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50-251 Turkey Point Plant, Unit 4, Florida Power and Light C 05000251
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SUBJECT: Turkey Points Units 3 & 4 Annual Radioactive Effluent
Release Rept for Jan-Dec 1993.W/940331 ltr.

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L-94-072
10 CFR 50.36(a)(2)

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Annual Radioactive Effluent Release Report

Attached is the Radioactive Effluent Release Report for the period of January 1, 1993, through December 31, 1993, for Turkey Point Units 3 and 4, as required by Technical Specification 6.9.1.4 and 10 CFR 50.36 (a)(2).

No gas storage tanks exceeded the limits allowed by Technical Specification 3.11.2.6 during the reporting period.

In accordance with the revisions of 10 CFR 20 and Turkey Point's Technical Specifications, changes were made to the Offsite Dose Calculation Manual. These changes are included in Attachment 1.

There were no continuous liquid effluent releases above the lower limit of detection for either Turkey Point Unit 3 or 4 during this period and therefore this information has not been included in this report.

In accordance with Technical Specification 6.9.1.4, the meterological data is available on site and shall be provided to the NRC upon request.

Should there be any questions or comments regarding this information, please contact us.

Very truly yours,

R. W. Pearce for

T. F. Plunkett
Vice President
Turkey Point Plant

TFP/RJT/rt

Attachment

cc: S. D. Ebneter, Regional Administrator, Region II, USNRC
T. P. Johnson, Sr. Resident Inspector, USNRC, Turkey Point Plant

IE48

Turkey Point Plant
Units 3 and 4

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

January 1993 through December 1993

Submitted by

NUCLEAR CHEMISTRY DEPARTMENT
TURKEY POINT PLANT
FLORIDA POWER AND LIGHT COMPANY

J.E. Berg, Radiochemistry Supervisor

For

R.N. Steinke, Chemistry Supervisor

D.E. Jernigan, Operations Manager

L..W. Pearce, Plant General Manager

9404110185

TURKEY POINT UNITS 3 AND 4
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
JANUARY 1993 THROUGH DECEMBER 1993

INDEX

1.0 REGULATORY LIMITS

1.1 Liquid Effluents

(a) The concentration of radioactive material released in liquid effluents to unrestricted areas shall not exceed the concentration specified in 10CFR20 Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained gases. For dissolved or entrained noble gases, the concentration shall not exceed 2.0E-04 micro curies per milliliter.

(b) The dose or dose commitment per reactor to a member of the public from any radioactive materials in liquid effluents released to unrestricted areas shall be limited as follows:

- During any calendar quarter, to less than or equal to 1.5 mrem to the total body and less than or equal to 5 mrem to any organ.
- During any calendar year, to less than or equal to 3 mrem to the total body and less than or equal to 10 mrem to any organ.

1.2 Gaseous Effluents

(a) The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary shall be limited to the following:

- Less than or equal to 500 mrem/year to the total body and less than or equal to 3000 mrem/year to the skin due to noble gases.
- Less than or equal to 1500 mrem/year to any organ due to I-131, I-133, tritium, and for all radioactive materials in particulate form with half-lives greater than 8 days.

(b) The air dose per reactor to areas at and beyond the site boundary due to noble gases released in gaseous effluents shall be limited to:

- During any calendar quarter, to less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation.
- During any calendar year, to less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

(c) The dose per reactor to a member of the public, due to I-131, I-133, tritium, and particulate with half-lives greater than 8 days in airborne effluents released to areas at and beyond the site boundary shall not exceed 7.5 mrem to any organ during any calendar quarter and shall not exceed 15 mrem to any organ during any calendar year.

2.0 MAXIMUM PERMISSIBLE CONCENTRATION

Water : In accordance with 10CFR20, Appendix B, Table II, Column 2, for entrained or dissolved noble gases as described in 1.1.a of this report.

Air : Release concentrations are limited to dose rate limits described in 1.2.a of this report.

3.0 AVERAGE ENERGY

The average energy of fission and activation gases in effluents is not applicable.

4.0 MEASUREMENTS AND APPROXIMATIONS OF TOTAL RADIOACTIVITY

All liquid and airborne discharges to the environment during this period were analyzed in accordance with Technical Specification requirements. The minimum frequency of analysis as required by Regulatory Guide 1.21 was met or exceeded.

When alpha, tritium and named nuclides are shown as " - - " curies on the following tables, this should be interpreted as 'no activity' was detected on the samples using the Plant Technical Specification analysis techniques to achieve the required Lower Limit of Detection ("LLD") sensitivity for radioactive effluents.

4.1 Liquid Effluents

Aliquots of representative pre-release samples, from waste disposal system, were isotopically analyzed for gamma emitting isotopes on a multichannel analyzer.

Frequent periodic sampling and analysis were used to conservatively determine if any radioactivity was being released via the steam generator blowdown system and the storm drain system.

Monthly and quarterly composite samples for the waste disposal system were prepared to give proportional weight to each liquid release made during the designated period of accumulation. The monthly composite was analyzed for tritium and gross alpha radioactivity. Tritium was determined by use of liquid scintillation techniques and gross alpha radioactivity was determined by use of a solid state scintillation system. The quarterly composite was analyzed for Sr-89, Sr-90 and Fe-55 by chemical separation.

All radioactivity concentrations determined from analysis of a pre-release composite were multiplied by the total represented volume of the liquid waste released to determine the total quantity of each isotope and of gross alpha activity released during the compositing period.

Aliquots of representative pre-release samples from the waste disposal system were analyzed on a per-release basis of gamma spectrum analysis. The resulting isotope concentrations were multiplied by the total volume released in order to estimate the total dissolved gases released.

The liquid waste treatment system is shared by both units at the site and generally all liquid releases are allocated on a 50%-50% basis to each unit respectively.

There were no continuous liquid effluent releases above the lower limit of detection for either Unit 3 or Unit 4 during this reporting period and therefore have been omitted from Table 2 of this report.

4.2 Gaseous Effluents

Airborne releases to the atmosphere occurred from : release of Gas Decay Tanks, the Containment Instrument Bleed Line, Containment Purges, and releases incidental to operation of the plant. The techniques employed in determining the radioactivity in airborne releases are:

a) Gamma spectrum analysis for fission and activation gases,
b) Removal of particulate material by filtration and subsequent gamma spectrum analysis, Sr-89, Sr-90 determination and gross alpha.
c) Absorption of halogen radionuclides on a charcoal filter and subsequent gamma spectrum analysis, and
d) Analysis of water vapor in a gas sample for tritium using liquid scintillation techniques.

All gas releases from the plant which were not accounted for by the above methods were conservatively estimated as curies of Xe-133 by use of the SPING-4 radiation monitor and the Plant Vent process monitor recorder chart and the current calibration curve for the monitor.

Portions of the gas waste treatment system are shared by both units and generally all gas releases from the shared system are allocated on a 50/50 basis to each unit.

Meteorological data for the period January 1993 through December 1993, in the form of Joint Frequency Distribution Tables, is maintained on-site.

4.3 Estimation of Errors

a) Sampling Error

The error associated with volume measurement devices, flow measuring devices, etc. , based on calibration data and design tolerances has been conservatively estimated to be collectively less than ±10%.

b) Analytical Error

Our quarterly Q.C. cross-check program involves counting unknown samples provided by an independent external lab. The errors associated with our analysis of these unknown samples, and reported to us by the independent lab, were used as the basis for deriving the following analytical error terms. The tritium results reported in Tables 1 and 3 for Units 3 and 4 of this report were increased by a factor of 1.32 due to a suspected error in the tritium analysis. A revision to this report will be issued to the NRC within thirty days of completion of the investigation.

NUCLIDE TYPE	AVERAGE ERROR	MAXIMUM ERROR
Liquid	±6.1%	±17.0%
Gaseous	±6.2%	±12.4%

5.0 BATCH RELEASES

5.1 LIQUID

	UNIT 3	UNIT 4
a) Number of releases	2.39E+02	2.39E+02
b) Total time period of batch releases,minutes	1.86E+04	1.86E+04
c) Maximum time period for a batch release, minutes	1.65E+02	1.65E+02
d) Average time period for a batch release, minutes	7.64E+01	7.64E+01
e) Minimum time for a batch release, minutes	3.00E+01	3.00E+01
f) Average stream flow during period of release of effluent into a flowing stream, liters-per-minute	3.01E+06	3.01E+06

5.2 GASEOUS

	UNIT 3	UNIT 4
a) Number of batch releases	1.20 E+01	1.10 E+01
b) Total time period of batch releases,minutes	8.70 E+02	6.30 E+02
c) Maximum time period for a batch release, minutes	2.40E+02	2.40E+02
d) Average time period for a batch release, minutes	7.25 E+01	5.72 E+02
e) Minimum time for a batch release, minutes	1.00E+01	1.00E+01

6.0 UNPLANNED RELEASES

6.1 Liquid

There were no unplanned liquid releases this period for either Unit 3 or Unit 4.

6.2 Gaseous

There were no unplanned gaseous releases this period for either Unit 3 or Unit 4.

7.0 REACTOR COOLANT ACTIVITY

7.1 Unit 3

Reactor coolant activity limits of 100/E-bar and 1.0 μCi/gram Dose Equivalent I-131 were not exceeded.

7.2 Unit 4

Reactor coolant activity limits of 100/E-bar and 1.0 μCi/gram Dose Equivalent I-131 were not exceeded.

8.0 SITE RADIATION DOSE

The assessment of radiation dose from radioactive effluents to the general public due to their activities inside the site boundary assumes a visitor was onsite at the "Red Barn" recreational area for twelve hours per day, two days each week of the year, receiving exposure from both Units at Turkey Point. The "Red Barn" is located approximately 0.39 miles NNE of the plant. Specific activities used in these calculations are the sum of these activities in Unit 3, Table 3, and Unit 4, Table 3. These dose calculations were made using historical, meteorological data.

Florida Power and Light established a temporary Day Care facility at the "Red Barn" recreational area during the entire 1993 period. The assessment of radiation dose from radioactive effluents to the occupants of the Day Care facility assumes that a person was at the facility ten hours per day, five days each week of the year, receiving exposure from both Units at Turkey Point. The "Red Barn" is located approximately 0.39 miles NNE of the plant. Specific activities used in these calculations are the sum of these

activities in Unit 3, Table 3, and Unit 4, Table 3. These dose calculations were made using historical, meteorological data.

Florida Power and Light established a satellite public school for kindergarten and first grade children. The assessment of radiation dose from radioactive effluents to the school of the satellite school assumes that a child was at the school ten hours per day, five days each week for twenty weeks of the year, receiving exposure from both Units at Turkey Point. The satellite school is located approximate 1.75 miles WNW of the plant. Specific activities used in these calculations are the sum of these activities in Unit 3, Table 3, and Unit 4, Table 3. These dose calculations were made using historical, meteorological data.

VISITOR DOSE

	ADULT INHALATION	CHILD INHALATION	CHILD INHALATION
	RED BARN	RED BARN	SATELLITE SCHOOL
	mrem	mrem	mrem
BONE	2.23E-08	1.07E-07	1.13E-07
LIVER	3.22E-08	5.07E-06	5.35E-06
THYROID	1.02E-05	4.02E-05	4.25E-05
KIDNEY	5.47E-08	3.34E-06	3.53E-06
LUNG	2.10E-08	4.96E-06	5.23E-06
GI-LLI	2.21E-08	4.97E-06	5.24E-06
TOTAL BODY	2.15E-08	5.03E-06	5.31E-06

	mrad	mrad	mrad
Gamma Air Dose	3.66E-03	3.66E-03	2.14E-03
Beta Air Dose	1.05E-02	1.05E-02	6.17E-03

9.0 OFFSITE DOSE CALCULATION MANUAL REVISIONS

Attachment 1 is the revision to the ODCM.

10.0 SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

No irradiated fuel shipments were made from the site. Common solid waste from Turkey Point Units 3 and 4 was shipped jointly. A summation of these shipments is given in Table 6 of this report.

11.0 PROCESS CONTROL PROGRAM REVISIONS

There were no changes to the Process Control Program during this reporting period.

12.0 INOPERABLE EFFLUENT MONITORING INSTRUMENTATION

12.1 Unit 3 Steam Jet Air Ejector Vent

The Steam Jet Air Ejector (SJAE) effluent monitoring instrumentation required by Technical Specification 3.3.3.3, Table 3.3-5, item 19.c, was declared out of service on June 10, 1993, due to moisture intrusion into the instrument. Alternate sampling equipment was placed in service for continuous monitoring of iodine and particulate activity, gas grab sample were obtained at twelve-hour intervals to monitor gaseous releases, and the alternate sampling flowrate measurements were made at the prescribed frequencies. Temporary modifications to the system were made to prevent moisture from entering the monitor. The SJAE was placed back into service on June 24, 1993. A permanent plant change, PC/M 93-136, was implemented on July 24, 1993, with no further incidents of moisture intrusion.

LIQUID EFFLUENTS SUMMARY

UNIT 3
TABLE 1

A. FISSION AND ACTIVATION PRODUCTS

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release (not including tritium,gases, alpha)	Ci	2.48E-02	8.86E-02	1.13E-01	4.98E-02	4.45
2. Average diluted concentration during the period	uCi/ml	7.08E-10	1.47E-10	3.52E-11	9.30E-11	
3. Percent of applicable limit	%	8.27E-03	2.32E-01	1.96E-01	1.31E-01	

B. TRITIUM

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	6.71E+01	4.98E+01	1.25E+02	4.55E+01	9.80
2. Average diluted concentration during the period	uCi/ml	1.95E-06	1.74E-06	3.76E-06	2.98E-06	
3. Percent of applicable limit	%	1.95E-01	1.74E-01	3.76E-01	2.98E-01	

C. DISSOLVED AND ENTRAINED GASES

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	9.20E-03	8.93E-03	3.44E-03	5.90E-04	4.45
2. Average diluted concentration during the period	uCi/ml	2.67E-10	3.11E-10	1.03E-10	3.86E-11	
3. Percent of applicable limit	%	1.33E-04	1.56E-04	5.15E-05	1.93E-05	

D. GROSS ALPHA RADIOACTIVITY

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	--	--	--	--	11.25

E. LIQUID VOLUMES

		Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Batch waste released, prior to dilution	LITERS	1.02E+06	1.98E+06	8.93E+05	6.77E+05	10.00
2. Continuous waste released, prior to dilution	LITERS	--	--	--	--	
3. Dilution water used during period	LITERS	3.45E+10	2.87E+10	3.34E+10	1.53E+10	

LIQUID EFFLUENTS SUMMARY

UNIT 3
TABLE 2

NUCLIDES RELEASED	UNITS	BATCH MODE			
		Qtr 1	Qtr 2	Qtr 3	Qtr 4
Na-24	Ci	--	--	--	--
Cr-51	Ci	4.28E-05	1.87E-04	1.83E-04	--
Mn-54	Ci	5.87E-03	6.32E-03	2.57E-03	2.43E-03
Fe-55	Ci	3.11E-04	3.86E-02	2.84E-02	4.12E-02
Co-58	Ci	3.28E-03	1.53E-02	6.81E-03	2.44E-02
Fe-59	Ci	--	6.95E-06	2.82E-05	5.35E-07
Co-60	Ci	5.85E-03	1.13E-02	6.11E-03	2.38E-03
Zn-65	Ci	--	--	--	--
Sr-90	Ci	--	3.27E-03	1.98E-04	8.56E-05
Nb-95	Ci	3.30E-05	1.53E-05	4.62E-04	8.95E-05
Ru-103	Ci	--	--	--	--
Ag-110	Ci	4.00E-03	2.08E-04	1.80E-03	5.22E-04
Sn-113	Ci	--	--	--	--
Sb-124	Ci	2.51E-04	3.31E-05	--	2.36E-05
Sb-125	Ci	2.28E-03	3.45E-04	5.03E-04	1.14E-03
I-131	Ci	5.83E-04	1.71E-04	4.09E-04	5.22E-04
I-133	Ci	--	6.15E-06	5.01E-05	--
Cs-134	Ci	2.92E-04	1.77E-04	2.51E-04	2.16E-04
I-134	Ci	--	--	--	--
Cs-137	Ci	1.66E-03	1.03E-03	1.56E-03	1.24E-03
La-140	Ci	9.89E-05	1.16E-05	1.61E-05	2.86E-05
W-187	Ci	1.48E-04	--	2.01E-04	7.20E-05
TOTAL FOR PERIOD	Ci	2.47E-02	7.70E-02	4.95E-02	7.43E-02

LIQUID EFFLUENTS - DISSOLVED GAS SUMMARY

UNIT 3
TABLE 2

NUCLIDES RELEASED	UNITS	BATCH MODE			
		Qtr 1	Qtr 2	Qtr 3	Qtr 4
Ar-41	Ci	--	--	--	--
Kr-85m	Ci	--	--	--	--
Kr-87	Ci	--	--	--	--
Xe-133	Ci	9.20E-03	8.93E-03	2.97E-04	5.90E-04
Xe-133m	Ci	--	--	--	--
Xe-135	Ci	--	--	--	--
TOTAL FOR PERIOD	Ci	9.20E-03	8.93E-03	2.97E-04	5.90E-04

LIQUID EFFLUENTS - DOSE SUMMATION

Age group : Teenager *Location : Cooling Canal*		
Shoreline Deposition	Dose (mrem)	% of Annual Limit
TOTAL BODY	1.34E-03	4.46E-02

GASEOUS EFFLUENTS SUMMARY

UNIT 3
TABLE 3

A. FISSION AND ACTIVATION PRODUCTS

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	3.23E+01	1.73E+02	6.55E+00	1.29E+02	5.25
2. Average release rate for the period	uCi/sec	4.02E-06	2.15E-05	8.15E-07	1.60E-05	
3. Percent of Technical Specification Limit	%	6.35E-10	3.47E-09	1.04E-10	4.81E-10	

B. IODINES

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	1.74E-04	5.99E-04	5.12E-05	6.26E-04	6.25
2. Average release rate for the period	uCi/sec	1.35E-09	4.57E-09	3.86E-10	4.73E-09	
3. Percent of Technical Specification Limit	%	3.38E-04	1.16E-03	9.90E-05	1.21E-03	

C. PARTICULATES

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Particulates with half-life >8 days	Ci	9.35E-07	--	--	8.50E-07	8.75
2. Average release rate for the period	uCi/sec	7.21E-12	--	--	6.42E-12	
3. Percent of Technical Specification Limit	%	†	†	†	†	
4. Gross Alpha Radioactivity	Ci	--	--	--	--	

D. TRITIUM

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	5.91E+00	--	--	--	9.90
2. Average release rate for the period	uCi/sec	4.56E-05	--	--	--	
3. Percent of Technical Specification Limit	%	†	†	†	†	

† NOTE : THESE PERCENTAGES ARE INCLUDED IN THE IODINE LIMIT CALCULATION

GASEOUS EFFLUENTS SUMMARY

UNIT 3
TABLE 4

A . FISSION GASES

NUCLIDES RELEASED	UNITS	BATCH MODE Qtr 1	Qtr 2	Qtr 3	Qtr 4
Kr-85m	Ci	1.15E-04	--	2.72E-06	8.90E-04
Kr-87	Ci	--	--	--	--
Xe-131m	Ci	2.20E-02	1.01E-01	2.20E-02	3.28E-01
Xe-133	Ci	9.25E+00	1.49E+00	3.76E-01	1.32E+01
Xe-133m	Ci	6.73E-02	--	4.56E-04	9.22E-02
Xe-135	Ci	2.63E-02	--	1.79E-04	1.23E-01
Xe-135m	Ci	--	--	--	--
TOTAL FOR PERIOD	Ci	9.37E+00	1.59E+00	3.99E-01	1.37E+01

NUCLIDES RELEASED	UNITS	CONTINUOUS MODE Qtr 1	Qtr 2	Qtr 3	Qtr 4
Ar-41	Ci	--	--	--	--
Kr-85	Ci	--	--	--	--
Kr-85m	Ci	1.03E-04	--	--	--
Kr-87	Ci	--	--	--	--
Kr-88	Ci	--	--	--	--
Xe-131m	Ci	--	2.41E+00	--	--
Xe-133	Ci	2.24E+01	1.62E+02	5.64E+00	6.75E+00
Xe-133m	Ci	--	--	--	--
Xe-135	Ci	4.81E-01	6.00E+00	1.39E-04	--
Xe-135m	Ci	--	--	--	--
Xe-138	Ci	--	--	--	--
TOTAL FOR PERIOD	Ci	2.29E+01	1.70E+02	5.64E+00	6.75E+00

B. IODINES

NUCLIDES RELEASED	UNITS	CONTINUOUS MODE Qtr 1	Qtr 2	Qtr 3	Qtr 4
Br-82	Ci	--	1.48E-05	--	--
I-131	Ci	1.66E-04	5.58E-04	9.60E-06	3.94E-04
I-133	Ci	8.10E-06	2.56E-05	--	2.32E-04
TOTAL FOR PERIOD	Ci	1.74E-04	5.99E-04	9.60E-06	6.26E-04

C. PARTICULATES

NUCLIDES RELEASED	UNITS	CONTINUOUS MODE Qtr 1	Qtr 2	Qtr 3	Qtr 4
Co-58	Ci	--	--	--	8.50E-07
Co-60	Ci	--	--	--	--
Cs-137	Ci	9.35E-07	--	--	--
TOTAL FOR PERIOD	Ci	9.35E-07	--	--	8.50E-07

DOSES DUE TO IODINE,TRITIUM, AND PARTICULATES

UNIT 3
TABLE 5

PATHWAY	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	SKIN	TOTAL BODY
Cow milk - Infant	4.76E-05	1.01E-04	1.79E-02	3.35E-05	4.45E-05	4.65E-05	- -	7.70E-05
Fruit & Veg Fresh	1.42E-06	9.57E-06	6.65E-04	1.10E-05	7.53E-06	8.07E-06	- -	8.69E-06
Ground Plane	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	5.74E-07	4.79E-07
Inhalation - Adult	9.88E-08	1.42E-07	4.50E-05	2.42E-07	9.29E-08	9.78E-08	9.01E-08	9.52E-08
TOTAL (mrem)	4.96E-05	1.11E-04	1.86E-02	4.52E-05	5.26E-05	5.52E-05	6.64E-07	8.62E-05
% of Annual Limit	3.31E-04	7.42E-04	1.24E-01	3.01E-04	3.51E-04	3.68E-04	4.43E-06	5.75E-04

DOSE DUE TO NOBLE GASES

	mrad	% of Annual Limit
Gamma Air Dose	1.70E-03	1.70E-02
Beta Air Dose	8.53E-03	8.53E-02

LIQUID EFFLUENTS SUMMARY

UNIT 4
TABLE 1

A. FISSION AND ACTIVATION PRODUCTS

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release (not including tritium,gases, alpha)	Ci	2.48E-02	8.86E-02	4.98E-02	7.53E-02	4.45
2. Average diluted concentration during the period	uCi/ml	3.54E-10	7.35E-11	1.76E-11	4.65E-11	
3. Percent of applicable limit	%	8.27E-03	2.32E-01	1.96E-01	1.31E-01	

B. TRITIUM

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	6.71E+01	4.98E+01	1.25E+02	4.55E+01	9.80
2. Average diluted concentration during the period	uCi/ml	1.95E-06	1.74E-06	3.76E-06	2.98E-06	
3. Percent of applicable limit	%	1.95E-01	1.74E-01	3.76E-01	2.98E-01	

C. DISSOLVED AND ENTRAINED GASES

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	9.20E-03	8.93E-03	3.44E-03	5.90E-04	4.45
2. Average diluted concentration during the period	uCi/ml	2.67E-10	3.11E-10	1.03E-10	3.86E-11	
3. Percent of applicable limit	%	1.33E-04	1.56E-04	5.15E-05	1.93E-05	

D. GROSS ALPHA RADIOACTIVITY

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	--	--	--	--	11.25

E. LIQUID VOLUMES

		Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Batch waste released, prior to dilution	LITERS	1.02E+06	1.98E+06	8.93E+05	6.77E+05	10.00
2. Continuous waste released, prior to dilution	LITERS	--	--	--	--	
3. Dilution water used during period	LITERS	3.45E+10	2.87E+10	3.34E+10	1.53E+10	

LIQUID EFFLUENTS SUMMARY

UNIT 4
TABLE 2

NUCLIDES RELEASED	UNITS	BATCH MODE Qtr 1	Qtr 2	Qtr 3	Qtr 4
Na-24	Ci	--	--	--	--
Cr-51	Ci	4.28E-05	1.87E-04	1.83E-04	--
Mn-54	Ci	5.87E-03	6.32E-03	2.57E-03	2.43E-03
Fe-55	Ci	--	4.10E-05	1.22E-05	1.38E-05
Co-58	Ci	3.28E-03	1.53E-02	6.81E-03	2.44E-02
Fe-59	Ci	--	6.95E-06	2.82E-05	5.35E-07
Co-60	Ci	5.85E-03	1.13E-02	6.11E-03	2.38E-03
Zn-65	Ci	--	--	--	--
Sr-90	Ci	3.30E-05	1.53E-05	4.62E-04	8.95E-05
Nb-95	Ci	5.63E-05	--	3.57E-05	1.58E-05
Ru-103	Ci	--	--	3.11E-05	--
Ag-110	Ci	--	--	--	--
Sn-113	Ci	4.00E-03	2.08E-04	1.80E-03	5.22E-04
Sb-124	Ci	--	--	--	--
Sb-125	Ci	--	--	--	--
I-131	Ci	2.51E-04	3.31E-05	--	2.36E-05
I-133	Ci	2.28E-03	3.45E-04	5.03E-04	1.14E-03
Cs-134	Ci	5.83E-04	1.71E-04	4.09E-04	5.22E-04
I-134	Ci	--	6.15E-06	5.01E-05	--
Cs-137	Ci	--	--	--	--
La-140	Ci	2.92E-04	1.77E-04	2.51E-04	2.16E-04
W-187	Ci	--	--	5.50E-06	--
TOTAL FOR PERIOD	Ci	2.25E-02	3.41E-02	1.93E-02	3.17E-02

LIQUID EFFLUENTS - DISSOLVED GAS SUMMARY

NUCLIDES RELEASED	UNITS	BATCH MODE Qtr 1	Qtr 2	Qtr 3	Qtr 4
Ar-41	Ci	--	--	--	--
Kr-85m	Ci	--	--	--	--
Kr-87	Ci	--	--	--	--
Xe-133	Ci	9.20E-03	8.93E-03	2.97E-04	5.90E-04
Xe-133m	Ci	--	--	--	--
Xe-135	Ci	--	--	--	--
TOTAL FOR PERIOD	Ci	9.20E-03	8.93E-03	2.97E-04	5.90E-04

LIQUID EFFLUENTS - DOSE SUMMATION

Age group : Teenager *Location : Cooling Canal*		
Shoreline Deposition	Dose (mrem)	% of Annual Limit
TOTAL BODY	1.34E-03	4.46E-02

GASEOUS EFFLUENTS SUMMARY

UNIT 4
TABLE 3

A. FISSION AND ACTIVATION PRODUCTS

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	1.45E+01	1.91E+02	6.63E+00	1.16E+01	5.25
2. Average release rate for the period	uCi/sec	1.81E-06	2.37E-05	8.25E-07	1.44E-06	
3. Percent of Technical Specification Limit	%	2.93E-10	3.90E-09	1.06E-10	2.00E-10	

B. IODINES

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	1.74E-04	5.99E-04	5.12E-05	6.26E-04	6.25
2. Average release rate for the period	uCi/sec	1.35E-09	4.57E-09	3.86E-10	4.73E-09	
3. Percent of Technical Specification Limit	%	3.38E-04	1.16E-03	9.90E-05	1.21E-03	

C. PARTICULATES

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Particulates with half-life >8 days	Ci	9.34E-07	--	--	8.50E-07	8.75
2. Average release rate for the period	uCi/sec	7.12E-12	--	--	6.49E-12	
3. Percent of Technical Specification Limit	%	†	†	†	†	
4. Gross Alpha Radioactivity	Ci	--	--	--	--	

D. TRITIUM

	UNITS	Qtr 1	Qtr 2	Qtr 3	Qtr 4	Est. Error (%)
1. Total Release	Ci	3.26E+00	2.09E-02	6.63E-02	--	9.90
2. Average release rate for the period	uCi/sec	2.52E-05	1.61E-07	5.11E-07	--	
3. Percent of Technical Specification Limit	%	†	†	†	†	

† NOTE : THESE PERCENTAGES ARE INCLUDED IN THE IODINE LIMIT CALCULATION

GASEOUS EFFLUENTS SUMMARY

UNIT 4
TABLE 4

Shoreline Depositi
TOTAL BODY

A . FISSION GASES

NUCLIDES RELEASED	UNITS	BATCH MODE Qtr 1	Qtr 2	Qtr 3	Qtr 4
Kr-85m	Ci	1.15E-04	4.10E-03	2.72E-06	8.90E-04
Kr-87	Ci	--	--	--	--
Xe-131m	Ci	2.20E-02	4.01E-01	2.20E-02	1.37E-01
Xe-133	Ci	1.25E+00	1.85E+01	4.50E-01	6.85E+00
Xe-133m	Ci	2.80E-03	1.62E-01	4.56E-04	3.69E-02
Xe-135	Ci	2.88E-03	1.06E-01	2.67E-03	1.99E-02
Xe-135m	Ci	--	--	--	--
TOTAL FOR PERIOD	Ci	1.28E+00	1.92E+01	4.75E-01	7.04E+00

NUCLIDES RELEASED	UNITS	CONTINUOUS MODE Qtr 1	Qtr 2	Qtr 3	Qtr 4
Ar-41	Ci	--	--	--	--
Kr-85	Ci	--	--	--	--
Kr-85m	Ci	--	--	--	--
Kr-87	Ci	--	--	--	--
Kr-88	Ci	--	--	--	--
Xe-131m	Ci	--	2.41E+00	--	--
Xe-133	Ci	1.28E+01	1.62E+02	5.64E+00	4.53E+00
Xe-133m	Ci	--	--	--	--
Xe-135	Ci	4.80E-01	6.00E+00	--	--
Xe-135m	Ci	--	--	--	--
Xe-138	Ci	--	--	--	--
TOTAL FOR PERIOD	Ci	1.33E+01	1.70E+02	5.64E+00	4.53E+00

B. IODINES

NUCLIDES RELEASED	UNITS	CONTINUOUS MODE Qtr 1	Qtr 2	Qtr 3	Qtr 4
Br-82	Ci	--	1.48E-05	--	--
I-131	Ci	1.66E-04	5.58E-04	9.60E-06	3.94E-04
I-133	Ci	8.10E-06	2.56E-05	--	2.32E-04
TOTAL FOR PERIOD	Ci	1.74E-04	5.99E-04	9.60E-06	6.26E-04

C. PARTICULATES

NUCLIDES RELEASED	UNITS	CONTINUOUS MODE Qtr 1	Qtr 2	Qtr 3	Qtr 4
Co-58	Ci	--	--	--	8.50E-07
Co-60	Ci	--	--	--	--
Cs-137	Ci	9.34E-07	--	--	--
TOTAL FOR PERIOD	Ci	9.34E-07	--	--	8.50E-07

DOSES DUE TO IODINE,TRITIUM, AND PARTICULATES

UNIT 4
TABLE 5

PATHWAY	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	SKIN	TOTAL BODY
Cow milk - Infant	5.44E-06	7.80E-06	2.50E-03	1.31E-05	1.69E-08	2.04E-06	- -	4.47E-06
Fruit & Veg Fresh	1.42E-06	2.04E-06	6.58E-04	3.46E-06	2.03E-09	5.43E-07	- -	1.16E-06
Ground Plane	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	4.79E-07	5.74E-07	4.79E-07
Inhalation - Adult	9.88E-08	1.42E-07	4.50E-05	2.42E-07	9.29E-08	9.78E-08	9.01E-08	9.52E-08
TOTAL (mrem)	7.43E-06	1.05E-05	3.20E-03	1.73E-05	5.91E-07	3.16E-06	6.64E-07	6.20E-06
% of Annual Limit	4.95E-05	6.98E-05	2.14E-02	1.16E-04	3.94E-06	2.11E-05	4.43E-06	4.13E-05

DOSES DUE TO NOBLE GASES

	mrad	% of Annual Limit
Gamma Air Dose	2.82E-03	2.82E-02
Beta Air Dose	4.48E-03	4.48E-02

DOSES DUE TO IODINE,TRITIUM, AND PARTICULATES

**SUMMATION
TABLE 5**

PATHWAY	BONE	LIVER	THYROID	KIDNEY	LUNG	GI-LLI	SKIN	TOTAL BODY
Cow milk - Infant	5.30E-05	1.09E-04	2.04E-02	4.66E-05	4.45E-05	4.85E-05	--	8.14E-05
Fruit & Veg Fresh	2.84E-06	1.16E-05	1.32E-03	1.44E-05	7.53E-06	8.61E-06	--	9.85E-06
Ground Plane	9.58E-07	9.58E-07	9.58E-07	9.58E-07	9.58E-07	9.58E-07	1.15E-06	9.58E-07
Inhalation - Adult	1.98E-07	2.85E-07	9.01E-05	4.84E-07	1.86E-07	1.96E-07	1.80E-07	1.90E-07
TOTAL (mrem)	5.70E-05	1.22E-04	2.18E-02	6.25E-05	5.32E-05	5.83E-05	1.33E-06	9.24E-05
% of Annual Limit	3.80E-04	8.12E-04	1.46E-01	4.17E-04	3.55E-04	3.89E-04	8.85E-06	6.16E-04

DOSES DUE TO NOBLE GASES

	mrad	% of Annual Limit
Gamma Air Dose	4.52E-03	4.52E-02
Beta Air Dose	1.30E-02	1.30E-01

TURKEY POINT UNITS 3 AND 4
ANNUAL RADIOACTIVE EFFLUENTS RELEASE REPORT
TABLE 6

	UNITS	VALUE

3. SOLID WASTE DISPOSITION

NUMBER OF SHIPMENTS	MODE OF TRANSPORT	DESTINATION
17 (Note 3)	Sole use truck	Oak Ridge, TN
11	Sole use truck	Barnwell, SC

B. IRRADIATED FUEL SHIPMENTS

None

RKR/eb/221

TURKEY POINT UNITS 3 AND 4
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
TABLE 6
SOLID WASTE SUPPLEMENT

Waste Classification	Total Volume Ft^3	(NOTE 4) Total Curie Quantity	(NOTE 5) Principal Radionuclides	(NOTE 6) Type of Waste	R.G. 1.21 Category	(NOTE 7) Type of Container	Solidification or Absorbent Agent
Class A	1866.6	0.518	None	Compactable Waste	1b.	Strong Tight	N/A
Class A	6.8	0.0004	None	Dewatered Resin	1a.	Strong Tight	N/A
Class A	199.4	0.042	None	Sludge	1a.	Strong Tight	Envirostone Gypsym Cement
Class A	547.6	10.8	None	Dewatered Resin, Filters	1a.	>Type A LSA	N/A
Class B	439.8	112.3	Ni-63 Sr-90 Cs-137	Dewatered Resin, Filters	1a.	>Type A, LSA	N/A
Class C	132.4	16.5	C-14 Co-60 Ni-63 Cs-137	Dewatered Filters	1a.	Type B	N/A

RKR/eb/255

TURKEY POINT UNITS 3 AND 4
ANNUAL RADIOACTIVE EFFLUENTS RELEASE REPORT
TABLE 6

SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A. SOLID WASTE SHIPMENT OFFSITE FOR BURIAL OR DISPOSAL

Note 1: Spent resin, filters, sludge, and evaporator bottoms volume indicates volume shipped directly to burial site.

Note 2: Dry compressible waste volume indicates volume shipped to burial site following reduction by a waste processing facility. Volume shipped to the waste processing facility was 1232.4 m^3

Note 3: Material transported to Oak Ridge, Tennessee, was consigned to licensed processing facilities for volume reduction and decontamination activities. The material remaining after processing was transported by the processor to Barnwell, South Carolina, for burial.

Note 4: The total curie quantity and radionuclide composition of solid waste shipped from the Turkey Point Plant Units 3 and 4 are determined using a combination of qualitative and quantitive techniques. The Turkey Point Plant follows the guidelines in the Low Level Waste Licensing Branch Technical Position on Radioactive Waste Classification (5/11/83) for these determinations.

The most frequently used techniques for determining the total activity in a package are the dose to curie method and inference from specific activity and mass or activity concentration and volume. Activation analysis may be applied when it is appropriate. The total activity determination by any of these methods is considered to be an estimate.

TURKEY POINT UNITS 3 AND 4
ANNUAL RADIOACTIVE EFFLUENTS RELEASE REPORT
TABLE 6

The composition of radionuclides in the waste is determined by both on-site analysis for principle gamma emitters and periodic off-site analyses for difficult to measure isotopes. The on-site analyses are performed either on a batch basis or on a routine basis using representative samples appropriate for the waste type. Off-site analyses are used to establish scaling factors or other estimates for difficult to measure isotopes.

Note 5: Principle radionuclide refers to those radionuclides contained in the waste in concentrations greater than 0.01 times the concentration of the nuclide listed in Table 1 or 0.01 times the smallest concentration of the nuclide listed in Table 2 of 10 CFR 61.

Note 6: Type of waste is specified as described in NUREG 0782, Draft Environment Impact Statement on 10 CFR 61 "Licensing Requirements for Land Disposal of Radioactive Waste".

Note 7: Type of container refers to the transport package.

RKR/eb/222

TURKEY POINT UNITS 3 AND 4
ANNUAL RADIOACTIVE EFFLUENTS RELEASE REPORT
ATTACHMENT 1

SUMMARY OF CHANGES TO THE PROCESS CONTROL PROGRAM

Pacific Nuclear, Inc., Waste Services Group Procedure PT-51-WS, Solidification Process Control Procedure, Revision 10, May 21, 1992, was deleted during this reporting period, at the end of a solidification campaign.

RKR/eb/224

Attachment 1

Revision to the ODCM

CONTROLLED DOCUMENT

NO. # 1

OFFSITE DOSE CALCULATION MANUAL

FOR

GASEOUS AND LIQUID EFFLUENTS

FROM THE

TURKEY POINT PLANT UNITS 3 AND 4

REVISION ~~3~~ 4

AMENDMENT 1

CHANGE DATED ~~11/12/92~~

Florida Power and Light Company

MHG # 93-225

PNSC APPROVAL [signature] DATE 1/4/93

PLT. MGR. APPROVAL [signature] DATE 1/4/93

OFFSITE DOSE CALCULATION MANUAL

FOR

GASEOUS AND LIQUID EFFLUENTS

FROM THE

TURKEY POINT PLANT UNITS 3 AND 4

<u>REVISION 4</u>

<u>AMENDMENT 1</u>

<u>CHANGE DATED 01/01/94</u>

Florida Power and Light Company

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OFFSITE DOSE CALCULATION MANUAL
FOR GASEOUS AND LIQUID EFFLUENT

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Appendix

1.0 Introduction

This manual describes methods which are acceptable for calculating radioactivity concentrations in the environment and potential offsite doses associated with liquid and gaseous effluents from the Turkey Point Nuclear Units. These calculations are performed to satisfy Technical Specifications and to ensure that the radioactive dose or dose commitment to any member of the public is not exceeded.

The radioactivity concentration calculations and dose estimates in this manual are used to demonstrate compliance with the Technical Specifications required by 10 CFR 50.36. The methods used are acceptable for demonstrating operational compliance with 10 CFR 20.106, 10CFR50 Appendix I, and 40CFR190. Only the doses attributable to Turkey Point Units 3 and 4 are determined in demonstrating compliance with 40CFR190 since there are no other nuclear facilities within 50 miles of the plant. Monthly calculations are performed to verify that potential offsite releases do not exceed Technical Specifications and to provide guidance for the management of radioactive effluents. The dose receptor is described such that the exposure of any member of the public is not likely to be substantially underestimated.

Quarterly and annual calculations of committed dose are also performed to verify compliance with regulatory limits on offsite dose. For these calculations, the dose receptor is chosen on the basis of applicable exposure pathways identified in a land use survey and the maximum ground level atmospheric dispersion factor (X/Q) at a residence, or on the basis of more conservative conditions such that the dose to any resident near the plant is not likely to be underestimated.

1.1 ODCM Review and Approval

1.1.1 Responsibility for Review

The Chemistry Department Supervisor or his designee shall perform a review of the ODCM annually.

1.1.2 Documentation of Reviews

Following the performance of the annual review required by Section 1.1.1, the individual performing the review shall submit a report for PNSC approval. This report should contain the following information:

1. A copy of the ODCM with any requested changes.

2. Information necessary to support the rationale for the requested changes.

3. A determination that the requested changes will not reduce the accuracy or reliability of dose calculations or setpoint determinations.

4. If no changes are being requested, no actions are required.

1.1.3 Institution of Changes

Changes to the ODCM shall become effective upon review and approval by the PNSC.

1.1.4 Submittal of Changes

Changes to the ODCM and any supporting documentation shall be submitted to the NRC in the Annual Radioactive Effluent Release Report for the period in which the changes were made effective. This submittal, per Technical Specification 6.14.2, shall contain the following information:

1. Sufficiently detailed information to totally support the rationale for the changes(s) without benefit of additional or supplemental information.

2. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered, dated and containing the revision number, together with appropriate analyses or evaluations justifying the change(s)

3. A determination that the change(s) will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

4. Documentation of the fact that the change(s) has been reviewed and found acceptable by the PNSC.

2.0 Liquid Effluents

2.1 Objectives

To provide calculational methodology needed to assure compliance with Technical Specification 3.11.1 which requires the following determinations and surveillances:

- o The concentration of radioactive materials released in liquid effluents.
- o The concentrations of radioactive materials released are maintained within the limits of Specification 3.11.1.1.
- o Quarterly and annual cumulative dose contributions to a member of the public from radioactivity in liquid effluents released from each unit to unrestricted areas are maintained within the limits of Specification 3.11.1.2.
- o Projected doses at least once per 31 days due to liquid releases to unrestricted areas are maintained within the limits of Specification 3.11.1.3.
- o Operation of appropriate portions of the Liquid Radwaste Treatment System if projected doses exceed limits of Specification 3.11.1.3.
- o Verification of operability of Liquid Radwaste System by meeting Specifications 3.11.1.1. and 3.11.1.2.

2.2 Bases

Radioactive liquid effluents from Turkey Point Units 3 and 4 are released through radiation monitors which provide an alarm and automatic termination of radioactive releases. There are three discharge points from the units: steam generator blowdown from each unit and a common radwaste monitor tank discharge.

The liquid effluent monitoring instrumentation and controls at Turkey Point for controlling and monitoring normal radioactive releases in accordance with Turkey Point Technical Specification 3.11.1. consist of the following:

2.2.1 Liquid Radwaste System

Potentially radioactive liquid waste from Units 3 and 4 chemistry laboratories, containment sumps, floor drains, showers and miscellaneous sources are collected in waste hold up tanks. These wastes are processed through a demineralizer system and the effluent stored in one of the three waste monitor tanks (Refer to Figure 2-1). Laundry wastes are normally segregated and sent to one of two monitor tanks. Liquid waste in the waste monitor tanks and monitor tanks are isolated and recirculated for a minimum of one(1) tank volume prior to sampling.

Liquids in these tanks are released after sampling and analysis in accordance with Technical Specification Table 4.11-1. The discharge from the waste monitor and monitor tanks is monitored by a radioactive liquid effluent monitor. Since these liquid effluents are a mixture from both Units 3 and 4, the measured releases from the common discharge point are apportioned to each unit as a ratio equal to the ratio of specific isotopic concentrations in the primary coolant of the two reactors to assure the effluents are within the allowable limits per reactor. An alternate method is to allocate effluent releases equally to both Units 3 and 4.

2.2.2 Steam Generator Blowdown

Units 3 and 4 steam generator blowdown can be discharged directly from the blowdown flashtanks to the condenser cooling water mixing basin. The activity of each steam generator blowdown discharge (a composite) is monitored prior to the Blowdown Flash Tank for Unit 3 and 4 respectively. Releases from the steam generator blowdown are sampled and analyzed in conformance with Technical Specification Table 4.11-1.

2.2.3 Storm Drains

Storm drains from Units 3 and 4 discharge into both the circulating water intake and the condenser cooling water mixing basin. Storm drains are sampled and analyzed in accordance with Technical Specification Table 4.11-1.

2.2.4 Radioactivity Concentration in Liquid Waste

The concentration of radionuclides in liquid waste is determined by sampling and analysis in accordance with Table 4.11-1 of the Technical Specifications. If a radionuclide is below its LLD, and the calculated LLD concentration is below the LLD concentration value specified in Technical Specification, Table 4.11-1 then it is not reported as being present in the sample. When the radionuclide's calculated LLD is greater than the LLD listed in Technical Specification Table 4.11-1, the calculated LLD should be assigned as the activity of the radionuclide.

2.2.5 Radioactivity Concentration in Water at the Restricted Area Boundary

Technical Specification 3.11.1.1 requires that the concentration of radioactive material, other than noble gases, in liquid effluent released into an unrestricted area not exceed 10 times the effluent concentration specified in 10 CFR Part 20, Appendix B, Table 2, Column 2. A maximum concentration, 2 x $10^{-4}\mu$Ci/ml, for noble gas entrained in aqueous releases into an unrestricted area applies separately since the potential exposure route, immersion in water, differs from that upon which Part 20, Appendix B is based.

Radioactive material in liquid effluent from Turkey Point is diluted by condenser cooling water from fossil units 1 and 2 and from nuclear units 3 and 4 in the condenser cooling water mixing basin. Water in the basin flows into an onsite closed cooling canal system. Liquid effluent does not actually leave the site in a surface discharge. For the purpose of compliance with Technical Specification 3.11.1.1, the total condenser cooling water flow from operating condenser cooling water pumps at the four units is assumed for dilution and the restricted area boundary is assumed to be at the end of the condenser cooling water mixing basin where water enters the cooling canal system.

Sections 2.3.1 and 2.3.2 describe methods used to assess compliance with Technical Specification 3.11.1.1. Effluent monitor alarm/trip setpoints are computed on the same basis, as described in section 2.6. If an alarm/trip setpoint is not exceeded, aqueous effluents are deemed to comply with Technical Specification 3.11.1.1.

2.3 Aqueous Concentration

The diluted concentration of radionuclides in the condenser cooling water mixing basin outflow is estimated with the equation

$$C_{zi} = C_i \cdot \frac{F_1}{F_2} \tag{1}$$

where:

C_{zi} = concentration of radionuclide i in the water in the condenser cooling water mixing basin outflow, (μCi/ml)

C_i = concentration of radionuclide i in liquid radwaste released, (μCi/ml)

F_1/F_2 = dilution

F_1 = flow in radioactive liquid discharge line (gal/min).*

F_2 = total condenser cooling water flow, (gal/min).* Value not greater than the rated total condenser cooling water flow from operating condenser cooling water pumps at the four units.

*F_1 and F_2 may have any suitable but identical units of flow, (volume/time).

2.3.1 Batch Release

A sample of each batch of liquid radwaste is analyzed before release for I-131 and other principal gamma emitters. With the activity concentration in a batch sample b based on the total isotopic activity, the fraction of the unrestricted area EC due to a batch release is derived by using the ratio of the individual isotopic concentrations and their related ECs. FEC_b is estimated with the equation

$$FEC_b = \frac{\sum_i \frac{C_{zi}}{EC_i}}{E_b} \qquad (2)$$

where:

FEC_b = fraction of the unrestricted area EC present in the condenser cooling water mixing basin outflow due to a batch release

C_{zi} = concentration of radionuclide i in the water in the condenser cooling water mixing basin out flow, (μCi/ml); determined from equation (1).

EC_i = Ten times the activity concentration limit in water of radionuclide i according to 10 CFR 20, Appendix B, Table 2, Column 2, (μCi/ml).

$$E_b = \frac{\text{(Quarterly average of the fraction of EC in the batch tank due to I-131 and principal gamma emitters)}}{\text{(Quarterly average of the fraction of EC in the batch tank due to all radionuclides measured.)}}$$

E_b is an adjustment to account for radionuclides not measured prior to release but measured in the quarterly sample per Technical Specification Table 4.11-1, i.e., Sr-89, Sr-90, Fe-55. The value of E_b is calculated from previously measured data (a conservative value of 0.5 has been estimated for E_b if a calculated value is not available), or FEC_b can be calculated by including a previous quarter's beta C_{zi} and EC_i into the calculation for each release, thus eliminating the E_b factor.

Alternately, the fraction of the unrestricted area EC due to a batch release can be estimated by:

$$FEC_b = \frac{C_b}{1\times10^{-7}} \qquad (3)$$

where:

$$C_b = \sum_i C_{zi}(\mu Ci/ml)$$

1×10^{-7} = unrestricted area EC for unidentified radionuclides in water, (μCi/ml).

2.3.2. Continuous Release

Continuous aqueous discharges are sampled and analyzed according to the schedule in Technical Specification Table 4.11-1. The fraction of the unrestricted area EC present in a continuously discharged radioactive stream, FEC_c, is derived from an isotopic analyses. The fraction of the unrestricted area EC can be derived using the ratio of the individual isotopic concentrations and their related ECs.

FEC_c is estimated with the equation

$$FEC_c = \frac{\sum_i \frac{C_{zi}}{EC_i}}{E_c} \qquad (4)$$

where:

FEC_c = fraction of the unrestricted area EC present in the condenser cooling water mixing basin outflow due to a continuous release

C_{zi} = concentration of radionuclide i in the water in the condenser cooling water mixing basin outflow determined from equation (1); (μCi/ml)

EC_i = Ten times the activity concentration limit in water of radionuclide i according to 10CFR20, Appendix B, Table 2, Column 2, (μCi/ml)

E_c = $\dfrac{\text{(Quarterly average fraction of EC due to I-131 and principal gamma emitters measured in samples of continuous releases during the quarter)}}{\text{(Quarterly average fraction of EC due to all radionuclides measured in samples of continuous releases)}}$

E_c is an adjustment to account for radionuclides not measured in individual samples of continuous releases but measured in the quarterly composite samples per Technical Specifications Table 4.11-1,i.e. Sr-89, Sr-90, Fe-55. The value of E_c is calculated from previously measured data (a conservative value of 0.5 has been estimated for E_c if a calculated value is not available), or FEC_c can be calculated by including a previous quarter's beta C_{zi} and EC_i into the calculation for each release, thus eliminating the E_c factor.

Alternately, the fraction of the unrestricted area EC present in the condenser cooling water mixing basin can be estimated by

$$FEC_c = \frac{C_c}{1x10^{-7}} \qquad (5)$$

Where:

$$C_c = \sum_i C_{zi} (\mu Ci/ml)$$

1×10^{-7} = unrestricted area EC for unidentified radionuclides in water, (μCi/ml)

2.3.3 Cumulative Release

To ensure that the unrestricted area EC is not exceeded during periods of multiple releases, the fraction of EC determined for each type of release is summed to determine a total release fraction using the following equation:

$$FEC_T = FEC_b + FEC_c \qquad (6)$$

Where:

FEC_T = the total fraction of the unrestricted area EC released.

FEC_b = the fraction of the unrestricted area EC due to batch releases.e.g.monitor tanks, storm drains etc.

FEC_c = the fraction of the unrestricted area EC due to continuous releases. e.g. steam generator blowdown.

2.4 Cumulative Dose

Technical Specification 3.11.1.2 requires the dose or dose commitment to a member of the public from radioactive materials released in liquid effluents from each unit to unrestricted areas be limited to ≤ 1.5 mrem to the whole body and ≤ 5 mrem to any organ during any calendar quarter and to ≤ 3 mrem to the whole body and ≤ 10 mrem to any organ during any calendar year.

Technical Specification 4.11.1.2 requires the dose or dose commitment to a member of the public due to radioactive material released in liquid effluent to be calculated on a cumulative quarterly and annual basis at least once per 31 days. The condenser cooling water basin and closed canal system which receives aqueous effluent is entirely on FP&L property, without surface discharge offsite, and FP&L does not

permit members of the public to use the water. As a result, potential exposure of a member of the public to radioactive material originating in aqueous effluent is limited to irradiation of persons by canal shoreline deposits.

Technical Specification 4.11.1.2 is satisfied by calculating the cumulative total body dose to a person who may be irradiated by radionuclides deposited on the cooling canal shoreline from radioactive liquid effluent. Compliance with the organ dose limit is assured as long as the total body dose is below its limit.

The model that is used to evaluate doses due to radioactivity in liquid effluents is

$$D = 0.23 \sum_{k} \sum_{i} A_i^{shoreline} \cdot \frac{C_{ik} \cdot F_{lk} \cdot t_k}{V \cdot \lambda_i^e} \tag{7}$$

where:

D = total body or organ dose due to irradiation by radionuclides on the shorelines which originated in a liquid effluent release, (mrem)

0.23 = units conversion constant =

$$\frac{1Ci}{10^6 \mu Ci} \times 60 \frac{min}{hr} \times 3785 \frac{ml}{gal}$$

A_i = transfer factor relating a unit aqueous concentration of radionuclide i to a dose commitment rate to specific organs and the total body of an exposed person. Values for A_i are tabulated in Appendix A, (mrem/Ci•gal/min)

C_{ik} = the concentration of radionuclide in the undiluted liquid waste to be discharged that is represented by sample k, (μCi/ml)

F_{lk} = liquid waste discharge flow during release represented by sample k, (gal/min)

V = cooling canal effective volume, approximately 3.75 X 10^9 gallons

λ_i^e = effective decay constant ($\lambda_i + F_3/V$, min^{-1}).

where:

λ_i = the radioactive decay constant

F_3 = canal-ground water interchange flow, approximately 2.25×10^5 gal/min

t_k = period of time (hours) during which liquid waste represented by sample k is discharged

Radionuclide concentrations (C_{ik}) in effluent are measured by the sampling and analysis program specified in Technical Specification Table 4.11-1. Typically, more than 90 percent of the potential irradiation from radionuclides deposited along the shoreline is due to Mn-54, Co-58, Co-60, Cs-134, and Cs-137. Of these radionuclides, Co-60 has the maximum dose transfer factor, A_i. Thus, for the purpose of assessing compliance with Technical Specification, 4.11.1.2, the radioactive effluent source term may be either:

a) principal gamma emitters measured by the effluent sampling and analysis program, or
b) Mn-54, Co-58, Co-60, Cs-134, and Cs-137 measured by the effluent sampling and analysis program and other identified gamma emitters assumed to be Co-60, or
c) all gamma emitters measured by the effluent sampling and analysis program assumed to be Co-60.

Use of principal gamma emitters measured by the effluent sampling and analysis program is preferred over the other alternates.

2.5 Projected Dose

Technical Specification 3.11.1.3 requires that the Liquid Radwaste Treatment System be operable and appropriate subsystems of the liquid radwaste treatment system shall be used to reduce the radioactive materials in liquid waste prior to their discharge when the projected doses from each unit to unrestricted areas due to liquid effluents, when averaged over a 31 day period, would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

Technical Specification 4.11.1.3.1 requires the doses, to unrestricted areas, due to radioactive material released in liquid effluent to be projected at least once per 31 days unless the liquid radwaste treatment system is being fully utilized.

This requirement is satisfied by extrapolating the dose to date during the current month to include the entire month. The dose to date is calculated as described in section 2.4.

The dose is projected with the relation:

$$P = \frac{31 \cdot D}{X} \qquad (8)$$

where:

P = the projected total body or organ dose during the month, (mrem).

31 = number of days in a calendar month, (days)

X = number of days in current month to date represented by available radioactive effluent sample, (days)

D = total body or organ dose to date during current month calculated according to section 2.4, (mrem)

Alternately, the monthly dose may be projected by computing the doses to the total body and most exposed organ accumulated during the most recent month and assuming the result represents the projected doses for the current month. The dose during the preceding month will be computed as described in section 2.4.

2.6 Method of Establishing Alarm and Trip Setpoints

The radioactive liquid effluent monitoring instrumentation should be operable in accordance with Specification 3.3.3.5, with its alarm/trip setpoints set to ensure the limit of Specification 3.11.1.1 are not exceeded.

The alarm/trip setpoint for each liquid effluent radiation monitor is derived from 10 times the effluent concentration limits provided in 10 CFR Part 20, Appendix B, Table 2, Column 2 applied in the condenser cooling water mixing basin outflow. Radiation monitoring and isolation points are located in the steam generator blowdown lines, R-3-19, R-4-19, and the liquid waste disposal system line, R-18, through which radioactive waste effluent is eventually discharged into the canal basin. See Figure 2-1.

The alarm setpoint for each liquid effluent monitor is based upon the measurements of radioactivity in a batch of liquid to be released or in the continuous aqueous discharge. Sample measurements are performed according to Technical Specificat ion Table 4.11-1. If the calculated setpoint is less than the existing setpoint, the setpoint shall be reduced to the new setpoint. If the calculated setpoint is greater than the existing setpoint, the setpoint may remain at the lower value or be increased to the calculated value.

2.6.1. Setpoint for a Batch Release

The liquid radwaste effluent line radiation monitor alarm setpoint for a batch release is determined with the equation below or a method which gives a lower setpoint value.

$$S_b = \frac{A_b \; S_f}{FEC_b} \cdot g_b + Bkg \qquad (9)$$

where:

S_b = radiation monitor alarm setpoint for a batch release, (cpm)

A_b = laboratory counting rate (cpm/ml) or activity concentration (μCi/ml) of sample from batch tank

FEC_b = fraction of unrestricted area EC present in the condenser cooling water mixing basin outflow due to a batch release; determined in section 2.3.1.

g_b = detection efficiency of monitor detector; ratio of effluent radiation monitor counting rate to laboratory counting rate or activity concentration in a given batch sample (cpm/cpm/ml or cpm/μCi/ml) which ever units are consistent with the units A_b.

Bkg = background (cpm)

S_f = A factor to allow for multiple sources from different or common release points. The allowable operating setpoints will be controlled administratively by assigning a fraction of the total allowable release to each of the release sources.

2.6.2 Setpoint for a Continuous Release

The liquid effluent line radiation monitor alarm setpoint for a continuous release is determined with the equation below or by a method which gives a lower setpoint value.

$$S_c = \frac{A_c\ S_f}{FEC_c} \cdot g_c + Bkg \qquad (10)$$

where:

S_c = radiation monitor alarm setpoint for a continuous release, (cpm)

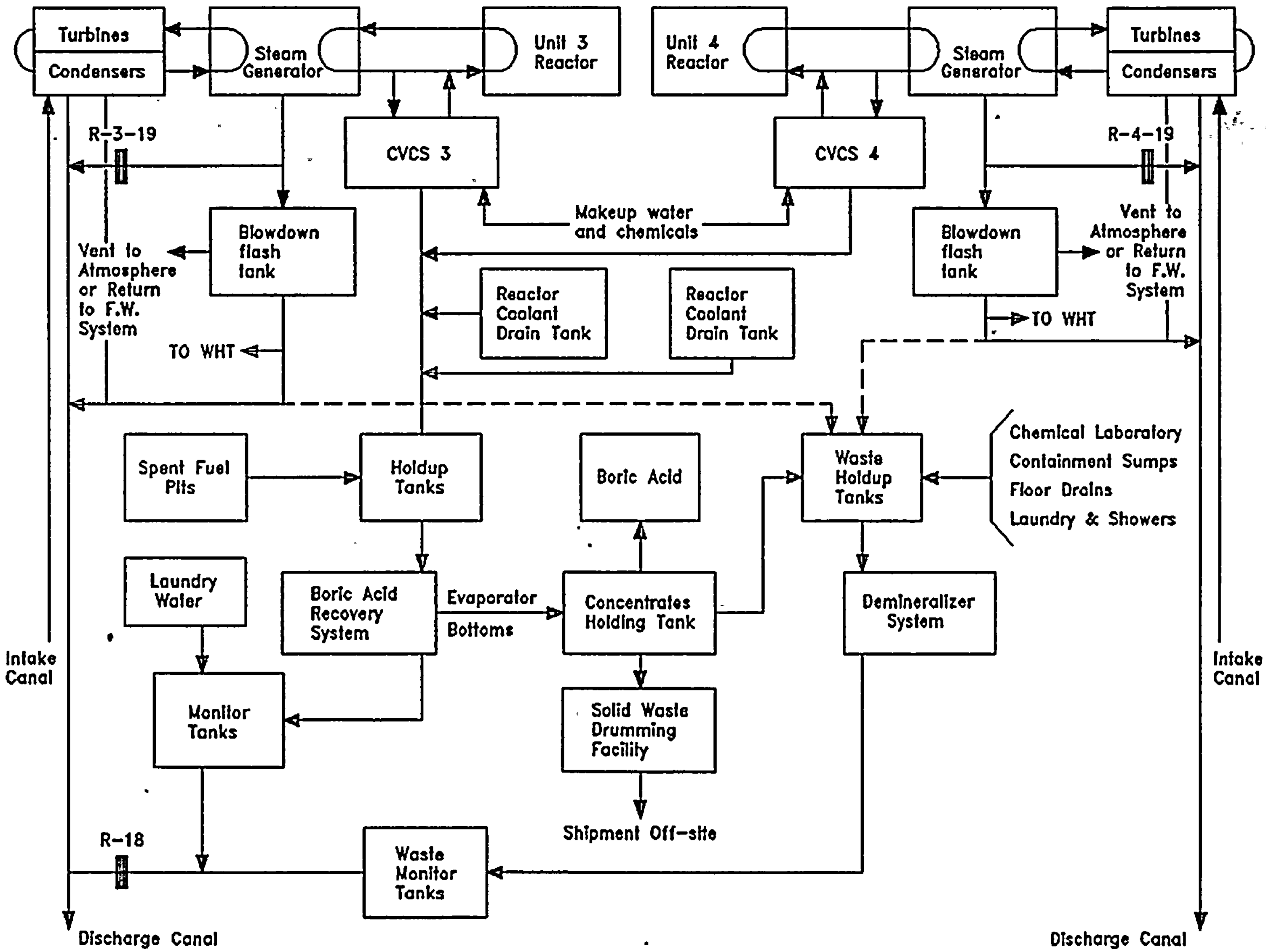
A_c = laboratory counting rate (cpm/ml) or activity concentration (μCi/ml) of sample from continuous release

FEC_c = fraction of unrestricted area EC present in the condenser cooling water mixing basin outflow due to a continuous release; determined in section 2.3.2.

g_c = detection efficiency of monitor detector; ratio of effluent radiation monitor counting rate to laboratory counting rate or activity concentration in a given continuous release sample, (cpm/cpm/ml or cpm/μCi/ml), whichever units are consistent with the units A_c.

S_f = A factor to allow for multiple sources from different or common release points. The allowable operating setpoints will be controlled administratively by assigning a fraction of the total allowable release to each of the release sources.

FIGURE 2-1



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3.0 Gaseous Effluent

3.1 Objectives

To provide calculational methodology needed to assure compliance with Technical Specification 3.11.2 which requires the following determinations and surveillances:

- o Radionuclide concentrations in gaseous effluents
- o The dose rate due to radioactive gaseous effluents to areas at and beyond the site boundary are maintained within the limits of Technical Specification 3.11.2.1.
  - Total body dose rate from radioactive noble gases
  - Skin dose rate from radioactive noble gases
  - Organ dose rate from radioiodines, tritium, and particulates with half-lives greater than 8 days.
- o Determine the cumulative quarterly and annual doses per reactor at and beyond the site boundary due to noble gases are maintained below the limits of Technical Specification 3.11.2.2 at least once per 31 days.
- o Determine that the cumulative quarterly and annual doses per reactor at and beyond the site boundary from radioiodines, tritium, and particulates with half-lives greater than 8 days are maintained below the limits of Technical Specification 3.11.2.3 at least once per 31 days.
- o Project the doses due to gaseous releases from each unit at least once per 31 days when gaseous radwaste treatment systems are not being fully utilized.

3.2 Bases

Radioactive gaseous effluents from Turkey Point Units 3 and 4 are released through four monitored release points; a common plant vent via a stack above the containment building, the Unit 3 spent fuel pit vent, and the condenser air ejector vents from each unit. Unmonitored radioactive airborne releases can also occur from the secondary steam systems of each unit if primary to secondary leakage is occurring. The effluent sources (refer to Figure 3-1) for each release point are tabulated in Table 3-1. The airborne releases from all these sources are treated as a mixed mode release from a single location for dose calculational purposes.

Compliance for beta and gamma dose limits at and beyond the site boundary for noble gas effluents is determined by assessing the dose rate and/or dose at the location where the minimum atmospheric dispersion occurs at the site boundary since the atmospheric dispersion will be higher at all other points off-site. This minimum dispersion occurs at the site boundary 1950 meters SSE of the plant where the dispersion factor is 5.8×10^{-7} sec/m^3.

The dose rate due to tritium, I-131, I-133, and radioactive particulates with half lives greater than 8 days at and beyond the site boundary is assessed by determining the dose rate to a hypothetical infant's thyroid via the inhalation pathway. The basis for this approach is NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants" which states: the dose factors are dependent on the specific organ and on the age group. The infant is the most restrictive age group for the dose rate calculations and the most restrictive organ is the thyroid via either the inhalation or grass-cow-milk pathway. The dose from tritium, I-131, I-133, and particulate is calculated by assuming a cow on pasture 4.5 miles west of the plant unless there is a milk producer in a more conservative location. At that location the reference atmospheric deposition factor, D/Q, is equal to 5×10^{-10} m^{-2}.

Sampling and analysis is performed as outlined in Technical Specification Table 4.11-2. Principle gamma emitters for batch gaseous effluents which are released via pathways (i.e. Plant Vent) with continuous radioiodine and particulate radionuclide sample trains are considered to be the Noble Gases.

3.2.1 Gaseous Radwaste System

Radioactive and potentially radioactive gases from units 3 and 4 containment buildings, the auxiliary building, unit 4 spent fuel pit, radwaste building and laundry area are released via the monitored plant vent after passing through filter systems. Radioactive waste gases from the primary systems (CVCS hold-up tanks) are stored in gas decay tanks to reduce activity levels by radioactive decay prior to release via the plant vent. The unit 3 spent fuel pit area is ventilated via its' own monitored vent after passing through a filtering system.

The steam jet air ejectors from each unit are vented through monitored release pathways. Other steam losses concurrent with primary to secondary leakage are unmonitored and gaseous activity must be accounted for.

3.2.2 Radioactivity in Gaseous Effluent

Radionuclides other than noble gases in the gaseous effluents are measured by the radioactive gaseous waste sampling and analysis program described in Technical Specification Table 4.11-2. Noble gas radionuclides are measured by continuous monitors in the four release points. The gaseous effluent streams monitoring points, and effluent discharge points are illustrated schematically in Figure 3-1.

The measured radionuclide concentrations in gaseous effluents from the plant are used for estimating off-site radionuclide concentrations and radiation doses. Sampling and analyses are performed consistent with the requirements of Technical Specification Table 4.11-2.

The radioactive iodines and particulate radionuclides from continuous releases and batch releases (Containment Purges and Gas Decay Tanks are released via the Plant Vent) are determined by charcoal and filter samples removed weekly from continuous sample trains installed at each release point (plant vent, condenser air ejectors and Unit 3 Spent Fuel Pit vent). Tritium activity is determined on monthly grab samples from the plant vent, condenser air ejector, and Unit 3 Spent Fuel Pit and by a grab sample from each containment purge.

Additional grab samples are obtained and analyzed if the conditions identified in Notes 4,5,6 and 7 of Technical Specification Table 4.11-2 exist, i.e., tritium grab samples once per 24 hours when the refueling canal is flooded, tritium grab samples at least weekly from the spent fuel pool ventilation exhaust when spent fuel is in the spent fuel pool, and sampling shall also be performed at least once per day for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in one (1) hour and analyses shall be completed within 48 hours of changing if:

(1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased by more than a factor of 3; and
(2) the noble gas activity monitor shows that the effluent activity has increased by more than a factor of 3.

Activities measured by these additional samples should be included in the cumulative dose calculations.

Noble gas activity released is measured by continuous noble gas monitors installed in each discharge point for release types listed in Technical Specification Table 4.11-2. The quantity of radioactive noble gas activity not accounted for by grab samples can be determined by integrating the release rate measurement from each effluent noble gas monitor. The total measured radioactivity discharged via a stack or vent during a specific time period can be determined from the effluent monitors by:

$$Q_j = \frac{N_j \cdot F}{3.53 \times 10^{-5} \cdot h} \tag{11}$$

where:

Q_j = total measured gaseous radioactivity release via a stack or vent during counting interval j, (μCi)

N_j = counts accumulated during counting interval j, (counts = N(cpm) x t (min))

F = discharge rate of gaseous effluent stream, (ft^3/min)

3.53×10^{-5} = conversion constant, (ft^3/cm^3)

h = effluent noble gas monitor calibration or counting rate response for noble gas gamma radiation, $\left(\frac{cpm}{\mu Ci/cm^3}\right)$

The distribution of radioactive noble gases in a gaseous effluent stream is determined by gamma spectrum analysis of gas samples from that stream. Results of previous analyses may be averaged to obtain a representative distribution.

If f_i represents the fraction of radionuclide i in a given effluent stream, based on the isotopic distribution of that stream, then the quantity of radionuclide i released in a given gaseous effluent stream during counting interval j is:

$$Q_{ij} = Q_j \cdot f_i \tag{12}$$

where:

Q_{ij} = quantity of radionuclide i released in a given gaseous effluent stream during counting interval j, (μCi)

f_i = the fraction of radionuclide i released in a given effluent stream

In the event the radioactive noble gas distribution is not obtainable from sample(s) taken during the current period the distribution will be obtained from recent data if available or from Table 3-2.

Some gaseous effluents from both Units 3 and 4, whose sources are identified in Table 3-1, discharge in common through the plant vent. To assure that the effluents are within allowable limits per reactor, the measured release from the plant vent is apportioned to each unit on a ratio equal to the ratio of specific isotopic concentrations in the primary coolant in the two reactors. An alternate method is to allocate effluent releases equally to both Units 3 and 4. Iodine and particulate release contributions will also be adjusted to account for specific containment purge releases.

3.3 Dose Rate Due to Gaseous Effluent

Technical Specification 3.11.2.1 provides that the dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary shall be limited to the following: ≤500 mrem/year to the total body and ≤3000 mrem/year to the skin due to noble gases and ≤1500 mrem/year to any organ due to I-131, I-133, tritium and all radioactive materials in particulate form with half-lives greater than 8 days.

Compliance with the limits on dose rate from noble gases is demonstrated by establishing effluent monitor alarm setpoints such that an alarm will occur at or before a dose rate limit of the combined releases for noble gases is reached for the release types listed in Technical Specification Table 4.11-2. If an alarm occurs when the monitor setpoint is at or below the limit, compliance may be assessed by comparing the monitor record with the setpoint (limit) calculated in accordance with Section 3.6 or a more conservative method. In the event an alarm occurs and the monitored release exceeds the setpoint limit, then compliance shall be evaluated by calculating dose rates in accordance with Sections 3.3.1 and 3.3.2.

The alarm setpoints shall be derived on the basis of the radionuclide distribution from a measured gamma spectrum, a historical gamma spectrum dominated by Xe-133 or by assuming the total noble gas activity is Xe-133. If Xe-133 is the dominant radioactive gas in the airborne effluent, the gamma dose rate to a person's body is expected to be a larger fraction of the 500 mrem/year limit than is the sum of beta and gamma dose rates to the skin limit of 3000 mrem/year. Thus, a gaseous effluent monitor setpoint may be derived on the basis of whole body gamma dose rate alone such that an alarm occurs at or before the whole body dose rate off-site exceeds 500 mrem/year as given in Technical Specification 3.11.2.1.

3.3.1 Total Body Dose Rate

The total body dose rate from radioactive noble gases may be calculated at any location off-site by assuming a person is immersed in and irradiated by a semi-infinite cloud of the noble gases. The dose rate is calculated using the equation

$$\dot{D}_{TB} = \frac{X}{Q} \cdot \frac{1}{t} \sum_i Q_i \cdot P_{\gamma i} \qquad (13)$$

where:

$\dot{D}_{TB}$ = Dose rate to total body from noble gases, (mrem/year)

$\frac{X}{Q}$ = atmospheric dispersion factor at the off-site location of interest, (sec/m^3)

t = Averaging time of release, i.e., increment of time during which Q_i was released, (year)

Q_i = quantity of noble gas radionuclide i released during the averaging time, (μCi)

$P_{\gamma i}$ = factor converting time integrated concentration of noble gas radionuclide i at ground-level to total body dose, $\left(\frac{mrem}{\mu Ci \cdot sec/m^3}\right)$; see Reference Table 3-4.

Since dose rate limits for airborne effluents apply everywhere off-site, compliance is assessed and alarm setpoints determined at the site boundary where the minimum atmospheric dispersion from the plant (maximum X/Q) occurs. Ordinarily, that location is selected on the basis of reference meteorology data in Table 3-6. According to those data, the minimum dispersion off-site occurs at the site boundary 1950 meters SSE of the plant where $X/Q = 5.8 \times 10^{-7}$ sec/m^3. Alternately, averaged meteorology data coincident with the period of release being evaluated may be used.

3.3.2 Skin Dose Rate

The dose rate to skin from radioactive noble gases may be calculated at any location off-site by assuming a person is immersed in and irradiated by a semi-infinite cloud of the noble gases. The dose rate to skin is calculated using the equation

$$\dot{D}_s=\frac{\chi}{Q}\cdot\frac{1}{t}\left[\sum_i Q_i\cdot S\beta_i+1.11\sum_i Q_i\cdot A_{\gamma i}\right] \tag{14}$$

where:

$\dot{D}_s$ = dose rate to skin from radioactive noble gases, (mrem/year)

$S\beta_i$ = factor converting time integrated concentration of noble gas radionuclide i at ground level, to skin dose from beta radiation, $\left(\frac{\text{mrem}}{\mu\text{Ci}\cdot\text{sec/m}^3}\right)$ Reference Table 3-4;

1.11 = ratio of tissue dose equivalent to air dose in a radiation field, (mrem/mrad)

$A_{\gamma i}$ = factor for converting time integrated concentration of noble gas radionuclide i in a semi-infinite cloud, to air dose from its gamma radiation, $\left(\frac{\text{mrad}}{\mu\text{Ci}\cdot\text{sec/m}^3}\right)$ listed in Table 3-3

Since dose rate limits for airborne effluents apply everywhere off-site, compliance is assessed and alarm setpoints determined at the site boundary where the minimum atmospheric dispersion from the plant (maximum χ/Q) occurs. Ordinarily, that location is selected on the basis of reference meteorology data in Table 3-6. According to those data, the minimum dispersion off-site occurs at the site boundary 1950 meters SSE of the plant where $\chi/Q = 5.8 \times 10^{-7}$ sec/m^3. Alternately, averaged meteorology data coincident with the period of release being evaluated may be used.

3.3.3 H-3, I-131, I-133 and Particulate Dose Rate

The dose rate to any organ due to H-3, I-131, I-133 and radioactive material in particulate form with a half life of more than 8 days is calculated with the equation.

$$D_{anp}=\frac{1}{3600t}\cdot\frac{\chi_d}{Q}\sum_k\sum_i Q'_{ik}\cdot TA_{anip} \qquad (15)$$

where:

D_{anp} = dose equivalent rate to body organ n of a person in age group a exposed via pathway p to radionuclides i identified in all analysis k of effluent air, (mrem/year)

3600 = conversion constant, (sec/hr)

t = period of time over which the effluent releases are averaged, (hr)

χ_d/Q = atmospheric dispersion factor, adjusted for depletion by deposition(sec/m^3).(Alternately χ/Q, unadjusted, may be used).

Q_{ik} = quantity of radionuclide i released during time increment t based on analysis k, (μCi).

TA_{anip} = a factor relating the airborne concentration time integral of radionuclide i to the dose equivalent to organ n of a person in age group a exposed via pathway p (inhalation),

$\left(\dfrac{mrem/yr}{\mu Ci/m^3}\right)$; See Appendix A.

When the dose rate due to H-3, I-131, I-133 and radionuclides in particulate form is calculated for the purpose of assessing compliance with Specification 3.11.2.1, a hypothetical infant located where the minimum atmospheric dispersion from the plant occurs is assumed as the receptor.

For the radioiodines and particulates with half-lives greater than eight days, the effective dose transfer factor, TA_{anip}, is based solely on the radioiodines (I-131, I-133). This approach was selected because the radioiodines contribute essentially all of the dose to the infant's thyroid via the inhalation and the grass-cow-milk pathway. The infant's thyroid via the inhalation pathway is the critical organ and controlling pathway respectively for the releases of radioiodines and particulates.

Ordinarily, the dose rate calculation will be based on the location of minimum dispersion adjusted for deposition according to the reference meteorology data in Table 3-7. According to those data, the minimum dispersion offsite occurs at the site boundary 1950 meters SSE of the plant and the χ_d/Q value is 5.0×10^{-7} sec/m^3. That location is identified in Figure 3-2. Alternately, averaged meteorological dispersion data coincident with the period of release may be used to evaluate the dose rate. These radionuclide concentrations in airborne effluents, Q_{ik}, are measured according to the sample and analysis schedule in Technical Specification Table 4.11-2.

3.4 Dose-Noble Gases

Technical Specification 3.11.2.2 requires that the air dose per reactor at and beyond the site boundary due to noble gases released in gaseous effluents shall be limited during any calendar quarter, to ≤5 mrad for gamma radiation and ≤10 mrad for beta radiation and during any calendar year, to ≤10 mrad for gamma radiation and ≤20 mrad for beta radiation.

3.4.1 Noble Gas Gamma Radiation Dose

Specification 4.11.2.2 requires the cumulative dose contributions be determined at least once per 31 days to verify that the accumulated air dose due to gamma radiation does not exceed the limits for the current quarter and year.

The gamma radiation dose to air offsite as a consequence of noble gas discharged from each unit can be calculated with the equation

$$D_\gamma = \frac{1}{0.8} \cdot \frac{\chi}{Q} \cdot A_{\gamma eff} \cdot \sum_j Q_j \tag{16}$$

where:

D_γ = noble gas gamma dose to air due to a mixed mode release, (mrad)

0.8 = a conservatism factor which, in effect, increases the estimated dose to compsate for variability in radionuclide distribution

χ/Q = atmospheric dispersion factor at the offsite location of interest, (sec/m^3)

$A_{\gamma eff}$ = effective gamma air dose factor converting time-integrated, ground-level, total activity concentration of radioactive noble gas, to air dose due to gamma radiation. This factor has been derived from noble gas radionuclide distributions in routine operational releases. Refer to Appendix B for a detailed explanation. The effective gamma air dose factor is:

$$A_{\gamma eff} = 1.4 \times 10^{-5} \quad \left(\frac{\text{mrad}}{\mu\text{Ci} \bullet \text{sec/m}^3}\right)$$

Q_j = the measured gaseous radioactivity released via a stack or vent during a single counting interval j, (μCi)

Specification 4.11.2.2 is satisfied by calculating the noble gas gamma radiation dose to air at the location identified in Figure 3-2. At that location, 1950 meters SSE of the Plant,the reference atmospheric dispersion factor to be used is $\chi/Q = 5.8 \times 10^{-7}$ sec/m^3.

Alternately,Specification 4.11.2.2 may be satisfied by calculating the gamma dose to air with the equation

$$D_\gamma = \frac{\chi}{Q} \sum_j \sum_i Q_j \cdot f_i \cdot A_{\gamma i} \tag{17}$$

where:

f_i = the fraction of radionuclide i released in a given effluent stream

$A_{\gamma i}$= factor converting time integrated, ground level concentration of noble gas radionuclide i to air dose from gamma radiation listed in Table 3-3, $\left(\frac{\text{mrad}}{\mu\text{Ci} \bullet \text{sec/m}^3}\right)$

3.4.2 Noble Gas Beta Radiation Dose

Technical Specification 4.11.2.2 requires an evaluation be performed once per 31 days to verify that the accumulated air dose due to beta radiation does not exceed the limits as given in 3.4 above.

The beta radiation dose to air offsite as a consequence of noble gas discharged from each unit can be calculated with the equation:

$$D_\beta = \frac{1}{0.8} \cdot \frac{\chi}{Q} \cdot A_{\beta eff} \cdot \sum_j Q_j \tag{18}$$

where:

D_β = noble gas beta dose to air due to a mixed mode release, (mrad)

0.8 = a conservatism factor which, in effect, increases the estimated dose to compensate for variability in radionuclide distribution

$A_{\beta eff}$ = effective beta air dose factor converting time-integrated, ground-level, total activity concentration of radioactive noble gas to air dose due to beta radiation. This factor has been derived from noble gas radionuclide distributions in routine operational releases. Refer to Appendix B for a detailed explanation. The effective beta air dose factor is:

$$A_{\beta eff} = 3.4 \times 10^{-5} \quad \left(\frac{\text{mrad}}{\mu\text{Ci} \cdot \text{sec/m}^3}\right)$$

Specification 4.11.2.2 is satisfied by calculating the noble gas beta radiation dose to air at the location identified in Figure 3-2. At that location, 1950 meters SSE of the Plant, the reference atmospheric dispersion factor to be used is $\chi/Q = 5.8 \times 10^{-7}$ sec/m^3.

Alternately, Specification 4.11.2.2 may be satisfied by calculating the beta radiation dose to air with the equation

$$D_\beta = \frac{\chi}{Q}\sum_j \sum_i Q_j \cdot f_i \cdot A_{\beta i} \tag{19}$$

where:

$A_{\beta i}$ = factor converting time-integrated, ground level concentration of noble gas radionuclide i to air dose from beta radiation, listed in Table 3-3:

$$\left(\frac{\text{mrad}}{\mu\text{Ci} \cdot \text{sec/m}^3}\right)$$

3.5 Dose Due to Iodine, Tritium, and Particulates in Gaseous Effluents

Technical Specification 3.11.2.3 requires the dose per reactor to a member of the public due to I-131, I-133, tritium, and particulates with half-lives greater than 8 days in airborne effluents released to areas at or beyond the site boundary shall not exceed 7.5 mrem to any organ during any calendar quarter and shall not exceed 15 mrem to any organ during any calendar year.

3.5.1 Determining the Quantity of Iodine, Tritium, and Particulates

Radionuclides, other than noble gases, in gaseous effluents that are measured by the radioactive gaseous waste sampling and analysis program described in Technical Specification Table 4.11-2 are used as the release term in dose calculations. Airborne releases are discharged either via a stack above the top of the containment building or via other vents and are treated as a mixed mode release from a single location. Releases of steam from the secondary system concurrent with primary to secondary leakage will also result in the release of activity to the atmosphere. For steam generator blowdown, using a blowdown sample analysis, it is assumed that 5% of the I-131 and I-133 and 33% of the tritium in the blowdown stream become airborne with the remainder staying in the liquid phase. For other unmonitored releases, the quantity of airborne releases may be determined by performining a steam mass balance. For each of these release combinations, samples are analyzed weekly, monthly, quarterly, or for each batch releases according to Table 4.11-2.

Each sample provides a measure of the concentration of specific radionuclides, C_i, in gaseous effluent discharged at flow, F, during a time increment, Δt. Thus, each release is quantified according to the relation

$$Q_{ik}=C_{ik}\cdot\sum_j F_j\cdot\Delta t_j \tag{20}$$

where:

Q_{ik} = the quantity of radionuclide i released in a given effluent stream based on analysis k, (μCi)

C_{ik} = concentration of radionuclide i in a gaseous effluent identified by analysis k, (μCi/cc)

Δt_j = time increment j during which radionuclide i at concentration C_{ik} is being discharged, (sec).

F_j = effluent stream discharge rate during time increment Δt_j, (cc/sec)

Note: A steam mass to determine other unmonitored releases may be determined using the following

$$F_j = M_w - (M_l + M_s) \quad (21)$$

where:

M_w = the measured mass of makeup water entering the secondary system during time interval Δt_j (gm /sec).

M_l = the measured mass of water discharged from the secondary system as liquid during time interval Δt_j. e.g. steam generator blowdown.

M_s = the measured mass of steam or non-condensible gases discharged from the secondary system during time interval Δt_j. e.g. air ejector discharge.

Note: it is assumed that all of the I-131, I-133 and tritium in the other unmonitored releases are discharged as airborne species. It also assumed that gm/sec is equivalent to cc/sec.

3.5.2 Calculating the Dose Due to Iodine, Tritium, and Particulates

A person may be exposed directly to an airborne concentration of radioactive material discharged in an effluent gaseous stream and indirectly via pathways involving deposition of radioactive material onto the ground. Dose estimates should account for the exposure via the following pathways:

- o direct radiation from airborne radionuclides except noble gases
- o inhalation
- o direct radiation from ground plane deposition
- o fruits and vegetables
- o air-grass-cow-meat
- o air-grass-cow-milk

Of all these pathways, the air-grass-cow-milk pathway is by far the controlling dose contributor. The radioiodines contribute essentially all of the dose, by this pathway, with I-131 typically contributing greater than 95%. The dose transfer factors for the radioiodines are much greater than for any of the other radionuclides. The critical organ is the infant's thyroid. For this reason, the potential critical organ dose via airborne effluents can be estimated by determining an effective dose transfer factor for the radioiodines based on the typical radioactive effluent distribution, the air-grass-cow-milk pathway, and the infant thyroid as the receptor. Then for conservatism the total cumulative release of all radioiodines and particulates can be used along with the effective dose transfer factor to determine a conservative estimate of the infant thyroid dose.

Technical Specification 4.11.2.3, requires an evaluation be performed once per 31 days to verify that the accumulated total body or organ dose for the current calendar quarter and calendar year does not exceed the limit as given in 3.5. Dose commitment due to iodines and particulates may be calculated by using the following equation

$$DM_k = \frac{3.17 x 10^{-8}}{0.8} \cdot \frac{D}{Q} \cdot TG_{131} \cdot \sum_i Q_{ik} \qquad (22)$$

where:

DM_k = the dose commitment to an infant's thyroid received from exposure via the air-grass-cow-milk pathway and attributable to iodines identified in analysis k of effluent air, (mrem)

3.17×10^{-8} = conversion constant, (yr/sec)

0.8 = a conservatism factor which, in effect, increases the estimated dose to compensate for variability in the radionuclide distribution.

D/Q = relative deposition rate onto ground from a mixed mode atmospheric release (m^{-2})

TG_{131} = factor converting ground deposition of radioiodines to the dose commitment to an infant's thyroid exposed via the grass-cow-milk pathway, $\left(\frac{mrem/yr}{\mu Ci/m^2 \bullet sec}\right)$

Q_{ik} = the quantity of radionuclide i (I-131 and I-133) released in a given effluent stream based on a single analysis k, (μCi)

Specification 4.11.2.3 is satisfied by calculating the dose to an infant from iodine and particulates discharged as airborne effluents via the air-grass-cow-milk pathway and is evaluated by assuming a cow on pasture 4.5 miles west of the plant. (There are no milk or meat animals within 5 miles). At that location the reference atmospheric deposition factor is $D/Q = 5 \times 10^{-10}\ m^{-2}$.

When equation 22 is used to estimate the critical organ dose commitment, the effective dose transfer factor is:

$$TG_{131} = 6.5 \times 10^{11} \left(\frac{mrem/yr}{\mu Ci/m^2 \bullet sec}\right)$$

The reference data from which TG_{131} was derived are summarized in Table B-2 of Appendix B.

Alternately, the requirement of Specification 4.11.2.3, to perform once per 31 days determinations of dose commitments due to radioiodine, tritium and radioactive particulates in effluent air may be made by using equations (22), (23), (24), and (25):

- The dose commitment from exposure to airborne concentrations of radioactive material other than noble gas from a release, Q_{ik}, via the inhalation and irradiation pathways is calculated with the equation

$$D_{ank} = 3.17x10^{-8} \cdot \frac{\chi_d}{Q} \cdot \sum_i Q_{ik} \cdot \sum_p TA_{anip} \qquad (23)$$

where:

D_{ank} = the dose commitment to organ n of a person in age group a due to radionuclides identified in analysis k of an air effluent, (mrem).

3.17×10^{-8} = conversion constant, (yr/sec)

χ_d/Q = atmospheric dispersion factor adjusted for depletion by deposition, (sec/m^3).

Q_{ik} = the quantity of radionuclide i released in a given effluent stream based on analysis k, (μCi).

TA_{anip} = a factor converting airborne concentration of radionuclide i to dose commitment to organ n of a person in age group a where exposure is directly due to airborne material via pathway p (inhalation, or external exposure to the plume), $\left(\frac{mrem/yr}{\mu Ci/m^3}\right)$; See Appendix A.

The dose to a person from iodine and particulates discharged as airborne effluents via the inhalation and irradiation pathways is evaluated at the nearest garden 3.6 miles west northwest of the plant. At that location, the reference atmospheric dispersion factor adjusted for depletion by deposition is $\chi_d/Q = 1 \times 10^{-7}$ sec/m^3, (Table 3-7).

- The dose commitment via exposure pathways involving radionuclide deposition from the atmosphere onto vegetation or the ground is calculated with the equation

$$D_{ank} = 3.17x10^{-8} \cdot \frac{D}{Q} \cdot \sum_i Q_{ik} \cdot \sum_p TG_{anip} \quad (24)$$

where:

D/Q = relative deposition rate onto ground from a mixed mode atmospheric release, (m^{-2})

TG_{anip} = factor converting ground deposition of radionuclide i to dose commitment to organ n of a person in age group a where exposure is due to radioactive material via pathway p (direct radiation from ground plane deposition, fruits and vegetables, air-grass-cow-meat, or air-grass-cow-milk), $\left(\frac{mrem/yr}{\mu Ci/m^2 \bullet sec}\right)$, See Appendix A.

- The dose to a person from iodine and particulates discharged as airborne effluents via the air-grass-cow-milk pathway is evaluated by assuming a cow on pasture 4.5 miles west of the plant. (There are no milk or meat animals within 5 miles). At this location, the reference atmospheric deposition factor is $D/Q = 5 \times 10^{-10}\ m^{-2}$ (Table 3-8).
- The concentration of tritium in vegetation is a function of the airborne concentration rather than the deposition. Thus, the dose commitment from airborne tritium via vegetation (fruit and vegetables), air-grass-cow-milk, or air-grass-cow-meat pathways is calculated with the equation

$$D_{ank} = 3.17x10^{-8} \cdot \frac{\chi}{Q} \cdot \sum_i Q_{ik} \cdot \sum_p TA_{anip} \quad (25)$$

where:

χ/Q = atmospheric dispersion factor at the off-site location of interest (sec/m^3)

- The dose to a person from tritium via the vegetation (fruit and vegetables), air-grass-cow-milk,or air-grass-cow-meat pathways is evaluated at the nearest garden (with residence assumed) 3.6 miles west northwest of the plant. At that location,the reference atmospheric dispersion factor is $\chi/Q = 1 \times 10^{-7}$ sec/m^3.
- The dose commitment via a given pathway as a result of measured discharges from a release point is accumulated with

$$D_{an} = \sum_k D_{ank} \tag{26}$$

where:

D_{an} = the dose commitment to organ n of a person in age group a

k = the counting index; it may represent either:

p, analysis of a grab sample

w, a weekly sample analysis

m, a monthly composite analysis, or

q, a quarterly composite analysis

3.6 Effluent Noble Gas Monitor Alarm Setpoint

The radioactive gaseous effluent monitoring instrumentation channels alarm setpoints are set in accordance with Specification 3.3.3.6, to ensure the limits of Specification 3.11.2.1 are not exceeded.

Each radioactive noble gas effluent monitor setpoint is derived either on the basis of total body dose equivalent rate or noble gas concentration, at or beyond the site boundary. The setpoint derivations assume that noble gas releases occur at ground-level.

For the purpose of deriving a setpoint, the distribution of radioactive noble gases in an effluent stream may be determined in one of the following ways:

- Preferably, the radionuclide distribution is obtained by gamma spectrum analysis of identifiable noble gases in effluent gas samples. Results of analysis of one or more samples may be averaged to obtain a representative spectrum.
- In the event a representative distribution is unobtainnable from measurements by the radioactive gaseous waste sampling and analysis program, it may be based upon a historical spectrum appearing in Table 3-2.
- Alternately, the total activity concentration of radioactive noble gases may be assumed to be Xe-133. This approach is valid because Xe-133 contributes about 99% of the noble gas activity.

A noble gas effluent monitor setpoint, based on dose rate, is calculated with the equation below, or a method which gives a lower setpoint value.

$$S = 1.06 \left[\frac{h \cdot S_f}{F \cdot \chi/Q}\right] \left[\frac{\sum_i C_i}{\sum_i C_i \cdot DF_i}\right] + Bkg \qquad (27)$$

where:

S = The alarm setpoint, (cpm)

1.06 = conversion constant; 500 mrem/yr$\cdot$60 sec/min $\cdot$ 35.37 ft^3/m^3$\cdot$ 1m^3/10^{6}cm^3

h = monitor response to activity concentration of effluent, $\left(\frac{\text{cpm}}{\mu\text{Ci/cm}^3}\right)$

F = flow of gaseous effluent stream, i.e., flow past the monitor, (ft^3/min)

χ/Q = atmospheric dispersion factor at the off-site location of interest, (sec/m^3)

C_i = concentration of radionuclide i in gaseous effluent (μCi/cc).

DF_i = factor converting ground-level or split-wake release of radionuclide i to the total body dose equivalent rate at the location of potential exposure, $\left(\frac{\text{mrem}}{\text{yr}\cdot\mu\text{Ci/m}^3}\right)$; See Table 3-5.

S_f = A factor to allow for multiple sources from different or common release points. The allowowable operating setpoints will be controlled administratively by assigning a fraction of the total allowable release to each of the release sources.

Each monitoring channel has a unique response, h, which is determined by the instrument calibration.

Atmospheric dispersion depends upon the local atmospheric conditions. For the purpose of calculating a radioactive noble gas effluent monitor setpoint, the atmospheric dispersion factor, χ/Q, will be based on prevailing meteorological conditions or on reference meteorological conditions. The minimum atmospheric dispersion off-site derived from reference meteorological conditions at the site boundary is 5.8×10^{-7} sec/m^3 at a location 1950 meters south southeast of the plant.

The applicable dose conversion factors, DF_i, for deriving setpoints are in Table 3-5.

The limiting factor for Equation 27 is the total body dose rate limit of 500 mrem/year which is included in the 1.06 conversion factor. The use of the total body dose assumes that the total body dose will be the controlling dose rate and the dominant contributor to this dose will be Xe-133.

Setpoints may also be calculated based on concentration using the equation below, or a method which gives a lower setpoint value.

$$S = \frac{EC \cdot h \cdot S_f}{4.7x10^{-4} \cdot F \cdot \chi/Q} + BKG \qquad (28)$$

where:

EC = the unrestricted area effluent concentration for the effluent noble gas mixture. The EC for noble gas is calculated from the distribution of noble gases in the release with the equation:

$$EC = \frac{\sum_i C_i}{\sum_i \frac{C_i}{EC_i}} \qquad (29)$$

where:

EC_i = Ten times the 10CFR20, Appendix B, Table 2, Column 1 value

4.7×10^{-4} = conversion constant, $\frac{1m^3}{35.37ft^3} \times \frac{1\ min}{60\ sec}$

h = monitor response to activity concentration of effluent, $\left(\frac{cpm}{\mu Ci/cm^3}\right)$

F = flow of gaseous effluent stream, i.e. flow past the monitor (ft^3/min).

X/Q = atmospheric dispersion factor at the off-site location of interest (sec/m^3).

S_f = A factor to allow for multiple sources from different or common release points.The allowable operating setpoints will be controlled administratively by assigning a fraction of the total allowable release to each of the release sources.

3.7 Projected Dose for Gaseous Effluents

Technical Specification 3.11.2.4 requires that the gas decay tank system shall be operable and used to reduce radioactive materials in gaseous waste prior to their discharge if the projected gaseous effluent dose per reactor due to gaseous effluent releases to areas at and beyond the site boundary when averaged over 31 days exceeds 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation, and the ventilation exhaust treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge if the projected gaseous effluent dose per reactor due to gaseous effluent releases to areas at and beyond the site boundary when averaged over 31 days exceeds 0.3 mrem to any organ.

Technical Specification 4.11.2.4.1 requires the doses, to areas at and beyond the site boundary, due to radioactive material released in gaseous effluent to be projected at least once per 31 days.

This requirement is satisfied by extrapolating the dose to date during the current month to include the entire month. The dose to date is calculated as described in

Sections 3.4.1, 3.4.2, and 3.5.2.

The dose is projected with the relation:

$$P = \frac{31 \cdot D}{X} \tag{30}$$

where:

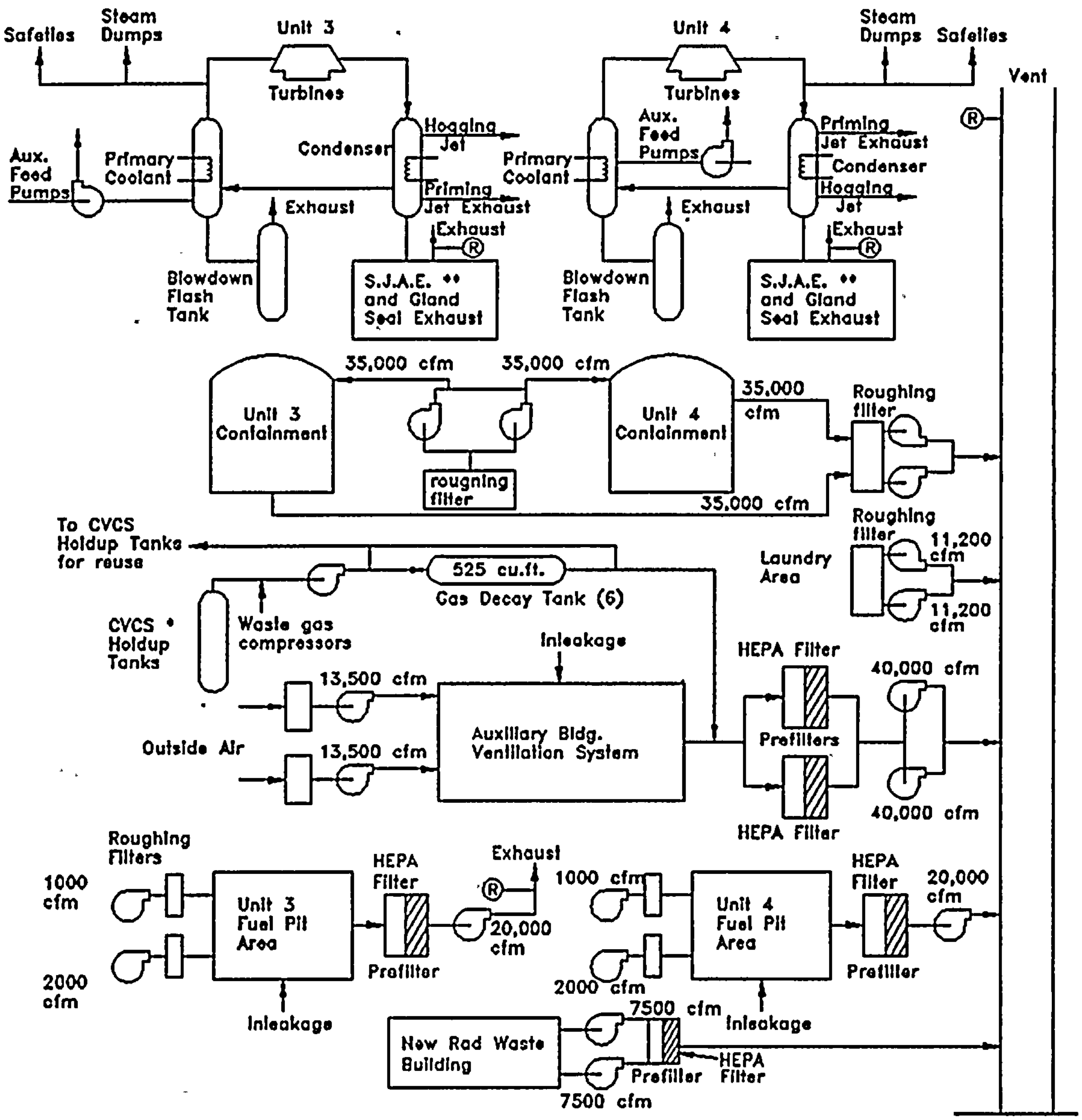
P = the projected dose during the month, (mrem)

31 = number of days in a calendar month, (days)

X = number of days in current month to date represented by available radioactive effluent sample, (days)

D = dose to date during current month calculated according to Sections 3.4.1, 3.4.2, and 3.5.2, (mrem), i.e., gamma, beta, or organ dose respectively.

Alternately, the monthly dose may be projected by computing the dose accumulated during the most recent month and assuming the result represents the projected dose for the current month. The dose during the proceeding month will be computed as described in Sections 3.4.1, 3.4.2, and 3.5.2.



* CVCS – Chemical and Volume Control System
** SJAE – Steam Jet Air Ejector
Ⓡ— – Effluent Monitoring Instrumentation

Figure 3-1

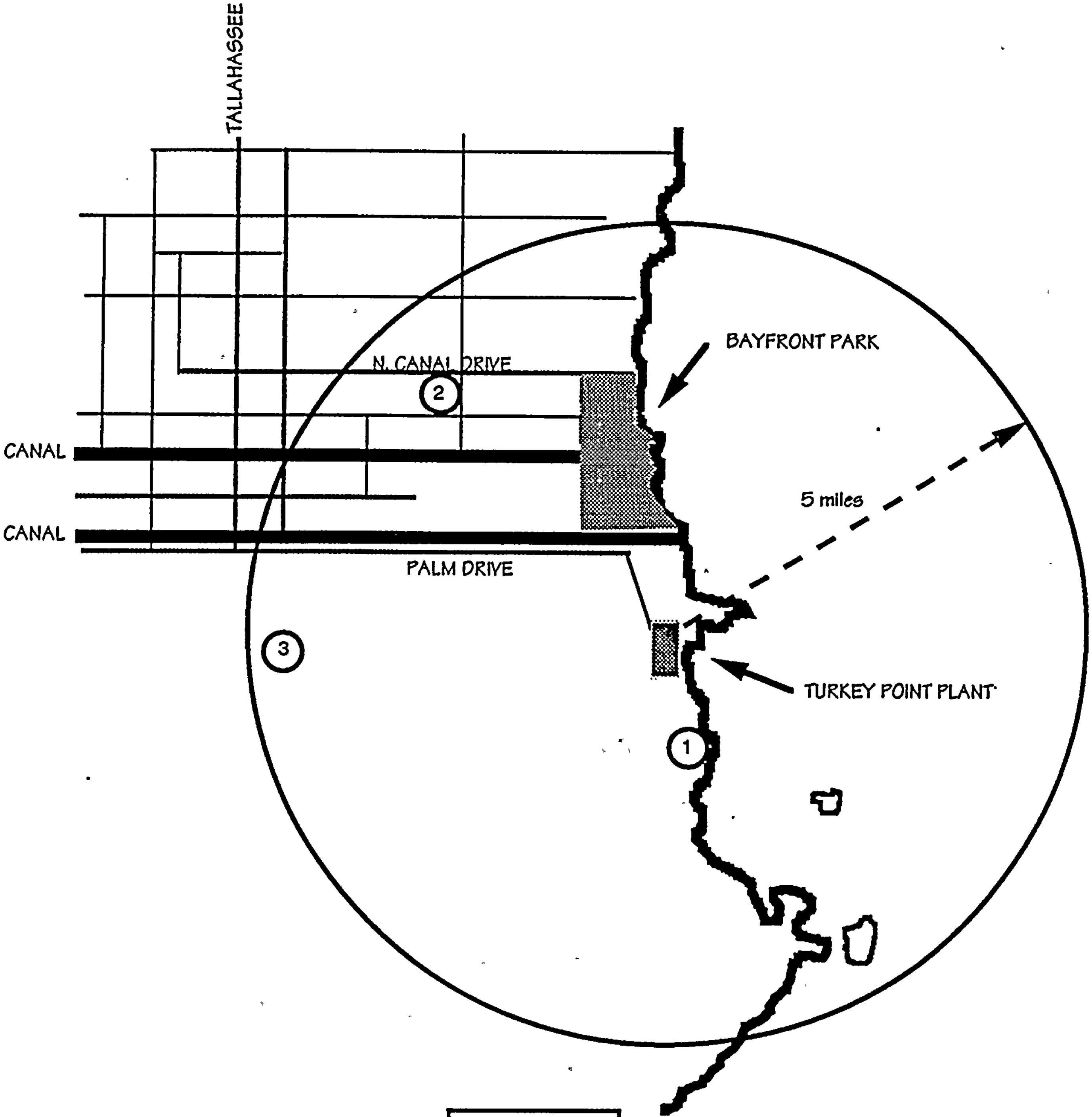


FIGURE 3.2

Locations at which doses due to airborne effluents from the Turkey Point Plant are calculated :

1. Beta and gamma doses to air, 1950 meters, SSE
2. Maximally exposed person, 5800 meters, WNW
3. Assumed beef and milk cow, 7250 meters, W

Table 3-1

Atmospheric Gaseous Release Points at the Turkey Point Units 3 and 4

Effluent Source	Release Point
Gas decay tanks	Plant vent
Radwaste Building	Plant vent
Auxiliary Building	Plant vent
Containment Purge	Plant vent
No. 4 spent fuel pit	Plant vent
No. 3 spent fuel pit	Spent fuel pit vent
Air ejectors	Turbine deck
Steam generator blowdown	Blowdown vent

Table 3-2

Distribution of Radioactive Noble Gases in Gaseous Effluent from Turkey Point Units 3 & 4

Nuclide	Release fraction [a,b]
Ar-41	9.2E-3
Kr-83m	--
Kr-85m	2.5E-4
Kr-85	2.5E-4
Kr-87	1.6E-4
Kr-88	2.1E-4
Xe-131m	4.4E-4
Xe-133m	1.2E-3
Xe-133	9.9E-1
Xe-135m	8.0E-4
Xe-135	3.4E-3
Xe-137	--
Xe-138	3.7E-4

a Based on measured discharge from Turkey Point Units 3 & 4 during 1978 through 1980.

b To estimate radionuclide concentrations in a sample in which only the total activity concentration has been measured, multiply the total activity concentration by the fraction of respective radionuclides listed here.

Table 3-3

Transfer Factors for Maximum Offsite Air Dose

Radionuclide	Air Dose Transfer Factors $A_{\gamma i}$ $\left(\frac{mrad}{\mu Ci\ sec/m^3}\right)$	$A_{\beta i}$ $\left(\frac{mrad}{\mu Ci\ sec/m^3}\right)$
Kr-83m	6.1E-7	9.1E-6
Kr-85m	3.9E-5	6.2E-5
Kr-85	5.4E-7	6.2E-5
Kr-87	2.0E-4	3.3E-4
Kr-88	4.8E-4	9.3E-5
Kr-89	5.5E-4	3.4E-4
Kr-90	5.2E-4	2.5E-4
Xe-131m	4.9E-6	3.5E-5
Xe-133m	1.0E-5	4.7E-5
Xe-133	1.1E-5	3.3E-5
Xe-135m	1.1E-4	2.3E-5
Xe-135	6.1E-5	7.8E-5
Xe-137	4.8E-5	4.0E-4
Xe-138	2.9E-4	1.5E-4
Ar-41	2.9E-4	1.0E-4

Ref: Regulatory Guide 1.109, Revision 1, Table B-1
Note: Values in the regulatory guide are in units of pCi*yr, to convert to units of µCi*sec multiply by a factor of 3.171 E-2.

Table 3-4

Transfer Factors for Maximum Dose to a
Person Offsite due to Radioactive Noble Gases

Radionuclide	Air Dose Transfer Factors $P_{\gamma i}$ $\left(\frac{mrem}{\mu Ci\ sec/m^3}\right)$	Air Dose Transfer Factors $S_{\beta i}$ $\left(\frac{mrem}{\mu Ci\ sec/m^3}\right)$
Kr-83m	2.4E-9	--
Kr-85m	3.7E-5	4.6E-5
Kr-85	5.1E-7	4.2E-5
Kr-87	1.9E-4	3.1E-4
Kr-88	4.7E-4	7.5E-5
Kr-89	5.3E-4	3.2E-4
Kr-90	4.9E-4	2.3E-4
Xe-131m	2.9E-6	1.5E-5
Xe-133m	8.0E-6	3.1E-5
Xe-133	9.3E-6	9.7E-6
Xe-135m	9.9E-5	2.3E-5
Xe-135	5.7E-5	5.9E-5
Xe-137	4.5E-5	3.9E-4
Xe-138	2.8E-4	1.3E-4
Ar-41	2.8E-4	8.5E-5

Ref: Regulatory Guide 1.109, Revision 1, Table B-1.

Note: Values in the regulatory guide are quoted in units of pCi*yr, to convert to units of μCi*sec multiply by a factor of 3.171 E-2.

Table 3-5

Dose Conversion Factors for Deriving Radioactive Noble Gas Effluent Monitor Setpoints

Radionuclide	Factor DF_i for Ground-level or Split-Wake Release $\left(\frac{\text{mrem}}{\text{yr}\ \frac{\mu\text{Ci}}{\text{m}^3}}\right)$
Kr-83m	7.56 E-2
Kr-85m	1.17 E3
Kr-85	1.61 E1
Kr-87	5.92 E3
Kr-88	1.47 E4
Kr-89	1.66 E4
Kr-90	1.56 E4
Xe-131m	9.15 E1
Xe-133m	2.51 E2
Xe-133	2.94 E2
Xe-135m	3.12 E3
Xe-135	1.81 E3
Xe-137	1.42 E3
Xe-138	8.83 E3
Xe-139	5.02 E3
Ar-41	8.84 E3

Table 3-6

REFERENCE METEOROLOGY
ANNUAL AVERAGE ATMOSPHERIC DISPERSION FACTORS

$\frac{X}{Q} \quad \frac{sec}{m^3}$

X/Q are annual averaged factors of atmospheric dispersion of a mixed mode gaseous release from the Turkey Point Plant at various distances and compass points from the plant.

Period of record: 01/01/76 to 12/31/77

BASE DISTANCE IN MILES / KILOMETERS

MILES	.25	.75	1.50	2.50	3.50	4.50	5.50	7.00
KM.	.40	1.21	2.41	4.02	5.63	7.24	8.85	11.26
NNE	8.9E-07	1.9E-07	8.3E-08	5.0E-08	3.0E-08	2.2E-08	1.9E-08	1.4E-08
NE	6.9E-07	1.5E-07	6.3E-08	3.8E-08	2.5E-08	2.1E-08	1.3E-08	1.0E-08
ENE	8.4E-07	1.4E-07	7.5E-08	3.9E-08	2.8E-08	2.3E-08	1.8E-08	1.3E-08
E	8.6E-07	1.9E-07	9.1E-08	5.1E-08	3.6E-08	2.7E-08	2.2E-08	1.7E-08
ESE	6.6E-07	1.5E-07	7.9E-08	4.5E-08	2.9E-08	2.3E-08	1.9E-08	1.2E-08
SE	1.6E-06	2.8E-07	1.1E-07	6.1E-08	4.2E-08	3.0E-08	2.6E-08	2.1E-08
SSE	4.9E-06	9.2E-07	3.6E-07	1.8E-07	1.1E-07	9.0E-08	7.1E-08	4.9E-08
S	2.9E-06	4.6E-07	1.8E-07	1.0E-07	7.8E-08	5.4E-08	4.6E-08	3.3E-08
SSW	6.5E-07	1.6E-07	6.5E-08	4.6E-08	2.4E-08	2.6E-08	1.8E-08	1.4E-08
SW	1.5E-06	3.2E-07	1.4E-07	7.9E-08	4.9E-08	3.2E-08	2.7E-08	1.9E-08

Table 3-6 continued

Page 2
REFERENCE METEOROLOGY
ANNUAL AVERAGE ATMOSPHERIC DISPERSION FACTORS

MILES	.25	.75	1.50	2.50	3.50	4.50	5.50	7.00
KM.	.40	1.21	2.41	4.02	5.63	7.24	8.85	11.26
WSW	2.9E-06	6.3E-07	2.3E-07	1.3E-07	7.6E-08	5.5E-08	4.2E-08	3.1E-08
W	6.3E-06	1.3E-06	5.2E-07	2.6E-07	1.7E-07	1.2E-07	9.2E-08	6.6E-08
WNW	4.1E-06	8.7E-07	3.4E-07	1.7E-07	1.2E-07	8.1E-08	6.3E-08	4.2E-08
NW	2.7E-06	6.0E-07	2.4E-07	1.2E-07	7.6E-08	5.1E-08	4.3E-08	3.2E-08
NNW	1.4E-06	2.9E-07	1.2E-07	6.8E-08	4.5E-08	3.0E-08	2.4E-08	1.5E-08
N	9.5E-07	2.1E-07	8.5E-08	4.5E-08	3.2E-08	2.2E-08	1.7E-08	1.3E-08

Table 3-6 continued

Page 3
REFERENCE METEOROLOGY
ANNUAL AVERAGE ATMOSPHERIC DISPERSION FACTORS

$$\frac{X}{Q} \cdot \frac{sec}{m^3}$$

X/Q are annual averaged factors of atmospheric dispersion of a mixed mode gaseous release from the Turkey Point Plant at various distances and compass points from the plant.

Period of record: 01/01/76 to 12/31/77

BASE DISTANCE IN MILES / KILOMETERS

MILES	9.00	11.00	.79	5.00	1.00	2.00	2.75	4.30
KM.	14.48	17.70	1.27	8.04	1.61	3.22	4.42	6.92
NNE	9.8E-09	6.6E-09	1.8E-07	2.0E-08	1.4E-07	6.2E-08	4.4E-08	2.3E-08
NE	7.3E-09	5.4E-09	1.5E-07	1.6E-08	1.1E-07	4.8E-08	3.5E-08	2.1E-08
ENE	1.1E-08	7.4E-09	1.4E-07	2.0E-08	1.0E-07	5.2E-08	3.6E-08	2.4E-08
E	1.3E-08	9.8E-09	1.7E-07	2.4E-08	1.3E-07	6.3E-08	4.6E-08	2.8E-08
ESE	1.1E-08	9.6E-09	1.4E-07	2.0E-08	1.2E-07	5.7E-08	4.0E-08	2.4E-08
SE	1.5E-08	1.3E-08	2.7E-07	2.7E-08	1.9E-07	7.8E-08	5.5E-08	3.1E-08
SSE	3.5E-08	2.7E-08	8.7E-07	7.9E-08	6.3E-07	2.5E-07	1.6E-07	9.4E-08
S	2.3E-08	1.8E-08	4.2E-07	5.0E-08	3.1E-07	1.3E-07	9.5E-08	5.8E-08
SSW	9.4E-09	7.1E-09	1.5E-07	2.1E-08	1.1E-07	5.4E-08	3.8E-08	2.5E-08
SW	1.4E-08	1.0E-08	3.0E-07	2.9E-08	2.3E-07	1.0E-07	6.9E-08	3.5E-08

Table 3-6 continued

Page 4
REFERENCE METEOROLOGY
ANNUAL AVERAGE ATMOSPHERIC DISPERSION FACTORS

BASE DISTANCE IN MILES / KILOMETERS

MILES	9.00	11.00	.79	5.00	1.00	2.00	2.75	4.30
KM.	14.48	17.70	1.27	8.04	1.61	3.22	4.42	6.92
WSW	2.2E-08	1.8E-08	5.9E-07	4.8E-08	4.3E-07	1.7E-07	1.0E-07	5.8E-08
W	4.5E-08	3.5E-08	1.2E-06	1.0E-07	9.0E-07	3.5E-07	2.3E-07	1.3E-07
WNW	2.9E-08	2.3E-08	8.1E-07	7.1E-08	5.9E-07	2.3E-07	1.6E-07	8.6E-08
NW	2.0E-08	1.5E-08	5.6E-07	4.7E-08	4.1E-07	1.6E-07	1.0E-07	5.6E-08
NNW	1.0E-08	8.3E-09	2.7E-07	2.6E-08	2.0E-07	9.1E-08	6.1E-08	3.2E-08
N	1.0E-08	7.2E-09	1.9E-07	2.0E-08	1.5E-07	5.9E-08	4.0E-08	2.3E-08

NUMBER OF VALID OBSERVATIONS = 16538
NUMBER OF INVALID OBSERVATIONS = 1006
NUMBER OF CALMS LOWER LEVEL = 195
NUMBER OF CALMS UPPER LEVEL = 383

Table 3-7

REFERENCE METEOROLOGY
DEPOSITION DEPLETED ANNUAL AVERAGE ATMOSPHERIC DISPERSION FACTORS

$$\frac{X_d}{Q} \quad \frac{sec}{m^3}$$

X_d/Q are annual averaged factors of atmospheric dispersion of a mixed mode gaseous release at various distances from the Turkey Point Plant which have been corrected for depletion from the plume by fallout and deposition.

Period of record: 01/01/76 to 12/31/77

BASE DISTANCE IN MILES / KILOMETERS

SECT	.25	.75	1.50	2.50	3.50	4.50	5.50	7.00
	.40	1.21	2.41	4.02	5.63	7.24	8.85	11.26
NNE	8.7E-07	1.7E-07	7.3E-08	4.4E-08	2.7E-08	1.9E-08	1.6E-08	1.2E-08
NE	6.9E-07	1.4E-07	5.5E-08	3.3E-08	2.2E-08	1.7E-08	1.2E-08	8.8E-09
ENE	8.0E-07	1.2E-07	6.5E-08	3.4E-08	2.4E-08	2.0E-08	1.6E-08	1.2E-08
E	8.6E-07	1.7E-07	7.6E-08	4.4E-08	3.1E-08	2.4E-08	1.9E-08	1.5E-08
ESE	6.1E-07	1.3E-07	6.9E-08	3.9E-08	2.5E-08	2:0E-08	1.6E-08	1.1E-08
SE	1.5E-06	2.6E-07	9.5E-08	5.2E-08	3.4E-08	2.4E-08	2.1E-08	1.7E-08
SSE	4.7E-06	8.2E-07	3.1E-07	1.5E-07	9.2E-08	7.4E-08	5.8E-08	3.8E-08
S	2.8E-06	4.2E-07	1.5E-07	8.5E-08	6.4E-08	4.4E-08	3.7E-08	2.6E-08
SSW	6.1E-07	1.4E-07	5.6E-08	3.9E-08	2.0E-08	2.2E-08	1.5E-08	1.2E-08
SW	1.3E-06	2.8E-07	1.3E-07	6.7E-08	4.2E-08	2.7E-08	2.3E-08	1.5E-08

Table 3-7

Page 2
REFERENCE METEOROLOGY
DEPOSITION DEPLETED ANNUAL AVERAGE ATMOSPHERIC DISPERSION FACTORS

BASE DISTANCE IN MILES / KILOMETERS

MILES	.25	.75	1.50	2.50	3.50	4.50	5.50	7.00
KM.	.40	1.21	2.41	4.02	5.63	7.24	8.85	11.26
WSW	2.7E-06	5.6E-07	2.1E-07	1.0E-07	6.4E-08	4.6E-08	3.5E-08	2.6E-08
W	5.9E-06	1.2E-06	4.4E-07	2.2E-07	1.4E-07	9.9E-08	7.6E-08	5.4E-08
WNW	3.8E-06	7.7E-07	2.9E-07	1.5E-07	9.8E-08	7.0E-08	5.4E-08	3.6E-08
NW	2.5E-06	5.4E-07	2.1E-07	1.1E-07	6.8E-08	4.5E-08	3.8E-08	2.8E-08
NNW	1.4E-06	2.6E-07	1.1E-07	6.0E-08	4.0E-08	2.6E-08	2.0E-08	1.3E-08
N	8.8E-07	1.9E-07	7.8E-08	3.9E-08	2.8E-08	1.9E-08	1.5E-08	1.1E-08

Table 3-7 continued

Page 3
REFERENCE METEOROLOGY
DEPOSITION DEPLETED ANNUAL AVERAGE ATMOSPHERIC DISPERSION FACTORS

BASE DISTANCE IN MILES / KILOMETERS

MILES	9.00	11.00	.79	5.00	1.00	2.00	2.75	4.30
KM.	14.48	17.70	1.27	8.04	1.61	3.22	4.42	6.92
NNE	8.5E-09	6.0E-09	1.6E-07	1.8E-08	1.2E-07	5.5E-08	3.8E-08	2.1E-08
NE	6.3E-09	4.5E-09	1.3E-07	1.4E-08	9.4E-08	4.2E-08	3.0E-08	1.8E-08
ENE	9.0E-09	6.7E-09	1.2E-07	1.8E-08	9.1E-08	4.5E-08	3.1E-08	2.0E-08
E	1.1E-08	7.9E-09	1.5E-07	2.1E-08	1.2E-07	5.5E-08	3.9E-08	2.4E-08
ESE	8.8E-09	8.3E-09	1.3E-07	1.8E-08	1.0E-07	5.0E-08	3.4E-08	2.0E-08
SE	1.3E-08	1.0E-08	2.4E-07	2.3E-08	1.7E-07	6.7E-08	4.7E-08	2.6E-08
SSE	2.7E-08	2.1E-08	7.7E-07	6.4E-08	5.6E-07	2.2E-07	1.3E-07	7.7E-08
S	1.9E-08	1.3E-08	3.8E-07	4.1E-08	2.7E-07	1.1E-07	7.8E-08	4.8E-08
SSW	7.9E-09	5.7E-09	1.4E-07	1.8E-08	9.6E-08	4.7E-08	3.2E-08	2.2E-08
SW	1.1E-08	8.6E-09	2.7E-07	2.4E-08	2.0E-07	9.1E-08	5.9E-08	2.9E-08

Table 3-7 continued

Page 4
REFERENCE METEOROLOGY
ANNUAL AVERAGE ATMOSPHERIC DISPERSION FACTORS

BASE DISTANCE IN MILES / KILOMETERS

MILES	9.00	11.00	.79	5.00	1.00	2.00	2.75	4.30
KM.	14.48	17.70	1.27	8.04	1.61	3.22	4.42	6.92
WSW	1.8E-08	1.4E-08	5.2E-07	4.0E-08	3.8E-07	1.4E-07	8.7E-08	4.8E-08
W	3.7E-08	2.8E-08	1.1E-06	8.6E-08	7.9E-07	3.1E-07	2.0E-07	1.0E-07
WNW	2.5E-08	2.0E-08	7.3E-07	6.1E-08	5.1E-07	2.0E-07	1.4E-07	7.4E-08
NW	1.8E-08	1.3E-08	5.1E-07	4.1E-08	3.6E-07	1.4E-07	8.9E-08	5.0E-08
NNW	9.1E-09	6.9E-09	2.4E-07	2.3E-08	1.8E-07	7.7E-08	5.4E-08	2.8E-08
N	8.7E-09	6.3E-09	1.8E-07	1.7E-08	1.3E-07	5.2E-08	3.5E-08	2.0E-08

NUMBER OF VALID OBSERVATIONS = 16538
NUMBER OF INVALID OBSERVATIONS = 1006
NUMBER OF CALMS LOWER LEVEL = 195
NUMBER OF CALMS UPPER LEVEL = 383

Table 3-8

REFERENCE METEOROLOGY
ANNUAL AVERAGED RELATIVE DEPOSITION RATE

$\frac{D}{Q} \quad \frac{1}{M^2}$

D/Q are annual averaged factors representing the fraction of a mixed mode airborne release from the Turkey Point Plant which is deposited on a square meter area of land at various distances and compass points from the plant.
Period of record: 01/01/76 to 12/31/77

BASE DISTANCE IN MILES / KILOMETERS

MILES	.25	.75	1.50	2.50	3.50	4.50	5.50	7.00
KM.	.40	1.21	2.41	4.02	5.63	7.24	8.85	11.26
NNE	6.4E-09	1.5E-09	4.7E-10	2.0E-10	9.1E-11	5.5E-11	4.1E-11	2.7E-11
NE	3.5E-09	8.7E-10	2.8E-10	1.2E-10	6.4E-11	4.3E-11	2.5E-11	1.7E-11
ENE	2.8E-09	5.1E-10	2.1E-10	7.6E-11	4.1E-11	2.9E-11	1.9E-11	1.2E-11
E	2.7E-09	6.6E-10	2.4E-10	1.1E-10	5.8E-11	3.7E-11	2.5E-11	1.6E-11
ESE	1.6E-09	4.2E-10	1.9E-10	7.7E-11	4.0E-11	2.7E-11	1.8E-11	1.0E-11
SE	5.3E-09	1.2E-09	3.7E-10	1.6E-10	9.0E-11	5.4E-11	4.2E-11	2.9E-11
SSE	2.6E-08	5.2E-09	1.8E-09	6.8E-10	3.5E-10	2.5E-10	1.8E-10	1.0E-10
S	1.2E-08	2.1E-09	6.7E-10	3.0E-10	2.0E-10	1.2E-10	9.1E-11	5.8E-11
SSW	2.3E-09	7.2E-10	2.4E-10	1.2E-10	5.3E-11	4.8E-11	2.8E-11	2.0E-11
SW	1.1E-08	2.7E-09	1.0E-09	4.3E-10	2.3E-10	1.2E-10	9.6E-11	5.5E-11

Table 3-8
Page 2
REFERENCE METEOROLOGY
ANNUAL AVERAGED RELATIVE DEPOSITION RATE

BASE DISTANCE IN MILES / KILOMETERS

MILES	.25	.75	1.50	2.50	3.50	4.50	5.50	7.00
KM.	.40	1.21	2.41	4.02	5.63	7.24	8.85	11.26
WSW	2.3E-08	5.0E-09	1.5E-09	6.1E-10	3.2E-10	2.0E-10	1.4E-10	8.5E-11
W	5.7E-08	1.2E-08	3.5E-09	1.4E-09	7.6E-10	4.9E-10	3.3E-10	2.1E-10
WNW	4.1E-08	9.6E-09	2.7E-09	1.0E-09	5.7E-10	3.4E-10	2.4E-10	1.4E-10
NW	2.4E-08	6.2E-09	1.7E-09	6.1E-10	3.1E-10	1.8E-10	1.3E-10	8.5E-11
NNW	1.2E-08	3.0E-09	9.5E-10	3.6E-10	2.0E-10	1.1E-10	7.5E-11	4.2E-11
N	5.8E-09	1.6E-09	4.8E-10	1.8E-10	9.6E-11	5.8E-11	4.0E-11	2.5E-11

Table 3-8 continued

Page 3
REFERENCE METEOROLOGY
ANNUAL AVERAGED RELATIVE DEPOSITION RATE

BASE DISTANCE IN MILES / KILOMETERS

MILES	9.00	11.00	.79	5.00	1.00	2.00	2.75	4.30
KM.	14.48	17.70	1.27	8.04	1.61	3.22	4.42	6.92
NNE	1.6E-11	9.3E-12	1.4E-09	4.7E-11	9.6E-10	2.8E-10	1.6E-10	6.2E-11
NE	9.9E-12	6.2E-12	8.1E-10	3.2E-11	5.6E-10	1.8E-10	1.1E-10	4.6E-11
ENE	8.1E-12	5.2E-12	5.0E-10	2.3E-11	3.6E-10	1.2E-10	6.4E-11	3.0E-11
E	1.0E-11	6.6E-12	5.9E-10	3.0E-11	4.3E-10	1.5E-10	8.8E-11	3.9E-11
ESE	7.5E-12	5.8E-12	4.1E-10	2.2E-11	3.1E-10	1.2E-10	6.5E-11	2.8E-11
SE	1.8E-11	1.3E-11	1.1E-09	4.7E-11	7.1E-10	2.3E-10	1.3E-10	6.0E-11
SSE	6.6E-11	4.5E-11	4.9E-09	2.1E-10	3.4E-09	1.1E-09	5.8E-10	2.6E-10
S	3.4E-11	2.3E-11	1.9E-09	1.0E-10	1.4E-09	4.4E-10	2.7E-10	1.3E-10
SSW	1.0E-11	6.6E-12	6.7E-10	3.6E-11	4.5E-10	1.7E-10	9.7E-11	4.8E-11
SW	3.5E-11	2.2E-11	2.5E-09	1.1E-10	1.9E-09	6.3E-10	3.6E-10	1.4E-10

Table 3-8 continued

Page 4
REFERENCE METEOROLOGY
ANNUAL AVERAGED RELATIVE DEPOSITION RATE

BASE DISTANCE IN MILES / KILOMETERS

MILES	9.00	11.00	.79	5.00	1.00	2.00	2.75	4.30
KM.	14.48	17.70	1.27	8.04	1.61	3.22	4.42	6.92
WSW	5.5E-11	3.8E-11	4.6E-09	1.6E-10	3.2E-09	9.7E-10	4.9E-10	2.2E-10
W	1.2E-10	8.7E-11	1.1E-09	3.9E-10	7.4E-09	2.2E-09	1.2E-09	5.0E-10
WNW	8.8E-11	6.1E-11	8.7E-09	2.8E-10	5.7E-09	1.6E-09	9.0E-10	3.8E-10
NW	4.5E-11	3.2E-11	5.6E-09	1.5E-10	3.7E-09	9.5E-10	5.0E-10	2.0E-10
NNW	2.5E-11	1.8E-11	2.7E-09	8.8E-11	1.8E-09	5.4E-10	3.0E-10	1.2E-10
N	1.7E-11	1.1E-11	1.4E-09	4.8E-11	1.0E-09	2.7E-10	1.5E-10	6.5E-11

NUMBER OF VALID OBSERVATIONS = 16538
NUMBER OF INVALID OBSERVATIONS = 1006
NUMBER OF CALMS LOWER LEVEL = 195
NUMBER OF CALMS UPPER LEVEL = 383

4.0 Dose Commitment from Releases Over Extended Time

4.1 Releases During 12 Months

Technical Specification 3.11.4 implements 40 CFR Part 190.102. It requires the annual (calendar year) dose or dose commitment to any member of the public from all uranium fuel cycles to be limited to less than or equal to 75 mrem to the thyroid and 25 mrem to the total body or any other organ.

Fuel cycle sources or nuclear power reactors other than the Turkey Point Plant itself do not measurably or significantly increase the radioactivity concentration in the vicinity of the Plant; therefore, only radiation and radioactivity in the environment attributable to the Plant itself are considered in the assessment of compliance with 40 CFR Part 190.102.

In the event a dose calculated for the purpose of assessing compliance with Specification 3.11.1.2, 3.11.2., or 3.11.2.3, exceeds 2 times the limit stated therein, then a calculation shall be made to determine whether any limit in 3.11.4 has been exceeded. The total dose calculated pursuant to Technical Specification 3.11.4 must include direct radiation contributions and the methodology for calculating direct radiation contribution must be indicated in the Annual Radioactive Effluent Release Report. These calculations should be made on the basis of radioactive effluents during the year-to-date and reference meteorological data or averaged meteorological data during completed quarters of the year-to-date.

Separately, an evaluation of doses due to effluents during the year is performed annually and reported in the Annual Radioactive Effluent Release Report submitted each year. This evaluation uses reference meteorological data or annual averaged meteorological data concurrent with the annual gaseous releases to evaluate atmospheric dispersion, deposition, and plume gamma exposure.

To assess compliance with Technical Specification 3.11.4, evaluations of dose due to liquid and gaseous effluents are calculated as described by the equations for:

- o total body dose due to liquid effluent via irradiation by radionuclides deposited on cooling canal shoreline as in Section 2.4 (Equation 7)

- o total body dose due to noble gas γ as in Section 3.4.1 (Equation 16)

- o skin dose due to noble gas ß as in Section 3.4.2 (Equation 17)

- o total body and maximally exposed organ doses due to gaseous effluents other than noble gases* as in Section 3.5.2 (Equation 22).

The doses are calculated on the basis of liquid and gaseous effluents from the Plant, sampled and analyzed in accord with Technical Specification Tables 4.11-1 and 4.11-2.

The receptor of the dose is described such that the dose to any member of the public is not likely to be underestimated. The receptor is selected on the basis of the combination of applicable pathways of exposure to gaseous effluent identified in the annual land use census and maximum ground level χ/Q at the residence. Conditions more conservative than appropriate for the maximally exposed person may be assumed in the dose assessment. Environmental pathway-to-dose transfer factors used in the dose calculations appear in Appendix A.

4.2 Environmental Measurements

When assessing compliance with 40 CFR Part 190 or 10 CFR Part 50 Appendix I dose limits, Radiological Environmental Monitoring Program results may be used to indicate actual radioactivity levels in the environment attributable to the Turkey Point Plant as an alternate to calculating the concentrations from radioactive effluent measurements. The measured environmental activity levels may thus be used to supplement the evaluation of doses to real persons for assessing compliance with 40 CFR Part 190 or 10 CFR Part 50 Appendix I.

4.3 Dose to a Person from Noble Gases

Technical Specification 3.11.4 requires the calculation of the annual (calendar year) dose or dose commitment to a person off-site exposed to radioactive liquid and gaseous effluents from the plant. One component of personal dose is total body irradiation by gamma rays from noble gases. Another is irradiation of skin by beta and gamma radiation from noble gases. The methods for calculating these doses are presented in Sections 4.3.1 and 4.3.2.

The amount of radioactive noble gas discharged is determined in the manner described in Section 3.3.

*Radioactive I-131, I-133, tritium, and radioactive material in particulate form having a half-life greater than 8 days.

4.3.1 Gamma Dose to Total Body

The gamma radiation dose to the whole body of a member of the public as a consequence of noble gas released from the Plant is calculated with the equation:

$$D_\gamma = \frac{\chi}{Q} \sum_i Q_i \cdot P_{\gamma i} \tag{31}$$

where:

D_γ = noble gas gamma dose to total body, (mrem)

Q_i = quantity of radioactive noble gas i discharged in gaseous effluent, (μCi)

χ/Q= atmospheric dispersion factor at the off-site location of interest, (sec/m^3)

$P_{\gamma i}$= factor converting time integrated, ground level concentration of noble gas nuclide i to total body dose from gamma radiation listed in Table 3-4, $\left(\frac{\text{mrem}}{\mu\text{Ci}\cdot\text{sec/m}^3}\right)$

When the total body dose due to gamma radiation from noble gas required by Technical Specification 3.11.4 is calculated, the most exposed receptor is located 3.6 miles west northwest of the plant where the reference meteorological dispersion factor, χ/Q, is 1×10^{-7} sec/m^3.

This calculation is the same technique used in Section 3.3.1, Equation 13, but is extrapolated to an annual release except the χ/Q value is for the most exposed receptor, not the minimum dispersion point off-site.

4.3.2 Dose to Skin

The radiation dose to the skin of a member of the public due to noble gas released from the Plant may be calculated with the equation:

$$D = \frac{\chi}{Q}\left[\sum_i Q_i \cdot S_{\beta i} + 1.11 \sum Q_i \cdot A_{\gamma i}\right] \quad (32)$$

where:

D = dose to skin due to noble gases, (mrem)

X/Q = atmospheric dispersion factor at the off-site location of interest, (sec/m^3).

Q_i = quantity of radioactive noble gas i discharged in gaseous effluent, (μCi).

$S_{\beta i}$ = factor converting time integrated ground level concentration of noble gas to skin dose from beta radiation listed in Table 3-4, $\left(\frac{mrem}{\mu Ci \cdot sec/m^3}\right)$

1.11 = ratio of tissue dose equivalent to air dose in a radiation field, (mrem/mrad)

$A_{\gamma i}$ = factor for converting time integrated, ground-level concentration of noble gas radionuclide i to air dose from its gamma radiation listed in Table 3-3, $\left(\frac{mrad}{\mu Ci \cdot sec/m^3}\right)$.

When the skin beta dose due to noble gas required by Specification 3.11.4 is calculated, the most exposed receptor is located 3.6 miles west northwest of the Plant where the reference meteorological dispersion factor, χ/Q, is 1×10^{-7} sec/m^3.

The total dose to the skin from noble gases is approximately equal to the beta radiation dose to the skin plus the gamma radiation dose to the total body.

This is the same technique used in Section 3.3.2, Equation 14, but is extrapolated to an annual release, except the χ/Q value is for the most exposed receptor rather than the minimum dispersion point off-site.

APPENDIX D

EXAMPLE CALCULATIONS

APPENDIX D

EXAMPLE CALCULATIONS

1. Determination of Radionuclide Concentration in the Condenser Cooling Water Mixing Basin, C_{zi}, from a Liquid Release (Section 2.3).

$$C_{zi} = C_i \cdot \frac{F_1}{F_2} \qquad (1)*$$

where:

C_i+ = concentration of radionuclide i in the liquid radwaste released, µCi/ml, obtained from nuclide analyses report for the liquid release sample taken prior to release.

F_1 = Flow rate from monitor tank = 100 gal/min.

F_2 = Total condenser cooling water flow = 156,000 gpm/circulating pump; total capacity Units 3 and 4 = 8 pumps x 156,000 = 1,248,000 gal/min.

+ Note: When determining actual release concentrations, contact units 1 and 2 to determine how many, if any, circulating pumps were running during release. The flow of these pumps must be included when determining F_2.

Example:

For a monitor tank analysis (from Nuclide Analysis Report), C_i is equal to the following concentrations:

Co-60	8×10^{-5} µCi/ml
Co-58	2×10^{-6} µCi/ml
Cr-51	7×10^{-7} µCi/ml
Mn-54	5×10^{-6} µCi/ml
Cs-137	5×10^{-7} µCi/ml
I-131	3×10^{-7} µCi/ml

F_1/F_2 = 100 gpm/1,248,000 gpm = 8×10^{-5}

*Note: Equation numbers refer to the equation listed by that number in the ODCM text.

	C_i	F_1/F_2	C_{zi}
Co-60	8×10^{-5}	8×10^{-5}	6.4×10^{-9}
Co-58	2×10^{-6}	8×10^{-5}	1.6×10^{-10}
Cr-51	7×10^{-7}	8×10^{-5}	5.6×10^{-11}
Mn-54	5×10^{-6}	8×10^{-5}	4.0×10^{-10}
Cs-137	5×10^{-7}	8×10^{-5}	4.0×10^{-11}
I-131	3×10^{-7}	8×10^{-5}	2.4×10^{-11}

2. **Determination of the Fraction of the Unrestricted Area EC from a Batch Release of Liquid Radwaste, FEC_b (Section 2.3.1).**

$$FEC_b = \frac{\sum_i \frac{C_{zi}}{EC_i}}{E_b} \qquad (2)$$

where:

C_{zi} = Radionuclide concentration in condenser cooling water mixing basin, µCi/ml

EC_i = Ten times the effluent concentration from 10 CFR 20 Appendix B, Table 2, Column 2, µCi/ml

E_b = 0.5; E_b is an adjustment to account for radionuclides not measured prior to release but measured in the monthly and quarterly sample per Technical Specification Table 4.11-1.

Example:

Σ FEC for a release must be less than 1 or the release cannot be made. Σ FEC for the batch release in example 1 above is calculated as follows:

Nuclide	C_{zi}	EC_i*	C_{zi}/EC_i	E_b	FEC_b
Co-60	6.4×10^{-9}	3×10^{-5}	2.1×10^{-4}	0.5	4.2×10^{-4}
Co-58	1.6×10^{-10}	8×10^{-5}	2.0×10^{-6}	0.5	4.0×10^{-6}
Cr-51	5.6×10^{-11}	5×10^{-3}	1.1×10^{-8}	0.5	2.2×10^{-8}
Mn-54	4.0×10^{-10}	3×10^{-4}	1.3×10^{-6}	0.5	2.6×10^{-6}
Cs-137	4.0×10^{-11}	1×10^{-5}	4.0×10^{-6}	0.5	8.0×10^{-6}
I-131	2.4×10^{-11}	1×10^{-5}	2.4×10^{-6}	0.5	4.8×10^{-4}
Σ	7.08×10^{-9}	--	2.20×10^{-4}	0.5	4.4×10^{-4}

*Use ten times the smaller value of the soluble(s) or insoluble (I) EC values given in 10 CFR 20, Appendix B, Table 2, Column 2.

The fraction of unrestricted area EC from a continuous release (Section 2.3.2) is calculated in the same manner as the batch release shown above.

3. **Determination of Cumulative Dose from Radioactive Liquid Effluents (Section 2.4).**

The dose or dose commitment to a member of the public from radioactive liquid effluent shall be calculated on a cumulative quarterly and cumulative annual basis at least once per 31 days.

The dose or dose commitment from radioactive liquid releases at Turkey Point is based on the irradiation of a child on the canal shoreline, the most restrictive age group and is calculated using equation 7.

$$D = 0.23 \sum_{k} \sum_{i} A_i^{shoreline} \cdot \frac{C_{ik} \cdot F_{ik} \cdot t_k}{V \cdot \lambda_i^e} \qquad (7)$$

where:

D = total body or organ dose due to irradiation by radionuclides on the shoreline which originated in a liquid effluent release, (mrem).

0.23 = units conversion constant = $\frac{1\ \text{Ci}}{10^6\ \mu\text{Ci}} \times 60\ \frac{\text{min}}{\text{hr}} \times 3785\ \frac{\text{ml}}{\text{gal}}$

A_i = transfer factor relating a unit aqueous concentration of radionuclide i (μCi) to dose commitment rate to specific organs and the total body of an exposed person tabulated in Appendix A, (mrem/Ci . min/gal).

C_{ik} = the concentration of radionuclide in the undiluted liquid waste to be discharged that is represented by sample k, (gal/min).

V = cooling canal effective volume, approximately 3.75×10^9 gallons.

t_k = period of time (hours) during which liquid waste represented by sample k is discharged.

λ_i^e = effective decay constant ($\lambda_i + F_3/V$, minute $^{-1}$).

where:

λ_i = the radioactive decay constant

F_3 = canal-ground water interchange flow, approximately 2.25×10^5 gal/min

Example:

The concentration of radionuclides in liquid waste discharges to the condenser cooling water mixing basin during the month of February was determined by summing the results of the radionuclide analysis sheets for each sample taken prior to the release. The total concentration of each radionuclide was:

Radionuclide	C_{ik} (μCi/mL)
Co-60	4×10^{-4}
Co-58	1×10^{-5}
Cr-51	4×10^{-6}
Cs-134	5×10^{-6}
Cs-137	2×10^{-6}
Mn-54	2×10^{-5}
I-131	1×10^{-6}

The average flow rate from the monitor tanks during the releases (F_{lk}) = 100 gpm.

The total period of time for the releases (t_k) was 15 hours.

The cumulative whole body dose to a child due to these releases is determined by summing the dose from each radionuclide as shown in the Example 3 Work sheet.

<u>EXAMPLE 3</u>

WORKSHEET FOR DOSE TO WHOLE BODY FROM LIQUID RELEASE

Radio-nuclide	C_{ik}	A_i	F_{lk}	t_t	$0.23A_i \cdot C_{ik} \cdot F_{lk} \cdot t_t$	λ_i	F_3/V	λ^e_i	$V\lambda^e_i$	D
Co-60	4E-4	9.45E+3	100	15	1.30E+3	2.53E-7	6.0E-5	6.02E-5	2.26E+5	5.8E-3
Co-58	1E-5	1.67E+2	100	15	5.76E-1	6.80E-6	6.0E-5	6.68E-5	2.50E+5	2.3E-6
Cr-51	4E-6	2.06E+0	100	15	2.84E-3	1.74E-5	6.0E-5	7.74E-5	2.90E+5	9.8E-9
Cs-134	5E-6	3.08E+3	100	15	5.31E+0	6.39E-7	6.0E-5	6.06E-5	2.27E+5	2.3E-5
Cs-137	2E-6	4.54E+3	100	15	3.13E+0	4.37E-8	6.0E-5	6.00E-5	2.25E+5	1.4E-5
Mn-54	2E-5	6.09E+2	100	15	4.20E+0	1.54E-6	6.0E-5	6.15E-5	2.31E+5	1.8E-5
I-131	1E-6	7.59E+0	100	15	2.62E-3	5.98E-5	6.0E-5	1.19E-4	4.46E+5	5.9E-9
ΣD =										5.9E-3

Total whole body dose to child from irradiation by radionuclides on the shoreline from radioactivity released in month of February is 5.9E-3 mrem. Cumulative dose for first quarter would be sum of January dose + February dose. Cumulative annual dose in this example would be the same as the quarterly dose.

In this case the organ dose is the same as the whole body dose since the dose transfer factors for direct radiation is the same.

4. Determination of the Projected Dose (Section 2.5)

The dose, to unrestricted areas, from liquid effluent must be projected at least once per 31 days when the liquid radwaste treatment system is not being fully utilized. The dose projection can be made using equation (8).

$$P = \frac{31 \cdot D}{X} \tag{8}$$

where:

P = the projected total body or organ dose during the month (mrem)

31 = number of days in a calendar month, (days)

X = number of days in current month to date represented by available radioactive effluent sample, (days)

D = total body or organ dose to date during current month calculated according to section 2.4, (mrem).

Example:

The whole body dose calculated as of March 15 was 7.5×10^{-2} mrem.

The projected dose for the 31 day period in March would be:

$$P = \frac{31\ x\ D}{15} = \frac{31\ x\ 7.5\ x\ 10^{-2}\ mrem}{15} = 1.55\ x\ 10^{-1}\ mrem$$

Thus, in accordance with Technical Specification 3.11.1.3, appropriate portions of the liquid radwaste treatment system must be used to reduce releases of radioactivity since the dose from each unit would exceed 0.06 mrem.

5. Liquid Radwaste Effluent Monitor Alarm Setpoint (Section 2.6.1).

The monitor alarm setpoint for liquid batch releases is based on the fraction of the unrestricted area EC (FEC) that will be present in the condenser cooling water mixing basin as a result of the activity concentration present in the liquid radwaste to be released.

The monitor setpoint can be determined using equation (9) for batch and continuous releases respectively.

Example:

$$S_b = \frac{A_b \cdot S_f}{FEC_b} \cdot g_b + Bkg \tag{9}$$

where:

S_b = radiation monitor alarm setpoint for a batch release, (cpm)

A_b = laboratory counting rate (cpm/ml) or activity concentration (μCi/ml) of sample from batch tank

FEC_b = fraction of unrestricted area EC present in the condenser cooling water mixing basin outflow due to a batch release; determined in section 2.3.1.

g_b = detection efficiency of monitor detector; ratio of effluent radiation monitor counting rate to laboratory counting rate or activity concentration in a given batch sample (cpm per cpm/ml or cpm per μCi/ml which ever units are consistent with the units A_b).

Bkg = background, (cpm)

S_f = A factor to allow for multiple sources from different or common release points. The allowable operating setpoints will be controlled administratively by assigning a fraction of the total allowable release to each of the release sources.

Determine the monitor setpoint when:

FEC_b = 6×10^{-4}

A_b = 8.85×10^{-5} µCi/ml

g_b = 15,000 cpm/µCi/ml

S_f = .8

Bkg = 10,000 cpm

$$S_b = \frac{8.85 \times 10^{-5} \times .8}{6 \times 10^{-4}} \times 1.5 \times 10^{4} + 1 \times 10^{4} = 11{,}770 \ cpm$$

6. **Determining the Total Body Dose Rate from Noble Gas (Section 3.3.1).**

The total body dose rate from the radioactive noble gases may be calculated at any location by assuming a person is immersed in and irradiated by a semi-infinite cloud of the noble gases. Compliance is assessed and alarm setpoints established based on the dose rate at the site boundary where the minimum atmospheric dispersion from the plant occurs. This location is 1950 meters SSE of the plant where $\chi/Q = 5.8 \times 10^{-7}$ sec/m^3. The dose rate D may be calculated using equation (13).

Example:

During a 31 day period the following noble gas activity was released from Unit 3. The total body dose rate is calculated by:

$$\dot{D}_{TB} = \frac{\chi}{Q} \cdot \frac{1}{t} \sum_{i} Q_i \cdot P_{\gamma i} \qquad (13)$$

where:

$\dot{D}_{TB}$ = Dose rate to total body from noble gases, (mrem/year)

χ/Q = atmospheric dispersion factor at the off-site location of interest, (sec/m^3)

t = Averaging time of release, i.e., increment of time during which Q_i was released, (year)

Q_i = quantity of noble gas radionuclide i released during the averaging time, (μCi)

$P_{\gamma i}$ = factor converting time integrated concentration of noble gas radionuclide i at ground level, to total body dose, $\left(\frac{\text{mrem}}{\mu\text{Ci}\cdot\text{sec/m}^3}\right)$; See Reference Table 3-4

The total body dose is summarized in the following table:

Radionuclide	Q_i	$P_{\gamma i}$	$Q_i P_{\gamma i}$
Kr-85m	3.6E-2	3.7E-5	1.33E-6
Kr-85	2.8E-1	5.1E-7	1.43E-7
Kr-87	2.5E-3	1.9E-4	4.75E-7
Kr-88	1.4E-2	4.7E-4	6.58E-6
Xe-131m	1.0E+1	2.9E-6	2.90E-5
Xe-133	4.3E+1	9.3E-6	4.00E-4
Xe-135	6.0E-1	5.7E-5	3.42E-5
Ar-41	7.7E-2	2.6E-4	2.00E-5

The value of $\Sigma Q_i P_{\gamma i}$ is equal to 4.92 E-4

D = 5.8 E-7 x 11.77 x 4.94 E-4 = 3.36 E-9 mRem/yr

Note: The time (t) is for 31 day period stated as years which equals 31d/365d/yr or 0.085 yr. The value in the table, 1/t is 1/0.085 = 11.77.

7. Determination of Skin Dose Rate from Noble Gases (Section 3.3.2)

The skin dose rate from radioactive noble gases may be calculated at any location in a manner similar to example 3.3.1 using Equation (14).

Example:

Using the noble gas release data given in Example 3.3.1 the skin dose rate is calculated by:

$$\dot{D}_S = \frac{\chi}{Q} \cdot \frac{1}{t}\left[\sum_i Q_i \cdot S\beta_i + 1.11 Q_i \cdot A_{\gamma i}\right] \tag{14}$$

where:

$\dot{D}_S$ = dose rate to skin from radioactive noble gases (mrem/year)

$S\beta_i$ = factor converting time integrated concentration of noble gas radionuclide i at ground-level, to skin dose from beta radiation, $\left(\frac{\text{mrem}}{\mu\text{Ci}\cdot\text{sec/m}^3}\right)$; Reference Table 3-4

1.11 = ratio of tissue dose equivalent to air dose in a radiation field, (mrem/mrad).

$A_{\gamma i}$ = factor for converting time integrated concentration of noble gas radionuclide i in a semi-infinite cloud, to air dose from its gamma radiation, $\left(\frac{\text{mrad}}{\mu\text{Ci}\cdot\text{sec/m}^3}\right)$; Listed in Table 3-3

The skin dose rate is summarized in the following table:

Nuclide	Q_i	$S\beta_i$	$Q_iS\beta i$	$A_{\gamma i}$	$Q_iA_{\gamma i}$
Kr-85m	3.6E-2	4.6E-5	1.7E-6	3.9E-5	1.40E-6
Kr-85	2.8E-1	4.2E-5	1.2E-5	2.0E-3	5.60E-4
Kr-87	2.5E-3	3.1E-4	7.8E-7	2.0E-4	5.00E-7
Kr-88	1.4E-2	7.5E-5	1.1E-6	4.8E-4	6.72E-6
Xe-131m	1.0E+1	1.5E-5	1.5E-4	4.9E-6	4.90E-5
Xe-133	4.3E-1	9.7E-6	4.2E-6	1.1E-5	4.73E-6
Xe-135	6.0E-1	5.9E-5	3.5E-5	6.1E-5	3.66E-5
Ar-41	7.6E-2	8.5E-5	6.5E-6	2.9E-4	2.20E-5

The value of $\Sigma Q_iS\beta_i$ = 2.11 E-4 and the value of $\Sigma Q_iA_{\gamma i}$ = 5.9 E-4

D =5.8E-7 x 11.77 (2.11E-4 + [1.11 x 5.9E-4]) = 5.58E-9 mrem/yr

Note: The value 1/t is 11.77 (see Example 6 table note), and X/Q is 5.8E-7 sec/m^3

8. Determining Dose Rate from Tritium, Iodines, and Particulates (Section 3.3.3)

The total body and/or organ dose rate due to tritium, radioiodines, and radioactive particulates with half-lives greater than 8 days released in the effluent air may be calculated at any location off-site using equation (15).

For assessing compliance with Technical Specification 3.11.2.1, the thyroid dose rate for a hypothetical infant located at the site boundary where the minimum atmospheric dispersion from the plant occurs is the assumed receptor.

Example:

During a calendar quarter (2184 hrs) the following activities were released from Unit 4. The dose rate from activity is calculated by:

$$D_{anp} = \frac{1}{3600t} \cdot \frac{\chi d}{Q} \sum_{k} \sum_{i} Q_{ik} \cdot TA_{anip} \qquad (15)$$

where:

D_{anp} = dose equivalent rate to body organ n of a person in age group a exposed via pathway p to radionuclide i identified in analysis k of effluent air, (mrem/year)

3600 = conversion constant, (sec/hr)

t = period of time over which the effluent releases are averaged, (2184 hrs/qtr)

χ_d/Q = quantity of radionuclide i released during time increment t based on analysis k, (μCi).

Q_{ik} = quantity of radionuclide i released during increment time t based on analysis k (uCi).

TA_{anip} = a factor relating the airborne concentration time integral of radionuclide i to the dose equivalent to organ n of a person in age group a exposed via pathway p,

$\left(\frac{\text{mrem/yr}}{\mu\text{Ci/m}^3}\right)$; See Appendix A

The dose rate from tritium; iodine and particulate is summarized in the following table.

Radionuclide	Q_{ik}	TA_{anip}	$Q_{ik}TA_{anip}$
H-3	1.6E+5	2.37E+3	3.79E+8
Cr-51	8.0E-6	1.8E+4	1.44E-1
Co-58	5.0E-7	0	0
Co-60	9.5E-7	0	0
I-131	3.5E-7	9.94E+11	3.48E+5
Cs-137	2.0E-6	0	0

Notes: The time factor 1/3600t = 1.27E-7 where t = 2184hrs/qtr
The value of $\Sigma Q_{ik}TA_{anip}$ = 3.8E+8
The value of X_d/Q = 5.8E-7

D_{anp} = 1.27E-7 x 5.8E-7 x 3.8E+8 = 2.8E-5 mrem/yr

9. Determining the Noble Gas Gamma Radiation Dose (Section 3.4.1)

The cumulative dose due to gamma radiation from radioactive noble gases discharged from the plant shall be calculated once per 31 days to verify the quarterly and annual limits will not be exceeded.

The gamma radiation dose from noble gases are calculated at the site boundary where the minimum atmospheric dispersion occurs, i.e., 1950 meters SSE of the plant where $\chi/Q = 5.8 \times 10^{-7}$ sec/m^3. The gamma dose is calculated using equation (16) or (17). The example given here uses equation (17).

Example:

The noble gas activity discharged during a 31 day period from gas decay tanks, containment purges, and the spent fuel pit vent were totaled as tabulated below. The gamma dose from the noble gas release is calculated as follows:

$$D_\gamma = \frac{\chi}{Q} \sum_j \sum_i Q_j \cdot f_i \cdot A_{\gamma i} \qquad (17)$$

where:

D_γ = The noble gas dose to air due to a mixed mode release (mrad).

χ/Q = The atmospheric dispersion factor for a mixed-mode discharge, (sec/m^3).

Q_j = The measured radioactivity released via stack or vent during a single counting interval, j (μCi).

f_i = The fraction of radionuclide i released in a given effluent stream.

$A_{\gamma i}$ = Factor converting time integrated, ground-level concentration of noble gas radionuclide i to air dose from gamma radiation listed in Table 3-3,

$$\left(\frac{\text{mrad}}{\mu\text{Ci} \cdot \text{sec/m}^3}\right)$$

The noble gas gamma radiation dose is summarized in the following table.

Radio-nuclide	Q_j	f_i	$A_{\gamma i}$	$Q_j f_i A_{\gamma i}$	χ/Q
Kr-85m	5.4E+1	6.7E-4	3.9E-5	1.4E-6	5.8E-7
Kr-85	5.4E+1	5.2E-3	5.4E-7	1.5E-7	5.8E-7
Kr-87	5.4E+1	4.6E-5	2.0E-4	5.0E-7	5.8E-7
Kr-88	5.4E+1	2.6E-4	4.8E-4	6.7E-6	5.8E-7
Xe-131m	5.4E+1	1.8E-1	4.9E-6	4.8E-5	5.8E-7
Xe-133	5.4E+1	8.0E-1	1.1E-5	4.8E-4	5.8E-7
Xe-135	5.4E+1	1.1E-2	6.1E-5	3.6E-5	5.8E-7
Ar-41	5.4E+1	1.4E-3	2.9-4	2.2E-8	5.8E-7

The value of $\Sigma Q_j f_i A_{\gamma i}$ = 5.95E-4

D_γ = 5.95E-4 x 5.8E-7 = 3.45-10

10. Determining Noble Gas Beta Radiation Dose (Section 3.4.2)

The beta air dose due to noble gases discharged from the plant shall be determined for the current calendar quarter and current calendar year at least once per 31 days. The beta air dose is calculated in the same manner as the gamma air dose in Sections 3.4.1 above using the effective beta air dose factor from Table 3-3 and Equation (18).

11. **Determining Dose Due to Iodine, Tritium, and Particulates (Section 3.5.2)**

Dose estimate should account for exposure of a person via the following pathways involving deposition of radioactivity on the ground.

- direct radiation from airborne radionuclides except noble gases
- inhalation
- direct radiation from ground plane deposition
- fruits and vegetables
- air-grass-cow-meat
- air-grass-cow-milk

The requirement to determine the dose commitments due to radioiodine, tritium, and radioactive particulates once per 31 days may be satisfied by using Equations (21), (22), (23), and (24).

Example:

The organ and total body dose to an infant from tritium inhalation and irradiation pathways and from radioiodines and particulates via the grass-cow-milk pathway is calculated using Equations 22 and 23. The major non-noble gas activities released over a 31 day period were used for the calculation. The atmospheric dispersion factor, χ_d/Q and deposition rate, D/Q values for a mixed mode release at 3.6 miles WNW and 4.5 miles west of the plant respectively were obtained from Tables 3-7 and 3-8. Factors TA_{anip} and TG_{anip} converting airborne activity to dose commitment are obtained from Appendix A for the organ, age group, and pathway.

$$D_{ank} = 3.17 \times 10^{-8} \frac{\chi_d}{Q} \sum_i Q_{ik} \sum_P TA_{anip} \tag{22}$$

$$D_{ank} = 3.17 \times 10^{-8} \frac{D}{Q} \sum_i Q_{ik} \sum_P TG_{anip} \tag{23}$$

where:

$\frac{\chi_d}{Q}$ = atmospheric dispension factor for a mixed mode release, adjusted for depletion by deposition, (sec/m3).

$\frac{D}{Q}$ = relative deposition rate onto ground from a mixed mode atmospheric release (m^{-2}).

Q_{ik} = the quantity of radionuclide i released in a given effluent stream based on analysis k, (μCi).

TA_{anip} = a factor converting airborne concentration of radionuclide i to a dose commitment to organ n of a person in age group a where exposure is directly due to airborne material via pathway P (inhalation or external exposure to the plume), $\left(\frac{mrem/yr}{\mu Ci/m^3}\right)$.

TG_{anip} = factor converting ground deposition of radionuclide i to dose commitment to organ n of a person in age group a where exposure is due to radioactive material via pathway P (direct radiation from ground plane deposition, fruits and vegateables, air-grass-cow-meat, or air-grass-cow-milk)$\left(\frac{mrem/yr}{\mu Ci/m^2 \bullet sec}\right)$.

D_{ank} = the dose commitment to organ n of a person in age group a due to radionuclides identified in analysis k of an air effluent, (mrem).

The organ and total body dose to an infant from radioiodines and particulates via the grass-cow-milk pathway is shown in the Example 10 Worksheet.

EXAMPLE 10 WORKSHEET - PAGE 1

GRASS-COW-MILK PATHWAY						
Organ Radio-nuclide	Q_{ik}	TA_{anip} or TG_{anip}	χ_d/Q or D/Q	3.17E-8	D_{ank}	Total Dose Sum of D_{ank} (mrem)
Bone						
H-3	2.0E+8	0			0	
Co-58	2.0E+1	0			0	
Co-60	1.7E+1	0			0	
I-131	3.9E+3	2.59E+9	5E-10	3.17E-8	1.6E-4	
Cs-137	6.1E+1	6.44E+10	5E-10	3.17E-8	6.2E-5	2.2E-4
Liver						
H-3	2.0E+8	2.37E+3	1E-7	3.17E-8	1.5E-3	
Co-58	2.0E+1	2.55E+7	5E-10	3.17E-8	8.1E-9	
Co-60	1.7E+1	8.73E+7	5E-10	3.17E-8	2.4E-8	
I-131	3.9E+3	3.09E+9	5E-10	3.17E-8	1.9E-4	
Cs-137	6.1E+1	7.21E+10	5E-10	3.17E-8	7.0E-5	1.8E-3
Thyroid						
H-3	2.0E+8	2.37E+3	1E-7	3.17E-8	1.5E-3	
Co-58	2.0E+1	0				
Co-60	1.7E+1	0				
I-131	3.9E+3	9.94E-11	5E-10	3.17E-8	6.1E-1	
Cs-137	6.1E+1	0				6.1E-1

EXAMPLE 10 WORKSHEET - PAGE 2

GRASS-COW-MILK PATHWAY						
Organ Radio-nuclide	Q_{ik}	TA_{anip} or TG_{anip}	χ_d/Q or D/Q	3.17E-8	D_{ank}	Total Dose Sum of D_{ank} (mrem)
Kidney						
H-3	2.0E+8	1.04E+3	1E-7	3.17E-8	6.6E-4	
Co-58	2.0E+1	0				
Co-60	1.7E+1	0				
I-131	3.9E+3	7.74E+8	5E-10	3.17E-8	4.8E-5	
Cs-137	6.1E+1	3.66E+9	5E-10	3.17E-8	3.5E-6	7.1E-4
Lung						
H-3	2.0E+8	2.37E+3	1E-7	3.17E-8	1.5E-3	
Co-58	2.0E+1	0				
Co-60	1.7E+1	0				
I-131	3.9E+3	0				
Cs-137	6.1E+1	8.69E+9	5E-10	3.17E-8	8.4E-6	1.5E-3
GI/LI						
H-3	2.0E+8	2.37E+3	1E-7	3.17E-8	1.5E-3	
Co-58	2.0E+1	6.6E+7				
Co-60	1.7E+3	2.16E+8				
I-131	3.9E+3	1.36E+8				
Cs-137	6.1E+1	1.86E+8	5E-10	3.17E-8	1.83E-7	1.5E-3

EXAMPLE 10 WORKSHEET - PAGE 3

GRASS-COW-MILK PATHWAY						
Organ Radio-nuclide	Q_{ik}	TA_{anip} or TG_{anip}	χ_d/Q or D/Q	3.17E-8	D_{ank}	Total Dose Sum of D_{ank} (mrem)
Total Body						
H-3	2.0E+8	2.37E+3	1E-7	3.17E-8	1.5E-3	
Co-58	2.0E+1	6.24E+7				
Co-60	1.7E+3	2.09E+8				
I-131	3.8E+3	1.81E+9				
Cs-137	6.1E+1	4.14E+9	5E-7	3.17E-8	4.0E-6	1.5E-3
Skin						
H-3	2.0E+8	0			0	
Co-58	2.0E+1	0			0	
Co-60	1.7E+1	0			0	
I-131	3.9E+3	0			0	
Cs-137	6.1E+1	0			0	0

12. Determining the Noble Gas Monitor Alarm Setpoint (Section 3.6)

Standard Technical Specifications require release setpoints to be based on a dose rate. Derivations used to determine setpoints assume that noble gas releases occur at ground level. The noble gas affluent monitor setpoint, based on dose rate is calculated using Equation (26).

$$S = 1.06\left[\frac{h \cdot S_f}{f \cdot \chi/Q}\right]\left[\frac{\sum_i C_i}{\sum_i C_i \cdot DF_i}\right] + Bkg \qquad (26)$$

where:

S = The alarm setpoint (CPM).

1.06 = Conversion factor;

$$\left(500\ \frac{\text{mrem}}{\text{yr}} \bullet \frac{60\ \text{sec}}{\text{min}} \bullet 35.37\ \frac{\text{ft}^3}{\text{m}^3} \bullet \frac{1\text{m}^3}{10^6\text{cm}^3}\right).$$

h = Monitor response to activity concentration of effluent $\left(\frac{\text{cpm}}{\mu\text{Ci/CM}^3}\right)$.

f = Flow of gaseous effluent stream past the monitor $\left(\frac{\text{ft}^3}{\text{min}}\right)$.

χ/Q = atmospheric dispersion factor at the offsite location of interest $\left(\frac{\text{sec}}{\text{m}^3}\right)$.

S_f = a factor to allow for multiple sources from different or common release points. The allowable operating setpoints will be controlled by assigning a fraction of the allowable release to each of the release sources.

DF_i = factor converting ground-level or split-wake release of radionuclide i to the total body dose equivalent rate at the location of potential exposure $\left(\frac{\text{mrem}}{\text{yr.}\mu\text{Ci/m}^3}\right)$; see Table 3-5.

C_i = concentration of radionuclide i in gaseous effluent (μCi/cc).

Bkg = monitoring instrument background (cpm).

Example:

The measured concentration of noble gases to be discharged to the atmosphere are:

Radionuclide	C_i (μCi/cc)
Kr-85m	3.6×10^{-5}
Kr-85	2.8×10^{-4}
Kr-87	2.5×10^{-6}
Kr-88	1.4×10^{-5}
Xe-131m	1.0×10^{-2}
Xe-133	4.3×10^{-2}
Xe-135	6.0×10^{-4}
Ar-41	7.7×10^{-5}

Determine the alarm setpoint, S (cpm) when:

$h = 3.0 \times 10^{-3} \dfrac{\text{cpm}}{\mu\text{Ci/cm}^3}$

$f = 1.7 \times 10^{4} \dfrac{\text{ft}^3}{\text{min}}$

$\chi/Q = 5.8 \times 10^{-7} \dfrac{\text{sec}}{\text{m}^3}$ (Note: This is the value at the point of minimum atmospheric dispersion which occurs at 1950 meters SSE of the plant).

$S_f = .25$

Bkg = 20 cpm

Calculate the effect of a ground level release as follows:

Radionuclide	C_i	DF_i	C_i x DF_i
Kr-85m	3.6×10^{-5}	1.17×10^{3}	4.2×10^{-2}
Kr-85	2.8×10^{-4}	1.61×10^{1}	4.5×10^{-3}
Kr-87	2.5×10^{-6}	5.92×10^{3}	1.5×10^{-2}
Kr-88	1.4×10^{-5}	1.47×10^{4}	2.1×10^{-1}
Xe-131m	1.0×10^{-2}	9.15×10^{1}	9.1×10^{-1}
Xe-133	4.3×10^{-2}	2.94×10^{2}	1.3×10^{1}
Xe-135	6.0×10^{-4}	1.81×10^{3}	1.1×10^{0}
Ar-41	7.7×10^{-5}	8.85×10^{3}	6.8×10^{-1}

$\Sigma C_i = 5.4 \times 10^{-2}$ $\Sigma C_i DF_i = 1.6 \times 10^{1}$

Calculate the setpoint as follows:

$$S = 1.06 \left[\frac{3.0 \times 10^{3} \cdot .25}{1.7 \times 10^{4} \cdot 5.8 \times 10^{-7}}\right]\left[\frac{5.4 \times 10^{-2}}{1.6 \times 10^{1}}\right] + 20$$

$= [8.06 \times 10^{4}]\,[3.4 \times 10^{-3}] + 20$

$= 294.0$ cpm

APPENDIX E

RADIOACTIVE EFFLUENT TECHNICAL SPECIFICATIONS

APPENDIX E

This Appendix contains all the radioactive effluent technical specifications and specification tables referenced in the Turkey Point Offsite Dose Calculation Manual (ODCM).

SECTION 1.0

DEFINITIONS

FREQUENCY NOTATION

1.12 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GAS DECAY TANK SYSTEM

1.13 A GAS DECAY TANK SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant system off gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MEMBER(S) OF THE PUBLIC

1.15 MEMBER(S) OF THE PUBLIC shall mean individual(s) in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual recieves an occupational dose.

OFFSITE DOSE CALCULATION MANUAL

1.16 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.18 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PURGE - PURGING

1.22 PURGE or PURGING shall be any 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

1.27 The SITE BOUNDARY shall mean that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

UNRESTRICTED AREA

1.34 An UNRESTRICTED AREA shall mean an area, access to which is neither limited nor controlled by the licensee.

VENTILATION EXHAUST TREATMENT SYSTEM

1.35 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

VENTING

1.36 VENTING shall be 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 not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

TABLE 1.2

OPERATIONAL MODES

MODE		REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1.	POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2.	STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3.	HOT STANDBY	< 0.99	0	$\geq$ 350°F
4.	HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5.	COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6.	REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

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* Excluding decay heat.

** Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS
CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS (See Figure 5.1-1) shall be limited to 10 times the concentrations specified in 10 CFR Part 20, Appendix B Table 2, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:
With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 4.11-1.

4.11.1.1.2 The results of the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

LIQUID RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)[1] ($\mu Ci/ml$)
1. Batch Waste Release Tanks[2]	P Each Batch	P Each Batch	Principal Gamma Emitters[3]	$5x10^{-7}$
			I-131	$1x10^{-6}$
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	P Each Batch	M Composite[4]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	P Each Batch	Q Composite[4]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
2. Continuous Releases[5]	W	W	Principal Gamma Emitters[3]	$5x10^{-7}$
			I-131	$1x10^{-6}$
a. Steam Generator Blowdown[7]	M[8]	M[8]	Dissolved and Entrained Gases (Gamma Emitters)	$1x10^{-5}$
	W[8]	M[8] Composite[6]	H-3	$1x10^{-5}$
			Gross Alpha	$1x10^{-7}$
	W[8]	Q[8] Composite[6]	Sr-89, Sr-90	$5x10^{-8}$
			Fe-55	$1x10^{-6}$
b. Storm Drain	M	M	Principal Gamma Emitters[3]	$5x10^{-7}$
			I-131	$1x10^{-6}$

TABLE 4.11-1 (continued)

TABLE NOTATIONS

(1) The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66 s_b}{(E)(V)(2.22\ X\ 10^6)(y)[EXP(-\lambda \Delta t)]}$$

where:

LLD = the "a priori" lower limit of detection (as μCurie per unit mass or volume).

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute).

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

$2.22 \times 10^{+6}$ = the number of disintegrations per minute per μCurie,

y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide and

Δt = the elapsed time between the midpoint of sample collection and the time of counting (for plant effluents, not environmental samples).

The value of S_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. Typical values of E, V, Y and Δt should be used in the calculation.

(2) A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated,and then thoroughly mixed to by a method described in the ODCM to assure representative sampling.

(3) The principal gamma emitters for which the LLD specification exclusively applies are 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 considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4.

(4) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

(5) A continuous release is the discharge of liquid wastes of a nondiscreet volume, e.g., from a volume of a system that has an input flow during the continuous release.

(6) Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

(7) Sampling and analysis of steam generator blowdown is not required during Mode 5 or 6.

(8) Sampling and analysis of steam generator blowdown on the applicable unit is only necessary for these species when primary to secondary leakage is occurring as indicated by the condenser air ejector monitor. (See Specification 3.3.3.6 in Table 3.3-8, Item 3a).

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (See Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (See Figure 5.1-1) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Radwaste Treatment System not in operation, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Liquid Radwaste Treatment Systems are not being fully utilized.

4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

RADIOACTIVE EFFLUENTS

3/4.11.2 GASEOUS EFFLUENTS
DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY (See Figure 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131, for Iodine-133, for tritium and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:
With the dose rate(s) exceeding the above limits, immediately restore the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

GASEOUS RELEASE TYPE	SAMPLING FREQUENCY	MINIMUM ANALYSIS FREQUENCY	TYPE OF ACTIVITY ANALYSIS	LOWER LIMIT OF DETECTION (LLD)(1) (μCi/cc)
1. Gas Decay Tank (Batch)	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters(2)	1×10^{-4}
2. Containment Purge or Venting (Batch)	P(6) Grab Sample	P Each PURGE(6)	Principal Gamma Emitters(2)	1×10^{-4}
			H-3	1×10^{-6}
3. Condenser Air Ejectors	M(6) Grab Sample	M(6) Gas Sample	Principal Gamma Emitters(2)	1×10^{-4}
			H-3	1×10^{-6}
4. Plant Vent (Includes Unit 4 Spent Fuel Pit Building Vent.)	M(6) Grab Sample	M(6) Gas Sample	Principal Gamma Emitters(2)	1×10^{-4}
	M(4),(5) Grab Sample	M	H-3	1×10^{-6}
5. Unit 3 Spent Fuel Pit Building Vent	M Grab Sample	M Gas Sample	Principal Gamma Emitters(2)	1×10^{-4}
	M(4),(5) Grab Sample	M	H-3	1×10^{-6}
6. All Release Types as listed in 3., 4., and 5. above	Continuous(3)	W(7) Charcoal Sample	I-131	1×10^{-12}
	Continuous(3)	W(7) Particulate Sample	Principal Gamma Emitters(2)	1×10^{-11}
	Continuous(3)	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous(3)	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous(3)	Noble Gas Monitor	Noble Gas Gross Beta or Gamma	1×10^{-6}

TABLE 4.11-2 (Continued)
TABLE NOTATIONS

(1) The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\, s_b}{E \bullet V \bullet (2.22 x 10^6) \bullet Y \bullet [\exp(-\lambda \Delta t)]}$$

Where:

LLD	=	the "a priori" lower limit of detection as defined above as a blank sample (microCurie per unit mass or volume),
s_b	=	the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),
E	=	the counting efficiency (counts per disintegration)
V	=	the sample size (units of mass or volume),
2.22×10^6	=	the number of disintegrations per minute per microCurie,
Y	=	the fractional radiochemical yield, when applicable,
λ	=	the radioactive decay constant for the particular radionuclide, and
Δt	=	the elapsed time between the midpoint of sample collection and the time of counting (for plant effluents, not environmental samples)

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. Typical values of E, V, Y, and Δt shall be used in the calculation.

TABLE 4.11-2 (Continued)
TABLE NOTATIONS (Continued)

(2) The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other gamma peaks that are measurable and indentifiable, together with the above nuclides, shall also be identified and reported pursuant to Specification 6.9.1.4.

Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD for that nuclide. When a radionuclide's calculated LLD is greater than its listed LLD limit, the calculated LLD should be assigned as the activity of the radionuclide; or, the activity of the radionuclide should be calculated using measured ratios with those radionuclides which are routinely identified and measured.

(3) The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2, and 3.11.2.3.

(4) When a Unit's refueling canal is flooded Tritium grab samples shall be taken on that Unit only from the following respective area(s) at least once per 24 hours:

For Unit 3 sample the plant vent and the Unit 3 spent fuel pool area ventilation exhaust.

For Unit 4 sample the plant vent only.

(5) When spent fuel is in the spent fuel pool, tritium grab samples shall be taken from the following respective area at least once per 7 days:

For Unit 3, sample the Unit 3 spent fuel pool area ventilation exhaust

For Unit 4, sample the plant vent.

(6) Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased by more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3.

TABLE 4.11-2 (Continued)
TABLE NOTATIONS (Continued)

(7) Sample collection media on the applicable Unit shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sample collection media on the applicable Unit shall also be changed at least once per 24 hours for at least 7 days following each shutdown, startup, or THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period and analyses shall be completed within 48 hours of changing if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has increased more than a factor of 3. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

RADIOACTIVE EFFLUENTS

DOSE-NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY (see figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION:

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the SITE BOUNDARY (See Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131, Iodine-133 tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the release and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The VENTILATION EXHAUST TREATMENT SYSTEM and the GAS DECAY TANK SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of any inoperable equipment or subsystems, and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Gaseous Radwaste Treatment Systems are not being fully utilized.

4.11.2.4.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM and GAS DECAY TANK SYSTEM shall be considered OPERABLE by meeting Specifications 3.11.2.1 and 3.11.2.2 or 3.11.2.3.

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY At all times.

ACTION

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limit of Specification 3.11.1.2a, 3.11.1.2b, 3.11.2.2a, 3.11.2.2b, 3.11.2.3a or 3.11.2.3b, calculations shall be made including direct radiation contributions from the units to determine whether the above limits of Specifications 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.2203(a)(4), shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the releases(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the units and the methodology used shall be indicated in the Annual Radioactive Effluent Release Report. This requirement is applicable only under conditions set forth in ACTION a. of Specification 3.11.4.

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.3-7 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.1.1 are not exceeded. The Alarm/Trip Setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

APPLICABILITY At all times, except as indicated in Table 3.3-7.

ACTION

a. With a radioactive liquid effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive effluents monitored by the affected channel, or declare the channel inoperable, or change the setpoint so it is acceptably conservative,

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-7. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, explain in the next Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected in a timely manor.

c. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-5.

TABLE 3.3-7

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1.	Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
	a. Liquid Radwaste Effluent Line	1*	35
	b. Steam Generator Blowdown Effluent Line	1**	36
2.	Flow Rate Measurement Devices		
	a. Liquid Radwaste Effluent Line	1*	37
	b. Steam Generator Blowdown Effluent Line	1** steam/generator	37

* Applicable during liquid effluent releases.
** Applicable during blowdown operations.

ACTION 35 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Specification 4.11.1.1.1,. and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this path-way.

ACTION 36 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are analyzed for gross (beta or/gamma) radioactivity at a lower limit of detection of no more than 1×10^{-7} μCurie/ml; or analyzed isotopically (Gamma) at a limit of detection of at least 5×10^{-7} μCurie/ml:

a. At least once per 12 hours when the specific activity of the secondary coolant is greater than 0.01 μCurie/gram DOSE EQUIVALENT I-131, or

b. At least once per 24 hours when the specific activity of the secondary coolant is less than or equal to 0.01 μCurie/gram DOSE EQUIVALENT I-131.

ACTION 37 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves may be used to estimate flow.

TABLE 4.3-5
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Liquid Radwaste Effluents Line	D	P	R(2)*	Q(1)
b. Steam Generator Blowdown Effluent Line	D	M	R(2)	Q(1)
2. Flow Rate Measurement Devices				
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R*	Q
b. Steam Generator Blowdown Effluent Lines	D(3)	N.A.	R	Q

*Channel calibration frequency shall be at least once per 18 months.

TABLE NOTATIONS

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measures levels above the Alarm/Trip Setpoint.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

TABLE NOTATIONS

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the Alarm/Trip Setpiont.

(2) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(3) CHANNEL CHECK shall consist of verifying indication of flow during periods of release. CHANNEL CHECK shall be made at least once per 24 hours on days on which continuous, periodic, or batch releases are made.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.3-8 shall be OPERABLE with their Alarm/Trip Setpoints set to ensure that the limits of Specifications 3.11.2.1 and 3.1.2.5 are not exceeded. The Alarm/Trip Setpoints of these channels meeting Specification 3.11.2.1 shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

APPLICABILITY As shown in Table 3.3-8

ACTION

a. With a radioactive gaseous effluent monitoring instrumentation channel Alarm/Trip Setpoint less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable or change the setpoint so it is acceptably conservative,

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take ACTION shown in Table 3.3-8. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful explain in the next Annual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4 why this inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECKS, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-6.

TABLE 3.3-8
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. GAS DECAY TANK SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Plant Vent Monitor)	1	*	45
b. Effluent System Flow Rate Measuring Device	1	*	46
2. WASTE GAS DISPOSAL SYSTEM (Explosive Gas Monitoring System)			
a. Hydrogen and Oxygen Monitors	1	**	49
3. CONDENSER AIR EJECTOR VENT SYSTEM			
a. Noble Gas Activity Monitor (SPING or PRMS).	1	#	47
b. Iodine Sampler	1	##	48
c. Particulate Sampler	1	##	48
d. Effluent System Flow Rate Measuring Device	1	##	46
e. Sampler Flow Rate Measuring Device	1	##	46

TABLE 3.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
4. Plant Vent System (Include Unit 4's Spent Fuel Pool)			
a. Noble Gas Activity Monitor (SPING or PRMS)	1	*	47
b. Iodine Sampler	1	*	48
c. Particulate Sampler	1	*	48
d. Effluent System Flow Rate Measuring Device	1	*	46
e. Sampler Flow Rate Measuring Device	1	*	46
5. Unit 3 Spent Fuel Pit Building Vent			
a. Noble Gas Activity Monitor	1	*	47
b. Iodine Sampler	1	*	48
c. Particulate Sampler	1	*	48
d. Sampler Flow Rate Measuring Device	1	*	46

TABLE 3.3-8 (Continued)
TABLE NOTATION

* - At all times.
** - During GAS DECAY TANK SYSTEM operation.
# - Applies during MODE 1, 2, 3 and 4.
- Applies during MODE 1, 2, 3 and 4 when primary to secondary leakage is detected as indicated by condenser air ejector noble gas activity monitor.

ACTION 45 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release:

a. At least two independent samples of the tank's contents are analyzed, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup;

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours.

ACTION 47 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity with 24 hours.

ACTION 48 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2 and analyzed at least weekly.

ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the GAS DECAY TANK SYSTEM may continue provided that grab samples are collected and analyzed for hydrogen and oxygen concentration at least a) once per 8 hours during degassing operations, and b) once per day during other operations.

TABLE 4.3-6

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. GAS DECAY TANK SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Plant Vent Monitor)	P	P	R(3)	Q(1)	*
b. Effluent System Flow Rate Measuring Device	P	N.A.	R	N.A.	*
2. GAS DECAY TANK SYSTEM (Explosive Gas Monitoring System)					
a. Hydrogen and Oxygen Monitors	D	N.A.	Q(4,5)	M	**
3. CONDENSER AIR EJECTOR VENT SYSTEM					
a. Noble Gas Activity Monitor (SPING or PRMS)	D	M	R(3)	Q(2)	#
b. Iodine Sampler	W	N.A.	N.A.	N.A.	##
c. Particulate Sampler	W	N.A.	N.A.	N.A.	##
d. Effluent System Flow Rate Measuring Device	D	N.A.	R	N.A.	##

TABLE 4.3-6 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Condenser Air Ejector Vent System (Continued)					
e. Sample Flow Rate Measuring Device	D	N.A.	R	N.A.	##
4. Plant Vent System (Include Unit 4's Spent Fuel Pool)					
a. Noble Gas Activity Monitor (SPING or PRMS)	D	M	(3,6)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Measuring Device	D	N.A.	(6)	N.A.	*
e. Sampler Flow Rate Measuring Device	D	N.A.	(6)	N.A.	*

TABLE 4.3-6 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Unit 3 Spent Fuel Pit Building Vent					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Measuring Device	D	N.A.	R	N.A.	*

TABLE NOTATION

* At all times.

** During GAS DECAY TANK SYSTEM operation.

\# Applies during MODE 1,2,3 and 4.

\## Applies during MODE 1,2,3 and 4 when primary to secondary leakage is detected as indicated by condenser air ejector noble gas activity monitor.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if the instrument indicates measured levels above the Alarm/Trip Setpoint.

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that if the instrument indicates measured levels above the Alarm Setpoint, alarm annunciation occurs in the control room (for PRMS only) and in the computer room (for SPING only).

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

TABLE 4.3-6 (Continued)

TABLE NOTATIONS (Continued)

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal.

a. One volume percent hydrogen, balance nitrogen, and

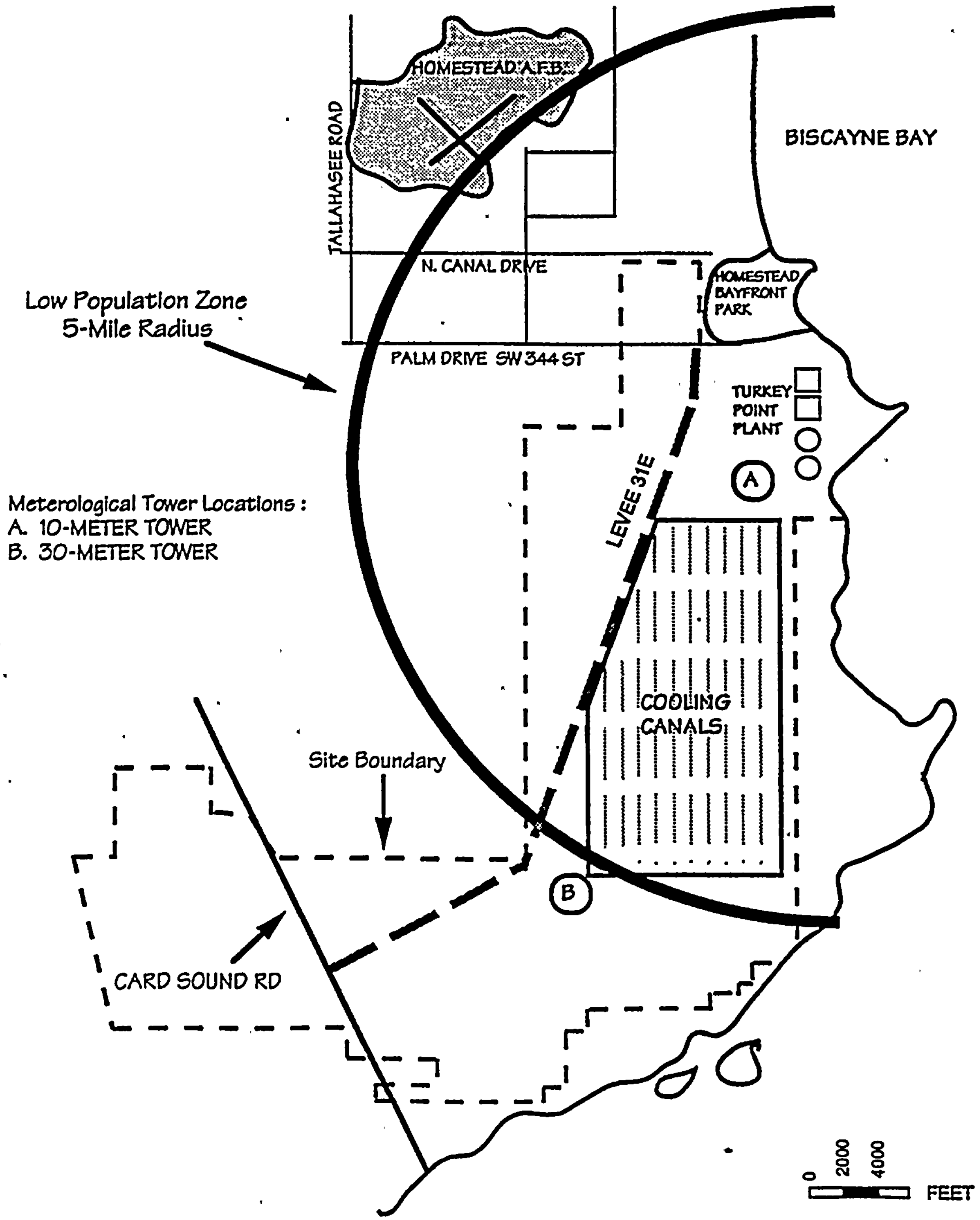
b. Four volume percent hydrogen, balance nitrogen.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

(6) CHANNEL CALIBRATION frequency shall be at least once per 18 months.



Site Area Map
Figure 5.1-1

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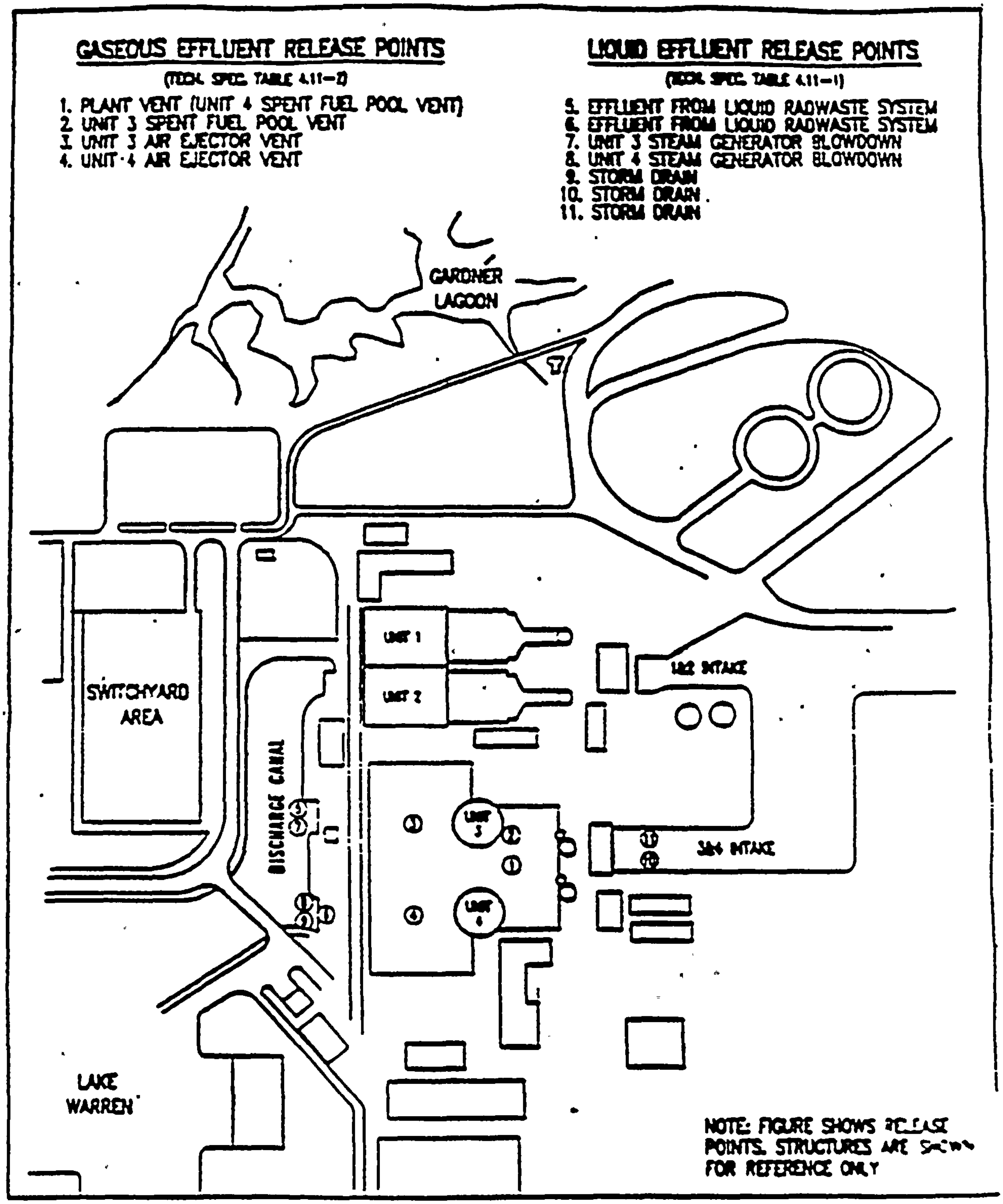

FIGURE 5.1-2 PLANT AREA MAP

TURKEY POINT

OFFSITE DOSE CALCULATION MANUAL

BASIS DOCUMENT

1.0 Introduction

The operation of a nuclear facility is regulated by requirements contained in the Code of Federal Regulations (CFR's). Section 50.36 of 10CFR50 requires that each nuclear power reactor operating license contain technical specifications that describe limits, operating conditions and other regulatory requirements imposed on the facility operation for protection of the health and safety of the public. At each site, conditions and limitations which are system dependent and site related must be incorporated in the technical specifications. These technical specifications are submitted to the Nuclear Regulatory Commission as part of the licensing process and upon approval are included in Appendix A of the facility operating license.

Technical specifications for a nuclear power plant require the operator to establish alarm and trip setpoints for each liquid and gaseous effluent release point. In addition, these setpoints must be maintained in auditable records and be determined in accordance with an Offsite Dose Calculation Manual (ODCM). The ODCM must also include the methodology and parameters used in the calculation of offsite doses due to radioactive liquid and gaseous effluents.

2.0 Offsite Dose Calculation Manual Methodology

The Nuclear Regulatory Commission (NRC) has developed dose calculation methodology which the NRC considers acceptable for use in the ODCM. The NRC guides are:

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.

Regulatory Guide 1.110 - "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors," March 1976.

Regulatory Guide 1.111 - "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors" (Revision 1), July 1977.

Regulatory Guide 1.112 - "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," April 1976.

Regulatory Guide 1.113 - "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I" (Revision 1), April 1977.

The NRC has also developed computer codes which may be used with these guides. The codes are:

NUREG-0017 - "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," April 1976.

NUREG-0324 - "XOQDOQ, Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations," September 1977.

NUREG-0133 - Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978.

Conformance with the NRC guidelines for dose calculation methodology is not required. However, if different mathematical models and parameters are used to calculate set points, release rates or dose estimates, the parameters and calculations used shall be substantiated in the ODCM.

The Turkey Point ODCM uses equations and models adopted from the methodology provided in the regulatory guides.

3.0 Definitions

The Technical Specifications contain terms which must be defined in order to clarify limits and the applicability of methodology employed in the ODCM. The terms and their definitions are as follows:

3.1 Frequency Notation

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in table 3.1.

3.2 Gas Decay Tank System

A GAS DECAY TANK SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

3.3 Identified Leakage

IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

3.4 Member(s) of the Public

MEMBER(S) OF THE PUBLIC shall mean individual(s) in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual recieves an occupational dose.

3.5 Offsite Dose Calculation Manual

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program.

3.6 Operable - Operability

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

3.7 Operational Mode - MODE

An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

3.8 Purge - Purging

PURGE or PURGING shall be any 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.

3.9 Site Boundary

The SITE BOUNDARY shall mean that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

3.10 Unrestricted Area

An UNRESTRICTED AREA shall mean an area, access to which is neither limited nor controlled by the licensee.

3.11 Ventilation Exhaust Treatment System

A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

3.12 Venting

VENTING shall be 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 not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

TABLE 3.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
NA	Not applicable.
P	Completed prior to each release.

TABLE 3.2

OPERATIONAL MODES

MODE		REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1.	POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2.	STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3.	HOT STANDBY	< 0.99	0	$\geq$ 350°F
4.	HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5.	COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6.	REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.
**Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

4.0 Liquid Radwaste Releases

Liquid radwaste from Turkey Point Nuclear Units 3 and 4 are discharged to the condenser cooling water mixing basin which in turn discharges to a closed cooling loop water canal system. Liquid radwaste releases may be discharged in batches from holding tanks, continuously from steam generator blowdown or through storm drains. Radwaste entering the mixing basin is mixed with and diluted by condenser cooling water from Fossil Units 1 and 2 and Nuclear Units 3 and 4 before being discharged to the canal.

At Turkey Point, all liquid radwaste are sampled and analyzed in accordance with Technical Specification Table 4.11-1. In addition, batch and continuous release are continuously monitored by in-line radiation monitors during release. Storm drain releases are not continuously monitored.

4.1 Technical Specifications for Liquid Effluents

The following Technical Specification requirements must be met when releasing radioactive liquid effluents and the methodology for calculating the specifications must be contained in the ODCM.

4.1.1 Liquid Effluent Concentrations

Technical Specification 3.11.1.1 requires that the radioactive concentration of liquid effluents discharged to the Unrestricted Area be limited to 10 times the radionuclide Effluent Concentrations (ECs) given in 10CFR20, Appendix B Table 2, Column 2 for radionuclides other than dissolved or entrained noble gases. A separate EC of 2×10^{-4} μCi/ml is given for noble gases dissolved or entrained in liquids.

For purposes of implementation at Turkey Point, the Unrestricted Area for liquid effluents begins where the water from the mixing basin enters the cooling canal.

The methodology for satisfying Technical Specification 3.11.1.1 is provided in ODCM equations 1 to 6. In addition to providing the required calculational techniques, some of these equations contain conservative factors to ensure that the requirements of Technical Specification 3.11.1.1 are not exceeded.

4.1.1.1 Diluted Radwaste Concentrations

Diluted radwaste concentrations are determined using ODCM equation 1:

$$C_{zi} = C_i \frac{F_1}{F_2} \qquad (1)$$

where:

C_{zi} = concentration of radionuclide i in the water in the condenser cooling water mixing basin outflow, (μCi/ml).

C_i = concentration of radionuclide i in liquid radwaste released, (μCi/ml).

F_1/F_2 = dilution.

F_1 = flow in radioactive liquid discharge line (gal/min).*

F_2 = total condenser cooling water flow, (gal/min).* Value not greater than the rated total condenser cooling water flow from operating condenser cooling water pumps at the four units.

* F_1 and F_2 may have any suitable but identical units of flow, (volume/time).

This equation is a simplification of the completely mixed model given in Regulatory Guide 1.113 for impoundments. As used at Turkey Point this equation provides conservative estimates of diluted radionuclide concentrations because:

- The volume of the mixing basin and the canal are not included in the total volume.

- The effects of radioactive decay are ignored.

In practice, the equation is used in the following way:

- To make pre-release estimates. These estimates are made using condenser cooling water flows from the nuclear units only as total flow (F_2) because cooling water flow for the fossil units is regulated by the Fossil Plant control room and may change during the period of release.

- To make post release estimates. These estimates are made using cooling water from both the fossil and the nuclear units as total flow (F_2). In general, post release estimates are less than or equal to pre-release estimates.

In addition to radioactive decay, ODCM equation 1 also ignores the rate of buildup of long-lived isotopes. Regulatory Guide 1.113 expresses this rate as:

$$C_o = \frac{W}{q_b}$$

where:

C_o = steady state concentration of non-decaying substances.

W = the rate of addition of radioactivity

q_b = pond blowdown or volume removal factor

Because of the closed nature of the cooling canal at Turkey Point, the only loss or removal factor (q_b) is evaporation which effects only volatile radionuclides and since, except for seasonal variations, the volume in the cooling canal remains relatively constant the rate of buildup maybe expressed as:

$$C_o = W$$

In other words the rate of buildup is equivalent to the steady state concentration at any point in time.

4.1.1.2 Effluent Concentrations (ECs)

Liquid radwaste activity release concentrations are determined by using ODCM equations 2 to 6. Equations 2, 3, 4 or 5 provide methods for determining the fractions of the EC in batch or continuous releases. Equation 6 provides a means of totalling the fractional ECs from all releases.

Equations 2, 3, 4 and 5 are simple fractional equations which compare the measured released concentrations of radionuclides in effluents to the limit or EC. For conservatism, the resulting fraction is further divided by an adjustment factor to account for radionuclides released but not measured prior to release. Since these equations are simple ratios, they are not directly related to or derived from the modeling equations used in the Regulatory Guides.

4.1.1.2.1 ODCM Equations 2 and 4

These two equations use the diluted nuclide concentration (C_{zi}) from ODCM equation 1 and 10 times the EC values for individual nuclides, i (EC_i) from 10CFR20, Appendix B, Table 2, Column 2 to determine the fraction of EC for individual nuclides (FEC) present in the cooling water mixing basin as a result of batch (FEC_b) and/or continuous releases (FEC_c). The equation(s) are:

$$FEC_{b(c)} = \frac{\sum_i \frac{C_{zi}}{EC_i}}{E_{b(c)}} \qquad (2/4)$$

where:

$FEC_{b(c)}$ = the fraction of the unrestricted area EC present in the condenser cooling water mixing basin outflow due to batch (b) or continuous (c) releases.

C_{zi} = the concentration of a radionuclide, i, in the condenser cooling water mixing basin outflow.

EC_i	=	10 times the activity concentration limit in water of radionuclide i according to 10CFR20 Appendix B Table 2, Column 2 (μCi/ml)
$E_{b(c)}$	=	$\dfrac{\text{Quarterly average of FEC in the batch tank (b) or continuous release (c) due to I-131 and principal gamma emitters.}}{\text{Quarterly average of FEC in the batch tank (b) or continuous release (c) due to all radionuclides measured.}}$

The term $E_{b(c)}$ is an adjustment to account for radionuclides such as Sr-89, Sr-90, Fe-55 etc. which are not measured prior to release but are measured in quarterly samples per Technical Specification Table 4.11-1. The value of $E_{b(c)}$ is calculated from previously measured data. If calculated data is not available, historical values may be used. Historical data quarterly analysis have established a value for $E_{b(c)}$ of about 0.8. To ensure conservatism and to allow for occasional concentrations which exceed the historical value, a value of 0.5 can be used as an alternative for $E_{b(c)}$.

Alternatively, the $E_{b(c)}$ factor can be eliminated by including a previous Quarter's beta activity and EC values into the calculations for each release.The addition of these values corrects for unmeasured activity making the $E_{b(c)}$ factor redundant and not required.

4.1.1.2.2 ODCM Equations 3 and 5

These two equations are an alternate means of determining the fraction of EC for a batch (b) or (c) continuous release. The equation(s) are:

$$FEC_{b(c)} = \frac{C_{b(c)}}{1 \times 10^{-7}} \qquad (3/5)$$

where:

$$C_{b(c)} = \sum_{i} C_{zi}$$

1×10^{-7} = ten times the unrestricted area EC for unidentified radionuclides in water from 10CFR20 Appendix B Table 2.

Equation 3/5 differs from equation 2/4 as follows:

- A gross value for radionuclides in water of 1×10^{-7} μCi/ml is used instead of EC values for individual radionuclides (i).
- The diluted concentrations (C_{zi}) of individual radionuclides are summed to produce a gross activity value.
- There is no adjustment for radionuclides not measured prior to release but measured in the monthly and quarterly samples.

As a result, the alternate equations 3/5 will generally yield fractional EC values that are less conservative than equations 2/4.

4.1.1.2.3 ODCM Equation 6

This equation is used to sum all of the fractional EC's to provide a cumulative total from all release paths, since simultaneous liquid radwaste releases may be occurring from several sources. This equation accounts for simultaneous releases from:

- Batch tanks.
- Continuous releases from Units 3 and 4.

The equation does not account for releases from storm drains although this release pathway may be considered a batch release and its contribution determined from monthly sample results.

4.1.2 Dose to a Member of the Public

Technical Specification 3.11.1.2 requires that the dose or dose commitment to a Member of the Public from radioactive materials in liquid effluents released from each unit to Unrestricted Areas shall be limited to:

- During any calendar quarter
  ≤ 1.5 mrems to the whole body
  ≤ 5.0 mrems to any organ

- During any calendar year
  ≤ 3.0 mrems to the whole body
  ≤ 10.0 mrems to any organ

The methodology for satisfying Technical Specification 3.11.1.2 is provided in ODCM equation 7. This equation considers only irradiation from shoreline as a dose factor at Turkey Point because of the nature of the cooling canal and restrictions on access to the site. At Turkey Point, both the condenser cooling water mixing basin and the closed loop cooling canal system are located entirely on FP&L property. Cooling water leaving the plant circulates through the canal system and returns to the plant cooling water inlet without offsite discharge. Since FP&L limits access to the canal and does not permit members of the public to use the water for drinking, bathing, gardening or any other purpose determination of dose is simplified. Dose factors that are important at other sites such as use of the water for drinking and gardens; the consumption of fish and shellfish etc. may be ignored at Turkey Point. Public access is limited by FP&L to occasional use of areas near the canal for camping by scout troops. As a result, the potential exposure of a member of the public to radioactive material in liquid effluent is limited to irradiation of campers by canal shoreline deposits.

The equation used in the ODCM is:

$$D = 0.23 \sum_{k} \sum_{i} A_i^{shoreline} \cdot \frac{C_{ik} \cdot F_{1k} \cdot t_k}{V \cdot \lambda_i^e} \qquad (7)$$

where:

D = total body or organ dose due to irradiation by radionuclides on the shorelines which originated in a liquid effluent release, (mrem)

0.23 = units conversion constant =

$$\frac{1 Ci}{10^6 \mu Ci} \times 60 \frac{\min}{hr} \times 3785 \frac{ml}{gal}.$$

A_i = transfer factor relating a unit aqueous concentration of radionuclide i (μCi) to dose commitment rate to specific organs and the total body of an exposed person (mrem/Ci·gal/min)

C_{ik} = the concentration of radionuclide in the undiluted liquid waste to be discharged that is represented by sample k, (μCi/ml)

F_{1k} = liquid waste discharge flow during release represented by sample k, (gal/min)

V = cooling canal effective volume, approximately 3.75×10^9 gallons

t_k = period of time (hours) during which liquid waste represented by sample k is discharged

λ_i^e = effective decay constant ($\lambda_i + F_3/V$, min $^{-1}$)

where:

λ_i = the radioactive decay constant

F_3 = canal-ground water interchange flow, approximately 2.25×10^5 gal/min

This equation is an adaptation of the shoreline dose equation in NUREG 1.109. This equation determines dose by combining the summed quantities of individual radionuclides i for three discrete terms.

The term A_i referred to as a transfer factor is a site related ingestion dose commitment factor to the total body or any organ for each identified principal gamma and beta emitter. The term A_i is an adaptation of ingestion dose data contained in Regulatory Guide 1.109 Appendix B. The other terms are related to specific site and radionuclide criteria for a given release of liquid effluent. In practice at Turkey Point only the dose to the whole body is determined because if this dose is within specification, organ dose will be below its limit.

4.1.3 Projected Dose

Technical Specification 4.11.1.2 requires that cumulative dose contributions from liquid effluents for the current calendar quarter and current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

In addition, Technical Specification 3.11.1.3 requires that the Liquid Radwaste Treatment system shall be Operable and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent from each unit to the Unrestricted areas would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ when averaged over a 31 day period.

Technical Specification 4.11.1.3.1 requires that doses due to liquid releases from each unit to Unrestricted Areas shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when the Liquid Radwaste Treatment Systems are not being fully utilized.

At Turkey Point, the Technical Specification requirements are satisfied with ODCM equation 8. The dose is projected with the relation:

$$P = \frac{31 \cdot D}{X} \qquad (8)$$

where:

P = the projected total body or organ dose during the month, (mrem)

31 = number of days in a calendar month, (days)

X = number of days in current month to date represented by available radioactive effluent sample, (days)

D = total body or organ dose to date during current month calculated according to ODCM dose equations, (mrem)

Alternately, the monthly dose may be projected by computing the doses to the total body and most exposed organ accumulated during the most recent month and assuming the result represents the projected doses for the current month.

This equation is a simplification of dose projection equations described in Regulatory Guide 1.109 and NUREG-0133.

4.1.4 Effluent Monitor Setpoints

Technical Specification 3.3.3.5 requires that liquid effluent monitoring instrumentation channels be Operable with their alarm/trip setpoints set to ensure that the limits of specifications 3.11.1.1 are not exceeded. The alarm/trip setpoints of these channels shall be determined and adjusted in accordance with the methodology and parameters in the ODCM.

The requirements of this Technical Specification are met by using ODCM equations 9 and 10. These equations are:

$$S_b = \frac{A_b\, S_f}{FEC_b} \cdot g_b + Bkg \qquad (9)$$

$$S_c = \frac{A_c\, S_f}{FEC_c} \cdot g_c + Bkg \qquad (10)$$

where:

$S_{b/c}$ = radiation monitor alarm setpoint for a batch release (b)/continuous release (c) (cpm)

$A_{b/c}$ = laboratory counting rate (cpm/ml) or activity concentration (μi/ml) of sample from batch tank (b)/continuous release (c)

$FEC_{b/c}$ = fraction of unrestricted area EC present in the condenser cooling water mixing basin outflow due to a batch release (b)/continuous release (c)

$g_{b/c}$ = detection efficiency of monitor detector; ratio of effluent radiation monitor counting rate to laboratory counting rate or activity concentration in a given batch sample (cpm/cpm/ml or cpm/μCi/ml) which ever units are consistent with the units A_b/A_c

Bkg = background (cpm)

S_f = A factor to allow for multiple sources from different or common release points. The allowable operating setpoints will be controlled administratively by assigning a fraction of the total allowable release to each of the release sources.

The setpoint equation(s) used at Turkey Point are derived from setpoint determinations provided in NUREG-0133 Addendum. The general equation has been altered to include a safety factor. This factor was added because there is a possibility of continuous releases from Units 3 and 4 occurring at the same time as a batch release. If this event occurs, the limits of Technical Specification 3.11.1.1 could be exceeded if the safety factor were not included.

5.0 Gaseous Radwaste Releases

Gaseous radwaste releases from Turkey Point Nuclear Units 3 and 4 are discharged from four monitored release points. These release points are:

- A common plant vent
- Unit 3 spent fuel pit vent
- Unit 3 and 4 air ejector vents

In addition, unmonitored gaseous releases occur from six release points in the Unit 3 and Unit 4 secondary system. These release points are:

- Blowdown flash tanks (2)
- Hogging jet exhausts (2)
- Water box priming jets (2)

Releases from the unmonitored points can result in discharges of radioactive gases if primary to secondary leakage occurs.

For calculational purposes, airborne releases from all discharge points are treated as a mixed mode, ground level release from a single location. The equations used for calculating gaseous release rates, atmospheric dispersion, dose rates and radiation monitor setpoints are adapted from models and equations given in regulatory guides and NUREG's. The principal references used in the Turkey Point ODCM for radioactive gaseous releases are Regulatory Guides 1.109 and 1.111 and NUREG-0133.

The standard technique is to use Turkey Point meteorological data from daily measurements and/or historical data with models from the regulatory guide to provide atmospheric dispersion factors for the plant. These dispersion factors are determined in 16 compass directions from the plant release point to locations within, at and beyond the site boundary. Because there are physical and chemical differences between the noble gases, radioiodines, tritium and particulates being released and dispersed in gaseous effluents, there are three dispersion factors which must be determined, these are referred to as:

- The atmospheric dispersion factor (χ/Q)
- The atmospheric dispersion factor adjusted for depletion by deposition (χ_d/Q)
- The relative deposition rate onto ground (D/Q)

5.1 Technical Specifications for Gaseous Effluents

The following Technical Specification requirements must be met when releasing radioactive gases and the

methodology for calculating releases must be contained in the ODCM.

5.1.1 Gaseous Effluent Dose Rates

Technical Specification 3.11.2.1 requires that the dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the site boundary shall be limited to the following:

- Noble Gases

  ≤ 500 mrem/yr to the whole body
  ≤ 3000 mrem/yr to the skin

- I-131, I-133, Tritium and Particulates with half lifes greater than 8 days.

  ≤ 1500 mrems/yr to any organ

The requirements of the specification reflect the differences between the behavior of noble gases as opposed to radioiodines, tritium and particulates. As a result of these differences, several equations are required in the ODCM to estimate the quantities of radionuclides released and the dose rate. The methodology for satisfying Technical Specification 3.11.2.1 is provided in ODCM equations 11 to 15.

5.1.1.1 Noble Gas Activity Release Quantities

The total measured quantity of noble gas activity released via a stack or vent during a specific time period can be determined using the appropriate gaseous effluent monitor data as follows:

$$Q_j = \frac{N_j \cdot F}{3.53x10^{-5} \cdot h} \qquad (11)$$

where:

Q_j = total measured gross gaseous radioactivity release via a stack or vent during counting interval j, (μCi)

N_j = counts accumulated during counting interval j, (counts = N(cpm) x t(min))

F = discharge rate of gaseous effluent stream, (ft^3/min)

3.53×10^{-5} = conversion constant, (ft^3/cm^3)

h = effluent noble gas monitor calibration or counting rate response for noble gas gamma radiation, $\left(\frac{cpm}{\mu Ci/cm^3}\right)$

The distribution of radioactive noble gases in a gaseous effluent stream is determined by gamma spectrum analysis of gas samples from that stream. Results of previous analyses may be averaged to obtain a representative distribution.

If f_i represents the fraction of radionuclide i in a given effluent stream, based on the isotopic distribution of that stream, then the quantity of radionuclide i released in a given gaseous effluent stream during counting interval j is:

$$Q_{ij} = Q_j \cdot f_i \qquad (12)$$

where:

Q_{ij} = quantity of radionuclide i released in a given gaseous effluent stream during counting interval j, (μCi)

f_i = the fraction of radionuclide i released in a given effluent stream

Equation 11 is an efficiency correction equation which converts the relative counts of a radiation monitor to an absolute release activity using the known monitor efficiency to make the conversion.

If a gamma spectrum analysis is available for the noble gases in a release, the relative ratios of the gases in the spectrum may be used to convert the gross activity, Q_j from equation 11 to specific radionuclide (i) activity as shown in equation 12. If a gamma spectrum is not available, historical noble gas activities may be averaged to produce release fractions for noble gases. These release fractions from historical Turkey Point noble gas data are given in ODCM Table 3-2.

5.1.1.2 Noble Gas Total Body Dose Rate

The total body dose rate due to noble gas releases is determined using the following equation:

$$\dot{D}_{TB} = \frac{\chi}{Q} \cdot \frac{1}{t} \sum_{i} Q_i \cdot P_{\gamma i} \qquad (13)$$

where:

D_{TB} = Dose rate to total body from noble gases, (mrem/year)

$\frac{\chi}{Q}$ = atmospheric dispersion factor at the off-site location of interest, (sec/m^3)

t = averaging time of release, i.e., increment of time during which Q_i was released, (year).

Q_i = quantity of noble gas radionuclide i released during the averaging time, (μCi)

$P_{\gamma i}$ = factor converting time integrated concentration of noble gas radionuclide i at ground-level to total body dose, $\left(\frac{\text{mrem}}{\mu\text{Ci} \cdot \text{sec/m}^3}\right)$

Equation 13 is an adaptation of the total body dose rate equations from Regulatory Guide 1.109. The atmospheric dispersion factor(s) χ/Q were developed from Turkey Point Meteorological data collected during calendar years 1976 and 1977 and atmospheric models from Regulatory Guide 1.111. The air dose transfer factor $P_{\gamma i}$ is derived from Regulatory Guide 1.109 Table B-1. Factors required for Turkey Point calculations are contained in ODCM tables. The factor(s) χ/Q is contained in ODCM Table 3-6 and the factor(s) $P_{\gamma i}$ in ODCM Table 3-4.

This equation assumes that the person subjected to the dose rate from noble gases is immersed in a semi-infinite cloud of the gases, which infers immersion in gases that are totally mixed and present at some uniform concentration. The limiting case for total body dose rates at or beyond the site boundary is the location at the

site boundary where the highest concentration of radioactive noble gases occurs. This location will be the point or quadrant where χ/Q data indicate that atmospheric dispersion is at a minimum. Present data indicate that minimum dispersion occurs at the site boundary 1950 meters SSE of the plant where χ/Q is equal to a value of 5.8×10^{-7} sec/m^3. This value will be used in equation 13 to determine total body dose rates from noble gases unless subsequent χ/Q data indicate that the minimum dispersion value and/or location at the site boundary has changed.

5.1.1.3 Noble Gas Skin Dose Rate

The skin dose rate due to noble gas releases is determined using the following equation:

$$\dot{D}_s = \frac{\chi}{Q} \cdot \frac{1}{t} \left[\sum_i Q_i \cdot S_{\beta i} + 1.11 \sum_i Q_i \cdot A_{\gamma i} \right] \qquad (14)$$

where:

$\dot{D}_s$ = dose rate to skin from radioactive noble gases, (mrem/year)

$S_{\beta i}$ = factor converting time integrated concentration of noble gas radionuclide i at ground level, to skin dose from beta radiation, $\left(\frac{\text{mrem}}{\mu\text{Ci} \cdot \text{sec/m}^3} \right)$

1.11 = ratio of tissue dose equivalent to air dose in a radiation field, (mrem/mrad)

$A_{\gamma i}$ = factor for converting time integrated concentration of noble gas radionuclide i in a semi-infinite cloud, to air dose from its gamma radiation, $\left(\frac{\text{mrem}}{\mu\text{Ci} \cdot \text{sec/m}^3} \right)$

Equation 14 is an adaptation of the skin dose rate equations from Regulatory Guide 1.109. This equation also uses historical χ/Q values from ODCM Table 3-6. The air dose transfer factor $A_{\gamma i}$ and $S_{\beta i}$ for gamma and beta doses respectively are derived from Regulatory Guide 1.109 Table B-1. The factor(s) $A_{\gamma i}$ is contained in ODCM Table 3-3 and the factor $S_{\beta i}$ in ODCM Table 3-4.

This equation also assumes a person immersed in a semi-infinite cloud of noble gases at the site boundary where minimum atmospheric dispersion occurs. As in equation 13, this location is 1950 meters SSE of the plant where χ/Q is equal to a value of 5.8×10^{-7} sec/m^3.

5.1.1.4 Tritium, I-131, I-133 and Particulate Dose Rate

The dose rate due to tritium, I-131, I-133 and particulates with a half-life greater than 8 days released in gaseous effluents is determined with the following equation:

$$D_{anp} = \frac{1}{3600t} \cdot \frac{\chi_d}{Q} \sum_k \sum_i Q_{ik} \cdot TA_{anip} \qquad (15)$$

where:

D_{anp} = dose equivalent rate to body organ n of a person in age group a exposed via pathway p. (mrem/year)

3600 = conversion constant, (sec/hr)

t = period of time over which the effluent releases are averaged, (hr)

χ_d/Q = atmospheric dispersion factor, adjusted for depletion by deposition (sec/m^3). (Alternatively χ/Q, unadjusted, may be used).

Q_{ik} = quantity of radionuclide i released during time increment t based on analysis k, (μCi).

TA_{anip}= a factor relating the airborne concentration time integral of radionuclide i to the dose equivalent to organ n of a person in age group a exposed via pathway p, $\left(\frac{\text{mrem/yr}}{\mu\text{Ci/m}^3}\right)$

Equation 15 is an adaptation of equations for radioiodines and other radionuclides discharged to the atmospheric contained in regulatory guide 1.109. This equation uses historical χ_d/Q values which are given in ODCM Table 3-7 and are derived

from Turkey Point historical meteorological data and the atmospheric models contained in Regulatory Guide 1.111. The dose transfer factor TA_{anip} is based on dose transfer factors given in Regulatory Guide 1.109 appendix E. These dose transfer factors are given for total body and organ doses to four categories of individuals, these are:

- Adult
- Teenager
- Child
- Infant

The doses to these individuals are expected to occur via two pathways, these are:

- Inhalation
- Ingestion

Pathway-dose transfer factors for Turkey Point are given in the ODCM Appendix A.

To ensure compliance with Technical Specification 3.11.2.1, a hypothetical infant located at the site boundary where the minimum atmospheric dispersion occurs is assumed as the receptor. This approach assures the most conservative estimate of dose. When this assumption is used, the infant's thyroid via the inhalation pathway is the critical organ and controlling pathway respectively. When estimating dose for radioiodines and particulates with half-lifes greater than 8 days, the dose transfer factor, TA_{anip}, is based solely on the radioiodines (I-131, I-133) because the radioiodines contribute essentially all of the dose to the infant's thyroid.

The limiting case for the dose rate due to iodine, tritium and particulates at or beyond the site boundary is the location at the site boundary where there is minimum dispersion adjusted for deposition. Present data from ODCM Table 3-7 indicate that minimum dispersion adjusted for deposition, χ_d/Q, occurs at the site boundary 1950 meters SSE of the plant where the χ_d/Q value is 5.0 x 10^{-7} sec/m^3.

5.1.2 Gaseous Effluent Dose From Noble Gases

Technical Specification 3.11.2.2 requires that the <u>air dose per reactor</u> at and beyond the site boundary due to noble gases in gaseous effluents shall be limited to:

- During any calendar quarter
  $\leq$ 5 mrad for gamma radiation
  $\leq$ 10 mrad for beta radiation

- During any calendar year
  $\leq$ 10 mrad for gamma radiation
  $\leq$ 20 mrad for beta radiation

In addition, Technical Specification 4.11.2.2 requires that the cumulative gamma and beta radiation dose be determined at least once per 31 days to verify that accumulated air dose due to gamma radiation and beta radiation does not exceed the limits for the current quarter and year.

5.1.2.1 Noble Gas Gamma Radiation Dose

The gamma radiation dose is calculated with the following equation:

$$D\gamma = \frac{1}{0.8} \cdot \frac{\chi}{Q} \cdot A_{\gamma eff} \cdot \sum_j Q_j \qquad (16)$$

where:

D_γ = noble gas gamma dose to air due to a mixed mode release, (mrad)

0.8 = a conservatism factor which, in effect, increases the estimated dose to compensate for variability in radionuclide distribution

$\frac{\chi}{Q}$ = atmospheric dispersion factor for a mixed mode discharge, (sec/m^3)

$A_{\gamma eff}$ = effective gamma air dose factor converting time-integrated, ground-level, total activity concentration of radioactive noble gas, to air dose due to gamma radiation. This factor has been derived from noble gas radionuclide distributions in routine operational releases. The effective gamma air dose factor is:

$$A_{\gamma eff} = 1.4 \times 10^{-5} \left(\frac{\text{mrad}}{\mu\text{Ci} \cdot \text{sec/m}^3} \right)$$

Q_j = the measured gaseous radioactivity released via a stack or vent during a single counting interval j, (μCi)

Equation 16 is derived from dose equations for noble gas gamma activity in Regulatory Guide 1.109. The measured gross activity value, Q_j is determined using ODCM equation 11. The atmospheric dispersion factor χ/Q was developed from Turkey Point historical meteorological data using Regulatory Guide 1.111. Turkey Point χ/Q data are given in ODCM Table 3-6. The conservatism value, 0.8, is based on Turkey Point historical noble gas data. This historical variability has been observed in both liquid and gas samples. In the case of liquids, the conservatism value was further reduced to 0.5 because of higher uncertainty of mixing in the liquids. The effective gamma air dose factor, $A_{\gamma eff}$, is based on historical Turkey Point noble gas data collected during the years 1978, 1979 and 1980. The technical basis for $A_{\gamma eff}$ is described in the ODCM Appendix B.

The limiting case for gamma dose from noble gases occurs at the location on the site boundary where minimum atmospheric dispersion, χ/Q, occurs. At Turkey Point, this location is at the site boundary 1950 meters SSE of the plant where the χ/Q value is 5.8×10^{-7} sec/m^3.

5.1.2.2 Noble Gas Beta Radiation Dose

The beta radiation dose is calculated with the following equation:

$$D_\beta = \frac{1}{0.8} \cdot \frac{\chi}{Q} \cdot A_{\beta eff} \cdot \sum_j Q_j \qquad (18)$$

where:

D_β = noble gas beta dose to air due to a mixed mode release, (mrad)

0.8 = a conservatism factor which, in effect, increases the estimated dose to compensate for variability in radionuclide distribution

$A_{\beta eff}$ = effective beta air dose factor converting time-integrated, ground-level, total activity concentration of radioactive noble gas to air dose due to beta radiation. This factor has been derived from noble gas radionuclide distributions in routine operational releases.

The effective beta air dose factor is:

$$A_{\beta eff} = 3.4 \times 10^{-5} \quad \left(\frac{mrad}{\mu Ci \cdot sec/m^3} \right)$$

This equation is identical in format to equation 16, except the effective beta air dose factor $A_{\beta eff}$ has been substituted for $A_{\gamma eff}$ to determine noble gas beta dose. The technical basis for the term $A_{\beta eff}$ is described in the ODCM Appendix B. Beta doses are also calculated for the location on the site boundary where minimum dispersion occurs, this location is 1950 meters SSE of the plant where χ/Q equals 5.8×10^{-7} sec/m^3.

5.1.2.3 Alternate Noble Gas Radiation Dose Calculations

The gamma and beta radiation doses from noble gases may also be calculated using the following equations:

$$D_{\gamma} = \frac{\chi}{Q} \sum_j \sum_i Q_j \cdot f_i \cdot A_{\gamma i} \qquad (17)$$

$$D_{\beta} = \frac{\chi}{Q} \sum_j \sum_i Q_j \cdot f_i \cdot A_{\beta i} \qquad (19)$$

where:

f_i = the fraction of radionuclide i released in a given effluent stream

$A_{\gamma i}$ = factor converting time integrated, ground level concentration of noble gas radionuclide i to air dose from gamma radiation, $\left(\frac{mrad}{\mu Ci \cdot sec/m^3} \right)$

$A_{\beta i}$ = factor converting time-integrated, ground level concentration of noble gas radionuclide i to air dose from beta radiation, $\left(\frac{\text{mrad}}{\mu\text{Ci} \cdot \text{sec/m}^3} \right)$

The difference between these equations and equations 16 and 18 is that $A_{\gamma eff}$ and $A_{\beta eff}$ are calculated $(A_i \cdot f_i)$ for each analysis as described in ODCM Appendix B. Since the factors are determined on each set of analysis data, the conservatism factor, 0.8 is not included in the equations because variability in the radionuclide distribution is reflected in sample analysis data.

5.1.2.4 Cumulative Noble Gas Gamma and Beta Radiation Dose Determinations

The cumulative gamma and beta radiation dose determinations required by Technical Specification 4.11.2.2 is satisfied by summing all the noble gas analysis performed on samples taken during releases using equations 16 or 17 and 18 or 19.

5.1.3 Gaseous Effluent Dose From Iodine, Tritium and Particulates

Technical Specification 3.11.2.3 requires that the dose to a member of the public from I-131, I-133, tritium and all radionuclides in particulate form with half-lifes greater than 8 days in gaseous effluent <u>released from each unit</u> to areas at and beyond the Site Boundary shall be limited to:

- During any calendar quarter
  $\leq$ 7.5 mrems to any organ

and

- During any calendar year
  $\leq$ 15 mrems to any organ

In addition, Technical Specification 4.11.2.3 requires that cumulative dose contributions during the current calendar quarter and current calendar year be determined at least once per 31 days.

5.1.3.1 Iodine, Tritium and Particulate Activity Release Quantities

The quantity of iodine, tritium and particulate activity released in gaseous effluents is determined with the following equation:

$$Q_{ik} = C_{ik} \cdot \sum_j F_j \cdot \Delta t_j \qquad (20)$$

where:

Q_{ik} = the quantity of radionuclide i released in a given effluent stream based on analysis k, (μCi)

C_{ik} = concentration of radionuclide i in a gaseous effluent identified by analysis k, (μCi/cc)

F_j = effluent stream discharge rate during time increment Δt_j, (cc/sec)

Δt_j = time increment j during which radionuclide i at concentration C_{ik} is being discharged, (sec).

Equation 20 is an integration equation used to determine the total activity entering the atmosphere at a known flow for a measured period of time. The term C_{ik} is the concentration values from sampling and analysis performed in accordance with Technical Specification Table 4.11-2 using weekly, monthly and/or quarterly analysis results. The value of C_{ik} may have to be adjusted, changing the value of Q_{ik} under certain circumstances. During normal operations, gaseous releases from stacks and vents require no adjustment of the term C_{ik}. However, if primary to secondary leakage is occurring radioactivity will be released to the atmosphere via gaseous releases from the secondary system. Under these circumstances C_{ik} is determined by sampling steam generator blowdown and assuming that 5% of the I-131 and I-133 and 33% of the tritium in the blowdown stream becomes airborne with the remainder staying in the liquid phase. This assumption has been validated during historical measurements of the blowdown liquid and steam phases.

For other unmonitored releases, the quantity of airborne releases may be determined by performing a steam mass balance using the following equation:

$$F_j = M_w - (M_L + M_S) \qquad (21)$$

where:

M_W = the measured mass of makeup water entering the secondary system during time interval Δtj. e.g. steam generator shutdown.

M_L = the measured mass of water discharged from the secondary system as liquid during time interval Δtj. e.g. steam generator blowdown.

M_S = the measured mass of steam or non-condensible gases discharged from the secondary system during time interval Δtj. e.g. air ejector discharge.

Equation 21 is a simple balance equation comparing input to losses. This equation assumes that when the water injected into the secondary system as makeup (M_W) is equal to the rate of known discharges of steam and gases (M_S) and liquid (M_L) that the discharge from the secondary system (F_j) will be zero. When F_j is a value greater than zero, it is assumed that the release rate is due to other unmonitored releases. For purposes of determining doses due to iodine, tritium and particulates it is further assumed that these other releases are as steam and their concentrations (C_{ik}) are the same as their concentrations in steam generator blowdown samples. This assumption is valid because of the large temperature and pressure differences between the operating secondary system and the ambient environment. Equation 20 is a great simplification of the complex mass balance equations in NUREG 1.109.

5.1.3.2 Determining Dose Due to Iodine, Tritium and Particulate Gaseous Releases

Doses from iodine, tritium and particulates discharged in gaseous effluents can result in exposure to a person by several pathways. These pathways are:

- Direct radiation from airborne radionuclides except noble gases
- Inhalation
- Direct radiation from ground plane deposition
- Fruits and vegetables
- Air-grass-cow-meat
- Air-grass-cow milk

Research, field studies and modeling indicate that of all these pathways, the air-grass-cow-milk pathway is by far the dominant and controlling dose factor. This occurs because:

- The dose factors for the radioiodines are much greater than dose factors for any of the other radionuclides

- The radioiodines contribute essentially all of dose by this pathway with I-131 typically contributing greater than 95%.

Since the air-grass-cow-milk pathway is the controlling pathway and radioiodine the controlling activity, the critical organ is the thyroid. To produce the most conservative result, doses are determined using effective dose transfer factors for radioiodine via the air-grass-cow-milk pathway and the infant thyroid as the receptor. An additional degree of conservatism is provided by totalling the cumulative release of <u>all radioiodines and particulates</u> with the radioiodine effective dose transfer factor to estimate infant thyroid dose.

Doses due to iodines and particulates are determined with the following equation:

$$DM_k = \frac{3.17x10^{-8}}{0.8} \cdot \frac{D}{Q} \cdot TG_{131} \cdot \sum_{i} Q_{ik} \qquad (22)$$

where:

DM_k = the dose commitment to an infant's thyroid received from exposure via the air-grass-cow-milk pathway and attributable to iodines identified in analysis k of effluent air, (mrem)

3.17×10^{-8} = conversion constant, (yr/sec)

0.8 = a conservatism factor which, in effect, increases the estimated dose to compensate for variability in the radionuclide distribution

D/Q = relative deposition rate onto ground from a mixed mode atmospheric release (m^{-2})

TG_{131}= factor converting ground deposition of radioiodines to the dose commitment to an infant's thyroid exposed via the grass-cow-milk pathway.

$$TG_{131} = 6.5 \times 10^{11} \left(\frac{mrem/yr}{\mu Ci/m^2 \cdot sec} \right)$$

Q_{ik} = the quantity of radionuclide i (I-131 and I-133) released in a given effluent stream based on a single analysis k, (μCi)

Equation 22 is adapted from radioiodine dose equations in Regulatory Guide 1.109. The conservatism factor, 0.8, is derived from historical radionuclide distributions observed in gas samples. The relative deposition rate onto ground D/Q is derived from Turkey Point historical meteorological data collected during calendar years 1976 and 1977 and atmospheric models from Regulatory Guide 1.111. The effective dose transfer factor for the air-grass-cow-milk-infant-thyroid pathway, TG_{131}, is based on historical data collected in 1978, 1979, and 1980. The technical basis for TG_{131} is given in the ODCM Appendix B. The quantity of radionuclide i released in a given effluent stream (Q_{ik}) is determined using ODCM equation 20.

The specifications for determining dose via the air-grass-cow-milk pathway given in NUREG-0133 states that the cow should be within 5 miles of the release point. At Turkey Point, there are no milk cows within 5 miles of the plant release point. Under these circumstances, NUREG-0133 states that a cow may be assumed between 4.5-5.0 miles in the worst sector. For the Turkey Point plant, the worst sector is the populated area due west of the plant. As a result, dose due to iodine, tritium, and particulates is determined for a phantom cow on pasture 4.5 miles west of the plant where the

relative deposition rate onto the ground D/Q is $5.0 \times 10^{-10} m^{-2}$.

5.1.3.3 Alternate Methods of Determining Dose Due to Airborne Iodines, Tritium and Particulates

In addition to determining dose due to the dominant air-grass-cow-milk pathway, the ODCM provides equations for evaluating dose via other pathways. These equations are based on examples described in Regulatory Guide 1.109 and NUREG-0133. Equations are provided to evaluate the following dose pathways:

- Inhalation and irradiation dose due to airborne concentrations of radioactive material other than noble gas. ODCM equation 23.
- Deposition from the atmosphere onto vegetation or the ground. ODCM equation 24. Deposition is from airborne concentrations of radioactive material other than noble gas.
- Dose from airborne tritium via vegetation, air-grass-cow-milk or air-grass-cow-meat. ODCM equation 25.
- Cumulative dose via a given pathway as a result of measured discharges from a release point. ODCM equation 26.

These alternate equations may be used to satisfy the requirements of Technical Specification 3.11.2.3.

Except for the cumulative dose equation (26), all of the dose equations share a common format as illustrated by equation 23:

$$D_{ank} = 3.17x10^{-8} \cdot \frac{\chi_d}{Q} \cdot \sum_i Q_{ik} \cdot \sum_p TA_{anip} \qquad (23)$$

where:

D_{ank} = the dose commitment to organ n of a person in age group a due to radionuclides identified in analysis k of an air effluent, (mrem)

3.17×10^{-8} = conversion constant, (yr/sec)

χ_d/Q = atmospheric dispersion factor adjusted for depletion by deposition, (sec/m^3)

Q_{ik} = the quantity of radionuclide i released in a given effluent stream based on analysis k, (μCi)

All of the equations are used to determine a dose (D) to an organ (n) of a person in a particular age group (a) identified in an analysis (k) of an effluent air sample. Although no specific age group (a) or organ (n) is identified in the equation, the most restrictive case is the infant for any organ. As a result the infant will be selected for purposes of making conservative dose estimates.

All of the equations use an atmospheric dispersion factor (χ/Q, χ_d/Q or D/Q). These factors have been determined using historical Turkey Point meteorological data and models from Regulatory Guide 1.111. The factors for sixteen compass sectors around the Turkey Point plant are given in the ODCM Tables 3-6 (χ/Q), 3-7(χ_d/Q) and 3-8 (D/Q). Each of the pathways uses a unique location to evaluate dose to a person, these are:

- The inhalation and irradiation and tritium pathways evaluate dose at the nearest garden (with residence assumed) which is 3.6 miles west of the plant where the χ_d/Q factor (for inhalation and irradiation) is 1×10^{-7} sec/m^3 and the χ/Q factor (tritium) is also 1×10^{-7} sec/m^3.

- The deposition from the atmosphere onto vegetation or the ground pathway evaluates dose at the phantom cow location 4.5 miles west of the plant where the D/Q value is 5.0×10^{-10} m^{-2}.

To determine conformance with Technical Specification 3.11.2.3, a cumulative dose calculation is made using the following equation:

$$D_{an} = \sum_{k} D_{ank} \qquad (26)$$

where:

D_{an} = the dose commitment to organ n of a person in age group (a)

k = the counting index; it may represent either:

p, analysis of a grab sample

w, a weekly sample analysis

m, a monthly composite analysis, or

q, a quarterly composite analysis

5.1.4 Projected Dose

Technical Specification 3.11.2.4 requires that the ventilation exhaust treatment system and gas decay tank system shall be operable and appropriate portions of these systems shall be used to reduce releases of radioactivity when the <u>projected doses in 31 days</u> due to gaseous effluent releases, <u>From each unit</u>, to areas at and beyond the site boundary <u>would exceed</u>:

- 0.2 mrad to air from gamma radiation or
- 0.4 mrad to air from beta radiation or
- 0.3 mrad to any organ of a Member of the Public

Technical Specification 4.11.2.4 further states that <u>doses from each unit</u> to areas at and beyond the site boundary shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Gaseous Radwaste Treatment Systems are not being fully utilized.

At Turkey Point, these Technical Specifications requirements are satisfied with ODCM equation 29 as follows:

$$P = \frac{31 \cdot D}{X} \qquad (30)$$

where:

P = the projected dose during the month, (mrem)

31 = number of days in a calendar month, (days)

X = number of days in current month to date represented by available radioactive effluent sample, (days)

D = dose to date during current month calculated according to ODCM dose equations

Alternately, the monthly dose may be projected by computing the dose accumulated during the most recent month and assuming the result represents the projected dose for the current month.

This equation, adapted from dose projection equations in Regulatory Guide 1.109 and NUREG-0133, extrapolates the dose to date in a current month to include the entire month. It should be noted that equation 30 is the same as ODCM equation 8 for liquids.

5.1.5 Effluent Noble Gas Monitor Setpoints

Technical Specification 3.3.3.6 requires that radioactive gaseous effluent monitoring instrumentation channels be operable with their alarm/trip setpoints set to ensure that the limits of specification 3.11.2.1 and 3.11.2.5 are not exceeded. The alarm/trip setpoints of the channels meeting specification 3.11.2.1 shall be determined and adjusted in accordance with methodology and parameters in the ODCM.

The requirements of this Technical Specification can be met using the following equation:

$$S = 1.06\left[\frac{h \cdot S_f}{F \cdot \chi/Q}\right]\left[\frac{\sum_i C_i}{\sum_i C_i \cdot DF_i}\right] + Bkg \qquad (27)$$

where:

S = The alarm setpoint, (cpm)

1.06 = conversion constant; 500 mrem/yr · 60 sec/min · 35.37 ft^3/m^3 · $1m^3/10^6cm^3$

h = monitor response to activity concentration of effluent, $\left(\frac{cpm}{\mu Ci/cm^3}\right)$

F = flow of gaseous effluent stream, i.e., flow past the monitor, (ft^3/min)

$\frac{\chi}{Q}$ = atmospheric dispersion factor at the offsite location of interest, (sec/m^3)

C_i = concentration of radionuclide i in gaseous effluent (μCi/cc)

DF_i = factor converting ground-level or split-wake release of radionuclide i to the total body dose equivalent rate at the location of potential exposure, $\left(\frac{mrem}{yr \cdot \mu Ci/m^3}\right)$

S_f = a factor to allow for multiple sources from different or common release points. The allowable operating setpoints will be controlled administratively by assigning a fraction of the total allowable release to each of the release sources.

Equation 27 is based on setpoint methodology described in NUREG-0133. This equation uses known factors about the gas monitor, atmospheric dispersion, and radionuclide distribution and background radiation to determine a setpoint. The required equation factors are:

- The monitor
  - An efficiency factor, h, must be known.
  - Gas flow past the monitor must be known. It should be noted that this is not the flow of the vent or stack through which gas is being discharged.

- Atmospheric dispersion (χ/Q)

  - The χ/Q values for Turkey Point are based on historical meteorological data and methodology from Regulatory Guide 1.111. Turkey Point χ/Q data are given in ODCM Table 3-6.

  - The air dose conversion factor DF_i is a factor which converts the ground level release of radionuclide i to the total body dose equivalent at the location of potential exposure. DF_i factors for Turkey Point are taken from Regulatory Guide 1.109 Table B-1 and are given in ODCM Table 3-5.

- Radionuclide distribution

  There are three acceptable means to determine radionuclide distribution, these are:

  - To perform a gamma spectrum analysis of the gas release.. Results of one or more analysis may be averaged to obtain a representative spectrum. This is the preferred way of determining release concentrations.

  - From the historical spectrum of noble gas distributions given in ODCM Table 3-2. This table is used in conjunction with a noble gas gross activity analysis.

  - By attributing the total or gross activity to Xe-133. This technique is valid because Xe-133 comprises about 99% of the noble gas activity.

- Background (Bkg)

  - The radioactive background in which the monitor operates should be known and added to the setpoint value to prevent setting the monitor setpoint too low.

The set point is determined by evaluating the location at the site boundary where minimum atmospheric dispersion occurs. This location is 1950 meters SSE of the plant where the χ/Q value is 5.8×10^{-7} sec/m^3.

The limiting factor when using equation 27 to determine a setpoint is the total body dose rate limit of 500 mrem/yr which is included in the 1.06 conversion factor. The use of total body dose assumes that the total body dose will be the controlling dose rate and the dominant contributor to this dose will be Xe-133.

The requirements of Technical Specification 3.3.3.6 can also be met by using the following equation:

$$S = \frac{EC \cdot h \cdot S_f}{4.7x10^{-4} \cdot F \cdot \chi/Q} \qquad (28)$$

where:

EC = the unrestricted area effluent concentration for the effluent noble gas mixture

4.7×10^{-4} = conversion constant, $\frac{1m^3}{35.37ft^3} \times \frac{1\ min}{60\ sec}$

h = monitor response to activity concentration of effluent, $\left(\frac{cpm}{\mu Ci/cm^3}\right)$.

F = flow of gaseous effluent stream, i.e., flow past the monitor (ft^3/min)

$\frac{\chi}{Q}$ = atmospheric dispersion factor at the offsite location of interest, (sec/m^3)

S_f = A factor to allow for multiple sources from different or common release points. The allowable operating setpoints will be controlled administratively by assigning a fraction of the total allowable release to each of the release sources.

The unrestricted area effluent concentration (EC) for the noble gases is determined from the distribution of noble gases in the release as follows:

$$EC = \sum_i C_i + \sum_i \frac{C_i}{EC_i} \qquad (29)$$

where:

C_i = concentration of radionuclide i in a gaseous effluent

EC_i = 10 times the unrestricted area effluent concentration for radionuclide i. Values of EC_i for the noble gases are given in 10CFR20, Appendix B, Table 2, Column 1.

The differences between equation 27 and equation 28 are:

- The dose rate in equation 27 represented by the term

$$\left[\frac{\sum_i C_i}{\sum_i C_i \cdot DF_i}\right]$$

  has been replaced by effluent concentration values based on noble gas release concentration as represented by the term EC which is derived using equation 28 and EC_i values for the noble gases from 10CFR20 Appendix B, Table 2, Column 1. As a result of this replacement, the air dose conversion constant DF_i is not required in equation 28.

- The background term included in equation 27 is not required in equation 28 because background is an inherent part of EC.

- The limiting factor of 500 mrem/yr total body dose in equation 27 has been replaced by EC. As a result the conversion factor changes in equation 28.

The atmospheric dispersion factor χ/Q for both equation 27 and equation 28 are the same. Setpoints using EC are also evaluated at the point of minimum atmospheric dispersion which is 1950 meters SSE of the plant where the χ/Q value is 5.8×10^{-7} sec/m^3.

6.0 Annual Dose Commitments

Technical Specification 3.11.4 requires that the annual (calendar year) dose or dose commitment to any Member of the Public due to releases of radioactivity and radiation from uranium fuel cycle sources shall be limited to:

≤ 25 mrems whole body or any organ except the thyroid
≤ 75 mrems to the thyroid

The requirements of Technical Specification 3.11.4 can be satisfied by applying the following equations from the ODCM.

- Total body dose due to liquid effluent deposited on the cooling canal shoreline.

$$D = 0.23 \sum_k \sum_i A_i^{shoreline} \cdot \frac{C_{ik} \cdot F_{ik} \cdot t_k}{V \cdot \lambda_i^e} \qquad (7)$$

- Total body dose due to noble gas gamma (γ).

$$\dot{D}_{TB} = \frac{X}{Q} \cdot \frac{1}{t} \sum_i Q_i \cdot P_{\gamma i} \qquad (13)$$

- Total body dose due to noble gas beta (ß).

$$\dot{D}_s = \frac{\chi}{Q} \cdot \frac{1}{t} \left[\sum_i Q_i \cdot S_{\beta i} + 1.11 \sum_i Q_i \cdot A_{\gamma i} \right] \qquad (14)$$

- Thyroid dose due to gaseous effluents other than noble gases.

$$DM_k = \frac{3.17x10^{-8}}{0.8} \cdot \frac{D}{Q} \cdot TG_{131} \cdot \sum_i Q_{ik} \qquad (22)$$

When equations 13 and 14 are used to assess compliance with Technical Specification 3.11.4, a different atmospheric dispersion factor χ/Q must be used. For determinations of annual dose, the χ/Q value is for the most exposed receptor not the minimum dispersion point at the site boundary. For Turkey Point the most exposed receptor is located 3.6 miles west northwest of the plant at the location of the nearest garden. The χ/Q value at that location is $1.0x10^{-7}$ sec/m^3.