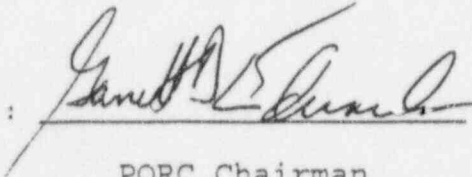


Offsite Dose Calculation Manual
Revision 9

Peach Bottom Atomic Power Station
Units 2 and 3

PECO Energy Company
Docket Nos. 50-277 & 50-278

PORC Approval :


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I. Purpose

The purpose of the Offsite Dose Calculation Manual is to establish methodologies and procedures for calculating doses to individuals in areas at and beyond the SITE BOUNDARY due to radioactive effluents from Peach Bottom Atomic Power Station. The results of these calculations are required to determine compliance with the requirements of Specification 5.5.4, "Radioactive Effluent Controls Program" of Appendix A to Operating Licenses DPR-44 and DPR-56, "Technical Specifications for Peach Bottom Atomic Power Station Units No. 2 and 3". The Radioactive Effluents Control Program is located in Appendix A of this Offsite Dose Calculation Manual and contains Offsite Dose Calculation Manual Specifications (ODCMS) and their associated Bases which implement the requirements of Technical Specification 5.5.4.

II. Setpoint Determination for Liquid & Gaseous Monitors

II.A Liquid Radwaste Activity Monitor Setpoint

Each tank of radioactive waste is sampled prior to release. A small liquid volume of this sample is analyzed for gross gamma activity in a NaI well counter. This NaI well counter activity is then converted to an equivalent liquid radwaste monitor reading.

$$\text{CPS (R/W Monitor)} = [\text{Net CPM/ml (well)} \times \text{Eff W/RW}] + \text{Background CPS}$$

Where:

$$\text{CPS (R/W Monitor)} = \text{liquid radwaste gross activity monitor reading in CPS}$$

$$\text{Net CPM/ml (well)} = \text{gross gamma activity for the radwaste sample tank [determined by the well counter]}$$

$$\text{Eff W/RW} = \text{conversion factor between well counter and liquid radwaste gross activity monitor [determined by calibrating both detectors with the same liquid radioactive source]}$$

$$\text{Background CPS} = \text{background reading of the liquid radwaste gross activity monitor in CPS}$$

Exceeding the expected response would indicate that an incorrect sample had been obtained for that release and the release is automatically stopped.

The alarm and trip pot setpoints for the liquid radwaste activity monitor are determined from a calibration curve for the alarm pot and trip pot. The alarm pot setting includes a factor of 1.25 to allow for analysis error, pot setting error, instrument error and calibration error. The trip pot setting includes a factor of 1.35 to allow for analysis error, pot setting error, instrument error and calibration error.

II.B Liquid Radwaste Release Flowrate Setpoint Determination

The trip pot setpoint for the liquid radwaste release flowrate is determined by multiplying the liquid radwaste flowrate (from Section III.A) by 1.2 and using this value on the appropriate calibration curve for the discharge flow meter to be used. The Peach Bottom radwaste system has two flow monitors - high flow (5 to 300 gpm) and low flow (0.8 to 15 gpm). The factor of 1.2 allows for pot setting error and instrument error. The flow rate determination includes a margin of assurance which includes consideration of this error such that the instantaneous release limit of 10 CFR 20 is not exceeded.

II.C Setpoint Determination for Gaseous Radwaste

The high and high-high alarm setpoints for the main stack radiation monitor, Unit 2 roof vent radiation monitor and Unit 3 roof vent radiation monitor are determined as follows:

High Alarm - the high alarm setpoint is set at approximately 3 x the normal monitor reading.

High-High Alarm - the high-high alarm setpoint is set at a release rate from this vent of approximately 30% of the instantaneous release limit of 10 CFR 20 as specified in ODCMS 3.8.C.1.a for the most restrictive case (skin or total body) on an unidentified basis. To determine these setpoints, solve the gaseous effluent dose rate equations in section IV.A of the ODCM to determine what main stack release rate and roof vent release rate will produce a dose rate of 150 mrem/yr to the total body (30% of the limit of 500 mrem/yr) and a dose rate of 900 mrem/yr to the skin (30% of the limit of 3000 mrem/yr) from each release point. Using the smallest (most restrictive) release rate for each release point determine monitor response required to produce this release rate assuming a normal vent flow rate and pressure correction factor. Set the high-high alarm for approximately this monitor response.

II.D. Setpoint Determination for Gaseous Radwaste

Flow Monitors

The alarm setpoint for the main stack flow monitor is as follows:

Low Flow Alarm - 10,000 CFM. - This setting ensures that the main stack minimum dilution flow as specified in ODCMS 3.8.C.4.a is maintained.

The alarm setpoints for the roof vent flow monitors are as follows:

Low Flow Alarm - 1.5×10^5 cfm

High Flow Alarm - 5.4×10^5 cfm

III. Liquid Pathway Dose Calculations

III.A Liquid Radwaste Release Flow Rate Determination

Peach Bottom Atomic Power Station Units 2 and 3 have one common discharge point for liquid releases. The following calculation assures that the radwaste release limits are met.

The flow rate of liquid radwaste released from the site to areas at and beyond the SITE BOUNDARY shall be such that the concentration of radioactive material after dilution shall be limited to the concentration specified in 10 CFR 20, Appendix B, Table II, Column 2 for radionuclides other than noble gases and $2E-4$ $\mu\text{Ci/ml}$ total activity concentration for all noble gases as specified in ODCMS 3.8.B.1. Each tank of radioactive waste is sampled prior to release and is quantitatively analyzed for identifiable gamma emitters as specified in Table 4.8.B.1 of the ODCMS. From this gamma isotopic analysis the maximum permissible release flow rate is determined as follows:

Determine a Dilution Factor by:

$$\text{Dilution Factor} = \sum_i \frac{\mu\text{Ci/ml } i}{ECL_i}$$

$\mu\text{Ci/ml } i$ = the activity of each identified gamma emitter in $\mu\text{Ci/ml}$

ECL_i = The effluent concentration specified in 10 CFR 20, Appendix B, Table II, Column 2 for radionuclides other than noble gases or 2×10^{-4} $\mu\text{Ci/ml}$ for noble gases.

III.A (Cont'd)

Determine the Maximum Permissible Release Rate with this Dilution Factor by:

$$\text{Release Rate (gpm)} = \frac{A \times 2.0 \times 10^5}{B \times C \times \text{Dilution Factor}}$$

- A = The number of circulating water pumps running which will provide dilution
- 2.0×10^5 = the flow rate in gpm for each circulating water pump running
- B = margin of assurance which includes consideration of the maximum error in the activity setpoint, the maximum error in the flow setpoint, and possible loss of 5 out of the 6 possible circulating water pumps during a release. The value used for B is 10.0.
- C = concentration gradient factor. The value used for C is 5.0 for discharge canal water levels less than 104' and 3.0 for canal water levels greater than 104'.

III.B ODCMS 4.8.B.2.1

Dose contributions from liquid effluents released to areas at and beyond the SITE BOUNDARY shall be calculated using the equation below. This dose calculation uses those appropriate radionuclides listed in Table III.A.1. These radionuclides account for virtually 100 percent of the total body dose and organ dose from liquid effluents.

The dose for each age group and each organ should be calculated to determine the maximum total body dose and organ dose for each quarter and the year, as appropriate. Cumulative dose files for quarterly and yearly doses should be maintained separately and the maximum total body and organ dose reported in each case. CM-1 NRC URI 88-33-01, T00353.

$$D = \sum_i \left[A_i \tau \sum_{l=1}^n \Delta t_l C_{il} F_l \right]$$

where:

- D = The cumulative dose commitment to the total body or any organ, from liquid effluents for the total time period $\sum_{l=1}^n \Delta t_l$, in mrem.
- Δt_l = The length of the lth time period over which C_{il} and F_l are averaged for the liquid release, in hours.

III.B (Cont'd)

- C_{il} = The average concentration of radionuclide, i , in undiluted liquid effluent during time period Δt from any liquid release, (determined by the effluent sampling analysis program, ODCMS Table 4.8.B.1), in $\mu\text{Ci/ml}$.
- $A_i \tau$ = The site related ingestion dose commitment factor to the total body or organ, τ , for each radionuclide listed in Table III.A.1, in mrem-ml per hr- μCi . See Site Specific Data.**
- F_i = The near field average dilution factor for C_{il} during any liquid effluent release. Defined as the ratio of the maximum undiluted liquid waste flow during release to the average flow through the discharge structure.

III.C ODCMS 4.8.B.4.1

Projected dose contributions from liquid effluents shall be calculated using the methodology described in section III.B.

** See Note 1 in Bases

TABLE III.A.1

LIQUID EFFLUENT INGESTION DOSE FACTORS
(DECAY CORRECTED)
A₁₇ DOSE FACTOR (MREM-ML PER HR- μ Ci)

<u>RADIO- NUCLIDE</u>	<u>TOTAL BODY</u>		
	<u>ADULT</u>	<u>TEEN</u>	<u>CHILD</u>
H-3	2.13E+00	1.53E+00	2.70E+00
NA-24	1.65E+02	1.70E+02	1.98E+02
P-32	5.93E+04	6.49E+04	8.33E+04
MN-54	9.82E+02	1.00E+03	1.08E+03
FE-55	1.31E+02	1.40E+02	1.96E+02
FE-59	1.14E+03	1.17E+03	1.36E+03
CO-58	2.59E+02	2.62E+02	3.17E+02
CO-60	7.40E+02	7.48E+02	9.07E+02
ZN-65	3.87E+04	3.95E+04	4.16E+04
SR-89	8.83E+02	9.45E+02	1.48E+03
SR-90	1.88E+05	1.56E+05	1.72E+05
TE-129M	2.01E+03	2.17E+03	2.79E+03
TE-131M	4.57E+02	4.81E+02	5.74E+02
TE-132	1.40E+03	1.44E+03	1.65E+03
I-131	1.86E+02	1.79E+02	2.36E+02
I-133	1.97E+01	2.03E+01	3.20E+01
CS-134	6.74E+05	3.88E+05	1.49E+05
CS-136	9.79E+04	9.15E+04	7.30E+04
CS-137	3.98E+05	2.20E+05	8.49E+04
BA-140	3.66E+01	3.62E+01	7.42E+01

NOTE: The listed dose factors are for radionuclides that may be detected in liquid effluents and have significant dose consequences. The factors are decayed for one day to account for the time between effluent release and ingestion of fish by the maximum exposed individual.

TABLE III.A.1
LIQUID EFFLUENT INGESTION DOSE FACTORS
(DECAY CORRECTED)

A_1 DOSE FACTOR (MREM-ML PER HR- μ Ci)

<u>RADIO- NUCLIDE</u>	<u>LIVER</u>		
	<u>ADULT</u>	<u>TEEN</u>	<u>CHILD</u>
H-3	2.13E+00	1.53E+00	2.70E+00
NA-24	1.65E+02	1.70E+02	1.98E+02
P-32	9.55E+04	1.04E+05	1.01E+05
MN-54	5.15E+03	5.06E+03	4.03E+03
FE-55	5.62E+02	6.01E+02	6.33E+02
FE-59	2.96E+03	3.02E+03	2.73E+03
CO-58	1.16E+02	1.14E+02	1.04E+02
CO-60	3.35E+02	3.32E+02	3.07E+02
ZN-65	8.55E+04	8.46E+04	6.69E+04
SR-89	no data	no data	no data
SR-90	no data	no data	no data
TE-129M	4.74E+03	5.09E+03	5.02E+03
TE-131M	5.48E+02	5.77E+02	5.40E+02
TE-132	1.48E+03	1.53E+03	1.36E+03
I-131	3.25E+02	3.32E+02	4.16E+02
I-133	6.48E+01	6.66E+01	8.45E+01
CS-134	8.25E+05	8.36E+05	7.06E+05
CS-136	1.36E+05	1.36E+05	1.13E+05
CS-137	6.07E+05	6.32E+05	5.75E+05
BA-140	7.00E-01	6.90E-01	1.11E+00

NOTE: The listed dose factors are for radionuclides that may be detected in liquid effluents and have significant dose consequences. The factors are decayed for one day to account for the time between effluent release and ingestion of fish by the maximum exposed individual.

TABLE III.A.1LIQUID EFFLUENT INGESTION DOSE FACTORS
(DECAY CORRECTED) A_1 DOSE FACTOR (MREM-ML PER HR- μ Ci)

<u>RADIO- NUCLIDE</u>	<u>BONE</u>		
	<u>ADULT</u>	<u>TEEN</u>	<u>CHILD</u>
H-3	no data	no data	no data
NA-24	1.65E+02	1.70E+02	1.98E+02
P-32	2.38E+05	2.58E+05	3.35E+05
MN-54	no data	no data	no data
FE-55	8.12E+02	8.47E+02	1.19E+03
FE-59	1.26E+03	1.30E+03	1.68E+03
CO-58	no data	no data	no data
CO-60	no data	no data	no data
ZN-65	2.69E+04	2.43E+04	2.51E+04
SR-89	3.08E+04	3.30E+04	5.19E+04
SR-90	7.67E+05	6.31E+05	6.78E+05
TE-129M	1.27E+04	1.37E+04	1.80E+04
TE-131M	1.12E+03	1.21E+03	1.56E+03
TE-132	2.29E+03	2.42E+03	3.07E+03
I-131	2.28E+02	2.38E+02	4.13E+02
I-133	3.72E+01	3.92E+01	6.84E+01
CS-134	3.47E+05	3.55E+05	4.30E+05
CS-136	3.45E+04	3.46E+04	4.10E+04
CS-137	4.44E+05	4.75E+05	6.01E+05
BA-140	5.57E+02	5.63E+02	1.27E+03

NOTE: The listed dose factors are for radionuclides that may be detected in liquid effluents and have significant dose consequences. The factors are decayed for one day to account for the time between effluent release and ingestion of fish by the maximum exposed individual.

TABLE III.A.1

LIQUID EFFLUENT INGESTION DOSE FACTORS
(DECAY CORRECTED)
A₁₇ DOSE FACTOR (MREM-ML PER HR- μ Ci)

KIDNEY

<u>RADIO- NUCLIDE</u>	<u>ADULT</u>	<u>TEEN</u>	<u>CHILD</u>
H-3	2.13E+00	1.53E+00	2.70E+00
NA-24	1.65E+02	1.70E+02	1.98E+02
P-32	no data	no data	no data
MN-54	1.53E+03	1.51E+03	1.13E+03
FE-55	no data	no data	no data
FE-59	no data	no data	no data
CO-58	no data	no data	no data
CO-60	no data	no data	no data
ZN-65	5.72E+04	5.41E+04	4.22E+04
SR-89	no data	no data	no data
SR-90	no data	no data	no data
TE-129M	5.31E+04	5.74E+04	5.29E+04
TE-131M	5.55E+03	6.01E+03	5.22E+03
TE-132	1.43E+04	1.47E+04	1.27E+04
I-131	5.57E+02	5.73E+02	6.82E+02
I-133	1.12E+02	1.16E+02	1.41E+02
CS-134	2.67E+05	2.66E+05	2.19E+05
CS-136	7.57E+04	7.42E+04	6.00E+04
CS-137	2.06E+05	2.15E+05	1.87E+05
BA-140	2.38E-01	2.34E-01	3.62E-01

NOTE: The listed dose factors are for radionuclides that may be detected in liquid effluents and have significant dose consequences. The factors are decayed for one day to account for the time between effluent release and ingestion of fish by the maximum exposed individual.

TABLE III.A.1

LIQUID EFFLUENT INGESTION DOSE FACTORS
(DECAY CORRECTED)
A₁₇ DOSE FACTOR (MREM-ML PER HR- μ Ci)

<u>RADIO- NUCLIDE</u>	<u>GI-LLI</u>		
	<u>ADULT</u>	<u>TEEN</u>	<u>CHILD</u>
H-3	2.13E+00	1.53E+00	2.70E+00
NA-24	1.65E+02	1.70E+02	1.98E+02
P-32	1.73E+05	1.41E+05	5.98E+04
MN-54	1.58E+04	1.04E+04	3.38E+03
FE-55	3.22E+02	2.60E+02	1.17E+02
FE-59	9.90E+03	7.15E+03	2.84E+03
CO-58	2.35E+03	1.56E+03	6.04E+02
CO-60	6.30E+03	4.33E+03	1.70E+03
ZN-65	5.38E+04	3.58E+04	1.18E+04
SR-89	4.94E+03	3.93E+03	2.01E+03
SR-90	2.22E+04	1.77E+04	9.13E+03
TE-129M	6.40E+04	5.15E+04	2.19E+04
TE-131M	5.44E+04	4.63E+04	2.19E+04
TE-132	7.02E+04	4.85E+04	1.37E+04
I-131	8.58E+01	6.57E+01	3.70E+01
I-133	5.82E+01	5.03E+01	3.40E+01
CS-134	1.44E+04	1.04E+04	3.80E+03
CS-136	1.55E+04	1.09E+04	3.96E+03
CS-137	1.18E+04	9.00E+03	3.60E+03
BA-140	1.15E+03	8.69E+02	6.43E+02

NOTE: The listed dose factors are for radionuclides that may be detected in liquid effluents and have significant dose consequences. The factors are decayed for one day to account for the time between effluent release and ingestion of fish by the maximum exposed individual.

IV. Gaseous Pathway Dose CalculationsIV.A. ODCMS 4.8.C.1.1 and 4.8.C.1.2

The dose rate in areas at and beyond the SITE BOUNDARY due to radioactive materials released in gaseous effluents shall be determined by the expressions below:

IV.A.1 Noble Gases:

The dose rate from radioactive noble gas releases shall be determined by either of two methods. Method (a), the Gross Release Method, assumes that all noble gases released are the most limiting nuclide - Kr-88 for total body dose (vent and stack releases) and skin dose (vent releases) and Kr-87 for skin dose (stack releases). Method (b), the Isotopic Analysis Method, utilizes the results of noble gas analyses required by ODCMS 4.8.C.1.1.

For normal operations, it is expected that method (a) will be used. However, if noble gas releases are close to the limits as calculated by method (a), method (b) can be used to allow more operating flexibility by using data that more accurately reflect actual releases.

a. Gross Release Method

$$D_{TB} = V \dot{Q}_{NS} + K (\overline{X/Q})_V \dot{Q}_{NV}$$

$$D_S = [L_S (\overline{X/Q})_S + 1.1B] \dot{Q}_{NS} + [L_V + 1.1M] (\overline{X/Q})_V \dot{Q}_{NV}$$

where:

The location is the site boundary, 1097m SSE from the vents. This location results in the highest calculated dose to an individual from noble gas releases.

D_{TB} = total body dose rate, in mrem/yr.

D_S = skin dose rate, in mrem/yr.

V = 4.72×10^4 mrem/yr per $\mu\text{Ci/sec}$; the constant for Kr-88 accounting for the gamma radiation from the elevated finite plume. This constant was developed using MARE program with plant specific inputs for PBAPS.

IV.A.1.a (Cont'd)

- \dot{Q}_{NS} = The gross release rate of noble gases from the stack determined by gross activity stack monitors averaged over one hour, in $\mu\text{Ci/sec}$.
- K = 1.47×10^4 mrem/yr per $\mu\text{Ci/m}^3$; the total body dose factor due to gamma emissions for Kr-88 (Reg. Guide 1.109, Table B-1).
- $(\overline{X/Q})_V$ = 5.33×10^{-7} sec/ m^3 ; the highest calculated annual average relative concentration for any area at or beyond the SITE BOUNDARY for all vent releases.
- \dot{Q}_{NV} = The gross release rate of noble gases in gaseous effluents from vent releases determined by gross activity vent monitors averaged over one hour, in $\mu\text{Ci/sec}$.
- L_V = 2.37×10^3 mrem/yr per $\mu\text{Ci/m}^3$; the skin dose factor due to beta emissions for Kr-88. (Reg. Guide 1.109, Table B-1).
- L_S = 9.73×10^3 mrem/yr per $\mu\text{Ci/m}^3$; the skin dose factor due to beta emissions for Kr-87. (Reg. Guide 1.109, Table B-1).
- $(\overline{X/Q})_S$ = 9.97×10^{-8} sec/ m^3 ; the highest calculated annual average relative concentration from the stack releases for any area at or beyond the SITE BOUNDARY.
- B = 1.74×10^{-4} mrad/yr per $\mu\text{Ci/sec}$; the constant for Kr-87 accounting for the gamma radiation from the elevated finite plume. This constant was developed using MARE program with plant specific inputs for PBAPS.
- M = 1.52×10^4 mrad/yr per $\mu\text{Ci/m}^3$; the air dose factor due to gamma emissions for Kr-88. (Reg. Guide 1.109, Table B-1).
- 1.1 = Unit conversion, converts air dose to skin dose, mrem/mrad.

IV.A.1. b. Isotopic Analysis Method

$$D_{TB} = \sum_i (V_i \dot{Q}_{is} + K_i (\overline{X/Q})_v \dot{Q}_{iv})$$

$$D_s = \sum_i [(L_i (\overline{X/Q})_s + 1.1B_i) \dot{Q}_{is} + (L_i + 1.1M_i) (\overline{X/Q})_v (\dot{Q}_{iv})]$$

where:

The location is the site boundary, 1097m SSE from the vents. This location results in the highest calculated dose to an individual from noble gas releases.

D_{TB} = total body dose rate, in mrem/yr.

D_s = skin dose, in mrem/yr.

V_i = The constant for each identified noble gas radionuclide for the gamma radiation from the elevated finite plume. The constants were developed using the MARE program with plant specific inputs for PBAPS. Values are listed on Table IV.A.1, in mrem/yr per $\mu\text{Ci/sec}$.

\dot{Q}_{is} = The release rate of noble gas radionuclide, i, in gaseous effluents from the stack determined by isotopic analysis averaged over one hour, in $\mu\text{Ci/sec}$.

K_i = The total body dose factor due to gamma emissions for each identified noble gas radionuclide. Values are listed on Table IV.A.1, in mrem/yr per $\mu\text{Ci/m}^3$.

$(\overline{X/Q})_v$ = $5.33 \times 10^{-7} \text{ sec/m}^3$; the highest calculated annual average relative concentration for any area at or beyond the SITE BOUNDARY for all vent releases.

\dot{Q}_{iv} = The release rate of noble gas radionuclide, i, in gaseous effluents from all vent releases determined by isotopic analysis averaged over one hour, in $\mu\text{Ci/sec}$.

L_i = The skin dose factor due to beta emissions for each identified noble gas radionuclide. Values are listed on Table IV.A.1, in mrem/yr per $\mu\text{Ci/m}^3$.

IV.A.1.b (Cont'd)

- $(\overline{X/Q})_s$ = 9.97×10^{-8} sec/m³; the highest calculated annual average relative concentration from the stack releases for any area at or beyond the SITE BOUNDARY.
- B_i = The constant for each identified noble gas radionuclide accounting for the gamma radiation from the elevated finite plume. The constants were developed using MARE program with plant specific inputs for PBAPS. Values are listed on Table IV.A.1, in mrad/yr per μ Ci/sec.
- M_i = The air dose factor due to gamma emissions for each identified noble gas radionuclide. Values are listed on Table IV.A.1, in mrad/yr per μ Ci/m³.
- 1 1 = Unit conversion, converts air dose to skin dose, mrem/mrad.

TABLE IV.A.1 - Constants for Isotopic Analysis Method
(corrected for decay during transit)

Radionuclide	Plume-Air Dose Factor B_i (mrad/yr per $\mu\text{Ci/sec}$)	Total Body Dose Factor K_i (mrem/yr per $\mu\text{Ci/m}^3$)	Skin Dose Factor L_i (mrem/yr per $\mu\text{Ci/m}^3$)	Gamma Air Dose Factor M_i (mrad/yr per $\mu\text{Ci/m}^3$)	Beta Air Dose Factor N_i (mrad/yr per $\mu\text{Ci/m}^3$)	Plume-Body Dose Factor V_i (mrem/yr per $\mu\text{Ci/sec}$)
Kr-85m	4.02E-05	1.17E+03	1.46E+03	1.23E+03	1.97E+03	3.76E-05
Kr-87	1.74E-04	5.92E+03	9.73E+03	6.17E+03	1.03E+04	1.66E-04
Kr-88	4.90E-04	1.47E+04	2.37E+03	1.52E+04	2.93E+03	4.72E-04
Xe-133	1.19E-05	2.94E+02	3.06E+02	3.53E+02	1.05E+03	1.11E-05
Xe-133m	1.09E-05	2.51E+02	9.94E+02	3.27E+02	1.48E+03	1.01E-05
Xe-135	6.37E-05	1.81E+03	1.86E+03	1.92E+03	2.46E+03	5.95E-05
Xe-135m	6.61E-05	2.53E+03	5.76E+02	2.72E+03	5.99E+02	6.17E-05
Xe-138	1.52E-04	6.98E+03	3.26E+03	7.28E+03	3.75E+03	1.46E-04

The values K_i , L_i , M_i , and N_i are taken from Reg. Guide 1.109, Table B-1. The values B_i and V_i were developed using the MARE program with plant specific inputs for PBAPS.

IV.A.2 Iodine-131, iodine-133, tritium and radioactive materials in particulate form, other than noble gases, with half-lives greater than eight days.

The dose rate shall be determined by either of two methods. Method (a), the Iodine-131 Method, uses the iodine-131 releases and a correction factor to calculate the dose rate from all nuclides released. Method (b), the Isotopic Analysis Method, utilizes all applicable nuclides.

For normal operations, it is expected that Method (a) will be used since iodine-131 dominates the critical pathway - thyroid. However, in the event iodine-131 releases are minimal (e.g., during long term shutdown) Method (b) will be used to provide accurate calculations. In the absence of iodine-131 releases, the lung is the critical organ.

a. Iodine-131 Method

$$D_T = (CF) P_I [W_S \dot{Q}_{IS} + W_V \dot{Q}_{IV}]$$

where:

The location is the site boundary, 1097m SSE from the vents.

D_T = dose rate to the thyroid, in mrem/yr.

CF = 1.09; the correction factor accounting for the use of iodine-131 in lieu of all radionuclides released in gaseous effluents including iodine-133.

P_I = 1.624×10^7 mrem/yr per $\mu\text{Ci}/\text{m}^3$; the dose parameter for I-131 via the inhalation pathways. The dose factor is based on the critical individual organ, thyroid, and most restrictive age group, child. All values are from Reg. Guide 1.109 (Tables E-5 and E-9).

W_S = 1.03×10^{-7} sec/ m^3 ; the highest calculated annual average relative concentration for any area at or beyond the SITE BOUNDARY from stack releases. (SSE boundary)

\dot{Q}_{IS} = The release rate of iodine-131 in gaseous effluents from the stack determined by the effluent sampling and analysis program (ODCMS Table 4.8.C.1) in $\mu\text{Ci}/\text{sec}$.

W_V = 4.78×10^{-7} sec/ m^3 ; the highest calculated annual average relative concentration for any area at or beyond the SITE BOUNDARY for all vent releases (SSE boundary).

IV.A.2. a. (Cont'd)

\dot{Q}_{IV} = The release rate of iodine-131 in gaseous effluents from all vent releases, determined by the effluent sampling and analysis program (ODCMS Table 4.8.C.1) in $\mu\text{Ci/sec}$.

IV.A.2. b. Isotopic Analysis Method

$$D_L = \sum_i P_i [W_S \dot{Q}_{is} + W_V \dot{Q}_{iv}]$$

where:

The location is the site boundary, 1097m SSE from the vents.

D_L = dose rate to the lung, in mrem/yr.

P_i = The dose parameter for radionuclides other than noble gases for the inhalation pathway. The dose factors are based on the critical individual organ-lung, and most restrictive age group-child. All values are from Reg. Guide 1.109 (Tables E-5 and E-9). Values are listed on Table IV.A.2, in mrem/yr per $\mu\text{Ci/m}^3$.

W_S = 1.03×10^{-7} sec/ m^3 ; the highest calculated annual average relative concentration for any area at or beyond the SITE BOUNDARY from stack releases. (SSE boundary)

\dot{Q}_{is} = The release rate of radionuclides; i, in gaseous effluents from the stack determined by the effluent sampling and analysis program (ODCMS Table 4.8.C.1) in $\mu\text{Ci/sec}$.

W_V = 4.78×10^{-7} sec/ m^3 ; the highest calculated annual average relative concentration for any area at or beyond the SITE BOUNDARY for all vent releases. (SSE boundary)

\dot{Q}_{iv} = The release rate of radionuclides, i, in gaseous effluents from all vent releases, determined by the effluent sampling and analysis program (ODCMS Table 4.8.C.1) in $\mu\text{Ci/sec}$.

TABLE IV.A.2 - CONSTANTS FOR ISOTOPIC ANALYSIS METHOD(mrem/yr per $\mu\text{Ci}/\text{m}^3$)

<u>Radionuclide</u>	<u>Pi - Inhalation Lung Dose Factor</u>
Mn-54	1.58×10^6
Cr-51	1.70×10^4
Co-58	1.11×10^6
Co-60	7.07×10^6
Zn-65	9.95×10^5
Sr-89	2.16×10^6
Sr-90	1.48×10^7
Ce-141	5.44×10^5
Cs-134	1.21×10^5
Cs-137	1.04×10^5
Ba-140	1.74×10^6

IV.B. ODCMS 4.8.C.2.1

The air dose in areas at and beyond the SITE BOUNDARY due to noble gases released in gaseous effluents shall be determined by the expressions below.

The air dose shall be determined by either of two methods. Method (a), the Gross Release Method, assumes that all noble gases released are the most limiting nuclide - Kr-88 for gamma radiation and Kr-87 for beta radiation. Method (b), the Isotopic Analysis Method, utilizes the results of noble gas analyses required by ODCMS 4.8.C.1.1.

For normal operations, it is expected that Method (a) will be used. However, if noble gas releases are close to the limits as calculated by Method (a), Method (b) can be used to allow more operating flexibility by using data that more accurately reflect actual releases.

IV.B.1 for gamma radiation:

a. Gross Release Method

$$D_{\gamma} = 3.17 \times 10^{-8} [M (\overline{X/Q})_v \overline{Q}_v + B \overline{Q}_s]$$

where:

The location is the SITE BOUNDARY 1097m SSE from the vents. This location results in the highest calculated gamma air dose from noble gas releases.

D_{γ} = gamma air dose, in mrad.

3.17×10^{-8} = years per second.

M = 1.52×10^4 mrad/yr per $\mu\text{Ci}/\text{m}^3$; the air dose factor due to gamma emissions for Kr-88. (Reg. Guide 1.109, Table B-1)

$(\overline{X/Q})_v$ = 5.33×10^{-7} sec/ m^3 ; the highest calculated annual average relative concentration from vent releases for any area at or beyond the SITE BOUNDARY.

\overline{Q}_v = The gross release of noble gas radionuclides in gaseous effluents from all vents, determined by gross activity vent monitors, in μCi . Releases shall be cumulative over the calendar quarter or year as appropriate.

IV.B.1. a (Cont'd)

B = 4.90×10^{-4} mrad/year per $\mu\text{Ci/sec}$; the constant for Kr-88 accounting for the gamma radiation from the elevated finite plume. The constant was developed using the MARE program with plant specific inputs for PBAPS.

\bar{Q}_s = The gross release of noble gas radionuclides in gaseous releases from the stack determined by gross activity stack monitor in μCi . Releases shall be cumulative over the calendar quarter or year as appropriate.

b. Isotopic Analysis Method

$$D_y = 3.17 \times 10^{-8} \sum_i [M_i (\bar{X}/\bar{Q})_v \bar{Q}_{iv} + B_i \bar{Q}_{is}]$$

where:

The location is the SITE BOUNDARY, 1097m SSE from the vents. This location results in the highest calculated gamma air dose from noble gas releases.

D_y = gamma air dose, in mrad.

3.17×10^{-8} = years per second.

M_i = The air dose factor due to gamma emissions for each identified noble gas radionuclide. Values are listed on Table IV.A.1, in mrad/yr per $\mu\text{Ci/m}^3$.

$(\bar{X}/\bar{Q})_v$ = 5.33×10^{-7} sec/ m^3 ; the highest calculated average relative concentration from vent releases for any area at or beyond the SITE BOUNDARY.

\bar{Q}_{iv} = The release of noble gas radionuclides, i , in gaseous effluents from all vents as determined by isotopic analysis, in μCi . Releases shall be cumulative over the calendar quarter or year, as appropriate.

B_i = The constant for each identified noble gas radionuclide accounting for the gamma radiation for the elevated finite plume. The constants were developed using the MARE program with plant specific inputs for PBAPS. Values are listed on Table IV.A.1, in mrad/yr per $\mu\text{Ci/sec}$.

IV.B.1. b. (Cont'd)

\bar{Q}_{is} = The release of noble gas radionuclides, i, in gaseous effluents from the stack determined by isotopic analysis, in μCi . Releases shall be cumulative over the calendar quarter or year, as appropriate.

IV.B.2. for beta radiation:

a. Gross Release Method

$$D_{\beta} = 3.17 \times 10^{-8} N [(\bar{X}/\bar{Q})_v \bar{Q}_v + (\bar{X}/\bar{Q})_s \bar{Q}_s]$$

where:

The location is the SITE BOUNDARY 1097m SSE from the vents. This location results in the highest calculated gamma air dose from noble gas releases.

D_{β} = beta air dose, in mrad.

3.17×10^{-8} = years per second.

N = 1.03×10^4 mrad/yr per $\mu\text{Ci}/\text{m}^3$; the air dose factor due to beta emissions for Kr-87.
(Reg. Guide 1.109, Table B-1)

$(\bar{X}/\bar{Q})_v$ = 5.33×10^{-7} sec/ m^3 ; the highest calculated annual average relative concentration from vent releases for any area at or beyond the SITE BOUNDARY.

\bar{Q}_v = The gross release of noble gas radionuclides in gaseous effluents from all vents determined by gross activity vent monitors, in μCi . Releases shall be cumulative over the calendar quarter or year, as appropriate.

$(\bar{X}/\bar{Q})_s$ = 9.97×10^{-8} sec/ m^3 ; the highest calculated annual average relative concentration from the stack releases for any area at or beyond the SITE BOUNDARY.

\bar{Q}_s = The gross release of noble gas radionuclides in gaseous releases from the stack determined by gross activity stack monitors, in μCi . Releases shall be cumulative over the calendar quarter or year, as appropriate.

IV.B.2. b. Isotopic Analysis Method

$$D_{\beta} = 3.17 \times 10^{-8} \sum_i N_i [(\overline{X/Q})_v \overline{Q}_{iv} + (\overline{X/Q})_s \overline{Q}_{is}]$$

3.17×10^{-8} = years per second.

N_i = The air dose factor due to beta emissions for each identified noble gas radionuclide. Values are listed on Table IV.A.1, in mrad/yr per $\mu\text{Ci}/\text{m}^3$.

$(\overline{X/Q})_v$ = 5.33×10^{-7} sec/ m^3 ; the highest calculated annual average relative concentration from vent releases for any area at or beyond the SITE BOUNDARY.

\overline{Q}_{iv} = The release of noble gas radionuclide, i, in gaseous effluents from all vents as determined by isotopic analysis, in μCi . Releases shall be cumulative over the calendar quarter or year, as appropriate.

$(\overline{X/Q})_s$ = 9.97×10^{-8} sec/ m^3 ; the highest calculated annual average relative concentration from the stack releases for any area at or beyond the SITE BOUNDARY.

\overline{Q}_{is} = The release of noble gas radionuclide, i, in gaseous effluents from the stack as determined by isotopic analysis, in μCi . Releases shall be cumulative over the calendar quarter or year, as appropriate.

IV.C ODCMS 4.8.C.3.1

The dose to an individual from iodine-131, iodine-133, tritium and radioactive materials in particulate form and radionuclides other than noble gases with half-lives greater than eight days in gaseous effluents released to areas at and beyond the SITE BOUNDARY.

The dose shall be determined by one of two methods. Method (a), the Iodine-131 Method, uses the iodine-131 releases and a correction factor to calculate the dose from all nuclides released. Method (b), the Isotopic Analysis Method, utilizes all applicable nuclides.

For normal operation, it is expected that Method (a) will be used since iodine-131 dominates the critical pathway - thyroid. However, in the event iodine-131 releases are minimal (e.g. during long term shutdown) Method (b) will be used to provide accurate calculations. In the absence of iodine-131 releases, the liver is the critical organ.

IV.C. a. Iodine - 131 Method

$$D_T = 3.17 \times 10^{-8} (CF) (0.5) R [W_S \bar{Q}_{IS} + W_V \bar{Q}_{IV}]$$

where:

Location is the critical pathway dairy 2103m SSW from vents.

D_T = critical organ dose, thyroid, from all pathways, in mrem.

3.17×10^{-8} = years per second.

CF = 1.09; the correction factor accounting for the use of Iodine-131 in lieu of all radio-nuclides released in gaseous effluents including Iodine-133.

0.5 = fraction of iodine releases which are nonelemental.

R = 3.08×10^{11} m² (mrem/yr) per μ Ci/sec; the dose factor for iodine-131. The dose factor is based on the critical individual organ, thyroid, and most restrictive age group, infant. See Site Specific Data.**

W_S = 4.95×10^{-10} meters⁻²; (\bar{D}/\bar{Q}) for the food pathway for stack releases.

\bar{Q}_{IS} = The release of iodine-131 from the stack determined by the effluent sampling and analysis program (ODCMS Table 4.8.C.1), in μ Ci. Releases shall be cumulative over the calendar quarter or year, as appropriate.

W_V = 1.14×10^{-9} meters⁻²; (\bar{D}/\bar{Q}) for the food pathway for vent releases.

\bar{Q}_{IV} = The release of iodine-131 from the vent determined by the effluent sampling and analysis program (ODCMS Table 4.8.C.1), in μ Ci. Releases shall be cumulative over the calendar quarter or year, as appropriate.

**See Note 2 in Bases.

IV.C. b. Isotopic Analysis Method

$$D_L = 3.17 \times 10^{-8} \sum_i R_i [W_S \bar{Q}_{is} + W_V \bar{Q}_{iv}]$$

where:

Location is the critical pathway dairy 2103m SSW from vents.

D_L = critical organ dose, liver, from all pathways, in mrem.

3.17×10^{-8} = years per second.

R_i = The dose factor for each identified radionuclide, i, based on the critical individual organ, liver and most restrictive age group, infant. Values are listed on Table IV.C.1, in m^2 (mrem/yr) per $\mu\text{Ci/sec}$.

W_S = 4.95×10^{-10} meters⁻²; (\bar{D}/\bar{Q})
for the food pathway for stack releases.

\bar{Q}_{is} = The release of radionuclides, i, in gaseous effluents from the stack determined by the effluent sampling (ODCMS Table 4.8.C.1), in μCi . Releases shall be cumulative over the calendar quarter or year, as appropriate.

W_V = 1.14×10^{-9} meters⁻²; (\bar{D}/\bar{Q})
for the food pathway for vent releases.

\bar{Q}_{iv} = The release of radionuclides, i, in gaseous effluents from the vents determined by the effluent sampling and analysis program (ODCMS Table 4.8.C.1) in μCi . Release shall be cumulative over the calendar quarter or year, as appropriate.

TABLE IV.C.1 - CONSTANTS FOR ISOTOPIC ANALYSIS METHOD(m² (mrem/yr) per μ Ci/sec)

<u>RADIONUCLIDE</u>	<u>R_i</u>
Mn-54	1.93×10^7
Cr-51	5.14×10^4 *
Co-58	9.58×10^6
Co-60	4.69×10^7
Zn-65	9.21×10^9
Sr-89	1.31×10^8 *
Sr-90	1.67×10^{10} *
Ce-141	8.68×10^3
Cs-134	3.54×10^{10}
Cs-137	3.24×10^{10}
Ba-140	7.09×10^4

* There is no liver dose factor given in R.G. 1.109 for these nuclides. Therefore, the whole body dose factor was used.

IV.D ODCMS 4.8.C.5.1

The projected doses from releases of gaseous effluents to areas at and beyond the SITE BOUNDARY shall be calculated in accordance with the following sections of this manual:

- a. gamma air dose - IV.B.1
- b. beta air dose - IV.B.2
- c. organ dose - IV.C

The projected dose calculation shall be based on expected release from plant operation. The normal release pathways result in the maximum releases from the plant. Any alternative release pathways result in lower releases and, therefore, lower doses.

IV.E Technical Requirements Manual Test Requirement (TR) 3.5.3

IV.E.1 The two types of recombiner hydrogen analyzers currently used at Peach Bottom are:

- a. Hays Thermal Conductivity type (Analyzers 4083, 4084, 4029, and 4047 on Unit 2. Analyzers 5083, 5084, 5029 and 5057 on Unit 3.)
- b. Scott Series 900 Helium - Immune type (Analyzers 4083X and 4084X on Unit 2. Analyzers 5083X and 5084X on Unit 3.)

These analyzers will be replaced, per Mod P61, with the Whittaker Electrochemical type (Analyzers 4083A and 4083B on Unit 2. Analyzers 5083A and 5083B on Unit 3).

IV.E.2 The calibration gases for the three types are:

- a. Hays Analyzers
Zero Gas - Air
Calibration Gas - 4% Hydrogen, Balance Nitrogen
- b. Scott Analyzers
Zero Gas - Air
Calibration Gas - 2% Hydrogen, Balance Air
- c. Whittaker Analyzers
Calibration Gas - 2% Hydrogen, Balance Air

V.A. ODCMS 4.8.D.1.1 and 4.8.D.1.2

If the doses as calculated by the equations in this manual do not exceed the limits given in ODCMS 3.8.B.2, 3.8.C.2, or 3.8.C.3 by more than two times, the conditions of ODCMS 3.8.D.1 have been met.

V.A. (Cont'd)

If the doses as calculated by the equations in this manual exceed the limits given in ODCMS 3.8.B.2, 3.8.C.2, or 3.8.C.3 by more than two times, the maximum dose or dose commitment to a real individual shall be determined utilizing the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977. Any deviations from the methodology provided in Regulatory Guide 1.109 shall be documented in the Special Report to be prepared in accordance with ODCMS 3.8.D.1.

The cumulative dose contribution from direct radiation from the two reactors at the site and from radwaste storage shall be determined by the following methods:

Cumulative dose contribution from direct radiation =
 Total dose at the site of interest (as evaluated by TLD measurement) -
 Mean of background dose (as evaluated by TLD's at background sites) -
 Effluent contribution to dose (as evaluated by ODCMS 4.8.D.1.1).

This evaluation is in accordance with ANSI/ANS 6.6.1-1979 Section 7. The error using this method is estimated to be approximately 8%.

VI.A. Unique Reporting Requirement ODCMS 3.10.3, Dose Calculations for the Radiation Dose Assessment Report

The assessment of radiation doses for the radiation dose assessment report shall be performed utilizing the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses To Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977. Any deviations from the methodology provided in Regulatory Guide 1.109 shall be documented in the radiation dose assessment report.

The meteorological conditions concurrent with the time of release of radioactive materials (as determined by sampling frequency of measurement) or approximate methods shall be used as input to the dose model.

The Radiation Dose Assessment Report shall be submitted within 120 days after January 1 of each year in order to allow time for the calculation of radiation doses following publication of radioactive releases in the Radioactive Effluent Release Report. There is a very short turnaround time between the determination of all radioactive releases and publication of the Radioactive Effluent Release Report. This would not allow time for calculation of radiation doses in time for publication in the same report.

VII.A ODCMS 4.8.E.1.1 and 4.8.E.1.2

The radiological environment monitoring samples shall be collected pursuant to Table VII.A.1 from the locations shown on Figures VII.A.1, VII.A.2, and VII.A.3, and shall be analyzed pursuant to the requirements of Table VII.A.1.

TABLE VII.A-1

ODCM - Peach Bottom Atomic Power Station
Radiological Environmental Monitoring Program

Pathway	Station Code	Distance and Direction from PBAPS Vents	Collection Method and Discusion	Analyses	
I. <u>Direct Radiation</u>					
Site Boundary	1L	1,256 feet NE of site	1 set of four (4) TLDs from each location at least quarterly Gamma dose quarterly	Gamma Dose quarterly	
	1A	1,396 feet SE of site			
	2	4,661 feet SE of site			
	1I	2,851 feet SSE of site			
	1C	4,513 feet SSE of site	TLD sites were chosen in accordance with Peach Bottom ODCMS Table 4.8.E.1 Item 1. Site Boundary stations all sectors except several along Conowingo Pond. These sectors are monitored by stations on the east side of Conowingo Pond. The 5 mile vicinity stations cover all sectors. The distant and special interest stations provide information in population centers and control locations.		
	1J	3,755 feet S of site			
	1F	2,707 feet SSW of site			
	40	7,050 feet SW of site			
	1NN	2,547 feet WSW of site			
	1H	3,104 feet W of site			
	1G	3,173 feet WNW of site			
	1B	2,587 feet NW of site			
1E	3,136 feet NNW of site				
Intermediate Distance	15	19,449 feet N of site			
	22	13,230 feet NNE of site			
	44	27,480 feet NE of site			
	32	15,213 feet ENE of site			
	45	18,524 feet ENE of site			
	14	10,397 feet E of site			
	17	21,966 feet ESE of site			
	31A	24,105 feet SE of site			
	4K	45,721 feet SE of site			
	23	5,276 feet SSE of site			
	27	13,859 feet S of site			
	48	25,772 feet SSW of site			

TABLE VII.A-1

ODCM - Peach Bottom Atomic Power Station
Radiological Environmental Monitoring Program

Pathway	Station Code	Distance and Direction from PBAPS Vents	Collection Method and Discussion	Analyses
	3A	19,114 feet SW of site		
	49	20,673 feet WSW of site		
	50	25,677 feet W of site		
	51	20,511 feet WNW of site		
	26	22,093 feet NW of site		
	6B	30,538 feet NW of site		
	42	21,954 feet NNW of site		
Distant and Special Interest	43	26,931 feet NNE of site		
	5	24,482 feet E of site		
	16	67,788 feet E of site		
	24	58,048 feet ESE of site		
	46	23,483 feet SSE of site		
	47	22,153 feet S of site		
	18	51,413 feet W of site		
	19 C	106,354 feet WNW of site		

II. Airborne

Particulates	1Z	1,396 feet SE of site	Approximately 1 cfm continuous flow through glass fiber filter which is collected weekly.	Gross beta analysis on each weekly sample. Gamma spectrometry shall be done when gross beta exceeds ten times the yearly mean of
	1B	2,587 feet NW of site		
	2	4,661 feet SE of site		
	3A	19,114 feet SW of site		
	22G1 C	162,565 feet NE of site		
			These stations provide for coverage of the	

TABLE VII.A-1

ODCM - Peach Bottom Atomic Power Station
Radiological Environmental Monitoring Program

Pathway	Station Code	Distance and Direction from PBAPS Vents	Collection Method and Discussion	Analyses
			highest annual average ground level D/Q near the site boundary, the community with the highest annual average D/Q and a control location.	control station value. Gross beta analysis done ≥ 24 hr after sampling to allow for Radon and Thoron daughter decay. Gamma Spec on quarterly composite by location.
Iodine	1Z 1B 2 3A 22G1 C	1,396 feet SE of site 2,587 feet NW of site 4,661 feet SE of site 19,114 feet SW of site 162,565 feet NE of site	A TEDA impregnated flow-through cartridge is connected to air sampler and is collected weekly at site filter change.	Iodine 131 weekly
III. <u>Waterborne</u>				
Surface	1LL C	1,256 feet NE of site	Sample collected from a	Gamma isotopic

TABLE VII.A-1

ODCM - Peach Bottom Atomic Power Station
Radiological Environmental Monitoring Program

Pathway	Station Code	Distance and Direction from PBAPS Vents	Collection Method and Discussion	Analyses
	1MM	5,470 feet SE of site	continuous water sampler, monthly. In event sampler is inoperable, daily grab samples will be collected until sampler returned to service.	analysis monthly; H-3 on quarterly composite
Drinking	4L 6I C	45,721 feet SE of site 30,337 feet NW of site	Sample collected from a continuous water sampler monthly. In event sampler is inoperable, weekly grab samples will be collected until sampler returned to service.	Gross beta and gamma isotopic monthly, H-3 on quarterly composite
Sediment	4J	7,346 feet SE of site	A sediment sample is taken down stream of discharge semi-annually.	Gamma isotopic analysis each sample
IV. <u>Ingestion</u>				
Milk	A C	30,493 feet WSW of site	Sample of fresh milk is collected from each farm biweekly when cows are on pasture (April through October), monthly at other times.	I-131 analyses on each sample
	G	7,855 feet SSW of site		
	J	5,119 feet W of site		
	O	12,241 feet SW of site		

Gamma isotopic
analysis or Cs-134, -

TABLE VII.A-1

ODCM - Peach Bottom Atomic Power Station
Radiological Environmental Monitoring Program

Pathway	Station Code	Distance and Direction from PBAPS Vents	Collection Method and Discussion	Analyses
				137 by chemical separation quarterly
Fish	4 6 C	7,162 feet SE of site 57,347 feet NW of site	Two species of recreationally important fish (predator and bottom feeder) sampled in season or semiannually if not seasonal.	Gamma isotopic analyses on edible portions.
Food Products	Two offsite locations of highest predicted annual average ground level D/Q and one location (15-30 km distance) in the least prevalent wind direction as determined using the results of the most recent annual Landuse Survey.		Samples of three (3) different kinds of broad leaf vegetation monthly when available if milk sampling is not performed	Gamma isotopic and I-131 analysis

C = Control Location

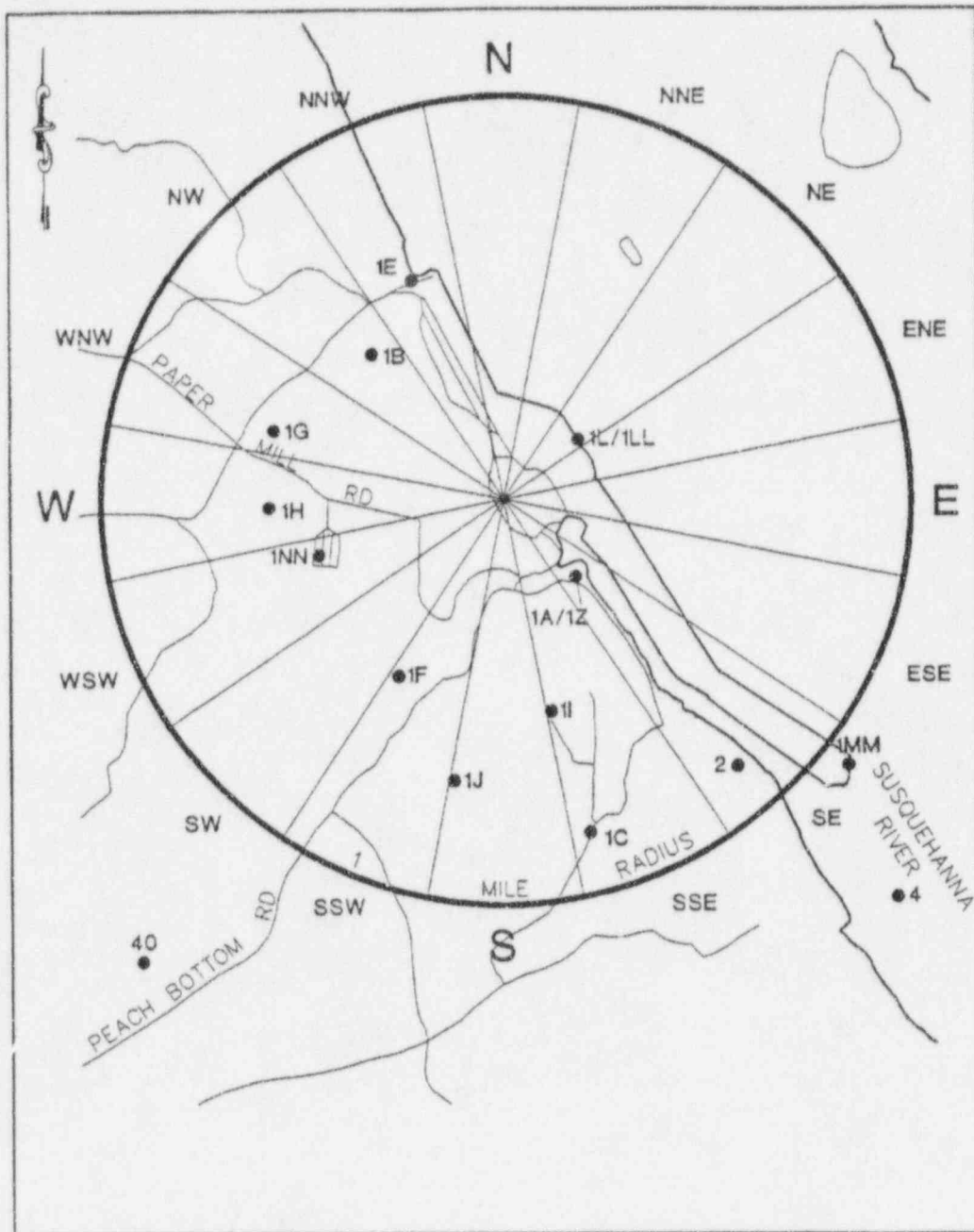


FIGURE VII.A.1
ENVIRONMENTAL SAMPLING STATIONS AT
SITE BOUNDARY AREA TO PEACH BOTTOM

INTERMEDIATE DISTANCES FROM PEACH BOTTOM SITE

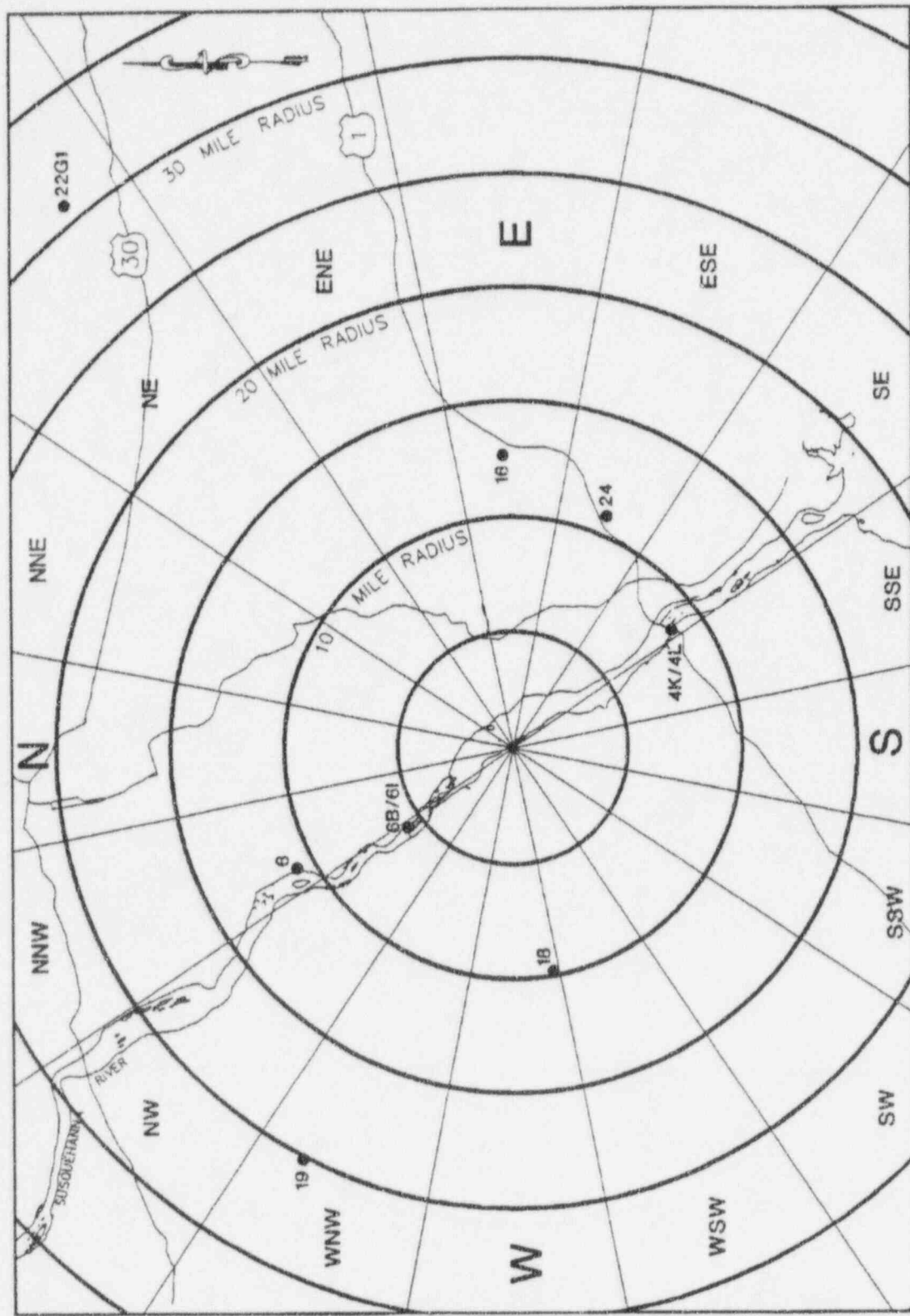
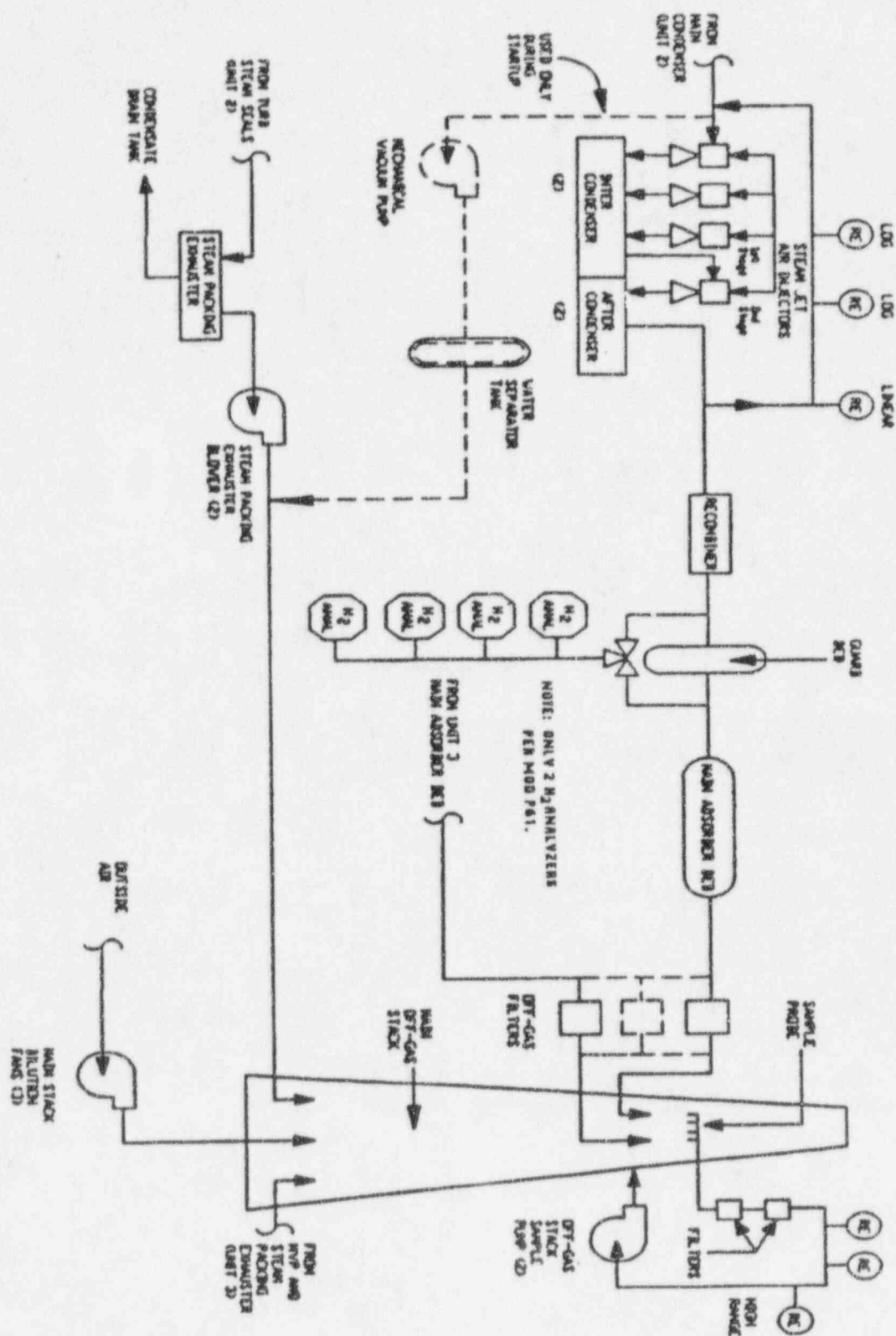
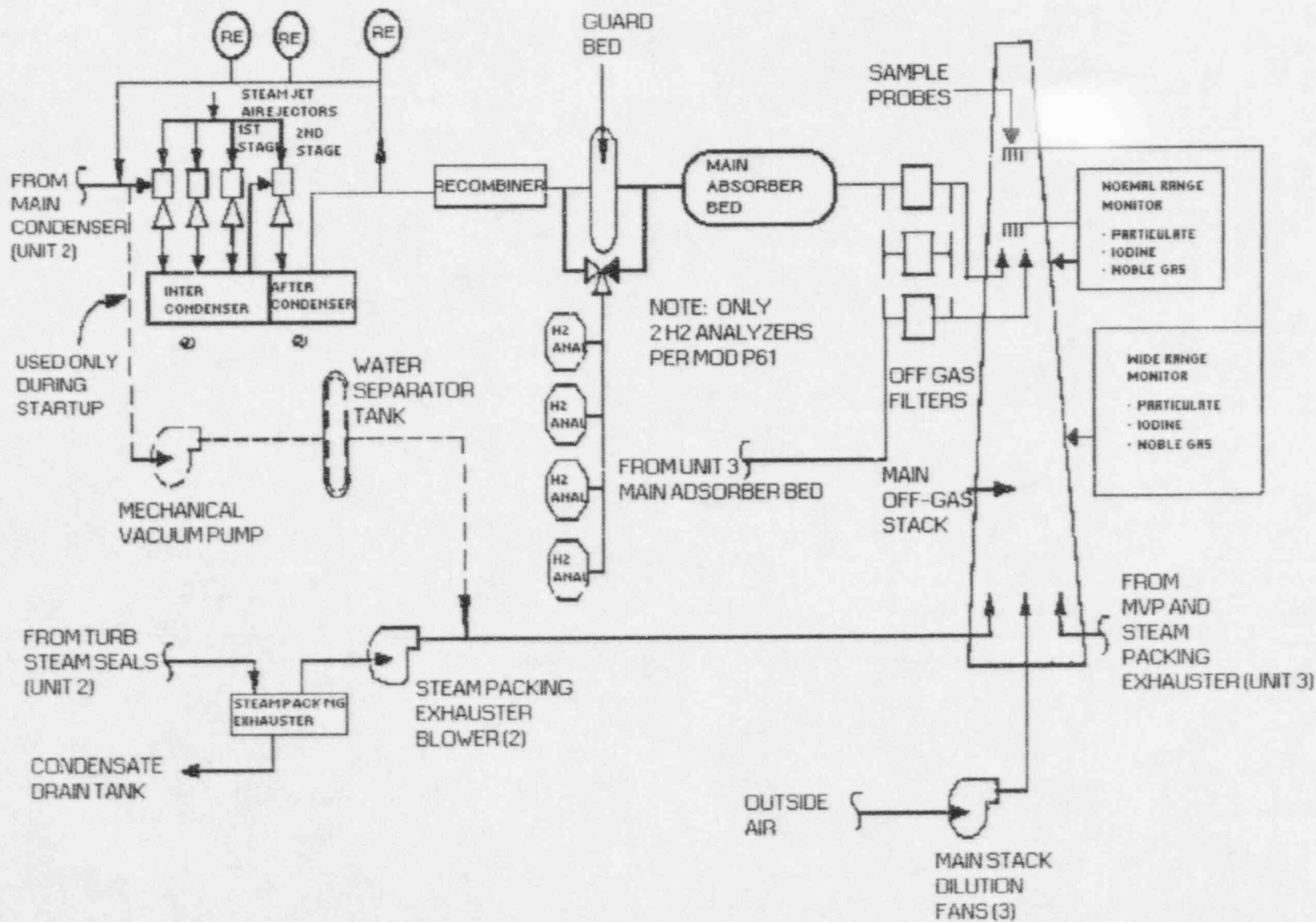


FIGURE VII.A.3
ENVIRONMENTAL SAMPLING STATIONS AT
REMOTE DISTANCES FROM PEACH BOTTOM SITE

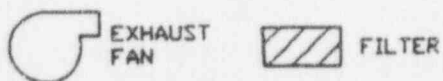
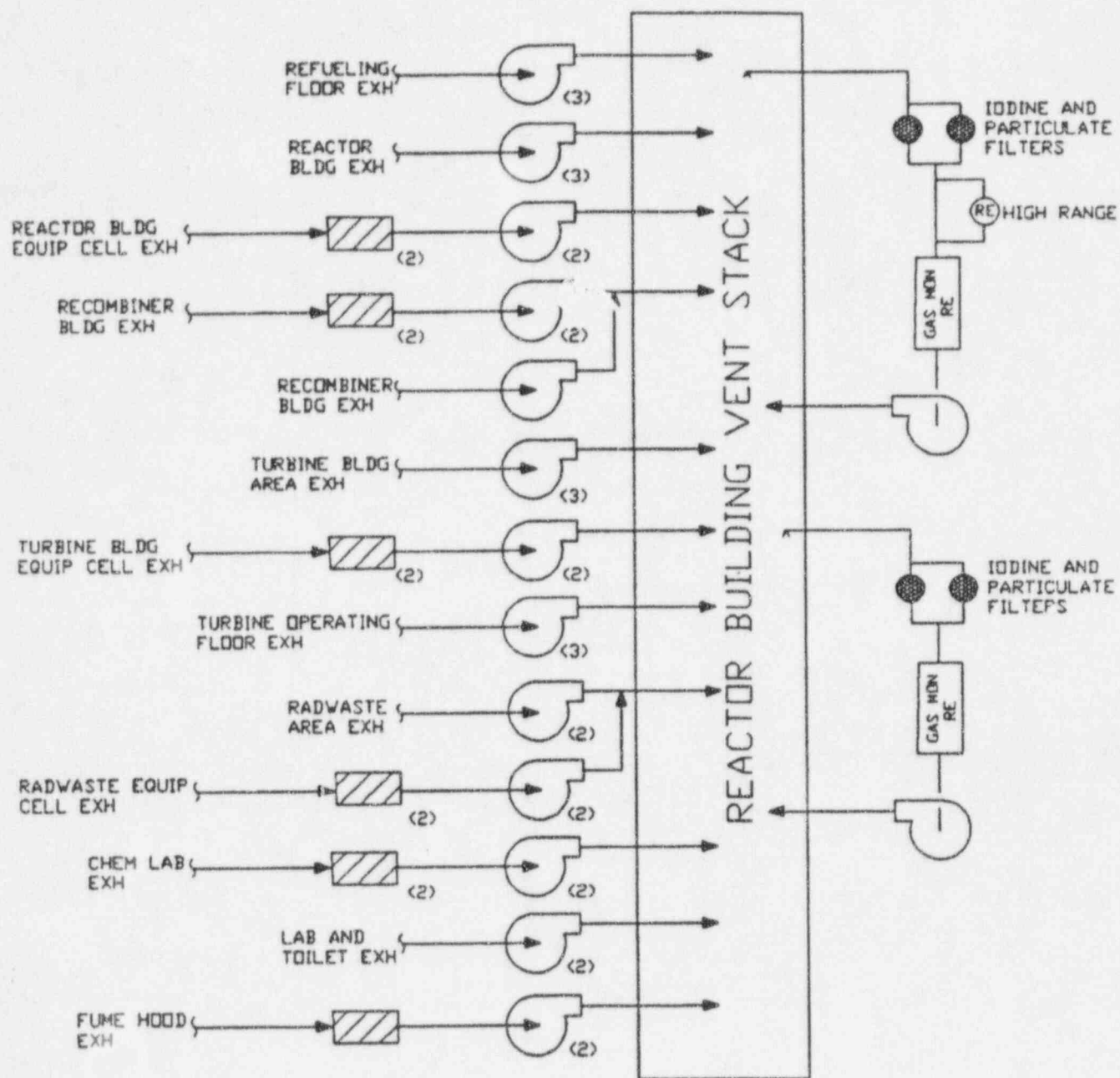


PECO ENERGY COMPANY
PBAPS UNITS 2 & 3

OFFGAS RADWASTE TREATMENT
SYSTEM
PRE MOD 5386

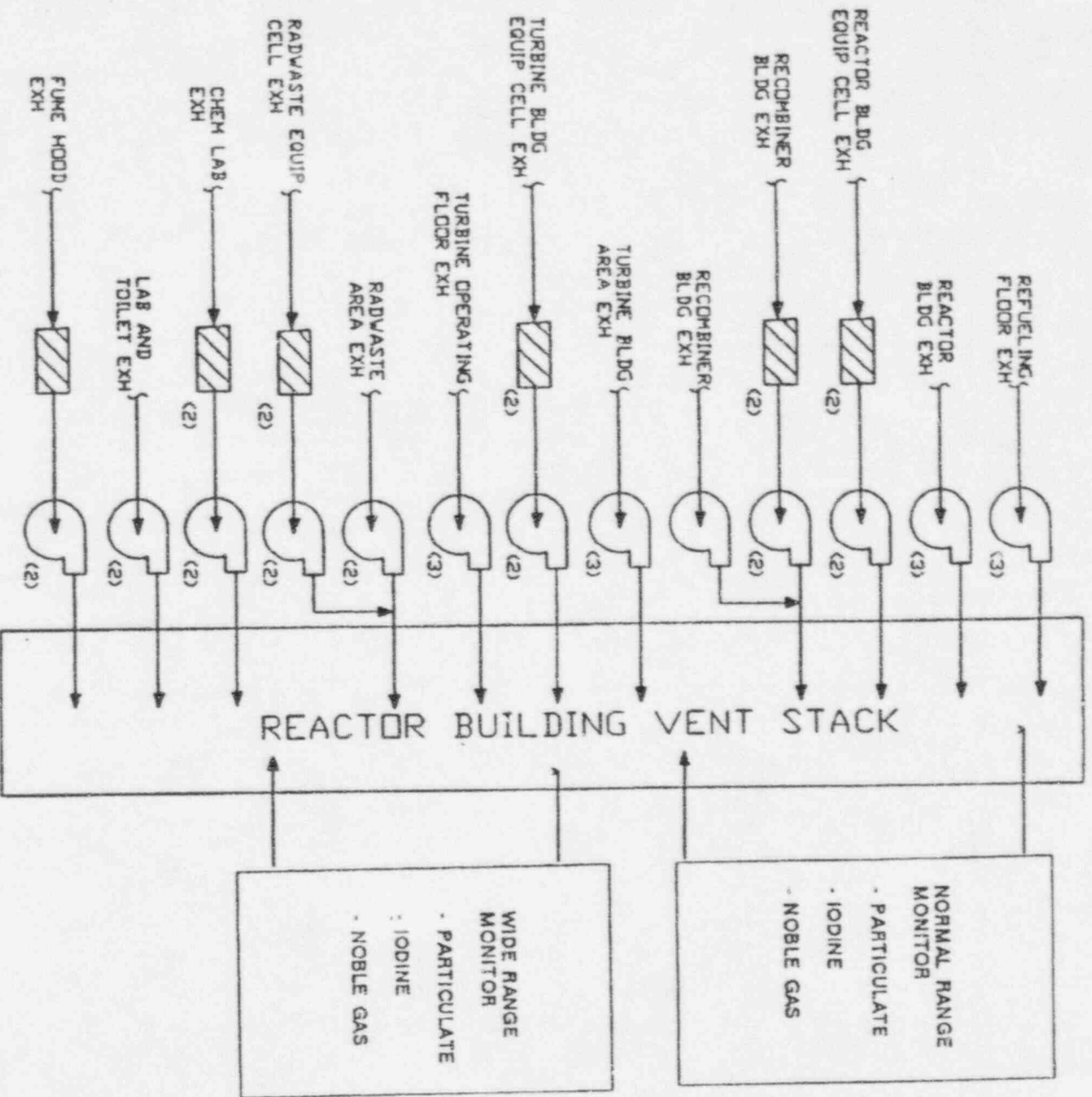


PECO ENERGY COMPANY
PBAPS UNITS 2 & 3
OFFGAS RADWASTE TREATMENT
SYSTEM
POST MOD 5386

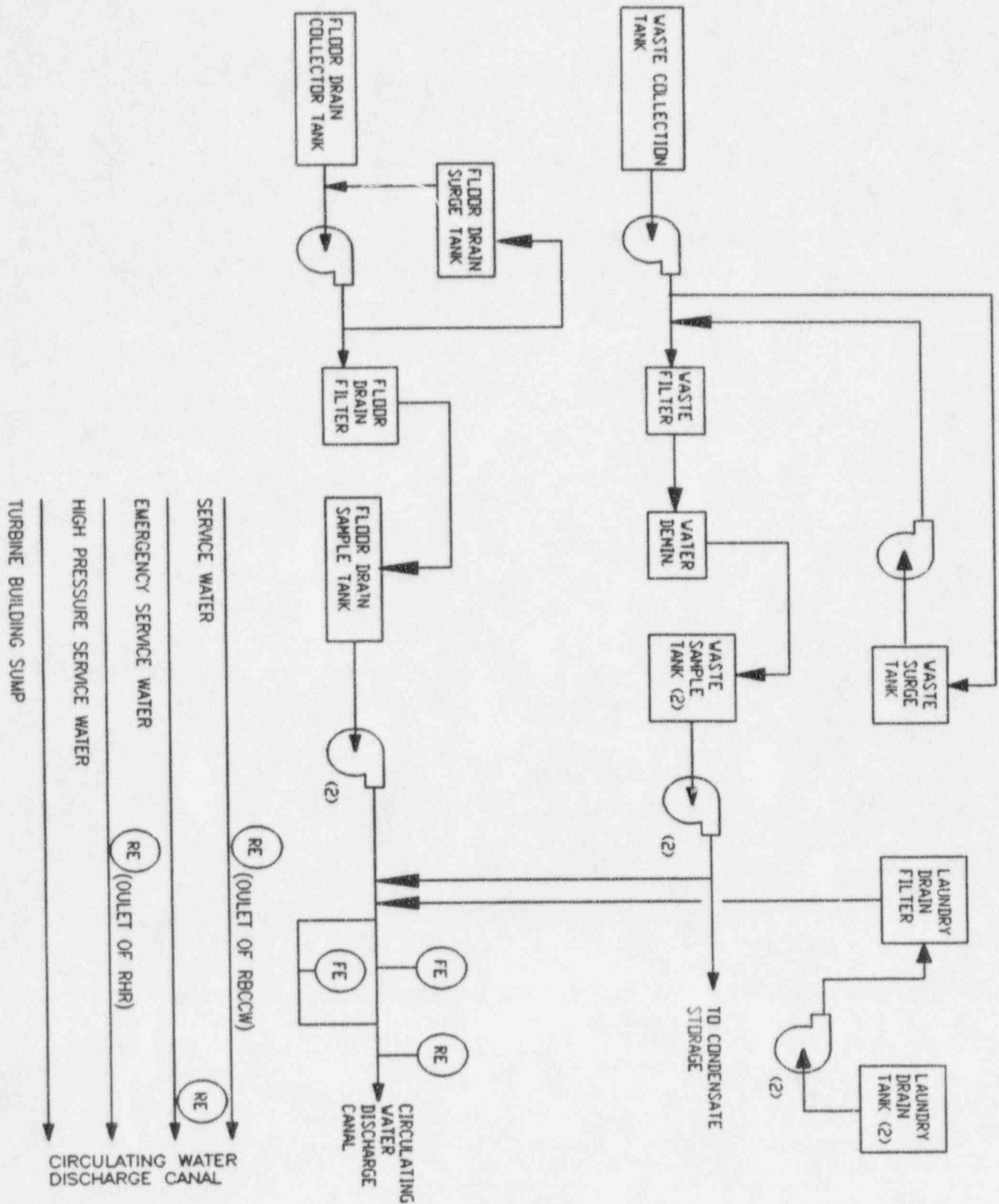


PECO ENERGY COMPANY
PBAPS UNITS 2 & 3

VENTILATION EXHAUST WASTE
TREATMENT SYSTEM
PRE MOD 5384



PECO ENERGY COMPANY
PBAPS UNITS 2 & 3
VENTILATION EXHAUST WASTE
TREATMENT SYSTEM
POST MOD 5384



PHILADELPHIA ELECTRIC CO.
PBAPS UNITS 2 & 3

LIQUID RADWASTE
TREATMENT SYSTEM

VIII. BasesSite Specific DataNOTE 1

Liquid dose factors, $A_1\tau$, for section III.B were developed using the following site specific data. The liquid pathways involved are drinking water and fish.

$$A_1\tau = (U_W/D_W + U_F \times BF_i) K_0 \times DF_i \times RC$$

U_W = liters per year; maximum age group usage of drinking water (Reg. Guide 1.109, Table E-5)

D_W = 5.4; average annual dilution at Conowingo intake

U_F = kg per year; maximum age group usage of fish (Reg. Guide 1.109, Table E-5)

BF_i = bioaccumulation factor for nuclide, i, in freshwater fish. Reg. Guide 1.109, Table A-1, except P-32 which uses a value of 3.0×10^3 pCi/kg per pCi/liter.

K_0 = $1.14 \times 10^5 = (10^6 \text{ pCi/uCi} \times 10^3 \text{ ml/l} + 8760 \text{ hr/yr})$ units conversion factor.

DF_i = dose conversion factor for nuclide, i, for the age group in total body or organ, as applicable. Reg. Guide 1.109, Table E-11, except P-32 bone which uses a value as indicated below.

$$3.0 \times 10^{-5} \text{ mrem/pCi}$$

RC = 1.16; reconcentration from PBAPS discharge back through PBAPS intake.

The data for D_W and RC were derived from data published in Peach Bottom Atomic Power Station Units 2 and 3 (Docket Nos. 50-277 and 50-278) Radioactive Effluent Dose Assessment, Enclosure A, September 30, 1976. All other data except P-32 BF and DF were used as given in Reg. Guide 1.109, Revision 1, October 1977. The P-32 BF and DF were used in accordance with information supplied in Branagan, E.F., Nichols, C.R., and Willis, C.A., "The Importance of P-32 in Nuclear Reactor Liquid Effluents", NRC, 6/82. The teen and child dose factors were derived by the ratio of the adult bone dose factors in Reg. Guide 1.109 and Branagan, et al.

VIII. (Cont'd)

NOTE 2

To develop constant R for section IV.C, the following site specific data were used:

$$R_i^C(D/Q) = K' \frac{Q_F(U_{ap})}{\lambda_i + \lambda_w} F_m(r) (DFL_i) \left[\frac{f_p f_s}{Y_p} + \frac{(1-f_p f_s) e^{-\lambda_i t_h}}{Y_s} \right] e^{-\lambda_i t_f}$$

where:

- K' = 10^6 pCi/ μ Ci; unit conversion factor
- Q_F = 50 kg/day; cow's consumption rate
- U_{ap} = 330 l/yr; yearly milk consumption by an infant
- λ_i = radioactive decay constant for nuclide of interest, sec^{-1} (e.g. $9.97 \times 10^{-7} \text{ sec}^{-1}$ for I-131)
- λ_w = $5.73 \times 10^{-7} \text{ sec}^{-1}$; decay constant for removal of activity in leaf and plant surfaces
- F_m = stable element transfer coefficient for nuclide of interest, day/liter (e.g. $6.0 \times 10^{-3} \text{ day/liter}$ for I-131)
- r = fraction of deposited nuclide retained in cow's feed grass, 1.0 for radioiodine; 0.2 for particulates
- DFL_i = ingestion dose factor in infant for nuclide of interest, mrem/pCi (e.g. $1.39 \times 10^{-2} \text{ mrem/pCi}$ for I-131)
- f_p = 0.6; the fraction of the year the cow is on pasture (average of all farms)
- f_s = 0.487; the fraction of cow feed that is pasture grass while the cow is on pasture (average of all farms)
- Y_p = 0.7 kg/m^2 ; the agricultural productivity of pasture feed grass
- Y_s = 2.0 kg/m^2 ; the agricultural productivity of stored feed
- t_f = $1.73 \times 10^5 \text{ sec}$ (2 days); the transport time from pasture, to cow, to milk, to receptor
- t_h = $7.78 \times 10^6 \text{ sec}$ (90 days); the transport time from pasture, to harvest, to cow, to milk, to receptor

VIII. (Cont'd)

The pathway is the grass-cow-milk ingestion pathway. These data were derived from data published in Peach Bottom Atomic Power Station Units 2 and 3 (Docket Nos. 50-277 and 50-278) Radioactive Effluent Dose Assessment, Enclosure A, September 30, 1976. All other data were used as given in Reg. Guide 1.109, Revision 1, October 1977.

ODCMS 4.8.B.2.1, Liquid Pathway Dose Calculations

The equations for calculating the doses due to the actual release rates of radioactive materials in liquid effluents were developed from the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977 and NUREG-0133 "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants", October 1978.

ODCMS 4.8.C.1.1 and 4.8.C.1.2Dose Noble Gases

The equations for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents were developed from the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977, NUREG-0133 "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants", August 1978, and the atmospheric dispersion model presented in Information Requested in Enclosure 2 to letter from George Lear to E. G. Bauer dated February 17, 1976, September 30, 1976. The specified equations provide for determining the air doses in areas at and beyond the SITE BOUNDARY based upon the historical average atmospheric conditions.

The dose due to noble gas release as calculated by the Gross Release Method is much more conservative than the dose calculated by the Isotopic Analysis Method. Assuming the release rates given in Radioactive Effluent Dose Assessment, September 30, 1976, the values calculated by the Gross Release Method for total body dose rate and skin dose rate are 6.0 times and 5.7 times, respectively, the values calculated by the Isotopic Analysis Method.

VIII. (Cont'd)

The model Technical Specification LCO of NUREG-0133 for all radionuclides and radioactive materials in particulate form and radionuclides other than noble gases requires that the instantaneous dose rate be less than the equivalent of 1500 mrem per year. For the purpose of calculating this instantaneous dose rate, thyroid dose from iodine-131 through the inhalation pathway will be used. Since the operating history to date indicates that iodine-131 releases have had the major dose impact, this approach is appropriate. The value calculated is increased by nine per cent to account for the thyroid dose from all other nuclides. This allows for expedited analysis and calculation of compliance with the requirements of ODCMS 3.8.C.1.

In the event that the plant is shutdown long enough so that iodine-131 is no longer present in gaseous effluents, an Isotopic Analysis Method is available. Since no iodines are present, the critical organ changes from the thyroid to the lung.

ODCMS 4.8.C.2.1Dose Noble Gases

The equations for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents were developed from the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977, NUREG-0133 "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants", August 1978, and the atmospheric dispersion model presented in Information Requested in Enclosure 2 to letter from George Lear to E. G. Bauer dated February 17, 1976, September 30, 1976. The specified equations provide for determining the air doses in areas at and beyond the SITE BOUNDARY based upon the historical average atmospheric conditions.

The dose due to noble gas releases as calculated by the Gross Release Method is much more conservative than the dose calculated by the Isotopic Analysis Method. Assuming the releases rates given in Radioactive Effluent Dose Assessment, September 30, 1976, the values calculated by the Gross Release Method for total body dose rate and skin dose rate are 4.3 times and 7.2 times, respectively, the values calculated by the Isotopic Analysis Method.

VIII. (Cont'd)

ODCMS 4.8.C.3.1Dose, Iodine-131, Iodine-133, Tritium, and Radioactive Material in Particulate Form

The equations for calculating the doses due to the actual release rates of radioiodines, radioactive material in particulate form, and radionuclides other than noble gases with half-lives greater than 8 days were developed using the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I", Revision 1, October 1977, NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants", October 1978, and the atmospheric dispersion model presented in Information Requested in Enclosure 2 to Letter from George Lear to E. G. Bauer dated February 17, 1976, September 30, 1976. These equations provide for determining the actual doses based upon the historical average atmospheric conditions.

Compliance with the 10 CFR 50 limits for radioiodines, radioactive materials in particulate form and radionuclides other than noble gases with half lives greater than eight days is to be determined by calculating the thyroid dose from iodine-131 releases. Since the iodine-131 dose accounts for 92 percent of the total dose to the thyroid, the value calculated is increased by nine percent to account for the dose from all other nuclides.

In the event that the plant is shutdown long enough so that iodine-131 is no longer present in gaseous effluents, an Isotopic Analysis Method is available. Since no iodines are present, the critical organ changes from the thyroid to the liver.

APPENDIX A

RADIOACTIVE EFFLUENTS CONTROLS PROGRAM

CONTAINING

OFFSITE DOSE CALCULATION MANUAL
SPECIFICATIONS (ODCMS) AND BASES

FOR

PEACH BOTTOM ATOMIC POWER STATION

UNITS 2 AND 3

REVISION 9

RADIOACTIVE EFFLUENTS CONTROLS PROGRAM

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ODCMS 3.8.A Definitions

-----NOTE-----

The defined terms of this section appear in capitalized type and are applicable throughout these Offsite Dose Calculation Manual Specifications and Bases.

<u>Term</u>	<u>Definition</u>
FUNCTIONAL TESTS	A FUNCTIONAL TEST is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).
GASEOUS RADWASTE TREATMENT SYSTEM	Any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.
INSTRUMENT CALIBRATION	An INSTRUMENT CALIBRATION means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. The known value of the parameter shall be injected into the instrument as close to the primary sensor as practicable.
INSTRUMENT CHECK	An INSTRUMENT CHECK is a qualitative determination of acceptable OPERABILITY by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.
INSTRUMENT FUNCTIONAL TEST	An INSTRUMENT FUNCTIONAL TEST means the injection of a simulated signal into the instrument as close to the primary sensor as practicable to verify the proper instrument response, alarm and/or initiating action.

(continued)

ODCMS 3.8.A Definitions (continued)

MEMBERS OF THE PUBLIC	MEMBERS OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.
OFFSITE DOSE CALCULATION MANUAL	Contains the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and describes the environmental radiological monitoring program.
OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and all necessary attendant instrumentation, controls, normal and emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PURGE - PURGING	PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.
SITE BOUNDARY	That line beyond which the land is not owned, leased, or otherwise controlled by licensee as defined in Figure 3.8.A.1.
SOURCE CHECK	A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

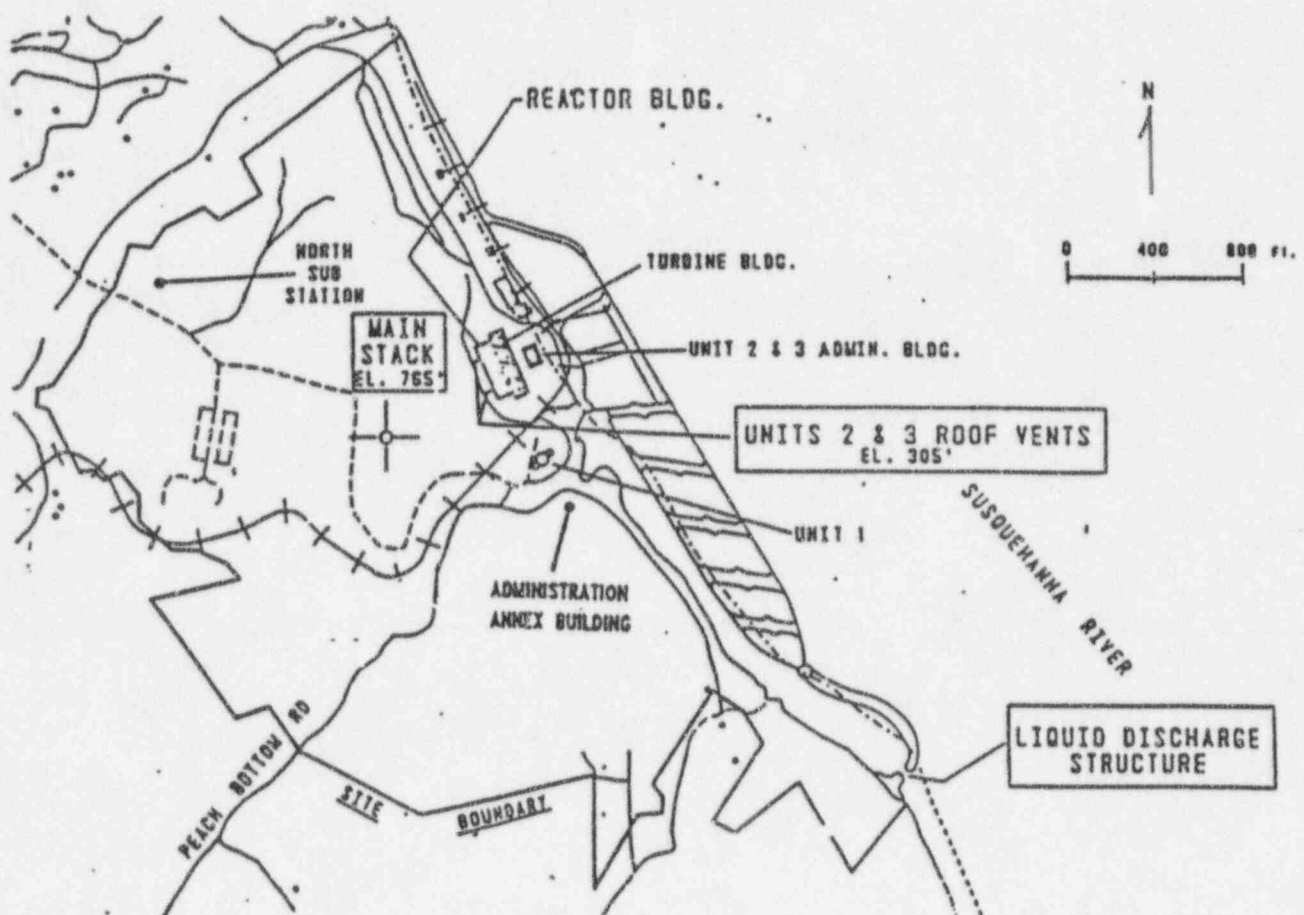


Figure 3.8.A.1

SITE BOUNDARY
(including gaseous and liquid effluent release points)

3.8.B Liquid Radwaste Effluents

ODCMS 3.8.B.1 The concentration of radioactive material released to areas at or beyond the SITE BOUNDARY shall be limited to:

- a. The concentration specified in 10 CFR 20 Appendix B, Table II, Column 2 for radionuclides other than noble gases; and
- b. $2 \times 10^{-4} \mu\text{Ci/ml}$ total activity concentration for all dissolved or entrained noble gases.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Concentration of radioactive material released to areas at or beyond the SITE BOUNDARY exceeding limits.	A.1 Decrease the release rate of radioactive materials to restore concentration to within limits.	Immediately
	<u>OR</u>	
	A.2 Increase the dilution flow rate to restore concentration within limits.	Immediately
	<u>OR</u>	
	A.3 Decrease the release rate of radioactive materials and increase the dilution flow rate to restore concentration within limits.	Immediately

ODCMS REQUIREMENTS

ODCMS REQUIREMENT		FREQUENCY
ODCMS 4.8.B.1.1	Take a sample of each batch of liquid effluent and analyze for the concentration of each significant gamma energy peak. Base the release rate on the circulating water flow rate at the time of discharge.	Prior to each release
ODCMS 4.8.B.1.2	Record radioactive concentrations and volume before dilution of each batch of liquid effluent released, the average dilution flow, and length of time over which each discharge occurred. These facility records shall be maintained.	During each release
ODCMS 4.8.B.1.3	Perform radioactive liquid waste sampling and activity analysis.	In accordance with Table 4.8.B.1

Table 4.8.B.1 (Page 1 of 2)
Radioactive Liquid Waste Sampling and Analysis

SAMPLE TYPE	SAMPLE FREQUENCY	SAMPLE ANALYSIS	SAMPLE LOWER LIMIT OF DETECTION (LLD) (a),(d),(e)
Waste Tank to be released	Each batch (b)	Quantitative Analysis of Identifiable Gamma Emitters	$5 \times 10^{-7} \mu\text{Ci/ml}$
		I-131	$1 \times 10^{-6} \mu\text{Ci/ml}$
Proportional Composite of Batches	31 days (c)	Fe-55	$1 \times 10^{-6} \mu\text{Ci/ml}$
		Tritium	$1 \times 10^{-5} \mu\text{Ci/ml}$
		Gross Alpha	$1 \times 10^{-7} \mu\text{Ci/ml}$
Proportional Composite of Batches	31 days (c)	Sr-89	$5 \times 10^{-8} \mu\text{Ci/ml}$
		Sr-90	$5 \times 10^{-8} \mu\text{Ci/ml}$
One Batch	31 days	Dissolved noble gases	$1 \times 10^{-5} \mu\text{Ci/ml}$

- (a) The Sample Lower Limit of Detection is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement. The values for the lower limit of detection are based on a 95% confidence level.
- (b) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analysis, each batch shall be isolated and thoroughly mixed to assure representative sampling.
- (c) A composite sample is one in which the quantity of the sample is proportional to the quantity of liquid waste discharged and in which the method of sampling results in a sample representative of the liquids released.

Table 4.8.B.1 (Page 2 of 2)
Radioactive Liquid Waste Sampling and Analysis

- (d) The principal gamma emitters for which the minimum detectable level specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the sample detectable limit for the analyses should not be reported as being present at the sample detectable limit level. When unusual circumstances result in sample detectable limits higher than required, the reasons shall be documented in the Annual Effluent Report. The values listed are believed to be attainable.
- (e) Certain mixtures of radionuclides may cause interference in the measurement of individual radionuclides at their detectable limit especially if other radionuclides are at much higher concentrations. Under these circumstances use of known ratios of radionuclides will be appropriate to calculate the levels of such radionuclides.

3.8.B Liquid Radwaste Effluents

ODCMS 3.8.B.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive material in liquid effluent releases from the two reactors at the site to the areas at or beyond the SITE BOUNDARY shall be limited to:

- a. ≤ 3.0 mrem to the total body and ≤ 10.0 mrem to any organ during any calendar quarter; and
- b. ≤ 6.0 mrem to the total body and ≤ 20.0 mrem to any organ during any calendar year.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Calculated dose from the release of radioactive materials in liquid effluents exceeds required limits.	A.1 Submit a Special Report to the NRC that identifies causes for exceeding limits, actions taken to reduce releases of radioactive materials in liquid effluents, corrective actions taken to assure subsequent releases are within limits, results of radiological analyses of the drinking water source, and the radiological impact on the potentially affected drinking water supplies with regard to 40 CFR 141, Safe Drinking Water Act.	31 days

ODCMS REQUIREMENTS

ODCMS REQUIREMENT		FREQUENCY
ODCMS 4.8.B.2.1	Determine cumulative dose contributions in accordance with the methodology and parameters in the ODCM.	31 days

3.8.B Liquid Radwaste Effluents

ODCMS 3.8.B.3 The following conditions shall be met:

- a. Dilution water flow rates necessary to satisfy requirements of ODCMS 3.8.B.1 shall be maintained;
- b. The gross activity monitor and the flow monitor on the waste effluent line shall be OPERABLE; and
- c. The liquid effluent control monitor shall be set in accordance with methodology and parameters in the ODCM to alarm and automatically close the waste discharge valve prior to exceeding limits in ODCMS 3.8.B.1.

APPLICABILITY: During release of radioactive wastes.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Gross activity monitor on the waste activity line inoperable.	A.1 Initiate action to restore monitor to OPERABLE status.	Immediately
	<u>AND</u>	
	A.2 Analyze two independent samples of tank's contents.	Prior to release
	<u>AND</u>	
	A.3 Perform independent verification of the release rate calculations and discharge line valve line-up using at least two technically qualified members of the facility staff.	Prior to release

(continued)

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
B. Flow monitor on the waste effluent line inoperable.	B.1 Initiate action to restore monitor to OPERABLE status.	Immediately
	<p><u>AND</u></p> <p>B.2 -----NOTE----- Pump performance curves may be used to estimate flow. -----</p> <p>Estimate waste effluent line flow rate.</p>	<p>4 hours</p> <p><u>AND</u></p> <p>Once per 4 hours thereafter</p>
C. Required Compensatory Measures and associated Completion Time of Condition A or B not met. <u>OR</u> Dilution water flow rates not satisfying requirements of ODCMS 3.8.B.1.	C.1 Suspend release of radioactive effluents via this pathway.	Immediately
D. Liquid radwaste effluents radiation monitors inoperable for > 30 days.	D.1 Explain the reason the inoperability was not corrected in a timely manner in the Annual Radioactive Effluent Release Report.	Prior to submittal of the next Annual Radioactive Effluent Release Report

ODCMS REQUIREMENTS

ODCMS REQUIREMENT		FREQUENCY
ODCMS 4.8.B.3.1	Perform INSTRUMENT CHECK of liquid radwaste effluents radiation monitor.	24 hours during release
ODCMS 4.8.B.3.2	Perform INSTRUMENT CHECK of liquid effluent flow monitor.	24 hours during release
ODCMS 4.8.B.3.3	Perform INSTRUMENT FUNCTIONAL TEST of liquid radwaste effluents radiation monitor, including OPERABILITY demonstration of the radwaste discharge automatic isolation valve and control room annunciation in response to alarm/trip setpoint being exceeded and an instrument INOP failure.	31 days
ODCMS 4.8.B.3.4	Perform SOURCE CHECK of liquid radwaste effluents radiation monitor.	92 days
ODCMS 4.8.B.3.5	Perform INSTRUMENT CALIBRATION of the liquid radwaste effluents radiation monitor with a known radiation source positioned in a reproducible geometry with respect to the sensor.	12 months
ODCMS 4.8.B.3.6	Perform INSTRUMENT CALIBRATION of the liquid effluents flow monitor.	12 months

3.8.B Liquid Radwaste Effluents

ODCMS 3.8.B.4 Liquid effluent releases shall be processed through one of the radwaste subsystems or combination of subsystems listed below:

- a. waste collector filter and demineralizer;
- b. floor drain filter;
- c. fuel pool filter demineralizer; or
- d. chemical/oily waste cleanup subsystem

APPLICABILITY: Prior to all liquid effluent release(s) whenever the release(s) would cause the projected dose, when it is averaged over 31 days to exceed 0.12 mrem to the total body or 0.4 mrem to any organ (combined total from the two reactors at the site).

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Liquid waste discharged without required processing.	A.1 Submit a Special Report to the NRC that includes, an explanation of why liquid radwaste was discharged without required processing, identification of any inoperable equipment or subsystems and the reason for the inoperability, the action taken to restore the inoperable equipment to OPERABLE status, and the action taken to prevent recurrence.	31 days

ODCMS REQUIREMENTS

ODCMS REQUIREMENT		FREQUENCY
ODCMS 4.8.B.4.1	Project doses due to liquid effluent releases to areas at and beyond the SITE BOUNDARY in accordance with the methodology and parameters in the ODCM.	31 days
ODCMS 4.8.B.4.2	Demonstrate OPERABILITY of the waste collector filter and demineralizer and the floor drain filter by analyzing the liquid processed through the subsystem and determining it meets the requirements of ODCMS 3.8.B.1.	92 days
ODCMS 4.8.B.4.3	Demonstrate OPERABILITY of the fuel pool filter demineralizer by analyzing the liquid processed through the subsystem and determining it meets the requirements of ODCMS 3.8.B.1.	Prior to release of liquid effluents processed by this subsystem
ODCMS 4.8.B.4.4	Demonstrate OPERABILITY of the chemical/oily waste cleanup subsystem by analyzing the liquid processed through the subsystem and determining it meets the requirements of ODCMS 3.8.B.1.	Prior to release of liquid effluents processed by this subsystem

3.8.C Gaseous Effluents

ODCMS 3.8.C.1 The dose rate at or beyond the SITE BOUNDARY due to radioactive materials in gaseous effluents released from the two reactors at the site shall be limited to the following:

- a. For noble gases, ≤ 500 mrem per year to the total body and ≤ 3000 mrem per year to the skin; and
- b. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days, ≤ 1500 mrem per year to any organ.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Dose rates at or beyond the SITE BOUNDARY exceed limits due to radioactive material in gaseous effluents.	A.1 Decrease release rates to comply with limits.	Immediately

ODCMS REQUIREMENTS

ODCMS REQUIREMENT	FREQUENCY
ODCMS 4.8.C.1.1 Verify the dose rate due to noble gases in gaseous effluents is within limits in accordance with methods and procedures of the ODCM.	In accordance with sampling and analysis program specified in Table 4.8.C.1

(continued)

ODCMS REQUIREMENTS (continued)

ODCMS REQUIREMENT	FREQUENCY
<p>ODCMS 4.8.C.1.2 Obtain representative samples and verify the dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents are within limits in accordance with the methods and procedures of the ODCM.</p>	<p>In accordance with sampling and analysis program specified in Table 4.8.C.1</p>

Table 4.8.C.1 (page 1 of 2)
Radioactive Gaseous Waste Sampling and Analysis from
Main Stack and Vent Stack^(e)

SAMPLE TYPE	SAMPLE FREQUENCY	SAMPLE ANALYSIS	SAMPLE LOWER LIMIT OF DETECTION (LLD) (a)(d)
Grab Sample	31 days (b)	Quantitative analysis of identifiable gamma emitters	$1 \times 10^{-4} \mu\text{Ci/cc(c)}$
Grab Sample	92 days	Tritium	$1 \times 10^{-6} \mu\text{Ci/cc}$
Charcoal Filters	7 days (c)	I-131	$1 \times 10^{-12} \mu\text{Ci/cc(c)}$
Particulate Filters	7 days (c)	Quantitative analysis of identifiable gamma emitters	$1 \times 10^{-10} \mu\text{Ci/cc(c)}$
		I-131	$1 \times 10^{-12} \mu\text{Ci/cc(c)}$
Particulate Filters (composite of weekly filters)	31 days	Gross Alpha	$1 \times 10^{-11} \mu\text{Ci/cc}$
Particulate Filters (composite of weekly filters)	31 days	Sr-89	$1 \times 10^{-11} \mu\text{Ci/cc}$
		Sr-90	$1 \times 10^{-11} \mu\text{Ci/cc}$
Noble Gas Monitor (Main Stack)	Continuously	Noble Gas Gross β or γ	$1 \times 10^{-6} \mu\text{Ci/cc (f)}$ $1 \times 10^{-3} \mu\text{Ci/cc (g)}$
Noble Gas Monitor (Roof Vents)	Continuously	Noble Gas Gross β or γ	$1 \times 10^{-6} \mu\text{Ci/cc}$

Table 4.8.C.1 (page 2 of 2)
Radioactive Gaseous Waste Sampling and Analysis from
Main Stack and Vent Stack^(e)

- (a) The sample lower limit of detection is defined as an a priori (before the fact) limit representing the capability of measurement system and not as an a posteriori (after the fact) limit for a particular measurement. The values for the lower limit of detection are based on a 95% confidence level.
- (b) Sampling and analysis shall be performed following shutdown, startup or a thermal power change exceeding 15% of rated thermal power within 1 hour from a steady state condition unless (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3, and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3.
- (c) Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing. Sampling shall also be performed at least once per 24 hours for at least 3 days following each shutdown, startup or thermal power change exceeding 15% of rated thermal power in 1 hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD may be increased by a factor of 10. This requirement does not apply if (1) analysis has shown that the dose equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3, and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.
- (d) Certain mixtures of radionuclides may cause interference in the measurement of individual radionuclides at their detectable limit especially if other radionuclides are at much higher concentrations. Under these circumstances use of known ratios of radionuclides will be appropriate to calculate the levels of such radionuclides. Nuclides which are below the sample detectable limit for the analyses should not be reported as being present at the sample detectable limit level.
- (e) Until completion of Modification 5384, the Reactor Building Vent Exhaust Stacks shall be considered to be the Vent Stacks.
- (f) After completion of Modification 5386.
- (g) Until completion of Modification 5386.

3.8.C Gaseous Effluents

ODCMS 3.8.C.2 The air dose at or beyond the SITE BOUNDARY due to noble gases in gaseous effluents released from the two reactors at the site shall be limited to the following:

- a. For gamma radiation, ≤ 10 mrad during any calendar quarter;
- b. For beta radiation, ≤ 20 mrad during any calendar quarter;
- c. For gamma radiation, ≤ 20 mrad during any calendar year; and
- d. For beta radiation, ≤ 40 mrad during any calendar year.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Calculated air dose at or beyond the SITE BOUNDARY from radioactive noble gases in gaseous effluents exceeds limits.	A.1 Submit a Special Report to the NRC that identifies causes for exceeding limits, corrective actions taken to reduce the releases, and corrective actions to assure that subsequent releases are within limits.	31 days

ODCMS REQUIREMENTS

ODCMS REQUIREMENT		FREQUENCY
ODCMS 4.8.C.2.1	Determine cumulative dose contributions for noble gases in accordance with the methodology and parameters in the ODCM.	31 days

3.8.C Gaseous Effluents

ODCMS 3.8.C.3 The dose to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY from iodine-131, iodine-133, tritium and all radionuclides in particulate form, with half lives > 8 days, in gaseous effluents released from the two reactors at the site shall be limited to:

- a. ≤ 15 mrem during any calendar quarter; and
- b. ≤ 30 mrem during any calendar year.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Calculated dose from the release of iodine-131, iodine-133, tritium and radionuclides in particulate form, with half-lives > 8 days in gaseous effluents exceeds limits.	A.1 Submit a Special Report to the NRC that identifies causes for exceeding limits, corrective actions taken, and proposed corrective actions to assure that subsequent releases are within limits.	31 days

ODCMS REQUIREMENTS

ODCMS REQUIREMENT		FREQUENCY
ODCMS 4.8.C.3.1	Determine cumulative dose contributions for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives > 8 days in accordance with the methodology and parameters in the ODCM.	31 days

3.8.C Gaseous Effluents

ODCMS 3.8.C.4 The following conditions shall be met:

- a. Main stack dilution flow shall be $\geq 10,000$ cfm;
- b. One vent stack noble gas monitor and one main stack noble gas monitor shall be OPERABLE and set to alarm in accordance with the methodology and parameters in the ODCM;
- c. One vent stack iodine filter and one main stack iodine filter and one vent stack particulate filter and one main stack particulate filter with their respective flow rate monitors shall be OPERABLE; and
- d. One vent stack flow rate monitor and one main stack flow rate monitor shall be OPERABLE and set to alarm in accordance with the methodology and parameters in the ODCM.

-----NOTE-----
Until completion of Modification 5384, the vent stack noble gas monitors, filters, and flow rate monitors shall be considered to be the reactor building exhaust vent noble gas monitors, filters, and flow rate monitors, respectively.

APPLICABILITY: During release of radioactive wastes.

COMPENSATORY MEASURES

-----NOTE-----
Separate condition entry is allowed for each instrument.

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. One required vent stack noble gas monitor inoperable. <u>OR</u> One required main stack noble gas monitor inoperable.	A.1	Collect two independent grab samples from the affected effluent release pathway.
		8 hours
		<u>AND</u>
		Once per 8 hours thereafter
	<u>AND</u>	
	A.2	-----NOTE----- Perform independent verification of the release rate calculations using at least two technically qualified members of the facility staff. -----
		Analyze grab samples from the affected release pathway for gross activity.
		24 hours from time of collection

(continued)

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
<p>B. One required vent stack iodine filter inoperable.</p> <p><u>OR</u></p> <p>One required vent stack particulate filter inoperable.</p> <p><u>OR</u></p> <p>One required main stack iodine filter inoperable.</p> <p><u>OR</u></p> <p>One required main stack particulate filter inoperable.</p>	<p>B.1 Collect samples from the affected effluent release pathway continuously with auxiliary sampling equipment such that each sample period duration is approximately 7 days.</p> <p><u>AND</u></p> <p>B.2 Analyze collected samples.</p>	<p>Immediately</p> <p>48 hours after end of each sampling period</p>
<p>C. One required vent stack flow rate monitor inoperable.</p> <p><u>OR</u></p> <p>One required main stack flow rate monitor inoperable.</p>	<p>C.1 Estimate flow rate from the affected effluent release pathway.</p>	<p>4 hours</p> <p><u>AND</u></p> <p>Once per 4 hours thereafter</p>
<p>D. Main off-gas stack dilution flow < 10,000 cfm.</p>	<p>D.1 Initiate action to re-establish main off-gas stack flow \geq 10,000 cfm.</p>	<p>Immediately</p>

(continued)

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
E. One or more required radioactive gaseous effluent monitoring instrumentation channels inoperable for > 30 days.	E.1 Explain the reason inoperability was not corrected in a timely manner in the Annual Radioactive Effluent Release Report.	Prior to submittal of next Annual Radioactive Effluent Release Report

ODCMS REQUIREMENTS

ODCMS REQUIREMENT		FREQUENCY
ODCMS 4.8.C.4.1	Perform INSTRUMENT CHECK of the vent stack and main stack noble gas radiation monitors.	24 hours
ODCMS 4.8.C.4.2	<p>-----NOTE----- The requirement to perform an INSTRUMENT CHECK of the reactor building exhaust vent sample flow rate monitors is only applicable until completion of Modification 5384. -----</p> <p>Perform INSTRUMENT CHECK of the vent stack and the main stack flow rate monitors, and the Unit 2 reactor building exhaust vent <u>sample</u> flow rate monitor.</p>	24 hours
ODCMS 4.8.C.4.3	<p>-----NOTE----- This requirement is only applicable until completion of Modification 5386. -----</p> <p>Perform INSTRUMENT CHECK of the main stack sample flow rate monitor.</p>	7 days

(Continued)

ODCMS REQUIREMENTS (continued)

ODCMS REQUIREMENT	FREQUENCY
<p>ODCMS 4.8.C.4.4 Perform INSTRUMENT FUNCTIONAL TEST of the vent stack and main stack noble gas radiation monitors, including control room alarm annunciation in response to alarm setpoint being exceeded and an instrument downscale failure.</p>	<p>92 days</p>
<p>ODCMS 4.8.C.4.5 -----NOTE----- This requirement is only applicable until completion of Modification 5386. ----- Perform INSTRUMENT FUNCTIONAL TEST of the main stack sample flow line Hi/Lo pressure switches.</p>	<p>184 days</p>

(Continued)

ODCMS REQUIREMENTS (continued)

ODCMS REQUIREMENT	FREQUENCY
<p>ODCMS 4.8.C.4.6 Perform INSTRUMENT CALIBRATION of the vent stack and main stack noble gas radiation monitors with a known radioactive source positioned in a reproducible geometry with respect to the sensor.</p>	<p>18 months</p> <p>AND</p> <p>-----NOTES-----</p> <p>1. Only applicable to vent stack monitors until completion of Modification 5384.</p> <p>2. Only applicable to main stack radiation monitors until completion of Modification 5386.</p> <p>-----</p> <p>12 months</p>

(continued)

ODCMS REQUIREMENTS (continued)

ODCMS REQUIREMENT	FREQUENCY
<p>ODCMS 4.8.C.4.7 Perform INSTRUMENT CALIBRATION of the vent stack and main stack flow rate monitors.</p>	<p>18 months</p> <p><u>AND</u></p> <p>-----NOTES-----</p> <p>1. Only applicable to vent stack flow monitors until completion of Modification 5384.</p> <p>2. Only applicable to main stack flow rate monitors until completion of Modification 5386.</p> <p>-----</p> <p>12 months</p>

(continued)

ODCMS REQUIREMENTS (continued)

ODCMS REQUIREMENT	FREQUENCY
<p>ODCMS 4.8.C.4.8 Perform INSTRUMENT CALIBRATION of the vent stack and main stack iodine and particulate sample flow rate monitors.</p>	<p>18 months</p> <p><u>AND</u></p> <p>-----NOTES-----</p> <p>1. Only applicable to vent stack sample flow rate monitors until completion of Modification 5384.</p> <p>2. Only applicable to main stack sample flow rate monitors until completion of Modification 5386.</p> <p>-----</p> <p>12 months</p>
<p>ODCMS 4.8.C.4.9 -----NOTE----- This requirement is only applicable until completion of Modification 5386. -----</p> <p>Perform INSTRUMENT CALIBRATION of the main stack sample flow line Hi/Lo pressure switches.</p>	<p>24 months</p>

3.8.C Gaseous Effluents

ODCMS 3.8.C.5 Gaseous effluents shall be processed through the Gaseous Waste Treatment System described below prior to discharge:

- a. Steam jet air ejector discharge shall be processed through the recombiner, holdup pipe, off-gas filter, and off-gas stack;
- b. Mechanical vacuum pump and gland steam exhauster discharge shall be processed through the off-gas stack;
- c. Reactor, turbine, radwaste, and recombiner building atmospheres shall be processed through permanently or temporarily installed equipment in the appropriate building ventilation system and the vent stack, with the exception of the following unmonitored exhausts:
 1. Recirculation M-G set and reactor building cooling water equipment rooms;
 2. Control room utility and toilet rooms;
 3. Cable spread room;
 4. Emergency switchgear rooms;
 5. 125/250 VDC battery rooms and the 250 VDC battery rooms, and
 6. Administration building maintenance decontamination area.

-----NOTE-----
Until completion of Modification 5304, the Reactor Building Ventilation Exhaust Stacks shall be considered to be the Vent Stacks.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Gaseous waste discharged without required processing.	A.1 Submit a Special Report to the NRC that explains the reasons gaseous radwaste was discharged without processing, identifies inoperable equipment or subsystems and the reasons for the inoperability, identifies actions taken to restore the inoperable equipment to OPERABLE status, and actions taken to prevent a recurrence.	31 days

ODCMS REQUIREMENTS

ODCMS REQUIREMENT	FREQUENCY
ODCMS 4.8.C.5.1 Project doses due to gaseous effluent releases at and beyond the SITE BOUNDARY in accordance with the methodology and parameters in the ODCM.	31 days
ODCMS 4.8.C.5.2 Collect and analyze air samples from each building area with an unmonitored exhaust.	31 days

(continued)

ODCMS REQUIREMENTS (continued)

ODCMS REQUIREMENT	FREQUENCY
<p>ODCMS 4.8.C.5.3 Demonstrate OPERABILITY of required gaseous radioactive waste system equipment by analyzing the gaseous waste processed through the required equipment to determine it meets the requirements of ODCMS 3.8.C.1</p>	<p>92 days</p>

3.8.C Gaseous Effluents

ODCMS 3.8.C.6 PURGING of the primary containment shall be:

- a. Through the Standby Gas Treatment System whenever primary containment is required to be OPERABLE; and
- b. Through the Reactor Building Ventilation Exhaust System or the Standby Gas Treatment System whenever primary containment is not required to be OPERABLE.

APPLICABILITY: During PURGING of the primary containment.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Requirement of ODCMS 3.8.C.6 not met.	A.1 Suspend PURGING.	Immediately

ODCMS REQUIREMENTS

ODCMS REQUIREMENT	FREQUENCY
ODCMS 4.8.C.6.1 Verify primary containment PURGING is through required effluent release pathway of ODCMS 3.8.C.6.	Prior to PURGING primary containment

3.8.D 40 CFR 190

ODCMS 3.8.D.1 The dose or dose commitment to a MEMBER OF THE PUBLIC over the calendar year from all uranium fuel cycle sources within 8 kilometers shall be:

- a. ≤ 25 mrem to the total body or any organ (except the thyroid); and
- b. ≤ 75 mrem to the thyroid.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Calculated dose to a MEMBER OF THE PUBLIC exceeds twice the limits specified in ODCMS 3.8.B.2 or ODCMS 3.8.C.2 or ODCMS 3.8.C.3.	-----NOTE----- Calculations shall include direct radiation contributions from reactor units and outside storage tanks. -----	
	A.1 Calculate dose or dose commitment to a MEMBER OF THE PUBLIC from all uranium fuel cycle sources within 8 kilometers to determine if specified limits were exceeded.	Immediately

(continued)

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
B. Calculated dose or dose commitment exceeds specified limits of ODCMS 3.8.D.1.	<p>B.1 -----NOTE----- Estimates of radiation exposure shall include the effects of all effluent pathways and direct radiation including releases covered by this Special Report. -----</p> <p>Submit a Special Report to the NRC that includes the corrective actions taken to prevent recurrence, the schedule for achieving conformance with required limits, estimates of radiation exposure to a MEMBER OF THE PUBLIC for the calendar year, descriptions of the levels of radiation and concentrations of radioactive material involved, and the cause of the exposure level or concentrations.</p> <p><u>AND</u></p>	<p>31 days</p> <p>(continued)</p>

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
B. (continued)	<p>B.2 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Only applicable if the release condition resulting in violation of 40 CFR 190 has not been corrected. 2. Special Report submittal is considered a timely request and a variance is granted until NRC action on the request is complete. <p>-----</p> <p>Submit a request for a variance in accordance with 40 CFR 190 in the Special Report to the NRC.</p>	31 days

ODCMS REQUIREMENTS

ODCMS REQUIREMENT	FREQUENCY
ODCMS 4.8.D.1.1 Determine cumulative dose contributions from liquid and gaseous effluents in accordance with methodology and parameters in the ODCM.	In accordance with ODCM

(continued)

ODCMS REQUIREMENTS (continued)

ODCMS REQUIREMENT	FREQUENCY
<p>ODCMS 4.8.D.1.2 Determine cumulative dose contributions from direct radiation from the reactor units and from radwaste storage in accordance with methodology and parameters in the ODCM.</p>	<p>In accordance with ODCM</p>

3.8.E Radiological Environmental Monitoring

ODCMS 3.8.E.1 Radiological environmental monitoring samples shall be collected at locations and analyzed as specified in Table 4.8.E.1 and the ODCM.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Milk samples permanently unavailable from any of the sample locations listed in the ODCM.	A.1 Identify locations for obtaining replacement samples and add them to radiological environmental monitoring program. Delete locations from which samples are unavailable.	31 days
	<p><u>AND</u></p> <p>A.2 Identify in the Radioactive Dose Assessment Report the cause of the unavailability of samples and new locations for obtaining samples and include in the report revised figures and tables for the ODCM reflecting the new locations.</p>	Prior to submittal in next Radioactive Dose Assessment Report

(continued)

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
<p>B. Level of radioactivity as a result of plant effluents in an environmental sampling medium at one or more locations specified in the ODCM exceeds the reporting levels of Table 3.8.E.1 when averaged over the calendar quarter.</p> <p><u>OR</u></p> <p>Level of radioactivity as a result of plant effluents in an environmental sampling medium at one or more locations specified in the ODCM exceeds the reporting levels of Table 3.8.E.1 when averaged over the calendar quarter and more than one radionuclide from Table 3.8.E.1 is detected and the equation in ODCMS 4.8.E.1.2 is satisfied.</p>	<p>B.1 Submit a Special Report to the NRC which includes an evaluation of any release conditions, environmental factors, or other aspects which caused the reporting level of Table 3.8.E.1 to be exceeded and the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year reporting level of Table 3.8.E.1.</p>	<p>31 days after end of affected calendar quarter</p>

(continued)

COMPENSATORY MEASURES (continued)

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
C. Radionuclides other than those in Table 3.8.E.1 detected and the potential annual dose to a MEMBER OF THE PUBLIC is greater than or equal to the calendar year limits of ODCMS 3.8.B.2, or ODCMS 3.8.C.2, or ODCMS 3.8.C.3, or ODCMS 3.8.D.1.	<p>C.1 -----NOTE----- Only required if the radionuclides detected are the result of plant effluents. -----</p> <p>Submit a Special Report to the NRC which includes an evaluation of any release conditions, environmental factors, or other aspects which caused the reporting level of Table 3.8.E.1 to be exceeded; and the corrective actions to be taken to reduce radioactive effluents so that potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year reporting level of Table 3.8.E.1</p> <p><u>AND</u></p>	<p>31 days after end of affected calendar quarter</p> <p>(continued)</p>

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
C. (continued)	<p>C.2</p> <p>-----NOTE----- Only required if the radionuclides detected are not the result of plant effluents. -----</p> <p>Describe the condition in the Annual Radiological Environmental Operating Report.</p>	<p>Prior to submittal of next Annual Radiological Environmental Operating Report</p>
D. Requirements of ODCMS 3.8.E.1 not met.	<p>D.1</p> <p>Prepare and submit, in the Annual Radiological Operating Report, a description of the reasons for not conducting the radiological environmental monitoring program as required and the actions taken to prevent recurrence.</p>	<p>Prior to submittal of next Annual Radiological Environmental Operating Report</p>

ODCMS REQUIREMENTS

ODCMS REQUIREMENT	FREQUENCY
<p>ODCMS 4.8.E.1.1 -----NOTES-----</p> <ol style="list-style-type: none"> 1. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailabilities, malfunction of automatic sampling equipment or other legitimate reasons. 2. If equipment malfunction occurs, an effort shall be made to complete corrective actions prior to the end of the next sampling period. <p>-----</p> <p>Collect radiological environmental monitoring samples at the locations and analyze as specified in Table 4.8.E.1 and the ODCM.</p>	<p>In accordance with Table 4.8.E.1</p>
<p>ODCMS 4.8.E.1.2 -----NOTE-----</p> <p>All radionuclides used in this evaluation shall be averaged on a calendar quarterly basis.</p> <p>-----</p> <p>Evaluate the concentration of radioactivity as a result of plant effluents in an environmental sampling medium against the following equation:</p> $\frac{\text{concentration}(1)}{\text{reporting level}(1)} + \frac{\text{concentration}(2)}{\text{reporting level}(2)} + \dots \geq 1.0$	<p>92 days</p>

Table 3.8.E.1 (page 1 of 1)
Reporting Levels for Radioactivity Concentration in Environmental Samples

Reporting Levels

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m ³)	FISH (pCi/kg. wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg. wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95	400				
I-131		0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140	200			300	

*For drinking water samples. This is 40 CFR part 141 value.

Radiological Environmental Monitoring
3.8.E.1

Table 4.8.E.1 (page 1 of 3)
Radiological Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES AND SAMPLE LOCATION(a)	SAMPLING AND COLLECTION FREQUENCY (b),(c)	TYPE AND FREQUENCY OF ANALYSIS
1. Direct Radiation(e)	At least 40 routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously to be placed as follows: 1) an inner ring of stations in the general area of the SITE BOUNDARY and an outer ring in the 3 to 6 mile range from the site. A station is in each sector of each ring except as dictated by local geography. The balance of the stations are in special interest areas such as population centers, nearby residences, schools and in areas to serve as control stations.	Every 31 days or Every 92 days	Every 31 days (Gamma dose) or Every 92 days (Gamma dose)
2. Airborne Radioiodine and Particulates	Samples from 5 locations: a. 3 samples from close to the SITE BOUNDARY locations (in different sectors) of the highest calculated annual average ground level D/Q. b. 1 sample from the vicinity of a community having the highest calculated annual average ground level D/Q. c. 1 sample from a control location unlikely to be affected by the plant.	Continuous sampler operation with sample collection every 7 days or required by dust loading, whichever is more frequent.	Radioiodine canisters: I-131 analysis every 7 days Particulate sampler: (f) gross beta radioactivity analysis following filter change; gamma isotopic analysis of composite (by location) 92 days
3. Waterborne a. Surface	a. 1 sample upstream b. 1 sample downstream	Composite (b) sample over 31 day period.	Gamma isotopic (d) analysis every 31 days. Composite for tritium analysis at least every 92 days
(continued)			

Table 4.8.E.1 (page 2 of 3)
Radiological Environmental Monitoring Program

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF SAMPLES AND SAMPLE LOCATION	SAMPLING AND COLLECTION FREQUENCY (b),(c)	TYPE AND FREQUENCY OF ANALYSIS
3. Waterborne (cont'd)			
b. Drinking	a. 1 sample of each of 1 to 3 of the nearest water supplies that could be affected by its discharge. b. 1 sample from a control location.	Composite (b) sample over 31 day composite period	Composite for gross beta and gamma (d) isotopic analyses every 31 days. Composite for tritium analysis at least every 92 days
c. Sediment from Shoreline	1 sample from downstream area with existing or potential recreational value.	Every 6 months	Gamma isotopic (d) analysis every 6 months
4. Ingestion			
a. Milk	a. Samples from milking animals in 3 locations within 3 miles distance having the highest dose potential. b. 1 sample from milking animals at a control location (unlikely to be affected by the plant).	Every 14 days when animals are on pasture, every 31 days at other times.	Gamma isotopic (d) analysis or 134, 137 Cs by chemical separation every 92 days. I-131 analysis of each sample.
b. Fish	a. 1 sample of each commercially and recreationally important species in vicinity of discharge point when available. b. 1 sample of same species in areas not influenced by plant discharge when available.	Sample in season, or every 6 months if they are not seasonal.	Gamma isotopic (d) analysis on edible portions.
c. Food Products	a. Samples of 3 different kinds of broad leaf vegetation grown nearest offsite garden of highest annual average ground level D/Q if milk sampling is not performed. b. 1 sample of each of the similar broad leaf vegetation grown 15 - 30 km distant in the least prevalent wind direction if milk sampling is not performed.	Every 31 days when available if milk sampling is not performed. Every 31 days when available if milk sampling is not performed	Gamma isotopic (d) and I-131 analysis Gamma isotopic (d) and I-131 analysis

Table 4.8.E.1 (page 3 of 3)
Radiological Environmental Monitoring Program

NOTES

- (a) Fixed sample locations are shown in the Offsite Dose Calculation Manual, Table VII.A.1 and Figures VII.A.1, VII.A.2, and VII.A.3. At times it may not be possible or practicable to obtain samples of the media of choice at the most desired location or time. In those instances suitable alternative media and locations may be chosen for the particular pathway in question. Such necessary deviations are reported in the Annual Environmental Radiological Operating Report.
- (b) Composite samples shall be collected by collecting an aliquot at intervals not exceeding 2 hours.
- (c) Sample collection frequencies are defined as follows:
- | | |
|----------|---------------------------------|
| 7 days | 7 calendar days \pm 2 days |
| 14 days | 14 calendar days \pm 3 days |
| 31 days | 31 calendar days \pm 8 days |
| 92 days | 92 calendar days \pm 10 days |
| 6 months | 6 calendar months \pm 20 days |
- (d) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.
- (e) Each phosphor is considered one thermoluminescent dosimeter.
- (f) If the gross beta activity in air attributable to plant operation is greater than ten times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

Table 4.8.E.2 (page 1 of 1)
Maximum Values for Minimum Detectable Levels of Activity (MDL)^(a)

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE (pCi/m ³)	FISH (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	2.5	.006				
3H	1200					
54 Mn	9		80			
59 Fe	18		160			
58, 60 Co	9		80			
65 Zn	18		160			
95 Zr-Nb	9					
131-I	---	.04		0.6	36	
134, 137 Cs	9, 11	.04	90	10	40	100
140 Ba-La	9			9		

(a) Analyses shall be performed in such a manner that the stated MDLs will be achieved under routine conditions at a 95% confidence level. Occasionally, background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides or other uncontrollable circumstances make these MDLs unachievable.

3.8.E Radiological Environmental Monitoring

ODCMS 3.8.E.2 A land use census shall be conducted and shall identify the location of the nearest milk animal in each of the 16 meteorological sectors within a distance of five miles.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

[illegible]

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. (continued)	<p>A.2 -----NOTE----- The indicator sampling location having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from the radiological environmental monitoring program after October 31 of the year in which land use census was conducted. -----</p> <p>Identify the new location in the Radioactive Dose Assessment Report and include the revised figures and tables for the ODCM reflecting the new location.</p>	<p>Prior to submittal of the next Radioactive Dose Assessment Report</p>
B. Land use census identifies a location which yields a calculated dose or dose commitment that is greater than values currently being calculated in ODCMS 3.8.C.3.	<p>B.1 Identify the new location in the Radioactive Dose Assessment Report.</p>	<p>Prior to submittal of the next Radioactive Dose Assessment Report</p>

ODCMS REQUIREMENTS

ODCMS REQUIREMENT		FREQUENCY
ODCMS 4.8.E.2.1	Conduct a land use census by a door-to-door survey or by consulting local agriculture authorities or by some other appropriate means.	12 months

3.8.E Radiological Environmental Monitoring

ODCMS 3.8.E.3 Analyses shall be performed on radioactive materials supplied as part of the EPA Environmental Radioactivity Intercomparison Studies Program, or another Interlaboratory Comparison Program approved by the NRC.

APPLICABILITY: At all times.

COMPENSATORY MEASURES

CONDITION	REQUIRED COMPENSATORY MEASURE	COMPLETION TIME
A. Requirements of ODCMS 3.8.E.3 not met.	A.1 Include corrective actions to prevent recurrence in the Annual Radiological Environmental Operating Report.	Prior to submittal of next Annual Radiological Environmental Operating Report

ODCMS REQUIREMENTS

ODCMS REQUIREMENT	FREQUENCY
ODCMS 4.8.E.3.1 Include a summary of the results of the Interlaboratory Comparison Program in the Annual Radiological Environmental Operating Report.	In accordance with Technical Specification 5.6.2

3.9 Major Changes to Radioactive Waste Treatment Systems

ODCMS 3.9.1 The radioactive waste treatment systems are those systems described in ODCMS 3.8.B.3, 3.8.B.4, 3.8.C.4, and 3.8.C.5, which are used to maintain control over radioactive materials in gaseous and liquid effluents.

ODCMS 3.9.2 Major changes to the radioactive waste systems shall be made by either of the following methods. For the purpose of this Specification "major changes" is defined in ODCMS 3.9.3.

a. Licensee initiated changes:

1. Licensee initiated changes shall be reported to the NRC as part of the Modification Report required by 10 CFR 50.59. The discussion of each change shall contain:
 - a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR Part 50.59;
 - b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;
 - c. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;
 - d. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid wastes, to the actual releases for the period prior to when the changes are to be made;
 - e. An estimate of the exposure to plant operating personnel as a result of the change; and
 - f. Documentation of the fact that the change was reviewed and found acceptable by the PORC.
2. The change shall become effective upon review and acceptance by both the PORC and NRB.

(continued)

3.9 Major Changes to Radioactive Waste Treatment Systems

ODCMS 3.9.2 (continued)

b. Commission initiated changes:

1. The applicability of the change to the facility shall be determined by the PORC after consideration of the facility design.
2. The licensee shall provide the Commission with written notification of its determination of applicability including any necessary revisions to reflect facility design.
3. The change shall be reviewed by the NRB at its next regularly scheduled meeting.
4. The change shall become effective on a date proposed by the licensee and confirmed by the Commission.

ODCMS 3.9.3

"Major Changes" to radioactive waste systems shall include the following:

- a. Changes in process equipment, components, structures, and effluent monitoring instrumentation from those described in the Final Safety Analysis Report (FSAR) and evaluated in the staff's Safety Evaluation Report (SER);
 - b. Changes in the design of radwaste treatment systems that significantly alter the characteristics and/or quantities of effluents released from those previously considered in the FSAR and SER;
 - c. Changes in system design which invalidate the accident analysis as described in the SER; and
 - d. Changes in system design that result in a significant increase in occupational exposure of operating personnel.
-

3.10 Reporting Requirements

ODCMS 3.10.1 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report shall be submitted in accordance with the requirements of Technical Specification 5.6.2. In addition to the requirements of Technical Specification 5.6.2, the Annual Radiological Environmental Operating Report shall include or reference from previous reports:

- a. A summary description of the radiological environmental monitoring program including sampling methods for each sample type, size, and physical characteristics of each sample type, sample preparation methods, and measuring equipment used;
- b. At least two maps of all sampling locations keyed to a table giving distances and directions from the midpoint between reactor vents;
- c. Results of the land use census required by ODCMS 3.8.E.2; and
- d. Results of the Interlaboratory Comparison Program and discussion of all analyses in which the LLD required by Tables 4.8.B.1 and 4.8.C.1 was not achievable.

ODCMS 3.10.2 Radioactive Effluent Release Report

The Radioactive Effluent Release Report shall be submitted in accordance with the requirements of Technical Specification 5.6.3. In addition to the requirements of Technical Specification 5.6.3, the Radioactive Effluent Release Report shall include:

- a. Information for each class of solid waste (as defined by 10 CFR part 61) shipped offsite during the report period:
 1. Container volume;
 2. Total curie quantity (specify whether determined by measurement or estimate);
 3. Principal radionuclides (specify whether determined by measurement or estimate);

(continued)

3.10 Reporting Requirements

ODCMS 3.10.2 Radioactive Effluent Release Report (continued)

4. Source of waste and processing employed (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms); and
 5. Type of container (e.g., LSA, Type A, Type B, Large Quantity).
- b. A list and description of unplanned releases from the site to areas at or beyond the SITE BOUNDARY of radioactive materials in gaseous and liquid effluents made during the reporting period.
- c. Changes made during the reporting period to the Offsite Dose Calculation Manual (ODCM) and a list of new locations for dose calculations and/or environmental monitoring identified by the land use census required by ODCMS 3.8.E.2.

ODCMS 3.10.3 Radiation Dose Assessment Reports

The radiation dose assessment reports shall be submitted within 120 days after January 1 of each year.

The Radiation Dose Assessment Report shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability. In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee will retain this summary of required meteorological data on site in a file that shall be provided to the NRC upon request. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluent to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The meteorological conditions

(continued)

3.10 Reporting Requirements

ODCMS 3.10.3 Radiation Dose Assessment Reports (continued)

concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. Approximate methods are acceptable. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual (ODCM).

The Radiation Dose Assessment Report shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous calendar year to show conformance with 40 CFR Part 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Guidance for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Revision 1, October 1977. If doses from plant effluents do not exceed twice the Appendix I limits, a statement to that effect shall constitute a 40 CFR 190 assessment.

B 3.8 ODCM SPECIFICATIONS

BASES

General

It is expected that releases of radioactive material in effluents will be kept at small fractions of the limits specified in Section 20.106 of 10 CFR, Part 20. At the same time, the licensee is permitted the flexibility of operation, compatible with consideration of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions which may temporarily result in releases higher than such small fractions, but still within the limits specified in Section 20.106 of 10 CFR, Part 20. It is expected that in using this operational flexibility under unusual operating conditions the licensee will exert his best efforts to keep levels of radioactive material in effluents as low as practicable.

Liquid Radwaste Effluents

Concentration

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents to areas at and beyond the SITE BOUNDARY will be within the concentration levels specified in 10 CFR, Part 20, Appendix B, Table II, Column 2. This instantaneous limitation provides additional assurance that the levels of radioactive materials in bodies of water in areas at or beyond the SITE BOUNDARY will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR, Part 50, to a MEMBER OF THE PUBLIC and (2) the limits of 10 CFR, Part 20.106(e) to the population. The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the International Commission on Radiological Protection (ICRP) Publication 2.

Dose

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. ODCMS 3.8.B.2 implements the guidance set forth in Section II.A of Appendix I and provides the required operating flexibility to implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as reasonably achievable." The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations described in the Offsite Dose Calculation Manual for calculating the doses due to the actual release rates

(continued)

BASES

Liquid Radwaste Effluents

Dose (continued)

of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50 Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977. This specification applies to the release of liquid effluents from the site.

Instrumentation

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential release of liquid effluents. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

System Operation

The OPERABILITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as reasonably achievable." This specification implements the requirements of 10 CFR, Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR, Part 50 and design objective Section II.D of Appendix I to 10 CFR, Part 50. The specified limits governing the use of appropriate portions of the Liquid Radwaste Treatment System were specified as a suitable fraction of the guide set forth in Section II.A of Appendix I, 10 CFR, Part 50, for liquid effluents.

The chemical/oily waste cleanup subsystem may include the portable Epicor oil/water emulsion system (as described in RW-625), an installed chemical/oily waste cleanup subsystem, or equivalent chemical/oil waste system.

(continued)

BASES (continued)

Gaseous Effluents

Dose

This specification is provided to ensure that the dose from radioactive materials in gaseous effluents at and beyond the SITE BOUNDARY will be within the annual dose limits of 10 CFR Part 20. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR part 20.106(b). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC with the appropriate occupancy factors are given in the ODCM. The specified limits restrict, at all times, the gamma and beta dose rates above background to a MEMBER OF THE PUBLIC, at or beyond the SITE BOUNDARY to ≤ 500 mrem/year to the total body or to ≤ 3000 mrem/year to the skin. These dose rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to ≤ 1500 mrem/year.

Dose, Noble Gases

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. ODCMS 3.8.C.2 implements the guidance set forth in Section II.B of Appendix I and provides the required operating flexibility to implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as reasonably achievable." ODCMS 4.8.C.2.1 implements the requirements in Section III.A of Appendix I that conformance with the guidance of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through the appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," Revision 1, October 1977 and the atmospheric dispersion model submitted on September 30, 1976, in a report titled: "Information Requested in Enclosure 2 to letter from George Lear to E. G. Bauer dated February 17, 1976." The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

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BASES

Gaseous Effluents (continued)

Dose - Iodine-131, Tritium and Radionuclides in Particulate Form

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. ODCMS 3.8.C.3 implements the guidance set forth in Section II.C of Appendix I and provides the required operating flexibility to implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as reasonably achievable." The ODCM calculational methods specified in the ODCMS 4.8.C.3.1 implements the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and the Atmospheric Dispersion Model submitted on September 30, 1976 in a report titled: "Information Requested in Enclosure 2 to letter from George Lear to E. G. Bauer dated February 17, 1976." These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for iodine-131, tritium, and radionuclides in particulate form with half lives greater than 8 days are dependent on the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

Instrumentation

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The OPERABILITY and use of instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR, Part 50.

(continued)

BASES

Gaseous Effluents (continued)

System Operation

The OPERABILITY of the Gaseous Radwaste Treatment System ensures that this system will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that appropriate portions of this system be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR, Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR, Part 50 and design objective Section II.D of Appendix I to 10 CFR, Part 50. The specified limits governing the use of appropriate portions of the Gaseous Radwaste Treatment System were specified as a suitable fraction of the guidance set forth in Section II.B and II.C of Appendix I, 10 CFR, Part 50, for gaseous effluents.

Containment PURGE

ODCMS 3.8.C.6 requires that the primary containment atmosphere receive treatment for the removal of gaseous iodine and particulates prior to release to provide reasonable assurance that PURGING operations will not result in exceeding the annual dose limits of 10 CFR Part 20 for areas at or beyond the SITE BOUNDARY.

40 CFR 190

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have now been incorporated into 10 CFR Part 30. This specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. It is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 kilometers (km) must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is

(continued)

BASES

Gaseous Effluents

40 CFR 190 (continued)

considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

Radiological Environmental Monitoring

Monitoring Program

The radiological environmental monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the two reactors at the sites. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways.

The required detection capabilities for environmental sample analyses are tabulated in terms of the Minimum Detectable Level (MDL). The MDL's required by Tables 4.8.B.1 and 4.8.C.1 of the specifications are considered optimum for routine environmental measurements in industrial laboratories. The monitoring program was developed utilizing the experience of the first seven years of commercial operation. Program changes may be initiated based on the additional operational experience.

Land Use Census

This specification is provided to ensure that significant changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50.

(continued)

BASES

Radiological Environmental Monitoring (continued)

Interlaboratory Comparison Program

The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices, are performed as part of the quality assurance program for environmental monitoring, in order to demonstrate that the results are reasonably valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

PORC	YES
SQR	YES
NQA	NO
50.59	YES
RESP MGR	YES

PECO Energy Company
Nuclear Generation Group

SOLID RADWASTE SYSTEM PROCESS
CONTROL PROGRAM (PCP)

1.0 PURPOSE

- 1.1 This program provides guidance and boundary conditions for preparation of specific procedures for processing, sampling, analysis, packaging, storage and shipment of solid radwaste in accordance with State and Federal Regulatory requirements.
CM-1

2.0 RESPONSIBILITY

NOTE: The station operating organization is outlined in each station's UFSAR.

- 2.1 PORC is responsible for reviewing all revisions to the PCP.
- 2.2 The Plant Manager is responsible for approving all revisions to the PCP.
- 2.3 The Nuclear Quality Assurance Department performs audits to verify compliance with the Nuclear Quality Assurance Program.
- 2.4 The Manager-Radwaste is responsible for:
- 2.4.1 Compliance with this Process Control Program.
- 2.4.2 Record keeping and document control of shipping and processing data.
- 2.4.3 Assuring the Radwaste personnel are appropriately trained and qualified.
- 2.4.4 Coordinating reviews and revisions to this document and corresponding implementing procedures.
- 2.4.5 Reviewing and revising this PCP and its implementing procedures.
- 2.4.6 Submitting revisions to the PCP for inclusion in the NRC Annual Radioactive Effluent Release Report.

- 2.5 The Sr. Manager-Operations is responsible for:
 - 2.5.1 Providing trained personnel to operate appropriate permanent radwaste process equipment.
 - 2.5.2 Defining those Operations positions which require training.
- 2.6 The Director-Training is responsible for:
 - 2.6.1 Development and implementation of training for designated personnel in accordance with Nuclear Training Division procedures.
- 3.0 PREREQUISITES
 - 3.1 The solid radioactive waste system shall be used in accordance with the PCP. **CM-1**
 - 3.2 Processing of solid radioactive waste shall be performed by qualified and trained personnel. Training records for operators of mobile vendor processing units shall be maintained by the Manager, Radwaste while the vendor is active on-site.
 - 3.3 Vendor services may be used for dewatering and solidification of radioactive waste processing provided the services are governed by a Topical Report and referenced in this PCP. Processing of radioactive waste by on-site vendors shall be performed in accordance with applicable Topical Report, procedures and NRC guidance. When providing solidification for stabilization, the vendor shall have an NRC approved topical report.
 - 3.4 The Topical Reports of vendor supplied Radwaste Processing Systems shall undergo review by the Manager, Radwaste (or designee). The review shall ensure the vendor supplied system will be compatible with plant operations. The review shall be performed in accordance with RW-C-106.
 - 3.5 On-site processing of radioactive waste shall be performed in accordance with approved station procedures.
 - 3.6 Quality Assurance shall be maintained as defined by the Nuclear Quality Assurance Plan.
- 4.0 PRECAUTIONS
 - 4.1 Changes to the PCP shall be submitted to the Nuclear Regulatory Commission in the Annual Radioactive Effluent Release Report for the period in which the change was made.
 - 4.2 Changes to resin dewatering implementing procedures or systems shall require verification that the free standing water content of the packaged product is within established regulatory limits.

- 4.3 Prior to revising or deleting a PCP implementing procedure, the preparer shall review the procedure against the PCP to ensure that there is no compromise or conflict.
- 4.4 Prior to modification to an in plant liquid Radwaste Processing System, the individual responsible for the MOD shall review to ensure that it is not in conflict with or compromises the PCP.
- 4.5 When processing does not meet storage requirements or shipping and transportation requirements, processing shall be suspended and the PCP reviewed for adequacy. The implementing procedures and/or the Solid Waste System shall be corrected as necessary to prevent recurrence.

5.0 APPARATUS

None

6.0 PROCEDURE

6.1 PROCESS DESCRIPTIONS

NOTE: Vendor services may be used to process any radioactive waste stream provided the services are performed with an acceptable Process Control Program.

- 6.1.1 Waste Sources (For diagrams of the Solid Radwaste Systems, see UFSAR 11.4 for LGS and 9.2 and 9.3 for PBAPS. Also refer to MOD 1750A for PBAPS).

1. Condensate Filter/Demineralizer Waste

- a. Condensate Filter/Demineralizer Waste is the waste product generated by the backwash of the condensate filter/demineralizers consisting of: 1) contaminated powdered ion exchange resins at varying degrees of exhaustion, fibrous filter media, and small concentrations of various solids and corrosion products for the precoatable element system, and 2) iron oxide and small concentrations of various other solids for the precoatless element system (LGS).
- b. A condensate filter/demineralizer backwash consists of approximately: 1) 9,000 gallons of slurry for the precoatable element system, and 2) 3,000 gallons of slurry for the precoatless element system (LGS).
- c. Backwashes are collected and settled in a Condensate Phase Separator. Clarified liquid is decanted until sufficient volume of settled spent resin and crud is obtained for processing.

- d. Phase separator contents are recirculated prior to transfer to the Dewatering System.
 - e. Slurry input to the Dewatering System is transferred to the system from the phase separators via the sludge mixing pumps.
2. Condensate Deep Bed Demineralizer Waste (Limerick Only)
- a. Condensate Deep Bed Demineralizer waste is the waste product generated by the backwash of the condensate deep bed demineralizers consisting of contaminated ion exchange bead resins at varying degrees of exhaustion and small concentrations of various solids and corrosion products.
 - b. Each condensate deep bed demineralizer backwash consists of approximately 2,800 gallons of slurry.
 - c. Each backwash contains 320 cubic feet of resin and is collected in an empty Spent Resin Receiver Tank.
 - d. During normal operations the Spent Resin Receiver Tank contents will be transferred to the Floor and Equipment Drain deep bed demineralizers for reuse.
 - e. Resins that are determined unacceptable for reuse or excess will be processed from the 1A Spent Resin Receiver Tank using vendor dewatering.
 - f. The 1A Spent Resin Receiver Tank Contents are recirculated for a minimum of 25 minutes to mix contents of the vessel, resulting in a homogeneous resin slurry.
 - g. The solids slurry is then fed to the External Processing Station for dewatering.
3. Waste Sludge Tank Waste
- a. Waste Sludge Tank Waste is the waste product generated by the backwash of the liquid radwaste and fuel pool filters and demineralizers consisting of contaminated powdered ion exchange resins and bead resins at varying degrees of exhaustion, fibrous filter media, carbon overlay material and small concentrations of various solids and corrosion products.
 - b. Backwashes from radwaste filter/demineralizers, radwaste deep bed demineralizers, and fuel pool filter/demineralizers are collected in the Waste Sludge Tank.

- c. A backwash from a radwaste filter consists of approximately 1,925 gallons of slurry (PB) or 1,500 gallons (LGS).
 - d. A backwash from the radwaste deep bed demineralizer consists of approximately 1,500 gallons of slurry.
 - e. At Peach Bottom, the Waste Sludge Tank contents are transferred to a Condensate Phase Separator for processing. From the Condensate Phase Separators, waste is processed using a Dewatering System.
 - f. At LGS, Waste Sludge Tank contents may be processed directly to the Dewatering System or transferred to a condensate phase separator for processing.
 - g. Tank contents are recirculated prior to transfer to the Dewatering System.
4. Reactor Water Cleanup Waste
- a. Reactor Water Cleanup waste is the waste product generated by the backwash of the Reactor Water Cleanup filter demineralizers consisting of contaminated powdered ion exchange resins at varying degrees of exhaustion, fibrous filter media, and small concentrations of various solids and corrosion products.
 - b. A RWCU filter-demineralizer backwash consists of approximately 1,100 gallons of slurry.
 - c. Backwashes are collected and settled in a phase separator. Clarified liquid is decanted until sufficient volume of settled spent resin and crud is obtained for processing.
 - d. Phase separator contents are recirculated prior to transfer to the Dewatering System.
5. Oily Waste
- a. Sludges and oils generated during operation and maintenance are collected in containers in appropriate approved areas throughout the plant. The filled and labeled containers are sealed and moved to available areas for temporary staging.
 - b. Sludges/oils may be decontaminated to below station free release limits and processed as non-radioactive waste.
 - c. Sludges/oils may be shipped off-site for processing/incineration.

6. Miscellaneous Waste

- a. Certain wet wastes may be allowed to dry and be treated as dry active waste or filters.

6.2 WASTE STREAM PROCESSING

6.2.1 Dewatering

1. Resin processing is normally performed using a Dewatering System.
 - a. The Dewatering System shall be operated in accordance with the system's Topical Report and applicable procedures.
 - b. The Dewatering System processes resin by using equipment described by the Topical Report to remove free standing water.
 - c. Dewatered resin is packaged in appropriately selected liners or High Integrity Containers based on waste classification.
 - d. Station to Dewatering System interfacing is addressed by an approved 10 CFR 50.59 Review.

6.2.2 Solidification

1. Resins, sludges and oily wastes may be solidified by a Mobile Solidification System.
 - a. The Mobile Solidification System shall be operated in accordance with the system's Topical Report and applicable procedures.
 - b. The Mobile Solidification System processes waste using equipment described by the Topical Report to remove free standing liquid.
 - c. Solidified waste is packaged in appropriately selected containers based on waste classification.
 - d. Station to Mobile Solidification system interfacing shall be addressed by an approved 10 CFR 50.59 Review.

6.2.3 Decontamination

1. Decontamination of oil may be performed on-site using a vendor service. The service shall be performed in accordance with the applicable 10 CFR 50.59 Review.

6.3 PRODUCT CONTROL

6.3.1 Routine sampling is performed by one of two techniques:

1. Direct sample from each batch. These samples are analyzed for activity and isotopic identity. If radionuclide distributions are shown to be consistent between similar batches, consideration may be given to decreasing the frequency of routine measurements. This constitutes routine sampling OR
2. Dose rate reading taken from a container of waste. Dose rates are converted to activity and isotopic breakdown based on annual samples.

6.3.2 Scaling factors for nuclides which are hard to identify are established for waste by analysis through an off-site vendor. Frequency of sampling is on an annual basis as a minimum.

6.3.3 Tests are performed on-site quarterly, as a minimum, to verify scaling factors. If the tests indicate that the scaling factors obtained through annual samples have changed by more than a factor of ten, consideration shall be given to increased off-site analysis.

6.3.4 When plant parameters affecting waste stream isotopics are altered (ex., fuel failure), consideration shall be given to further waste stream sampling.

6.3.5 Administrative controls for preventing unsatisfactory waste forms from being released for shipment are described in applicable station procedures. If the provisions of the PCP are not satisfied, then suspend shipments of defectively packaged solid radioactive waste from the site. **CM-1**

6.3.6 Liquid wastes or wastes containing liquids shall be converted to a form that contains as little free standing and non-corrosive liquid as is reasonably achievable. The liquid shall not exceed 1% of the volume of the waste, or 0.5% of the waste volume for solidified waste.

6.3.7 Processed resin is tested annually as a minimum to verify the free standing water content of the packaged product is within applicable regulatory limits. In addition to the annual verification, the free standing water content is determined whenever process changes occur that may significantly alter system performance. Tests are performed to verify operability and for ensuring the waste form is similar to that indicated in prequalification testing.

6.3.8 To prevent accumulation of radiolytically generated combustible gas in greater than Type A radioactive waste packages containing water and/or organic substances, one or more of the following measures are taken:

1. The container is equipped with a vent to prevent accumulation of such gas.

2. A determination by calculation, test, or measurement is performed to ensure that hydrogen generation is limited to a molar quantity that would be no more than 5% by volume of the secondary container gas void at standard temperature and pressure (STP) over a period of time that is twice the expected shipment time.

3. The cask cavity will be inerted.

6.3.9 Sufficient analyses shall be performed to verify that the quality of waste forms prepared for disposal by vendor's on-site processing shall be similar to vendor's test results.

6.4 WASTE CHARACTERISTICS FOR LAND DISPOSAL

The following are minimum requirements for all classes of waste and are intended to facilitate handling at the disposal site, provide protection of health and safety of personnel at the disposal site, and ensure that wet radioactive wastes are processed to meet shipping and burial ground requirements:

CM-1

6.4.1 Waste shall not be packaged for disposal in cardboard or fiberboard containers.

6.4.2 Waste shall not be readily capable of detonation or of explosive decomposition or reaction at normal pressure and temperatures, or of explosive reaction with water.

6.4.3 Waste shall not contain, or be capable of generating, quantities of toxic gases, vapors, or fumes harmful to persons transporting, handling, or disposing of the waste.

6.4.4 Waste shall not be pyrophoric. Pyrophoric materials contained in the waste shall be treated, prepared, and packaged to be nonflammable.

6.4.5 Waste containing hazardous, biological, pathogenic or infectious material shall be handled in a manner that minimizes the potential hazard from the non-radiological materials.

6.4.6 Each waste shipment shall be accompanied by a shipping manifest giving a physical description of the waste, the volume, the radionuclide identity and quantity, the total radioactivity, the principal chemical form, and waste class.

6.5 WASTE STABILITY

The following requirements are intended to provide stability and are applicable to Class B and Class C wastes:

6.5.1 Waste shall have structural stability which can be provided by the waste form itself, processing the waste to a stable form, or placing the waste in a disposal container or structure that provides stability.

6.5.2 Void spaces within the waste and its package should be reduced to the extent practicable.

6.5.3 Waste requiring stabilization is normally packaged in approved High Integrity Containers (HIC). The HICs are handled and stored in accordance with the applicable Certificate of Conformance (C of C). Upon arrival at the burial site, Class B and C wastes are placed in concrete overpacks for structural stability.

6.6 SURVEILLANCE REQUIREMENTS FOR SOLIDIFICATION

Perform the following prior to the shipment of solidified radioactive waste from the site: **CM-1**

6.6.1 At least one representative test specimen from at least every tenth batch of waste shall be solidified.

6.6.2 If any test specimen fails to verify solidification, the solidification of the batch under test shall be suspended until additional test specimens can be obtained, alternative solidification parameters can be determined and a subsequent test verifies solidification.

6.6.3 If the initial test specimen from a batch of waste fails to verify solidification, representative test specimens from consecutive batches shall be tested until at least three consecutive test specimens demonstrate solidification. The solidification process control program shall be modified as required to assure solidification of subsequent batches.

6.7 QUALITY ASSURANCE PROGRAM

6.7.1 Quality Assurance shall be maintained as defined by the Nuclear QA Plan.

6.7.2 Audits by personnel independent of the activities are performed and reviewed by appropriate management personnel.

6.7.3 Administrative controls for preventing unsatisfactory waste forms from being released for shipment are described in applicable station procedures.

6.7.4 Procedures are reviewed to ensure compliance with the requirements and process parameters of this PCP.

6.7.5 Radioactive wastes not described within this document shall be evaluated for inclusion in this PCP or in a vendor Process Control Program prior to processing.

6.8 REVISIONS

6.8.1 Revisions to the PCP shall be approved in accordance with applicable station or Common Nuclear Generation Group Procedures.

6.8.2 Any changes to the Solid Radwaste Process Control Program shall be submitted for input to the Annual Radioactive Effluent Release Report.

6.9 DOCUMENTATION

6.9.1 Records of all data, tests, analysis results, and records of training, inspection, and audits are maintained in accordance with applicable procedures.

6.9.2 Waste classification records, waste form records, and other records required for the preparation of the Annual Radioactive Effluent Release Report shall be prepared and retained in accordance with the requirements of 10CFR20, 10CFR71, 49CFR170-178, and Station Technical Specifications.

6.9.3 Sufficient documentation shall be maintained to demonstrate compliance with this PCP.

| 7.0 REFERENCES

7.1 SOURCE DOCUMENTS

7.1.1 Low Level Waste Licensing Branch Technical Position on Waste Form.

7.1.2 Low Level Waste Licensing Branch Technical Position on Waste Classification.

7.1.3 Limerick Generating Station Unit 1 and 2 Technical Specifications, 6.5.1.6, 6.8.1, 6.13.

7.1.4 Updated Final Safety Analysis Report Peach Bottom Atomic Power Station Units 2 and 3, and Limerick Generating Station Units 1 and 2.

| 7.1.5 PBAPS Technical Specifications Section 5.6.3.

| 7.1.6 PBAPS ODCMS 3.9.3.

7.1.7 49 CFR Parts 170 through 178.

7.1.8 10 CFR Parts, 20, 50, 61, and 71.

7.1.9 Standard Review Plan 11.4, including Branch Technical Position ETSB11-3.

7.1.10 Pacific Nuclear: NuPac Services Topical Report for Dewatering System TP-02-P-A.

7.1.11 General Criteria for High Integrity Containers (SCDHEC).

7.1.12 Westinghouse-Hittman Nuclear, Incorporated (SEG) Topical Report STD-R-05-011NP-A for Mobile In Container Dewatering and Solidification System (MDSS).

7.1.13 CM-1, Letter from G. A. Hunger, Jr. dated Sept. 29, 1994 transmitting TSCR 93-16 (Reference A/R A0905923, E02)

7.2 CROSS REFERENCES

7.2.1 RW-C-106, Review of Vendor Topical Report for Waste Processing

7.2.2 PCP Implementing Procedures

NOTE: PCP Implementing Procedures are those which apply directly to processing, packaging, sampling, analyzing, and shipping radwaste.

1. Limerick and Peach Bottom Radwaste Procedures (RW) or Applicable Common Procedure.
2. Limerick and Peach Bottom Surveillance Test Procedures (ST) and Routine Test Procedures (RT).
3. Limerick System (S) and Peach Bottom System Operating (SO) Procedures.

7.2.3 A-C-930, Radwaste and Radioactive Material QA Program

7.2.4 Generic Letter 89-01 (Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Off-site Dose Calculation Manual or to the Process Control Program), 1/31/89.

7.2.5 Off Site Dose Calculation Manual

7.2.6 Limerick and Peach Bottom Chemistry Procedures or Applicable Common Procedures.

7.2.7 Safety Evaluation for Oil Decontamination, MOD 1259A (Peach Bottom only).

7.2.8 Safety Evaluation for MOD 1750A, Radwaste Dewatering Facility Peach Bottom Atomic Power Station (Peach Bottom only).

7.2.9 NRC Generic Letter 91-02, "Reporting Mishaps Involving LLW Forms Prepared for Disposal.

7.2.10 PECO 10CFR50.59 Review for Scientific Ecology Group Dewatering System Operation.

7.2.11 I.E. Circular 80-18, 10CFR50.59 Safety Evaluation for Changes to Radioactive Waste Treatment Systems.

8.0 ATTACHMENTS

None