	2/year
Holdup Tank	Hydrogen and oxygen	2/year
Spent Resin Tank	Isotopic analysis	1/year
Containment Vent Header	Isotopic analysis	5/year
Gas Surge Tank	Hydrogen, oxygen and isotopic analysis	1/day
Gas Decay Tank	Hydrogen, oxygen, and isotopic analysis	1/week

When obtaining local liquid samples, it is important to obtain representative samples. All local sample points are taken from vertically run pipe or from the top of horizontal run pipe. The local sample lines are as short as possible and the sampling isolation valves are located as close to the source as possible to limit the amount of purge water required before a representative sample is obtained.

The sample line is connected by plastic tubing to a 5 gallon poly bottle which may be vented via a HEPA-charcoal filters, if necessary. After purging, the sample is collected in a small poly bottle by diverting the sample flow. After filling, the sample bottle is sealed and marked as to sample location and date and time of sampling.

11.4.3.1 Preconcentrator and Waste Filters

The influent and effluent samples are periodically taken and analyzed to ensure that the filters are removing suspended solids. A sample of suspended solids is obtained by filtering the liquid sample through a millipore filter. A count of filtered activity (gross count) is made and an analysis of crud weight is done by visual comparison with a known standard or the sample is dried and weighed. Only the portion of activity in particulate form is of direct interest, but dissolved activity would also normally be measured. Crud weight concentration at the filter inlets is normally 10 to 20 ppb but could be much higher at times. Effluent activity crud concentrations will normally be only a small percentage of influent values. The filters are normally changed on high radio-activity or high differential pressure. They are also replaced if testing reveals low decontamination factor (DF).

The procedures and calculations for determining suspended solids activity and crud weight are as follows:

a) Determination

A sample aliquot is filtered onto a (0.45 micron) membrane filter paper. Depending on the expected concentration, the actual value is either determined by visual comparison with a known standard, or the sample is dried and weighed.

b) Limitations

The accuracy of these methods is dependent on the actual concentration involved, the drying time, and weighing accuracy.

c) Procedure

1) A specific procedure is used for the determination of suspended solids by actually weighing the sample. If the sample to be analyzed contains radioactive substances the appropriate precautions are observed.

The weight of particulate matter is adjusted for the difference between final and initial weight of the test filter as "weight of particulate matter" and recorded.

2) An alternate procedure is used for the determination of suspended solids by color comparison.

d) Calculations

1) S/S (ppm) = $\frac{W}{V}$ x 1000

where: S/S = Suspended Solids

W = Weight of residue on filter paper

V = Liters of sample

2) The concentration is obtained directly from the membrane filter comparison chart in ASTM D1888.

e) Reference
ASTM, D1888

11.4.3.2 Flash Tank

The influent and effluent samples are taken to verify the performance capability of the flash tank. Periodically, gas samples are automatically taken from the holdup tanks which give indication of flash tank performance during normal operation. An increase in hydrogen content in the holdup tank gas space indicates malfunction of the flash tank. On increase in hydrogen content the flash tank controls would be checked and maintenance performed. If it is necessary to continue processing letdown waste, the flash tank can be bypassed by diverting the influent directly to the holdup tanks. The contents of the holdup tanks can be recycled later through the flash tank as operational conditions permit. The hydrogen concentration in the influent is expected to be approximately 10 to 35 cc (H_2) /kg (H_2O) and the effluent is expected to be equal to or less than 5 cc (H_2) /kg (H_2O). A typical procedure for testing for dissolved hydrogen is described in ASTM E-260-65T.

11.4.3.3 Preconcentrator Ion Exchanger

Influent and effluent samples are taken to determine the ion exchanger decontamination factor and to determine performance effectiveness. The decontamination factor is determined by measuring the gross beta-gamma activity of coolant entering the ion exchanger and comparing it to the gross beta-gamma activity of coolant leaving the ion exchanger. Influent and effluent concentrations by nuclide are identified in Table 11.2-3. Measurements for lithium content are also made. If lithium breakthrough should occur, lithium will build up in the boric acid concentrate. In such a case the ion exchanger resin will be replaced on high lithium or high soluble activity breakthrough. The procedures used for lithium and decontamination tests are as follows:

Determination of Lithium in Water

a) Determination

Lithium standards and the sample are aspirated into a flame through which a beam of light of specific wavelength is passed. The amount of absorption that occurs by the lithium atoms in the flame is related to the concentration of lithium in the solution. By use of a curve of concentration versus absorbance based on known standards the concentration of the unknown sample can be determined.

b) Limitations

In a water matrix of low dissolved solids (<1 percent) no interferences are known. In higher concentrations of dissolved solids the rate of aspiration will decrease as the percent of dissolved solids increases and low results will be obtained. To correct for high dissolved solids, standards of the approximate salt concentration of the sample may be prepared and run with the sample or the method of addition may be analyzed.

The sensitivity for lithium is 0.03 ppm for 1 percent absorption.

Determination of gross beta-gamma activity in water.

a) Determination

The gross beta-gamma activity of the sample is measured by evaporating a two milliliter aliquot in a metal planchet and beginning a 5 minutes count 1 hour after sample was obtained.

b) Limitations

Some volatile activities are lost through the heating-evaporation cycle. Also since this sample is counted in a G-M counter, a high counting rate sample may suffer statistical counting losses due to the long dead time inherent in G-M counter.

c) Calculations

1) For results in dpm/cc =

$$\frac{\text{Gross counts in cpm} - \text{Background Count in cpm}}{\text{Efficiency of detector in } \frac{\text{cpm}}{\text{dpm}} \times \text{Sample Volume in cc}}$$

2) For results in μCi/cc =

$$\frac{\text{dpm}}{\text{cc}\left(2.22\times10^{6} \frac{\text{dpm}}{\mu\text{Ci}}\right)} =$$

$$\frac{\text{Gross counts in cpm} - \text{background count in cpm}}{\left(\text{Efficiency of detector in } \frac{\text{cpm}}{\text{dpm}}\right) \text{sample volume in cc} \left(2.22\times10^{6} \frac{\text{dpm}}{\mu\text{Ci}}\right)}$$

11.4.3.4 Boric Acid Concentrator

The boric acid concentrator bottoms and distillate samples are obtained to determine if the unit is functioning as designed. The bottoms sample verifies that the boric acid is at the desired concentration required for transfer to the boric acid makeup tanks in the chemical and volume control system. This analysis will assure that the concentrator controls are operating adequately and that the automatic operation of the unit is satisfactory. During each batch that is processed from the holdup tanks it is anticipated that five bottoms samples will be analyzed/however, this frequency may be reduced as experience is gained on the unit's operation. The distillate is sampled on a less frequent schedule to verify that the boron carryover is minimal and to estimate ion exchanger exhaustion rate. Wide variations in operation are not expected.

By sampling both the bottoms and distillate for gross beta-gamma activity the concentrator DF can be verified. The same approach is used for the preconcentrator ion exchanger.

The procedure and calculations to be accomplished for boron measurement are as follows:

a) Determination

Boric acid is a weak acid that cannot be directly titrated accurately with standard alkali, but can be transformed into a relatively strong acid by complexing with a polyhydroxy alchol, such as mannitol. The resultant acid complex is titrated with sodium hydroxide to an equivalence point of pH 8.5.

b) Calculations

1) Standardization $$N_S = \frac{N_B \times V_B}{V_S}$$

where: N_S = Normality of standard sodium hydroxide solution

N_B = Normality of standard boron solution

V_S = Volume of standard sodium hydroxide solution

V_B = Volume of standard boron solution

2) Samples

$$\text{ppm B} = \frac{V \times N \times 10.811 \times 10^3}{\text{ml sample}}$$

where: V = Volume of standard sodium hydroxide used

N = Normality of standard sodium hydroxide solution

3) Accuracy of Method

The data given below illustrates the accuracy of the method for determining boron concentration.

Concentration (ppm boron)	Standard Deviation (± ppm boron)
50	1.38
100	1.64
200	1.05
500	0.55
1000	2.17
2000	3.93

11.4.3.5 Boric Acid Condensate Ion Exchanger

Influent and effluent samples are taken to determine the effectiveness for boron removal and to verify the performance capability. Liquid samples are taken from the boric acid condensate tanks prior to discharge to the circulating water discharge, which would indicate functionability of the boric acid condensate ion exchangers. An increase in boron content or activity would indicate malfunction of the ion exchanger. The ion exchanger is designed for long operation periods and any carryover would be extremely infrequent. If high boron or high activity breakthrough should occur the ion exchanger resin will be replaced. The contents of the tank would then be recycled through the recharged ion exchanger prior to discharge, if necessary. The tests for boron content are the same as defined in Section 11.4.3.4 and the test for ion exchanger decontamination factor are indicated in Section 11.4.3.3.

11.4.3.6 Boric Acid Condensate Tank

The contents of the boric acid condensate tank are recirculated, a representative sample of the tank contents is obtained, and a determination of the isotopic inventory is made. A sample is obtained by recycling the contents of the tank until a homogeneous mixture is attained and then a sample is drawn off the recirculation line. If the analysis indicates that release can be made within permissible limits, the quantity of activity to be released is recorded on the basis of liquid volume in the tank and its activity concentration. If release cannot be made within permissible limits, the waste is recycled for further processing. The concentrations of various nuclides are indicated in Table 11.2-3. The method used for analysis is gamma spectrometry in accordance with ASTM D2459-69.

11.4.3.7 Circulating Water Discharge

This sample point is located downstream of the process radiation monitor. The purpose of obtaining the sample here is to provide verification that the process radiation monitor is functioning properly. A gross-beta analysis similar to the procedures utilized

for determination of ion exchanger decontamination factor can be followed. This measurement also serves as a redundant verification of the isotopic analysis made in the boric acid condensate tanks and the gross-beta analysis performed by the process radiation monitor. The concentrations by nuclide would be the same as those in the boric acid condensate tanks.

11.4.3.8 Boric Acid Holding Tank

A sample is obtained by recirculating the contents of the tank to achieve a homogeneous mixture prior to drawing off a sample in the recirculation line. The analytical technique to be employed will be identical to that for the boric acid concentrator bottoms determination and is discussed in Section 11.4.3.4. When concentration of boron matches that in the boric acid makeup tank the liquid can be transferred to the makeup tank, or if capacity is not available and boric acid is contaminated it can be pumped to the drumming station for disposal. This station can also be used for taking a sample for activity analysis.

11.4.3.9 Equipment Drain Tank

A sample is obtained by recirculating the contents of the tank to achieve mixing prior to drawing off a sample. The sample is to be analyzed for gross beta and boron content and analytical techniques are discussed in Section 11.4.3.3. The analytical results will determine if the liquid can be either processed by the waste concentrator or boric acid concentrator, or if activity is below the release limits of 10CFR20 it can be directly discharged via the circulating water discharge.

11.4.3.10 Chemical Drain Tank

The contents of the chemical drain tank are recirculated to achieve mixing prior to drawing off a sample.

The only analysis that is required is a gross beta-gamma count and the technique to be employed is discussed in Section 11.4.3.3. for gross beta-gamma activity in water. The gross activity count will determine if liquid can be discharged directly or if it should be processed in the waste concentrator.

11.4.3.11 Laundry Drain Tank

The sampling and analytical techniques to be used for this tank is identical to that for the chemical drain tank. However the liquid from this tank is discharged directly to the circulating water canal after passing through a filter.

11.4.3.12 Waste Concentrator

The bottoms and distillate of the waste concentrator are sampled for boron content and gross beta-gamma count to check the per-

formance of the unit. The analytical techniques to be employed are given in Section 11.4.3.4 for boron analyses and Section 11.4.3.3. for the gross beta-gamma count.

11.4.3.13 Waste Ion Exchanger

The discharge of the waste ion exchanger is sampled and analyzed for boron and gross beta-gamma count. The sample is drawn off at the provided connection and the analytical technique to be employed for boron is given in Section 11.4.3.4 and Section 11.4.3.3 for gross beta-gamma.

The sample analyzed at this point along with the sample for the distillate on the waste concentrator will determine the performance of the ion exchanger.

11.4.3.14 Waste Condensate Tank

The contents of the waste condensate tank are recirculated to achieve mixing prior to sampling. The samples are analyzed for a gross beta-gamma count prior to discharging the contents to the circulating water canal. However, if the count exceeds the acceptable limit the liquid is pumped to the waste concentrator or waste ion exchanger for additional processing. The analytical technique employed is discussed in Section 11.4.3.3.

11.4.3.15 Gas Analyzer

A gas analyzer is used to determine the concentration of hydrogen and oxygen in the gas to be sampled from the various sources as given in Section 11.4.3. The gas analyzer has an automatic timing device that periodically samples the gas from the tanks and vents by opening and closing appropriate solenoid valves. After the system is lined up to a particular sampling station and flow has started, entrained water is removed from the sample by flowing through a water separator trap. The excess water is drained from the trap and sent to the equipment drain tank, while the gas is pumped to a refrigeration unit where saturated water is removed. From the refrigeration unit the gas flows to the oxygen and hydrogen sensing devices, and then another pump is used to discharge the gas to the gas surge header. The hydrogen and oxygen sensing elements are installed in parallel with a flow meter installed upstream of each sensor so that the operator can manually balance the flow to each sensor. The readout signal from the sensor is sent to a recorder where the oxygen and hydrogen concentrations are recorded. If an explosive mixture is being approached the gas analyzer alarm will annunciate in the control room to alert the operator that immediate action is required. The explosive mixture can be avoided by either purging or diluting with nitrogen.

Calibration gases for the gas analyzer are required. A zero gas (prepurified nitrogen) is required for the hydrogen and oxygen sensors and a span gas (45 percent hydrogen in nitrogen)

is required for the hydrogen sensor. These gases are required to periodically calibrate the instrument. Oil-free air is required to purge the sensors prior to and after calibrating.

Sample connections are available in the system downstream of the pump that returns the gas to the gas surge header where a sample container (1 liter) is attached for taking a sample for isotopic analysis. An option is available to the operator for either bypassing the sensors to take the radioactive sample or the radioactive sample can be the same gas that has passed through the sensors for the hydrogen and oxygen determination.

11.4.4 CALIBRATION AND MAINTENANCE

All radiation monitoring systems are calibrated periodically on a schedule shown in Section 16.4 of the Technical Specifications. Any maintenance required will be conducted in accordance with the instructions provided by the instrument manufacturer.

11.5 SOLID WASTE SYSTEM

11.5.1 DESIGN BASES

The solid waste system is designed to collect and prepare solid radioactive wastes and certain radioactive liquid wastes for offsite shipment. The waste forms and quantities to be processed are given in Table 11.5-1. The waste activities to be processed are given in Tables 11.5-2, 11.5-3 and 11.5-4.

TABLE 11.5-1

INPUTS TO SOLID WASTE SYSTEM PER CORE CYCLE
BASED ON CONTINUOUS OPERATION WITH 1 PERCENT FAILED FUEL

Liquids	Quantity	Operation
Concentrator Bottoms, gal/yr	8,000	miscellaneous waste processing
Concentrator Bottoms, gal/yr	3,700	steam generator blowdown processing estimated normal operation
Concentrator Bottoms, gal/yr	22,000	steam generator blowdown processing anticipated operational occurrence
Solids		
IX Resins, dewatered, ft^3	256	sluice ion exchangers once per year
Filter Elements	14 per year	change each filter cartridge twice per year
Miscellaneous Compressible Solid Waste, ft^3	1,000 to 2,500	compact
Miscellaneous Noncompressible Waste, ft^3	500 to 1,000	place in suitable containers

11.5.2 SYSTEM DESCRIPTION

The solid waste system is shown on Figure 11.2-4. Concentrate from the waste concentrator is pumped directly from the concentrator to the drum fill station. As shown on Figure 11.2-3, the boric acid concentrator bottoms from either concentrator can be bypassed around the boric acid holding tank directly to the drumming station while the other boric acid concentrator bottoms are being processed through the boric acid holding tank. This bypass can be used to return re-concentrated boric acid directly to the boric acid makeup tanks in the chemical and volume control system (CVCS) in the event of problems with the holding tank, but the bypass will more likely be used when a boric acid concentrator is used for miscellaneous waste processing thereby avoiding contamination of the holding tank.

At the drumming station, the concentrate from either the waste concentrator or one of the boric acid concentrators enters a shipping drum as described in Section 11.5.5. The liquid waste quantity from the concentrators is based on processing the 156,000 gallons of miscellaneous liquid waste per Table 11.2-1. This waste is concentrated by a minimum factor of approximately 20 resulting in approximately 8,000 gallons or 1,000 cubic feet of concentrate to be solidified per year. When mixed with cement and absorbents in the shipping drums, the shipped volume is approximately 2,400 cubic feet requiring approximately 330-55-gallon drums.

Because most of the 156,000 gallons of water that enters the liquid waste system is a result of decontamination operations, and floor washdowns, the radioactivity is diluted significantly by uncontaminated wash water. Consequently, the aerated waste influent activity is approximately 1 percent of the reactor coolant activity.

No credit is taken for activity removal by the waste filter in evaluating concentrator bottoms activities. Except for tritium, the concentrate activities are a factor of 20 above influent activities. Tritium concentration is one hundredth of Table 11.1-1. Steam generator blowdown due to primary to secondary leakage is processed similarily to liquid wastes. Approximately 3,700 gallons per year of waste concentrator bottoms are generated as a result of a continuous 20 gpd steam generator leak for normal operation (0.1 percent failed fuel). The shipped volume of solidified waste is approximately 1,100 ft^3 requiring 160-55-gallon drums. The anticipated operational occurrence with a steam generator leak of 120 gpd results in approximately 22,000 gal/yr of bottoms and 6,600 ft^3 of solidified waste. Drumming station influent activities from these sources are given on Table 11.5-2. Concentrator bottom activity after day decay period is shown on Table 11.5-3.

Packaging requirements based on these activities in concentrator bottoms quantities are defined in Section 11.5.5.

TABLE 11.5-2

CONCENTRATOR BOTTOMS INPUT ACTIVITY TO SOLID WASTE SYSTEM PER CORE CYCLE

	Normal Operation		Anticipated Operational Occurrences	
Nuclide	Waste Concentration (μCi/cc at 70 F)	Total Curies/year	Waste Concentration (μCi/cc at 70 F)	Total Curies/year
H-3	2.21(-3)	9.86(-2)	1.39(-2)	1.60
Sr-89	1.67(-4)	7.45(-3)	8.68(-3)	.998
Sr-90	1.07(-5)	4.77(-4)	5.5 (-4)	6.33(-2)
Y-90	4.34(-6)	1.94(-4)	3.6 (-4)	4.14(-2)
Sr-91	1.35(-8)	6.02(-7)	9.12(-5)	1.05(-2)
Y-91	3.78(-3)	1.69	1.97(-1)	22.66
Mo-99	9.3 (-3)	.415	7.44(-1)	85.56
Ru-103	1.28(-4)	5.71(-3)	6.70(-2)	7.71
Ru-106	1.02(-5)	4.55(-4)	5.24(-3)	.603
I-129	2.981(-9)	1.33(-7)	1.52(-7)	1.75(-5)
I-131	5.72(-2)	2.55	3.32	382.80
Te-132	1.87(-3)	8.34(-2)	1.38(-1)	15.87
I-133	1.56(-3)	6.96(-2)	5.54(-1)	63.71
Cs-134	4.1 (-3)	.183	2.1 (-1)	24.15
I-135	3.08(-7)	1.37(-5)	2.74(-3)	.315
Cs-136	5.08(-4)	2.27(-2)	2.8 (-2)	3.22
Cs-137	1.12(-2)	.500	6.54(-1)	75.21
Ba-140	1.2 (-4)	5.35(-3)	6.6 (-3)	.759
La-140	1.06(-5)	4.73(-4)	1.29(-3)	.148
Pr-143	1.19(-4)	5.31(-3)	6.52(-3)	.750
Ce-144	1.69(-4)	7.54(-3)	8.72(-3)	1.003
Co-60	2.14(-4)	9.54(-3)	1.1 (-3)	.127
Fe-59	3.36(-6)	1.50(-4)	1.35(-5)	1.55(-3)
Co-58	1.63(-3)	7.27(-2)	8.42(-3)	.968
Mn-54	1.21(-5)	4.50(-4)	5.8 (-5)	6.67(-3)
Cr-51	1.05(-3)	4.68(-2)	5.56(-3)	.639
Zr-95	3.22(-7)	1.44(-5)	1.67(-6)	1.23(-4)
Total		5.78		689

TABLE 11.5-3

CONCENTRATOR BOTTOMS ACTIVITY AFTER 90-DAY DECAY

Nuclide	Normal Operation Curies	Anticipated Operational Occurrence Curies
H-3	9.71(-2)	1.575
Sr-89	2.17(-3)	0.291
Sr-90	4.74(-4)	6.30(-2)
Y-91	0.587	7.87
Ru-103	1.20(-3)	1.62
Ru-106	3.84(-4)	0.508
I-129	1.33(-7)	1.75(-5)
I-131	1.10(-3)	0.165
Cs-134	0.168	22.2
Cs-136	1.87(-4)	2.65(-2)
Cs-137	0.497	74.8
Ba-140	4.09(-5)	5.80(-3)
Pr-143	5.60(-5)	7.92(-3)
Ce-144	6.05(-3)	0.804
Co-60	9.24(-3)	0.123
Fe-59	3.75(-5)	3.88(-4)
Co-58	3.02(-2)	0.402
Mn-54	3.69(-4)	5.46(-3)
Cr-51	4.64(-3)	6.34(-2)
Zr-95	5.51(-6)	4.71(-5)
Total	1.38	110

The ion exchanger resin activity buildup is based on inlet activity concentration in Table 11.1-1, the maximum removal of each nuclide for each process ion exchanger and the expected flow rate through each ion exchanger. The chemical and volume control system purification ion exchanger is assumed to continuously process 40 gpm letdown over the full core cycle after being used in the previous core cycle for lithium removal. The chemical and volume control system purification ion exchanger used for lithium removal is assumed to process 40 gpm letdown over 20 percent of the core cycle which is the expected usage of the unit to control lithium concentration. The chemical and volume control system deborating ion exchanger is assumed to process letdown over 2.4 percent of the core cycle. This is the expected usage of the unit to remove the final 30 ppm of boron in the reactor coolant to compensate for fuel burnup. Each of the two WMS preconcentrator ion exchangers are assumed to process one half of the 843,000 gallons of clean waste per year. The waste was assumed to be letdown reactor coolant which has already passed through a chemical and volume control system ion exchanger. The boric acid condensate and the waste ion exchangers will not retain a significant amount of activity due to the very low concentration of activity in the inlet process fluid. The fuel pool ion exchanger is assumed to remove the portion of reactor coolant system end of life activity which enters the fuel pool water, assuming complete mixing, after being diluted by the refueling water. The activities on these ion exchangers by nuclide after one core cycle of operation with 1 percent failed fuel (anticipated operational occurrence)are given in Table 11.5-4. Normal operation with 0.1 percent failed fuel would result in 0.1 of the activity given for anticipated operational occurrence. Actually the chemical and volume control system purification bed activities represent the curies accumulated in two years of operation with one percent failed fuel. During the first year of operation, the resin is gradually converted from the $H-BO_3$ form to the $Li-BO_3$ form as lithium, produced by the B10 (η, α) Li^7 reaction in the core, is controlled below its upper concentration limit. During this time, monovalent cations such as cesium are removed by this bed from the coolant. By the end of one core cycle ($\sim$ 20 percent usage factor) this bed becomes saturated with lithium. Although the monovalent cations on the bed are retained, the bed loses capability to remove these cations from the coolant. When the bed is run continuously during the second core cycle, the activities of Cs, Mo, and Y decrease due to radioactive decay.

Activities on the various ion exchange resin beds after six months and one year decay are shown on Table 11.5-5. Cesium-137 dominates the activity that must be shipped, but without significant failed fuel, Co-60 would dominate. Assuming eight beds are disposed of per year (one purification, one deborating, two preconcentrator, two boric acid condensate, one waste condensate, and one fuel pool purification) the mixed activity would be about 23,000 Ci or 90 Ci/ft^3. Packaging requirements for this activity concentration and resin quantity are defined in Section 11.5.5.

Filter vessels used in removing particulates from process streams are installed behind labyrinthed concrete cubicles roofed over by a concrete slab. The roof slab is fitted with a removable concrete hatch. Holes in the hatch permit long handled tools to operate valves for draining vessel and to undo head bolt nuts to permit vessel head to swing open in a vertical arc. A monorail system above the cubicle roof allows for removing the hatch and setting it aside. Filters of high activity levels require a shielding device simulating an inverted bell to be positioned over the opened hatch. A winch device and appropriate opening allow the filter cartridge to be withdrawn from the filter vessel and into the cartridge (inverted bell) shield. The cartridge shield is replaced on a transport device and removed to the drumming area where the cartridge will be prepared for disposal.

Low level activity cartridges may be removed without the use of a cartridge shield. Upon removal through the hatch it would be deposited in a shipping container positioned to receive the filter. The container would then be removed to the drumming area for final preparation for disposal.

The activity level on the filter elements is based on a conservatively high inlet concentration, the maximum removal for each nuclide and the expected flow rate through each unit. The purification filters 1A and 1B are each assumed to process 40 gpm in series (1A upstream and 1B downstream of ion exchangers) continuously over the entire core cycle. The fuel pool purification filter is assumed to remove the portion of reactor coolant system end of life activity which enters the fuel pool water, assuming complete mixing, after being diluted by the refueling water. The waste management system waste filter and the laundry filter are not expected to retain a significant amount of activity due to the very low concentration of activity in the inlet process fluid and the low quantity of liquid each unit must process. Each of the two pre-concentrator filters are assumed to process one half of liquid waste processed by the boron recovery system. This waste is assumed to be reactor coolant which has been processed by the chemical and volume control system.

The activity on these filters after one core cycle with source activities equivalent to one percent failed fuel are given in Table 11.5-6. These same filter activities after six months and one year decay are shown in Table 11.5-7. The total activity for a given filter may be divided between a number of cartridges that could be packaged separately. Packaging requirements for filter cartridges based on the activities of Table 11.5-7 are defined in Section 11.5.5.

Miscellaneous compressible solid waste such as contaminated clothing, plastic sheeting, and tape, accumulates as a result of health physics and maintenance activities. Volumes before compression typically range from 1000 to 2500 cubic feet per year at operating plants. Activity typically ranges from 0.5 to 1 curie per year.

Noncompressible solid waste such as tools, contaminated equipment, typically ranges from 500 to 1000 cubic feet per year as packaged. Activity typically ranges from 5 to 15 curies per year.

TABLE 11.5-4

ACTIVITY BUILDUP ON ION EXCHANGE RESINS PER CORE CYCLE WITH 1% FAILED FUEL (CURIES(1))

Nuclide	CVCS (2) Purification	CVCS Deborating	WMS Pre Conc.(each)	FPS Purification
Br-84	0.455	1.1×10^{-2}	9.1×10^{-6}	5.11×10^{-2}
Rb-88	14	0	2.8×10^{-6}	1.66
Rb-89	0.29	0	5.8×10^{-8}	3.44×10^{-2}
Sr-89	110	0	0.22	0.473
Sr-90	24.5	0	4.9×10^{-2}	3.09×10^{-2}
Y-90	0.215	0	1.08×10^{-2}	4.27×10^{-2}
Sr-91	0.6	0	1.2×10^{-5}	5.94×10^{-2}
Y-91	500	0	25	4.24×10^{-2}
Mo-99	450	0	22.5	63.4
Ru-103	65	0	0.13	0.373
Ru-106	16.5	0	3.3×10^{-2}	2.54×10^{-2}
Te-129	0.47	1.11×10^{-2}	9.4×10^{-4}	6.34×10^{-2}
I-129	1.1×10^{-2}	5.24×10^{-6}	2.2×10^{-7}	7.92×10^{-6}
I-131	14,000	160.5	0.28	297
Te-132	430	8.12	0.86	18.7
I-132	45.5	1.03	9.1×10^{-4}	5.24
I-133	2150	52	4.3×10^{-2}	167
Te-134	0.305	7.2×10^{-3}	6.1×10^{-4}	3.92×10^{-2}
I-134	10	0.24	2.0×10^{-4}	1.18
Cs-134	4000	0	200	222
I-135	330	7.95	6.6×10^{-3}	33.5
Cs-136	37	0	1.85	2.46
Cs-137	18.500	0	925	90.3
Cs-138	1.3	0	6.5×10^{-2}	0.786
Ba-140	34	0	6.8×10^{-4}	0.497
La-140	3.95	0	7.9×10^{-3}	0.258
Pr-143	32	0	6.4×10^{-2}	0.321
Ce-144	265	0	0.53	0.382
Fission Product Total	41,000 Ci(2)	230 Ci	1150 Ci (3)	900 Ci
Cr-51	6.0	0.144	1.1×10^{-2}	7.9×10^{-2}
Mn-54	0.300	0	0	0
Co-58	18.9	0.454	3.5×10^{-2}	0.107
Fe-59	5.6×10^{-2}	0	0	0
Co-60	8.30	0.199	1.3×10^{-2}	1.3×10^{-2}
Zr-95	3.49×10^{-3}	0	0	0
Corrosion Product Total	32.6	0.797	5.9×10^{-2}	0.199
Overall Total	41,000	231	1150	900

Notes: (1) Activity levels for normal operation (.1% failed fuel) would be 0.1 of the 1% failed fuel values.
(2) Based on operation for two core cycles. See Text.
(3) One-half of total.

TABLE 11.5-5

ACTIVITY ON ION EXCHANGER RESINS AFTER DECAY (CURIES)

Nuclide	CVCS Purification		CVCS Deborating		WMS Pre Conc.(each)		FPS Purification	
	6 mo	1 yr	6 mo	1 yr	6 mo	1 yr	6 mo	1 yr
Br-84	0	0	0	0	0	0	0	0
Rb-88	0	0	0	0	0	0	0	0
Rb-89	0	0	0	0	0	0	0	0
Sr-89	9.0	0.736	0	0	1.81(-2)*	1.47(-3)	3.88(-2)	3.17(-3)
Sr-90	24.2	24.0	0	0	4.84(-2)	4.78(-2)	3.05(-2)	3.01(-2)
Y-90	0	0	0	0	0	0	0	0
Sr-90	0	0	0	0	0	0	0	0
Y-91	58.5	6.85	0	0	2.93	0.343	4.96(-3)	5.81(-4)
Mo-99	0	0	0	0	0	0	0	0
Ru-103	2.74	1.16(-2)	0	0	5.50(-3)	2.32(-4)	1.57(-2)	6.65(-4)
Ru-106	11.7	8.25	0	0	2.33(-2)	1.65(-2)	1.79(-2)	1.27(-2)
Te-129	0	0	0	0	0	0	0	0
I-129	1.10(-2)	1.10(-2)	5.24(-6)	5.24(-6)	2.27(-7)	2.2(-7)	7.92(-6)	7.92(-6)
I-131	2.08(-3)	0	2.38(-5)	0	0	0	4.41(-5)	0
Te-132	0	0	0	0	0	0	0	0
I-132	0	0	0	0	0	0	0	0
I-133	0	0	0	0	0	0	0	0
Te-134	0	0	0	0	0	0	0	0
I-134	0	0	0	0	0	0	0	0
Cs-134	3380	2850	0	0	169	142.5	188	159
I-135	0	0	0	0	0	0	0	0
Cs-136	2.18(-3)	0	0	0	1.51(-4)	0	1.16(-4)	0
Cs-137	18,280	18,040	0	0	914	904	89.4	88.4
Cs-138	0	0	0	0	0	0	0	0
Ba-140	1.72(-3)	0	0	0	0	0	2.52(-5)	0
La-140	0	0	0	0	0	0	0	0
Pr-143	3.10(-3)	0	0	0	0	0	3.11(-5)	0
Ce-144	170	109	0	0	0.338	0.216	0.245	0.157
Fission Product Total	21,900	21,000	2.91(-5)	5.24(-6)	1065	1050	280	250
Cr-51	6.3(-2)	0	1.52(-3)	0	0	0	0	0
Mn-54	0.20	0.133	0	0	0	0	0	0
Co-58	3.18	0.534	7.65(-3)	1.28(-2)	0	0	1.8(-2)	3.05(-3)
Fe-59	0	0	0	0	0	0	0	0
Co-60	7.77	7.28	0.186	0.174	1.22(-2)	1.14(-2)	1.22(-2)	1.14(-2)
Zr-95	0	0	0	0	0	0	0	0
Corrosion Product Total	11.21	7.95	0.264	0.187	1.22(-2)	1.14(-2)	3.02(-2)	1.45(-2)

Note: * denotes power of 10

TABLE 11.5-6

ACTIVITY BUILDUP ON FILTER CARTRIDGE PER CORE CYCLE (CURIES)

Nuclide	CVCS Purification 1A	CVCS Purification 1B	WMS Pre-Concentrator (each)	WMS Waste	FPS Purification
CO-60	40.0	4.00	7.4(-2)*	**	7.33(-2)
Fe-59	0.317	3.17(-2)	**	**	**
Co-58	106.3	10.63	1.97(-1)	**	6.05(-1)
Mn-54	1.472	0.1472	**	**	**
Cr-51	34.5	3.45	6.3(-2)	**	4.46(-1)
Zr-95	1.98(-2)	**	**	**	**
Total	182	18.2	3.34(-1)	**	1.12

* () power of 10

** less than 10^{-2}

TABLE 11.5-7

ACTIVITY ON FILTER CARTRIDGES AFTER DECAY (CURIES)

Nuclide	1A		1B		WMS-Pre-Concentrator (Each)		WMS Waste		FPS Purification	
	6 mo	1 yr	6 mo	1 yr	6 mo	1 yr	6 mo	1 yr	6 mo	1 yr
Co-60	37.4	35.0	3.74	3.5	6.92(-2)*	6.48(-2)	**	**	6.86(-2)	6.42(-2)
Fe-59	1.90(-2)	**	**	**	**	**	**	**	**	**
CO-58	17.9	3.00	1.79	0.330	3.32(-2)	**	**	**	1.02(-1)	1.71(-2)
Mn-54	0.98	0.655	9.8(-2)	6.55(-2)	**	**	**	**	**	**
Cr-51	0.364	**	3.64(-2)	**	**	**	**	**	**	**
Zr-95	2.82(-2)	**	**	**	**	**	**	**	**	**
Total	56.6	38.7	5.66	3.87	1.024(-1)	6.48(-2)	**	**	1.706(-1)	8.13(-2)

** less than 10^{-2}

* ( ) power of 10

11.5.3 EQUIPMENT DESCRIPTION

A data summary of the solid waste system components is given in Table 11.5-8.

Two drum rollers are used to roll the 55-gallon drums to mix cement and concentrator bottoms. Cement is first added to the drum in a predetermined amount based on the amount of concentrator bottoms to be added to the drum. The concentrator bottoms are then added to the drum in an amount determined by concentrate activity level and shipping limitations. The cover is secured to the drum manually and placed on the horizontal drum roller using a monorail chain hoist. After rolling for approximately 15 to 30 minutes, the drum may be transferred for curing. After approximately 24 hours, the drum cover is removed and absorbent material is added if necessary to absorb any free liquid. After the cover is replaced, the drum is transferred to storage (Section 11.5.6) for decay and ultimate offsite shipment. It is expected that two to four 55-gallon drums could be filled per hour giving a capacity of 30 to 80 gallons of concentrator bottoms per hour based on 20 gallons of concentrate per drum. Drum rollers for mixing radioactive liquid waste and cement have been used at Connecticut Yankee without difficulty.

The spent resin tank can hold the equivalent of approximately eight to ten beds of spent resin from the various plant ion exchangers, and therefore, storage capacity in excess of one year is normally available. Higher activity resin will normally be sluiced to a shielded shipping container, but sluicing to 55 gallon drums is also possible. Approximately 256 cubic feet per year of resin will be sluiced to the shipping containers at flow rates from 3 to 6 ft^3/min. in a slurry containing approximately 2 parts water to 1 part resin. Sluicing resin in the stated manner is typically done at operating nuclear power plants such as Connecticut Yankee without difficulty.

To sluice resin from the spent resin tank, water from the holdup pumps or reactor makeup water enters the tank through a bottom retention screen. The tank fills with water compressing any air in the tank forcing a water and resin slurry up the tank outlet line. The discharge pipe extends upward from near the tank bottom thus minimizing chances of resin binding while allowing efficient removal of nearly all the resin from the tank. A conductance type level sensor allows level to be monitored. The resin and water slurry enters the shipping container and the water is drawn off by the dewatering pump. The dewatering suction line is fitted with a disposable length of pipe attached to a filter element lying on the bottom of the container. The filter element serves as a resin retention element preventing resin or loosened crud from leaving with the dewatering flow. The resin dewatering pump is self priming and air drawn through the suction piping during priming operations is vented through a water trap to the equipment drain tank. Flush water is returned to the holdup tank.

Before sluicing resin from an ion exchanger to the spent resin tank, the tank is vented and partially drained of water by pumping to the holdup tanks. After resin is sluiced in, water is added at least above

the resin level and the vent valve is closed. A record is kept of all resin additions and discharges from the spent resin tank as a check on resin inventory in the tank.

Fresh ion exchange resin is added to the various ion exchangers by connecting the resin addition tank, essentially a funnel, to the resin fill flange and pouring the as-received resin directly into the ion exchanger vessel.

The waste baler employs a hydraulic ram to compact compressible solids. A volume reduction from five to ten is normally possible. The drums are loosely filled and placed into the baler, fastened in position, and the unit is sealed. An exhaust fan serves to exhaust any activity that may be forced airborne by the bursting of plastic bags through a HEPA filter to the ventilation system. Waste material addition and compression operations may be repeated until the drum is essentially filled. Balers of this type have been used for this service in many nuclear installations without problems.

No special equipment is employed for handling miscellaneous noncompressible solid waste.

TABLE 11.5-8

DESIGN DATA FOR SOLID WASTE SYSTEM COMPONENTS

1. Drum Roller

Quantity	2
Type	Horizontal, direct drive
Drum size, gal	55
Capacity, lb	1000
Code	None

2. Baler

Quantity	1
Type	Hydraulic, totally enclosed
Drum size, gal	55
Compacting force, psi	8.3
Exhaust fan capacity, CFM	40
Exhaust filter	Prefilter and HEPA filter
Code	None

3. Spent Resin Tank

Quantity	1
Type	Vertical
Volume, gal	3200
Design pressure, psig	50
Design temperature, F	200
Material	SS
Code	ASME Boiler and Pressure Vessel Code 1968, Section III, Class C

4. Dewatering Pump

Quantity	1
Type	Self-priming centrifugal
Design flow, gpm	100
Design head, ft	95
Design temperature, F	200
Material	SS
Code	None

5. Resin Addition Tank

Quantity	1
Volume, gal	100
Design pressure, psig	Atmospheric
Design temperature, F	N.A.
Material	SS
Code	None

11.5.4 EXPECTED VOLUMES

The concentrator bottoms input is estimated at approximately 11,700 gallons per year as described in Section 11.5.2. To solidify this liquid requires approximately two pounds of cement per pound of liquid. Using a liquid density of 10 pounds per gallon, 117,000 pounds of liquid requires 235,000 pounds of cement powder for a total weight of 352,000 pounds. The density of the cured mixture is approximately 100 pounds per cubic foot resulting in a solidified waste volume of 3520 cubic feet. Approximately 490 55-gallon drums will be required per year to dispose of this waste. The principle radionuclides in the waste are Cs-137 or Co-60 depending on the extent of failed fuel. Curie content of the shipped waste could be as great as indicated in Table 11.5-2.

The ion exchange resins will be shipped dewatered and, therefore, the shipped volume is approximately 256 cubic feet per year with curie content per Table 11.5-5.

The baler compacts the compressible solid waste with volume reduction ranging from 5 to 10 thus producing 100 to 500 cubic feet per year as shipped. Shipped curie content is expected to range from 0.5 curie to 1 curie per year.

Miscellaneous noncompressible solid waste is estimate to range from 500 to 1000 cubic foot per year based on general plant operating experience. Need to dispose of a large component could significantly increase this estimate however. Shipped content ranges from 5 Ci to 15 Ci per year.

11.5.5 PACKAGING

As discussed in previous sections, liquid wastes are solidified by mixing with cement in standard, 55-gallon steel drums. The drums are preloaded with dry cement, transported via a monorail chain hoist system to the fill station, and filled with a predetermined volume of concentrator bottoms. Activity levels in the drums will be controlled such that shipments will not exceed the limits specified in Title 49 of Code of Federal Regulations.

Spent resins are sluiced to a shipping container from the spent resin tank and dewatered via the resin dewatering pump. Activity levels will be controlled such that shipments will not exceed the limits specified in Title 49 of Code of Federal Regulations.

11.5.6 STORAGE FACILITIES

No plans have been made for storage of high level activity waste. Low activity waste is stored temporarily in the drumming station in the reactor auxiliary building with local shielding provided to reduce the dose rates to the operating personnel to below 10CFR20 limits.

11.5.7 SHIPMENT

Solid radwastes will be shipped from the site by truck, which carriers will be licensed contractors operating in accordance with Department of Transportation and other applicable regulations.

Shipping containers or vehicles will not normally be stored on site. If short term storage does occur on site, the allowed locations will be selected such that the security and radiological surveillance procedures for plant operation are met with respect to the activity involved.

11.5.8 INSTRUMENT APPLICATION

Table 11.5-9 lists the parameters used to monitor the waste management system.

TABLE 11-5-9

WASTE MANAGEMENT SYSTEM INSTRUMENTATION APPLICATION

System Parameter & Location	Indication		Alarm[1]		Rec[1]	Control Function	Inst. Range	Normal Operating Range	Inst. Accuracy
	Local	Contr Room	High	Low					
Boric Acid Holding Tank Temperature	*		*[2]	*[2]		Boric Acid Holding Tank Heaters	20-200 F		± 4 F
Gas Decay Tanks Temp.	*						40-250 F		±1.0 F
Gas Surge Tank Temp.	*						40-200 F		±2.0 F
Reactor Drain Tank Pressure		*	*	*			0-25 psig		±0.2 psig
Flash Tank Pressure		*	*	*			0-15 psig		±0.1 psig
Holdup Tank 1A Pressure		*	*	*			0-10 psig		±0.1 psig
Holdup Tanks 1B, 1C, 1D Pressure	*		*	*			0-10 psig	2 psig	±0.1 psig
Spent Resin Tank Press.	*		*[2]				0-60 psig	0-15 psig	±0.5 psig
Gas Decay Tanks Press.	*		*[2]				0-200 psig	0-165 psig	±1.0 psig
Gas Surge Tank Press.	*		*[2]			Waste Gas Compressor	0-30 psig	0-5 psig	±0.3 psig
Boric Acid Holding Pumps Discharge Press.	*						0-100 psig	0-60 psig	±0.5 psig
Reactor Drain Pump Discharge Pressure	*						0-100 psig	20-60 psig	±0.5 psig
Flash Tank Pumps Discharge Pressure	*						0-100 psig	20-60 psig	±0.5

[1] All alarms and recorders are in the control room unless otherwise indicated.

[2] Alarmed on local Waste Management System annunciator panel.

TABLE 11-5-9 (Continued)

WASTE MANAGEMENT SYSTEM INSTRUMENTATION APPLICATION

System Parameter & Location	Indication		Alarm[1]		Rec[1]	Control Function	Inst. Range	Normal Operating Range	Inst. Accuracy
	Local	Contr Room	High	Low					
Holdup Drain Pumps Discharge Pressure	*						0-100 psig	20-60 psig	±0.5
Boric Acid Condensate Pumps	*						0-100 psig	60 psig	±0.5
Equipment Drain Pump Discharge Pressure	*						0-100 psig	60 psig	±0.5
Chemical Drain Pump Discharge Pressure	*						0-100 psig	0-60 psig	±0.5psig
Laundry Drain Pumps Discharge Pressure	*						0-100 psig	0-60 psig	±0.5 "
Waste Condensate Pumps Discharge Pressure	*						0-100 psig	0-60 psig	±0.5 "
Resin Dewatering Pump Discharge Pressure	*						0-100 psig	0-60 psig	±0.5 "
Waste Gas Compressors Discharge Pressure	*						0-200 psig	0-165 psig	±1.0 "
Waste Gas Discharge Pressure	*						0-160 psig	0-15 psig	±0.8 "
Nitrogen Supply Press.	*		*[2]	*[2]			0-600 psig	200-240 "	±3.0 "
Hydrogen Supply Press.	*		*[2]	*[2]			0-600 psig	90-110 "	±3.0 "

[1] All alarms and recorders are in the control room unless otherwise indicated.

[2] Alarmed on local Waste Management System annunciator panel.

TABLE 11-5-9 (Continued)

WASTE MANAGEMENT SYSTEM INSTRUMENTATION APPLICATION

System Parameter & Location	Indication		Alarm[1]		Rec[1]	Control Function	Inst. Range	Normal Operating Range	Inst. Accuracy
	Local	Contr Room	High	Low					
Pre-Concentrator Filters ΔPs	*						0-50 psid	0-10 psid	±.25 psid
Pre-Concentrator Ion Exchangers ΔPs	*						0-50 psid	1-15 psid	±.25 "
Boric Acid Condensate Ion Exchangers ΔPs	*						0-50 psid	1-15 psid	±.25 "
Boric Acid Discharge Strainer ΔP	*						0-50 psid	0-10 psid	±.25 "
Laundry Filter ΔP	*						0-50 psid	1-10 psid	±.25 "
Waste Filter ΔP	*						0-50 psid	1-10 psid	±.25 "
Waste Ion Exchanger ΔP	*						0-50 psid	1-15 psid	±.25 "
Reactor Drain Tank Level		*	*	*		Reactor Drain Pumps	0-100%		±0.5%
Flash Tank Level	*	*	*	*		Flash Tank Pumps, Isolate Tank & Bypass to Holdup Tanks	0-100%		±0.5%
Holdup Tanks Levels		*	*	*		Holdup Drain Pumps	0-100%		±0.5%
Boric Acid Holding Tank Level	*		*[2]	*[2]		Boric Acid Holding Pumps	0-100%		±2.0%
Boric Acid Condensate Tanks Levels	*		*[2]	*[2]		Boric Acid Condensate Pumps	0-100%		±2.0%

1 All alarms and recorders are in the control room unless otherwise indicated.

2 Alarmed on local Waste Management System annunciator panel.

TABLE 11-5-9 (Continued)

WASTE MANAGEMENT SYSTEM INSTRUMENTATION APPLICATION

System Parameter & Location	Indication Local	Indication Contr Room	Alarm[1] High	Alarm[1] Low	Rec[1]	Control Function	Inst. Range	Normal Operating Range	Inst. Accuracy
Waste Condensate Tanks Levels	*		*[2]	*[2]		Waste Condensate Pumps	0-100%		±2.0%
Equipment Drain Tank Level	*		*[2]	*[2]		Equipment Drain Pump	0-100%		±2.0%
Chemical Drain Tank Level	*		*[2]	*[2]		Chemical Drain Pump	0-100%		±2.0%
Laundry Drain Tanks Levels	*		*[2]	*[2]		Laundry Drain Pump	0-100%		±2.0%
Spent Resin Tank Level	*		*[2]				0-100%		±0.5%
Gas Surge Tank Level			*				-		-
Flash Tank N_2 Supply Flow	*						0-5 scfm	1.5 scfm	±0.1 scfm
Flash Tank Flow						Flash Tank Pumps, N_2 Supply Flow	-	-	-
Sluicing Water Flow	*						0-150 gpm	0-100 gpm	±3.0 gmp
Liquid Waste Discharge Flow		*			*	Liquid Waste Discharge Isolation Valves	0-50 gpm	10-50 gpm	±.25 gpm
Waste Gas Discharge Flow	*	*			*		0-10 scfm	.5-2.0 scfm	±0.2 scfm

[1] All alarms and recorders are in the control room unless otherwise indicated.

[2] Alarmed on local Waste Management System annunciator panel.

TABLE 11-5-9 (Continued)

WASTE MANAGEMENT SYSTEM INSTRUMENTATION APPLICATION

System Parameter & Location	Indication		Alarm[1]		Rec[1]	Control Function	Inst. Range	Normal Operating Range	Inst. Accuracy
	Local	Contr Room	High	Low					
Liquid Waste Discharge Radiation		*	*		*	Liquid Waste Discharge Isolation Valves	10^{-7}-10^{-1} mc/cc	Variable	± 0.5%
Waste Gas Discharge Radiation		*	*		*	Waste Gas Discharge Isolation Valve	10^{-5}-10^{3} mc/cc	Variable	± 0.5%
Boric Acid Concentrator 1A	*		*[3]				-	-	-
Boric Acid Concentrator 1B	*		*[3]				-	-	-
Radioactive Waste Concentrator	*		*[3]				-	-	-
Waste Management System Annunciator Panel			*[3]				-	-	-
Gas Analyzer	*		*[3]				-	-	-

[1] All alarms and recorders are in the control room unless otherwise indicated.

[2] Alarmed on local Waste Management System annunciator panel.

[3] Master alarm annunciated in control room for any individual system alarm condition.

11.6 OFF-SITE RADIOLOGICAL MONITORING PROGRAM

11.6.1 EXPECTED BACKGROUND

A preoperational radioactivity monitoring program is being conducted to determine the magnitude and nature of the radioactivity in the environment surrounding the site prior to the startup of the plant. The information obtained will serve as a base line in evaluating any changes in environmental radioactivity level that may possibly be attributed to the plant operation.

The preoperational program is being closely coordinated with the State program for monitoring radioactivity levels. Discussions held with State agencies assist in formulating an acceptable preoperational environmental radioactivity monitoring program. The resulting program will be reviewed periodically to assure maximum effectiveness.

The expected annual whole body exposure from natural sources is 120 mrem.

11.6.2 CRITICAL PATHWAYS

Details of the potential exposure pathways to man are presented in Section 2.3.7.3 of the Hutchinson Island Environmental Report.

Reconcentration of specific radionuclides by various trophic levels of marine ecology are considered in the environmental monitoring program by including samples of Indian River and Atlantic Ocean water, fish and shellfish, and bottom sediments. Interpretation of surveillance results will be based on selected radionuclides, such as Cobalt 60, Manganese 54, Cesium 137 and Iodine 131. Factors to be considered in the selection of the radionuclides are:

a) The estimated composition and relative concentration of the various radionuclides in the plant liquid effluent

b) The reconcentration factors for these radionuclides in the species of fish and shellfish caught in the Indian River and in the ocean

c) The appropriate limits established by regulatory requirement.

The Applicant has a program to study marine ecology which will include radioactivity effects. Fish and shellfish samples will be analyzed for gross radioactivity and for the controlling radionuclides. Recognition of the reconcentration factors, in conjunction with the 10CFR20 requirements, will assure that discharges are within acceptable limits for environmental exposure.

11.6.3 SAMPLING MEDIA, LOCATIONS AND FREQUENCY

Sampling stations and sites are established on the basis of population density and distribution, meteorological, hydrological and topography conditions.

A comprehensive sampling program is being established including definition of types of samples, number of samples, frequency of collections, and methods of analysis, and will be initiated at least 12 months prior to plant startup. Section 15.4, Technical Specifications, summarizes this program.

Marine organisms will be collected where available near the site, with emphasis on obtaining samples near the plant discharge.

Bottom sediment samples will be collected at the plant outfall. Surface water samples from the outfall will be collected and composited for analysis. Ground water samples will be collected from an appropriate nearby well. Soil samples will be collected from the immediate plant environment. Air radiation dosimeters or air particulate samplers will be located at various onsite or offsite positions. Selection of these positions will be made after considering the prevailing wind directions.

11.6.4 ANALYTICAL SENSITIVITY

The collected samples will include samples of river, ocean and well water, soil, air particulates, farm and dairy products, fish and shellfish, and river and ocean bottom sediments from the vicinity of the plant. Sample radioactivity analyses, based on the type of sample and information desired, will include one or more of the following: Gross alpha, gross beta, gross beta-gamma, Potassium 40, Iodine 131, Strontium 90 and others as appropriate.

The operational environmental radioactivity program will be similar to the preoperational program. A sampling and analysis schedule will be established based on the level of activity found in the plant effluent and in the environment samples. If the gross beta-gamma analysis shows an increase in gross gamma, a gamma spectrum analysis would be made to identify the nuclides present. Results of the sample analyses will be evaluated to insure the effectiveness of plant radiation control and compliance with the requirements of all regulatory agencies, both Federal and State. This program will be reviewed and revised as required.

Minimum levels of detectability in gaseous and liquid media for the following radionuclides assumes continuous monitoring with the level cited measurable within a four hour period are as follows:

Nuclide	Minimum Concentration in Gas, µCi/ml	Minimum Concentration in Liquid, µCi/ml
H-3	$5x10^{-6}$	-
Co-58	$2x10^{-11}$	$9x10^{-7}$
Kr-58	$3x10^{-7}$	$8x10^{-5}$
Sr-90	$4x10^{-12}$	-
I-131	$4x10^{-12}$	$3x10^{-7}$
Xe-133	$5x10^{-7}$	$6x10^{-6}$
Cs-137	$5x10^{-12}$	$4x10^{-7}$

(Proposed ANSI standard, Subcommittee N/13/42, submitted 1/12/72)

11.6.5 DATA ANALYSIS AND PRESENTATION

Environmental monitoring data shall be evaluated in terms of potential human radiation exposure. The value selected as the permissible whole body exposure to individuals living in the vicinity of the plant may be 5 or 10 mrem per year, corresponding to an "as low as practicable" consensus value.

11.6.6 PROGRAM STATISTICAL SENSITIVITY

Because of the normally fractional level of operations emission compared with background radioactivity in the plant vicinity, the program statistical sensitivity appears to be poor on an absolute activity level basis. Sufficient spectrographic analysis is done on enough samples to provide reasonable assurance that potential human exposure will not exceed 5 mrem per year ±100 percent from plant operations.

TABLE OF CONTENTS

CHAPTER 12
RADIATION PROTECTION

Section	Title	Page
12.1	SHIELDING	12.1-1
12.1.1	DESIGN OBJECTIVES	12.1-1
12.1.2	DESIGN DESCRIPTION	12.1-3
12.1.2.1	Reactor Building Primary Shield	12.1-3
12.1.2.2	Reactor Building Secondary Shield	12.1-4
12.1.2.3	Shield Building	12.1-4
12.1.2.4	Fuel Handling Building Shielding	12.1-4
12.1.2.5	Reactor Auxiliary Building Shielding	12.1-5
12.1.2.6	Control Room Shielding	12.1-5
12.1.2.7	Emergency Core Cooling System Area Shielding	12.1-6
12.1.2.8	Materials and Structural Requirements	12.1-6
12.1.3	SOURCE TERMS	12.1-6
12.1.4	AREA MONITORING	12.1-16
12.1.4.1	Design Bases	12.1-16
12.1.4.2	System Description	12.1-16
12.1.4.3	Design Evaluation	12.1-19
12.1.4.4	Testing and Inspection	12.1-21
12.1.5	OPERATING PROCEDURES	12.1-22
12.1.6	ESTIMATES OF EXPOSURE	12.1-22
12.2	VENTILATION	12.2-1
12.2.1	DESIGN OBJECTIVES	12.2-1
12.2.2	DESIGN DESCRIPTION	12.2-1
12.2.2.1	General	12.2-1
12.2.2.2	Containment Airborne Radioactivity Removal System	12.2-3
12.2.2.3	Containment Purge System	12.2-4
12.2.2.4	Reactor Auxiliary Building Main Ventilation System	12.2-5
12.2.2.5	Fuel Handling Building Ventilation System	12.2-6
12.2.2.6	Control Room Ventilation System	12.2-7
12.2.3	SOURCE TERMS	12.2-8

TABLE OF CONTENTS (CONT'D)

CHAPTER 12

RADIATION PROTECTION

Section	Title	Page
12.2.4	AIRBORNE RADIOACTIVITY MONITORING	12.2-9
12.2.4.1	Containment Atmosphere Radiation Monitoring System	12.2-9
12.2.4.2	Plant Vent Radiation Monitoring System	12.2-11
12.2.5	OPERATING PROCEDURES	12.2-13
12.2.6	ESTIMATES OF INHALATION DOSES	12.2-13
12.3	HEALTH PHYSICS PROGRAM	12.3-1
12.3.1	PROGRAM OBJECTIVES	12.3-1
12.3.2	FACILITIES AND EQUIPMENT	12.3-1
12.3.3	PERSONNEL DOSIMETRY	12.3-3

LIST OF TABLES

Table	Title	Page
12.1-1	MAXIMUM GAMMA SPECTRA (γ/cm^2-sec) OUTSIDE REACTOR VESSEL	12.1-7
12.1-2	MAXIMUM NEUTRON SPECTRA (n/cm^2-sec) OUTSIDE REACTOR VESSEL	12.1-8
12.1-3	REACTOR COOLANT ACTIVITY SHUTDOWN CONDITIONS (70F)	12.1-10
12.1-4	FISSION PRODUCT GAMMA SOURCE IN CONTAINMENT BUILDING (MEV/SEC) FOLLOWING MAXIMUM HYPOTHETICAL ACCIDENT	12.1-11
12.1-5	WASTE MANAGEMENT SYSTEM (WMS) MAXIMUM NUCLIDE CONCENTRATIONS DURING ANTICIPATED OPERATIONAL OCCURRENCES (1.0% failed fuel) (μCi/cc)	12.1-12
12.1-6	WASTE MANAGEMENT SYSTEM (WMS) MAXIMUM NUCLIDE CONCENTRATIONS DURING NORMAL OPERATIONS (0.1% failed fuel) (μCi/cc)	12.1-14
12.1-7	AREA RADIATION MONITORING SYSTEM	12.1-17
12.1-8	AREA RADIATION MONITORING ENVIRONMENTAL AND POWER SUPPLY DESIGN CONDITIONS	12.1-20
12.2-1	AREAS WITH AIRBORNE RADIOACTIVITY SOURCES	12.2-2
12.2-2	DESIGN CHARACTERISTICS - CONTAINMENT RADIATION MONITORING SYSTEM	12.2-10
12.3-1	PORTABLE RADIATION SURVEY INSTRUMENTS	12.3-5

LIST OF FIGURES

Figure	Title
12.1-1	MAXIMUM DOSE RATE LEVELS - REACTOR CONTAINMENT BUILDING FLOOR ELEVATION 18', 23' and 62'
12.1-2	MAXIMUM DOSE RATE LEVELS - REACTOR CONTAINMENT BUILDING FLOOR ELEVATION 45'
12.1-3	MAXIMUM DOSE RATE LEVELS - REACTOR AUXILIARY BUILDING FLOOR ELEVATION - 0.5'
12.1-4	MAXIMUM DOSE RATE LEVELS - REACTOR AUXILIARY BUILDING FLOOR ELEVATION 19.5'
12.1-5	MAXIMUM DOSE RATE LEVELS - REACTOR AUXILIARY BUILDING FLOOR ELEVATION 43'
12.1-6	CONTROL ROOM LAYOUT ISOMETRIC
12.1-7	ISOMETRIC VIEW OF CONTROL ROOM SHIELDING AND POST-LOCA RADIATION SOURCES
12.1-8	CUMULATIVE POST-MHA WHOLE BODY DOSE IN THE CONTROL ROOM
12.1-9	AREA RADIATION MONITORING SYSTEM FUNCTIONAL BLOCK DIAGRAM
12.2-1	CONTAINMENT RADIATION MONITORING SYSTEM
12.2-2	PLANT VENT RADIATION MONITORING SYSTEM

CHAPTER 12

RADIATION PROTECTION

12.1 SHIELDING

12.1.1 DESIGN OBJECTIVES

Radiation shielding is designed for continuous safe operation at a core power level of 2700 Mwt with system activity levels stemming from fuel cladding defects in the equivalent of one percent of the fuel rods.

In addition, the shielding ensures that, in the event of a maximum hypothetical accident (MHA), the integrated radiation exposures off-site and in the control room, due to the contained activity, do not result in radiation doses to plant personnel or the general public in excess of 10CFR100 limits.

The applicable portions of 10CFR20 and 10CFR100 are used as the bases for defining the following limits on acceptable exposures in areas within and beyond the site boundary for normal operation and for accident conditions, respectively.

Location	Max Whole Body Dose Rate (mrem/hr)
a) Site Boundary	
Normal Operation	0.001
Following MHA	25 Rem in 2 hrs
b) Service Building	
Normal Operation	0.05
c) Turbine Building	
Normal Operation	0.05
d) Reactor Auxiliary & Fuel Handling Buildings	
Continuous Occupancy	
Outside Controlled Access Areas	0.5
Inside Controlled Access Areas	2.5

Location	Max Whole Body Dose Rate (mrem/hr)
Controlled Occupancy	
Occupancy for 6 hr/wk	15.0
Occupancy determined by Health Physics Staff	100.0
e) Limited Access Areas in Containment Structure during Operation at Full Power	100.0
f) Control Room	
Normal Operation	0.5
Following MHA	3 rem integrated whole body dose over 90 days after accident

Areas with dose rates greater than 100 mrem/hr are isolated and access to these areas is controlled in accordance with 10CFR20 by means described below.

The calculated maximum doserate levels (based on 1 percent failed fuel) in the plant are shown in Figures 12.1-1 through 12.1-5. For purposes of determining length of occupancy, the plant has been divided into the following zones:

Zone	Access	Dose Rate (mr/hr)
I	No restriction	< 0.5
II	Occupational Access	$0.5 \rightarrow 2.5$
III	Periodic Access	$2.5 \rightarrow 15.0$
IV	Limited Access	$15.0 \rightarrow 100.0$
V	Restricted Access	> 100.0

Zone II is a restricted radiation area to which plant personnel can have continuous access during the regular 40 hr/wk work schedule, without exceeding the allowable whole body dose of 1-1/4 rems per calendar quarter (10CFR20.101). This zone requires no posting.

Zones III and IV are posted with "Caution-Radiation Area" signs. The length of occupancy varies with the actual radiation level in the room, the nature of the radiation present, and the past radiation history of the personnel entering.

Zone V is posted with "Caution-High Radiation Area" signs, and access to it is strictly controlled. Zone V represents inaccessible restricted areas, entry to which is permissible only with the approval of the shift supervisor. Each high radiation area therefore, is provided with a barrier or barriers which are kept locked. Access is by keys distributed from the control room.

Neutron and gamma radiation surveys are performed in all accessible areas of the plant as required to determine shielding adequacy. Areas such as the containment operating floor, reactor vessel head, reactor coolant loop compartments and spent fuel handling areas are surveyed prior to access for refueling or after shutdown and a time-limited work schedule established.

12.1.2 DESIGN DESCRIPTION

Plant site plan and general arrangement drawings showing shielding and location of equipment are shown on Figures 1.2-1 through 1.2-19.

Design criteria for penetrations through shield walls and for acceptable radiation levels at valve stations for process equipment containing radioactive fluids are identical to the shielding criteria for the shield walls and process equipment involved.

12.1.2.1 Reactor Building Primary Shield

The primary shield function is to limit radiation emanating from the reactor vessel; this radiation during operation consists of neutrons (both fast and slow) emitted from the core, prompt fission gammas, fission product gammas, and gamma radiation resulting from neutron capture in the core internals and vessel. Following shutdown, only fission product gammas, and gamma radiation from neutron activation of the coolant and corrosion products are present.

The primary shield consists of 7'-3" of reinforced concrete surrounding the reactor vessel. The annular cavity between the primary shield and the reactor vessel is air cooled to prevent overheating and dehydration of the concrete shield (see Section 9.4.7.2.3).

The primary shield arrangement and thickness (shown in Figures 1.2-8, 1.2-9 and 1.2-10 are designed to:

a) attentuate the neutron flux to prevent excessive activation of unit components and structures

b) reduce the contribution of radiation from the reactor to obtain a reasonable division of the shielding function between the primary and secondary shield

c) reduce residual radiation from the core to a level which permits access to the region between the primary and secondary shields at a reasonable time after shutdown

d) permit access during shutdown for inspections

12.1.2.2 Reactor Building Secondary Shield

The secondary shield reduces the radiation activity from the reactor coolant system to a level which allows limited access to the containment during operation and to supplement primary shielding. The controlling radiation source in the design of the secondary shield is N-16 resulting from the (n,p) reaction with the oxygen in the coolant.

The secondary shield is reinforced concrete 4'-0" thick surrounding the reactor coolant piping, pumps, steam generators, and pressurizer.

In addition, a partial 2'-0" thick reinforced concrete wall is located midway between the primary and secondary shields in the main steam and feedwater lines penetration region, to prevent streaming of gamma radiation from the reactor coolant system through the penetration openings.

The entire secondary shield system is shown in Figures 1.2-8, -9 and -10.

12.1.2.3 Shield Building

The steel containment structure is enclosed by a reinforced concrete shield building with 3'-0" thick cylindrical walls and a 2'-6" thick dome. In conjunction with the primary and secondary shields, the shield building limits the radiation level outside the structure from all sources within the containment to less than 0.5 mrem/hr at 2700 Mwt with 1 percent failed fuel.

The combination of shield structures ensures that radiation doses at the site boundary due to direct radiation from the activity within the containment are below the recommended guideline values of 10CFR100 in the event of the MHA.

12.1.2.4 Fuel Handling Building Shielding

Shielding is provided for radiation protection of plant personnel during all phases of spent fuel removal, storage and preparation for off-site shipment. Operations requiring shielding of personnel are: spent fuel removal from the reactor, spent fuel transfer through the refueling canal and transfer tube, spent fuel storage, and spent fuel shipping cask loading prior to transporation.

All spent fuel removal, transfer and shipping cask loading operations are done under a minimum of 10'-4" of borated water.

The refueling cavity above the reactor vessel flange is flooded to elevation 60 ft to provide 24 ft of water shielding above the reactor vessel flange. This height assures 132 inches of water above the active portion

of a withdrawn fuel assembly at its highest point of travel. Under these conditions, the dose rate from the spent fuel assembly is less than 2 mrem/hr at the water surface.

The fuel assembly removed from the reactor vessel is moved to the upender and horizontally transferred to the fuel pool by the fuel transfer mechanism inside the fuel transfer tube. A 6'-0" concrete shield around the refueling cavity protects the refueling personnel from radiation from the spent fuel assemblies and reactor internals.

The fuel pool is flooded to provide shielding as the fuel is being withdrawn from the fuel transfer tube and being raised by the spent fuel handling machine for insertion in the spent fuel rack. The concrete sides of the fuel pool are 6'-0" thick to ensure a dose rate of less than 0.5 mrem/hr at the outer surface of the structure.

12.1.2.5 Reactor Auxiliary Building Shielding

The reactor auxiliary building shield walls are designed to protect personnel working near various system components, such as those in the chemical and volume control, waste management, shutdown cooling, and sampling systems.

In addition, major pieces of potentially radioactive equipment are housed within shielded compartments so that access to any component for repair or inspection is permissible without shutdown and decontamination of the entire system.

The concrete thickness of each compartment shield wall is sufficient to reduce the dose rate outside the compartment at nearby normally accessible areas to less than 2.5 mrem/hr. The individual enclosures also serve to contain any major spill of radioactive liquid from any of the tanks or other components in the system.

A floor drainage system conveys any liquid spills to floor sumps from which they are pumped to the waste management system for processing.

The floor drains are embedded in concrete so as to contain any spills from leaking lines and/or to shield the line in access areas.

12.1.2.6 Control Room Shielding

Direct radiation contribution to the dose rates in the control room following a MHA stem from the following:

a) airborne activity in the containment (TID releases are assumed to be uniformly dispersed throughout the containment free volume)

b) shield building ventilation system filters

c) hydrogen purge system filters

The control room layout and location relative to the sources of direct radiation is shown in Figures 12.1-6 and 12.1-7 respectively. The

nearest filter is located at least 70 ft away with a minimum of 4 ft of concrete intervening. The shield thickness between the control room and the containment, located 31 ft away, is a minimum of 6 ft equivalent concrete (2 inch thick steel pressure vessel, 3 ft thick concrete shield building and 2 ft thick control room wall).

The cumulative dose to the control room following a MHA is shown in Figure 12.1-8. The dominant contribution comes from the containment.

12.1.2.7 <u>Emergency Core Cooling System Area Shielding</u>

Shielding of the emergency core cooling system rooms has been designed for normal operation and shutdown conditions. Under normal plant operating conditions only the reactor drain pump and the low pressure safety injection pumps are sources of radiation. Sufficient shielding is achieved with 1 foot thick concrete walls. A 2 foot thick concrete wall separates the two rooms. One room houses one low pressure safety injection pump, one high pressure safety injection pump, one containment spray pump and two reactor drain tank pumps; the other room houses one low pressure safety injection pump, two high pressure safety injection pumps and one containment spray pump. Refer to Figure 1.2-12.

This arrangement allows inspection and servicing of the safety feature systems during plant operation and shutdown, and also some accessibility during the long term cooling following the postulated accident.

12.1.2.8 <u>Materials and Structural Requirements</u>

The concrete for the primary and secondary shield, shield building, reactor auxiliary building and fuel handling building shield walls has a density of 138 lb/ft^3. Since the primary and secondary shield walls serve as support for the reactor coolant system components and provide missile protection and support for the refueling apparatus, reinforced concrete is used.

12.1.3 SOURCE TERMS

Plant shielding is designed to attenuate neutron and gamma radiation emanating from the following sources:

a) reactor vessel

b) reactor coolant loops

c) auxiliary systems equipment

d) spent fuel assemblies

e) radioactive material released during postulated accidents

The gamma and neutron sources utilized in the design of the primary shield are shown in Tables 12.1-1 and 12.1-2. The gamma sources include capture gammas from the reactor core, internals and vessel.

TABLE 12.1-1

MAXIMUM GAMMA SPECTRA (γ/cm^2-sec)
OUTSIDE REACTOR VESSEL

E(mev)	Side	Top	Bottom
10.00	1.40(+7)*	1.57(+3)	7.34(+3)
9.00	1.22(+8)	1.16(+4)	1.08(+7)
8.00	2.29(+8)	1.23(+4)	1.93(+7)
7.00	2.28(+8)	2.07(+4)	3.42(+7)
6.00	2.41(+8)	2.53(+4)	4.91(+7)
5.00	2.99(+8)	3.21(+4)	6.71(+7)
4.00	3.80(+8)	4.19(+4)	8.88(+7)
3.00	5.74(+8)	5.47(+4)	1.25(+8)
2.00	9.27(+8)	8.02(+4)	1.94(+8)
1.38	5.46(+8)	4.82(+4)	1.14(+8)
1.00	5.01(+8)	4.33(+4)	1.07(+8)
0.75	6.88(+8)	5.97(+4)	1.51(+8)
0.50	7.45(+8)	9.66(+4)	2.51(+8)
0.25	9.65(+8)	1.18(+5)	3.21(+8)

* Denotes power of ten.

TABLE 12.1-2

MAXIMUM NEUTRON SPECTRA (n/cm^2-sec)
OUTSIDE REACTOR VESSEL

E(mev)	Side	Top	Bottom
18	9.04(+3)*		1.09(-2)
14	5.22(+5)		8.08(-1)
10	1.34(+7)		1.35(+1)
8	2.88(+7)		1.00(+1)
6	5.13(+7)		1.06(+1)
4	4.72(+7)	insignificant	1.16(+1)
3	3.30(+7)		1.33(+1)
2	3.49(+7)		1.17(+1)
1	3.99(+7)		1.53(+1)
0.33	4.67(+7)		3.42(+1)

* Denotes power of ten.

The activity of N-16 in the reactor coolant, 3.6 X 10^6 γ/cc-sec at the vessel outlet nozzles, determines the amount of shielding required around the reactor coolant loops during full power operation.

During shutdown the major sources of activity in the reactor coolant loops are the fission and corrosion products reported in Table 12.1-3, corrected to account for the change in coolant density from operating of shutdown conditions.

The fission product gamma radiation source strength in the containment at various times following the maximum hypothetical accident is shown in Table 12.1-4.

Fission product and corrosion activity also determine the shielding requirements for the auxiliary systems. The total quantity of the principal nuclides in process equipment that contains or transports radioactivity is identified for selected locations in Tables 12.1-5 and 12.1-6. The selected locations are indicated by the numbers within ellipses on Figures 11.2-1, 11.2-3, 11.2-4, 11.3-1, 9.3-4 and 9.3-5. Expected maximum values, i.e., those used as a design basis for shielding requirements, correspond to the equivalent activity values calculated to exist due to fuel cladding defects in 1.0 percent of the fuel rods and are shown in Table 12.1-5. Expected average values correspond to fuel cladding defects in 0.1 percent of the fuel rods and are shown in Table 12.1-6.

The activity in the chemical and volume control system prefilter, ion exchangers and afterfilter is determined by assuming a letdown flow of 40 gpm and retention of at least 90 percent of the nonvolatile fission products, except cesium, yttrium, molybdenum and tritium; for which no removal is assumed. No credit is taken for radioactive decay.

In the boron recovery system, the activity in the preconcentrator ion exchangers is determined by assuming retention of at least 90 percent of all incoming radionuclides, except tritium, and a batch flow rate of 20 gpm. The corresponding average annual flow rate is 1.5 gpm based on processing 780,000 gallons, equivalent to 10 reactor volumes, from sources listed in Table 11.2-1. Reactor coolant pump seal flow is not included in this average. A decontamination factor (DF) of 10^4 (ratio of bottoms to distillate activity) is assumed for the boric acid concentrator for all nuclides (except tritium). The influent activity is taken to be concentrated in the bottoms by a factor of 300. No operation of the boric acid condensate ion exchanger is assumed.

In the waste management system, the activity present in the gas decay tanks is that of the noble gases present in the coolant, with no credit taken for radioactive decay.

No radioactive waste or shipping casks are stored outside of buildings. See Section 11.5.7.

The routing of piping carrying potentially radioactive matter in quantities significant enough to affect the amount of exposure to plant personnel is done during the design stage to ensure that the piping is

TABLE 12.1-3

REACTOR COOLANT ACTIVITY

SHUTDOWN CONDITIONS (70 F)

Nuclide	Half Life	Specific Activity (μCi/cc) Anticipated Operational Occurrence (1.0% Failed Fuel)	Specific Activity (μCi/cc) Normal Operation (1.0% Failed Fuel)	Nuclide	Half Life	Specific Activity (μCi/cc) Anticipated Operational Occurrence (1.0% Failed Fuel)	Specific Activity (μCi/cc) Normal Operation (1.0% Failed Fuel)
H-3	12.3y	0.182	0.148	I-133	21h	7.81	0.781
Br-84	32m	6.43(-2)*	6.43(-3)	Xe-133	5.3d	250.0	25.0
Kr-85m	4.4h	2.05	0.205	Te-134	42m	3.62(-2)	3.62(-3)
Kr-85	10.8y	1.22	1.22(-1)	I-134	52m	0.855	8.55(-2)
Kr-87	76m	1.12	1.12(-1)	Cs-134	2.1y	0.138	1.38(-2)
Kr-88	2.8h	3.58	0.358	I-135	6.7h	3.72	0.372
Rb-88	18m	3.52	0.352	Xe-135	9.2h	10.4	1.04
Rb-89	15m	8.83(-2)	8.83(-3)	Cs-136	13d	3.52(-2)	3.52(-3)
Sr-89	51d	7.0(-3)	7.0(-4)	Cs-137	30y	0.441	4.41(-2)
Sr-90	28.8y	3.6(-4)	3.6(-5)	Xe-138	17m	0.497	4.97(-2)
Y-90	64h	1.41(-3)	1.41(-4)	Cs-138	32m	0.952	9.52(-2)
Sr-91	9.7h	4.9(-3)	4.9(-4)	Ba-140	12.8d	8.42(-3)	8.42(-4)
Y-91	59d	0.153	1.53(-2)	La-140	40.2h	8.07(-3)	8.07(-4)
Mo-99	67h	2.80	0.280	Pr-143	13.6d	8.05(-3)	8.05(-4)
Ru-103	39.6d	5.7(-3)	5.7(-4)	Ce-144	285d	5.7(-3)	5.7(-4)
Ru-106	367d	3.42(-4)	3.42(-5)	Co-60**	5.2y	7.16(-4)	7.16(-4)
Te-129	67m	3.46(-2)	3.46(-3)	Fe-59**	45d	2.94(-5)	2.94(-5)
I-129	1.7(7)y	9.95(-8)	9.95(-9)	Co-58**	71d	6.43(-3)	6.43(-3)
I-131	8.0d	5.48	0.548	Mn-54**	312d	3.8(-5)	3.8(-5)
Ke-131m	12d	2.04	0.204	Cr-51**	27d	5.24(-3)	5.24(-3)
Te-132	78h	0.455	4.55(-2)	Zr-95**	65d	1.29(-6)	1.29(-6)
I-132	2.3h	1.41	0.141				

* Denotes Power of ten

** Corrosion products

TABLE 12.1-4

FISSION PRODUCT GAMMA SOURCE IN CONTAINMENT BUILDING (MEV/SEC) FOLLOWING MAXIMUM HYPOTHETICAL ACCIDENT

Energy Interval (Mev)

Time	0.1 - 0.4	0.4 - 0.9	0.9 - 1.35	1.35 - 1.8	1.8 - 2.2	2.2 - 2.6	2.6
0	2.11 (18)*	1.26 (19)	5.00 (18)	9.39 (18)	6.10 (18)	4.39 (18)	1.99 (18)
0.5 hr	2.01 (18)	1.11 (19)	4.56 (18)	3.44 (18)	3.17 (18)	3.56 (18)	2.54 (17)
1 hr	1.94 (18)	9.81 (18)	3.98 (18)	3.05 (18)	2.14 (18)	2.95 (18)	9.32 (16)
2 hr	1.84 (18)	7.78 (18)	3.22 (18)	2.44 (18)	1.47 (18)	2.09 (18)	3.00 (16)
8 hr	1.43 (18)	4.39 (18)	1.48 (180	1.08 (18)	4.76 (17)	3.95 (17)	1.07 (16)
24 hr	7.93 (17)	3.05 (18)	5.95 (17)	4.27 (17)	1.59 (17)	7.08 (16)	1.76 (14)
1 wk	2.10 (17)	7.32 (17)	1.12 (17)	1.20 (17)	4.03 (16)	2.13 (16)	1.19 (14)
1 mo	2.81 (16)	8.76 (16)	2.17 (15)	1.66 (16)	1.00 (15)	1.27 (15)	3.78 (13)
2 mo	4.10 (15)	5.39 (16)	3.68 (14)	2.95 (15)	5.29 (14)	1.95 (14)	1.22 (13)
4 mo	5.15 (14)	2.98 (16)	1.36 (14)	3.34 (14)	4.42 (14)	2.34 (13)	5.29 (12)

* Denotes power of ten

TABLE 12.1-5

WASTE MANAGEMENT SYSTEM (WMS) MAXIMUM NUCLIDE CONCENTRATIONS DURING ANTICIPATED OPERATIONAL OCCURRENCES (1.0% failed fuel) (μCi/cc)

Nuclide	CVCS 6 WMS 1	CVCS 7	CVCS 10 WMS 2,3,4,5,6,7,8	WMS 9,10,11	WMS 12,13,14	WMS 15	WMS 18,19,20,21
H-3	1.2(-1)*	1.2(-1)	1.2(-1)	1.2(-1)	1.2(-1)	1.2(-1)	1.2(-1)
Br-84	4.11(-3)	4.11(-3)	4.11(-6)	4.11(-6)	0	0	0
Kr-85m	1.57(0)	1.57(0)	1.57(0)	7.85(-1)	0	0	0
Kr-85	1.14(0)	1.14(0)	1.14(0)	5.7(-1)	4.72(-1)	4.72(-1)	2.36(-1)
Kr-87	8.0(-1)	8.0(-1)	8.0(-1)	4.0(-1)	0	0	0
Kr-88	2.68(0)	2.68(0)	2.68(0)	1.34(0)	0	0	0
Rb-88	5.96(-1)	5.96(-1)	5.96(-1)	5.96(-1)	0	0	0
Rb-89	9.02(-3)	9.02(-3)	9.02(-3)	9.02(-3)	0	0	0
Sr-89	3.8(-3)	3.8(-3)	3.8(-5)	3.8(-5)	3.36(-5)	3.36(-6)	1.18(-8)
Sr-90	1.84(-1)	1.84(-1)	1.84(-3)	1.84(-3)	1.84(-3)	1.84(-4)	9.2(-7)
Y-90	6.71(-4)	6.71(-4)	6.71(-4)	6.71(-4)	6.58(-5)	6.58(-6)	3.29(-8)
Sr-91	1.69(-3)	1.69(-3)	1.69(-5)	1.69(-5)	3.36(-12)	3.36(-13)	1.18(-15)
Y-91	2.48(-2)	2.48(-2)	2.48(-2)	2.48(-2)	2.23(-2)	2.23(-3)	1.12(-5)
Mo-99	1.21(0)	1.21(0)	1.21(0)	1.21(0)	1.25(-1)	1.25(-2)	6.25(-5)
Ru-103	3.24(-3)	3.24(-3)	3.24(-4)	3.24(-4)	2.77(-4)	2.77(-5)	1.39(-7)
Ru-106	1.67(-4)	1.67(-4)	1.67(-5)	1.67(-5)	1.64(-5)	1.64(-6)	8.2(-9)
Te-129	1.6(-2)	1.6(-2)	1.6(-3)	1.6(-3)	0	0	0
I-129	4.0(-8)	4.0(-8)	4.0(-11)	4.0(-11)	4.0(-11)	4.0(-14)	2.0(-16)
I-131	1.7(0)	1.7(0)	1.7(-3)	1.7(-3)	7.84(-4)	7.84(-7)	3.92(-9)
Xe-131m	1.18(0)	1.18(0)	1.18(0)	5.9(-1)	2.89(-1)	2.89(-1)	5.78(-2)
Te-132	2.31(-1)	2.31(-1)	2.31(-3)	2.31(-3)	3.39(-4)	3.39(-5)	1.2(-7)
I-132	3.46(-1)	3.46(-1)	3.46(-4)	3.46(-4)	0	0	0
I-133	1.88(0)	1.88(0)	1.88(-3)	1.88(-3)	1.51(-6)	1.51(-9)	7.55(-12)
Xe-133	1.97(+2)	1.97(+2)	1.97(+2)	9.85(+1)	2.49(+1)	2.49(+1)	4.98(0)
Te-134	4.61(-3)	4.61(-3)	4.61(-4)	4.61(-4)	0	0	0
I-134	6.95(-2)	6.95(-2)	6.95(-5)	6.95(-5)	0	0	0
Cs-134	2.25(-1)	2.25(-1)	2.25(-1)	2.25(-1)	2.25(-1)	2.25(-3)	1.13(-5)

TABLE 12.1-5 CONTINUED

Nuclide	CVCS 6 WMS 1	CVCS 7	CVCS 10 WMS 2,3,4,5,6,7,8	WMS 9,10,11	WMS 12,13,14	WMS 15	WMS 18,19,20,21
N-135	6.39(-1)	6.39(-1)	6.39(-4)	6.39(-4)	0	0	0
Xe-135	6.49(0)	6.49(0)	6.49(0)	3.25(0)	0	0	0
Cs-136	3.47(-2)	3.47(-2)	3.47(-2)	3.47(-2)	2.15(-2)	2.15(-4)	1.08(-6)
Cs-137	9.58(-1)	9.58(-1)	9.58(-1)	9.58(-1)	9.58(-1)	9.58(-3)	4.79(-5)
Xe-138	3.4(-1)	3.4(-1)	3.4(-1)	1.7(-1)	0	0	0
Cs-138	3.48(-1)	3.48(-1)	3.48(-1)	3.48(-1)	0	0	0
Ba-140	4.64(-3)	4.64(-3)	4.64(-5)	4.64(-5)	2.85(-5)	2.85(-6)	1.93(-8)
La-140	4.58(-3)	4.58(-3)	4.58(-4)	4.58(-4)	1.1(-4)	1.1(-5)	5.5(-8)
Pr-143	4.01(-3)	4.01(-3)	4.01(-4)	4.01(-4)	2.53(-4)	2.53(-5)	1.27(-7)
Ce-144	2.57(-3)	2.57(-3)	2.57(-4)	2.57(-4)	2.46(-4)	2.46(-5)	1.23(-7)
Co-60 **	8.64(-4)	8.64(-5)	8.64(-5)	8.64(-5)	8.64(-5)	8.64(-5)	4.32(-7)
Fe-59 **	3.54(-5)	3.54(-6)	3.54(-6)	3.54(-6)	3.08(-6)	3.08(-6)	1.54(-8)
Co-58 **	7.74(-3)	7.74(-4)	7.74(-4)	7.74(-4)	7.14(-4)	7.14(-4)	3.57(-6)
Mn-54 **	4.57(-5)	4.57(-6)	4.57(-6)	4.57(-6)	4.47(-6)	4.47(-6)	2.24(-8)
Cr-51 **	6.32(-3)	6.32(-4)	6.32(-4)	6.32(-4)	5.02(-4)	5.02(-4)	2.51(-6)
Zr-95	1.56(-7)	1.56(-8)	1.56(-8)	1.56(-8)	1.42(-8)	1.42(-8)	7.1(-11)

Denotes power of ten

** Corrosion products

TABLE 12.1-6

WASTE MANAGEMENT SYSTEM (WMS) MAXIMUM NUCLIDE CONCENTRATIONS DURING NORMAL OPERATIONS (0.1% failed fuel) (μCi/cc)

Nuclide	CVCS 6 WMS 1	CVCS 7	CVCS 10 WMS 2,3,4,5,6,7,8	WMS 9,10,11	WMS 12,13,14	WMS 15	WMS 18,19,20,21
H-3	8.4(-2)*	8.4(-2)	8.4(-2)	8.4(-2)	8.4(-2)	8.4(-2)	8.4(-2)
Br-84	4.11(-4)	4.11(-4)	4.11(-7)	4.11(-7)	0	0	0
Kr-85m	1.57(-1)	1.57(-1)	1.57(-1)	7.85(-2)	0	0	0
Kr-85	1.14(-1)	1.14(-1)	1.14(-1)	5.7(-2)	4.72(-2)	4.72(-2)	2.36(-2)
Kr-87	8.0(-2)	8.0(-2)	8.0(-2)	4.0(-2)	0	0	0
Kr-88	2.68(-1)	2.68(-1)	2.68(-1)	1.34(-1)	0	0	0
Rb-88	5.96(-2)	5.96(-2)	5.96(-2)	5.96(-2)	0	0	0
Rb-89	9.02(-4)	9.02(-4)	9.02(-4)	9.04(-4)	0	0	0
Sr-89	3.8(-4)	3.8(-4)	3.8(-6)	3.8(-6)	3.36(-6)	3.36(-7)	1.18(-9)
Sr-90	1.84(-2)	1.84(-2)	1.84(-4)	1.84(-4)	1.84(-4)	1.84(-5)	9.2(-8)
Y-90	6.71(-5)	6.71(-5)	6.71(-5)	6.71(-5)	6.58(-6)	6.58(-7)	3.29(-9)
Sr-91	1.69(-4)	1.69(-4)	1.69(-6)	1.69(-6)	3.36(-13)	3.36(-14)	1.18(-16)
Y-91	2.48(-3)	2.48(-3)	2.48(-3)	2.48(-3)	2.23(-3)	2.23(-4)	1.12(-6)
Mo-99	1.21(-1)	1.21(-1)	1.21(-1)	1.21(-1)	1.25(-2)	1.25(-3)	6.25(-6)
Ru-103	3.24(-4)	3.24(-4)	3.24(-5)	2.77(-5)	2.77(-5)	2.77(-6)	1.39(-8)
Ru-106	1.67(-5)	1.67(-5)	1.67(-6)	1.67(-6)	1.64(-6)	1.64(-7)	8.2(-10)
Te-129	1.6(-3)	1.6(-3)	1.6(-4)	1.6(-4)	0	0	0
I-129	4.0(-9)	4.0(-9)	4.0(-12)	4.0(-12)	4.0(-12)	4.0(-15)	2.0(-17)
I-131	1.7(-1)	1.7(-1)	1.7(-4)	4.7(-4)	7.84(-5)	7.84(-8)	3.92(-10)
Xe-131m	1.18(-1)	1.18(-1)	5.9(-2)	2.89(-2)	2.89(-2)	2.89(-2)	5.78(-3)
Te-132	2.31(-2)	2.31(-2)	2.31(-4)	2.31(-4)	3.39(-5)	3.39(-6)	1.2(-8)
I-132	3.46(-2)	3.46(-2)	3.46(-5)	3.46(-5)	0	0	0
I-133	1.88(-1)	1.88(-1)	1.88(-4)	1.88(-4)	1.51(-7)	1.51(-10)	7.55(-13)
Xe-133	1.97(+1)	1.97(+1)	1.97(+1)	9.85(0)	2.49(0)	2.49(0)	4.98(-1)
Te-134	4.61(-4)	4.61(-4)	4.61(-5)	4.61(-5)	0	0	0
I-134	6.95(-3)	6.95(-3)	6.95(-6)	6.95(-6)	0	0	0
Cs-134	2.25(-2)	2.25(-2)	2.25(-2)	2.25(-2)	2.25(-2)	2.25(-4)	1.13(-6)

TABLE 12.1-6 CONTINUED

Nuclides	CVCS 6 WMS 1	CVCS 7	CVCS 10 WMS 2,3,4,5,6,7,8	WMS 9,10,11	WMS 12,13,14	WMS 15	WMS 18,19,20,21
I-135	6.39(-2)	6.39(-2)	6.39(-5)	6.39(-5)	0	0	0
Xe-135	6.49(-1)	6.49(-1)	6.49(-1)	3.25(-1)	0	0	0
Cs-135	3.47(-3)	3.47(-3)	3.47(-3)	3.47(-3)	2.15(-3)	2.15(-5)	1.08(-7)
Cs-137	9.58(-2)	9.58(-2)	9.58(-2)	9.58(-2)	9.58(-2)	9.58(-4)	4.79(-6)
Xe-138	3.4(-2)	3.4(-2)	3.4(-2)	1.7(-2)	0	0	0
Cs-138	3.48(-2)	3.48(-2)	3.48(-2)	3.48(-2)	0	0	0
Ba-140	4.64(-4)	4.64(-4)	4.64(-6)	4.64(-6)	2.85(-6)	2.85(-7)	1.93(-9)
La-140	4.58(-4)	4.58(-4)	4.58(-5)	4.58(-5)	1.1(-5)	1.1(-6)	5.5(-9)
Pr-143	4.01(-4)	4.01(-4)	4.01(-5)	4.01(-5)	2.53(-5)	2.53(-6)	1.27(-8)
Ce-144	2.57(-4)	2.57(-4)	2.57(-5)	2.57(-5)	2.46(-5)	2.46(-6)	1.23(-8)
Co-60 **	8.64(-4)	8.64(-5)	8.64(-5)	8.64(-5)	8.64(-5)	8.64(-5)	4.32(-7)
Fe-59 **	3.54(-5)	3.54(-6)	3.54(-6)	3.54(-6)	3.08(-6)	3.08(-6)	1.54(-8)
Co-58 **	7.74(-3)	7.74(-4)	7.74(-4)	7.74(-4)	7.14(-4)	7.14(-4)	3.57(-6)
Mn-54 **	4.57(-5)	4.57(-6)	4.57(-6)	4.57(-6)	4.47(-6)	4.47(-6)	2.24(-8)
Cr-51 **	6.32(-3)	6.32(-4)	6.32(-4)	6.32(-4)	5.02(-4)	5.02(-4)	2.51(-6)
Zr-95 **	1.56(-7)	1.56(-8)	1.56(-8)	1.56(-8)	1.42(-8)	1.42(-8)	7.1(-11)

* Denotes power of ten

** Corrosion products

adequately shielded. Any such piping designed in the field is checked for shielding adequacy.

12.1.4 AREA MONITORING

12.1.4.1 Design Bases

The area radiation monitoring system is designed to:

a) warn of abnormal gamma radiation levels in areas where radioactive material may be stored, handled or inadvertently introduced

b) warn plant personnel whenever abnormal concentrations of airborne radioactive materials exist

c) supplement other systems, including the process radiation monitoring system (Section 11.4) and the leak detection system (Section 5.2.4), in detecting abnormal migrations or accumulations of radioactive material

d) initiate a containment isolation signal in the event of abnormal radiation inside the containment

The area radiation monitoring system is a complex of radiation monitors and alarms which provide operating personnel with a continuous record of radiation levels and integrated airborne gamma activity concentrations at selected locations within the plant. The system assists in detecting unauthorized or inadvertent movement of radioactive material in the plant and in helping operating personnel decide on deployment of personnel in the event of an accident resulting in the release of radioactive material.

12.1.4.2 System Description

The area radiation monitoring system consists of 27 channels that are located at selected places in and around the plant to detect and record the radiation levels and, if necessary, annunciate abnormal conditions. The areas where the gamma monitors are located are shown in Table 12.1-7. Indication, annunciation and recording are all done in the control room. Indication and annunciation are also provided wherever there is a local indicator. The control room has readout modules for all 27 channels.

The gamma area radiation monitoring system is shown as a functional block diagram in Figure 12.1-9. A typical channel consists of a gamma sensitive halogen quenched Geiger Müller (GM) detector, an indicator and an alarm/trip unit, a power supply and a shared multipoint recorder. Some channels have, in addition, a local audio alarm auxiliary unit as indicated in Table 12.1-7.

Channels 2, 7, 8, 9, 12, 15, 19, 21 are provided with a remote alarm meter. This remote alarm unit consists of a five decade readout meter (10^{-1} mr/hr to 10^{4} mr/hr), a red high-high radiation light, and a high radiation horn which activates with the alarm light.

TABLE 12.1-7

AREA RADIATION MONITORING SYSTEM

Channel Number	Location and General Arrangement Figure Number	Range mr/hr	Local Meter and Alarm
1	Containment Annulus Figure 1.2.8	0.1-1,000	No
2	On Refueling Machine Figure 1.2-8	0.1-1,000	Yes
3*	In Containment Cooling System Ring Duct Header Figure 1.2-8	10-100,000	No
4*	In Containment Cooling System Ring Duct Header Figure 1.2-10	10-100,000	No
5*	In Containment Cooling System Ring Duct Header	10-10,000	No
6*	In Containment Cooling System Ring Duct Header Figure 1.2-8	10-100,000	No
7	Control Room Figure 1.2-17	0.1-1,000	Yes
8	H&V Room el 43.0 ft Figure 1.2-14 Column RA2-RAE	0.1-1,000	Yes
9	Hot Chemistry Lab Figure 1.2-13	0.1-1,000	Yes
10	Sample Room Figure 1.2-13	0.1-1,000	No
11	Figure 1.2-13, Column RAA-RA2 el 19.5 ft	0.1-1,000	No
12	Drumming Station Figure 1.2-13 Column RA2-16	0.0-1,000	Yes

* The asterisk indicates the area monitors that are classified as seismic Class I electrical equipment and supply the containment isolation signal (CIS). For further discussion refer to Section 7.3.

TABLE 12.1-7 (cont'd)

Channel Number	Location and General Arrangement Figure Number	Range mr/hr	Local Meter and Alarm
13	Boric Acid Storage Tank Figure 1.2-13 Column RAE-RA4 el 19.5 ft	0.1-1,000	No
14	Machine Shop Figure 1.2-13	01-1,000	No
15	ECCS Rooms A&B Figure 1.2-12	0.1-1,000	Yes
16	Containment Spray Pumps Figure 1.2-12	0.1-1,000	No
17	Charging Pumps Figure 1.2-12	0.1-1,000	No
18	Gas Decay Tank Figure 1.2-12	0.1-1,000	No
19	Main Corridor el 0.5 ft at Columns RAK-RA4 Figure 1.2-12	0.1-1,000	Yes
20	Pipe Penetration Room el 19.50 ft Figure 1.2-13	0.1-1,000	No
21	Spent Fuel Handling Machine Figure 1.2-18	0.1-1,000	Yes
22	New Fuel Storage Area Fuel handling bldg	0.1-1,000	No
23	Decontamination Area Figure 1.2-18	0.1-1,000	No
24	NE Corner of Site Figure 1.2-2	0.1-1,000	No
25	NW Corner of Site Figure 1.2-2	0.1-1,000	No
26	SW Corner of Site Figure 1.2-2	0.1-1,000	No
27	SE Corner of Site	0.1-1,000	No

The log readout module associated with these channels has a similar dynamic range of five decades. It is provided with a 0-10 mv recorder output, a 0-50 mv computer output and a meter output required to operate field mounted indicators.

Channels 3, 4, 5, 6 are provided with a 4-20 ma dc output for containment isolation. Moreover, they are protected to continuously operate under a direct spray solution containing 0.97 percent boric acid.

12.1.4.3 Design Evaluation

The system provides five full decades (10^{-1} mr/hr to 10^{4} mr/hr) of dynamic range. This system provides the accuracy and reliability of a pulse-counting circuit, without the necessity of compressing the fifth decade, or operating in a combination pulse-counting and current-sensing mode, to achieve the five decade dynamic range. The energy response has been empirically determined by the manufacturer to be ± 15 percent from 100 Kev to 1.5 Mev. The halogen quenched GM detector is mounted inside of an aluminum casting which provides safe operation of the detector in external pressure in excess of 40 psig. This detector is provided with a live zero to preclude spurious fail alarms when the background radiation level in which the detector is operating is less than 0.1 mr/hr. Natural background radiation and check source contribution hold the system in the non-failure condition. If for any reason the system stops responding to radiation a failure condition will be indicated on the readout module. A built-in anti-saturation circuit prevents the system readings from falling off full scale during overrange conditions.

To prevent a loss of monitoring of an entire area of the plant due to a loss of power, each of the 27 channels is connected to a separate power supply in conformance with the AEC non-common mode failure criteria. Except for the containment area radiation monitors which initiate the containment isolation signal (CIS), all other monitors have power supplies fed from an interruptible source and are inoperative during a loss of off-site power for a maximum period of 30 seconds until the diesel generator sets start. The four containment area radiation monitors which initiate the CIS are fed from four seismic Class I power supplies (125v dc system).

These four containment area radiation monitors identified by channel numbers 3, 4, 5 and 6 in Table 12.1-7 are designed as seismic Class I and can withstand LOCA conditions including operation under direct containment spray containing 0.97 percent boric acid for a period of at least 15 minutes after an accident. See Section 3.11 for discussion of the environmental qualification of these components. All other area radiation monitors are designed to withstand the environmental and power supply conditions shown in Table 12.1-8

These four containment monitors are physically and electrically separated from each other in accordance with the criteria at forth in IEEE 279, August 1968, and IEEE 308, November 1970. The output signals from the monitors (labeled MA, MB, MC and MD) feed the engineered safety features logic to make up the CIS. The four monitors are positioned in sets of

TABLE 12.1-8

AREA RADIATION MONITORING
ENVIRONMENTAL AND POWER
SUPPLY DESIGN CONDITIONS

PARAMETER	AT SENSOR LOCATION	EQUIPMENT IN CONTROL ROOM
TEMPERATURE	-20 F -140 F 264 F*	32 F -120 F
HUMIDITY	0 - 100%	0 - 95%
RADIATION	10^5 RADS 10^7*	———
POWER	———	115 V 60 cps
PRESSURE	30 psig 40 psig*	———

* For Channels 3, 4, 5, 6, under LOCA conditions

two 180^{o} apart inside the containment cooling system ring header duct to assure homogeneity of the monitored atmosphere and to preclude the possibility of spurious actuation due to direct radiation from equipment. The air flow through the ring header duct during normal operation is 180,000 cfm. The radiation monitors level alarm is set at 10 R/hr. A setpoint of 10 R/hr was selected based on response time for the high containment pressure setpoint of 5 psig since both signals feed the CIS circuitry.

The readout module incorporates three alarms as follows:

a) Alert Alarm: A white, push-button light goes "ON" and an alarm sounds when the radiation level exceeds a preset point. The level is adjustable over the entire range. Depressing the push-button light causes the meter to indicate where the alarm level has been set;

b) High Alarm: A red, push-button light, which is identical to the alert alarm described above, except the setting is higher;

c) Failure/Reset Alarm: An amber, push-button light, which remains "ON" when the system is in normal operation, and goes "OFF" if high voltage, or line voltage fails. When this push-button light is depressed, it automatically resets the alert and high alarms.

Two checking systems are provided to check the integrity of the area radiation monitoring system; a continuous one which is described above as a failure/reset alarm, and an on-command system which activates a check source into position. This causes an up-scale reading verifying system integrity.

The integral check source provided with this detector assembly consists of approximately 1 uCi of depleted uranium. This source is activated from the readout module located on the control room radiation monitoring panel.

The environmental and power supply design conditions for the monitors are given in Table 6.2-8.

12.1.4.4 Testing and Inspection

An internal trip test circuit adjustable over the full range of the trip circuit is a design feature. The test signal is fed into the indicator and trip unit so that a meter reading is provided in addition to a real trip. A check source that is remotely controlled from the control room is an integral part of each area radiation monitoring channel.

The function of the containment area monitors is discussed in Section 7.3. Environmental qualifications of the containment area radiation monitors is discussed in Section 3.11.

12.1.5 OPERATING PROCEDURES

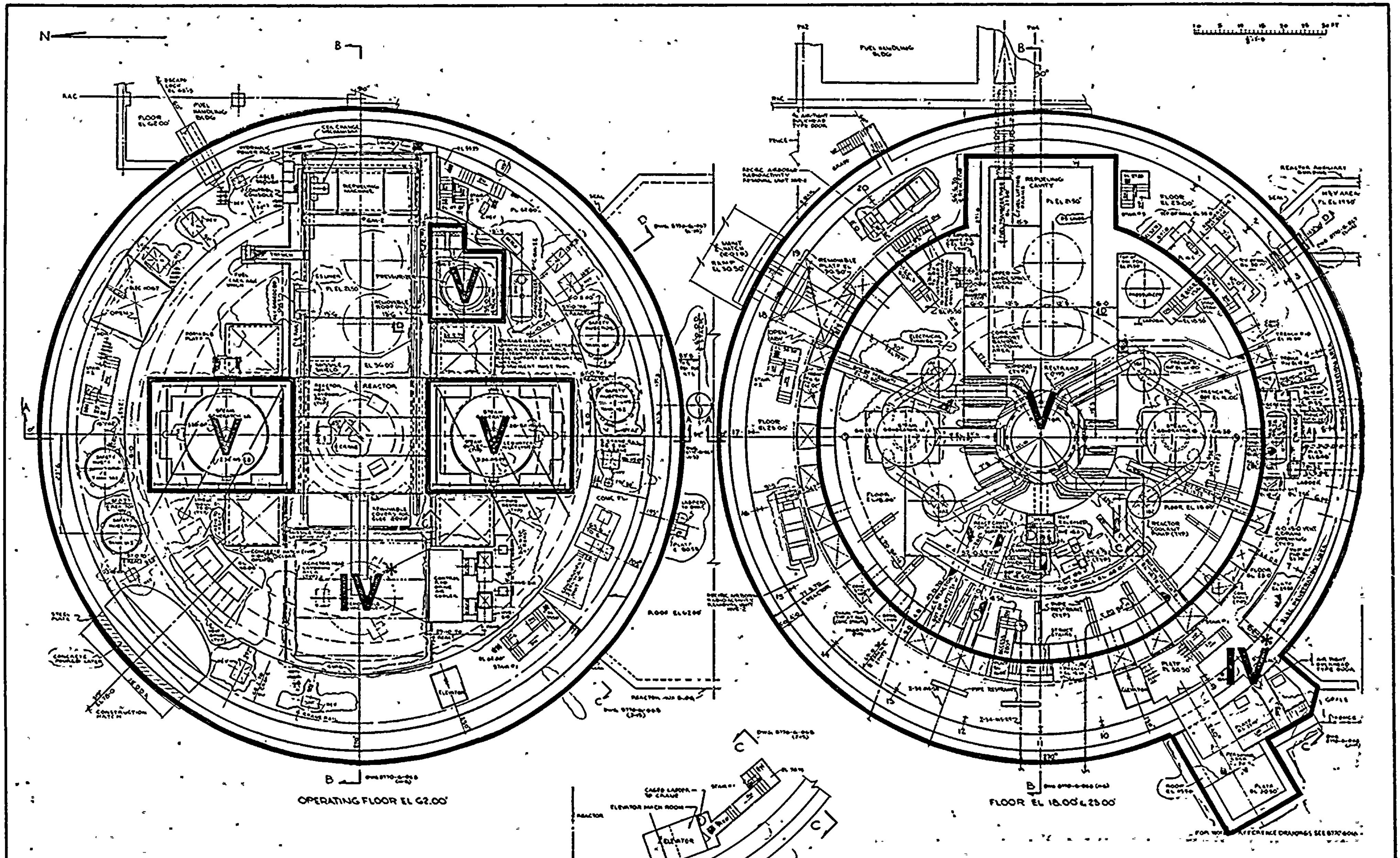
Permanent shielding is installed wherever radioactive sources are expected to occur and wherever operating and maintenance functions are normally required. Both regular and special temporary radiation shields are available for use in all normal operational situations where radioactivity containing equipment must be transported or repaired. Time-radiation dose schedules will be developed and followed to formalize operations to be executed in unusually high gamma radiation fields in such a manner that exposure to operating and maintenance personnel are minimized.

12.1.6 ESTIMATES OF EXPOSURE

The expected maximum dose rate levels for various in-plant locations are listed below. The zones correspond to those shown in Figures 12.1-1 through 12.1-5 but the dose rates vary since the expected values are based on 0.1 percent failed fuel. During normal power operation, Zones IV and V areas within the containment are not represented by the values listed below since the dose rates for those areas are determined by N-16 activity and not percentage of failed fuel.

Zone	Dose Rate (mr/hr)
I	<.05
II	.05 ⟶ .25
III	.25 ⟶ 1.5
IV	1.5 ⟶ 10.0
V	>10.0

The physical models used to determine the shielding required are approximations to the physical layout, i.e., cylindrical sources were used to approximate tanks, line sources for pipes and so forth. The methodology outlined in T. Rockwell III, "Reactor Shielding Design Manual," are employed to determine dose rates from the modelled equivalent sources. Conservative assumptions were made in the shielding analysis such as: waste tanks completely full; neglect wall thickness of some pipes and neglect any air attenuation. Source data utilized in the calculation of shielding thicknesses are given in Section 12.1.3.



DIRECT RADIATION DOSE LEVELS
NORMAL OPERATION

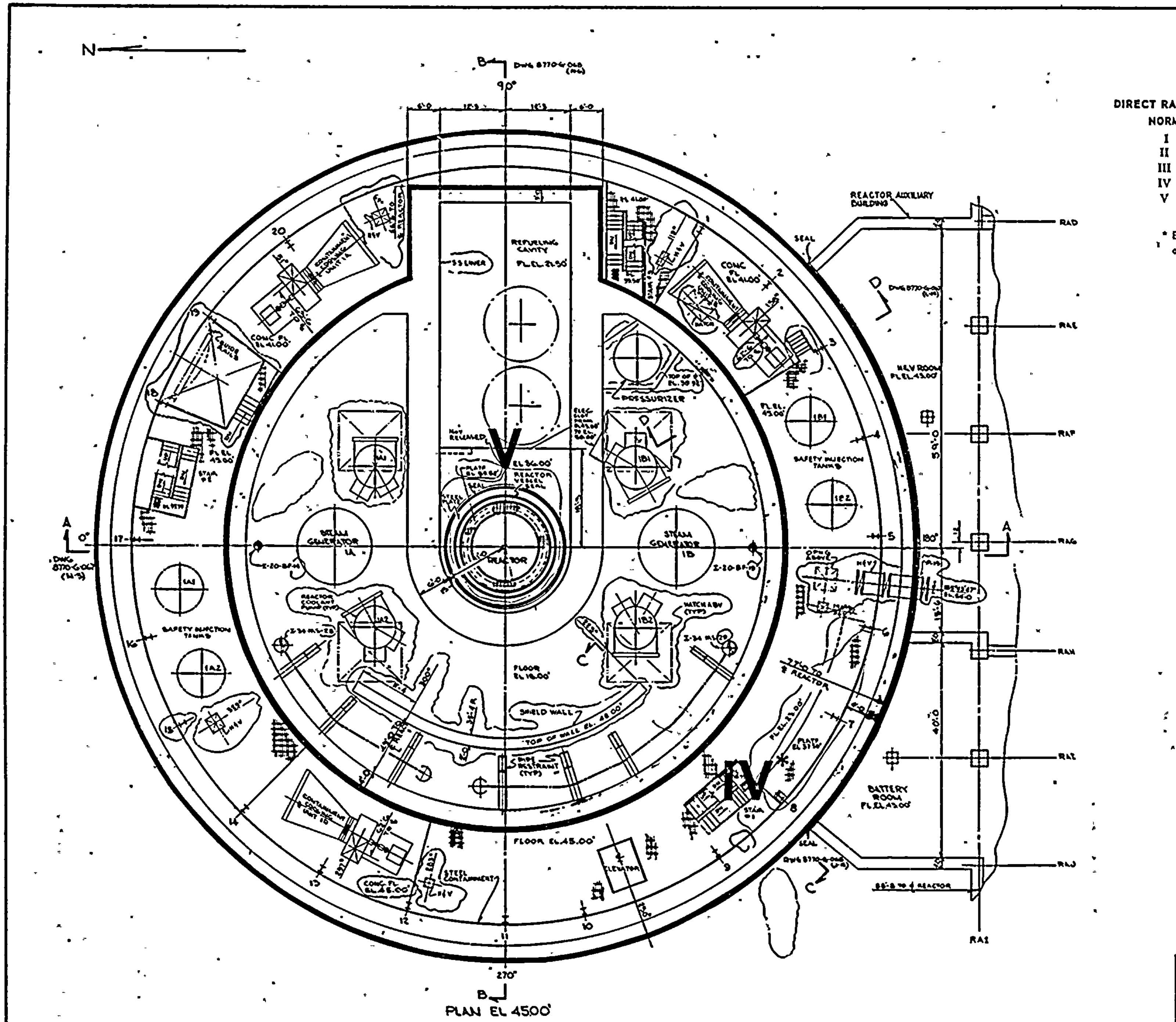
I ≤0.5 MR/HR
II 0.5-2.5
III 2.5-15.0
IV. 15.0-100
V >100

* Becomes III 8 Hrs. after shutdown.

8770-G-065

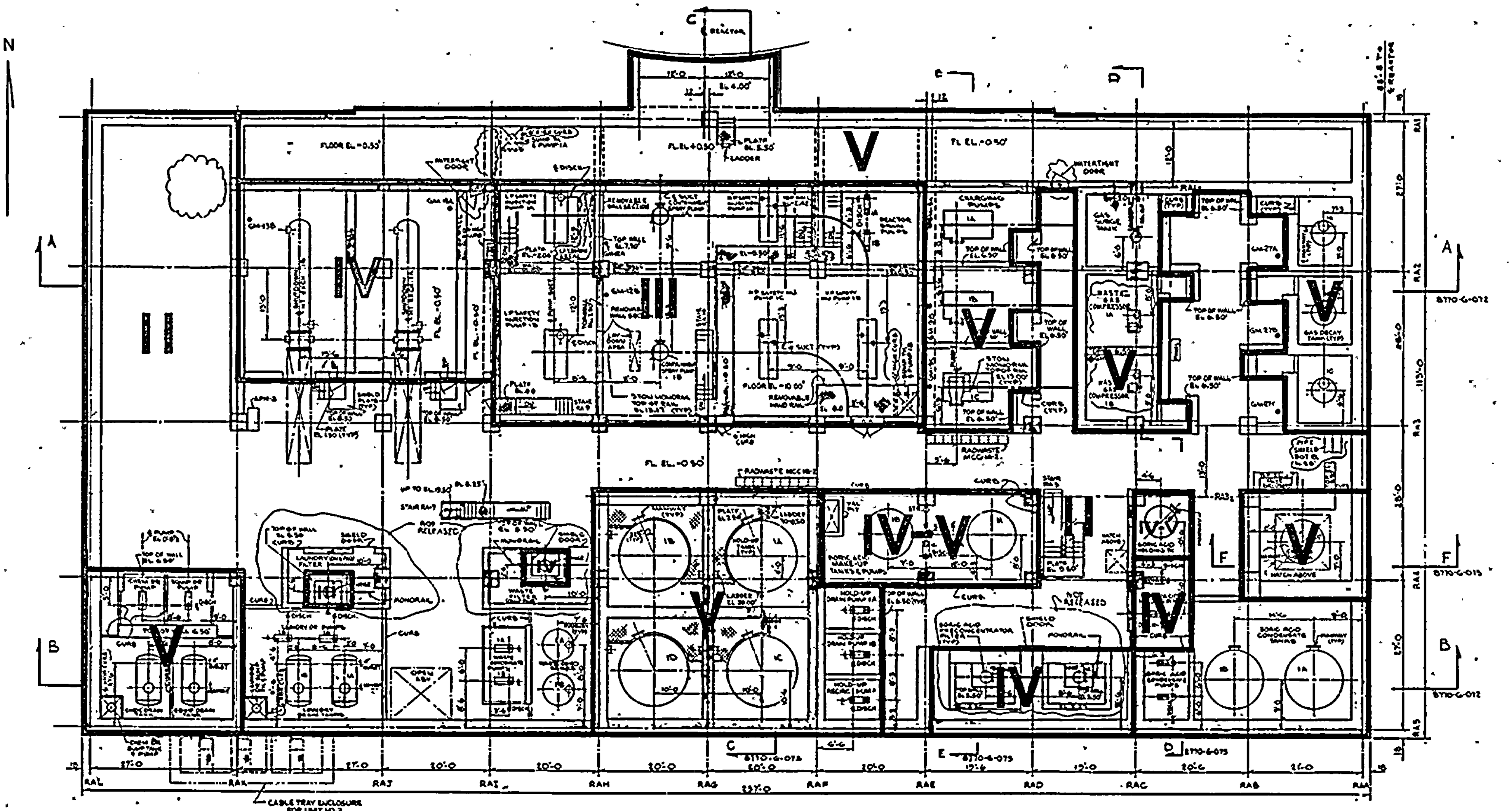
FLORIDA POWER & LIGHT COMPANY
Hutchinson Island Plant

MAXIMUM DOSE RATE LEVELS – REACTOR CONTAINMENT BUILDING FLOOR ELEVATION 18', 23' and 62'
FIGURE 12.1-1



FLORIDA POWER & LIGHT COMPANY
Hutchinson Island Plant

MAXIMUM DOSE RATE LEVELS – REACTOR CONTAINMENT BUILDING FLOOR ELEVATION 45'
FIGURE 12.1-2



PLAN EL.-0.50

LEGEND

● GM-GEIGER-MULLER MONITOR
APM-AIR PARTICLE MONITOR

DIRECT RADIATION DOSE LEVELS
NORMAL OPERATION

I	$\leq$0.5 MR/HR
II	0.5-2.5
III	2.5-15.0
IV	15.0-100
V	> 100

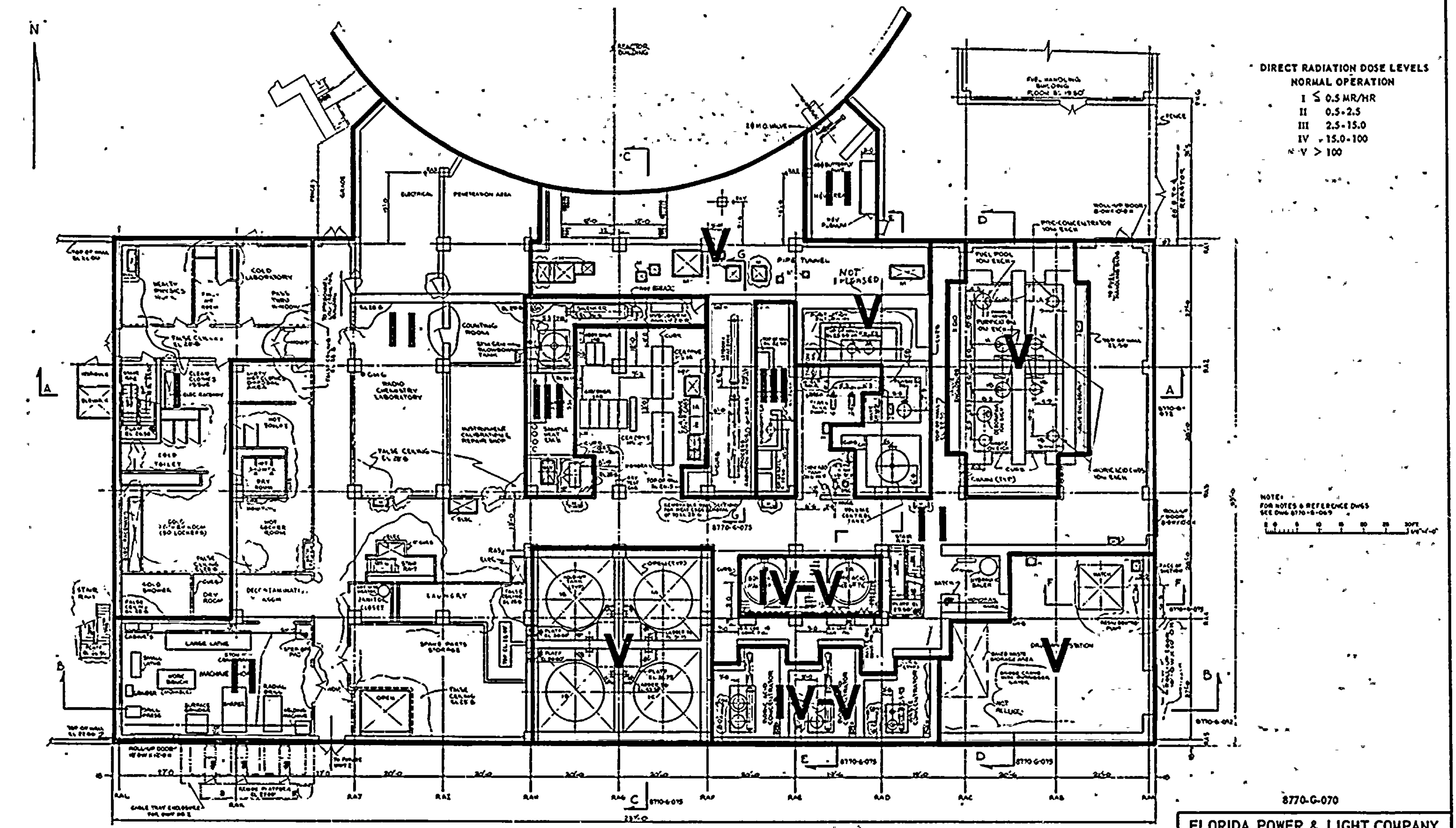
REFERENCE DRAWINGS

LIST OF DRAWINGS	8770-G-146
SITE PLAN	8770-G-056
ENLARGED PLOT PLAN	8770-G-059
GEN. ARR.-REACTOR AUXILIARY BUILDING-PLAN-SH 2	8770-G-070
GEN. ARR.-REACTOR AUXILIARY BUILDING-PLAN-SH 3	8770-G-071
GEN. ARR.-REACTOR AUXILIARY BUILDING-SECTIONS-SH 1	8770-G-072
GEN. ARR.-REACTOR AUXILIARY BUILDING-SECTIONS-SH 2	8770-G-073
GEN. ARR.-REACTOR AUXILIARY BUILDING-MISC. PLANS (SECT.	8770-G-076

8770-G-069

FLORIDA POWER & LIGHT COMPANY
Hutchinson Island Plant

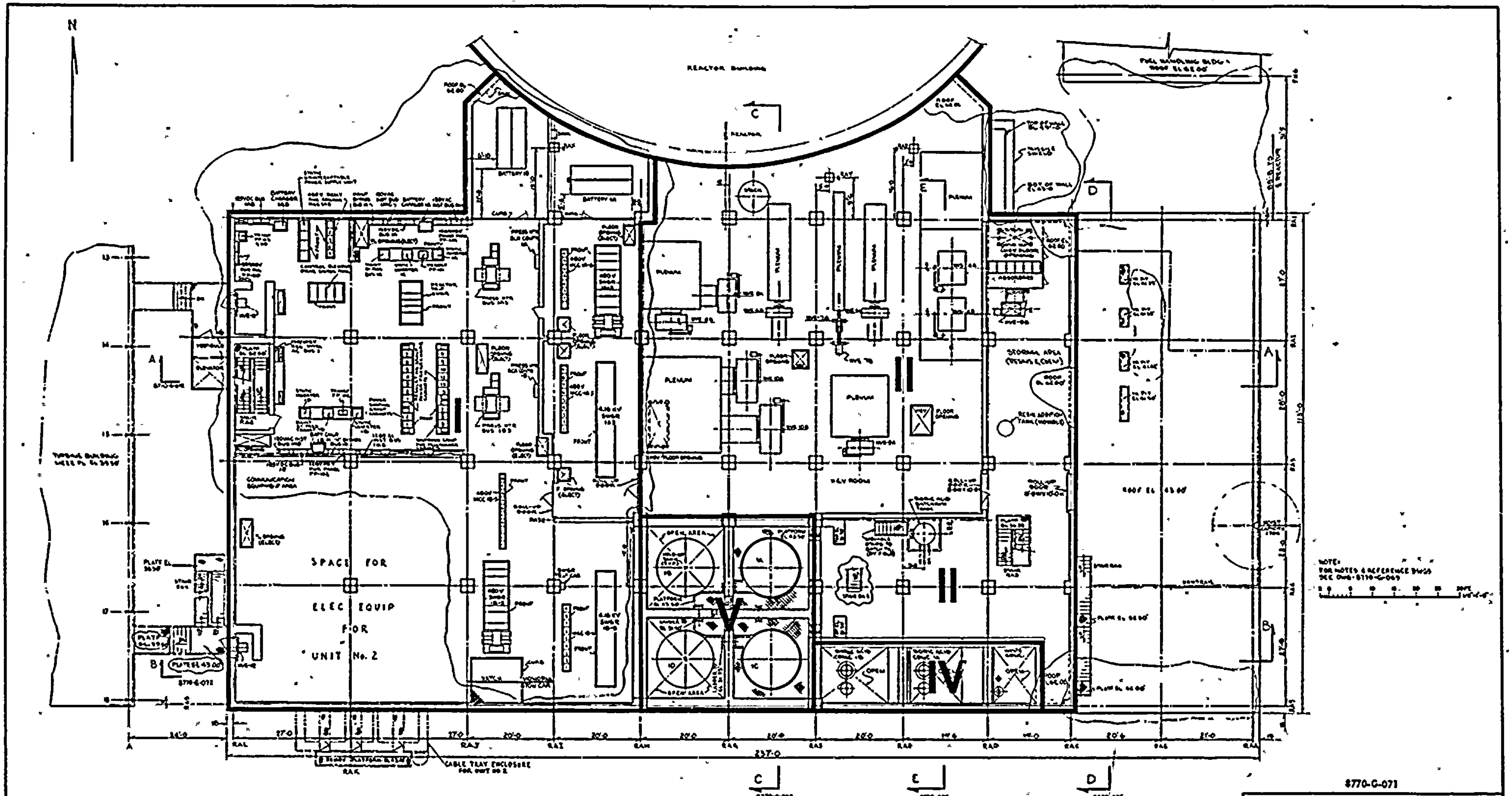
MAXIMUM DOSE RATE LEVELS - REACTOR AUXILIARY BUILDING FLOOR ELEVATION - 0.5'
FIGURE 12.1-3



FLORIDA POWER & LIGHT COMPANY
Hutchinson Island Plant

MAXIMUM DOSE RATE LEVELS – REACTOR AUXILIARY BUILDING FLOOR ELEVATION 19.5'

FIGURE 12.1-4

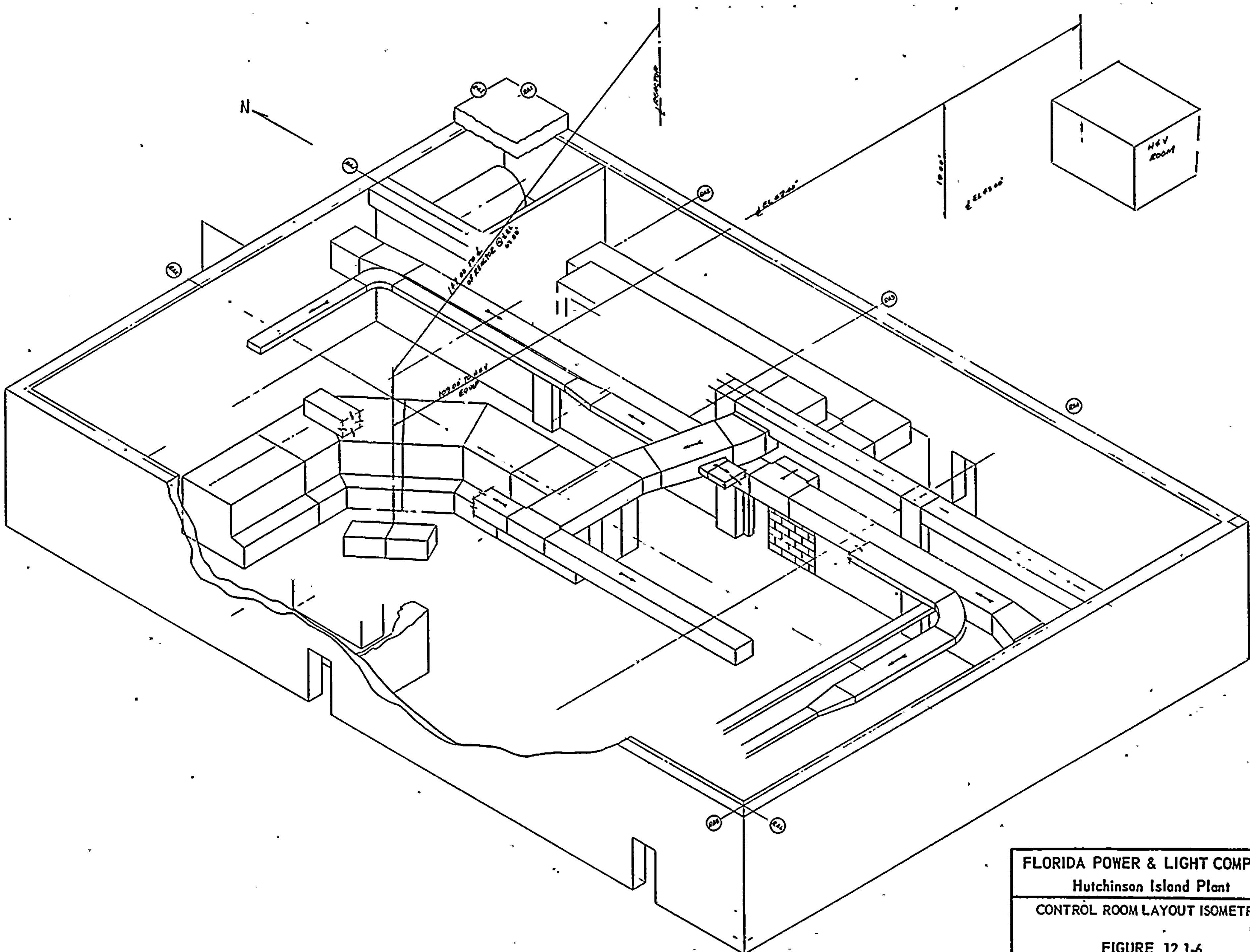


DIRECT RADIATION DOSE LEVELS NORMAL OPERATION

I ≤0.5 MR/HR
II 0.5-2.5
III 2.5-15.0
IV 15.0-100
V >100

FLORIDA POWER & LIGHT COMPANY
Hutchinson Island Plant

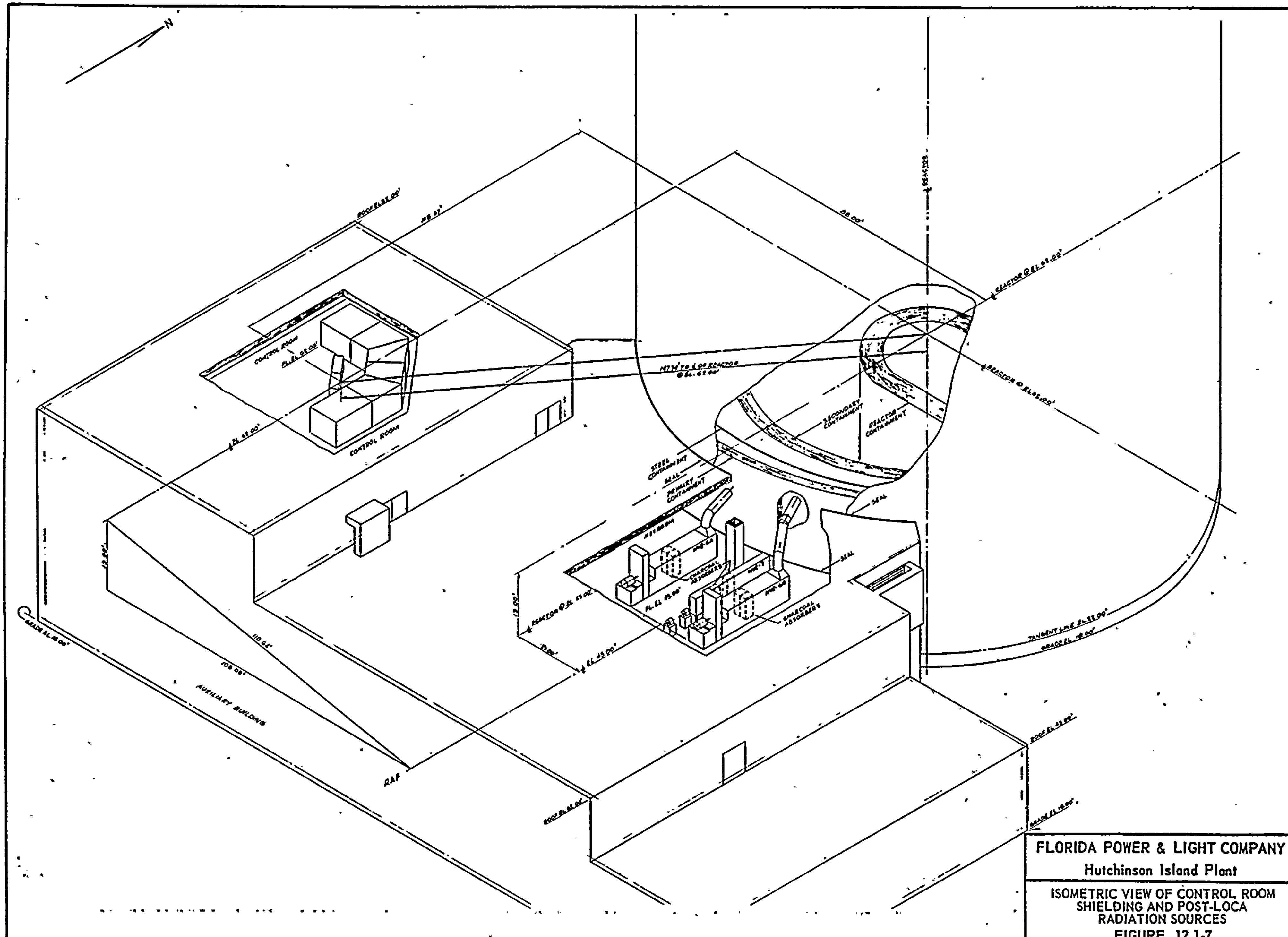
MAXIMUM DOSE RATE LEVELS –
REACTOR AUXILIARY BUILDING
FLOOR ELEVATION 43'
FIGURE 12.1-5



FLORIDA POWER & LIGHT COMPANY
Hutchinson Island Plant

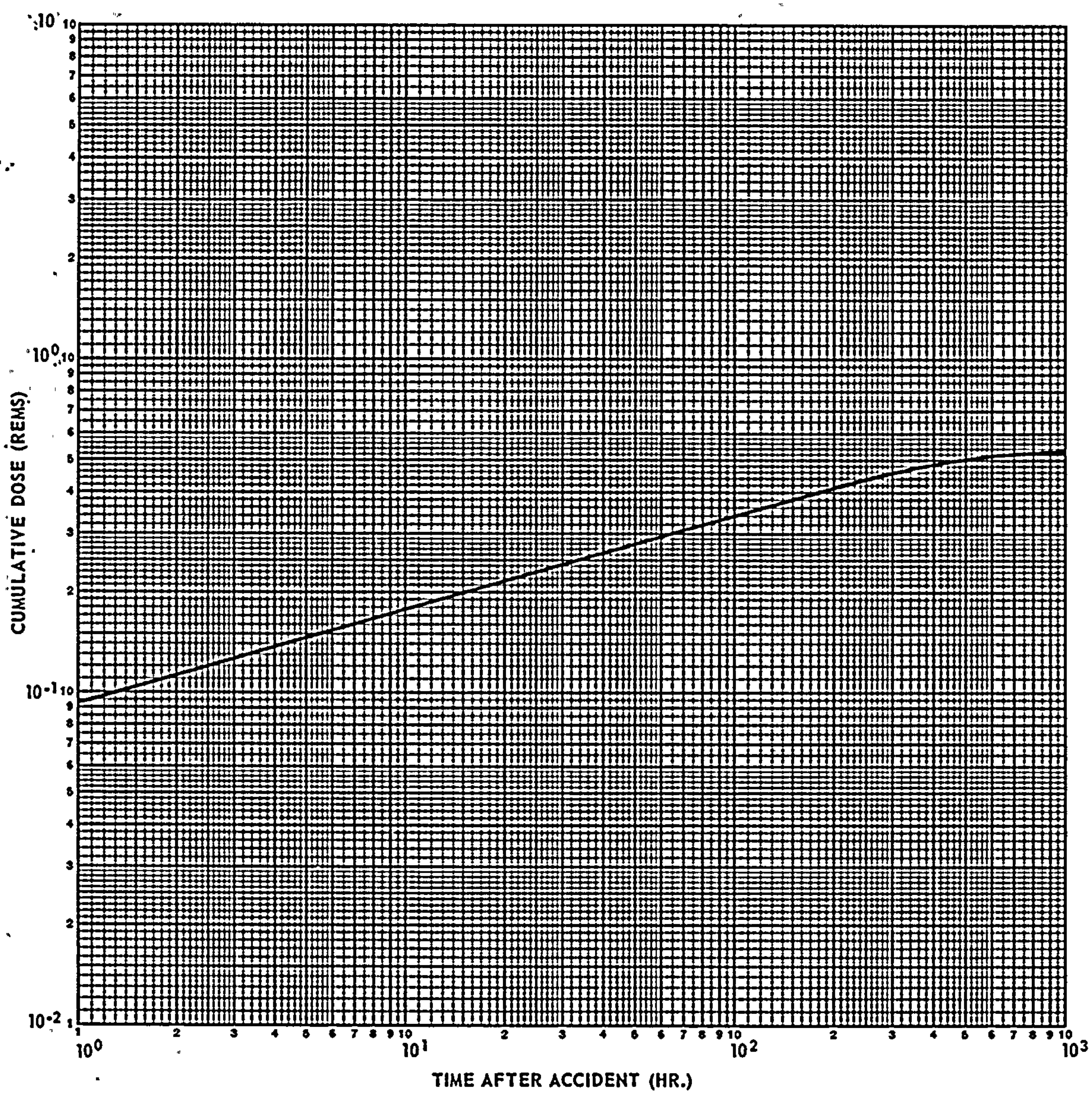
CONTROL ROOM LAYOUT ISOMETRIC

FIGURE 12.1-6



FLORIDA POWER & LIGHT COMPANY
Hutchinson Island Plant

ISOMETRIC VIEW OF CONTROL ROOM SHIELDING AND POST-LOCA RADIATION SOURCES
FIGURE 12.1-7

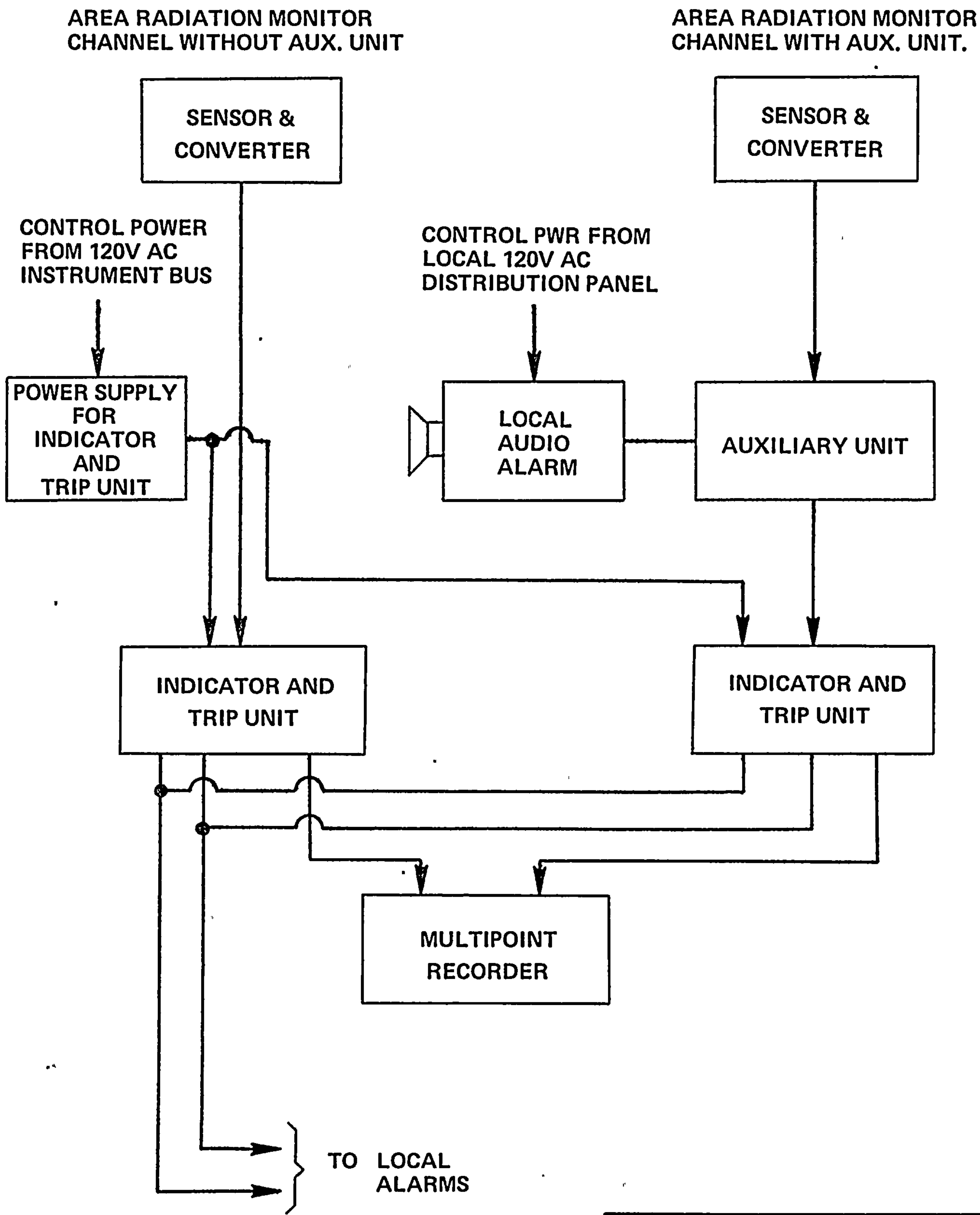


FLORIDA POWER & LIGHT COMPANY

Hutchinson Island Plant

CUMULATIVE POST-MHA WHOLE BODY DOSE IN THE CONTROL ROOM

FIGURE 12.1-8



FLORIDA POWER & LIGHT COMPANY

HUTCHINSON ISLAND PLANT
UNITS 1 & 2

AREA RADIATION MONITORING SYSTEM
FUNCTIONAL BLOCK DIAGRAM
FIGURE 12.1-9

12.2 VENTILATION

12.2.1 DESIGN OBJECTIVES

The design objectives for the plant ventilation systems as they relate to radiation protection are to:

a) Limit the buildings of airborne radioactivity to levels that will permit access required for operation, maintenance, inspection and testing without exceeding the requirements of 10CFR20.

b) Assure migration of air toward areas having sources of airborne radioactivity.

Table 12.2-1 summarizes the areas subjected to contamination by airborne radioactivity and shows the effect of ventilation system operation on the summation of the ratios of nuclide activity, (C), to their respective maximum permissible concentrations, (MPC), as set forth in 10CFR20, Appendix B.

The information developed is based on Table 11.1-1 entitled "Reactor Coolant Activities." Table 11.1-1 is based on 0.1 percent failed fuel.

12.2.2 DESIGN DECRIPTION

12.2.2.1 General

Charcoal adsorbers, where referred to in the system descriptions, consist of banks composed of tray type units rated at 333 cfm capacity each. Trays are of stainless steel construction and consist of two carbon beds of two inch thickness, separated by an interspace.

The charcoal is iodine impregnated with the capability of trapping a minimum of 99.9 percent of the iodines at accident concentrations with 5 percent as methyl iodide when operating at 70 percent relative humidity and 150 F. Test clips are provided for periodic testing to monitor charcoal for aging effects. Freon tests are periodically applied in accordance with USAEC Report DP-870 to assure minimum in-place leak test efficiencies of 99.9 percent.

High efficiency particulate (HEPA) air filters, where referred to, consist of banks assembled in 24" wide x 24" high x 11-1/2" deep extended media dry type units. Each unit is rated at 1000 cfm. Filters are fabricated from a continuous sheet of asbestos or glass-asbestos media, closely pleated over aluminum separators, and have a minimum efficiency of 99.97 percent when tested with thermally generated DOP droplets of uniform 0.3 micron size. Filters located within the containment are provided with composition-asbestos separators in lieu of aluminum. Periodic in-place leak tests are applied, using DOP aerosol to assure filter integrity.

Medium efficiency filters, where referred to, are of the dry-extended media type, having an average National Bureau of Standards atmospheric

TABLE 12.2-1

AREAS WITH AIRBORNE RADIOACTIVITY SOURCES

Area(1)	Air Flow (cfm)	Leak Rate (gpd)	Source(3) Strength Factor	$\sum$C/MPC	Maximum Whole Body Dose from Internal Exposure(2) (rem/year)
AUXILIARY BUILDING, ELEVATION -0.5'					
Reactor Drain Pumps	10,000	1.0	1.0	4.17 (-2)	2.09 (-1)
Charging Pumps	6,000	5.0	1.0	3.48 (-1)	1.74
Spent Resin Tank	225	0.01	50.0	9.26 (-1)	4.63
Boric Acid Holding Tank and Pumps	1,025	0.01	50.0	2.03 (-1)	1.02
Boric Acid Condensate Tanks	1,500	0.01	0.01	2.70 (-5)	1.35 (-4)
Boric Acid Preconcentrator Filter	300	0.01	0.1	1.39 (-2)	6.95 (-2)
Holdup Drain Pumps	1,500	0.1	50.0	1.39	6.95
Chemical Drain Tank	1,200	5.0	0.1	1.74 (-1)	8.68 (-1)
Remaining Areas	6,410	1.0	0.1	6.50 (-3)	3.25 (-2)
AUXILIARY BUILDING, ELEVATION +19.5'					
Radio Chemistry Lab. and Counting Room	1,000	0.1	1.0	4.17 (-2)	2.09 (-1)
Sample HX and H_2 Analyzer	1,525	1.0	0.01	2.73 (-2)	1.36 (-1)
Boronometer	475	0.1	1.0	8.77 (-2)	4.38 (-1)
Purificaton Filter, Flash Tank and Flash Tank Pumps	2,200	0.1	1.0	1.89 (-2)	9.45 (-2)
Fuel Pool Ion Exch., Boric Acid Condensate Ion Exch. Deborating Exch.	800	0.1	50.0	2.606	13.0
Drumming Station	2,250	2.0	10.0	3.706	18.5
Boric Acid and Waste Concentrator	1,800	0.1	50.0	1.158	5.78
Boric Acid Makeup Tank and Pumps	2,100	0.1	50.0	9.92 (-1)	4.96
Holdup Tanks	2,800	0.1	50.0	7.44 (-1)	3.72

(1) Refer to Figures 12.1-3, 12.1-4 and 12.1-5 for radiation zone classification for access.
(2) Assumes continuous exposure for a 40 hr week, 50 weeks per year.
(3) Ratio of nuclide radioactivity concentrations of leak source to those of the reactor coolant.

type test efficiency of 35 percent. Filter units are 2 in thick and either 20" high x 20" wide or 24" high x 24" wide.

12.2.2.2 Containment Airborne Radioactivity Removal System

The containment airborne radioactivity removal system consists of two units as shown on Figures 9.4-1 and 9.4-3. The system is designed to remove airborne radioactivity in particulate and iodine form by recirculating containment atmosphere through high efficiency filters and charcoal adsorbers. The system is used for radioactivity removal during normal operation only and serves no function for post-LOCA dose reduction. Each of the two units located at elevation 23' includes a suction connection from the containment cooling system ring duct header located at elevation 98'; a bank HEPA air filters, 2 units wide by 5 units high; a bank of unit tray type charcoal adsorbers, 2 units wide by 15 units high; and a belt driven, single width single inlet, freely discharging centrifugal fan. Fan and filters for each of the two units are rated at 10,000 cfm with a fan static pressure of 6 in. wg which permits rated operation with fully loaded filters.

The containment airborne radioactivity removal system is manually energized from the control room. Charcoal adsorber temperature and HEPA filter pressure drop are annunciated and monitored in the control room. Low flow annunciation is also provided.

All active components of the system are suitable for operation in 120 F ambient temperature and 1 rad/hr of radiation. Ambient temperatures are limited to a maximum of 120 F by normal operation of three containment fan cooling units.

System design and calculational evaluation is based on normal plant operation only, assuming a 1200 lb/day reactor coolant leakage into the containment. Specific equilibrium activities of the coolant are listed in Table 11.1-1. Other assumptions made in the calculation model are:

a) Failed fuel is 0.1 percent.

b) Iodine partition coefficient is 0.10.

c) 100 percent of all noble gases become airborne at the leakage or generation rate.

d) Tritium, which is in water molecular form, concentrates in the atmosphere up to the level determined by the specific humidity (lb of water vapor per pound of dry air at 120 F dry bulb and 50 percent relative humidity).

e) Mixing efficiency of all airborne constituents is 90 percent.

f) Because a quantity of leakage remains in liquid form, there is little opportunity, barring liquid surface disturbances, atomization or residue following complete evaporation, for particulates to become airborne. Therefore, the assumption of all particulates being airborne is conservative.

After 90 days with the assumed leakage rate of 1200 lb/day of reactor coolant into the containment with all airborne radiation cleanup systems inactive, the sum of C/MPC for all nuclides is 1860. At this time, the sum of all nuclide activities is 0.00193 Ci/cc. Iodines alone amount to 647 MPC, particulates amount to 1142 MPC, and noble gases including tritium amount to 71 MPC. Pulldown time for the airborne radioactivity removal system is closely a function of air change rate which, when both units are operating, is 0.48 containment net free volume air changes per hour. With this capacity and following 26 hours of system operation, the gross C/MPC level is reduced to a minimum of 80.5, with gross activity at 0.00185 μCi/cc. However, within 10 hours, 99 percent of the total pulldown from 1860 to 71 MPC hass occurred. At minimum MPC level, iodine, particulate and noble gas concentrations in MPC's have been reduced to 7.5, 0.7, and 71.3 respectively.

12.2.2.3 Containment Purge System

The containment purge system, which exhausts the containment atmosphere to the environment, is rated at 42,000 cfm and is operated following reduction of iodine and particulate activity by the containment airborne radioactivity removal system. The system is shown on Figures 9.4-1 and 9.4-3. It serves to further reduce the residual iodine and particulate activity as well as to reduce the activity of the nonfilterable noble gases and tritium. Where only short term access to the containment is required, the system is not operated. For extended time, as on a week-end shutdown, the purge exhaust system is used. Although it is likely that the reactor will be subcritical at these times, the calculational model conservatively assumes full power operation.

The suction side of the purge system is connected through a 48" x 48" duct to the containment cooling system ring duct header to assure uniform purging of the containment. A 36" x 14" branch duct is connected through an automatic damper to forty air inlets located above the water line in the refueling cavity. This exhaust branch is activated during refueling. During a normal nonrefueling purge, the containment air from the ring header is drawn through butterfly isolating valves FCV-25-4,5 and 6 into a filter casing that is common to two belt-driven, single width single inlet, top annular upblast fans that discharge to the plant vent. Each fan is rated a 42000 cfm and 9 in. wg static pressure.

The filter casing and high efficiency filter mounting frames, common to both fans, are of all welded construction and include, in the direction of flow, a set of medium efficiency prefilters and a bank of HEPA filters, 7 units wide by 6 units high.

The air makeup side of the purge system includes, in the direction of flow, a 12' wide x 10' high air intake louver, a bank of medium efficiency filters, 6 wide by 5 high, and three 48 in. diameter butterfly type isolating valves, designated V-25-1,-2 and -3.

The system is manually energized from the control room. When the switch is moved to the "start" position, exhaust butterfly valves FCV-25-4,-5,

and -6 open and through valve limit switches start a fan. Fan motor starter interlocks and a differential pressure switch permit opening of makeup air butterfly valves FCV-25-1, -2, and -3 only when a slightly negative pressure differential has been established in the containment. This prevents unfiltered blowback through the makeup air valves. The purge fans are designed to trip off if a high containment vacuum condition occurs. All containment purge system isolation valves (FCV-25-1, -2,-3,-4,-5 and -6) close automatically on activation of the containment isolation signal.

Calculational assumptions noted for the containment airborne radioactivity removal system apply as well to the purge system, which may be energized following radioactivity cleanup. The final values for MPC and activity concentration resulting from the recirculation phase of cleanup (i.e, the airborne radioactivity removal system) are used as the initial input to the computer calculation for the purge phase cleanup.

The result is that after operation of the airborne radioactivity removal system and the purge system, the sum of C/MPC for all nuclides is equal to 3.73. This corresponds to a gross activity level of 0.108×10^{-4} μCi/cc. Iodines alone amount to 2.3 MPC, particulates amount to 0.9 MPC, and noble gases including tritium amount to 0.5 MPC. Approximately 1 containment volume per hour is processed by the purge system. This permits 99 percent of the total cleanup to the above levels to occur after 9.6 hours of operation.

12.2.2.4 Reactor Auxiliary Building Main Ventilation System

The reactor auxiliary building main ventilation system is of the nonrecirculating type and consists of a conventional central air supply system with a Class I seismic rating and a nonseismic Class I filtered exhaust. The system is shown on Figure 9.4-1.

The air supply system includes a screened outside air intake louver, a bank of medium efficiency filters, steam heating coils, and two belt-driven fans that discharge into an air duct distribution system. The fans are double width double inlet, top angular upblast type. Supply fans HVS 4A and 4B are each rated for full system flow which is 69,000 cfm and 3 in. wg. static pressure. A system of ductwork exhausts areas having sources of airborne radioactivity at a rate that is sufficient to reduce activity to levels permitting required weekly access during power operation. Refer to Table 12.2-1 for a listing of these areas. In most cases, the air flow required to limit temperature rise exceeds radioactivity control requirements.

The main exhaust system includes exhaust ductwork to the main plenum; a bank of 12 wide by 6 high medium efficiency prefilters; a bank of HEPA filters 12 wide by 6 high; and two belt-driven, single width single inlet, upblast centrifugal fans (HVE-10A and 10B) which discharge to the plant vent through backdraft dampers. Each of the two exhaust fans are rated at full system capacity of 71,600 cfm and 7 in. wg static pressure. The exhaust system is not required to operate following an accident, this function being performed by emergency exhaust system fans HVE 9A and 9B.

Supply and exhaust fans, except for HVE 9A and 9B, are manually energized from the control room.

Air flow stoppage is annunciated by means of flow switches located in the discharge of each fan. HEPA filter pressure drop is indicated in the control room and annunciated when in excess of 3 in. wg.

Other areas within the reactor auxiliary building that are free of airborne radioactivity contaminating sources are exhausted directly to the environment. Such areas include laboratories, offices, electrical equipment areas, cold locker rooms, battery rooms, cold shower, cold area toilet, clean clothes issue room and cable enclosure.

The total volume of the reactor auxiliary building is 978,770 cu ft of which 638,370 cu ft are areas exhausted to the plant vent.

In most cases the minimum air change rate in any space within the reactor auxiliary building is 4 volumes per hour. The overall air change rate for portions of the building exhausted to the plant vent is 6.73 volumes per hous. Air change rates for specific areas having sources of airborne contamination are listed in Table 12.2-1.

12.2.2.5 Fuel Handling Building Ventilation System

The fuel handling building ventilation system reduces plant personnel doses due to potential airborne activity resulting from diffusion of fission products through the fuel pool water. The system consists of (a) a separate air supply and air exhaust network serving the fuel pool area and (b) a separate air supply and air exhaust network serving the fuel pool equipment areas and the new fuel storage area. The system is shown on Figures 9.4-1 and 9.4-2.

Each air supply system includes a wall intake, disposable roughing filters, 15 psi steam heating coils, and a cabinet type centrifugal fan. The first supply system is rated at 9800 cfm; the second supply system is rated at 8300 cfm. These supply systems provide hourly air changes of 6.8 volumes and 4.7 volumes for the fuel pool and equipment areas, respectively.

The fuel pool is exhausted by 36, 8" x 4" air inlet ducts spaced 4'-0" on centers around the pool periphery and 1'-0" above the pool surface. Four such inlet ducts are located 8'-0" on centers in the south wall of the refueling canal. The fuel pool exhaust system includes the concrete-embedded welded steel pool exhaust ducts; the duct connection to the filter casing; an all welded filter casing that includes a bank of medium efficiency filters, 9 wide x 3 high, and a bank of HEPA filters 3 wide x 3 high; and two belt-driven, single width single inlet, upblast fans. Each fan is rated to deliver full system airflow of 10,000 cfm at 6 in. wg. static pressure which corresponds to an average velocity of 1250 ft/min through the 36 fuel pool exhausts. Discharge is to a concrete penthouse with a 7'-0" wide x 8'-0" high fixed louver located on its north face.

Air is supplied to the fuel pool at a rate of 9800 cfm from 10'-0" above the 62'-0" floor elevation forcing migration of all air downward to the fuel pool surface. The amount of air supplied, being less than that exhausted, prevents out-leakage from the pool area to the equipment spaces within the building and to the environment.

The fuel pool equipment areas are exhausted through a filter system of the same arrangement used in the fuel pool exhaust system. A single width single inlet, upblast centrifugal fan exhausts the filter casing and discharges into the concrete penthouse. The fan is rated for 8100 cfm flow and 4 in. wg. static pressure.

12.2.2.6 Control Room Ventilation System

The design of the control room reflects the limits of acceptable exposures to plant personnel as documented in 10CFR100 for normal and emergency operation. General Design Criterion 19 stipulates that radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposure in excess of 5 rem whole body, or its equivalent, to any part of the body for the duration of the accident. As stated in Section 12.1.1, the control room shielding is designed to limit the maximum whole body dose rate to 0.5 mrem/hr during normal operation and the maximum integrated whole body dose to 3 rem for 90 days following MHA. The actual expected cumulative whole body dose rate following a MHA is shown on Figure 12.1-8.

The control room ventilation system consists of three split-system direct expansion air conditioning units, a ducted air intake and air distribution system, and an emergency filteration section with HEPA filters and charcoal absorbers. The system is shown on Figures 9.4-1 and 9.4-2.

Each of the three indoor units includes a fail-open electrically operated inlet damper, medium efficiency filters, direct expansion refrigerant cooling coil, steam coil, cabinet type centrifugal fan rated at 7500 cfm and 1.25 in. wg. external static pressure, and a gravity damper to prevent recirculation through deenergized units. Each of the indoor sections is served by a weather resistant, roof mounted condensing unit which consists of an air cooled condenser section assembled to a refrigerant compressor section.

The emergency filtration system processes control room air through charcoal filters if the plant site is subject to airborne contamination. This system includes a manual shut-off damper, a 1 unit wide x 2 unit high bank of HEPA filters, and a 1 unit wide x 6 unit high bank of charcoal adsorbers feeding into two full capacity centrifugal booster fans arranged in parallel, each rated at 2000 cfm and 4 inches wg. static pressure. Motorized inlet and gravity outlet dampers are provided at each booster fan. Two electric motor operated 16 in. butterfly isolation valves are located on each duct leading from the outside air intake hoods which are mounted on the north and south walls of the reactor auxiliary building at elevation 78'-4". All ductwork is of welded construction.

During normal operation the control room ventilation system operates in a basic recirculation mode with the emergency filteration system deenergized. Outside makeup air bypasses the emergency filter train and mixes with the control room return air before it is conditioned by the cooling or heating coils.

Upon receipt of a CIS, all four outside air intake valves close within five seconds preventing entry of potentially contaminated outside air. The CIS also energizes both filtration booster fans which, through interlocks, open the respective fan inlet dampers. One of the two filtration booster fans may be manually deenergized and placed on standby. Control room air is then recirculated at a rate of 2000 cfm through the HEPA and charcoal filters, resulting in the cleanup of residual airborne radioactivity that may be present.

Based on a control room volume of 95,472 cu ft which conservatively includes space above the hung ceiling and adjacent offices, 1.26 air changes per hour are recirculated through the filters. Assuming 90 percent efficiency, activities of filterable nuclides are reduced to 1 percent of their initial levels in 3.2 hours, assuming no in-leakage.

The outside air intakes may remain closed for a period of time dictated by maximum allowable CO_2 concentrations and minimum levels of oxygen. Because of the high ratio of control room volume per occupant, considerable time would pass before levels of either constituent could endanger personnel. CO_2 level should be kept at 1 percent or below for continuous exposure. Levels of 2 or 3 percent can be tolerated for one or two hours with some discomfort but not permanent harm.[1] Oxygen should not be less than 17 percent for prolonged occupancy. With the outside air intake closed, and assuming 10 occupants, the CO_2 level would reach 1 percent in 44 hours and the oxygen level would reduce to 17 percent in 133 hours. This is based on a breathing rate, oxygen consumption and carbon dioxide production rate of 30 cu ft/hr, 1.20 cu ft/hr and 1.00 cu ft/hr, respectively, and on a net control room volume of 44,100 cu ft. Net control room volume excludes space above the hung ceiling, panel volume and adjacent rooms and for this calculation, is conservative. Upon analysis of favorable post-LOCA site radioactivity information, one of the outside air intake ducts will be opened. Approximately 750 cfm of outside air is then mixed with 1250 cfm of return air and drawn through the HEPA and charcoal filter section. Determination of whether to draw outside makeup air from either the north or south side of the reactor auxiliary building is based on the readings of the radiation monitors located at each intake.

When air flow through any of the three control room supply system discharge ducts or through either of two emergency filter booster fan discharges falls below a preset limit, it is annunciated in the control room.

12.2.3 SOURCE TERMS

The source terms that form the basis for plant ventilation system design and operation are derived from Tables 12.1-3, 12.1-5 and 12.1-6 which

list the specific activities in process fluids and Table 12.2-1 which lists estimates of leakage points and rates in the process systems. When combined properly, the tables yield the source terms necessary for the inhalation dose calculations discussed in Section 12.2.6.

12.2.4 AIRBORNE RADIOACTIVITY MONITORING

12.2.4.1 Containment Atmosphere Radiation Monitoring System

The containment atmosphere radiation monitoring system is designed to provide a continuous indication in the control room of the particulate and gaseous radioactivity levels inside the containment. Radioactivity in the containment atmosphere indicates either the presence of fission products due to a reactor coolant leak or the presence of activation products due to neutron leakage from the reactor vessel. Refer to Figure 12.2-1 for the following discussion.

The sample flow is withdrawn from the plant vent through two isokinetic nozzles. The nozzles are designed such that the sampling velocity is the same as that in the containment cooling system ring duct so that preferential selection does not occur, i.e., so that the weights of the radioactive particles do not become a factor in obtaining a representative sample.

One isokinetic stream passes through a moving paper filter which collects particulates greater than 1 micron in size; it then passes through a gas sampler where it is monitored for radio-gas content before being pumped back to the containment. The moving filter assembly consists of a fixed position detector scanning the moving filter for any deposited activity. The other isokinetic sample stream passes through an iodine charcoal filter holder before being pumped back to the containment. The filter is periodically replaced and analyzed in the laboratory. Lead shielding arranged in a 4π geometry around the detector prevents interference from background radiation. Refer to Table 12.2-2 for design characteristics of the system.

The pumping system associated with the containment airborne radiation monitoring system consists of 2 root pump assemblies. One 5 cfm pump is connected to the iodine charcoal filter to which the containment air sample is drawn through an isokinetic nozzle. A second 10 cfm pump draws air sample after it has passed through the particulate and gaseous monitors described above.

Each detector system is processed at the factory through a primary radionuclide calibration. In this process, two or more National Bureau of Standards calibrated isotopes are prepared in the proper form to simulate the effluent for which the system will be used. The system readout is then compared with the actual activity being observed. The activity is then removed from the system, and with all other conditions exactly the same, a point source (approximately 0.8 μCi) is attached to the sensitive area of the detector and the detector is placed in its normal operating position. The response of the readout is now recorded. Since all other conditions and settings are identical to the primary

TABLE 12.2-2

DESIGN CHARACTERISTICS - CONTAINMENT RADIATION MONITORING SYSTEM

1. Air Particulate Channel

   a. Detector

Type	beta plastic scintillator
Sensitivity to Sr^{90} in a 0.1 mR/hr background, $\mu Ci/cc$	3.2×10^{-12}
Dynamic range, $\mu Ci/cc$	3.2×10^{-12} to 1.5×10^{-7}
Design pressure, psig	50
Design temperature range, F	32 to 122
Amperage, voltage, frequency, phase	10 amps, 110V ac, 60 Hz, single phase
Check source*	Cs^{137} - 8 μCi

   b. Ratemeter

Type	log count
Accuracy, percent	± 10 of scale
Range, cpm	10 to 10^6
Voltage, frequency, power	100-130V ac, 50-60 Hz, 10W

   c. Moving Filter Paper**

Absorption efficiency, percent	99 for particulates 0.3 microns in diameter
Shield	2 in. 4π geometry lead

2. Gaseous Radiation Monitoring Channel

   a. Detector

Type	beta plastic scintillator
Sensitivity to Xe^{133} in a 0.1 mR/hr background, $\mu Ci/cc$	2.8×10^{-7}
Dynamic range, $\mu Ci/cc$	2.8×10^{-7} to 1.8×10^{-2}
Pressure, psig	50
Temperature range, F	32 to 122
Check source*	Cs^{137} - 8 μCi

   b. Ratemeter (same as air particulate channel)

3. Iodine Charcoal Filter Holder Channel

Type.	Activated charcoal filter
Efficiency, percent	95

*Check source is activated from ratemeter located in control room.

**Filter paper movement may be selected for continuous or programmed step advance.

calibration, the point source is directly referenced to the primary National Bureau of Standards calibrated source. A calibration of the system at periodic intervals can now be easily achieved.

The push button operated check sources that come along with the monitoring system are used during the routine calibration process at the factory, at which time the response caused by the check source is recorded. This provides a quick method of checking the system to ascertain that its still in approximate calibration after installation and operation.

Power sources, indicating devices and recorders are all located on the same main control room radiation panel. All radiation channels are indicated, recorded and annunciated on the control room radiation monitoring panel. Abnormal radiation levels are indicated both visually and audibly.

The sensitivity of the containment atmosphere monitoring system for detecting reactor coolant leakage is discussed in Section 5.2.4.

12.2.4.2 Plant Vent Radiation Monitoring System

The plant vent radiation monitoring system is designed to representatively sample, monitor, indicate and record the radioactivity level in the plant effluent gases being discharged from the plant vent. It provides a continuous indication in the control room of the activity levels of radioactive materials released to the environs so that determination of the total amount of activity release is possible.

A schematic diagram of the plant vent radiation monitoring system is shown in Figure 12.2-2.

The plant vent radiation monitoring system continuously monitors the plant vent exhaust for particulates, iodine and gases at the point of release to the atmosphere. Its function is to confirm that releases of radioactivity do not exceed the predetermined limits set by 10OCFR20.

The sample flow is withdrawn from the plant vent through isokinetic nozzles located at a minimum of 8 vent diameters from the last point of radioactivity entry. The nozzles are designed such that the sampling velocity is the same as that in the plant vent so that preferential selection does not occur, i.e., so that the weights of the radioactive particles do not become a factor in obtaining a representative sample.

One stream passes through a moving filter tape which captures particulates that are greater than one micron in size. The stream then passes through a gas sampler where it is monitored for radiogas content. A 10 scfm root pump draws the sample through the monitors before discharging to the atmosphere. The other stream passes through a fixed filter sampler and then through a 5scfm root pump assembly before it is discharged to the atmosphere.

The particulate and gaseous detectors used for the vent plant monitoring system have the same technical descriptions as those for the containment atmosphere radiation described in Table 12.2-2. The sampler and associ-

ated ratemeters are similar to those previously described. The fixed sampler and its associated detector have the following characteristics:

Electrical power requirement	10 amps, 110 v ac, single phase, 60 Hz
Sample line OD, in.	1
Effluent sample Temperature limits, F	32 to 122
Pressure limits, psig	150 psig with well shield installed; 15 without the well
Sensitive volume, $in.^3$	191 with well installed
Shielding	2" Pb 4π geometry
Filter diameter, in.	2
Iodine cartridge size	
Diameter, in.	2
Thickness, in.	3/4
Detector type	NaI scintillator, 1-1/2" dia. 1" thick, with 10-stage photomultiplier

Each detector system is processed at the factory through a primary radionuclide calibration. In this process, two or more National Bureau of Standards calibrated isotopes are prepared in the proper form to simulate the effluent for which the system will be used. The system readout is then compared with the actual activity being observed. The activity is then removed from the system, and with all other conditions exactly the same, a point source (approximately 0.8 μCi) is attached to the sensitive area of the detector and the detector is placed in its normal operating position. The response of the readout module is now recorded. Since all other conditions and settings are identical to the primary calibration, the point source is directly referenced to the primary National Bureau of Standards calibrated source. A recalibration of the system at periodic intervals can now be easily achieved.

The push button operated check sources that come along with the monitoring system are used during the routine calibration process at the factory, at which time the response caused by the check source is recorded. This provides a quick method of checking the system to ascertain that it is still in approximate calibration after installation and operation.

The plant vent particulate monitor is designed to have a minimum sensitivity for Sr^{90} from 3×10^{-12} to 3×10^{-7} μCi per cc. The radioactive gas monitor is designed to have a sensitivity for X^{133} from 4×10^{-6} to 4×10^{-2} μCi per cc.

12.2.5 OPERATING PROCEDURES

Stationary and portable detectors are provided for monitoring airborne activity in operating and maintenance locations susceptible to radioactive contamination. Where and when practicable, operational procedures will explicitly specify the use of health physics respiratory equipment during performance of maintenance and repair involving potential activity releases.

12.2.6 ESTIMATES OF INHALATION DOSES

Expected airborne activity concentrations for various auxiliary building areas are provided in Table 12.2-1 in terms of ratio of MPC's. Also provided for reference are the maximum whole body doses from internal exposure which would be realized if an individual spent his entire 40 hour work week, 50 weeks per year, in any of the designated areas. In practice this can not and will not occur because the total whole body dose from both external and internal sources will be maintained well below the occupational exposure limit of 5 rem per year by the plant health physics programs. Areas with potentially high concentrations of airborne activity is so marked and access controlled.

12.3 HEALTH PHYSICS PROGRAM

12.3.1 PROGRAM OBJECTIVES

The program objectives are:

a) to implement a radiation monitoring program for protecting the health and safety of plant personnel

b) to implement a medical examination program for plant personnel to confirm the effectiveness of the radiation monitoring program.

12.3.2 FACILITIES AND EQUIPMENT

Health physics personnel have the responsibility for use and maintenance of specially provided facilities and equipment for the protection of plant personnel, including the personnel radiation dosimetry program and the radiation worker medical examination program.

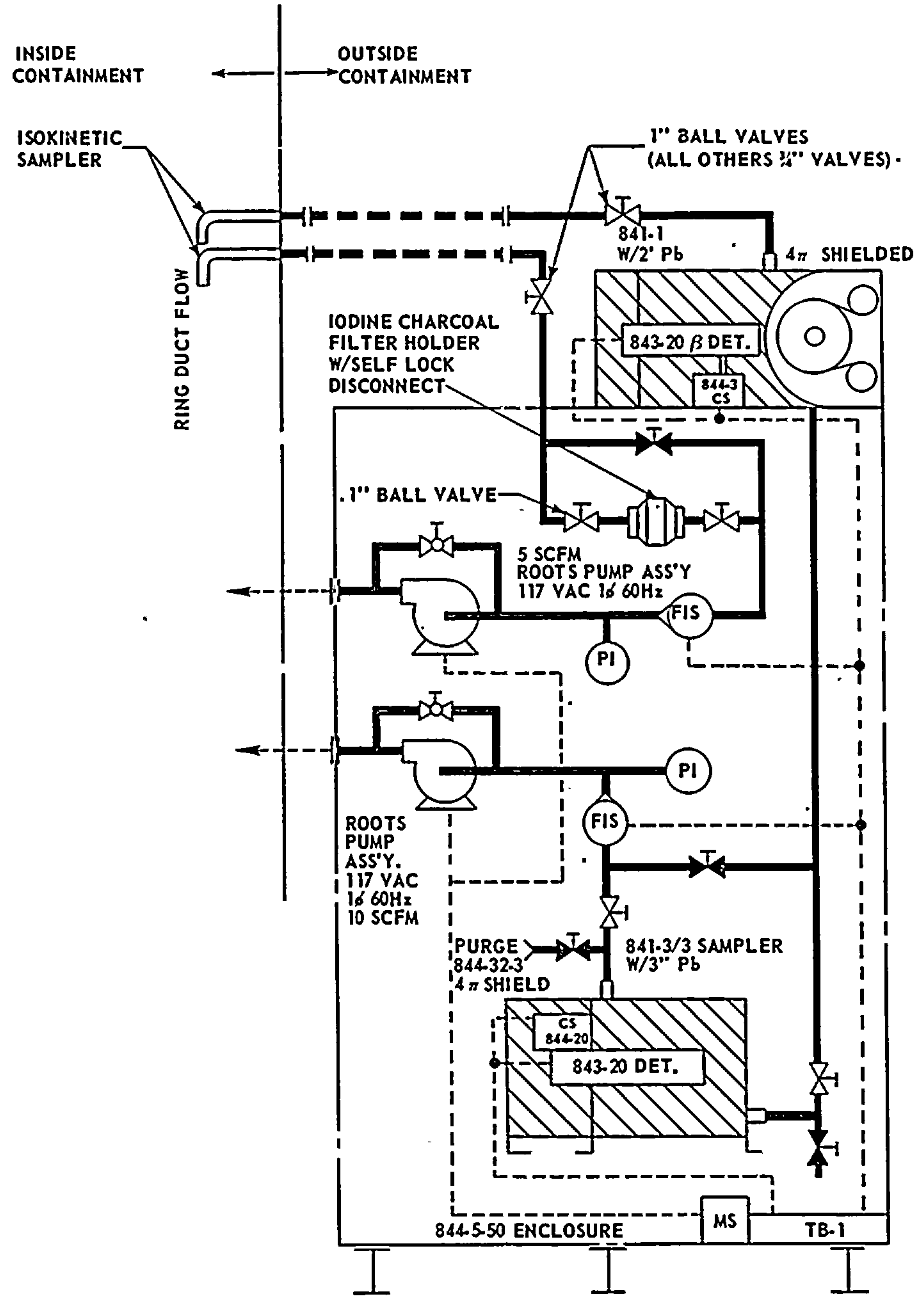
A comprehensive medical examination program for radiation workers is established for maintaining medical records of the physical status of each employee. Subsequent medical examinations will be held as determined necessary. This program is design to preserve the health of employees and to confirm the effectiveness of the radiation control methods employed at the plant.

The general arrangement of the locker room facilities in the reactor auxiliary building is designed to provide adequate personnel decontamination and change areas as shown in Figure 1.2-13. The cold locker room is used to store personal items not required or allowed in the radiation controlled areas. The hot locker room is employed as a change area and storage area for potentially contaminated clothing. Personnel monitors are located at the access point(s). All personnel will survey themselves on leaving radiation controlled areas. Showers, sinks and necessary monitoring equipment also are provided in the hot locker room to aid in the decontamination of personnel.

All personnel entering radiation controlled areas are required to wear protective clothing. The nature of the work to be done is the governing factor in the selection of protective clothing to be worn by individuals. The protective apparel available includes shoe covers, head covers, gloves, and coveralls or lab coats. Additional items of specialized apparel such as plastic or rubber suits, face shields, safety glasses and respirators are also available. In all cases, health physics-trained personnel will evaluate the radiological conditions and specify the required items of protective clothing to be worn.

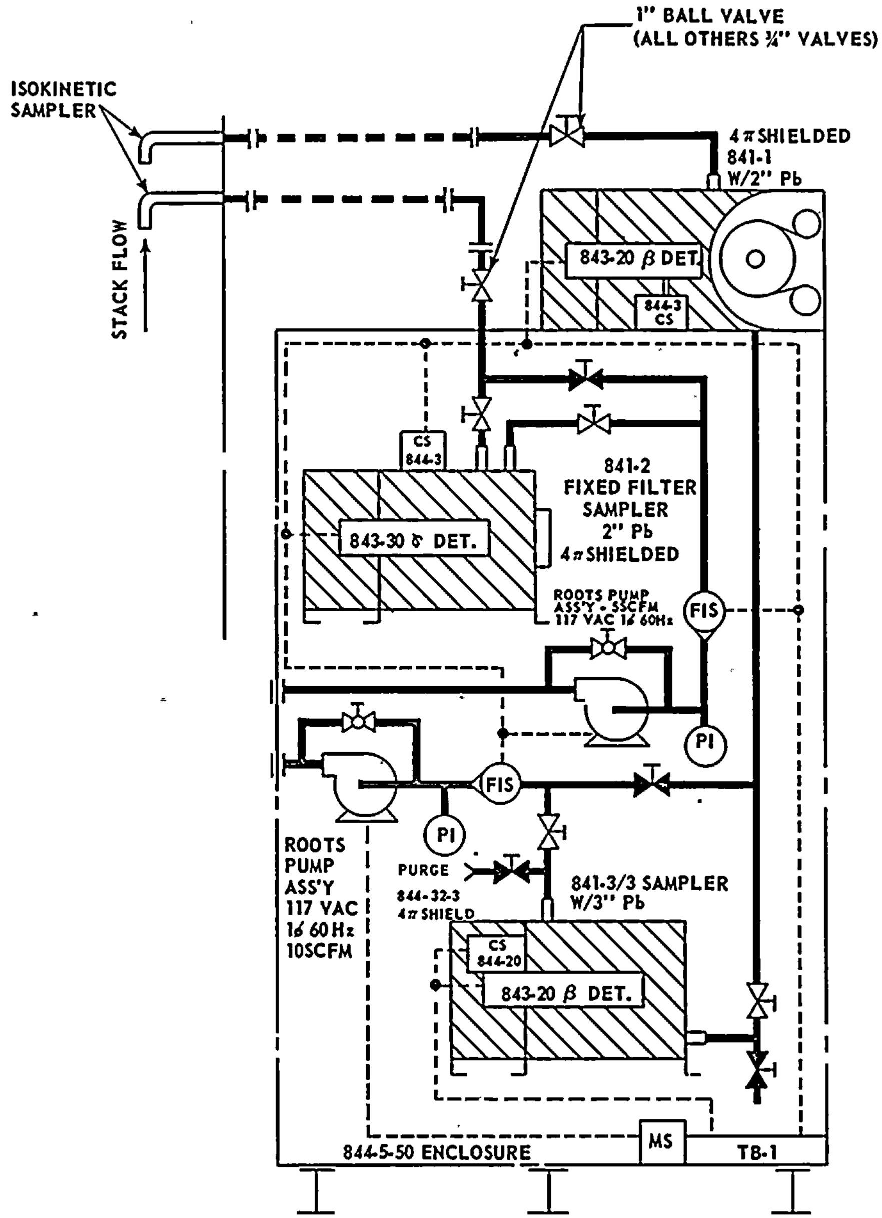
Appropriate written procedures govern the proper use of protective clothing, where and how it is to be worn and removed, and how the change room and decontamination facilities for personnel equipment, and plant areas are to be used.

Provisions are made for decontamination of work areas throughout the plant. The spent fuel cask wash area has facilities to handle decontamination of



FLORIDA POWER & LIGHT COMPANY
Hutchinson Island Plant

CONTAINMENT RADIATION MONITORING SYSTEM
FIGURE 12.2-1



FLORIDA POWER & LIGHT COMPANY
Hutchinson Island Plant

PLANT VENT RADIATION
MONITORING SYSTEM

FIGURE 12.2-2

12.3 HEALTH PHYSICS PROGRAM

12.3.1 PROGRAM OBJECTIVES

The program objectives are:

a) to implement a radiation monitoring program for protecting the health and safety of plant personnel

b) to implement a medical examination program for plant personnel to confirm the effectiveness of the radiation monitoring program.

12.3.2 FACILITIES AND EQUIPMENT

Health physics personnel have the responsibility for use and maintenance of specially provided facilities and equipment for the protection of plant personnel, including the personnel radiation dosimetry program and the radiation worker medical examination program.

A comprehensive medical examination program for radiation workers is established for maintaining medical records of the physical status of each employee. Subsequent medical examinations will be held as determined necessary. This program is design to preserve the health of employees and to confirm the effectiveness of the radiation control methods employed at the plant.

The general arrangement of the locker room facilities in the reactor auxiliary building is designed to provide adequate personnel decontamination and change areas as shown in Figure 1.2-13. The cold locker room is used to store personal items not required or allowed in the radiation controlled areas. The hot locker room is employed as a change area and storage area for potentially contaminated clothing. Personnel monitors are located at the access point(s). All personnel will survey themselves on leaving radiation controlled areas. Showers, sinks and necessary monitoring equipment also are provided in the hot locker room to aid in the decontamination of personnel.

All personnel entering radiation controlled areas are required to wear protective clothing. The nature of the work to be done is the governing factor in the selection of protective clothing to be worn by individuals. The protective apparel available includes shoe covers, head covers, gloves, and coveralls or lab coats. Additional items of specialized apparel such as plastic or rubber suits, face shields, safety glasses and respirators are also available. In all cases, health physics-trained personnel will evaluate the radiological conditions and specify the required items of protective clothing to be worn.

Appropriate written procedures govern the proper use of protective clothing, where and how it is to be worn and removed, and how the change room and decontamination facilities for personnel equipment, and plant areas are to be used.

Provisions are made for decontamination of work areas throughout the plant. The spent fuel cask wash area has facilities to handle decontamination of

large equipment. This area contains a decontamination enclosure and service facilities. A decontamination room external to the machine shop in the reactor auxiliary building is used for the decontamination of hand tools and small equipment.

Respiratory protective devices are required in any situation arising from operations in which airborne radioactivity exists or is expected. In such cases, the airborne concentrations are evaluated by health physics trained personnel and the necessary protective devices specified according to concentration and type of airborne contaminants present.

Respiratory devices available for use include:

a) half-face respirator (filter type)
b) full-face respirator (filter, gas canister, or supplied air)
c) self-contained breathing apparatus

Self-contained or supplied air breathing apparatus is used in any situation involving exposure to gaseous activity or oxygen deficient atmospheres.

The following tabulation of airborne concentrations expected to be in excess of maximum permissible concentration (MPC) limits specified in 10 CTR 20, Appendix B, Table I, is used to determine the appropriate type of respiratory protection equipment required.

Airborne Concentration	Type
Particulate activity less than 10 times MPC and gas activity of 1 MPC or less	Half-face respirator (filter type)
Particulate activity less than 50 times MPC and gas activity of 1 MPC or less	Full-face respirator (filter type)
Particulate activity greater than 50 times MPC or gas activity greater than 1 MPC	Self-contained or supplied air breathing apparatus

Respirators are maintained by checking for mechanical defects, contamination, and cleanliness by health physics trained personnel.

A health physics office and radiochemistry laboratory are located in the reactor auxiliary building (el. 19.5'). These facilities include both laboratory and shielded counting rooms. These are equipped to analyze routine air samples and contamination swipe surveys. Portable radiation survey instruments, respiratory protection equipment and contamination control supplies are stored in the reactor auxiliary building.

Portable personnel radiation monitors are located in the following controlled access points:

a) Auxiliary Building
b) Reactor Containment (during shutdown)
c) Fuel Handling Building

A portal monitor is located in the main guard station at the exit from the plant and provides a final radiation survey of all personnel levaing the generating station area.

The types and minimum quantities of portable radiation survey instruments available for routine monitoring functions are listed in Table 12.3-1.

Survey instruments are calibrated periodically and maintanance records are maintained for each instrument.

In order to protect personnel from access to high radiation areas that may exist temporarily as a result of plant operations and maintenance, warning signs, audible and visual indicators, barricades, and locked doors are used as necessary.

Administrative and physical security measures are employed to prevent unauthorized entry of personnel to any designated high radiation area or contaminated area. These measures include the following:

a) Areas accessible to personnel as listed in Section 12.1.1 in which radiation levels are such that a major portion of the body might receive in any one hour a dose in excess of 100 mrem are barricaded and conspicuously posted as "high radiation areas." Administrative controls require the issuance of a "radiation work permit" prior to entry to any high radiation area or contamination area. Locked wire mesh doors are provided to prevent unauthorized entry into these areas in which the intensity of radiation is greater than 1000 mr/hr.

b) Any individual or group of individuals entering a high radiation area is provided with a portable survey meter which continuously indicates the dose rate in the area.

c) All personnel are required to wear protective clothing for entry into designated contaminated areas. The areas involved will be decontaminated as soon as practicable to prevent the spread of contamination. Decontamination will be performed under the direction of health physics personnel.

12.3.3 PERSONNEL DOSIMETRY

The official and permanent record of accumulated external radiation exposure received by individuals is obtained principally from the interpretation of thermoluminescent dosimeters (TLD) and film badges. Direct reading dosimeters provide day-by-day indication of external radiation exposure.

All persons subject to occupational radiation exposure are issued beta-gamma TLD's and are required to wear them at all times. A badge with neutron sensitive film is issued to personnel whenever a significant neutron exposure is possible. Neutron film badges are issued at the controlled

access point or the main guard station. The TLD's and film badges are processed on a routine basis at monthly intervals. The TLD's and film badge of any individual is processed by special handling whenever it appears that an overexposure may have occurred.

Self reading dosimeters are issued, in addition to the TLD, to personnel where work conditions make desirable a day-to-day indication of exposure. Dosimeters are read, recorded and rezeroed regularly. Dosimeter records furnish the exposure data for the administrative control of radiation exposure.

Special or additional personnel monitors are issued as may be required under unusual conditions. These devices are issued at the discretion of health physics personnel.

Visitors to the controlled access areas are escorted by qualified personnel and are issued appropriate personnel monitoring devices including self reading dosimeters. An escort may not be required for those who have received the necessary radiation protection training when this arrangement is approved by the Health Physicist and authorized by the Plant Superintendent.

Records of radiation exposure history and current occupational exposure will be maintained for each individual for whom personnel monitors are issued.

TABLE 12.3-1

PORTABLE RADIATION SURVEY INSTRUMENTS

Quantity	Range or Capacity	Type	Sensitivity
2	0-0.1, 1, 10, 100 mr/hr 0-500, 5K, 50K, 500K cpm	Low range beta-game survey meters	1500 cpm/mr/hr 1400 cpm/mr/hr
4	0-25, 250, 2500, 25000 mr/hr 0-5, 50, 500, 5K, 50K, 500K mr/hr 0-5, 50, 500 mr (Integrated)	Intermediate range beta-gamma survey meters	$\alpha>3.5$ Mev:$\beta>40$ Kev γ-40 Kev to 2 Mev
2	0-10, 100, 1000 R/hr 0-0.1, 1, 10R (integrated)	High range beta-gamma survey meters	$\alpha>1.7$ Mev; $\beta>25$ Kev γ-10 Kev to 1.3 Mev n-Thermal to 10 Mev
2	0-10, 000 mr/hr	Neutron survey meters	
1	18 cfm, nominal	High volume air particulate sampler	
2	1 cfm, nominal	Low volume air particulate sampler	
2	160-7,000 cpm, alarm set	Beta-gamma portal monitors	$\simeq$3700 cpm/mr/hr γ-80 Kev to 3 Mev
50	0-200 mr	Self reading dosimeters	

TABLE OF CONTENTS

CHAPTER 13
CONDUCT OF OPERATIONS

Section	Title	Page
13.1	ORGANIZATIONAL STRUCTURE OF APPLICANT	13.1-1
13.1.1	CORPORATE ORGANIZATION	13.1-1
13.1.1.1	Function and Responsibility	13.1-1
13.1.1.2	Management and Technical Support Staff	13.1-1
13.1.1.3	Organizational Relations Among Applicant, NSSS and A-E	13.1-5
13.1.1.4	Operations Staff	13.1-5
13.1.2	OPERATING ORGANIZATION	13.1-6
13.1.2.1	Plant Organization	13.1-6
13.1.2.2	Administration	13.1-7
13.1.3	QUALIFICATION REQUIREMENTS FOR NUCLEAR FACILITY PERSONNEL	13.1-9
13.1.3.1	Plant Superintendent	13.1-9
13.1.3.2	Assistant Plant Superintendent - Operations	13.1-10
13.1.3.3	Assistant Plant Superintendent - Maintenance	13.1-10
13.1.3.4	Assistant Plant Superintendent - Technical	13.1-10
13.1.3.5	Plant Engineer - Instrument and Control	13.1-11
13.1.3.6	Reactor Engineer	13.1-11
13.1.3.7	Radiochemist	13.1-12
13.1.3.8	Health Physicist	13.1-12
13.1.3.9	Plant Supervisor	13.1-12
13.1.3.10	Watch Engineer	13.1-13
13.1.3.11	Control Center Operator	13.1-13

13.1.3.12	Nuclear Operator	13.1-14
13.1.3.13	Turbine Operator	13.1-14
13.1.3.14	Technicians	13.1-14
13.1.3.15	Maintenance Personnel	13.1-15
13.1.3.16	Succession of Responsibility	13.1-15
13.2	TRAINING PROGRAM	13.2-1
13.2.1	PROGRAM DESCRIPTION	13.2-1
13.2.1.1	Formal Nuclear Training Program Content	13.2-1
13.2.1.2	Schedule Chart	13.2-1
13.2.1.3	On-the-Job Reactor Plant Operation	13.2-2
13.2.1.4	Plant Simulator Training	13.2-3
13.2.1.5	Previous Nuclear Training	13.2-3
13.2.1.6	Formal On-Site Training	13.2-3
13.2.1.7	Non-licensed Personnel Training	13.2-4
13.2.1.8	General Employee Training	13.2-4
13.2.1.9	Responsible Training Coordinator	13.2-4
13.2.2	RETRAINING PROGRAM	13.2-4
13.2.3	REPLACEMENT TRAINING	13.2-5
13.2.4	TRAINING RECORDS	13.2-5
13.3	EMERGENCY PLANNING	13.3-1
13.3.1	EMERGENCY ORGANIZATION	13.3-1
13.3.1.1	Authorities, Responsibilities and Duties	13.3-1
13.3.1.2	Emergency Notification Means	13.3-4
13.3.2	RADIOACTIVE MATERIALS RELEASE DETERMINATION	13.3-7
13.3.2.1	Magnitude Criteria for Notification of Governmental Agencies	13.3-7
13.3.2.2	Protective Measures and Levels of Action	13.3-8

13.3.3	NOTIFICATION PROCEDURES	13.3-8
13.3.3.1	Local Agreements	13.3-8
13.3.3.2	State Agreements	13.3-10
13.3.3.3	Federal Agencies Agreements	13.3-12
13.3.4	UPDATING PROVISIONS	13.3-13
13.3.4.1	Organization	13.3-13
13.3.4.2	Procedures	13.3-14
13.3.4.3	Emergency Personnel Lists	13.3-14
13.3.5	EMERGENCY FIRST AID AND DECONTAMINATION FACILITIES	13.3-14
13.3.5.1	On-Site Provisions	13.3-14
13.3.5.2	Transportation Arrangements	13.3-15
13.3.6	OFF-SITE TREATMENT ARRANGEMENTS	13.3-16
13.3.6.1	University of Miami School of Medicine	13.3-16
13.3.6.2	Fort Pierce Memorial Hospital	13.3-16
13.3.7	PERSONNEL	13.3-17
13.3.7.1	Licensee Employees	13.3-17
13.3.7.2	Other Assistants	13.3-18
13.3.8	RADIATION EMERGENCY PERIODIC DRILLS	13.3-18
13.3.9	POST-EMERGENCY REENTRY CRITERIA	13.3-19
13.4	COMPANY NUCLEAR REVIEW BOARD	13.4-1
13.4.1	CONSTITUENTS AND FUNCTIONS	13.4-1
13.4.2	REVIEW & AUDIT - CONSTRUCTION	13.4-1
13.4.3	REVIEW & AUDIT - TEST AND OPERATION	13.4-1
13.5	PLANT PROCEDURES	13.5-1
13.5-1	PROCEDURE GENERATION, REVIEW AND MAINTENANCE	13.5-1
13.5-2	OPERATING PROCEDURES	13.5-3
13.5.3	EMERGENCY PROCEDURES	13.5-3
13.5-4	MAINTENANCE PROCEDURES	13.5-5

13.5.5	INSTRUMENT TEST AND CALIBRATION PROCEDURES	13.5-5
13.5.6	CHEMICAL - RADIOCHEMICAL CONTROL PROCEDURES	13.5-6
13.6	<u>PLANT RECORDS</u>	13.6-1
13.7	<u>INDUSTRIAL SECURITY</u>	13.7-1
13.7.1	PERSONNEL AND PLANT DESIGN	13.7-1
13.7.2	SECURITY PLAN	13.7-1

LIST OF FIGURES

FIGURE	TITLE
13.1-1	COMPANY ORGANIZATION CHART
13.1-2	PLANT ORGANIZATION DIAGRAM
13.2-1	EMPLOYEE TRAINING SCHEDULE
13.3-1	EMERGENCY ORGANIZATION

13. CONDUCT OF OPERATIONS

13.1 ORGANIZATIONAL STRUCTURE OF APPLICANT

13.1.1 CORPORATE ORGANIZATION

13.1.1.1 Function and Responsibility

The FPL Company organization chart is shown in Figure 13.1-1. The engineering and operations functions are under the Operating Vice President. The design and engineering of power plants and coordination with construction is the responsibility of the Vice President of Engineering.

Technical support capability for nuclear plant operation exists in the Engineering and Quality Assurance Departments and their consultants.

13.1.1.2 Management and Technical Support Staff

The Quality Assurance Department is under the direction of a Vice President, Dr. James Coughlin. Dr. Coughlin received a Ph.D. from Purdue University Mechanical Engineering School where he was on the teaching staff for seven years. In 1955 he joined the Babcock & Wilcox Atomic Energy Division and, as Project Coordinator for the Consolidated Edison Indian Point Unit 1, his activities involved reactor design, operation and control. In 1957 he was named Supervisor of Reactor Thermal Design, and a year later became Assistant Section Chief of Reactor Engineering, with collateral responsibility for preparation of research and development programs. In 1960 he became Nuclear Systems Specialist, serving as liaison with marketing sales and engineering in preparing both technical and economic development of nuclear steam system products. He joined FPL in 1967 and has the responsibility for nuclear power activities.

Dr. Coughlin is a registered professional engineer and a member of numerous technical organizations including ASME, ANS and AIF.

Reporting to the Vice President, Nuclear Power Activities, is Joseph W. Williams, Jr., Manager of Quality Assurance, who was appointed to this position in May, 1972. He graduated from the University of Florida in 1950 with a Bachelor of Chemical Engineering Degree.

Mr. Williams has twenty-one years of experience in the operation, maintenance and supervision of modern oil and gas fueled high pressure steam power plants, including participation in the start up of three high pressure boiler and turbine generator units. This experience includes Student Engineer, Training and Operations Departments, 1950 and 1951; Betterment Engineer, Plant Betterment Foreman, and Plant Results Foreman, Palatka Plant, 1951 to 1955; Assistant Plant Superintendent - Operations, Riviera Plant, Palatka Plant and Cutler Plant, 1955 to 1960; and Plant Superintendent, Palatka Plant, Turkey Point Plant Units 1 and 2, 1960 to May, 1972. As Plant Superintendent, he was responsible for operating and maintaining the fossil fuel units in a safe, reliable, and efficient manner.

He has completed the following nuclear industry related courses:

Nuclear Power Reactor - University of Florida, 1966
Nuclear Fuel Management - NUS, 1966
Radiological Health - Public Health Service, 1966
Reactor Safety & Hazards Evaluation - Public Health Service, 1967
Advanced Nuclear Technology - University of Florida, 1967

Mr. Williams is a registered professional engineer.

The Vice President - Engineering is Mr. James A. Lasseter. Mr. Lasseter graduated from the Georgia Institute of Technology in 1939 with a BS degree in Electrical Engineering. He joined FPL in 1946 and held various positions relating to distribution system design, equipment development, failure analysis and system studies. In 1953 he became System Distribution Engineer with overall engineering responsibility for the Company's distribution system. In 1960 he became System Engineer with responsibilities relating to distribution systems and to installation and maintenance of the Company's transmission system relaying, telemetering, load control and supervisory control equipment.

In 1964 he was named Assistant Chief Engineer. In 1965 he was appointed Manager of Transmission and Distribution which included overall responsibility for construction, operation and maintenance of the Company's transmission and distribution system.

In 1972 he was appointed Vice President - Engineering and is responsible for directing the Company's engineering activities relating to its generation, transmission and distribution facilities.

Reporting to the Vice President of Engineering is Mr. Walter H. Rogers, Jr., Chief Engineer - Power Plants. A 1947 graduate of Rensselaer Polytechnic Institute with a BME degree, Mr. Rogers was employed by Foster Wheeler from 1947 to 1952 in the Research and Development Department, participating in steam generator design and development.

In 1952 he joined the Company and participated in the design of numerous power plants now operating. In 1965 he was assigned to the Turkey Point Project participating in plant design. In 1968 he was appointed to the job he now holds, with responsibility for procurement and engineering of power plants. He is a registered professional engineer.

Reporting to Mr. Rogers are two nuclear Project Managers, each with about twenty years experience in both nuclear and fossil power plant operation, testing and engineering, who coordinate engineering, design and quality assurance on power plants assigned to them.

The Hutchinson Island Project Manager, Mr. E. H. O'Neal graduated from the University of Florida with a BME degree in 1949. Mr. O'Neal was employed by General Electric Company during 1949 and 1950 with various assignments on G.E.'s training program including steam turbine development testing. In late 1950 he joined Florida Power & Light Company and has been involved in plant design and equipment procurement for over 20 units installed on Florida Power & Light Company's system. He is a registered professional engineer.

The Turkey Point Project Manager, Mr. J. B. Olmstead, graduated from Rensselaer Polytechnic Institute in 1950 and was employed by the Babcock and Wilcox Company from 1950 to 1968 in various assignments including start-up of steam generation equipment, and participation in the start-up of SRE, Shippingport and S3G. From 1964 to early 1968 he was Manager of the Nuclear Power Generation Department Proposal Section participating in the design and marketing of nuclear steam systems.

In early 1968 he joined FP&L as the Project Manager for Turkey Point. He is a registered professional engineer.

Reporting to the Project Managers are Project Engineers. Two of these engineers have nuclear engineering degrees. Mr. Clifford S. Kent, Jr., received an MSE in 1967 from the University of Florida, Department of Nuclear Engineering and joined the Company then. Mr. Sidney G. Brain graduated from the University of Florida with a Bachelor of Science in Engineering Sciences (Nuclear Option) in 1966. He joined the Company in 1966 and has participated in the Turkey Point Project since then.

The Engineering Department, Power Plants staff engineers include specialists in the fields of nuclear, mechanical and electrical engineering, and plant construction and quality assurance.

The staff engineer, nuclear specialty, is Mr. Francis P. Green who graduated from the University of Illinois with a Master of Science Degree in Electrical Engineering in 1949, and in 1958 he received a Master's Certificate in Nuclear Engineering from the Oak Ridge School of Reactor Technology.

From 1949 through 1952 he was a development engineer in Van de Graaff accelerator engineering and nuclear physics research with the NEPA Project and with the Union Carbide Corporation at Oak Ridge National Laboratory. Continuing at the National Laboratory, from 1952 through 1960 he was a development engineer in the Instrumentation and Controls Department, primarily associated with the use of analog computation in the design and development of reactor control systems. During the years 1961 through 1965, he served Union Carbide Corporation, as Staff Assistant to the Director of Research at the Parma Research Center and then as Advanced Planning and Criticality Officer in the development and manufacture of graphite-matrix nuclear reactor fuel. From 1966 through part of 1969 he was a Senior Engineer with the Plant Safeguards and Licensing Department of the Nuclear Energy Systems Division, Westinghouse Electric Corporation, responsible for the preparation of Safety Analysis Reports for Turkey Point Units 3 and 4, the Carolina Power and Light, Robinson Unit 2 and the Rochester Gas and Electric Corporation, Ginna Unit.

Mr. Green joined Florida Power and Light in 1969 and serves as nuclear consultant and is involved in design review and power plant licensing.

A staff engineer, mechanical specialty, is Mr. Ben F. Gilbert who attended Georgia Institute of Technology for three years in Mechanical Engineering and who graduated from the University of Miami with a Bachelor of Science Degree in Mechanical Engineering in 1965.

He was a FPL summer employee, Engineer Trainee, at the Cutler Power Plant and a Student Engineer at the Lauderdale Power Plant while attending the University of Miami. He joined FPL in February, 1965, as Engineer Trainee at the Cape Kennedy Power Plant, where he progressed to Plant Test Engineer in 1965, Assistant Plant Engineer in 1966, and Plant Engineer in 1967. He has participated in the start-up of two large high pressure boilers and turbine generators.

Mr. Gilbert participated in the procedures writing group at Turkey Point in 1970 and following three months experience in the electrical engineering start-up group, joined the Engineering Department, Power Plants in 1971.

A second staff engineer, mechanical specialty is Fred I. Brown who attained his Associate Degree in Applied Science and Bachelor of Science Degree in Mechanical Engineering at Purdue University.

His experience with summer employment while attending Purdue included two summers as production machinist at Zolner Piston Corporation, one summer as a copper tube bender at Wayne Fabricating Company, Fort Wayne, Indiana, and one summer in the Purdue University tin shop. He also has one year experience in design and testing as a Junior Engineer at the Research and Development Laboratory, Anaconda Wire & Cable Co.

He joined FPL in June, 1967, as Engineer Trainee and was assigned to the Cutler Power Plant. He became Assistant Plant Engineer in January, 1968, and participated in the start-up of one large oil and gas fueled boiler and turbine generator. He transferred to the Turkey Point Plant April, 1970, and worked in the writing group for Nuclear Units 3 and 4. In June, 1972, he joined the Engineering Department, Power Plants.

Mr. Brown is a registered professional engineer.

Among the consultants available to Power Plant Engineering is Mr. Raymond L. Lyerly who spends over 50 per cent of his time on Company work. Mr. Lyerly has a MS in ME from Columbia University, is a registered professional engineer and has had a private consulting practice since 1971.

Mr. Lyerly has had 22 years of experience in the fields of heat transfer and nuclear engineering, largely in the power generation field. His experience includes manufacturing, mechanical designs, experimental programs, analyses, plant start-up and operations, and consulting assignments. the last six years of his work has been heavily oriented toward nuclear safety.

Mr. Lyerly's early career was related to special heat transfer equipment for power plants, stationary gas turbines, and early nuclear design programs.

In 1959 he joined General Nuclear Engineering Corporation as a Senior Heat Transfer Specialist. In this capacity he worked on several reactor designs, principally concentrating on fuel and core design, for both gas-cooled and water-cooled reactor types. He was assigned to the BONUS Nuclear Power Plant Program during the plant's final construction period (Rincon, Puerto Rico) and assisted in its power startup operations.

Mr. Lyerly joined Southern Nuclear Engineering, Inc., a design and consulting firm as a founding member in 1964. His regular activities have included safety evaluation, accident analyses, criteria development, design reviews, and other assignments for various utility clients and Government agencies.

13.1.1.3 Organizational Relations Among Applicant, NSSS and A-E

The working interrelationships and organizational interfaces among the applicant, the nuclear steam supply system manufacturer, the architect-engineer and other contractors is described in detail in Chapter 17.

13.1.1.4 Operations Staff

The Power Resources Department, which is headed by the Director of Power Resources, Mr. A. D. Schmidt, reports to the Operating Vice President, Mr. Loftin Johnson. The Managers of Power Resources report to Mr. Schmidt.

Mr. Schmidt has approximately 30 years experience in the operation of electric generating plants. He received a BME from the University of Florida in 1942 and is a registered professional engineer. He has taken courses in nuclear engineering given by the University of Florida for Company employees and has completed the course in Management of Radiation Accidents given by EPA.

Mr. Bensen received a BME degree from Tulane University in 1946 and joined the Company in 1947. He has approximately twenty five years experience in the operation, maintenance and supervision of power plants.

He has taken nuclear engineering courses at the University of Florida, completed the Westinghouse Reactor Operator Training Program, the on-site Reactor Operator Training Program at Turkey Point Plant, and the EPA course in the management of radiation accidents.

13.1.2 OPERATING ORGANIZATION

13.1.2.1 Plant Organization

The organization chart for the Hutchinson Island Plant is shown on Figure 13.1-2. The nuclear plant organization for Hutchinson Island Plant follows the same basic plan as other power plants throughout the Florida Power & Light Company system. This results in a staff of approximately 25 full time operating employees and 30 technical and maintenance employees. These employees are grouped according to their duties in operation, maintenance and technical staff. Other support personnel are available from the Florida Power & Light Company system for specific work assignments.

The Plant Superintendent reports to the Manager of Power Resources - Nuclear on all matters concerning the operation and maintenance of the Hutchinson Island Plant. Assistant Plant Superintendents report to the Plant Superintendent in their areas of responsibility. The normal operating shift consists of a Plant Supervisor, Watch Engineer, Control Center Operator, Nuclear Operator and Turbine Operator. The Plant Supervisor is directly responsible to the Assistant Plant Superintendent-Operations for all operations on his shift. Personnel will hold AEC licenses as shown in Figure 13.1-3. During and following initial core loading a licensed Senior Operator will be on site at any time fuel is being handled in the refueling facilities and/or in the reactor, or when fuel is in the reactor.

The initial start-up, including system check outs, preoperational testing, fuel loading, initial criticality, and approach to full power operation will be performed by the regular Hutchinson Island plant personnel with assistance from the Production Department staff, Combustion Engineering and Ebasco as needed.

The maintenance force consists of approximately 15 men experienced in mechanical and electrical maintenance of large steam electric generating plants headed by the Assistant Plant Superintendent-Maintenance. The nucleus of this group will have gained nuclear plant maintenance experience at the Turkey Point Plant.

The Technical staff supervised by the Assistant Plant Superintendent - Technical, consists of plant engineers, instrument and control personnel and technicians who will function in the areas of reactor physics, radio chemistry, radiological protection, water chemistry, instrument-control maintenance and plant performance.

All three groups are supported and augmented as necessary by the Power Resources Department's staff of mechanical, electrical, chemical and nuclear engineers, the Power Plant Engineering Department, the Production Test Group, and the Central Chemical Laboratory.

Technical expertise will be augmented in the areas of reactor physics, fuel management, radio chemistry, radiological protection and metallurgy by the use of consultants in the particular field, or vendor advisory services during the early years of operation.

13.1.2.2 Administration

Administrative policies will be established pertaining to personnel conduct, method of conducting operations, and preparing and retaining plant documents. The rules and instructions will provide a clear understanding of operating philosophy and management policies. In particular, written administrative policies will be provided to control the issurance of documents, including changes, which prescribe activities affecting safety-related structures, systems, or components, such as operating procedures, test procedures, equipment control and work permit procedures, maintenance or modification procedures, and refueling procedures. These policies will assure that documents, including revisions or changes are reviewed for accuracy and approved for release by authorized personnel and are distributed to and used by the personnel performing the prescribed activity.

The general responsibilities and authorities of the plant staff will be delineated. These include:

a) The operator's authority and responsibility to shut down the reactor when he determines that the safety of the reactor is in jeopardy or when operating parameters exceed the reactor protective system setpoints and automatic shutdown does not occur.

b) The responsibility of the senior reactor operator to determine the circumstances, analyze the cause, and determine or recommend to the appropriate level of management that operations can proceed safely before directing the return of the reactor to power after a trip or an unscheduled or unexplained power reduction.

c) The senior reactor operator's responsibility to be present at the plant and to provide direction for returning the reactor to power following a trip or an unscheduled or unexplained power reduction.

d) The operator's responsibility to believe and respond to instrument indications until they are proven to be incorrect.

e) The operator's responsibility to adhere to the plant Technical Specifications.

In the event that a situation should arise which could affect the health or safety of the public, the Plant Superintendent has the authority to shut down the unit. In his absence the Assistant Plant Superintendents - Operation and Technical, the Plant Supervisor, the Watch Engineer and the Control Center Operator have authority to shut down the unit. The Health Physicist can effectively recommend a shutdown directly to any of the Supervisors. All plant supervisory personnel will have sufficient training in radiation safety to appreciate recommendations from the Health Physicist and to recognize conditions requiring unit shutdown.

Administrative controls will be established to assure that all operations, maintenance procedures, tests and emergencies are handled in accordance with written procedures which have been reviewed and approved through established channels. The Plant Superintendent has the responsibility and authority to operate the units within the limits of the facility license. Within these limits, he has the authority to approve procedures and their subsequent revisions.

Temporary changes which do not violate the intended purpose of the original written procedures may be made provided such changes are reviewed for safety considerations and are approved by two knowledgeable members of the plant supervisory staff, at least one of whom shall hold a Senior Operator License. Such changes shall be documented and reviewed at the next regular meeting of the Plant Nuclear Safety Committee and approved by the Plant Superintendent.

Significant changes will be made only with authorization from the Company Nuclear Review Board. Significant changes are those changes which conflict with the intent of the Operating License and Technical Specification.

Written procedures for safety related systems will be reviewed by the Plant Nuclear Safety Committee and recommendation for revision made to the Plant Superintendent. Written revisions will then be approved and issued by the Plant Superintendent. Current written procedures will be available to the operator at all times.

Routine inspection of operating logs, charts and other data will be made by engineers on the plant staff. In addition all abnormal or nonroutine operations will be reviewed by engineers of the Production Department staff. They will make periodic visits to the plant for general review of operations and assistance with any problems that might arise.

Work of the plant Health Physicist will be reviewed by his direct supervisors.

Periodic meetings will be held by plant supervisory and technical personnel to keep all personnel appraised of current conditions and to review plant safety procedures.

Review and audit of plant operations is performed by the Plant Nuclear Safety Committee. Its description, responsibilities and authority are stated in detail in the Technical Specifications.

13.1.3 QUALIFICATION REQUIREMENTS FOR NUCLEAR FACILITY PERSONNEL

All key personnel assigned to the Hutchinson Island Plant will have had extensive experience in nuclear and fossil fueled electric generating plants in their respective areas of responsibility, or will have received nuclear training to prepare them for their specific assignments. All employees at the plant will have training in health physics. Licensed personnel will receive training in the technical, license, and legal requirements for protection of the health and safety of the public. Operating shift personnel will be trained concerning the effect of radiation, the theory and use of radiation detection devices and procedures for determining and establishing radiation work areas. A licensed operator will be responsible for implementing radiation protection procedures on each shift. All minimum qualification requirements for facility personnel satisfy ANSI 18.1, "Standard for Selection and Training of Personnel for Nuclear Power Plants."

13.1.3.1 Plant Superintendent

a) Function and Responsibility

The Plant Superintendent reports to the Director of Power Resources and has direct responsibility for operating and maintaining the plant in a safe, reliable and efficient manner. He is responsible for protection of the plant staff and the general public from avoidable radiation exposure and/or any other consequences of an accident at the plant. He bears the responsibility for compliance with the facility operating license. He has authority to take any action necessary, without consultation, to prevent or mitigate the consequences of an accident.

b) Minimum Requirements

The Plant Superintendent shall have ten years of responsible power plant experience of which a minimum of three years shall be nuclear power plant experience. A maximum of four years of the remaining seven years may be fulfilled by academic training on a one-for-one time basis. This academic training shall be in an engineering or scientific field generally associated with power production. He shall have acquired the experience and training normally required for examination by the AEC for a Senior Operator License whether or not the examination is taken.

13.1.3.2 Assistant Plant Superintendent - Operations

a) Function and Responsibility

The Assistant Plant Superintendent - Operations has the responsibility for directing the actual day-to-day operation of the plant and shall hold a Senior Operator License. He reports directly to the Plant Superintendent and directs the plant operating staff. He coordinates operation related to maintenance activities with the Assistant Plant Superintendent - Maintenance. He assumes all of the Plant Superintendent's responsibilities and authority in his absense. He is responsible for over all supervision of fuel handling operations.

He has the authority to shut down the unit if in his opinion conditions warrant this action and he has the authority to initiate the Emergency Plans.

b) Minimum Requirements

The Assistant Plant Superintendent - Operations shall have a minimum of eight years of responsible power plant experience of which a minimum of three years shall be nuclear power plant experience. A maximum of two years of the remaining five years of power plant experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis. He will hold a Senior Operator License.

13.1.3.3 Assistant Plant Superintendent - Maintenance

a) Function and Responsibility

Responsible for supervision of mechanical and electrical maintenance and for the equipment maintenance records.

b) Minimum Requirements

The Assistant Plant Superintendent - Maintenance shall have a minimum of seven years of responsible power plant experience or applicable industrial experience, a minimum of one year of which shall be nuclear power plant experience. A maximum of two years of the remaining six years of power plant or industrial experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis. He further should have non-destructive testing familiarity, craft knowledge, and an understanding of electrical, pressure vessel, and piping codes.

13.1.3.4 Assistant Plant Superintendent - Technical

a) Function and Responsibility

Supervises the instrument and control, chemistry and health physics, and reactor engineering groups.

He has the authority to shut down the unit if in his opinion conditions warrant this action and he has the authority to initiate the Emergency Plans.

He assists the Assistant Plant Superintendent - Operations in the supervision of fuel handling operations.

b) Minimum Requirements

The Assistant Plant Superintendent - Technical should have a minimum of eight years in responsible positions, of which one year shall be nuclear power plant experience. A maximum of four years of the remaining seven years of experience should be fulfilled by satisfactory completion of academic training.

13.1.3.5 Plant Engineer - Instrument and Control

a) Function and Responsibility

He is responsible for supervision of maintenance, calibrations and installation of all instrument and control equipment and for maintenance of instrument and control records.

He has the authority to shut down the unit if in his opinion conditions warrant this action and he has the authority to initiate the Emergency Plans.

b) Minimum Requirements

The Plant Engineer - Instrument and Control shall have a minimum of five years experience in instrumentation and control, of which a minimum of six months shall be in nuclear instrumentation and control. A minimum of two years of this five years experience should be related technical training. A maximum of four years of this five years experience may be fulfilled by related technical or academic training.

13.1.3.6 Reactor Engineer

a) Function and Responsibility

The Reactor Engineer conducts or supervises tests, accumulates and evaluates data and maintains records of the performance of all plant equipment including core performance and fuel records.

b) Minimum Requirements

The Reactor Engineer shall have a minimum of a Bachelor's Degree in Engineering or the Physical Sciences and two years experience in such areas as reactor physics, core measurements, core heat transfer and core

physics testing programs.

13.1.3.7 Radiochemist

a) Function and Responsibility

The Radiochemist conducts or supervises the chemical and radiochemical analyses of water, gas and solid samples. He evaluates test results, and directs corrective measures incident to these analyses.

b) Minimum Requirements

The Radiochemist shall have a minimum of five years experience in chemistry of which a minimum of one year shall be in radiochemistry. A minimum of two years of this five years experience should be related technical training. A maximum of four years of this five years experience may be fulfilled by related technical or academic training.

13.1.3.8 Health Physicist

a) Function and Responsibility

The Health Physicist conducts or supervises surveys and monitoring programs to detect, measure and assess radiation levels within the facility and maintains records and reports of all radiation surveys and monitoring programs. He instructs or assists in the instruction of all personnel in the basic principles of radiation protection. He assists or supervises decontamination operations.

b) Minimum Requirements

The Health Physicist shall have a minimum of five years experience in radiation protection at a nuclear reactor facility. A minimum of two years of this five years experience should be related technical training. A maximum of four years of this five years experience may be fulfilled by related technical or academic training.

13.1.3.9 Plant Supervisor

a) Function and Responsibility

The Plant Supervisor is responsible for the actual operation of the plant on his assigned shift. He directs the activities of the operators on his shift and must be cognizant of all maintenance activity being performed while he is on duty. The Plant Supervisor on duty has the authority to shut down the unit if, in his opinion, conditions warrant this action. He has the authority to initiate the Emergency Plans.

During fuel handling operations he may direct the operation or operate fuel handling equipment.

b) Minimum Requirements

The Plant Supervisor shall have a minimum of a high school diploma or equivalent and four years of responsible power plant experience of which a minimum of one year shall be nuclear power plant experience. A maximum of two years of remaining three years of power plant experience may be fulfilled by academic or related technical training on a one-for-one time basis. He shall hold a Senior Operator License.

13.1.3.10 Watch Engineer

a) Function and Responsibility

The Watch Engineer is the working operating foreman and is responsible for plant operations on his shift. Upon assignment he may assume the responsibilities of Plant Supervisor. During fuel handling operations he shall direct or operate fuel handling equipment.

The Watch Engineer on duty has the authority to shut down the unit if in his opinion conditions warrant this action. He has the authority to initiate the Emergency Plans.

b) Minimum Requirements

The Watch Engineer shall have a high school diploma or equivalent and four years of responsible power plant experience of which a minimum of one year shall be nuclear power plant experience. He shall hold a Senior Operator License.

13.1.3.11 Control Center Operator

a) Function and Responsibility

The Control Center Operator operates controls and monitors instruments located in the control room containing reactor, turbine, generator and transmission line control boards and under direct or general supervision directs the operation of all plant equipment as required to maintain proper operating conditions. He executes or directs the execution of orders received from the Plant Supervisor, Watch Engineer, and Dispatcher. He has the authority to shut down the unit if in his opinion conditions warrant this action. During fuel handling operations he may operate fuel handling equipment under general supervision. He may operate radiation survey instruments.

b) Minimum Requirements

The Control Center Operator shall have a high school diploma or equivalent and two years of power plant experience, of which a minimum of one year shall be nuclear power plant experience. He shall hold an Operator License.

13.1.3.12 Nuclear Operator

a) Function and Responsibility

Operates nuclear reactor auxiliary equipment and turbine-generator auxiliary equipment under the direction of a licensed Operator or Senior Operator. Performs inspections of operating equipment and systems including but not limited to the reactor coolant system, chemical and volume control system, component cooling, shutdown heat removal and spent fuel pool cooling systems, waste management system, engineered safety features systems and components, radiation detection equipment, containment and radioactive area ventilation and purge systems, primary water makeup system, refueling water tank, gas supply systems, liquid and gas sampling and analysis equipment and chemical feed addition equipment. He performs operating adjustments and services, records operating data, and operates radiation survey instruments. During fuel handling operations he may operate fuel handling equipment under direct supervision of a licensed operator.

He may operate other plant auxiliary equipment under the direction of a Plant Supervisor, Watch Engineer, or Control Center Operator.

b) Minimum Requirements

The Nuclear Operator shall have a high school diploma or equivalent and should possess a high degree of manual dexterity and mature judgment.

13.1.3.13 Turbine Operator

a) Function and Responsibility

The Turbine Operator operates turbine controls and serves as turbine-generator attendent. He may be assigned additional duties and assist the Nuclear Equipment Operator. Performs operating adjustments and services. Records operating data.

During fuel handling operations he may operate fuel handling equipment under the direct supervision of a licensed operator.

b) Minimum Requirements

The Turbine Operator shall have a high school diploma or equivalent and should possess a high degree of manual dexterity and mature judgment.

13.1.3.14 Technicians

a) Minimum Requirements

Technicians in responsible positions shall have a minimum of two years of working experience in their specialty and should have a minimum of one year of related technical training in addition to their experience.

13.1.3.15 Maintenance Personnel

a) Minimum Requirements

Maintenance personnel shall have a minimum of three years experience in one or more crafts. They shall possess a high degree of manual dexterity and ability and should be capable of learning and applying basic skills in maintenance operations.

b) Maintenance Personnel Experience

Mechanics and electricians who will perform maintenance and repair of the nuclear plant and its components and auxiliaries have met the requirements of the four year joint IBEW-FPL apprentice program or have equivalent maintenance experience.

13.1.3.16 Succession of Responsibility

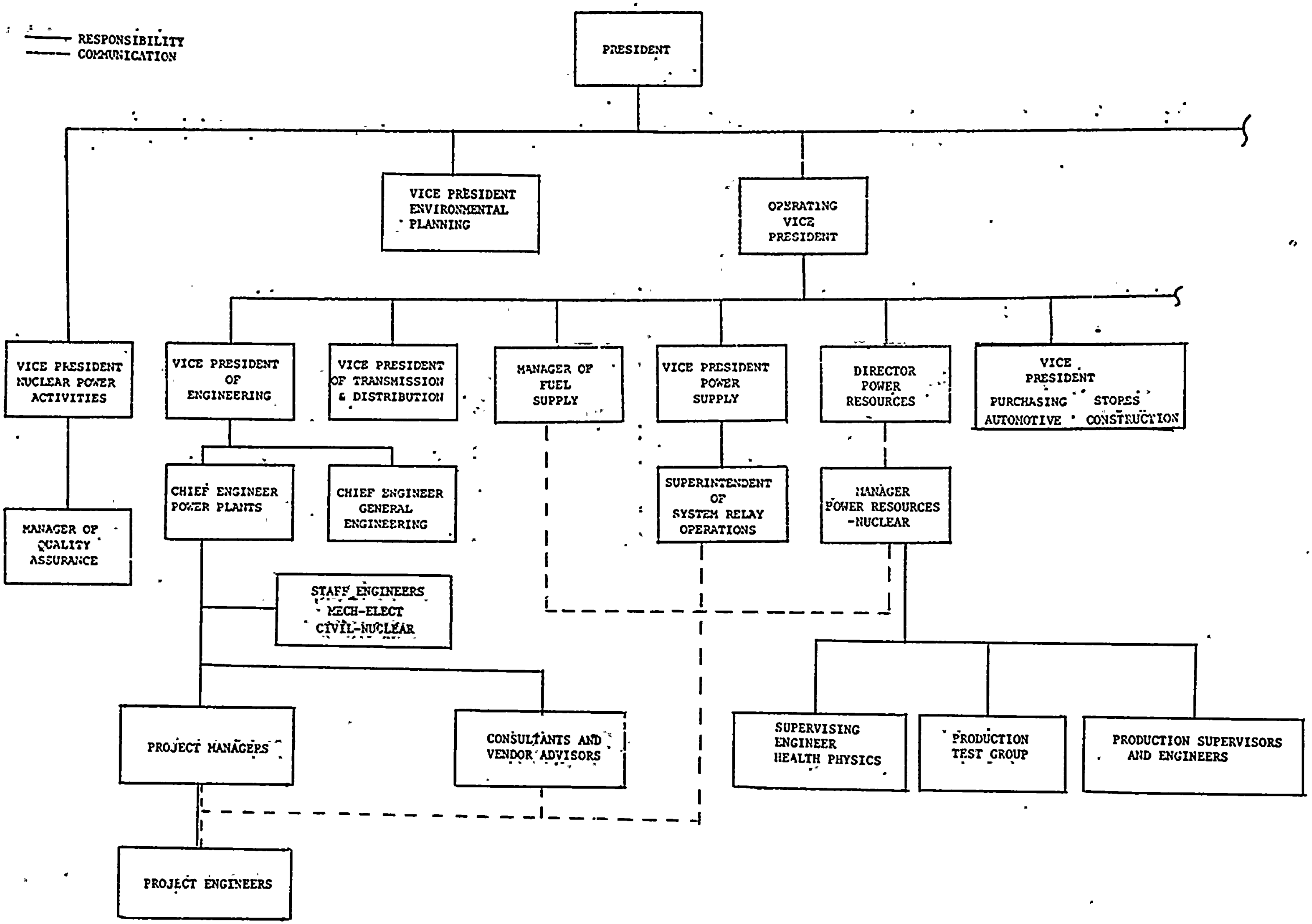
The following table lists the normal sequence of assignment of responsibilities in the event of absence or unavailability.

Plant Superintendent
1. Assistant Plant Superintendent - Operations
2. Assistant Plant Superintendent - Technical

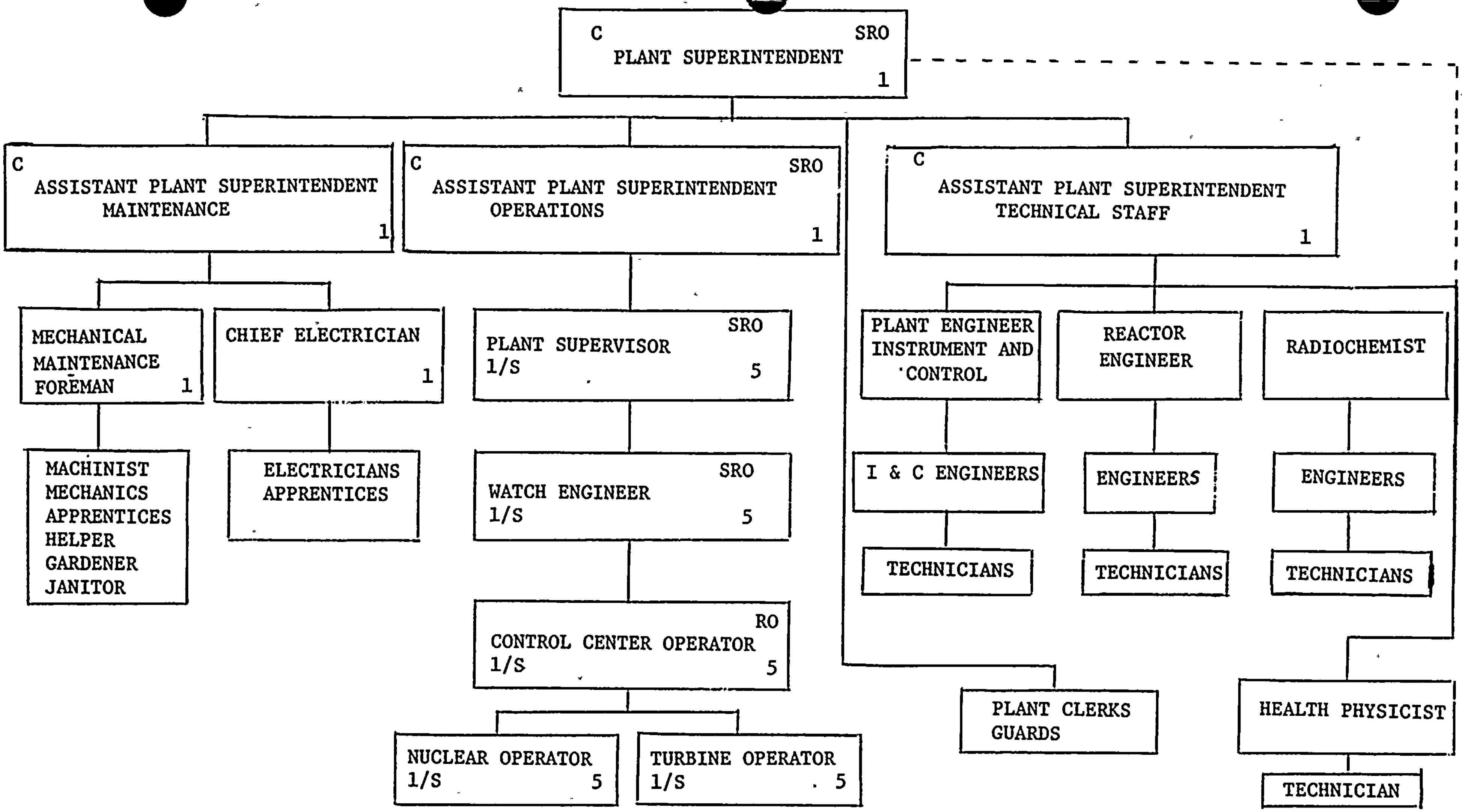
Assistant Plant Superintendent - Operations
1. Assistant Plant Superintendent - Technical
2. Plant Supervisor
3. Watch Engineer

Assistant Plant Superintendent - Technical
1. Assistant Plant Superintendent - Operations
2. Plant Engineer - Reactor
3. Plant Engineer - Instrumentation and Control

Assistant Plant Superintendent - Maintenance
1. Assistant Plant Superintendent - Operations
2. Assistant Plant Superintendent - Technical
3. Maintenance Supervisor
4. Maintenance Foreman



FLORIDA POWER & LIGHT COMPANY
HUTCHINSON ISLAND PLANT
COMPANY ORGANIZATION CHART
FIGURE 13.1-1



Legend

C	RO
Position Title	
1/S	5

C College Graduate
RO AEC Reactor Operator License
SRO AEC Senior Reactor Operator License
1/S Number Required Per Shift
5 Total Number in Classification

HUTCHINSON ISLAND PLANT
PLANT ORGANIZATION DIAGRAM
FIGURE 13.1-2

13.2 TRAINING PROGRAM

13.2.1 PROGRAM DESCRIPTION

The overall objective of the training program is to provide technical development, specialized training and operating experience to those FP&L personnel whose actions, orders or decisions can determine the design or affect or change the operating characteristics of the plant. The Plant Superintendent is responsible for the conduct and administration of the initial on-site training program and for the training of replacement plant personnel.

Beginning with the solicitation of proposals from nuclear equipment suppliers for Turkey Point Plant which will start up before the Hutchinson Island Plant, Florida Power & Light Company senior engineering and operating staff members have undertaken a program of technical development designed to provide them with a high level of proficiency in the evaluation of nuclear plant design. This program has included short courses, seminars, and visitation to operating nuclear power plants. Members of this group have been given responsibility for the evaluation of the designs submitted by the engineers and manufacturers. Guidance on the training program content was obtained from ANSI N-18.1 and the AEC Licensing Guide, "Operating Licenses, Division of Reactor Licensing, November 1965."

13.2.1.1 Formal Nuclear Training Program Content

A nuclear engineering course of 80 class hours was conducted for approximately 40 employees of the Florida Power & Light Company by the University of Florida, Department of Nuclear Engineering. This group of employees included all senior engineering and operating personnel plus a plant supervisory group from whom some of the Hutchinson Island plant supervisory personnel will be chosen. The course was followed by an 80 hour laboratory course at Gainesville, Florida, utilizing the University of Florida's test reactor. An advanced Nuclear Technology course was then given to approximately 20 employees by the University of Florida, Department of Nuclear Engineering. This group included senior engineering and plant supervisory personnel.

The program for formal education and training of the Hutchinson Island Plant staff is designed to meet the individual needs of the participants, depending upon their background, previous training, and expected job assignment. Company personnel who compose the initial group to receive training include those from whom the plant staff will be selected. These people will then serve as instructors for the training program to be given to the remainder of the plant personnel.

13.2.1.2 Schedule Chart

Figure 13.2-1 shows the training schedule for each employee in relation to the schedule for preoperational testing and fuel loading.

Most times indicated are tentative and constitute expected minimum requirements.

13.2.1.3 On-the-Job Reactor Plant Operation

The training program will be divided into several phases of activity. They are:

a) Initial Training at Turkey Point

It is planned that the cold license personnel for Hutchinson Island will be trained and be licensed at Turkey Point Plant Prior to the on-site phase at Hutchinson Island Plant. Technical training for candidates for AEC dold examinations at the reactor operator level will cover the following subjects:

1) principles of reactor operation
2) design features of the nuclear power plant
3) general operating characteristics of the nuclear power plant involved
4) instrumentation and control systems
5) safety and emergency systems
6) standard and emergency operating procedures
7) radiation control and safety provisions

In addition to the above subjects, technical training for candidates for AEC cold examinations at the senior reactor operator level will cover the following subjects:

8) reactor theory
9) handling and disposal of, and hazards associated with, radioactive materials
10) specific operating characteristics of the nuclear power plant
11) fuel handling and core parameters and
12) administrative procedures, conditions, and limitations.

This training will be conducted by the regular Turkey Point Plant staff instructors. The course will follow the same outline that has been successfully used for training Turkey Point licensed operators.

Following the formal training program the operators will be examined and receive hot licenses at Turkey Point. Selection of candidates for cold licenses at Hutchinson Island will be made primarily from those who hold Turkey Point licenses.

b) Hutchinson Island Plant Design Lectures

Following the Turkey Point training phase, the Hutchinson Island Plant supervisory staff will participate in a lecture series related to the NSSS design conducted by Combustion Engineering. The lectures will include as a minimum the following subjects:

Core Mechanical Design
Thermal and Hydraulics
Reactor Physics
Instrument and Control Systems Design
Mechanical Systems Design
Electrical Systems Design
NSSS Response
Safety Analysis and Technical Specifications
Operating and Emergency Procedures
Start-up Test Program

c) Plant Simulator Training

Each cold license candidate will participate in training on a plant simulator which will be programmed to simulate Hutchinson Island Plant parameters.

Approximately two months prior to initial core loading additional simulator time will be given each license candidate using updated core parameters and set points.

d) Previous Nuclear Training

It is expected that in addition to training and experience received at Turkey Point, some license candidates will have received training in military service or at other reactor locations.

Some of the Turkey Point formal training may be eliminated for these persons, based on the judgment of the Plant Superintendent. These license candidates will however participate in the on-site and simulator training sessions.

e) Formal On-Site Training

Following the simulator training the cold license candidates will receive on-the-job training and participate in plant checkout. This phase of the training program will be organized to train and familiarize the non-supervisory personnel in the theory, design, installation of equipment, testing, startup, and operation of the plant. During this phase, supervisory and non-supervisory personnel will assist in developing and writing instruction manuals and test programs, testing of individual components and systems, developing and writing operating, refueling and emergency procedures, preparation of maintenance procedures and participation in maintenance work.

Instrument and control, radiochemistry and health physics personnel will be assigned several months prior to initial fuel loading and will be trained as required in their respective responsibilities.

All personnel will receive training in radiation protection and emergency procedures prior to fuel loading.

f) Non-licensed Personnel Training

The nucleus of the non-licensed staff including supervisors, engineers, technicians, operators, and maintenance personnel, will be drawn from personnel experienced at the Turkey Point Plant. In addition to on-the-job training at both Turkey Point and Hutchinson Island, all site personnel will be given training in radiation protection. In addition, some engineers and technicians will receive training at locations other than Turkey Point and Hutchinson Island when deemed necessary by the Plant Superintendent.

g) General Employee Training

Industrial Safety and First Aid training are conducted as a part of the regular company safety program at monthly meetings. This program will be implemented to include topics applicable to the nuclear plant.

h) Training Coordinator

The Training Coordinator will be responsible for the employee training program. Initially, this will be an experienced Combustion Engineering training expert. He will be assisted by a FP&L supervisor who will assume the position after the completion of initial operator training.

13.2.2 RETRAINING PROGRAM

Formal retraining will be given to all licensed operators in the form of classroom study, covering the initial on-site training material, and oral and written examinations. Personnel will be periodically enrolled in appropriate refresher courses or courses covering new concepts to ensure the continued safe operation of the plant. Refresher courses will be conducted at approximately two year intervals.

The retraining program will include:

a) Plant start-up and shutdown precedures
b) Normal plant operating conditions and procedures
c) Operational limitations, precautions, and set points
d) Emergency plans and security procedures
e) Off normal condition operating procedures
f) Emergency shutdown systems

g) Changes in equipment and operating procedures
h) General safety, first aid, and radiation safety
i) Alarms and instrumentation signals
j) Operation of selected auxiliary systems important to overall plant safety.

Periodic written, oral and walk-through type examinations will be given to all operators to evaluate the status of their training.

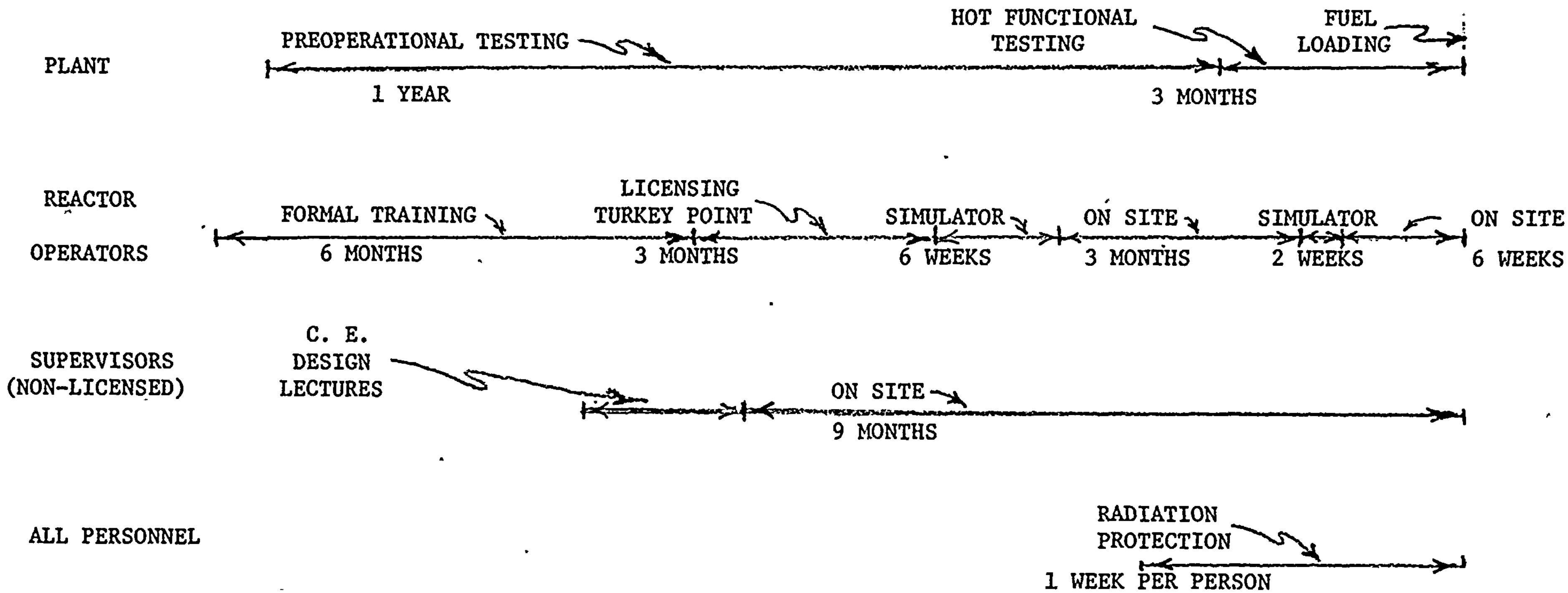
13.2.3 REPLACEMENT TRAINING

A continuing training program during the life of the plant will assure that each replacement employee who requires an AEC operator license receives the same general material, follows the same fundamental program used for the original personnel and knows his specific duties and responsibilities for normal and emergency operations. Capable personnel from other FP&L plants will be considered for these replacement positions on a selective basis. The supervisory and technical staff will be the primary source of instructors for the continuing training program but outside assistance will be obtained as necessary to assure competent replacement personnel.

13.2.4 TRAINING RECORDS

The training records will be set up and maintained by the Training Coordinator. The annual training program evaluation will be the responsibility of the Plant Superintendent.

EMPLOYEE TRAINING SCHEDULE



HUTCHINSON ISLAND PLANT
EMPLOYEE TRAINING SCHEDULE

FIGURE 13.2-1

13.3 EMERGENCY PLANNING

A plan is being developed to cover all foreseeable emergencies such as fire, personnel injuries and illness, radiation exposure, contamination and other conditions that could result from either nuclear or non-nuclear incidents. In order to assure continuing familiarity with the emergency plan, practice drills will be held from time to time for training purposes. It is intended that the plant be self-sufficient insofar as possible in handling emergency conditions, but when necessary, the Police, Fire Department, AEC, and other local, state and federal agencies will be called upon. Agreements are being made with outside agencies concerning their expected role in any emergency.

13.3.1 EMERGENCY ORGANIZATION

13.3.1.1 Authorities, Responsibilities and Duties

The Company organization charts relating to the Hutchinson Island Plant operation and to the review and approval of emergency plans and implementation are shown in Figures 13.3-1 through 13.3-3.

a) Operating Vice President

1) Administer the overall planning and implementation of the Emergency Plans

2) Make policy decisions and expend funds to implement the plan

3) Contract or arrange for the outside services and agreements required to support the provisions of the plan

4) Conduct liaison with Federal, State, and Local Agencies, and coordinate Company emergency planning with these agencies to ensure a minimum of conflicting actions and authority in order to avoid confusion and provide for efficient emergency protective actions

5) Ensure the physical security of the plant site

6) Act as Chairman of the Company Emergency Plan Review Committee to ensure the adequacy of Company plans to cope with foreseeable emergencies.

b) Director of Power Resources

1) Provide plans, personnel, equipment and facilities to minimize the consequences of all foreseeable incidents involving FPL nuclear power facilities

2) Periodically up-date the provisions of the Emergency Plans to reflect both on-site and off-site emergency response capability

3) Act as the off-site Emergency Control Officer

4) In event of accidents or radiation incidents as described in the Emergency Plans, Section VI, NOTIFICATION AND REPORT REQUIREMENTS, notify and send required reports to the appropriate Federal and State agencies indicated in Section VI

5) Obtain emergency support from off-site organizations

6) Provide for the continuation of the activities of the Off-Site Emergency Organization Headquarters, Ryan Building, Miami, through the plant re-entry and recovery operations as required. If recovery is a long term operation, the Headquarters may be relocated to the FPL General Office Building, Miami

c) Plant Superintendent

1) Implement the provisions of the Plan and all Annexes and Appendices

2) Procure and store emergency equipment and supplies in appropriate locations

3) Appoint emergency team leaders and alternates for each work shift and provide for back-up manning to assure adequate emergency coverage

4) Direct plant personnel in long-term post-accident planning and recovery operations

5) Limit the radiation exposures of personnel to reasonable values consistent with the nature of the incident and the urgency to take action involving personnel exposure risk

6) Periodically up-date the provisions of the Emergency Plans to ensure on-site emergency response capability

7) Provide for appropriate training and refresher courses for employees assigned to emergency reponse teams

8) Provide for semi-annual drills, on a non-interference basis, to exercise all elements comprising the emergency response capability of the Company, the Plant, the Medical Support personnel and hospital treatment facilities, and those Federal, State, and Local agencies which may be involved

9) Submit a written evaluation of the drills to the Operating Vice President through established reporting channels within seven working days

10) Submit other verbal and written reports as required

d) Plant Supervisor

1) Act as Emergency Coordinator until relieved by a succeeding Shift Supervisor or by command of higher authority

2) Declare the existence of an emergency and classify as to type

3) Evacuate on-site personnel as required

4) Act to control the emergency condition

5) Establish an Emergency Control Center

6) Obtain assistance from off-site groups as required

7) Follow, generally, the Emergency Action Sequence Guidelines

e) Emergency Team Leaders

Emergency Team Leaders and Back-Up Emergency Team Leaders will be prepared to implement all aspects of their emergency assignments. They will:

1) Be familiar with the detailed procedures and standing orders necessary to implement the Emergency Plans.

2) Ensure the currency of their team roster to include the name, home address, telephone number, emergency assignment, training, drill experience, and physical limitations of team members.

3) Be prepared to assist the Emergency Coordinator in obtaining back-up emergency assistance during off-shift hours.

f) Off-Site Emergency Organization

The Off-Site Emergency Organization is composed of Company officials and a staff of assistants and is responsible for all off-site emergency control activities of FPL. It will provide assistance to the Plant in administration, public relations, security, engineering, and technical matters.

1) Emergency Control Officer (Director of Power Resources)

a. Direct the Off-Site Emergency Organizations

b. Activate the required members of the Organization upon receipt of notification of an emergency condition from the Plant Duty Call Supervisor or the Division Load Dispatcher

c. Obtain, as required, the assistance of off-site support personnel of the Company, Local agencies, State and Federal agencies, and other support agreements or arrangements

d. Notify the U. S. Atomic Energy Commission in accord with the provisions of 10 CFR 20.402 and 20.403, and the State Division of Health in accord with the FPL - Division of Health Agreement of November 1, 1971, as amended

2) Emergency Administrative Officer (Operating Vice President)

a. Make any new policy decisions required

b. Authorize unusual expenditures of funds for emergency activities

c. Conduct liaison with regulatory agencies on administrative matters not covered by established emergency plans

3) Emergency Information Officer (Miami Area Vice President)

Release information to the public and maintain liaison with local agencies and the news media for public relations purposes

4) Emergency Security Officer (Executive Assistant - Security)

Provide and maintain liaison with local security and law enforcement agencies

5) Emergency Technical Officer (Vice President Engineering)

Provide advice and technical information regarding engineering design and as-built construction details for the Hutchinson Island Plant

13.3.1.2 Emergency Notification Means

Licensee Personnel

Any individual who discovers an emergency condition, or a condition which could become an emergency, shall take the following general actions:

a) Notify the Plant Supervisor by telephone or in person and give him the details of the emergency condition found

b) Take any immediate action which he is qualified to perform that will aid in controlling and/or minimizing the effects of the emergency; keep other personnel away from the affected area

c) Withdraw to a safe area, and if the possibility of radioactive contamination exists, remain in the safe area until monitored

The Plant Supervisor, acting as Emergency Coordinator, shall use the plant P.A. system to announce that an emergency condition exists and to activate the Emergency Teams. If the emergency condition is a fire, or one that requires evacuation of the containment, he will sound the fire alarm or containment evacuation alarm and then make the P.A. announcement.

As soon as possible after determining that an emergency condition exists, the Emergency Coordinator, or his designee, shall notify the Duty Call Supervisor and request that he notify the Plant Superintendent, the Emergency Team Leaders and the Emergency Control Officer.

The Duty Call Supervisor will notify the other members of the Plant Supervisory staff.

The Emergency Team Leaders will notify the members of their teams. The Emergency Control Officer will notify the required members of the Off-Site Emergency Organization and any required State or Federal agency. If for any reason the Emergency Coordinator cannot establish communications with the Duty Call Supervisor, the System Load Dispatchers in Miami will be notified and will then make the notifications requested by the Emergency Coordinator.

<u>State and Federal Personnel</u>

Notification and reporting requirements of State and Federal regulatory agencies will be implemented by the Off-Site Emergency Organization.

a) <u>State of Florida - Division of Health</u>

1) Notification of the Division of Health under the criteria shown below is in accord with the agreement between the Division of Health and the Company and does not constitute a request for assistance. <u>However, assistance may be obtained if specifically requested by the Company</u>.

2) The State Division of Health, Radiological Emergency Staff Duty Officer will be notified in event of incidents involving radionuclides which:

a. Cause or threaten to cause, within the Generating Station Area, serious injury to personnel, exposure to ionizing radiation in excess of limits specified in 10 CFR 20, or excessive and/or uncontrolled spills or gaseous releases.

b. Result in any unplanned release of radioactive material to an area occupied by members of the general public, and which may require that protective measures be taken to prevent or minimize personnel exposure.

c. Involve the evacuation of contaminated or potentially contaminated personnel to off-site areas under conditions which could result in the spread of significant quantities of radionuclides to normally uncontaminated areas.

d. Require or may require the assistance of the Florida Radiological Response Team in on-site or off-site areas.

e. Notification will be made and the following information will be provided:

1) Name, location, and telephone number of individual calling.

2) Type of incident: Liquid or gaseous release; liquid or solid spill; personnel exposure or contamination.

3) Exact location of incident.

4) Time and date of incident.

b) <u>Atomic Energy Commission</u>

Director, Region II, Directorate of Regulatory Operations
U. S. Atomic Energy Commission
Suite 818
230 Peachtree Street, N. W.
Atlanta, Georgia 30303

Phone: A.C. 404, 526-4503 Day or Night

1) Notify the Atomic Energy Commission by TELEPHONE OR TELEGRAPH, as shown below, in event of incidents involving by-product or special nuclear material which may have caused or threatens to cause:

a. <u>Immediate Notification - Serious Incident</u>

1. Loss or theft of licensed material in quantities that may result in a substantial hazard to persons in unrestricted areas; or

2. Exposure of any individual equal to or exceeding either:

Whole body-------------------------------- 25 Rem
Skin of whole body------------------------150 Rem
Feet, ankles, hands, or forearms----------375 Rem; or

3. Release of radioactive material which, if averaged over a period of 24 hours, would exceed 5000 times the limits specified in 10 CFR 20, Appendix B, Table II; or

4. A loss of one working week or more of the operation of any facility affected; or

5. Damage to property in excess of $100,000.

b. Twenty-Four Hour Notification - Reportable Incident

1. Exposure of any individual equal to or exceeding either:

Whole body	5 Rem
Skin of whole body	30 Rem
Feet, ankles, hands, or forearms	75 Rem; or

2. Release of radioactive material which, if averaged over a period of 24 hours would exceed 500 times the limits specified in 10 CFR 20, Appendix B, Table 2; or

3. A loss of one day or more of the operation of any facilities affected; or

4. Damage to property in excess of $1,000.

c. Notification - Other Incidents

Any incident not covered in 1) or 2) above, but which, in the judgment of the Emergency Control Officer, would create serious public relations problems.

d. Any written report filed with the Commission pursuant to 1) and 2) above, shall be prepared so that names of individuals who have received exposure to radiation will be stated in a separate part of the report.

13.3.2 RADIOACTIVE MATERIALS RELEASE DETERMINATION

13.3.2.1 Magnitude Criteria for Notification of Governmental Agencies

The Off-Site Emergency Organization will act as the implementing clearing house on emergency conditions telephoned or communicated by radio by judging from the received technical data which local areas if any could be affected by a radioactive material release.

Nomograms and area maps will be used to successively generate an environmental picture of a developing off-site situation. Low population zone local evacuation procedures would be followed in case that projected radiation levels were to exceed a predetermined value as delineated on a meteorological dispersion overlay. Cooperating local and State agencies would be notified to initiate Civil Defense type activities appropriate to the projected conditions.

13.3.2.2 Protective Measures and Levels of Action

The Hutchinson Island Emergency Plans consider the classes of accidents specified in the proposed annex to 10 CFR 50, Appendix D "Discussion of Accidents in Applicants Environmental Reports: Assumptions." Six additional types of accidents or emergencies are also considered. Classes of accidents which would not require initiation of the Emergency Plans are not discussed here.

In the analysis of hypothetical radiological accidents, some minor substitution of analytical assumptions were made. For example, the maximum site boundary exposure values are based on 1.0% failed fuel, where applicable, and atmospheric dispersion coefficients (X/Q) are 10 times the highest annual average sector value at the site. The protective action levels established are designed to prevent any individual within the exclusion area from being exposed to a whole body radiation dose in excess of 2.5 Rems, or from being exposed to a concentration of airborne radioactive material which, if averaged over a period of 24 hours, would exceed 500 times the limits specified for such materials in 10 CFR 20, Appendix B, Table II.

Each of the classes of accidents considered includes the:

a) description of the accident

b) protective action level at which the Emergency Plans will be put into effect

c) protective measures taken to minimize the effects of the accident

d) consequences of the accident

Tables in the Emergency Plans show the location of the various fixed radiation monitors and their alarm set points.

13.3.3 NOTIFICATION PROCEDURES

13.3.3.1 Local Agreements

a) Medical Support

A formal signed agreement between FPL and the University of Miami provides for medical support in event of actual, suspected, or alleged radiation injury and/or radionuclide contamination of persons resulting from an accident at the Hutchinson Island Plant. The University of Miami, through its School of Medicine, Department of Radiology, Division of Nuclear Medicine, developed the Radiation Emergency Evaluation Facility (REEF) to carry out the terms of its agreement with FPL. REEF will:

1) Provide immediate availability of fully equipped local medical facilities in Fort Pierce and Miami, Florida, with an adequate staff of physicians, nurses, and technical personnel skilled in the diagnosis and treatment of radiation injury and personnel contamination.

2) Provide fully equipped and staffed "regional" medical facilities to provide definitive medical care back-up for serious cases of radiation exposure. "Regional" facilities are provided through an agreement between REEF and the Oak Ridge Associated Universities, Oak Ridge, Tennessee.

3) Provide training for FPL personnel in first aid related to radiation injury and radionuclide contamination; assist FPL in radiation related injury investigation and exposure evaluation; participate in radiation accident drills; provide consultation in radiation related matters; conduct pre-placement medical examinations for certain job positions and maintain records of such examinations.

4) An agreement between the Fort Pierce Memorial Hospital and the Radiation Emergency Evaluation Facility (REEF) of the University of Miami School of Medicine provides that the Fort Pierce Memorial Hospital will:

   a. Provide immediate availability on a 24 hour basis, adequately equipped medical facilities, staffed with physicians, nurses, and technical personnel trained in the treatment of radiation injury and injury accompanied by radionuclide contamination. Such availability shall be for the benefit of FPL employees, and at the request of FPL, members of the public who are involved in actual or suspected radiation exposure or contamination resulting from an incident occurring at the Hutchinson Island Plant.

   b. Establish plans, procedures, and training programs for casualty reception, diagnosis, and treatment; such plans, procedures, and programs shall be compatible with those of the Radiation Emergency Evaluation Facility (REEF) of the University of Miami School of Medicine, and shall be subject to the approval of FPL. Plans would be updated as required to reflect changes in personnel or other situations bearing on the effectiveness of the Plans.

   c. Function, generally, as an auxiliary facility to REEF in radiation emergency casualty treatment, and be responsive to the decisions of the REEF Director with respect to such casualty treatment.

d. Maintain permanently and for ready reference the medical records of all persons examined at the request of FPL in relation to actual, suspected or alleged incidents involving possible exposure to radiation or radionuclide contamination. Copies of these records will be sent to FPL and will be made available to REEF.

b) <u>Sheriff's Departments - St. Lucie and Martin Counties</u>

The County Sheriff's Departments will provide for traffic control and security in off-site areas.

c) <u>Fire Departments, Ft. Pierce and Stuart</u>

Will provide fire control support as back-up for the Hutchinson Island Plant fire fighting capability.

13.3.3.2 <u>State Agreements</u>

a) <u>State of Florida, Radiological Response Plan</u>

A formal signed agreement between FPL and the Florida Division of Health, Department of Health and Rehabilitative Services provides for off-site support in event of a radiological incident at the Hutchinson Island Plant. The Department of Health and Rehabilitative Services Radiological Response Plan, which is implemented by the Florida Division of Health, is the official State Plan for use by all agencies of the State in responding to radiological emergencies in Florida.

In addition to the specific assistance to be provided by the Division of Health, the facilities and services of other State agencies are available to the Division of Health. These include, but are not limited to those under the jurisdiction of the Departments of Commerce, (Nuclear and Space Commission), Insurance (Fire Marshall's Office), Community Affairs (Civil Defense), Safety and Motor Vehicles (Florida Highway Patrol), Natural Resources (Marine Patrol), Transportation and Pollution Control.

The agreement with the Division of Health, and provisions of their Radiological Response Plan, provide that the Division of Health will:

1) Provide for a single point of contact between FPL and the State which will serve to activate all State and County emergency organizations supporting this plan.

2) Receive and record all reports of radiological incidents at the Hutchinson Island Plant, furnish radiological

response teams, assess the impact or potential impact of such incidents and activate appropriate provisions of the Radiological Response Plan to assess levels of radioactivity in off-site areas.

3) Coordinate emergency actions such as the evacuation of of off-site public areas and the establishment of off-site exclusion areas.

4) Formulate decontamination procedures and supervise the decontamination of off-site areas to the extent required to protect public health.

5) Institute controls of dietary ingestion of radionuclides by humans to limit intake levels and take action to reduce the dietary intake to acceptable levels.

6) Coordinate its Radiological Response Plan with similar plans of Federal, State, and Local governmental agencies having legal jurisdiction in St. Lucie and Martin Counties.

7) Implement the provisions of the Plan for using the available resources of other State agencies.

8) Train selected representatives of State and Local governmental agencies and private organizations to familiarize them with the health hazards and emergency operating procedures applicable to radiological incidents, and to insure adequate and coordinated emergency response capability.

9) Assist State and Local governmental agencies in planning for any required evacuation and/or exclusion of persons from public or private areas, and the housing and feeding of evacuees.

10) Provide radiological laboratory capability, including mobile laboratory facilities, and field radiological instrumentation, equipment, and supplies to insure that measures outlined above are properly and effectively carried out.

11) In the event of a radiological emergency caused by on-site activities of FPL, the Division of Health will aid and advise FPL in its efforts to contain the spread of contamination resulting therefrom and to decontaminate where necessary any on-site or off-site areas. Supervision of decontamination in off-site areas is the responsibility of the Division of Health.

b) <u>Florida Highway Patrol - Department of Safety and Motor Vehicles</u>

The Florida Highway Patrol will:

1) Use their radio network, as an alternate means, to contact the Division of Health Radiological Emergency Duty Officer at the request of FPL, and to contact local law enforcement agencies and local health authorities

2) Render emergency assistance at the scene of off-site radiation emergencies within their jurisdiction

3) Transport the State Radiological Response Team on request

4) Within their capability, evacuate and exclude individuals from designated public and private areas

5) Aid in traffic and crowd control in off-site areas in accordance with the Florida Radiological Response Plan

c) <u>Department of Community Affairs, Division of Emergency Government (Civil Defense)</u>

Civil Defense organizations in both <u>St. Lucie</u> and <u>Martin Counties</u> will aid in personnel evacuation, feeding, and shelter in off-site areas. The decision to evacuate an area will normally be made by the Florida Division of Health.

d) <u>Florida Marine Patrol - Department of Natural Resources</u>

The Marine Patrol, at the request of the Florida Division of Health, will assist in the evacuation of water areas affected by a radiological incident and will control access to these areas.

13.3.3.3 <u>Federal Agencies Agreements</u>

a) <u>U. S. Coast Guard</u>

The U. S. Coast Guard will employ helicopter, amphibious aircraft and surface craft to implement their plan to "provide to any person or governmental authority any assitance that constitutes the rescue, aid, or evacuation of persons in danger, and the protection of property threatened by any type of disaster.

b) <u>U. S. Atomic Energy Commission - Region II, Directorate of Regulatory Operations, Atlanta, Georgia</u>

Will receive notification of incidents and conduct investigative activities.

c) <u>U. S. Atomic Energy Commission - Savannah River Operations Office, Aiken, South Carolina</u>

Will provide a Radiological Assistance Team to aid in evaluating radiological hazards, provide consultation and render services for on-site and off-site radiation surveys in accord with the provisions of their Radiological Assistance Plan.

13.3.4 UPDATING PROVISIONS

An annual review of the Emergency Plan organization, procedures and personnel lists will be conducted by the Administrative Assistant and the Nuclear Supervising Engineer, who report to the Director of Power Resources.

13.3.4.1 <u>Organization</u>

Emergency Information Kits will be prominently and consistently labeled and color coded. The contents of the kit will be posted on the exterior of the kit, and will consist of:

a) Emergency Plans and Emergency Operating Procedures.

b) Emergency Health Physics Instructions.

c) Charts and overlays for estimating downwind radiation dose and dose rates, and delineating the areas affected.

d) Appropriate drawings of the plant site and generating station area.

e) List of items to be considered in emergency situations.

f) List showing the location and available supply of portable radiation survey instruments, personnel protective equipment and clothing, and decontamination supplies.

Emergency Information Kits will be maintained in the following locations:

a) Control Room (Primary On-Site Emergency Control Center)

b) Plant Office (First Alternate On-Site Emergency Control Center)

c) Plant Supervisor's Office

d) Health Physics Office

e) Reactor Auxiliary Building

f) Main Guard Station

g) Site Boundary Station (Off-Site Emergency Control Center)

h) Alternate Off-Site Emergency Control Center (Fort Pierce)

i) Off-Site Emergency Organization Headquarters (Ryan Building, Miami)

13.3.4.2 Procedures

Emergency Operating Procedures are provided to ensure the proper implementation of the Emergency Plans. The procedures provide detailed information and instructions on:

a) The specific authority, responsibility, duties and actions of individuals, plant emergency teams and off-site emergency support elements.

b) The classification of radiation emergencies; radiation emergency evacuation criteria; and personnel evacuation, assembly and accountability procedures.

c) Re-entry following a radiation emergency evacuation.

d) Communication network operation.

e) Maintenance of the emergency personnel roster, and emergency accident records.

f) Emergency drills and emergency alarm tests.

13.3.4.3 Emergency Personnel Lists

The Emergency Plans contain the current emergency team lists and telephone numbers for licensee and for outside organizations. These are dated and formally revised at least annually to reflect the inevitable personnel changes in any viable organization.

13.3.5 EMERGENCY FIRST AID AND DECONTAMINATION FACILITIES

13.3.5.1 On-Site Provisions

Personnel monitoring, decontamination and medical equipment and personnel emergency equipment and supplies will be maintained in "units" at two locations within the plant area, at the Site Boundary Station and the Alternate Off-Site Emergency Control Center. The "units" will consist of radiation survey instruments, air samplers, protective clothing, respiratory protective equipment, decontamination supplies, ropes and signs, and other miscellaneous equipment and supplies which may be required for reentry into the Hutchinson Island site.

A full description of the contents of each unit will be posted on the unit containers; all units will be inspected on a quarterly schedule and provisions will be made to record the date on which the units were inspected and the name of the inspector.

In all serious or potentially serious injury cases not requiring immediate action to save life, competent medical advice will be sought before any action is taken which could have a seriously complicating effect.

To avoid the spread of radionuclide contamination to normally uncontaminated on-site or off-site areas, and when such action is feasible, persons with minor injuries accompanied by radionuclide contamination will be decontaminated and treated at the plant by trained first-aid personnel or by medical personnel called to the site. Subsequent treatment and tests for radionuclide absorption will be conducted as recommended by the Chief, Department of Radiology, Fort Pierce Memorial Hospital and/or the REEF Director.

When the nature of the injury precludes first-aid treatment at the plant first-aid station, contaminated clothing and equipment will be removed, if possible, and the injured person will be evacuated to Fort Pierce Memorial Hospital.

Persons will be taken to or referred to the Fort Pierce Memorial Hospital and/or REEF for tests, evaluation, and/or treatment when:

a) Actual, suspected, or alleged radiation exposure is in excess of: 5 Rems - whole body; 30 Rems - skin of whole body; 75 Rems - feet, ankles, hands, or forearms.

b) External contamination cannot be removed to acceptable levels by normal plant first-aid decontamination procedures.

c) In the judgment of plant health physics personnel, the inhalation or ingestion of radionuclides is in excess of established administrative control limits. Judgment, in inhalation cases, will generally be based on the respiratory protection used, the concentration of the specific radionuclide(s) involved as shown by monitoring equipment or calculation, and the duration of exposure.

13.3.5.2 <u>Transportation Arrangements</u>

a) <u>Transportation of Injured Persons</u>

1) A company owned vehicle is maintained at the plant site for the transportation of injured persons to off-site medical facilities.

2) When radionuclide contamination is not involved or can be readily contained, ambulance service is available from the Universal Ambulance Service in Ft. Pierce.

3) Personal vehicles may be used when dictated by the circumstances.

4) Under certain circumstances not involving emergency actions personnel decontamination information should accompany persons sent to off-site medical treatment facilities.

13.3.6 OFF-SITE TREATMENT ARRANGEMENTS

13.3.6.1 University of Miami School of Medicine

The Radiation Emergency Evaluation Facility (REEF), Miami, Florida, will:

a) Provide, on 24-hour per day basis, for immediate availability of adequately equipped and staffed local (Fort Pierce Memorial Hospital) medical facilities for the treatment of radiation injury, or physical injury accompanied by radionuclide contamination.

b) Provide for immediate availability of fully equipped medical facilities (Miami) with an adequate staff of nurses and technical personnel skilled in the diagnosis and treatment of radiation injury and personnel contamination, and fully equipped and staffed regional facilities (Oak Ridge, Tennessee) to provide definitive medical care for serious cases of radiation exposure.

c) Designate the physical facilities and equipment to be used for initial emergency care and subsequent definitive care and treatment of personnel with radiation injuries.

d) Coordinate the efforts of the multiple medical diciplines within the University of Miami School of Medicine which are committed to support the radiation emergency medical treatment effort of REEF.

e) Designate physicians, basic science personnel and alternate, for special hospital and REEF emergency teams to handle radiation emergency patients.

f) Assist FPL in the investigation and evaluation of incidents involving actual, suspected, or alleged radiation injury or radionuclide inhalation or contamination, and provide consultation on nuclear-medical matters.

13.3.6.2 Fort Pierce Memorial Hospital

The Fort Pierce Memorial Hospital, Fort Pierce, Florida, will:

a) Provide, on a 24-hour per day basis, adequately equipped and

staffed medical facilities for the treatment of physical injury accompanied by radionuclide contamination.

b) Function, generally, as an auxiliary facility to REEF in radiation emergency casualty treatment, be responsive to the decisions of the REEF Director with respect to such treatment, and release to REEF any patients requiring long-term treatment or evaluation.

13.3.7 PERSONNEL

13.3.7.1 Licensee Employees

The Training Coordinator appointed by the Plant Superintendent will ensure that each person assigned to work at Hutchinson Island on a permanent or temporary basis receives a thorough orientation on all emergency plans and procedures required to ensure his safety. All persons assigned to an emergency team or who will be expected to assist in event of an emergency will receive additional training appropriate to his assignment.

The frequency of scheduled regular training courses or refresher courses will be determined by the Training Coordinator and will be based on the requirement for maintaining fully trained emergency teams at all times. Personnel will be informed of changes in emergency plans and procedures at scheduled monthly safety meetings, and refresher course for previously trained personnel will be given semi-annually.

Training courses will be conducted by or under the supervision of plant personnel as follows:

a) Radiation: Plant Health Physicist

b) Fire: Assistant Plant Superintendent - Maintenance

c) First Aid and Personnel Decontamination: Plant Radiochemist

d) Security: Plant Security Supervisor

Special Medical First Aid Training

a) In addition to training given by the FPL Radiochemist (See c. above), personnel of the University of Miami School of Medicine, under the general direction of the Director of REEF (Radiation Emergency Evaluation Facility), will train selected FPL employees in:

   1) The treatment of traumatic injury which may or may not be accompanied by radiation exposure and/or radionuclide contamination

   2) Decontamination procedures for cases of gross personnel

contamination

3) Personnel evacuation procedures, for 1) and 2) above, to off-site medical facilities

b) Details of the medical support available to FPL are given in the Medical Plan, Personnel Evacuation and Treatment.

13.3.7.2 Training of Non-FPL Personnel

a) County, Municipal, and Military groups or agencies which are included in the emergency plan or who may become involved in event of an emergency affecting off-site areas will be offered orientation and training in radiological emergency response by State of Florida Division of Health personnel.

b) All non-FPL groups or agencies who may participate in on-site emergency activities will be encouraged to become familiar with the physical lay-out of the Hutchinson Island Plant, and will be invited to attend appropriate training and emergency plan orientation courses conducted by or for FPL.

13.3.8 RADIATION EMERGENCY PERIODIC DRILLS

Practice emergency drills will be held as required to assure emergency team proficiency, but at least semi-annually by each (shift) Emergency Team and by each Back-Up Emergency Team. The drills will exercise all appropriate sections of the Emergency Plan including coordination with State, County, Local, and Military agencies which may be expected to function in support of Hutchinson Island emergency teams under actual emergency conditions. The extent of participation of non-FPL groups will depend upon the nature and extent of the problems involved in the simulated emergency.

Semi-annually, a drill will be held to exercise all elements of the Plant emergency teams and the non-FPL emergency support teams and groups.

Planning, Execution, and Evaluation

a) The Assistant Plant Superintendent-Operations, assisted by other key personnel, will:

1) Plan all emergency drills and will coordinate the planning with all State, County, Local, Military, and FPL support groups involved

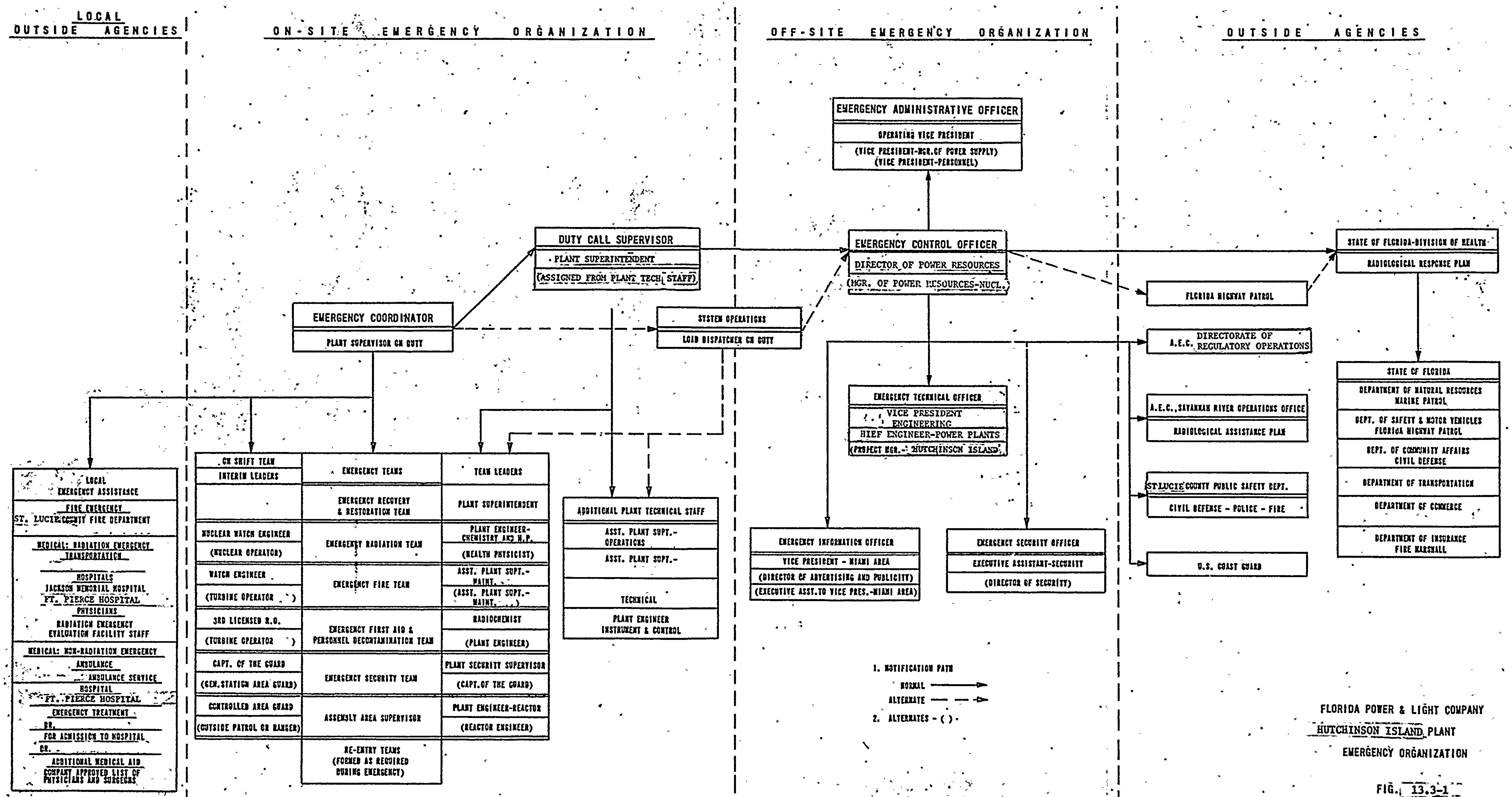
2) Schedule the execution of drills and insure that all facets of the drills are observed by qualified supervisory and/or management personnel

3) Conduct a critique with the Plant Nuclear Safety Committee immediately following each drill to evaluate its effectiveness and to determine the adequacy of personnel response and of the emergency procedures involved. The Plant Nuclear Safety Committee will be responsible for effecting any necessary changes in the Emergency Operating Procedures.

4) Submit to the Plant Superintendent a written evaluation of the drill.

13.3.9 POST-EMERGENCY REENTRY CRITERIA

The Emergency Coordinator will assess the radioactive contamination or other adverse environmental condition levels obtained from monitoring instruments for the area to be reentered. After specifying what protective equipment is required, appropriate personnel will enter the area and audit status of equipment and structures involved in the specified emergency. When found or placed within normal or abnormal condition status, operations under established procedures may be reinitiated and the Emergency Coordinator will announce termination of the emergency.



FLORIDA POWER & LIGHT COMPANY
HUTCHINSON ISLAND PLANT
EMERGENCY ORGANIZATION

FIG. 13.3-1

13.4 COMPANY NUCLEAR REVIEW BOARD

Review and audit of design, construction and operation of nuclear units are performed by the Company Nuclear Review Board. Its purpose, authority, responsibility, and composition are described in detail in the Technical Specifications.

13.4.1 CONSTITUENTS AND FUNCTIONS

The Company Nuclear Review Board consists of the Chairman, Vice President, Nuclear Power Activities; the Vice-Chairman, Chief Engineer - Power Plants; four members, Director of Power Resources, Manager of Power Resources - Nuclear, Manager of Quality Assurance and Project Manager; the Secretary, Project Engineer; and one or more non-voting members including Plant Superintendent and staff consultants. The Board operates under a written charter with the authority to enforce safe operation and maintenance of the nuclear units by critical examination and advisement of all aspects of operation including contemplated actions and after-the-fact anomalies.

13.4.2 REVIEW & AUDIT - CONSTRUCTION

Review and audit of the construction of the Hutchinson Island plant is the responsibility of the Florida Power & Light Company Engineering Department - Power Plants and the Company Nuclear Review Board.

Specific guidelines for review and audit are detailed in the Florida Power & Light Company, Quality Assurance Department "Quality Assurance Program."

13.4.3 REVIEW & AUDIT - TEST AND OPERATION

Review and audit of plant test and operation is the responsibility of the Plant Superintendent and the Company Nuclear Review Board. The Florida Power & Light Quality Assurance Department will provide an independent review and audit of all tests and operations which might affect the safety of the plant. The Company Nuclear Review Board has the power to confer with any consultants it believes necessary both from within FPL and from outside agencies.

The Plant Nuclear Safety Committee will advise the Plant Superintendent on matters of plant test and operation. Its description, responsibilities, and authority is stated in detail in the Technical Specifications.

13.5 PLANT PROCEDURES

13.5.1 PROCEDURE GENERATION, REVIEW AND MAINTENANCE

The plant operating and technical staff will have prepared written procedures prior to fuel loading to be used for start-up, normal operation and anticipated off normal and emergency operating conditions. The initial procedures will be reviewed by the Plant Nuclear Safety Committee for safety considerations and reviewed by Combustion Engineering, Inc. for conformance to the intended purpose. The Plant Superintendent will then give approval and issue the written procedures.

Initial procedures will be used in the operator training program as course material and will be included in the examinations. Subsequent to the initial training program, procedures will be distributed to a controlled distribution list. Feedback will be required from each recipient that he has read and studied the procedure and has instructed personnel under his supervision in the use of the procedure. Periodic audits will be made by plant supervision to determine the effectiveness of this instruction.

Temporary changes which do not violate the intended purpose of the original written procedure may be made provided such changes are reviewed for safety considerations and are approved by two members of the plant supervisory operating staff, at least one of whom shall hold Senior Operator License. Such changes shall be documented and reviewed at the next regular meeting of the Plant Nuclear Safety Committee and approved by the Plant Superintendent.

Significant changes will be made only with authorization from the Company Nuclear Review Board and from the AEC. Significant changes are those changes which conflict with the intent of the Operating License and Technical Specifications.

Written procedures will be reviewed by the Plant Nuclear Safety Committee and recommendation for revision made to the Plant Superintendent. Written revisions will then be approved and issued by the Plant Superintendent. Current written procedures will be available to the operator at all times.

Check off lists will be provided for procedures as required. Where practical, the procedure itself will be in the form of a check off sheet which will be signed by the operator who completed it.

Copies of all plant operating procedures and plans will be located in the plant office, control room, and the Watch Engineer and Plant Supervisor offices. In addition, copies of selected pertinent procedures will be maintained in the reactor auxiliary building and turbine operating area.

Copies of maintenance procedures will be located in the maintenance office. During fuel handling or maintenance operations in containment pertinent instructions and procedures will be maintained in the containment.

Written procedures shall include the following subjects:

a) Normal start-up, operation and shutdown of the reactor and of all systems and components involving nuclear safety of the facility.
b) Refueling operations.
c) Actions to be taken to correct specific and foreseen potential malfunctions of systems or components, including responses to alarms, suspected primary system leaks and abnormal reactivity changes.
d) Emergency conditions involving potential or actual release of radioactivity.
e) Emergency and off-normal condition procedures.
f) Preventive or corrective maintenance operations which could have an effect on the safety of the reactor.
g) Surveillance and testing requirements.
h) Plant Security Plan implementation.
i) Emergency Plans implementation.

13.5.2 OPERATING PROCEDURES

Operating procedures cover the normal and off-normal operation of all systems and components.

Normal operating procedures cover the operation of a unit from a cold shutdown condition to power and return to cold shutdown. Additional procedures cover start-up, operation and shutdown of individual systems and components. These include check lists for establishing the necessary condition of system components to perform the specific procedure, and precautions to be followed during the procedure. Radiation control procedures specify the requirements and precautions necessary to maintain good radiation protection for plant personnel and the public. Included are responsibilities, exposure limits, use of radiation work permits, protective clothing requirements and contamination control, requirements for training of personnel and standards for protection against radiation.

Off-normal condition operating procedures cover the range of equipment and system troubles. These are procedures that describe actions to be taken to prevent equipment malfunction or to prevent a perturbation from resulting in a situation of more serious consequence. Typical conditions included are excessive system leakage, pump failure, loss of off-site power, control air failure, and a stuck or faulty CEA assembly. An annunciator verification list relates the actions to be performed should off-normal conditions occur in the operation of systems or equipment. This list provides the guidelines for a preplanned course of response to alarms under certain conditions; however, the particular situation will govern the extent to which each suggested action is carried out. These guidelines include (a) the possible cause of the alarm; (b) alarm set points and signal source; (c) immediate action to be taken by the control operator; and (d) subsequent action based on off-normal procedure.

13.5.3 EMERGENCY PROCEDURES

Emergency procedures are developed by the Administrative Assistant to the Director of Power Resources to ensure that proper action is taken in the event of an emergency.

The emergency procedures provide each individual with sufficient detailed information so that he has a thorough understanding of his duties and responsibilities. Periodic drills will be conducted to maintain a high state of proficiency in performing the emergency procedures.

Two general types of Emergency Procedures will be developed:

a) Emergency Operations Procedures provide the operator with instructions required to stop the degradation of conditions and mitigate their consequences and to guide operations during potential emergencies. They will be written so that a trained operator will know in advance the expected course of events that will identify an emergency and the immediate action he should take. Since

emergencies may not follow anticipated patterns, the procedures will provide sufficient flexibility to accommodate variations.

Emergency operation procedures that cover actions for manipulations of controls to prevent accidents or lessen their consequences will be based on a general sequence of observations and actions. Emphasis will be placed on operator responses to observations and indications in the control room; that is, when immediate operator actions are required to prevent or mitigate the consequences of a serious condition, procedures will require that those actions be implemented promptly. Supplemental background information will be appended to these procedures to further aid operators in taking proper emergency actions.

The following categories of events are examples of potential emergencies for which procedures will be written:

1) Loss of coolant from identified and unidentified sources, from small loss to maximum hypothetical accident loss
2) Reactor transients and excursions
3) Failure of vital equipment
4) Loss or degradation of vital power sources
5) Emergency shutdown
6) Reactor trips
7) Abnormally high radiation levels

b) Emergency Plans Implementing Procedures will provide the following:

1) Individual assignment of authorities and responsibilities for performance of specific tasks
2) Protective action levels specified for the emergency identified that require implementation of the protective measures outlined in the Emergency Plans and the actions required for such protective measures
3) Specific actions dealing with the measures to be taken by coordinating support groups
4) Procedures for medical treatment and handling of contaminated individuals
5) Special equipment requirements for items such as medical treatment, emergency personnel removal, specific radiation detection, personnel dosimetric and rescue operations. Procedures for making this equipment available. Operating instructions for such equipment and provisions for periodic inspection and maintenance
6) Identification of communications networks for the emergency organization, including communications required for effective coordination of all support groups and notification lists
7) Description of alarm signals
8) Procedures required to restore the plant to normal following an emergency
9) Requirements for periodically testing implementing procedures, communications network, and alarm systems to assure that they function properly

13.5.4 MAINTENANCE PROCEDURES

Procedures will be written for maintenance of equipment expected to require frequent attention. Examples of such equipment are control rod drives, pump seals, important filters and strainers, diesel generator sets, major valves and steam generators. As experience is gained in operation of the plant, routine maintenance will be altered to improve equipment performance, and procedures written for repair of equipment will be improved, if required. Since the probability of failure is usually unknown and the time and mode of failure are usually unpredictable for most equipment, procedures will not be written for repair of most equipment before failure.

Radiation protection measures will be prescribed before the task begins.

Permission to release equipment or systems for maintenance will be granted by responsible operating personnel. Prior to granting permission, such operating personnel will verify that the equipment or system can be released, and, if so, how long it may be out of service.

After permission has been granted, equipment will be made safe to work on. Measures will provide for protection of equipment and workers. Equipment and systems in a controlled status will be clearly identified. Strict control measures for such equipment will be enforced.

Conditions considered in preparing equipment for maintenance will include, for example, shutdown margin; method of emergency core cooling; establishment of a path for decay heat removal; temperature and pressure of the system; valves between work and hazardous material; venting, draining, and flushing; entry into closed vessels; hazardous atmospheres; handling hazardous materials; electrical hazards; and physical barriers, as required.

The procedures will contain enough detail to permit the maintenance work to be performed safely and expeditiously.

Instructions will be included for returning the equipment to its normal operating status. Operating personnel will place the equipment in operation and verify its functional acceptability. Special attention will be given to restoration of normal conditions, such as removal of signals used in maintenance or testing, and to systems that can be defeated by leaving valves or breakers mispositioned or by leaving switches in "Test" or "Manual" positions. All jumpers will be controlled. When placed into service, the equipment will receive special surveillance until a run-in period has ended.

13.5.5 INSTRUMENT CALIBRATION AND TEST PROCEDURES

Procedures will be provided for periodic calibration and testing of plant instrumentation such as interlocks, alarm devices, sensors, signal conditioners, and protective circuits. The procedures will have provisions for meeting surveillance schedules and for assuring measurement accuracies adequate to keep safety parameters within operational and safety limits.

Procedures will be provide for proper control, periodic calibration and adjustment of measuring and test equipment to maintain accuracy within necessary limits.

13.5.6 CHEMICAL-RADIOCHEMICAL CONTROL PROCEDURES

Procedures will be provided for chemical and radiochemical control activities. They will include, for example, the nature and frequency of sampling and analyses; instructions for maintaining coolant quality within prescribed limits; limitations on concentrations of agents that could cause corrosive attack, foul heat transfer surfaces, or become sources of radiation hazards due to activation; and the control, treatment, and disposal of radioactive wastes.

13.6 PLANT RECORDS

The Plant Superintendent has direct responsibility for records management. Record books will be maintained on a shift basis to record principal operating events, any unusual occurrences and the actions taken. Pertinent daily records will be used to monitor and evaluate routine and/or specific operating conditions and periodic tests will be conducted to audit plant performance.

Complete records for the first year of operation will be retained permanently. Unit operating log sheets, recorder charts and routine daily reports will be retained for the current year and the preceding six years. Manual recording methods are used for all records except those recorded by plant instrumentation. Items to be retained include logs of plant operation and records of:

a) Normal operation (e.g., power level, fuel exposure, shutdowns)
b) Principal maintenance activities
c) Abnormal occurrences
d) Checks, inspections, tests, and calibrations of components and systems
e) Reviews of changes made to procedures or equipment or reviews of tests and experiments to comply with 10 CFR 50.59
f) Radioactive shipments
g) Gaseous and liquid radioactive waste released to environs
h) Offsite environmental monitoring surveys
i) Fuel inventories and transfers
j) Plant radiation and contamination surveys
k) Radiation exposures for all plant personnel
l) Reactor coolant system inservice inspections
m) Plant changes reflected in updated, corrected, and as-built drawings of the plant
n) Minutes of meetings of the Plant Nuclear Safety Committee and the Company Nuclear Review Board.

Retention periods of sufficient duration to assure the ability to reconstruct significant events and satisfy any statutory requirements which apply will be specified.

13.7 INDUSTRIAL SECURITY

A Security Plan and implementing procedures following the intent of AEC Safety Guide 17 will be prepared and in effect prior to loading fuel, setting forth plans and procedures for the prevention or mitigation of consequences of industrial sabotage and other acts such as vandalism, arson and civil disturbance which are inimical to the safe operation of the plant and the health and safety of the public. The Security Plan will be compatible with the Emergency Plan.

13.7.1 PERSONNEL AND PLANT DESIGN

A plant Security Supervisor reporting to the Plant Superintendent will be designated and the overall security program will be under the supervision of the Company Director of Security. The Director of Security will report to the Director of Power Resources and to the Vice President, Personnel. The Plant Security Supervisor is responsible for monitoring security administrative procedures, supervises the Security Guard Force and verifies alarm system operation.

To the extent operationally feasible, the plant will be designed and arranged to enhance industrial security and reduce the vulnerability of the plant to industrial sabotage and other acts which may adversely affect the plant operation and the health and safety of the public. When this is not feasible, compensatory steps will be taken in the form of physical barriers, administrative controls, or electrical/electronic protection devices and alarms to assure the viability of the plan. Control and warning signs designating perimeter boundaries and conditions of entry will be posted.

A personnel screening program will provide an evaluation of all employees with regard to stability, integrity, safety, reliability, ability to perform the job to which assigned, and aptitude and ability to progress to higher classification. All employees will be provided training in the Plant Industrial Security Program. The Security Guard Force provides the enforcement medium for the Security Plan, and will have the ability to qualify with side-arm.

13.7.2 SECURITY PLAN

All personnel and material entry into the plant will be controlled through the use of access coded identification badges, material passes, the Facility Access Log, security guards and escorts. The plant will be enclosed with a lighted barrier and entrance to internal vital areas such as the control building, containment, switchgear rooms, diesel generators and intake pump areas shall be controlled by alarm and electrical/electronic devices and/or security guards. Containers and vehicles are subject to inspection by the Security Force on entry to or exit from the site.

All plant personnel will be issued permanent photo identification cards which will be issued, maintained and retrieved by the Security Guard Force at the Main Gate.

The occasional visitors categories are:

a) Sightseers, delivery trucks and other occasional visitors
b) Salesmen and visitors requiring admittance to the plant office only
c) Manufacturer's Service Representatives, contractors and their employees, and AEC personnel
d) Company employees other than regular plant staff.

Category a visitors will be issued a badge that indicates that they must be escorted. They will be provided with an escort.

Category b visitors will be issued a badge identifying them as being authorized to visit the office only and will be directed to the office after receiving authorization from a member of the plant staff. Should it be necessary for them to visit other areas of the plant they will be provided with an escort and the appropriate identification badge and monitoring devices.

Category c visitors who are visiting the plant for the first time will be issued a badge identifying them as authorized to visit the office only.

Category d visitors will identify themselves to the guard and, depending on their reason for visiting the plant and the extent of their radiation training, will be treated in the same manner as Category a or c visitors.

Badges will be worn visibly when practicable at all times when within the plant area. Unfamiliar or unidentified persons on-site are subject to challenge by plant personnel.

The Security Guard Force is responsible for controlling access to the plant during emergency conditions. The Security Guard Force will receive instruction in specific situations from the Emergency Coordinator. All personnel entering the plant during emergency conditions will be accounted for, issued radiation monitoring devices and escorted, if required.

Frequent inspection trips will be made by operating personnel through operating, standby and emergency equipment areas to survey potential unauthorized status changes in equipment or structures.

Liaison will be established with Federal, state and local law enforcement agencies by the Plant Security Supervisor and/or the Director of Security. A procedure will set forth instructions for action to be taken in emergency situations involving security. Incidents involving attempted or actual breach of industrial security controls will be reported to the AEC within twenty-four hours.

Procedures will be issued relating to the Security Guard Force, Personnel Identification and Control of Movement, Employee Security Orientation Program, The Lock and Key System, Inspections and Audits, Security Emergencies and Safeguard Equipment Status Control.

The inspections and audits will occur periodically to provide reports of the security program effectiveness, to develop employee awareness and to assure training program adequacy.

Security incidents, including unauthorized entry, introduction of dangerous materials and acts affecting national defense support capabilities, will be investigated by all involved agencies and reports made to accompany officials and the AEC.