Pacific Gas and Electric Company
Humboldt Bay Power Plant
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March 29, 2002

PG&E Letter HBL-02-005

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 50-133, OL-DPR-7
Humboldt Bay Power Plant Unit 3
Annual Radioactive Effluent Release Report for 2001

Dear Commissioners and Staff:

Enclosure 1 contains the Humboldt Bay Power Plant Unit 3 "Annual Radioactive Effluent Release Report," covering the period January 1 through December 31, 2001. This report is required by Section VII.J.3 of the Unit 3 Technical Specifications.

Enclosure 2 contains Revision 5 to the "SAFSTOR Offsite Dose Calculation Manual" as required by Specification Section 4.2 of the "SAFSTOR Offsite Dose Calculation Manual."

Sincerely,

Tom A. Moulia

Tom A. Moulia

cc: Drew Holland
Ellis W. Merschoff

Enclosures

IE48

Enclosure 1
PG&E Letter HBL-02-005

HUMBOLDT BAY POWER PLANT UNIT 3
ANNUAL RADIOACTIVE
EFFLUENT RELEASE REPORT

January 1 through December 31, 2001

PACIFIC GAS AND ELECTRIC COMPANY
HUMBOLDT BAY POWER PLANT
DOCKET NO. 50-133, LICENSE NO. DPR-7

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT
JANUARY 1, 2001 THROUGH DECEMBER 31, 2001

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HUMBOLDT BAY POWER PLANT

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT 2001

INTRODUCTION

This report summarizes gaseous and liquid radioactive effluent releases from Humboldt Bay Power Plant Unit 3 for the four quarters of 2001. The report includes calculated potential radiation doses from these radioactive effluents and a comparison with the numerical guidelines of 10 CFR 50, Appendix I, as well as a summary of shipments of solid radioactive waste. The concentrations of plant effluent releases during the reporting period were well below Technical Specification (TS) limits.

The information is reported as required by Section VII.J.3 of the TS, and it is presented in the general format of Regulatory Guide 1.21, Appendix B (except for the topics identified below).

Meteorology

The meteorological data logging system was removed from service in 1967 so the information specified by Regulatory Guide 1.21, Appendix B, Section F, is not available. Previous Humboldt Bay Power Plant Annual Radioactive Effluent Release Reports summarized the cumulative joint frequency distribution of wind speed, direction, and atmospheric stability for the period April 1962 through June 1967, when the meteorological data logging system was in service.

Short-lived Nuclides

The Unit was last operated on July 2, 1976. Due to the long decay time since operation, short-lived radionuclides are neither expected nor reported. This includes Iodines and noble gases other than Krypton-85.

Air Particulate Filter Composites - Sr-90

Air particulate sample filters are combined for approximately monthly intervals and analyzed off-site for Sr-90.

Air Particulate Filter Composites - Gross Alpha

Each weekly sample filter is individually counted for gross alpha activity, rather than analyzing a monthly composite of the filters, as described in Regulatory Guide 1.21.

Gaseous Effluents - Tritium

Tritium releases during plant operation were less than detection levels. Since the plant was permanently shutdown in 1976, current tritium release levels are less than the release levels that occurred during plant operations. Therefore, no tritium samples were collected during this reporting period.

Liquid Effluents - Sr-90

Batch releases may be analyzed individually, or composited and analyzed monthly, rather than analyzed as a quarterly composite as described in Regulatory Guide 1.21.

Average Energy

For HBPP, calculations for the average energy of gaseous releases of fission and activation gases are not required to be performed or reported.

Errata For Previous Report

None.

I. SUPPLEMENTAL INFORMATION

A. Regulatory Limits

1. Gaseous Effluents

a. Noble Gas Release Rate Limit

The radioactive noble gas release rate limit is based on concentration limits from 10 CFR 20, divided by an annual average dispersion factor for the sector with the least favorable atmospheric dispersion. The applicable annual average dispersion factor is 1.0E-5 seconds per cubic meter.

b. Iodine Release Rate Limit

Due to the long decay time since the Unit was shutdown, the license does not define an iodine release rate limit.

c. Particulate Release Rate Limit

The radioactive particulate release rate limit is based on concentration limits from 10 CFR 20, divided by an annual average dispersion factor for the sector with the least favorable atmospheric dispersion. The applicable annual average dispersion factors for elevated releases and for ground-level releases are 1.0E-5 and 6.59E-3 seconds per cubic meter, respectively. The radionuclide mixture used to determine the limits is estimated from the mixture observed prior to installing the stack HEPA filter (44% Co60, 4% Sr90, 52% Cs137), after decay correction for approximately 3 years.

When both elevated and ground-level releases occur, the "percent of applicable limit" in Table 1 is the sum of the values for "percent of applicable limit" for each of the release paths.

2. Liquid Effluents

a. Concentration Limit

Concentration limits for liquid effluent radioactivity released to Humboldt Bay are taken from 10 CFR 20.

B. Maximum Permissible Concentrations

1. Gaseous Effluents

Maximum Permissible Concentrations for gaseous effluents are taken from 10 CFR 20, Appendix B, Table 2, Column 1.

2. Liquid Effluents

Maximum Permissible Concentrations for liquid effluents taken from 10 CFR 20, Appendix B, Table 2, Column 2.

C. Measurements and Approximations of Total Radioactivity

1. Gaseous Effluents - Elevated Release

a. Fission and Activation Gases

All ventilation and system vents are routed to the Unit 3 stack. A continuous monitor equipped with a beta scintillator, with its response calibrated for Kr-85, monitors the gaseous activity released from the stack.

The "less than" value reported for Kr-85 is based on the estimated sensitivity of the stack Kr-85 monitor.

The estimated sensitivity of the stack Kr-85 monitor permits detection of Kr-85 at approximately 50% of applicable LLD presented in the ODCM.

b. Iodines

Due to the long decay time since operation (shutdown July 2, 1976), no detectable releases of radioactive Iodines can be expected. Therefore, the TS do not require that these radionuclides be monitored.

c. Particulates

Radioactive particulates released from the plant stack are monitored by continuous sample collection on particulate filters. Filter papers are removed from the stack sampling system weekly, and are analyzed for the concentration of gamma-emitting nuclides (intrinsic germanium detector). All statistically significant gamma peaks are identified.

After decaying at least 7 days, the filters are analyzed for gross alpha radioactivity (internal proportional counter or scintillation counter).

Filters are composited monthly and analyzed monthly for Strontium-90 (the only Strontium present). The monthly composite results are averaged together to produce the quarterly composite.

The estimated error of the reported particulate release values is based on uncertainty in sample flow rate, stack flow rate, detector calibration, and typical sample counting statistics.

The Minimum Detectable Activity (MDA) for all particulate filter samples was less than the applicable LLD presented in the ODCM.

2. Gaseous Effluents - Ground-level Release

a. Fission and Activation Gases

All ventilation and system vents were routed to the Unit 3 stack during the report period. Refer to the discussion for elevated releases.

b. Iodines

All ventilation and system vents were routed to the Unit 3 stack during the report period. Refer to the discussion for elevated releases.

c. Particulates

All ventilation and system vents were routed to the Unit 3 stack during the report period. Refer to the discussion for elevated releases.

3. Liquid Effluents

a. Batch Releases

Water from contaminated plant systems was collected, filtered, treated with Cesium-specific ion-exchange media, and analyzed before discharge (on a batch basis) through the liquid radwaste process monitor. Analysis of weekly composite samples from the plant effluent canal did not detect any additional release of radioactive liquids during the report period.

Samples of liquid waste batches were analyzed for the concentration of gamma-emitting nuclides (intrinsic germanium detector). All statistically important peaks were identified. All batches, or composites of batches, were analyzed for radioactive strontium (Sr-90), gross alpha and tritium.

The error of the reported release values is estimated based on uncertainty in sample volume, batch volume, detector calibration, and typical sample counting statistics.

The MDA for all batch samples was less than the applicable LLD presented in the ODCM.

b. Continuous Releases

There were no continuous liquid effluent releases during this report period.

D. Batch Release Statistics

1. Liquid

a. Number of batch releases.. 8

b. Total time period for batch releases .... 1.36E3 minutes

c. Maximum time period for a batch release........ 1.82E2 minutes

d. Average time period for a batch release.......... 1.70E2 minutes

e. Minimum time period for a batch release......... 1.50E2 minutes

2. Gaseous

a. Number of batch releases.. 0

b. Total time period for batch releases .... N/A

c. Maximum time period for a batch release.......................... N/A

d. Average time period for a batch release............................ N/A

e. Minimum time period for a batch release........................... N/A

E. Abnormal Release Statistics

1. Liquid

a. Number of abnormal releases 0

b. Total activity released... N/A

2. Gaseous

a. Number of abnormal releases 0

b. Total activity released... N/A

II. GASEOUS AND LIQUID EFFLUENTS

A. Gaseous Effluents

Table 1 summarizes the total quantities of radioactive gaseous effluents. Table 2A presents the quantities of each of the nuclides determined to be released from the stack (elevated release point). Table 2B presents the quantities of each of the nuclides determined to be released by other routes (ground level release points).

B. Liquid Effluents

Table 3 summarizes the total quantities of radioactive liquid effluents. Table 4 presents the quantities of each of the nuclides determined to be released.

HUMBOLDT BAY POWER PLANT

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT 2001

TABLE 1

GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES

	Units	First Quarter	Second Quarter	Third Quarter	Fourth Quarter	Est. Total Error, %
A. Fission & Activation Gases						
1. Total release	Ci	<5.13E1	<5.19E1	<5.25E1	<5.25E1	3.20E1
2. Average release rate	μCi/sec	<6.60E0	<6.60E0	<6.60E0	<6.60E0	
3. Percent of applicable limit	%	<9.40E-3	<9.40E-3	<9.40E-3	<9.40E-3	
4. Applicable limit	μCi/cc	7.00E-7	7.00E-7	7.00E-7	7.00E-7	
B. Particulates						
1. Total release	Ci	<1.96E-6	6.26E-7	5.35E-7	<2.31E-6	2.80E1
2. Average release rate	μCi/sec	<2.51E-7	7.96E-8	6.73E-8	<2.90E-7	
3. Percent of applicable limit	%	<2.90E-6	9.17E-7	7.75E-7	<3.34E-6	
4. Applicable limit	μCi/cc	8.68E-11	8.68E-11	8.68E-11	8.68E-11	
5. Gross alpha radioactivity	Ci	<4.87E-8	<5.73E-8	<5.42E-8	<5.67E-8	

Note: The < symbol used in this table means that a majority of the measurements contributing to the result were less than the Minimum Detectable Activity (MDA) for the analyses. Data for individual nuclides combines detected and non-detected results as if all values were detected. The < symbol is applied if less than 50% of the combined value is made up of detected results. When combining detected and non-detected results for different nuclides (e.g. activity totals of multiple nuclides), values with the < symbol are ignored (i.e. treated as zero). When combining non-detected results for different nuclides (e.g. activity totals of multiple nuclides, when none were detected), all values with the < symbol are used.

Since the particulate releases for this period were mostly "less than" values, the limits are based on the typical mixture for 1998, decay corrected by approximately 3 years. For the year 2001, the mixture is estimated to be 61% Cs-137, 34% Co-60 and 5% Sr-90.

TABLE 2A

GASEOUS EFFLUENTS - ELEVATED RELEASE - NUCLIDES RELEASED

Nuclides Released	Unit	Continuous Mode			
		First Quarter	Second Quarter	Third Quarter	Fourth Quarter
1. Fission Gasses					
Krypton-85	Ci	<5.13E1	<5.19E1	<5.25E1	<5.25E1
Total for period	Ci	<5.13E1	<5.19E1	<5.25E1	<5.25E1
2. Particulates					
Cobalt-60	Ci	<7.65E-7	<6.40E-7	<6.17E-7	<8.28E-7
Strontium-90	Ci	<4.04E-8	<3.71E-8	<3.50E-8	<3.50E-8
Cesium-137	Ci	<6.17E-7	6.26E-7	5.35E-7	<5.77E-7
Am-241	Ci	<5.34E-7	<5.63E-7	<4.31E-7	<8.67E-7
Total for period	Ci	<1.96E-6	6.26E-7	5.35E-7	<2.31E-6

Note: The < symbol used in this table means that a majority of the measurements contributing to the result were less than the Minimum Detectable Activity (MDA) for the analyses. Data for individual nuclides combines detected and non-detected results as if all values were detected, but the < symbol is applied if less than 50% of the combined value is made up of detected results. When combining detected and non-detected results for different nuclides (e.g. activity totals of multiple nuclides), values with the < symbol are ignored (i.e. treated as zero). When combining non-detected results for different nuclides (e.g. activity totals of multiple nuclides, when none were detected), all values with the < symbol are used.

HUMBOLDT BAY POWER PLANT

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TABLE 2B

GASEOUS EFFLUENTS - GROUND-LEVEL RELEASES
NUCLIDES RELEASED

Nuclides Released	Unit	Continuous Mode			
		First Quarter	Second Quarter	Third Quarter	Fourth Quarter
1. Fission Gasses					
Krypton-85	Ci	N/A	N/A	N/A	N/A
Total for period	Ci	N/A	N/A	N/A	N/A
2. Particulates					
Cobalt-60	Ci	N/A	N/A	N/A	N/A
Strontium-90	Ci	N/A	N/A	N/A	N/A
Cesium-137	Ci	N/A	N/A	N/A	N/A
Am-241	Ci	N/A	N/A	N/A	N/A
Total for period	Ci	N/A	N/A	N/A	N/A

Note: The < symbol used in this table means that a majority of the measurements contributing to the result were less than the Minimum Detectable Activity (MDA) for the analyses. Data for individual nuclides combines detected and non-detected results as if all values were detected, but the < symbol is applied if less than 50% of the combined value is made up of detected results. When combining detected and non-detected results for different nuclides (e.g. activity totals of multiple nuclides), values with the < symbol are ignored (i.e. treated as zero).

N/A - There were no ground level gaseous effluents during the report period.

HUMBOLDT BAY POWER PLANT

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TABLE 3

LIQUID EFFLUENTS - SUMMATION OF ALL RELEASES

	Units	First Quarter	Second Quarter	Third Quarter	Fourth Quarter	Est. Total Error, %
A. Fission & Activation Products						
1. Total release (not including tritium, gases, alpha)	Ci	7.63E-5	2.62E-5	5.37E-5	5.90E-5	1.30E1
2. Average diluted concentration	µCi/ml	3.21E-12	1.43E-12	2.28E-12	2.28E-12	
3. Percent of applicable limit	%	4.13E-4	2.32E-4	2.80E-4	3.04E-4	
4. Applicable limit	µCi/ml	7.77E-7	6.18E-7	8.14E-7	7.50E-7	
B. Tritium						
1. Total release	Ci	1.47E-3	3.51E-4	1.47E-3	1.42E-3	1.50E1
2. Average diluted concentration	µCi/ml	6.18E-11	1.92E-11	6.23E-11	5.52-11	
3. Percent of applicable limit	%	6.18E-6	1.92E-6	6.23E-6	5.52E-6	
4. Applicable limit	µCi/ml	1.00E-3	1.00E-3	1.00E-3	1.00E-3	
C. Gross Alpha Radioactivity						
1. Total release	Ci	2.29E-6	6.23E-7	3.20E-7	2.19E-7	1.00E1
D. Volume of waste released (prior to dilution)	Liters	7.18E4	2.62E4	4.92E4	5.19E4	3.00E0
E. Volume of dilution water	Liters	2.38E10	1.83E10	2.36E10	2.58E10	1.50E1

Note: The < symbol used in this table means that a majority of the measurements contributing to the result were less than the Minimum Detectable Activity (MDA) for the analyses. Data for individual nuclides combines detected and non-detected results as if all values were detected, but the < symbol is applied if less than 50% of the combined value is made up of detected results. When combining detected and non-detected results for different nuclides (e.g. activity totals of multiple nuclides), values with the < symbol are ignored (i.e. treated as zero).

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TABLE 4

LIQUID EFFLUENTS - NUCLIDES RELEASED

Nuclides Released	Unit	Batch Mode			
		First Quarter	Second Quarter	Third Quarter	Fourth Quarter
Strontium-90	Ci	2.32E-5	1.64E-5	1.69E-5	2.07E-5
Cesium-137	Ci	5.13E-5	9.40E-6	3.00E-5	3.67E-5
Cesium-134	Ci	<5.77E-7	<1.98E-7	<4.44E-7	<3.39E-7
Cobalt-60	Ci	1.86E-6	3.83E-7	6.87E-6	1.61E-6
Am-241	Ci	<9.20E-7	<2.97E-7	<2.01E-6	<5.93E-7
Total for period	Ci	7.63E-5	2.62E-5	5.37E-5	5.90E-5

Nuclides Released	Unit	Continuous Mode			
		First Quarter	Second Quarter	Third Quarter	Fourth Quarter
Strontium-90	Ci	N/A	N/A	N/A	N/A
Cesium-137	Ci	N/A	N/A	N/A	N/A
Cesium-134	Ci	N/A	N/A	N/A	N/A
Cobalt-60	Ci	N/A	N/A	N/A	N/A
Am-241	Ci	N/A	N/A	N/A	N/A
Total for period	Ci	N/A	N/A	N/A	N/A

Notes: The < symbol used in this table means that a majority of the measurements contributing to the result were less than the Minimum Detectable Activity (MDA) for the analyses. Data for individual nuclides combines detected and non-detected results as if all values were detected, but the < symbol is applied if less than 50% of the combined value is made up of detected results. When combining detected and non-detected results for <u>different</u> nuclides (e.g. activity totals of multiple nuclides), values with the < symbol are ignored (i.e. treated as zero).

N/A - There were no continuous mode liquid effluents during the report period.

III. SOLID RADIOACTIVE WASTE

Table 5 summarizes the disposal of solid radioactive waste made during the report period. The volume reported is the 'as-buried' quantity.

HUMBOLDT BAY POWER PLANT

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT 2001

TABLE 5

SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A. Solid Waste Shipped Offsite For Burial Or Disposal

1. Type of Waste	Unit	12 Month Period	Estimated Total Error, %
a. Spent resins, filter sludges, evaporator bottoms, etc.	Cubic Meter	0	N/A
	Ci	0	N/A
b. Dry compressible waste, contaminated equipment, etc.	Cubic Meter	0	N/A
	Ci	0	N/A
c. Irradiated components, control rods, etc.	Cubic Meter	0	N/A
	Ci	0	N/A
d. Other (Processed Waste)	Cubic Meter	5.73E0	1.00E1
	Ci	2.41E-2	5.60E1

2. Estimate of major nuclide composition (by type of waste)	Unit	Nuclide	12 Month Period
d. Other (Processed Waste)	%	H-3	5.73E-5
	%	C-14	3.22E-4
	%	Fe-55	2.42E0
	%	Co-60	1.31E0
	%	Ni-63	1.96E0
	%	Sr-90	7.27E0
	%	Tc-99	2.86E-3
	%	I-129	1.61E-7
	%	Cs-137	7.62E1
	%	Pu-238	6.72E-2
	%	Pu-239/40	2.49E-1
	%	Pu-241	6.96E0
	%	Am-241	1.21E0
	%	Cm-244	1.02E-1

3. Solid Waste Disposition	Number of Shipments	Mode of Transportation	Destination
	4	Truck	Clive, Utah

B. Irradiated Fuel Shipments

1. Irradiated Fuel Disposition	Number of Shipments	Mode of Transportation	Destination
	None	N/A	N/A

IV. RADIOLOGICAL IMPACT ON MAN

A comparison of calculated doses from various paths has shown that the offsite doses are primarily due to direct radiation and to the consumption of aquatic foods. Maximum doses to individuals (for the maximally exposed organs and age groups) are summarized in Table 6. These doses comply with 40 CFR 190 as there are no other uranium fuel cycle facilities within 8 km of the Humboldt Bay Power Plant.

A. Doses to the average individual in the population from all receiving-water-related pathways were calculated for detected releases, based on the guidance of Regulatory Guide 1.109. The highest results were less than 0.001 mrem/yr (total body) for the Adult age group, and 0.001 mrem/yr for the bone of the Adult age group.

   These doses are well below the 10 CFR 50, Appendix I numerical guidelines for limiting effluents as low as is reasonably achievable (ALARA) (3 mrem/yr to the total body and 10 mrem/yr to any organ).

B. Total body doses to the average individual in the population from gaseous effluents to a distance of 50 miles from the site are not calculated, but this dose is less than the total body dose to an average individual present at the maximally exposed location. For an average individual at the maximally exposed location, the total body dose (calculated with the same dispersion and deposition parameters as were used to calculate maximum exposure) was less than 0.001 mrem/yr.

   This maximum calculated dose is well below the 10 CFR 50, Appendix I numerical ALARA guidelines (10 mrem/yr for gamma radiation and 20 mrad/yr for beta radiation from noble gases and <15 mrem/yr to any organ from tritium and radionuclides in particulate form).

C. Total body doses (to the average individual in unrestricted areas from direct radiation from the facility) are based on TLD results of stations at the site boundary, using the shoreline occupancy factors given in Regulatory Guide 1.109 for the highest average potential individual (Teen age group). For this group, direct radiation would result in an exposure of 0.008 mrem/yr.

   This maximum potential dose is well below the 10 CFR 20.1302(b)(2)(ii) limit of 50 mrem/yr from external sources necessary to demonstrate compliance with the 10 CFR 20.1301 dose limit for individual members of the public.

HUMBOLDT BAY POWER PLANT

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TABLE 6

RADIATION DOSE FOR MAXIMALLY EXPOSED INDIVIDUALS

	Dose, milli-rem				
Dose Source	First Quarter	Second Quarter	Third Quarter	Fourth Quarter	Annual Total
Liquid Effluents					
Water-related Pathways (1)	<0.01 (5) <0.01 (6)	<0.01 (5) <0.01 (7)	<0.01 (5) <0.01 (7)	<0.01 (5) <0.01 (6)	<0.01 (5) <0.01 (6)
Airborne Effluents					
Particulates (2)	0.00 (8) 0.00 (8)	<0.01 (5) <0.01 (7)	<0.01 (5) <0.01 (7)	0.00 (8) 0.00 (8)	<0.01 (5) <0.01 (7)
Noble Gases (3)	N/A	N/A	N/A	N/A	N/A
Direct Radiation (4)	0.01	0.01	0.02	0.01	0.01

Notes

1. Maximum total body and organ doses to individuals in unrestricted areas from receiving-water-related exposure pathways were calculated from the average concentrations of liquid releases detected during the report period, following the applicable portions of Regulatory Guide 1.109.

2. Maximum total body and organ doses to individuals in unrestricted areas from airborne-particulate-related exposure pathways were calculated from the average concentrations of airborne particulate releases detected during the report period, following the applicable portions of Regulatory Guide 1.109.

3. Total body and skin doses to potentially exposed individuals located at the point of maximum offsite ground-level concentrations of radioactive gaseous effluents were not calculated because there were no detected releases of radioactive noble gases, and because the total body doses would be less than 0.005 milli-rem/yr at the level at which the releases could be detected.

4. Total body doses (to the maximum individual in the population) are based on TLD results of stations at the site boundary, using the shoreline occupancy factors of Regulatory Guide 1.109 for the maximum potential individual (Teen age group).

5. Total body (Adult age group).

6. Bone (Adult age group).

7. Bone (Child age group).

8. There were no detectable stack releases in the first and fourth quarters.

V. CHANGES TO THE OFFSITE DOSE CALCULATION MANUAL (ODCM)

The ODCM was revised once during the report period. The change maintained the level of radioactive effluent control and dose commitment required by regulation. The changes did not adversely affect the accuracy or reliability of effluent, dose or setpoint calculations.

The revision involved the following change:

- For the REMP report format shown in Table 4-1 (page I-46), the row titled "DIRECT RADIATION" was revised to show data from the four off-site TLD stations as the control locations for monitoring direct radiation, rather than showing the data for "onsite" and "offsite" stations as separate items.
- Added a clarification to the Liquid Effluent Monitor Setpoint Calculation (Section 1.1.6, page II-3), that a minimum of one circulating water pump must be in operation during radioactive liquid discharges to ensure that the alarm setpoint is conservative for release mixtures that are not 100% Cs-137.
- Added a policy statement to Section 9.0, "Program Changes" (added new Section 9.3, page II-38) to maintain the existing "non-quality-related" environmental monitoring program shown as either "State of California" or "PG&E/HBPP Elective" in Table 2-7.
- Added new Section 10.0, "Commitments" (on page II-38) to provide the reference for the source of the commitment in Section 9.3, as noted above.

Revision 5 to the ODCM was reviewed by the Plant Staff Review Committee (PSRC) on 11/29/01 and approved by the Plant Manager on 11/29/01. It was effective on 12/13/01.

VI. CHANGES TO THE PROCESS CONTROL PROGRAM (PCP)

There were no changes to the Process Control Program in 2001

VII. CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

There were no major changes to the gaseous, liquid or solid radwaste treatment systems during the report period.

VIII. INOPERABLE EFFLUENT MONITORING INSTRUMENTATION

No effluent monitoring instrumentation was INOPERABLE for periods of 30 days or more during this reporting period.

HUMBOLDT BAY POWER PLANT UNIT 3

CHANGES TO THE

SAFSTOR OFFSITE DOSE CALCULATION MANUAL

DURING 2001

(Procedure Classification - Quality Related)

INTRODUCTION

The SAFSTOR Off-site Dose Calculation Manual (ODCM) is provided to support implementation of the Humboldt Bay Power Plant (HBPP) Unit 3 radiological effluent controls and radiological environmental monitoring. The ODCM is divided into two parts, Part I - Specifications and Part II - Calculational Methods and Parameters.

Part I contains the specifications for liquid and gaseous radiological effluents (RETS) developed in accordance with NUREG-0473, *Draft Radiological Effluent Technical Specifications - BWR*, by License Amendment Request (LAR) 96-02 and the radiological environmental monitoring program (REMP). Both the RETS and the REMP were relocated from the Technical Specifications by LAR 96-02 in accordance with the provisions of Generic Letter 89-01, *Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program*, issued by the NRC in January, 1989.

Implementation of the LAR revised the instantaneous liquid concentration limits based on "old" 10 CFR 20 maximum permissible concentrations (MPCs) to 10 times the "new" 10 CFR 20, Appendix B, Table 2, Column 2 effluent concentration limits (ECLs) and replaced the gaseous effluent instantaneous concentration limits at the site boundary with annual dose rate limits equating to the doses associated with the annual average concentrations of "old" 10 CFR 20, Appendix B, Table II, Column 1. The LAR also established limits for doses to members of the public from radiological effluents based on the as low as reasonably achievable (ALARA) design objectives of 10 CFR 50, Appendix I as applicable to a nuclear power plant which has been shut down in excess of 20 years and is in SAFSTOR Decommissioning. These dose limits were established following the guidance of NUREG-0133, *Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants*, and NUREG-0473. This guidance was modified, as appropriate, to reflect the SAFSTOR decommissioning licensing basis contained in the HBPP SAFSTOR Decommissioning Plan, the Environmental Report submitted as Attachment 6 to the HBPP SAFSTOR licensing amendment request and NUREG-1166, *HBPP Final Environmental Statement.*

The ODCM contains the REMP required by Technical Specification VII.G. This program consists of monitoring stations and sampling programs based on the SAFSTOR Decommissioning Plan and the Environmental Report which established baseline conditions for soil, biota and sediments. The REMP also includes requirements to participate in an interlaboratory comparison program.

Part II of the ODCM contains the calculational methods developed, following the above guidance, to be used in determining the dose to members of the public resulting from routine radioactive effluents released from HBPP during the SAFSTOR period. Part II also contains the methodology used to determine effluent monitor alarm/trip setpoints which assure that releases of radioactive materials remain within specified concentrations.

The ODCM also contains the Process Control Program (PCP) for solid radioactive wastes which is required by Technical Specification VII.E.1.j. The ODCM also contains administrative controls regarding the content of the Annual Radiological Environmental Monitoring Report and the Annual Radioactive Effluent Release Report which are required by Technical Specifications VII.J.1 and VII.J.3 and administrative controls regarding major changes to radioactive waste treatment systems.

The ODCM shall become effective after review by the Plant Staff Review Committee and approval by the Plant Manager in accordance with Technical Specification Section VII.O. Changes to the ODCM shall be documented and records of reviews performed shall be retained. This documentation shall contain sufficient information to support the change (including analyses or evaluations), and a determination that the change will maintain the level of radioactive effluent control required by the regulations listed in Technical Specification VII.F and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

Changes shall be submitted to the NRC in the form of a complete and legible copy of the entire ODCM as part of, or concurrent with, the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed.

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PART I - SPECIFICATIONS

1.0 DEFINITIONS

1.1 ACTION

ACTION shall be that part of a control that prescribes remedial measures required under designated conditions.

1.2 BASELINE COMPARISON

A BASELINE COMPARISON shall be a comparison of cumulative radioactivity releases for a stated period with the baseline radioactivity release conditions established by the ENVIRONMENTAL REPORT.

1.3 CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

1.4 CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

1.5 CHANNEL FUNCTIONAL TEST

a. Analog channels - one injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY including required alarms, interlocks, display, and trip functions.

b. Bistable channels - the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including alarm and trip functions.

1.6 ENVIRONMENTAL REPORT

Submitted as Attachment 6 to the SAFSTOR license amendment request, the ENVIRONMENTAL REPORT established baseline radiological environmental conditions for soil, biota and sediments. In accordance with the NRC approved SAFSTOR Decommissioning Plan, these baseline conditions will only need to be reestablished prior to DECON if a significant release during SAFSTOR occurs as the result of an accident.

1.7 FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1-1.

1.8 INDEPENDENT VERIFICATION

INDEPENDENT VERIFICATION is a separate act of confirming or substantiating that an activity or condition has been completed or implemented, in accordance with specified requirements, by an individual not associated with the original determination that the activity or condition was completed or implemented in accordance with specified requirements.

1.9 INSTANTANEOUS CONCENTRATION

INSTANTANEOUS CONCENTRATION is the concentration averaged over one hour of radioactive materials in effluents.

1.10 LIQUID RADWASTE TREATMENT SYSTEM

The LIQUID RADWASTE TREATMENT SYSTEM shall be any available equipment (e.g., filters, evaporators, demineralizers, or contractor services) capable of reducing the quantity of radioactive material, in liquid effluents, prior to discharge.

1.11 MEMBER OF THE PUBLIC

MEMBER OF THE PUBLIC means an individual in any area located beyond the boundary of the restricted area controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials and within, at, or beyond the SITE BOUNDARY. However, an individual is not a member of the public during any period in which the individual receives an onsite occupational dose.

1.12 OFFSITE DOSE CALCULATION MANUAL

The OFFSITE DOSE CALCULATION MANUAL contains the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm Trip Setpoints, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM also contains the Radioactive Effluent Controls and Radiological Environmental Monitoring Program and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports. The ODCM also contains the Process Control Program (PCP) for solid radioactive wastes which is required by Technical Specification VII.E.1.j.

1.13 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). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, 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).

1.14 PROCESS CONTROL PROGRAM

The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

1.15 PURGE - PURGING

PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

1.16 SITE BOUNDARY

The SITE BOUNDARY shall be the boundary of the unrestricted area used in the offsite dose calculations for gaseous and liquid effluents as defined in Technical Specification II.B.

1.17 SOLIDIFICATION

SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a homogeneous (uniformly distributed), monolithic, immobilized solid with definite volume and shape, bounded by a stable surface of distinct outline on all sides (free-standing).

1.18 SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.19 UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area located beyond the boundary of the restricted area controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials and within, at, or beyond the SITE BOUNDARY.

1.20 URANIUM FUEL CYCLE

As defined in 40 CFR Part 190.02(b), "URANIUM FUEL CYCLE means the operations of milling of uranium ore, chemical conversion of uranium, isotopic enrichment of uranium, fabrication of uranium fuel, generation of electricity by a light-water-cooled nuclear power plant using uranium fuel, and reprocessing of spent uranium fuel, to the extent that these directly support the production of electrical power for public use utilizing nuclear energy, but excludes mining operations, operations at waste disposal sites, transportation of any radioactive material in support of these operations, and the reuse of recovered non-uranium special nuclear and by-product materials from the cycle."

1.21 VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is 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 release to the environment (such a system is not considered to have any effect on noble gas effluents).

1.22 VENTING

VENTING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during VENTING. Vent, used in system names, does not imply a VENTING process.

Table 1-1
FREQUENCY NOTATION

Notation	Frequency	[1]Extension Period
D	At least once per 24 hours.	None
W	At least once per 7 days.	42 hours
M	At least once per 31 days.	7 days
Q	At least once per 92 days.	22 days
SA	At least once per 184 days.	45 days
A	At least once per 365 days.	91 days
P	Completed prior to each release.	
N.A.	Not applicable.	

[1]The extension period for a frequency of a week or longer is 25% with a maximum tolerance of 325% for three consecutive periods.

2.0 SPECIFICATIONS

2.1 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITIONS

2.1.1 The radioactive liquid effluent monitoring instrumentation channels shown in Table 2-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 2.3 are not exceeded.

APPLICABILITY: At all times

ACTION:

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required above, without delay suspend the release of radioactive liquid effluents monitored by the affected channel, or change the setpoint so that it is acceptably conservative, or declare the channel inoperable.

b. With one or more radioactive liquid effluent monitoring instrumentation channels inoperable, take the ACTION shown in Table 2-1. For the instrumentation covered by items 1 and 2 of the table, exert best efforts to return the inoperable instrument(s) to OPERABLE status within 30 days. If the affected instrument(s) cannot be returned to OPERABLE status within 30 days, provide information on the reasons for inoperability and lack of timely corrective action in the next Radioactive Effluent Release Report.

SURVEILLANCE REQUIREMENTS

2.1.2 Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 2-2.

Table 2-1
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels OPERABLE	ACTION
1. Gross Radioactivity Monitors Providing Automatic Termination of Release		
a. Process Water Monitor	1	21
2. Flow Rate Measurement Devices		
a. None		

Table Notation

ACTION 21 With less than the required number of OPERABLE channels, 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 2.3.1, and

b. An INDEPENDENT VERIFICATION of release rate calculations is performed, and

c. An INDEPENDENT VERIFICATION of discharge valve lineup is performed.

Otherwise, suspend releases of radioactive materials via this pathway.

Table 2-2
RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release				
a. Process Water Monitor	D	Q	A	Q(1)(2)
2. Flow Rate Measurement Devices				
a. None				

Table Notation

(1) Alarm functions and background readings shall be checked weekly. If a background reading exceeds the equivalent of 5×10^{-5} micro-Ci/ml of Cs-137, the cause will be investigated and remedial measures taken to reduce the background reading.

(2) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the alarm setpoint.

b. Circuit failure.

c. Instrument indicates a downscale failure.

2.2 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITIONS

2.2.1 The radioactive gaseous effluent monitoring instrumentation channels shown in Table 2-3 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of specification 2.6 are not exceeded.

APPLICABILITY: Whenever the ventilation system is in operation.

ACTION:

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than required above, without delay suspend the release of radioactive gaseous effluents monitored by the affected channel, or change the setpoint so that it is acceptably conservative, or declare the channel inoperable.

b. With one or more radioactive gaseous effluent monitoring instrumentation channels inoperable, take the ACTION shown in Table 2-3. For the instrumentation covered, exert best efforts to return the inoperable instrument(s) to OPERABLE status within 30 days. If the affected instrument(s) cannot be returned to OPERABLE status within 30 days, provide information on the reasons for inoperability and lack of timely corrective action in the next Radioactive Effluent Release Report.

SURVEILLANCE REQUIREMENTS

2.2.2 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 2-4.

Table 2-3
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Instrument	Minimum Channels OPERABLE	ACTION
1. Stack Gas Monitoring System		
a. Noble Gas Activity Monitor	1	23,24
b. Iodine Sampler*	N.A.	
c. Particulate Sampler	1	23,25
d. Effluent System Flow Rate Monitor	1	26
e. Sampler Flow Rate Monitor**	1	

Table Notation

ACTION 23 The monitor may be taken out of service for calibration or maintenance, but shall be returned to service as soon as practicable within the 30 day period allowed by ACTION 2.2.1.b.

ACTION 24 With the number of channels OPERABLE less than that required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 12 hours and these samples are analyzed for noble gas activity within 24 hours.

ACTION 25 With the number of channels OPERABLE less than that required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided samples are continuously collected as required in Table 2-6.

ACTION 26 With the number of channels OPERABLE less than that required by the Minimum Channels OPERABLE requirement, the effluent system default flow rate may be used for effluent calculations.

* Not included in the stack gas monitoring system.

** Loss of sampler flow would result in alarm and failure of Noble Gas activity monitor and particulate sampler.

Table 2-4
RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Instrument	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Stack Gas Monitoring System				
a. Noble Gas Activity Monitor	D	M	A	Q(1)
b. Iodine Sampler*	N.A.	N.A.	N.A.	N.A.
c. Particulate Sampler	W	N.A.	N.A.	N.A.
d. Effluent System Flow Rate Monitor	W	N.A.	A	N.A.
e. Sampler Flow Rate Monitor	Q	N.A.	N.A.	N.A.

Table Notation

(1) The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the alarm setpoint .

b. Circuit failure.

c. Instrument indicates a downscale failure. **

* Although this sampler is normally required for nuclear plant monitoring, it is not required or included in the HBPP stack gas monitoring system due to the long decay time since operation.

** Although this is a normal requirement of the CHANNEL FUNCTIONAL TEST for operating plants, no downscale failure indication is provided on this instrument at HBPP, and downscale failure indication is not required for the monitor to be OPERABLE.

2.3 LIQUID EFFLUENT - CONCENTRATION

LIMITING CONDITIONS

2.3.1 The instantaneous concentration of radioactive material released beyond the SITE BOUNDARY shall be less than or equal to 10 times the concentrations specified in 10 CFR Part 20, Appendix B, Table 2, Column 2.

APPLICABILITY: At all times.

ACTION:

With the instantaneous concentration of radioactive materials released beyond the SITE BOUNDARY exceeding the above limits, without delay restore the concentration of radioactive materials being released beyond the SITE BOUNDARY to within the above limits.

SURVEILLANCE REQUIREMENTS

2.3.2 Radioactive liquid wastes shall be sampled and analyzed in accordance with the sampling and analysis program of Table 2-5.

2.3.3 The results of the radioactivity analyses shall be used with the calculational methods in Part II of the ODCM to assure that the concentrations of radioactive material released to Humboldt Bay are maintained within the limits of Specification 2.3.1.

Table 2-5
RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (µCi/ml)[a]
A. Batch Waste Release Tanks[c] 1. Treated Waste Hold Tank(2) 2. Waste Receiver Tanks(3)	P Each Batch	P Each Batch	Principal Gamma Emitters[e]	5×10^{-7}
	P Each Batch	M Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[b]	Sr-90	5×10^{-8}
B. Plant Continuous Releases[d] 1. Caisson Sump	D Grab Sample	W Composite[b]	Principal Gamma Emitters[e]	5×10^{-7}
	D Grab Sample	M Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	D Grab Sample	Q Composite[b]	Sr-90	5×10^{-8}

Table Notation

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

* For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\, s_b}{(E)(V)\left(2.22 \times 10^6\right)\left(e^{-\lambda \Delta t}\right) Y}$$

Where:

LLD is the lower limit of detection as defined above (as microcurie per unit mass or volume),

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

E is the counting efficiency (as counts per disintegration),

Table 2-5 (Continued)

Table Notation (Continued)

E is the counting efficiency (as counts per disintegration),

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

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

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

Typical values of E, V, Y, and Δt shall be used in the calculation.

The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

b 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 which is representative of the liquids released.

c 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 assure representative sampling.

d A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume or system that has an input flow during the continuous release.

e The principal gamma emitters for which the LLD specification applies exclusively are Co-60 and Cs-137. This does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are not detected for the analyses shall be reported as "less than" the nuclide's LLD, and shall not be reported as being present at the LLD level for that nuclide. The "less than" values shall not be used in the required dose calculations.

2.4 LIQUID EFFLUENT - DOSE

LIMITING CONDITIONS

2.4.1 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released beyond the SITE BOUNDARY shall be limited as follows:

a. 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.

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

APPLICABILITY: At all times.

ACTION:

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, a Special Report pursuant to Administrative Control 4.3, which includes:

a. Identification of the cause for exceeding the limit(s);

b. Corrective action taken to reduce the release of radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the remainder of the current calendar year so that the dose or dose commitment to a MEMBER OF THE PUBLIC from this source is less than or equal to 3 mrem total body and less than or equal to 10 mrem to any organ during the calendar year.

SURVEILLANCE REQUIREMENTS

2.4.2 At least once per 31 days, perform a BASELINE COMPARISON for liquid effluent radioactivity released to date for the current calendar quarter and current calendar year. IF the comparison indicates that the activity released to date exceeds the Environmental Report baseline annual release, THEN a dose calculation shall be performed for the current calendar quarter and the current calendar year.

2.5 LIQUID WASTE TREATMENT

LIMITING CONDITIONS

2.5.1 The LIQUID RADWASTE TREATMENT SYSTEM shall be used, as appropriate, to reduce radioactive materials in liquid wastes prior to their discharge, when projected monthly doses due to liquid effluents discharged to Humboldt Bay would exceed the action levels of 0.06 mrem whole body or 0.2 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

When radioactive liquid waste, in excess of the above action levels, is discharged without prior treatment, prepare and submit to the Commission within 30 days, a Special Report pursuant to Administrative Control 4.3, which includes the following information:

a. Identification of inoperable equipment and the reasons for inoperability.

b. Actions taken to restore the inoperable equipment to OPERABLE status.

c. Actions taken to prevent recurrence.

SURVEILLANCE REQUIREMENTS

2.5.2 Before approving any release, perform a BASELINE COMPARISON for liquid effluent radioactivity released (or projected to be released) during the 31 day period prior to and including the projected release. IF the comparison indicates that the activity released will exceed the Environmental Report baseline monthly release, THEN a dose calculation shall be performed for comparison with Specification 2.5.1.

2.6 GASEOUS EFFLUENTS - DOSE RATE

LIMITING CONDITIONS

2.6.1 The dose rate at or beyond the SITE BOUNDARY, due to radioactive materials released in gaseous effluents, shall be limited as follows:

a. Noble gases: less than or equal to 500 mrem/year total body and less than or equal to 3000 mrem/year to the skin.

b. Tritium and radioactive particulates with half-lives of greater than 8 days: less than or equal to 1500 mrem/year to any organ.

APPLICABILITY: At all times.

ACTION:

With dose rate(s) exceeding the above limits, without delay decrease the dose rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

2.6.2 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits. This determination has been established by the Environmental Report.

2.6.3 The dose rate due to radioactive materials specified above, other than noble gases, in gaseous effluents shall be determined to be within the above limits by obtaining representative samples and performing analyses in accordance with Table 2-6 and comparing cumulative activity released with the Environmental Report baseline conditions.

Table 2-6
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
Plant Stack	Q[b] Grab Sample	Q[b]	Noble Gas (Kr-85)	1×10^{-4}
	Continuous[d]	W[c] Particulate Sample	Principal Gamma Emitters[e]	1×10^{-11}
	Continuous[d]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[d]	Q Composite Particulate Sample	Sr-90	1×10^{-11}
	Continuous[d]	Continuous Noble Gas Monitor	Noble Gas Gross Beta	1×10^{-6}

Table Notation

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

* For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\, s_b}{(E)(V)\left(2.22 \times 10^6\right)\left(e^{-\lambda \Delta t}\right) Y}$$

Where:

LLD is the lower limit of detection as defined above (as microcurie per unit mass or volume),

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

Table 2-6 (Continued)

Table Notation (Continued)

E is the counting efficiency (as counts per disintegration),

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

2.22×10^6 is the number of disintegrations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

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

Typical values of E, V, Y, and Δt shall be used in the calculation.

The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as an a posteriori (after the fact) limit for a particular measurement.

[b] Analyses shall also be performed following an occurrence which could alter the mixture of radionuclides.

[c] Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler).

[d] 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 the Specifications 2.6, 2.7, and 2.8.

[e] The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-85 for gaseous emissions and Co-60 and Cs-137 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are not detected for the analyses shall be reported as "less than" the nuclide's LLD, and shall not be reported as being present at the LLD level for that nuclide. The "less than" values shall not be used in the required dose calculations.

2.7 GASEOUS EFFLUENTS: DOSE - NOBLE GASES

LIMITING CONDITIONS

2.7.1 The air dose at or beyond the SITE BOUNDARY due to radioactive noble gases released in gaseous effluents shall be limited to :

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

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

APPLICABILITY: At all times.

ACTION:

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, a Special Report, pursuant to Administrative Control 4.3, which includes:

a. Identification of the cause for exceeding the limit(s).

b. Corrective action taken to reduce the release of radioactive noble gases in gases effluents during the remainder of the current calendar quarter and during the remainder of the current calendar year so that the average dose during the calendar year is less than or equal to 10 mrad gamma and 20 mrad beta radiation.

SURVEILLANCE REQUIREMENTS

2.7.2 Compliance with these Specifications for normal SAFSTOR conditions has been established on a licensing basis by the Environmental Report and NUREG-1166, *Final Environmental Statement for Decommissioning Humboldt Bay Power Plant, Unit No. 3*, issued by the NRC. If an accident involving spent fuel occurs during the SAFSTOR period, the noble gas activity released in gaseous effluents shall be compared with the baseline conditions established by the Environmental Report. IF the comparison indicates that the activity released will exceed the Environmental Report baseline release, THEN a dose calculation shall be performed.

2.8 GASEOUS EFFLUENTS: DOSE - TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITIONS

2.8.1 The dose to a MEMBER OF THE PUBLIC from the release of tritium and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released beyond the SITE BOUNDARY shall be limited as follows:

a. During any calendar quarter: less than or equal to 7.5 mrem to any organ, and

b. During any calendar year: less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times .

ACTION:

With the calculated dose from the release of tritium and radioactive materials 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, a Special Report, pursuant to Administrative Control 4.3, which includes:

a. Identification of the cause for exceeding the limit(s).

b. Corrective action taken to reduce the release of tritium and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents during the remainder of the current calendar quarter and during the remainder of the current calendar year so that the average dose to any organ is less than or equal to 15 mrem.

SURVEILLANCE REQUIREMENTS

2.8.2 At least once per 31 days, perform a BASELINE COMPARISON for gaseous effluent radioactivity (tritium and particulate form) released to date for the current calendar quarter and current calendar year. IF the comparison indicates that the activity released to date exceeds the Environmental Report baseline annual release, THEN a dose calculation shall be performed for the current calendar quarter and the current calendar year.

2.9 SOLID RADIOACTIVE WASTE

LIMITING CONDITIONS

2.9.1 The solid radwaste system shall be used in accordance with a PROCESS CONTROL PROGRAM to process wet radioactive wastes to meet shipping and burial ground requirements.

APPLICABILITY: At all times.

ACTION:

With the provisions of the a PROCESS CONTROL PROGRAM not satisfied, suspend shipments of defectively processed or defectively packaged solid radioactive wastes from the site.

SURVEILLANCE REQUIREMENTS

2.9.2 The PROCESS CONTROL PROGRAM, as defined in Section 1.0, shall be used to verify that processed wet radioactive wastes (e.g., filter sludges, spent resins and evaporator bottoms) meet the shipping and burial ground requirements with regard to solidification and dewatering.

2.10 TOTAL DOSE

LIMITING CONDITIONS

2.10.1 The 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 less than or equal to 25 mrem to the total body or any organ (except the thyroid, which shall be limited to less than or equal to 75 mrem).

APPLICABILITY: At all times.

ACTION:

With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 2.4.1.a, 2.4.1.b, 2.7.1.a, 2.7.1.b, 2.8.1.a, or 2.8.1.b, calculations should be made, which include direct radiation contributions from the reactor, to determine whether the above limits of Specification 2.10 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Administrative Control 4.3, 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 Part 20.2203, 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 release(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 considered granted until staff action on the request is complete.

SURVEILLANCE REQUIREMENTS

2.10.2 DOSE CALCULATIONS - Annual dose contributions from liquid and gaseous effluents shall be calculated in accordance with dose calculation methodology provided for Specifications 2.4.1, 2.7.1, and 2.8.1.

2.11 REMP MONITORING PROGRAM

LIMITING CONDITIONS

2.11.1 The radiological environmental monitoring program shall be conducted as specified in Table 2-7.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 2-7, prepare and submit to the Commission, in the Annual Radiological Environmental Monitoring Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity, resulting from plant effluents, in an environmental sampling medium exceeding the reporting levels of Table 2-8 when averaged over any calendar quarter, prepare and submit to the Commission, within 30 days of obtaining analytical results from the affected sampling period, a Special Report pursuant to Administrative Control 4.3, which includes an evaluation of any release conditions, environmental factors or other aspects which caused the limits of Table 2-8 to be exceeded. When more than one of the radionuclides in Table 2-8 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Monitoring Report.

When radionuclides other than those in Table 2-8 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is greater than or equal to the calendar year limits of Specifications 2.4, 2.7, and 2.8. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Monitoring Report.

2.11 REMP MONITORING PROGRAM - Continued

SURVEILLANCE REQUIREMENTS

2.11.2 The radiological environmental monitoring samples shall be collected pursuant to Table 2-7 from the locations given in Tables 2-7 and 2-10 and Figures 2-1, 2-2, 2-3, 2-4 and 2-5 and shall be analyzed pursuant to the requirements of Tables 2-7 and 2-9.

Table 2-7
HBPP RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	PROGRAM DESCRIPTION: Number of Samples and Locations[a]	PROGRAM DESCRIPTION: Sampling and Collection Frequency	PROGRAM DESCRIPTION: Type of Analysis	PROGRAM BASIS: ODCM Specs (QR)	PROGRAM BASIS: State of California (NQR)	PROGRAM BASIS: PG&E/HBPP Elective (NQR)
AIRBORNE	1 offsite location	Continuous sampler operation with sample collection at least once per 7 days[1]	Gross beta radioactivity following filter change[2] Gamma isotopic[c] analysis on quarterly composite (by station)[2]			X
DIRECT RADIATION[b]	16 onsite stations with TLDs	TLDs exchanged quarterly[1]	Gamma exposure[3]	X		
	1 offsite control station with TLD	TLDs exchanged quarterly[1]	Gamma exposure[3]			X
	4 offsite stations with TLDs	TLDs exchanged quarterly[1]	Gamma exposure[3]	X	X	
	24 offsite stations with TLDs	TLDs exchanged quarterly[1]	Gamma exposure[3]			X
WATERBORNE Surface Water	Discharge canal effluent	Continuous sampler operation with sample collection weekly[1] Dip samples if sampler inoperable[1]	Gamma isotopic[c] and Tritium analysis of weekly sample[2] Sample submitted to the State Department of Health Services monthly[1]	X	X	
Groundwater	5 groundwater spent fuel pool monitoring wells	Quarterly[1]	Tritium and gamma isotopic[c] analysis[2]	X		
			Alpha and Beta Analysis			X
Sediment	3 locations located in Humboldt Bay	Quarterly[4]	Gamma isotopic[c] analysis[2]			X
Algae	3 stations located in Humboldt Bay	Quarterly, subject to availability[4]	Gamma isotopic[c] analysis[2]			X

Table 2-7 (Continued)

Exposure Pathway and/or Sample	PROGRAM DESCRIPTION Number of Samples and Locations[(a)]	Sampling and Collection Frequency	Type of Analysis	PROGRAM BASIS ODCM Specs (QR)	State of California (NQR)	PG&E/HBPP Elective (NQR)
INGESTION						
Milk	Pedrotti Dairy	Annually[(1)]	Gamma isotopic[(c)] analysis[(2)]			X
	Holgerson Dairy	Annually[(1)]	Gamma isotopic[(c)] analysis[(2)]			X
Fish and Invertebrates	1 sample of fish from Station 55	Quarterly, subject to availability[(4)]	Gamma isotopic[(c)] analysis[(2)]			X
	1 sample of clams from Station 59	Quarterly, subject to availability[(4)]	Gamma isotopic[(c)] analysis[(2)]			X
	1 sample of oysters from Station 65	Quarterly, subject to availability[(4)]	Gamma isotopic[(c)] analysis[(2)]			X
TERRESTRIAL						
Soil	2 locations, one near the plant and one from a control location	Quarterly[(4)]	Gamma isotopic[(c)] analysis[(2)]			X

Table Notations

QR - Quality Related
NQR - Non-Quality Related

(1)Performed by HBPP
(2)Performed by TES
(3)Performed by DCPP
(4)Performed by Humboldt State University

(a) Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment and other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the quality-related sampling schedule shall be documented in the Annual Radiological Environmental Monitoring Report. It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances, suitable specific alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the REMP, and submitted in the next Annual Radioactive Effluent Release Report, including a revised figure(s) and table for the REMP reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples for that pathway and justifying the section of the new location(s) for obtaining samples. Note: This reporting requirement applies only to the quality-related portion of the REMP.

(b) For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges should not be used as dosimeters for measuring direct radiation.

(c) Gamma isotopic analysis means the identification and quantification of gamma emitting radionuclides that may be attributable to the effluents from the facility.

Table 2-8
REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Analysis	Water (pCi/L)
H-3	20,000*
Co-60	300
Cs-137	50

* For drinking water samples. This is the 40CFR141 value. If no drinking water pathway exists, a value of 30,000 pCi/L may be used.

Table 2-9
DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a] [b]
LOWER LIMITS OF DETECTION (LLD)[c]

Analysis	Water (pCi/L)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/L)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000[d]					
Co-60	15		130			
Cs-137	18	0.06	150	18	80	180

Table Notations

[a] This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report.

[b] Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13, Revision 1, July 1977.

[c] The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66 S_b}{E \times V \times 2.22 \times Y \times \exp(-\lambda \Delta t)}$$

Where:

LLD = the "a priori" lower limit of detection as defined above (as pCi 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 (as counts per minute)

Table 2-9 (Continued)

Table Notations (Continued)

E = the counting efficiency (as counts per transformation)

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

2.22 = the number of transformations per minute per pico-Curie

Y = the fractional radiochemical yield (when applicable)

λ = the radioactive decay constant for the particular radionuclide

Δt = the elapsed time between sample collection (or end of the sample collection period) and time of counting

The value of S_b used in the calculation of the LLD for a detection system will 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. In calculating the LLD for a radionuclide determined by gamma ray spectrometry, the background will include the typical contributions of other radionuclides normally present in the samples (e.g., potassium 40 in milk samples).

Analyses will be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Environmental Radiological Operating Report.

Typical values of E, V, Y and t should be used in the calculation. It should be recognized that the LLD is defined as a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.

(d) For surface water samples, a value of 3000 pCi/l may be used.

Table 2-10
DISTANCES AND DIRECTIONS TO ENVIRONMENTAL MONITORING STATIONS

Station No.	Code	Station Name	Radial Direction Sector	Radial Direction By Degrees	Radial Distance from Plant (Miles)
***1**	ΔO	**King Salmon Picnic Area**	**W**	**270**	**0.3**
***2**	Δ	**180 Dinsmore Drive, Fortuna**	**SSE**	**158**	**9.4**
3	Δ□	Humboldt Hill Road at Bret Harte Lane	SSE	158	0.9
4	Δ	Wood and K Street, Eureka	NNE	42	4.0
5	O	Redwood Avenue, Arcata	NE	45	12.3
6	Δ	Table Bluff and Clough Road	S	180	5.7
7	Δ	College of the Redwoods	S	180	2.6
8	Δ	Humboldt Hill Road near TV Station	SSE	170	1.8
9	Δ	2376 Harbor View Drive	SSE	165	1.6
10	Δ	B Street, Fields Landing	SSW	200	1.2
11	Δ	Whittier Court & Irving Humboldt Hill	SSE	175	1.1
12	Δ	Bell Hill Road and Sauters	SSW	195	0.7
***14**	Δ	**South Bay School Parking Lot**	**S**	**180**	**0.4**
16	ΔO	Elk River Road/PG&E Gas Reg/Pedrotti Dairy	ENE	72	1.4
17	Δ	Bassford Road at Grauer's Lane	E	90	2.0
18	Δ	6418 Elk River Road	ESE	112	2.0
19	Δ	5399 Noe Avenue	NE	45	1.9
21	ΔO	PG&E Well 2, HH Road	ESE	128	0.5
22	Δ	Station B - 14th Street	NNE	23	4.0
24	Δ	Pole at 7th and L Street	NNE	32	5.0
***25**	Δ	**Irving Drive, Humboldt Hill**	**SSE**	**175**	**1.3**
27	Δ	6700 Berta Road	ESE	125	1.9
28	Δ	7200 Berta Road	SSE	142	2.1
29	Δ	Vista Road, Humboldt Hill	SSE	148	1.5
31	Δ	King Salmon Road East of Freeway	SSE	170	0.4
32	Δ	Loma Road and Volpis	SSW	185	0.5
33	ΔO	110 kV Line No. 1 Well	ESE	110	0.1
34	Δ	King Salmon Road and RR Track	SSW	185	0.3
36	Δ	Plant Entrance Road	WSW	230	0.2
45	Δ	Humboldt Substation (T17)	ENE	61	5.9
48	O	Holgerson Dairy	S	180	5.1
55	O	HBPP Outfall Canal	NNW	338	0.1
56	O	1000 ft North of Outfall Canal Discharge	NE	45	0.2
57	O	1000 ft South of Outfall Canal Discharge	W	270	0.2
59	O	Hookton Channel	SW	225	0.8
65	O	Coast Oyster Company	NNE	23	4.6

Table Notations

Code: Δ Dosimetry Station □ Air Particulate Station O Biological Station
Note: ***Quality Related Station**

Figure 2-1
HBPP ONSITE TLD LOCATIONS

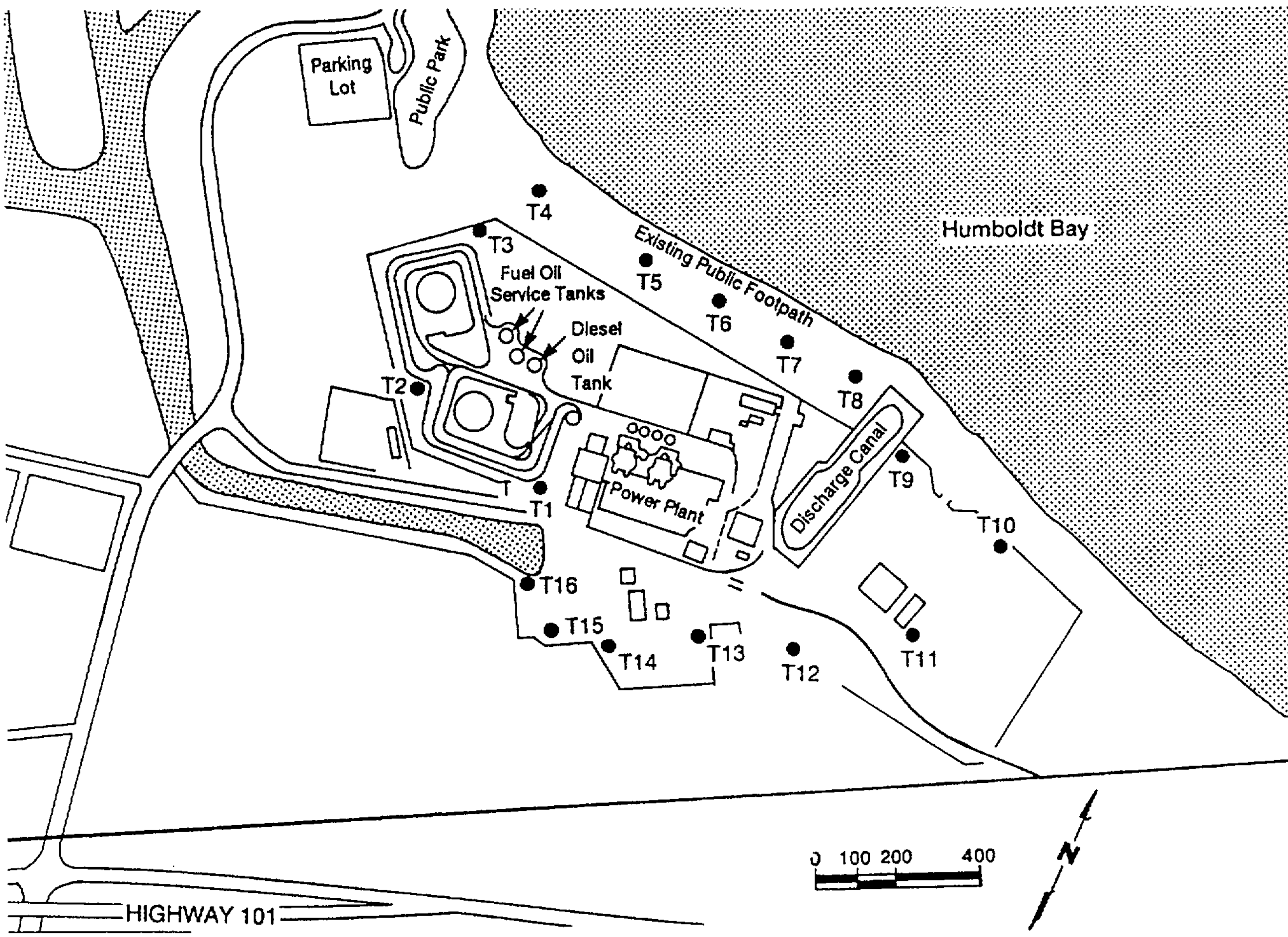


Figure 2-2
HBPP ONSITE MONITORING WELL LOCATIONS

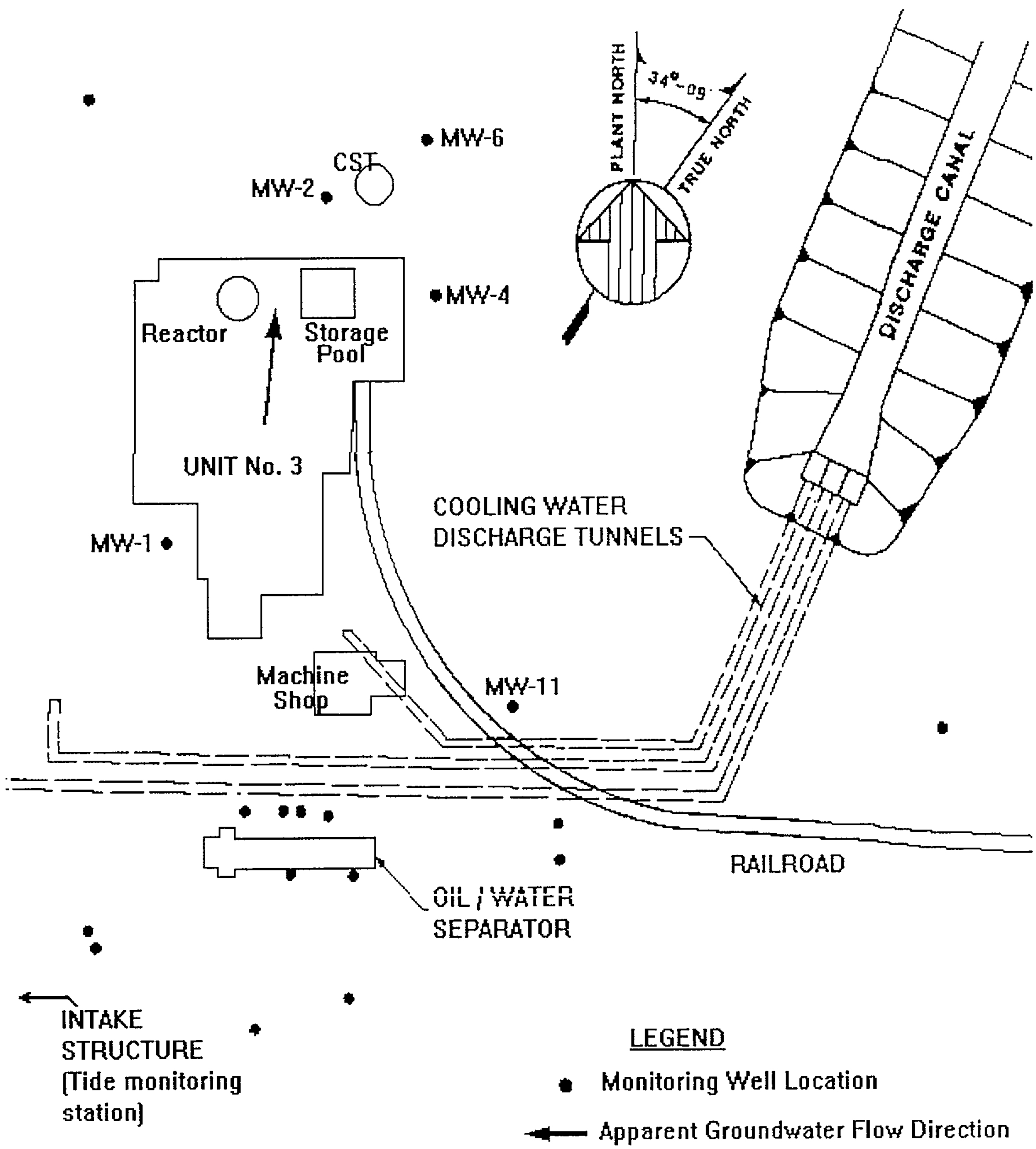


Figure 2-3
HBPP OFFSITE SAMPLING LOCATIONS

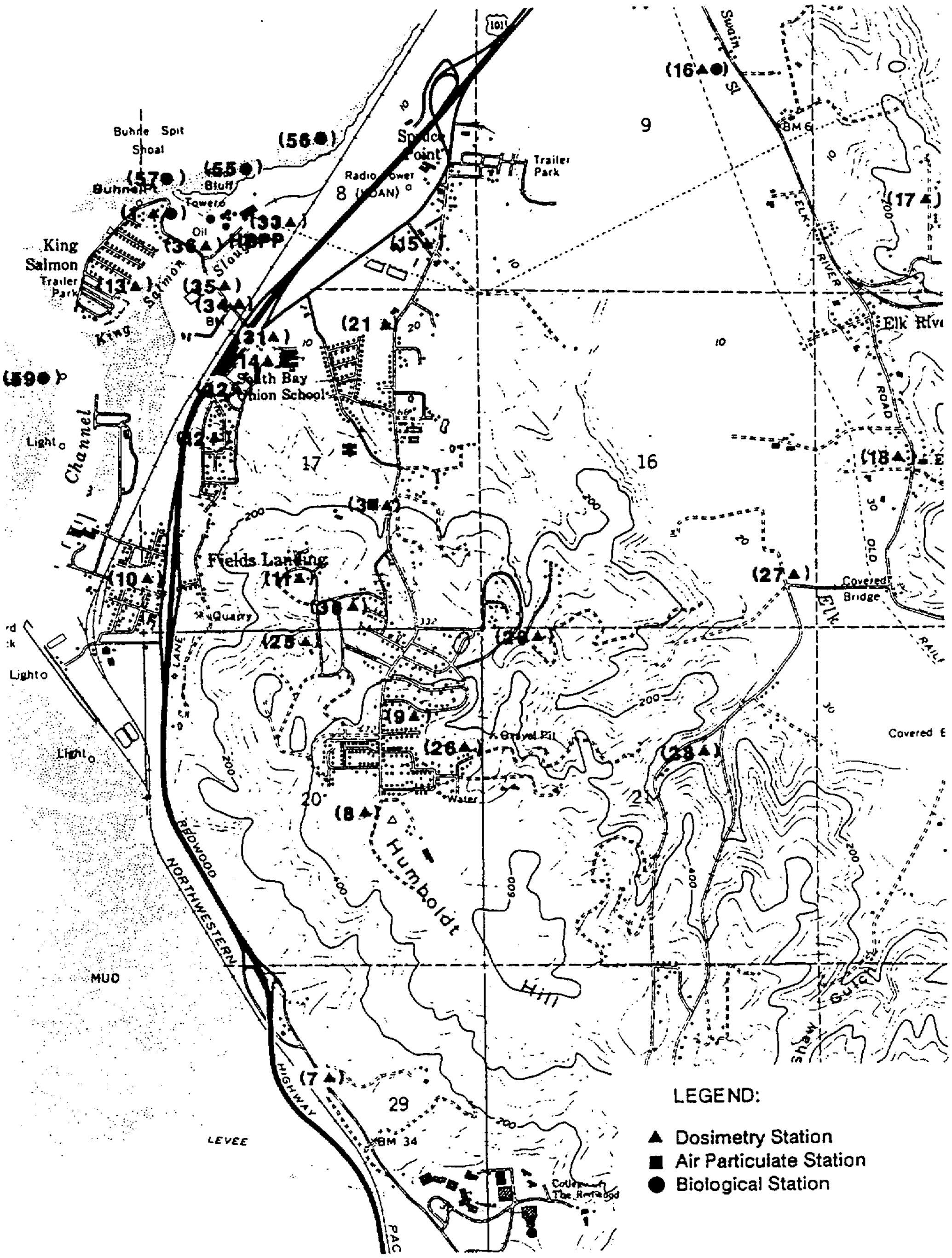


Figure 2-4
HBPP OFFSITE SAMPLING LOCATIONS (CONTINUED)

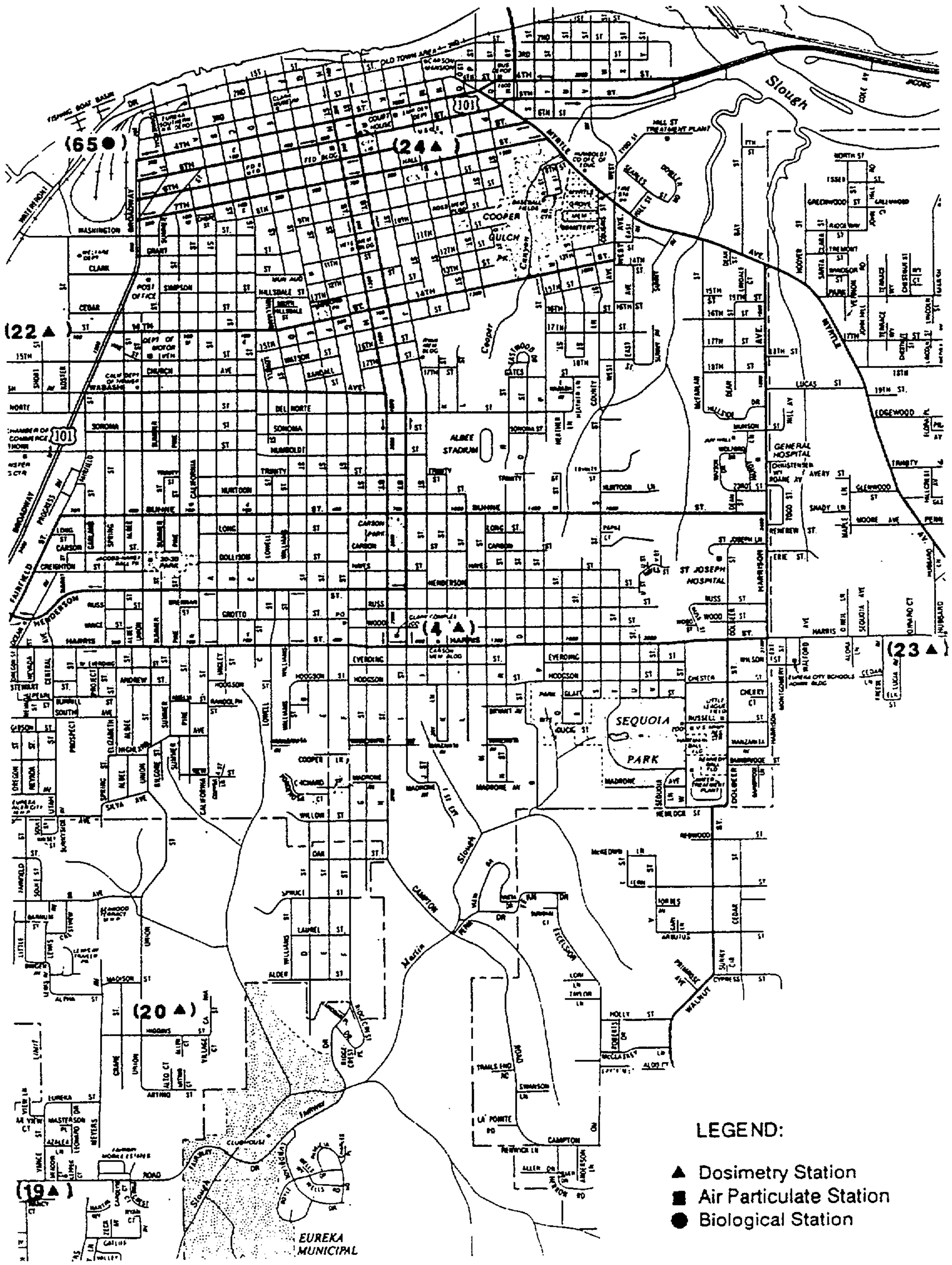
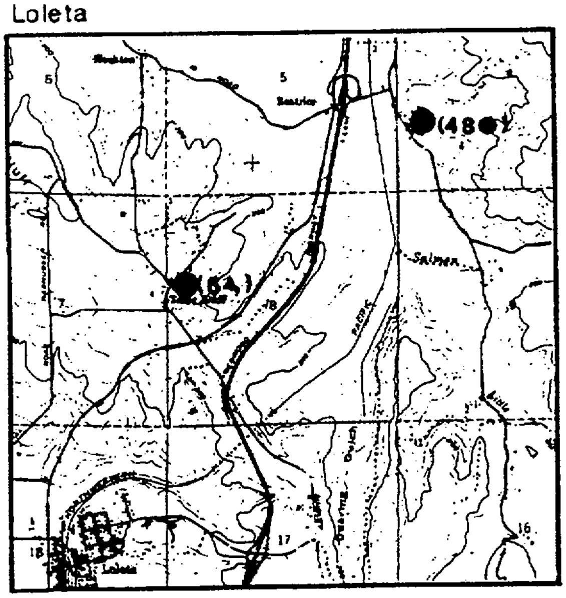
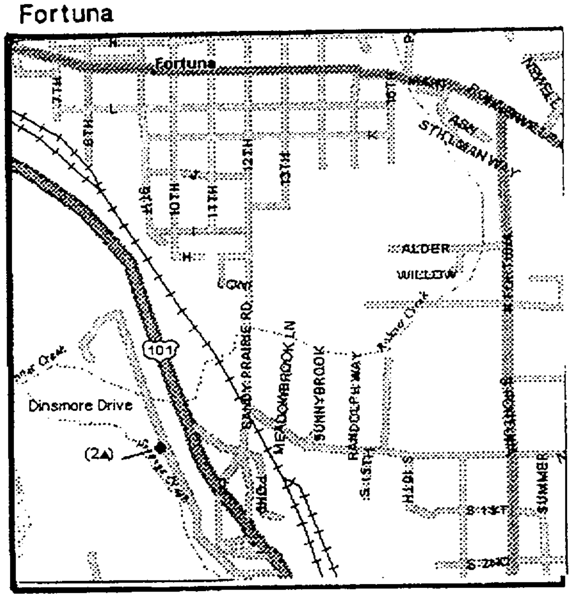


Figure 2-5
HBPP OFFSITE SAMPLING LOCATIONS (CONTINUED)

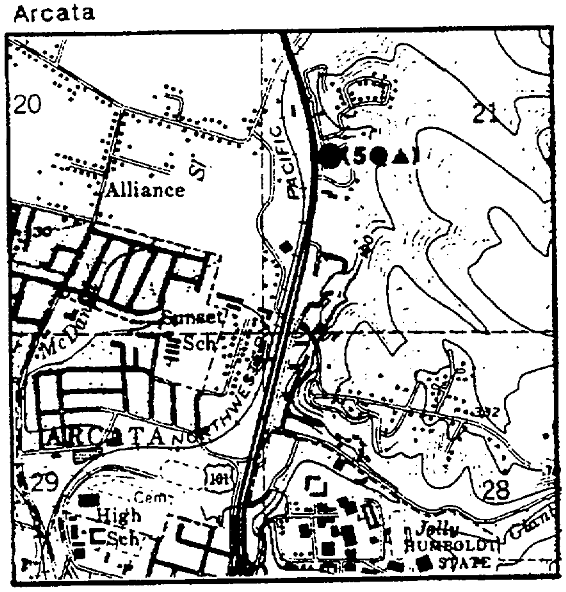
Loleta



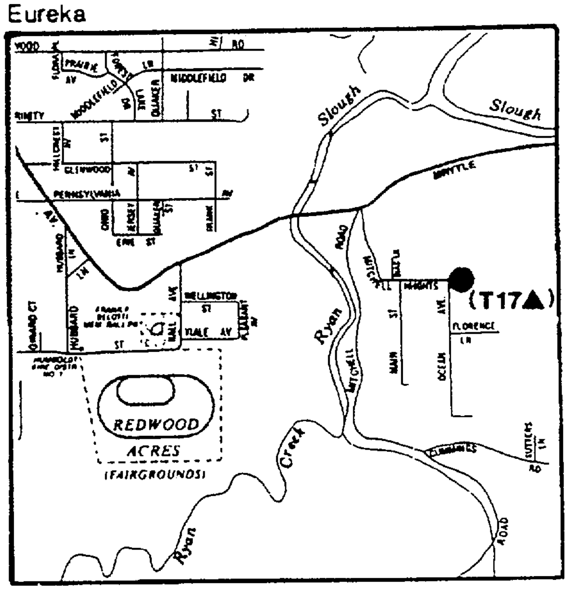
Fortuna



Arcata



Eureka



2.12 REMP INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITIONS

2.12.1 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program.

APPLICABILITY: At all times.

ACTION:

With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Monitoring Report.

SURVEILLANCE REQUIREMENTS

2.12.2 A summary of the results obtained from this program shall be included in the Annual Radiological Environmental Monitoring Report pursuant to Administrative Control 4.1.

3.0 SPECIFICATION BASES

3.1 Radioactive Liquid Effluent Monitoring Instrumentation Basis

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with Part II of the ODCM to ensure that the alarm/trip will occur prior to exceeding 10 times the effluent concentration limits of 10 CFR Part 20 for releases to Humboldt Bay. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3.2 Radioactive Gaseous Effluent Monitoring Instrumentation Basis

The radioactive gaseous effluent instrumentation is provided to monitor the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents from the plant stack. The alarm setpoints for these instruments are calculated in accordance with Part II of the ODCM to ensure that the alarm will occur prior to exceeding a radioactive material concentration corresponding to gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY of less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR Part 50.

3.3 Liquid Effluent Concentration Basis

This specification is provided to ensure that the instantaneous concentration of radioactive materials released in liquid waste effluents beyond the SITE BOUNDARY will be less than 10 times the effluent concentration limits specified in 10 CFR Part 20. The specification provides operational flexibility for releasing liquid effluents in concentrations to follow the Section II.A and II.C design objectives of Appendix I to 10 CFR 50. This limitation provides reasonable assurance that the levels of radioactive materials released to Humboldt Bay will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, to a MEMBER OF THE PUBIC and (2) the limits of 10 CFR 20.1302 to the population. This specification does not affect the requirement to comply with the annual limitations of 10 CFR 20.1301(a).

3.4 Liquid Effluent Dose Basis

This specification is provided to implement the requirements of Sections II.A. III-A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statement provides the required operating flexibility and at that same time implements the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable" (ALARA). The dose calculations in the OFFSITE DOSE CALCULATION MANUAL (ODCM) implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the OFFSITE DOSE CALCULATIONAL MANUAL (ODCM) for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

3.5 Liquid Waste Treatment Basis

The requirement that these systems be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as reasonably achievable" (ALARA). This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were selected as one quarter of the dose design objectives (on a monthly basis) set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents (3 mrem/yr; 10 mrem/yr to any organ).

3.6 Gaseous Effluents Dose Rate Basis

This specification provides reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA either within or outside the SITE BOUNDARY in excess of the design objectives of Appendix I to 10 CFR 50. The annual dose rate limits are the doses associated with the annual average concentrations of "old" 10 CFR 20, Appendix B, Table II, Column 1. The specification provides operational flexibility for releasing gaseous effluents to satisfy the Section II.A and II.C design objectives of Appendix I to

10 CFR 50.

For a MEMBER OF THE PUBLIC who may at times be within the SITE BOUNDARY, the period of occupancy (which is bounded by the maximum occupational period while working in Units 1 or 2) will be sufficiently low to compensate for the reduced atmospheric dispersion of gaseous effluents relative to that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. This specification does not affect the requirement to comply with the annual limitations of 10 CFR 20.1301(a).

The dose assessment contained in NUREG-1166 has established that neither the routine release of tritium and radioactive particulates with half-lives of greater than 8 days during SAFSTOR nor the occurrence of an analyzed accident during SAFSTOR will exceed the 1500 mrem/year dose rate limit.

The only tritium source term is the spent fuel pool. Assuming a conservative tritium spent fuel pool concentration of 1.0E-4 microcuries/ml, an evaporation rate of 50 gal per day and a ventilation flow rate of 32,000 cfm, the airborne tritium concentration is well below the required LLD of 1.0E-6 microcuries/ml. Therefore tritium is not sampled in the plant stack effluent stream.

3.7 Gaseous Effluents: Noble Gases Dose Basis

This Specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as reasonably achievable" (ALARA). The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculations established for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculational of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

3.8 Gaseous Effluents: Tritium and Radionuclides in Particulate Form Dose Basis

This specification is provided to implement the requirements of Sections II.C, III.A, and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluent will be kept "as low as is reasonably achievable" (ALARA). The calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by
calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The methods for calculating the dose due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and 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. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioactive particulates with half-life greater than eight days are dependent on the existing radionuclide pathways to man, in areas at and beyond the SITE BOUNDARY. The pathways which were examined in the development of these calculations were: 1) Individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leaf vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3.9 Solid Radioactive Waste Basis

This Specification ensures that radioactive wastes that are transported from the site shall meet the solidification requirements specified by the burial ground licensee of the respective states to which the radioactive material will be shipped. It also implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3.10 Total Dose Basis

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have now been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR part 190.11 and 10 CFR Part 20.2203a4, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190 and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 2.3, 2.4, 2.6, 2.7 and 2.8. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3.11 REMP Monitoring Program Basis

The quality-related portion of the REMP satisfies the requirements in 10 CFR Parts 20 and 50 that radiological environmental monitoring programs be established to provide data on measurable levels of radiation and radioactive materials in the site environs. It supplements the SAFSTOR Environmental Report baseline environmental conditions by conducting onsite and offsite environmental monitoring to evaluate routine conditions during SAFSTOR and to document any increased nuclide concentrations and/or radiation levels resulting from accidents during SAFSTOR.

The non quality-related portion of the HBPP REMP fulfills commitments for environmental monitoring made to the state of California and conducts additional environmental monitoring which PG&E/HBPP has elected to continue from the REMP which was being implemented prior to approval of the SAFSTOR Decommissioning Plan. Normally, non quality-related environmental monitoring (including sample collection and analysis) is conducted in accordance with the programmatic controls established for the quality-related environmental monitoring; however, this monitoring is not subject to the program requirements for radiological environmental monitoring contained in the NRC Radiological Assessment Branch's Branch Technical Position which was issued as Generic Letter 79-65 nor is it subject to the HBPP

Decommissioning Quality Assurance Program requirements including adherence to Regulatory Guide 4.15, *Quality Assurance for Radiological Monitoring Programs (Normal Operations)--Effluent Streams and the Environment.*

The SAFSTOR Environmental Report, submitted to the NRC as Attachment 6 to the SAFSTOR license amendment request, established baseline conditions for soil, biota and sediments. In accordance with the NRC approved SAFSTOR Decommissioning Plan, these baseline conditions will only need to be reestablished prior to DECON if a significant release during SAFSTOR occurs as the result of an accident.

The LLD's required by Table 2-9 are considered optimum for routine environmental measurements in industrial laboratories. The LLD's for drinking water meet the requirements of 40 CFR 141.

3.12 REMP Interlaboratory Comparison Program Basis

The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

4.0 ADMINISTRATIVE CONTROLS

4.1 Annual SAFSTOR Radiological Environmental Monitoring Report

A report on the SAFSTOR Radiological Environmental Monitoring Program shall be prepared annually in accordance with the NRC Branch Technical Position and submitted to the NRC within 90 days of January 1 of each year. This report shall be included as a separate section to the Annual Radiological Monitoring Report required by Technical Specification VII.J.1.

The Annual Radiological Environmental Report shall include:

a. Summaries, interpretations, and an analysis of trends of the results of the quality related Radiological Environmental Monitoring Program activities for the report period.

b. A comparison with the baseline environmental conditions established in the Decommissioning Environmental Report.

c. The results of analysis of quality related environmental samples and of quality related environmental radiation measurements taken during the period pursuant to the locations specified in Table 2-7 summarized and tabulated in the format of Table 4-1, Radiological Environmental Monitoring Program Annual Report Summary, or equivalent.

d. A summary description of the SAFSTOR Radiological Environmental Monitoring Program.

e. Legible maps covering all sampling locations keyed to a table giving distances and directions from Unit 3.

f. The results of licensee participation in the Interlaboratory Comparison Program and the corrective action taken if the specified program is not being performed as required in accordance with Specification 2.12.

g. The reason for not conducting the quality related portion of the Radiological Environmental Monitoring Program as required, and discussion of all deviations from the quality related sampling schedule of Table 2-7, including plans for preventing a recurrence in accordance with Specification 2.11.

h. A discussion of quality related environmental sample measurements that exceed the reporting levels of Table 2-8, Reporting Levels for Radioactivity Concentrations in Environmental Samples, but are not the result of plant effluents (i.e., demonstrated by comparison with a control station or the SAFSTOR Environmental Report).

i. A discussion of all analyses in which the LLD required by Table 2-9 was not achievable.

Table 4-1
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM ANNUAL REPORT SUMMARY - EXAMPLE

Name of Facility: Humboldt Bay Power Plant Unit 3 Docket No.: 50-133, OL-DPR-7

Location of Facility: Humboldt County, California (County, State) Reporting Period: January 1 - December 31, 1997

Medium or Pathway Sampled [Unit of Measurement]	Type and Total Number of Analyses Performed	Lower Limit of Detection[a] (LLD)	All Indicator Locations Mean, (Fraction) & [Range][b]	Location with Highest Annual Mean: Name, Distance and Direction	Location with Highest Annual Mean: Mean, (Fraction) & [Range][b]	Control Locations Mean, (Fraction) & [Range][b]	Number of Nonroutine Reported Measurements
AIRBORNE Radioiodine and Particulates	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
DIRECT RADIATION [mR/quarter]	Direct radiation (64)	3	13.6 ± 0.1 (64/64) [11.8 - 17.5]	Station T7	15.4 ± 0.2 (4/4) [13.8 - 17.5]	12.7 ± 0.3 (4/4) [12.5 - 12.9]	0
WATERBORNE Surface Water (Discharge canal effluent) [pCi/l]	Gamma isotopic (54)	Co-60: 15 Cs-137: 18	<MDA (0/54) [N/A]	N/A	N/A	Not Required	2
	Tritium (54)	500	<MDA (0/54) [N/A]	N/A	N/A	Not Required	2

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TABLE 4-1 (Continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM ANNUAL REPORT SUMMARY

Medium or Pathway Sampled [Unit of Measurement]	Type and Total Number of Analyses Performed	Lower Limit of Detection[a] (LLD)	All Indicator Locations	Location with Highest Annual Mean		Control Locations	Number of Nonroutine Reported Measurements
			Mean, (Fraction) & [Range][b]	Name, Distance and Direction	Mean, (Fraction) & [Range][b]	Mean, (Fraction) & [Range][b]	
WATERBORNE (continued) Groundwater (Monitoring wells) [pCi/l]	Gross Alpha (22)	3	7 ± 6 (1/22) [7 - 7]	Monitoring Well No. 2	7 ± 6 (1/4) [7 - 7]	N/A (0/4) [N/A]	2
	Gross Beta (22)	4	8 ± 2 (9/22)	Monitoring Well No. 11	10 ± 3 (3/6) [7 - 15]	10 ± 3 (3/6) [7 - 15]	2
	Gamma isotopic (22)	Co-60: 15 Cs-137: 18	<MDA (0/20) [N/A]	N/A	N/A	N/A (0/4) [N/A]	2
	Tritium (22)	500 (15/22) 200 (7/22)[c]	461 ± 64 (7/22) [299 - 601]	Monitoring Well No. 1	484 ± 94 (3/5) [409 - 589]	444 ± 88 (4/5) [299 - 601]	2
Drinking Water	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
Sediment	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
Algae	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
INGESTION Milk	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
Fish and invertebrates	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
TERRESTRIAL Soil	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A

TABLE 4-1 (Continued)
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM ANNUAL REPORT SUMMARY

[a] The LLD is defined as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95 percent probability with only 5 percent probability of falsely concluding that a blank observation represents a "real" signal.
LLD is defined as the a priori lower limit of detection (as pCi per unit mass or volume) representing the capability of a measurement system and not a the a posteriori (after the fact) limit for a particular measurement. (Current literature defines the LLD as the detection capability for the instrumentation only, and the MDA, minimum detectable concentration, as the detection capability for a given instrument, procedure and type of sample.) The actual MDA for these analyses was at or below the LLD.

[b] The mean and the range are based on detectable measurements only. The fraction of detectable measurements at specified locations is indicated in parentheses; e.g., (10/12) means that 10 out of 12 samples contained detectable activity. The range of detected results is indicated in brackets; e.g., [23-34].

[c] Tritium samples taken 10/24/97 and 11/18/97 were analyzed to a lower than normal LLD of 200 pCi/l.

Not Required - not required by the HBPP Unit 3 Technical Specifications. Baseline environmental conditions for this parameter were established in the Environmental Report as referenced by the SAFSTOR Decommissioning Plan.

N/A - Not applicable
Note: The example data are based on the 1997 monitoring results and are provided for illustrative purposes only.

4.2 Annual Radioactive Effluent Release Report

This report shall be submitted prior to April 1 of each year as required by SAFSTOR Technical Specification VII.J.3. The following information shall be included:

a. A summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant as outlined in Regulatory Guide 1.21, *Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants*, (Rev. 1, 1974) with data summarized on a quarterly basis following the format of Appendix B thereof.

b. For each type of solid waste shipped off-site:

   1. Container Volume

   2. Total Curie Quantity (specified as measured or estimated)

   3. Principal Radionuclides (specified as measured or estimated)

   4. Type of Waste (e.g., spent resin, compacted dry waste)

   5. Solidification Agent (e.g., cement)

c. A list and description of unplanned releases beyond the SITE BOUNDARY.

d. Information on the reasons for inoperability and lack of timely corrective action for any radioactive liquid or gaseous monitoring instrumentation inoperable for greater than 30 days in accordance with Specifications 2.1 and 2.2.

e. A summary description of changes made to:

   1. Process Control Program (PCP)

   2. Radioactive Waste Treatment Systems

f. A complete, legible copy of the entire ODCM if any change to the ODCM was made during the reporting period. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

4.3 Special Reports

The originals of Special Reports shall be submitted to the Document Control Desk with a copy sent to the Regional Administrator, NRC Region IV, within the time period specified for each report. These reports shall be submitted covering the activities identified below to the requirements of the applicable Specification.

a. Radioactive Effluents - Specifications 2.4, 2.5, 2.7, 2.8 and 2.10.

b. Radiological Environmental Monitoring - Specification 2.11.

4.4 Major Changes to Radioactive Waste Treatment Systems

a. Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid) shall be reported to the NRC in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed. The changes shall be reviewed and concurred with by the Plant Staff Review Committee and approved by the Plant Manager.

b. The following information shall be available for review:

1. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59,

2. Sufficient information to totally support the reason for the change,

3. A description of the equipment, components and processes involved and the interfaces with other plant systems,

4. A evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously estimated in the Environmental Report submitted to the NRC as Attachment 6 to the SAFSTOR license amendment request,

5. An evaluation of the change which shows the expected maximum exposures to an individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the Environmental Report,

6. An estimate of the exposure to plant personnel as a result of the change, and

7. Documentation of the fact that the change was reviewed and approved in accordance with plant procedures.

PART II - CALCULATIONAL METHODS AND PARAMETERS

1.0 EFFLUENT MONITOR SETPOINT CALCULATIONS

1.1 LIQUID EFFLUENT MONITORS

Specification 2.1 requires that the process water monitor and the caisson sump monitor be set to alarm to ensure that the limits of Specification 2.3 are not exceeded (the instantaneous concentration of radioactive material released to UNRESTRICTED AREAS shall be less than or equal to 10 times the concentrations specified in 10 CFR Part 20, Appendix B, Table 2, Column 2).

1.1.1 The alarm setpoint (countrate) for each monitor is calculated as:

$$A = \left[\left(\frac{F_3}{F_1 + F_2}\right) \times 10 \times (ECL_C) \times K \times 0.85\right] + B \qquad (1\text{-}1)$$

where:

A = The alarm setpoint, counts per minute, of the process water monitor or the caisson sump monitor.

F_1 = Flow rate past the process water monitor.

F_2 = Flow rate past the caisson sump monitor.

F_3 = Flow rate of the effluent canal into Humboldt Bay (F1 + F2 + circulating water flow - minimum flow with one Unit 1 or Unit 2 circulating water pump in operation is 12,500 gpm).

K = Calibration factor for the monitor, with units of cpm per micro-Ci/ml. Baseline calibration of the process water monitor (on 9/20/88) found this factor to be within ±15% of 3.06×10^8 cpm per micro-Ci/ml.

0.85 = Conservatism factor (85 percent of the Specification 2.3 concentration limits to allow for 15% monitor calibration uncertainty).

B = The monitor background reading (prior to any discharge) in counts per minute.

ECL_C = Composite Effluent Concentration Limit (ECL) for the mix of radionuclides (micro-Ci/ml).

1.1.2 The composite ECL for the mix of radionuclides is calculated as follows:

$$ECL_C = \frac{\sum_i C_i}{\sum_i \frac{C_i}{ECL_i}} = \frac{\sum_i f_i}{\sum_i \frac{f_i}{ECL_i}} \tag{1-2}$$

where:

ECL_i = ECL for radionuclide "i" from 10 CFR 20, Appendix B, Table 2, Column 2 (micro-Ci/ml).

C_i = Concentration of radionuclide "i" in the mixture.

f_i = Fraction of radionuclide "i" in the mixture.

1.1.3 Table 2-2 of Specification 2.1 requires that if a background reading exceeds the equivalent of 5×10^{-5} micro-Ci/ml of Cs-137, the cause will be investigated and remedial measures taken to reduce the background reading. Therefore, the maximum background allowable (B_{max}, cpm) is:

$$B_{max} = K \times \left(5 \times 10^{-5}\right) \text{cpm} \tag{1-3}$$

1.1.4 The most conservative background limit is calculated as if the calibration factor was 2.60×10^{8} cpm per micro-Ci/ml (-15% tolerance). This background limit would be 13,005 cpm. It is plant policy to use a background limit (slightly lower) at 13,000 cpm to ensure that this limit is satisfied. Note that if the background setting exceeds 13,000 cpm, the monitor should be declared INOPERABLE until the background has been reduced.

1.1.5 For continuous direct caisson sump discharges, the monitor should be set to alarm at or below 7.5 times the Cs-137 ECL from 10 CFR 20, Appendix B, Table 2, column 2 (75 percent of the Specification 2.3 limit for Cs-137), assuming no circulating water pump flow and that no liquid radwaste discharge is in progress (i.e., Equation 1-1 is solved with $F_1 = 0$ and $F_3 = F_2$).

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1.1.6 If the Specification 2.3 alarm setting is calculated for Cs-137, -15% tolerance, no dilution and for zero background, the alarm setting would be 26,000 cpm. Because the actual mixture may have a limit that is lower than that of Cs-137, and may also provide a reduced detector response, it is plant policy to maintain the alarm setting at or below 25,000 cpm, and to run at least one circulator during discharges, to ensure that this limit is satisfied. Refer to section 1.1.7 for the administrative (lower) alarm settings.

1.1.7 For routine liquid radwaste batch discharges, it is plant policy to set the process monitor alarm no higher than necessary in order to provide protection against inadvertent releases. With at least one circulator operating, the alarm should be set according to the following table, and in any case, no higher than 25,000 cpm. The table is based approximately on the on the sum of twice the typical background[1] and 130% of the predicted countrate for the batch[2], with the alarm point rounded up to the next higher meter mark (for more precise settings).[3]

Table 1-1
Liquid Effluent Monitor Alarm Setpoints

Predicted Process Monitor Reading (Net cpm)	Alarm Setting (cpm)	Equivalent Cs-137 Concentration (micro-Ci/ml for 600 cpm background)
Up to 2,692	5000	1.4×10^{-5}
2,692 up to 3,461	6000	1.8×10^{-5}
3,461 up to 4,231	7000	2.1×10^{-5}
4,231 up to 5,000	8000	2.4×10^{-5}
5,000 up to 5,769	9000	2.7×10^{-5}
5,769 up to 6,538	10,000	3.1×10^{-5}
6,538 up to 10,385	15,000	4.7×10^{-5}
10,385 up to 14,231	20,000	6.3×10^{-5}
14,231 up to 18,077	25,000	8.0×10^{-5}

[1]This table is based on a nominal background of 625 cpm. As of 4/15/97, the background reading is about 600 cpm. The extra 25% provides an allowance related to the uncertainty of reading the background.

[2] See section 2.4 of TBD-206. The 30% tolerance is for a combination of analytical and process monitor uncertainties and a 10% margin between the ratemeter and chart recorder.

[3] Each decade is marked at 1, 1.5, 2, 2.5, 3, 4, 5, 6, 7, 8 and 9.

1.2 GASEOUS EFFLUENT MONITOR

1.2.1 Specification 2.2 requires that the Stack Gas Monitoring System be set to alarm to ensure that the limits of specification 2.6 are not exceeded (the dose rate at or beyond the SITE BOUNDARY, due to noble gases released in gaseous effluents, shall be limited to less than or equal to 500 mrem/year total body and less than or equal to 3000 mrem/year to the skin).

1.2.2 Therefore, the alarm setpoint for this limiting condition is the lesser of A_{TB} or A_{SK} calculated for Kr-85. A_{TB} is calculated as:

$$A_{TB} = \left(\frac{500 \times C_1 \times C_2}{F \times \left(\frac{\chi}{Q}\right) \times D_A \times K} \right) + B \qquad (1\text{-}4)$$

and A_{SK} is calculated as :

$$A_{SK} = \left(\frac{3000 \times C_1 \times C_2}{F \times \left(\frac{\chi}{Q}\right) \times \left[L + (1.1 \times M)\right] \times K} \right) + B \qquad (1\text{-}5)$$

where:

A_{TB} = The alarm setpoint, cpm, for the stack noble gas radioactivity monitor, measuring the radioactivity concentration in the stack (prior to release) based on total body dose.

A_{SK} = The alarm setpoint, cpm, for the stack noble gas radioactivity monitor, measuring the radioactivity concentration in the stack (prior to release) based on skin dose.

500 = Whole body dose limit in mrem/year.

3000 = Skin dose limit in mrem/year.

C_1 = Conversion factor, 10^{-6} micro-Ci/pico-Ci.

C_2 = Conversion factor, 10^{-6} m^3/cc.

F = The flowrate, cubic meters per second, of the Unit No. 3 ventilation system discharge to the stack. This parameter is nominally 11.8 cubic meters per second (25,000 cfm) for the 50 foot stack.

$\frac{\chi}{Q}$ = The "instantaneous" atmospheric dispersion parameter, seconds per cubic meter.

= 6.52×10^{-4} seconds/cubic meter for releases from the 50 foot stack at a flow rate of 25,000 cfm. Refer to Appendix B.

D_{TB} = The total body dose factor due to gamma exposure in a semi infinite cloud, mrem/year per pico Curie/cubic meter. The value of this parameter is given in Table B-1 of Regulatory Guide 1.109 as 1.61×10^{-5} for Kr-85.

L = The skin dose factor due to beta exposure in a semi infinite cloud, mrem/year per pico Curie/cubic meter. The value of this parameter is given in Table B-1 of Regulatory Guide 1.109 as 1.34×10^{-3} for Kr-85.

M = The air dose factor due to gamma exposure in a semi infinite cloud, mrad/year per pico Curie/cubic meter. The value of this parameter is given in Table B-1 of Regulatory Guide 1.109 as 1.72×10^{-5} for Kr-85. The associated factor of 1.1 is a unit conversion from mrad to mrem.

K = Calibration factor for the monitor. As discussed in Appendix C, the calibration factor is 3.1×10^{-8} micro-Ci/cc per cpm.

B = The monitor background reading due to ambient background radiation and natural radioactive noble gasses, cpm. This parameter is generally not significant, since the typical reading is 20 ± 10 cpm.

1.2.3 Using the parameters above, the alarm point for the stack monitors should be set at or below 9,200 cpm for the 50 foot stack. It is plant policy to set it at 1,000 cpm. Note that changes to these values affect EPIP R-6 (Volume 3), EDOI H-11 (Volume 2) and STP 3.16.4 (Volume 6).

2.0 LIQUID EFFLUENT DOSE CALCULATIONS

2.1 CALENDAR QUARTER

Specification 2.4.1.a requires that the dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS during any calendar quarter shall be limited to less than or equal to 1.5 mrem to the total body and less than or equal to 5 mrem to any organ .

2.1.1 Compliance with Specification 2.4.1.a has been established on a licensing basis by the Environmental Report submitted to the NRC as Attachment 6 to the SAFSTOR licensing amendment request and NUREG-1166, *Final Environmental Statement for Decommissioning Humboldt Bay Power Plant, Unit No. 3*, issued by the NRC .

2.1.2 These reports have demonstrated that neither the routine release of radioactive materials in liquid effluents during SAFSTOR nor the occurrence of an analyzed accident during SAFSTOR would exceed the dose specification of Specification 2.4.1.a.

2.1.3 Therefore, calculation of dose due to the release of radioactive materials in liquid effluents during any calendar quarter is not necessary.

2.1.4 IF a comparison performed at least once per 31 days indicates that the activity due to the release of radioactive materials in liquid effluents will exceed the Environmental Report baseline release for the current calendar quarter, THEN a dose calculation for the current calendar quarter shall be performed.

2.2 CALENDAR YEAR

Specification 2.4.1.b requires that the dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS during any calendar year shall be limited to less than or equal to 3 mrem to the total body and less than or equal to 10 mrem to any organ .

2.2.1 Compliance with Specification 2.4.1.b has been established on a licensing basis by the Environmental Report and NUREG-1166.

2.2.2 These reports have demonstrated that the routine release of radioactive materials in liquid effluents during SAFSTOR will not exceed the dose specification of Specification 2.4.1.b.

2.2.3 Therefore, calculation of dose during any calendar year due to the routine release of radioactive materials in liquid effluents is not necessary.

2.2.4 IF a comparison indicates that the activity due to the release of radioactive materials in liquid effluents will exceed the Environmental Report baseline release for the current calendar year, THEN a dose calculation for the current calendar year shall be performed.

2.3 LIQUID EFFLUENT DOSE CALCULATION METHODOLOGY

The dose contribution to the total body and each individual organ (bone, liver, kidney, lung and GI-LLI) of the maximum and average exposed individual (adult, teen, child, and infant) will be calculated for the nuclides detected in effluents. The dose to an organ of an individual from the release of a mixture of radionuclides will be calculated as follows:

$$D = \sum_{i=1}^{n}\left[C_i \times DF \times \left\{\left(B_{Fish,i} \times U_{Fish}\right)+\left(B_{Inv,i} \times U_{Inv}\right)\right\}\right] \qquad (2\text{-}1)$$

where:

D = The dose commitment, mrem per year, to an organ (or to the whole body) due to consumption of aquatic foods.

C_i = The average diluted effluent concentration, pico-Curie/liter, for radionuclide, i. This will be estimated by dividing the total activity of the nuclide discharged during the quarter, pico-Curies, by the total circulating water discharge flow during the quarter, liters. Note that the resulting dose commitment is the annual dose for the case of four quarters with this average concentration.

DF = The dose conversion factor, mrem/pico-Curie for the nuclide, organ, and age group being calculated. This factor is taken from Tables 2-1, 2-2, and 2-3.

$B_{Fish,i}$ = The bioaccumulation factor, pico-Curie/kilogram per pico-Curie/liter, in fish for the radionuclide in question. This value is taken from Table 2-4.

$B_{Inv,i}$ = The bioaccumulation factor, pico-Curie/kilogram per pico-Curie/liter, in invertebrates for the radionuclide in question. This value is taken from Table 2-4.

U_{Fish} = Usage factor (consumption) of fish, kilogram/year, for the age group and individual (average or maximum) in question. This factor is derived from Table 2-5 or 2-6.

U_{Inv} = Usage factor of invertebrates, kilogram/year, for the applicable age group and individual (average or maximum). This factor is from Table 2-5 or 2-6.

The total exposure to an organ (or whole body) is found from the summation of the contributions of each of the individual nuclides calculated. Note that the infant age group is not considered to consume either fish or other seafood, and exposure to this age group need therefore not be calculated.

Table 2-1
Ingestion Dose Factors for Adult Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-11

	Organ					
Nuclide	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	1.05×10^{-7}	1.05×10^{-7}	1.05×10^{-7}	1.05×10^{-7}	1.05×10^{-7}
Co-60	No Data	2.14×10^{-6}	4.72×10^{-6}	No Data	No Data	4.02×10^{-5}
Sr-90	7.58×10^{-3}	No Data	1.86×10^{-3}	No Data	No Data	2.19×10^{-4}
Cs-137	7.97×10^{-5}	1.09×10^{-4}	7.14×10^{-5}	3.70×10^{-5}	1.23×10^{-5}	2.11×10^{-6}
Y-90	9.62×10^{-9}	No Data	2.58×10^{-10}	No Data	No Data	1.02×10^{-4}

Table 2-2
Ingestion Dose Factors for Teen Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-12

	Organ					
Nuclide	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	1.06×10^{-7}	1.06×10^{-7}	1.06×10^{-7}	1.06×10^{-7}	1.06×10^{-7}
Co-60	No Data	2.81×10^{-6}	6.33×10^{-6}	No Data	No Data	3.66×10^{-5}
Sr-90	8.30×10^{-3}	No Data	2.05×10^{-3}	No Data	No Data	2.33×10^{-4}
Cs-137	1.12×10^{-4}	1.49×10^{-4}	5.19×10^{-5}	5.07×10^{-5}	1.97×10^{-5}	2.12×10^{-6}
Y-90	1.37×10^{-8}	No Data	3.69×10^{-10}	No Data	No Data	1.13×10^{-4}

Table 2-3
Ingestion Dose Factors for Child Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-13

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	2.03×10^{-7}	2.03×10^{-7}	2.03×10^{-7}	2.03×10^{-7}	2.03×10^{-7}
Co-60	No Data	5.29×10^{-6}	1.56×10^{-5}	No Data	No Data	2.93×10^{-5}
Sr-90	1.70×10^{-2}	No Data	4.31×10^{-3}	No Data	No Data	2.29×10^{-4}
Cs-137	3.27×10^{-4}	3.13×10^{-4}	4.62×10^{-5}	1.02×10^{-4}	3.67×10^{-5}	1.96×10^{-6}
Y-90	4.11×10^{-8}	No Data	1.10×10^{-9}	No Data	No Data	1.17×10^{-4}

Table 2-4
Bioaccumulation Factors for Saltwater Environment
(pCi/kg per pCi/liter)
Selected Nuclides from Regulatory Guide 1.109, Table A-1

Element	Fish	Invertebrate
H	9.0×10^{-1}	9.3×10^{-1}
Co	1.0×10^{2}	1.0×10^{3}
Sr	2.0	2.0×10^{1}
Cs	4.0×10^{1}	2.5×10^{1}
Y	2.5×10^{1}	1.0×10^{3}

Table 2-5
Average Individual Foods Consumption for Various Age Groups
(kilo-gram/year or liters/year)
From Regulatory Guide 1.109, Table E-4

Age Group	Fish	Other Seafood (Invertebrates)	Fruits and Vegetables	Milk	Meat
Adult	6.9	1.0	190	110	95
Teen	5.2	0.75	240	200	59
Child	2.2	0.33	200	170	37
Infant	0	0	0	0	0

Table 2-6
Maximum Individual Foods Consumption for Various Age Groups
(kilo-gram/year or liters/year)
From Regulatory Guide 1.109, Table E-5

Age Group	Fish	Other Seafood (Invertebrates)	Fruits and Vegetables	Milk	Meat
Adult	21	5.0	520	310	110
Teen	16	3.8	630	400	65
Child	6.9	1.7	520	330	41
Infant	0	0	0	330	0

3.0 LIQUID WASTE TREATMENT

3.1 TREATMENT REQUIREMENTS

3.1.1 ODCM Specification 2.5

Specification 2.5 requires that liquid radwaste shall be treated, as required, to reduce radioactive materials in liquid wastes prior to their discharge, when projected monthly doses due to liquid effluents discharged to UNRESTRICTED AREAS would exceed 0.06 mrem whole body or 0.2 mrem to any organ.

3.1.2 NPDES Waste Discharge Requirement

NPDES Permit No. CA0005622, issued by the California Regional Water Quality Control Board - North Coast Region, requires that the discharge of liquid wastes "shall not cause bottom deposits in the receiving waters." The permit also identifies Discharge Serial No. 001E (liquid low level radioactive waste) that indicates that the waste may be treated prior to discharge. The permit does not mandate treatment.

3.2 TREATMENT CAPABILITIES

3.2.1 Liquid Waste Collection System

Liquid waste is collected in either the turbine building drain tank (TBDT), reactor equipment drain tank (REDT), reactor caisson sump or radwaste building sump.

a. Turbine Building Drain Tank

The TBDT, turbine building floor drain pump and TBDT pumps are located at elevation -14 feet in the reactor caisson in a shielded vault beneath the new fuel storage vault. The contents of the 3,000 gallon capacity tank may be pumped to a radwaste receiver tank or drained to the REDT via the caisson floor drain system.

b. Reactor Equipment Drain Tank

The REDT and associated REDT pumps are located at the -66 foot level of the reactor caisson access shaft. The contents of this 500 gallon capacity tank are pumped automatically to the radwaste treatment system using either of the two REDT pumps.

c. Reactor Caisson Sump

The reactor caisson sump and its associated reactor caisson sump pumps are located at the -66 foot level of the access shaft. The sump, which collects groundwater in-leakage, has a capacity of 50 gallons. The pump may transfer its contents automatically through a liquid effluent monitor to the Discharge Canal, or may be valved to the radwaste treatment system if necessary for compliance with Specification 2.5 due to groundwater contamination.

d. Radwaste Building Sump

The radwaste building sump tank, with a capacity of 250 gallons, is located beneath the radwaste building floor and receives liquids from drains in the vicinity of the radwaste building. The sump pump is located on the operating floor of the radwaste building (elevation +12 feet) over the sump tank. This pump automatically maintains the level of the tank and discharges to one of the waste receiver tanks.

3.2.2 Liquid Waste Treatment System

The liquid waste treatment system processes, stores and provides for disposal of radioactively contaminated wastes and other liquid wastes that are potentially radioactively contaminated. These wastes are first collected by the radwaste collection system and are then pumped to the radwaste building on the north side of the refueling building. The major components of the liquid waste treatment system which are available for use to comply with Specification 2.5 include the:

- waste receiver tanks (3)
- radwaste concentrator
- radwaste demineralizer
- resin disposal tank
- concentrated waste tanks (2)
- waste hold tanks (2)
- radwaste filters (2)
- concentrator drip receiver tank

a. Waste Receiver and Waste Hold Tanks

The three 7,500 gallon carbon steel radwaste receiver tanks are for wastes coming from the radwaste collection system. Two 7,500 carbon steel waste hold tanks are for storing treated wastes for retreatment or disposal. The tanks are located in an external section of the radwaste building, but are within the prefabricated steel radwaste enclosure.

b. Radwaste Concentrator

The radwaste concentrator was designed to concentrate 7,500 gallons per week. The concentrator consists of a vessel about 14 feet high and 24 inches in diameter with a 40 square foot, callandria-type evaporating section near the bottom. Steam from the Unit 1 or Unit 2 auxiliary steam system is fed to the callandria outside of the tubes. Evaporation takes place within the tubes. The concentrator is located in a shielded cubicle in the radwaste building.

Concentrator vapor goes to a condenser which is cooled with water from an independent cooling loop, and the condensate goes to the drip receiver tank for collection for further treatment or disposal. The concentrated radwaste is discharged to one of the two concentrated waste storage tanks.

Concentration by evaporation is generally the most appropriate method for treatment of liquids containing high total dissolved solids (TDS).

c. Radwaste Demineralizer

The radwaste demineralizer is a single, mixed bed unit with a nominal flow of 20 gpm and a flow capacity of 50 gpm. The demineralizer tank is 24 inches in diameter and was designed for 75 psig in accordance with the ASME Code. There are no provisions for regeneration; spent resins are sluiced to the resin disposal tank. The demineralizer is located in a shielded cubicle in the radwaste building.

Demineralization is generally not an appropriate method to treat high TDS liquids, but selective ion-exchange media may be used to reduce the concentration of specific radioactive ions in high DTS liquids.

d. Resin Disposal Tank

This 10,000 gallon tank is located in an individual shielded vault within the radwaste building. It is accessed through a hatch in the top of the vault. All spent resins from the various demineralizers on site are routed to this tank.

e. Concentrated Waste Tanks

Two 5,000 gallon storage tanks are located in a shielded vault in the radwaste building. These tanks receive concentrated wastes from the concentrator. These tanks have no inherent means for draining and must be pumped down through access ports in the top of the tank.

f. Radwaste Filters

Two radwaste filters are available in the radwaste building. These are cartridge-type filters which can remove particles down to 25 microns in diameter.

g. Concentrator Drip Receiver Tank

A concentrator drip receiver tank is provided to collect the condensed vapors from the concentrator. The concentrator drip receiver pump either recirculates water in the tank for sample mixing purposes, or it discharges to the treated waste pump discharge header for final disposition.

3.2.3 Mobile Liquid Waste Treatment Systems

Various mobile liquid waste treatment systems are available from vendors for use if necessary. These include systems such as high pressure filtration, demineralization, reverse osmosis and solidification.

Mobile liquid waste treatment systems are available for treatment of both high and low TDS liquids.

4.0 GASEOUS EFFLUENT DOSE CALCULATIONS

4.1 DOSE RATE

4.1.1 Noble Gases

Specification 2.6.1.a requires that the dose rate at or beyond the SITE BOUNDARY, due to noble gases released in gaseous effluents, shall be limited to less than or equal to 500 mrem/year total body and less than or equal to 3000 mrem/year to the skin.

a. Compliance with Specification 2.6.1.a has been established on a licensing basis by the Environmental Report and NUREG-1166.

b. These reports have demonstrated that neither the routine release of noble gases during SAFSTOR nor the occurrence of an analyzed accident involving spent fuel assemblies during SAFSTOR would exceed the dose rate specification of Specification 2.6.1.a.

c. Therefore, further methodology for the determination of dose rate due to noble gases is not necessary.

4.1.2 Tritium and Radioactive Particulates

Specification 2.6.1.b requires that the dose rate at or beyond the SITE BOUNDARY, due to tritium and radioactive particulates with half-lives of greater than 8 days released in gaseous effluents, shall be limited to less than or equal to 1500 mrem/year to any organ.

a. Compliance with Specification 2.6.1.b has been established on a licensing basis by the baseline gaseous effluent releases established in the Environmental Report and dose assessment contained in NUREG-1166.

b. These reports have demonstrated that neither the routine release of tritium and radioactive particulates with half-lives of greater than 8 days during SAFSTOR nor the occurrence of an analyzed accident during SAFSTOR will exceed the dose rate specification of Specification 2.6.1.b.

c. Therefore, further methodology for the determination of dose rate due tritium and radioactive particulates with half-lives of greater than 8 days released in gaseous effluents is not necessary.

4.2 DOSE - NOBLE GASES

4.2.1 Calendar Quarter

Specification 2.7.1.a requires that the air dose in UNRESTRICTED AREAS during any calendar quarter due to radioactive noble gases released in gaseous effluents shall be limited to less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation.

a. Compliance with Specification 2.7.1.a has been established on a licensing basis by the Environmental Report and NUREG-1166.

b. These reports have demonstrated that the routine release of noble gases in gaseous effluents during SAFSTOR will not exceed the dose specification of Specification 2.7.1.a.

c. Therefore, calculation of dose during any calendar quarter due to radioactive noble gases released in gaseous effluents is not necessary for the routine release of noble gases during SAFSTOR.

d. IF a comparison performed following an accident involving spent fuel indicates that the noble gases released in gaseous effluents will exceed the Environmental Report baseline release for the current calendar quarter, THEN a dose calculation for the current calendar quarter shall be performed.

4.2.2 Calendar Year

Specification 2.7.1.b requires that the air dose in UNRESTRICTED AREAS during any calendar year due to radioactive noble gases released in gaseous effluents shall be limited to less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

a. Compliance with Specification 2.7.1.b has been established on a licensing basis by the Environmental Report and NUREG-1166.

b. These reports have demonstrated that the routine release of noble gases in gaseous effluents during SAFSTOR will not exceed the dose specification of Specification 2.7.1.b.

c. Therefore, calculation of dose during any calendar year due to radioactive noble gases released in gaseous effluents is not necessary for the routine release of noble gases during SAFSTOR.

d. IF a comparison performed following an accident involving spent fuel indicates that the radioactive noble gases released in gaseous effluents will exceed the Environmental Report baseline release for the current calendar year, THEN a dose calculation for the current calendar year shall be performed.

4.2.3 Noble Gas Dose Calculation Methodology

Both dose to the whole body (gamma dose) and dose to the skin (beta dose) due to the release of radioactive noble gas effluents are calculated. However, due to the decay time since last operation, Kr-85 is the only radioactive noble gas that remains in the fuel. The equations for calculating the maximum hypothetical radiation exposure at an offsite location are as follows:

$$D_{WB} = Q \times (\chi/Q) \times K \qquad (4\text{-}1)$$

$$D_S = Q \times (\chi/Q) \times [L + (1.1 \times M)] \qquad (4\text{-}2)$$

where:

D_{WB} = Whole body (gamma) dose, mrem/year.

D_S = Skin (beta + gamma) dose, mrem/year .

χ/Q = The atmospheric dispersion parameter, seconds per cubic meter.
= 1.0×10^{-5} seconds per cubic meter for releases from the 50 foot stack with a flowrate of 25,000 cfm. Refer to Appendix B.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B.

K = The total body dose factor due to gamma exposure in a semi-infinite cloud, mrem/year per pico-Curie/cubic meter. The value of this parameter is given in Table B-1 of Regulatory Guide 1.109 as 1.61×10^{-5} for Kr-85.

L = The skin dose factor due to beta exposure in a semi-infinite cloud, mrem/year per pico-Curie/cubic meter. The value of this parameter is given in Table B-1 of Regulatory Guide 1.109 as 1.34×10^{-3} for Kr-85.

M = The air dose factor due to gamma exposure in a semi-infinite cloud, mrad/year per pico-Curie/cubic meter. The value of this parameter is given in Table B-1 of Regulatory Guide 1.109 as 1.72×10^{-5} for Kr-85. The associated factor of 1.1 is a unit conversion from mrad to mrem.

Q = The average release rate of Kr-85 in gaseous releases, pico-Curies/sec.

Note that this is the exposure to a hypothetical individual continuously located at the maximum ground level exposure location.

4.3 DOSE - TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

4.3.1 Calendar Quarter

Specification 2.8.1.a requires that the organ dose to a MEMBER OF THE PUBLIC from the release of tritium and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released to UNRESTRICTED AREAS shall be limited to less than or equal to 7.5 mrem during any calendar quarter.

a. Compliance with Specification 2.8.1.a has been established on a licensing basis by the Environmental Report and NUREG-1166.

b. These reports have demonstrated that the routine release of tritium and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents during SAFSTOR will not exceed the dose specification of Specification 2.8.1.a.

c. Therefore, calculation of dose during any calendar quarter due to tritium and radioactive materials in particulate form with half-lives greater than 8 days released in gaseous effluents is not necessary for the routine release of noble gases during SAFSTOR.

d. IF a comparison performed at least once per 31 days indicates that the tritium and radioactive materials in particulate form with half-lives greater than 8 days released in gaseous effluents will exceed the Environmental Report baseline release for the current calendar quarter, THEN a dose calculation for the current calendar quarter shall be performed.

4.3.2 Calendar Year

Specification 2.8.1.b requires that the organ dose to a MEMBER OF THE PUBLIC from the release of tritium and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents released to UNRESTRICTED AREAS shall be limited to less than or equal to 15 mrem during any calendar year.

a. Compliance with Specification 2.8.1.b has been established on a licensing basis by the Environmental Report and NUREG-1166.

b. These reports have demonstrated that the routine release of tritium and radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents during SAFSTOR will not exceed the dose specification of Specification 2.8.1.b.

c. Therefore, calculation of dose during any calendar year due to tritium and radioactive materials in particulate form with half-lives greater than 8 days released in gaseous effluents is not necessary for the routine release of noble gases during SAFSTOR.

d. IF a comparison indicates that the tritium and radioactive materials in particulate form with half-lives greater than 8 days released in gaseous effluents will exceed the Environmental Report baseline release for the current calendar year, THEN a dose calculation for the current calendar year shall be performed.

4.3.3 Particulate Organ Dose Calculation Summation Methodology

The releases of radioactive materials in particulate form with half-lives greater than 8 days in gaseous effluents will be essentially limited to Cs-137, Co-60, and Sr-90. The annual dose commitment will be calculated for any organ of an individual age group as follows:

$$D = \sum_{i=1}^{n}\left[Q_i \times \left(R_{Inh,i} + R_{GP,i} + R_{Meat,i} + R_{Milk,i} + R_{Veg,i}\right)\right] \qquad (4\text{-}3)$$

where:

D = Annual dose commitment, mrem/year.

Q_i = The average release rate of the nuclide in question, pico-Curies/second.

$R_{Inh,i}$ = The dose factor for the inhalation pathway for the radionuclide, i, in units of mrem/year per pico-Curie/sec.

$R_{GP,i}$ = The dose factor for the ground plane (direct exposure from deposition) pathway for the radionuclide, i, in units of mrem/year per pico-Curie/sec.

$R_{Meat,i}$ = The dose factor for the grass-cow-meat pathway for the radionuclide, i, in units of mrem/year per pico-Curie/sec.

$R_{Milk,i}$ = The dose factor for the grass-cow-milk pathway for the radionuclide, i, in units of mrem/year per pico-Curie/sec.

$R_{Veg,i}$ = The dose factor for the pathway of deposition on vegetation for the radionuclide, i, in units of mrem/year per pico-Curie/sec.

In general, the calculations for these pathways give results that represent trivial radiation exposure. The values calculated for typical anticipated SAFSTOR releases range from about 0.002 mrem/year (fruit/vegetable consumption pathway) to less than 1×10^{-6} mrem/year (for direct radiation exposure from material deposited on the ground).

4.3.4 Particulate Inhalation Pathway Dose Calculation Methodology

$$R_{Inh,i} = (\chi/Q) \times BR_a \times DF_{i,a} \tag{4-3a}$$

where:

χ/Q = The atmospheric dispersion parameter, seconds/cubic meter.
= 1.0×10^{-5} seconds/cubic meter for releases from the 50 foot stack. Refer to Appendix B.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B.

BR_a = The breathing rate of the receptor age group (a), cubic meters per year. The values to be used are 1400, 3700, 8000, and 8000 cubic meters/year for the infant, child, teen, and adult age groups, respectively.

$DF_{i,a}$ = The organ (or total body) inhalation dose factor, mrem/pico-Curie, for the receptor age group, a, for the radionuclide, i. The dose factors are given in Tables 4-1, 4-2, 4-3, and 4-4.

4.3.5 Particulate Ground Plane Pathway Dose Calculation Methodology

$$R_{GP,i} = (D/Q) \times SF \times DF_i \times K \times W \quad (4\text{-}3b)$$

where:

K = unit conversion constant, 8760 hr/yr.

DF_i = The ground plane dose conversion factor for radionuclide, i, in mrem/hr per pCi/m^2 from Table 4-5.

SF = The shielding factor (dimensionless). Table E-15 of Regulatory Guide 1.109 suggests values of 0.7 for the maximum individual.

D/Q = The atmospheric deposition factor, with units of inverse square meters.
= 3.0×10^{-8} inverse square meters for releases from the 50 foot stack. Refer to Appendix B.
= 5.39×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B.

W = Weathering factor. This is the reciprocal of the weathering time constant given in Regulatory Guide 1.109, for a 14 day removal half-life. In this equation, W has the value of 1.74×10^6 seconds.

4.3.6 Particulate Grass-Cow-Milk Pathway Dose Calculation Methodology

$$R_{Milk,i} = (D/Q) \times \left(\frac{Q_F \times U_a \times F_m \times DF_{i,a} \times W}{Y}\right) \quad (4\text{-}3c)$$

where:

Q_F = The cow's vegetation consumption rate. This is given as 50 kg/day per Regulatory Guide 1.109, Table E-3.

U_a = The receptor's milk consumption rate, liters/year for the age group in question. See Tables 4-6 and 4-7.

Y = The agricultural productivity by unit area of pasture. This parameter is 0.7 kg/m^2 per Regulatory Guide 1.109, Table E-15.

$DF_{i,a}$ = The ingestion dose factor for radionuclide, i, for the receptor in age group (a), in units of mrem/pico-Curie, from Tables 4-8, 4-9, 4-10, or 4-11.

F_m = The fraction of the cow's intake of a nuclide which appears in a liter of milk, with units of days/liter. This parameter is given by Table 4-12.

D/Q = The atmospheric deposition factor, with units of inverse square meters.
= 3.0×10^{-8} inverse square meters for releases from the 50 foot stack. Refer Appendix B.
= 3.29×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B.

W = Weathering factor. This is the reciprocal of the weathering time constant given in Regulatory Guide 1.109, for a 14 day removal half-life. In this equation, W has the value of 1.74×10^6 seconds.

4.3.7 Particulate Grass-Cow-Meat Pathway Dose Calculation Methodology

$$R_{Meat,i} = (D/Q) \times \left(\frac{Q_F \times U_a \times F_f \times DF_{i,a} \times W}{Y}\right) \qquad (4\text{-}3d)$$

where:

Q_F = The cow's vegetation consumption rate of 50 kg/day per Regulatory Guide 1.109, Table E-3.

U_a = The receptor's meat consumption rate, kilogram/year. Refer to Tables 4-5 and 4-7.

Y = The agricultural productivity by unit area of pasture. This parameter is 0.7 kg/m^2 per Regulatory Guide 1.109, Table E-15.

$DF_{i,a}$ = The ingestion dose factor for radionuclide, i, for the receptor in age group (a), in mrem/pCi, from Tables 4-8, 4-9, or 4-10, as appropriate. Note that this path is not considered to apply to the infant age group.

Ff = The fraction of the animal's intake of a nuclide which finally appears in meat, days/kilogram. This parameter is given in Table 4-13.

D/Q = The atmospheric deposition factor, with units of inverse square meters.
= 3.0×10^{-8} inverse square meters for releases from the 50 foot stack. Refer to Appendix B.
= 3.29×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B.

W = Weathering factor. This is the reciprocal of the weathering time constant given in Regulatory Guide 1.109, for a 14 day removal half-life. In this equation, W has the value of 1.74×10^{6} seconds.

4.3.8 Particulate Vegetation Pathway Dose Calculation Methodology

$$R_{Veg,i} = (D/Q) \times \left(\frac{U_T \times DF_{i,a} \times W}{Y} \right) \qquad (4\text{-}3e)$$

where:

U_T = The total consumption rate of fruits and vegetables, kilogram/year. This parameter is determined with the default values from Regulatory Guide 1.109, as reproduced in Tables 4-6 and 4-7.

D/Q = The atmospheric deposition factor, with units of inverse square meters.
= 3.0×10^{-8} inverse square meters for releases from the 50 foot stack. Refer to Appendix B.
= 3.29×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B.

W = Weathering factor. This is the reciprocal of the weathering time constant given in Regulatory Guide 1.109, for a 14 day removal half-life. In this equation, W has the value of 1.74×10^{6} seconds.

Y = The agricultural productivity by unit area of pasture. This parameter is 0.7 kg/m^2 per Regulatory Guide 1.109, Table E-15.

Note: this equation probably overestimates exposures, since it assumes that all of the deposition on a plant remains on the plant, while the Regulatory Guide allows a factor of 0.25. Also, the quantities assumed consumed include grain (none is grown in the vicinity of the plant), as well as vegetables and fruit grown in other areas (imported to Humboldt county).

4.3.9 Tritium Organ Dose Calculation Methodology

The annual dose commitment may be calculated for any organ of an individual age group as follows:

$$D = Q_{H3} \times \left(R_{Inh,\,H3} + R_{GP,\,H3} + R_{Meat,\,H3} + R_{Milk,\,H3} + R_{Veg,\,H3}\right) \quad (4\text{-}4)$$

where:

D = Annual dose commitment, mrem/year.

Q_{H3} = The average release rate of H-3, pico-Curies/second.

$R_{Inh,\,H3}$ = The dose factor for the inhalation pathway for H-3, mrem/year per pico-Curie/sec.

$R_{Meat,\,H3}$ = The dose factor for the grass-cow-meat pathway for H-3, mrem/year per pico-Curie/sec.

$R_{Milk,\,H3}$ = The dose factor for the grass-cow-milk pathway for H-3, mrem/year per pico-Curie/sec.

$R_{Veg,\,H3}$ = The dose factor for the vegetation consumption pathway, mrem/year per pico-Curie/sec.

This pathway results in trivial offsite calculated radiation exposures. A very conservative assumption of Tritium release is that Spent Fuel Pool water at 1 x 10^{-2} micro-Curies/ml H-3 is lost to the stack at a rate of 50 gallons/day. With this assumption, the calculated maximum offsite exposure is 0.0013 mrem/year.

4.3.10 Tritium Inhalation Pathway Dose Calculation Methodology

$$R_{Inh,\,H3} = \left(\chi/Q\right) \times BR_{a} \times DF_{H3,\,a} \quad (4\text{-}4a)$$

where:

χ/Q = The atmospheric dispersion parameter, seconds/cubic meter.
= 1.0×10^{-5} seconds/cubic meter for releases from the 50 foot stack. Refer to Appendix B.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B.

BR_a = The breathing rate of the receptor age group (a), cubic meters per year. The values to be used are 1400, 3700, 8000, and 8000 cubic meters/year for the infant, child, teen, and adult age groups, respectively.

$DF_{H3,a}$ = The organ (or total body) inhalation dose factor for the receptor age group, a, for H-3. This is given in units of mrem/pico-Curie by Tables 4-1, 4-2, 4-3, and 4-4.

4.3.11 Tritium Grass-Cow-Milk Pathway Dose Calculation Methodology

The concentration of tritium in milk is based on the airborne concentration rather than the deposition:

$$R_{Milk,\,H3} = \left(\chi/Q\right) \times \left(\frac{0.75 \times 0.5}{H}\right) \times Q_F \times U_a \times F_m \times DF_a \qquad (4\text{-}4b)$$

where:

Q_F = The cow's vegetation consumption rate. This is 50 kg/day per Regulatory Guide 1.109, Table E-3.

U_a = The receptor's milk consumption rate for age group, a, from Regulatory Guide 1.109. See Tables 4-6 or 4-7.

DF_a = The ingestion dose factor for H-3, for the reference group, mrem/pico-Curie, from Tables 4-8, 4-9, 4-10, and 4-11.

F_m = The fraction of the cow's intake of a nuclide which appears in a liter of milk, with units of days/liter. This parameter is given by Table 4-12.

0.75 = The fraction of total feed that is water.

0.5 = The ratio of specific activity of the feed grass to the atmospheric water.

H = Absolute humidity of the atmosphere, 0.008 kilograms/cubic meter, according to Regulatory Guide 1.109.

χ/Q = The atmospheric dispersion parameter, seconds/cubic meter.
= 1.0×10^{-5} seconds/cubic meter for releases from the 50 foot stack. Refer to Appendix B.
= 3.29×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B.

4.3.12 Tritium Grass-Cow-Meat Pathway Dose Calculation Methodology

$$R_{Meat,\,H3} = \left(\chi/Q\right) \times \left(\frac{0.75 \times 0.5}{H}\right) \times Q_F \times U_a \times F_M \times DF_a \quad \text{(4-4 c)}$$

Equation (C-9) from Regulatory Guide 1.109

where:

Q_F = The cow's vegetation consumption rate: 50 kg/day per Regulatory Guide 1.109, Table E-3.

U_a = The receptor's meat consumption rate. See Table 4-6 and Table 4-7.

DF_a = The ingestion dose factor for H-3, for the receptor in age group (a), in mrem/pCi, from Tables 4-8 through 4-11.

F_M = The fraction of the animal's intake of H-3 which appears in a kilogram of meat, with units of days/kilogram. This parameter is given by Table 4-13.

0.75 = The fraction of total feed that is water.

0.5 = The ratio of specific activity of the feed grass to the atmospheric water.

H = Absolute humidity of the atmosphere, 0.008 kilograms/cubic meter, according to Regulatory Guide 1.109.

χ/Q = The atmospheric dispersion parameter, seconds/cubic meter.
= 1.0×10^{-5} seconds/cubic meter for releases from the 50 foot stack. Refer to Appendix B.
= 3.29×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B.

4.3.13 Tritium Vegetation Pathway Dose Calculation Methodology

The concentration of tritium is based on the airborne concentration rather than the deposition:

$$R_{Veg,H3} = \left(\chi/Q\right) \times \left(\frac{0.75 \times 0.5}{H}\right) \times U_T \times DF_a \quad (4\text{-}4d)$$

where:

U_T = The total consumption rate of fruits and vegetables, kilogram/year. This parameter is given in Tables 4-6 and 4-7.

H = Absolute humidity of the atmosphere, 0.008 gm/m^3 per Regulatory Guide 1.109.

0.75 = The fraction of total feed that is water.

0.5 = The ratio of specific activity of H-3 in the feed grass to the specific activity in atmospheric water.

DF_a = The ingestion dose factor for H-3, for the receptor in age group (a), in mrem/pCi, from Tables 4-8 through 4-11.

χ/Q = The atmospheric dispersion parameter, seconds/cubic meter.
= 1.0×10^{-5} seconds/cubic meter for releases from the 50 foot stack. Refer to Appendix B.
= 3.29×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B.

Table 4-1
Inhalation Dose Factors for Adult Age Group
(mrem/pico-Curie inhaled)
Selected Nuclides from Regulatory Guide 1.109, Table E-7

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	1.58×10^{-7}	1.58×10^{-7}	1.58×10^{-7}	1.58×10^{-7}	1.58×10^{-7}
Co-60	No Data	1.44×10^{-6}	1.85×10^{-6}	No Data	7.46×10^{-4}	3.56×10^{-5}
Sr-90	1.24×10^{-2}	No Data	7.62×10^{-4}	No Data	1.20×10^{-3}	9.02×10^{-5}
Cs-137	5.98×10^{-5}	7.76×10^{-5}	5.35×10^{-5}	2.78×10^{-5}	9.40×10^{-6}	1.05×10^{-6}
Y-90	2.61×10^{-7}	No Data	7.01×10^{-9}	No Data	2.12×10^{-5}	6.32×10^{-5}

Table 4-2
Inhalation Dose Factors for Teen Age Group
(mrem/pico-Curie inhaled)
Selected Nuclides from Regulatory Guide 1.109, Table E-8

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	1.59×10^{-7}	1.59×10^{-7}	1.59×10^{-7}	1.59×10^{-7}	1.59×10^{-7}
Co-60	No Data	1.89×10^{-6}	2.48×10^{-6}	No Data	1.09×10^{-3}	3.24×10^{-5}
Sr-90	1.35×10^{-2}	No Data	8.35×10^{-4}	No Data	2.06×10^{-3}	9.56×10^{-5}
Cs-137	8.38×10^{-5}	1.06×10^{-4}	3.89×10^{-5}	3.80×10^{-5}	1.51×10^{-5}	1.06×10^{-6}
Y-90	3.73×10^{-7}	No Data	1.00×10^{-8}	No Data	3.66×10^{-5}	6.99×10^{-5}

Table 4-3
Inhalation Dose Factors for Child Age Group
(mrem/pico-Curie inhaled)
Selected Nuclides from Regulatory Guide 1.109, Table E-9

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	3.04×10^{-7}	3.04×10^{-7}	3.04×10^{-7}	3.04×10^{-7}	3.04×10^{-7}
Co-60	No Data	3.55×10^{-6}	6.12×10^{-6}	No Data	1.91×10^{-3}	2.60×10^{-5}
Sr-90	2.73×10^{-2}	No Data	1.74×10^{-3}	No Data	3.99×10^{-3}	9.28×10^{-5}
Cs-137	2.45×10^{-4}	2.23×10^{-4}	3.47×10^{-5}	7.63×10^{-5}	2.81×10^{-5}	9.78×10^{-7}
Y-90	1.11×10^{-6}	No Data	2.99×10^{-8}	No Data	7.07×10^{-5}	7.24×10^{-5}

Table 4-4
Inhalation Dose Factors for Infant Age Group
(mrem/pico-Curie inhaled)
Selected Nuclides from Regulatory Guide 1.109, Table E-10

	Organ					
Nuclide	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	4.62×10^{-7}	4.62×10^{-7}	4.62×10^{-7}	4.62×10^{-7}	4.62×10^{-7}
Co-60	No Data	5.73×10^{-6}	8.41×10^{-6}	No Data	3.22×10^{-3}	2.28×10^{-5}
Sr-90	2.92×10^{-2}	No Data	1.85×10^{-3}	No Data	8.03×10^{-3}	9.36×10^{-5}
Cs-137	3.92×10^{-4}	4.37×10^{-4}	3.25×10^{-5}	1.23×10^{-4}	5.09×10^{-5}	9.53×10^{-7}
Y-90	2.35×10^{-6}	No Data	6.30×10^{-8}	No Data	1.92×10^{-4}	7.43×10^{-5}

Table 4-5
External Dose Factors for Standing on Contaminated Ground
(mrem/hour per pico-Curie/square meter)
Selected Nuclides from Regulatory Guide 1.109, Table E-6

		Total
Nuclide	Skin	Body
H-3	0	0
Co-60	2.00×10^{-8}	1.70×10^{-8}
Sr-90	2.60×10^{-12}	2.20×10^{-12}
Cs-137	4.90×10^{-9}	4.20×10^{-9}
Y-90	2.60×10^{-12}	2.20×10^{-12}

Table 4-6
Average Individual Foods Consumption for Various Age Groups
(kilo-gram/year or liters/year)
From Regulatory Guide 1.109, Table E-4

Age Group	Fish	Other Seafood (Invertebrates)	Fruits and Vegetables	Milk	Meat
Adult	6.9	1.0	190	110	95
Teen	5.2	0.75	240	200	59
Child	2.2	0.33	200	170	37
Infant	0	0	0	0	0

Table 4-7
Maximum Individual Foods Consumption for Various Age Groups
(kilo-gram/year or liters/year)
From Regulatory Guide 1.109, Table E-5

Age Group	Fish	Other Seafood (Invertebrates)	Fruits and Vegetables	Milk	Meat
Adult	21	5.0	520	310	110
Teen	16	3.8	630	400	65
Child	6.9	1.7	520	330	41
Infant	0	0	0	330	0

Table 4-8
Ingestion Dose Factors for Adult Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-11

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	1.05×10^{-7}	1.05×10^{-7}	1.05×10^{-7}	1.05×10^{-7}	1.05×10^{-7}
Co-60	No Data	2.14×10^{-6}	4.72×10^{-6}	No Data	No Data	4.02×10^{-5}
Sr-90	7.58×10^{-3}	No Data	1.86×10^{-3}	No Data	No Data	2.19×10^{-4}
Cs-137	7.97×10^{-5}	1.09×10^{-4}	7.14×10^{-5}	3.70×10^{-5}	1.23×10^{-5}	2.11×10^{-6}
Y-90	9.62×10^{-9}	No Data	2.58×10^{-10}	No Data	No Data	1.02×10^{-4}

Table 4-9
Ingestion Dose Factors for Teen Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-12

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	1.06×10^{-7}	1.06×10^{-7}	1.06×10^{-7}	1.06×10^{-7}	1.06×10^{-7}
Co-60	No Data	2.81×10^{-6}	6.33×10^{-6}	No Data	No Data	3.66×10^{-5}
Sr-90	8.30×10^{-3}	No Data	2.05×10^{-3}	No Data	No Data	2.33×10^{-4}
Cs-137	1.12×10^{-4}	1.49×10^{-4}	5.19×10^{-5}	5.07×10^{-5}	1.97×10^{-5}	2.12×10^{-6}
Y-90	1.37×10^{-8}	No Data	3.69×10^{-10}	No Data	No Data	1.13×10^{-4}

Table 4-10
Ingestion Dose Factors for Child Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-13

	Organ					
Nuclide	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	2.03×10^{-7}	2.03×10^{-7}	2.03×10^{-7}	2.03×10^{-7}	2.03×10^{-7}
Co-60	No Data	5.29×10^{-6}	1.56×10^{-5}	No Data	No Data	2.93×10^{-5}
Sr-90	1.70×10^{-2}	No Data	4.31×10^{-3}	No Data	No Data	2.29×10^{-4}
Cs-137	3.27×10^{-4}	3.13×10^{-4}	4.62×10^{-5}	1.02×10^{-4}	3.67×10^{-5}	1.96×10^{-6}
Y-90	4.11×10^{-8}	No Data	1.10×10^{-9}	No Data	No Data	1.17×10^{-4}

Table 4-11
Ingestion Dose Factors for Infant Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-14

	Organ					
Nuclide	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	3.08×10^{-7}	3.08×10^{-7}	3.08×10^{-7}	3.08×10^{-7}	3.08×10^{-7}
Co-60	No Data	1.08×10^{-5}	2.55×10^{-5}	No Data	No Data	2.57×10^{-5}
Sr-90	1.85×10^{-2}	No Data	4.71×10^{-3}	No Data	No Data	2.31×10^{-4}
Cs-137	5.22×10^{-4}	6.11×10^{-4}	4.33×10^{-5}	1.64×10^{-4}	6.64×10^{-5}	1.91×10^{-6}
Y-90	8.69×10^{-8}	No Data	2.33×10^{-9}	No Data	No Data	1.20×10^{-4}

Table 4-12
Stable Element Transfer Data For Cow-Milk Pathway
(days/liter)
Selected Nuclides from Regulatory Guide 1.109, Table E-1

Element	F_m
H	1.0×10^{-2}
Co	1.0×10^{-3}
Sr	8.0×10^{-4}
Cs	1.2×10^{-2}
Y	1.0×10^{-5}

Table 4-13
Stable Element Transfer Data For Cow-Meat Pathway
(days/kilo-gram)
Selected Nuclides from Regulatory Guide 1.109, Table E-1

Element	F_f
H	1.2×10^{-2}
Co	1.3×10^{-2}
Sr	6.0×10^{-4}
Cs	4.0×10^{-3}
Y	4.6×10^{-3}

5.0 URANIUM FUEL CYCLE CUMULATIVE DOSE

5.1 WHOLE BODY DOSE

Specification 2.10 limits the whole body dose equivalent from the Uranium fuel to no more than 25 mrem/year. The whole body dose is determined by summing the calculated doses from the following:

a. Stack Noble gas releases, using equation (4-1).

b. Stack Particulate releases, using equation (4-3).

c. Stack Tritium releases, using equation (4-4).

d. Liquid releases, using equation (2-1).

To this calculated exposure is added potential direct radiation exposure to an individual at the site boundary. The only portion of the site boundary where there is significant direct radiation is near the radwaste facilities at the [PG&E] North edge of the site. Due to the possibility that an individual at the shoreline (fishing, bird watching, etc.) may use the path at the brow of the cliff for access, the TLD stations along the path are used to estimate an annual radiation exposure. The time period used for this estimate is 67 hours/year, given by Table E-5 of Regulatory Guide 1.109, as the maximum time for shoreline recreation for the Teen age group.

5.2 SKIN DOSE

Specification 2.10 limits the dose to any organ (thyroid excepted) to less than or equal to 25 mrem/year. The dose to the skin is determined by summing the calculated doses from the following:

a. Stack Noble gas releases, using equation (4-2).

b. Stack Tritium releases, using equation (4-4). (For H-3, the exposure to all organs is essentially equal, so the whole body value may be used for skin.)

c. Liquid Tritium releases, using equation (2-1). (Use whole body value, as above, for H-3).

d. The potential direct radiation exposure to an individual at the site boundary base on TLD stations, as determined in Section 5.1 above.

5.3 DOSE TO OTHER ORGANS

Specification 2.10 limits the dose to any organ (thyroid excepted) to less than or equal to 25 mrem/year. The dose to any individual other than skin organ is determined by summing the calculated doses from the following:

a. Stack Noble gas releases, using equation (4-1).

b. Stack Tritium releases, using equation (4-4).

c. Liquid Tritium releases, using equation (2-1).

d. The potential direct radiation exposure to an individual at the site boundary base on TLD stations, as determined in Section 5.1 above.

5.4 DOSE TO THE THYROID

Specification 2.10 limits the dose to the thyroid to less than or equal to 75 mrem/year. Since Unit 3 has not operated since July 2, 1976, there is an insufficient radioactive iodine source term remaining onsite to approach this limit. Therefore, calculation of dose to the thyroid is not required.

6.0 PROCESS CONTROL PROGRAM FOR RADIOACTIVE WASTE REQUIRING SOLIDIFICATION

6.1 SCOPE

This section pertains to radioactive waste containing a total specific activity which exceeds the burial ground criteria for solidification, or which exceeds the concentration limits for Class A waste as defined in 10 CFR 61. These wastes must be stabilized by solidification and contain no freestanding liquids prior to shipment offsite for land burial, or else be packaged in a high integrity container in accordance with Section 7.0.

6.2 PROGRAM ELEMENTS

For the land burial disposal of radioactive waste requiring solidification, HBPP shall implement the following steps:

6.2.1 Contract vendor solidification service may be utilized. The contract vendor solidification service may consist of solidification by the contractor or supply of materials, procedures and process control program (PCP) for HBPP solidification.

6.2.2 This vendor service shall include transmittal to HBPP of copies of their solidification procedure and PCP prior to performing the solidification.

6.2.3 The process parameters included in the PCP may include, but are not limited to, waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents and mixing and curing times.

6.2.4 The vendor solidification procedure and PCP shall be incorporated into a Plant Manual procedure that will be effective during the solidification process. This procedure will identify all Plant interfaces with the vendor's equipment (e.g., flush water, fire protection, shielding requirements, etc.), as well as identify the actions to be taken if excess free standing liquids are observed. This procedure shall require at least one representative test specimen from at least every tenth batch of waste processed to ensure solidification. The procedure should also include the actions to be taken if the test specimen fails to solidify.

6.2.5 This procedure shall be reviewed per plant procedures for adequacy in meeting applicable State, Federal, Department of Transportation and burial ground regulatory requirements and approved by the Plant Manager or designee prior to its implementation. This review shall ensure that the stability requirements of 10 CFR 61.56(b) for wastes exceeding Class A concentrations are met by the vendor solidification program.

7.0 PROCESS CONTROL PROGRAM FOR RADIOACTIVE WASTE PACKAGED IN HIGH INTEGRITY CONTAINERS

7.1 SCOPE

This section pertains to radioactive waste containing specific activity which exceeds the burial ground criteria for solidification, or which exceeds the concentration limits for Class A waste as defined in 10 CFR 61. These wastes must be stabilized by packaging in dewatered form in a high-integrity container which meets burial ground and regulatory requirements, or else be solidified in accordance with Section 6.0.

7.2 PROGRAM ELEMENTS

For land burial disposal of radioactive waste requiring a high-integrity container, HBPP shall implement the following steps:

7.2.1 A contract vendor high-integrity container shall be used.

7.2.2 The container shall be demonstrated to have been approved or have a current Certificate of Compliance prior to acceptance for use by HBPP. This shall include provision by the vendor to HBPP of documentation reflecting this authorization.

7.2.3 The material placed in the high-integrity container shall meet all applicable burial ground and regulatory waste form requirements for waste which is packaged in this manner.

7.2.4 The above criteria shall be met by following Plant Manual procedures which will be reviewed and approved by the Plant Manager or designee in accordance with Plant Manual administrative procedures prior to implementation at the time of packaging and disposal.

8.0 PROCESS CONTROL PROGRAM FOR LOW ACTIVITY DEWATERED RESINS AND OTHER WET WASTES

8.1 SCOPE

This section pertains to bead-type spent radioactive demineralizer resin and other wet wastes shipped for land burial which contain a total specific activity less than the burial ground criteria for solidification, and which does not exceed the concentration limits for Class A waste as defined in 10 CFR 61.

8.2 PROGRAM ELEMENTS

8.2.1 The dewatered resin or wet wastes must meet the requirements of 10 CFR 61.56 or those of the burial ground (whichever is more restrictive) for freestanding, noncorrosive liquid.

8.2.2 For bead resins, the preceding criterion will be met by following approved Plant Manual procedures for dewatering resin.

8.2.3 Liquid waste, that will not be thermal treated to remove freestanding liquid, must be solidified.

9.0 PROGRAM CHANGES

9.1 PURPOSE OF THE OFFSITE DOSE CALCULATION MANUAL

The Offsite Dose Calculation Manual was developed to support the implementation of the Radiological Effluent Technical Specifications required by 10 CFR 50, Appendix I, and 10 CFR 50.36. The purpose of the manual is to provide the NRC with sufficient information relative to effluent monitor setpoint calculations, effluent related dose calculations, and environmental monitoring to demonstrate compliance with radiological effluent controls.

9.2 CHANGES TO THE OFFSITE DOSE CALCULATION MANUAL

It is recognized that changes to the ODCM may be required during the SAFSTOR period. All changes shall be reviewed and approved by the PSRC and the Plant Manager prior to implementation. The NRC shall be informed of all changes to the ODCM by providing a description of the change(s) in the first Annual Radioactive Effluent Release Report following the date the change became effective. Records of the reviews performed on change to the ODCM should be documented and retained for the duration of the possession only license.

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9.3 HBPP does not intend to modify or reduce the environmental monitoring requirements as specified in the ODCM during the periods of SAFSTOR and decommissioning activities. This applies to those environmental samples and analysis identified in Table 2-7 as either quality or non-quality samples. (CTS-291)

10.0 COMMITMENTS

Revised 12/13/01

The following commitment is implemented by this procedure. The section number that implements to commitment is noted parenthetically.

CTS-291 (Section II, 9.3)

11.0 PROCEDURE OWNER

11.1 Director, Radiation Protection

APPENDIX A

SAFSTOR BASELINE CONDITIONS

1.0 LIQUID AND GASEOUS EFFLUENTS

1.1 LIQUID EFFLUENTS

Baseline levels of radioactive materials contained in liquid effluents during the SAFSTOR period were established in the Environmental Report submitted as Attachment 6 to the SAFSTOR license amendment request. These values are presented for cumulative annual release and average monthly discharge in Table A-1.

1.2 GASEOUS EFFLUENTS

Baseline levels of radioactive materials contained in gaseous effluents established in the Environmental Report are presented for cumulative annual and average monthly release in Table A-2.

Table A-1
Baseline Liquid Effluent Activity

Type of Activity	Annual Release (Curies)	Monthly Average Release (Curies)
Tritium	8.6E-2	7.2E-3
Principal Gamma Emitters (total)	1.85E-1	1.54E-2
Strontium-90	3.28E-4	2.73E-5

Table A-2
Baseline Gaseous Effluent Activity

Type of Activity	Annual Release (Curies)	Monthly Average Release (Curies)
Tritium	<4.0E-2	<3.3E-3
Particulate Gamma Emitters (total)	3.16E-4	2.63E-5
Strontium-90	3.38E-6	2.82E-7

APPENDIX B

BASES FOR ATMOSPHERIC DISPERSION AND DEPOSITION VALUES

1.0 BASIS FOR DISPERSION/DEPOSITION VALUES - 50' STACK

1.1 The instantaneous atmospheric dispersion factor (X/Q) is taken from meteorological parameter calculations performed to evaluate reducing the height of the Unit 3 stack. The calculation report is number N238C, Revision 0, titled "Determine Effect of Humboldt Bay Power Plant Unit 3 Stack Reconfiguration on Downwind Effluent Concentrations". This calculation is microfilmed (with calculations N238A & N238B), at microfilm reel/frame location (RLOC) 07175-4939 thru 5359. Table 1 (frame number 5140) of the calculation (N238C) provides "1 hour" values for the instantaneous X/Q for the 50' stack for various stack flow rates, based on an EPA model named "ISCST". The instantaneous X/Q value used in the ODCM (6.52×10^{-4}) is based on a stack flow of 25,000 cfm.

1.2 The annual average atmospheric dispersion factor (X/Q) is taken from meteorological parameter calculations performed to evaluate reducing the height of the Unit 3 stack. The calculation report is number N238C, Revision 0, titled "Determine Effect of Humboldt Bay Power Plant Unit 3 Stack Reconfiguration on Downwind Effluent Concentrations". This calculation is microfilmed (with calculations N238A & N238B), at microfilm reel/frame location (RLOC) 07175-4939 thru 5359. Table 1 (frame number 5140) of the calculation (N238C) provides annual maximum values for X/Q for the 50' stack for various stack flow rates, based on an NRC model named "XOQDOQ". The annual average X/Q value used in the ODCM (1.00×10^{-5}) is based on a stack flow of 25,000 cfm.

1.3 The annual average atmospheric deposition factor (D/Q) is taken from meteorological parameter calculations performed to evaluate reducing the height of the Unit 3 stack. The calculation report is number N238C, Revision 0, titled "Determine Effect of Humboldt Bay Power Plant Unit 3 Stack Reconfiguration on Downwind Effluent Concentrations". This calculation is microfilmed (with calculations N238A & N238B), at microfilm reel/frame location (RLOC) 07175-4939 thru 5359. Table 1 (frame number 5140) of the calculation (N238C) provides annual maximum values for D/Q for the 50' stack for various stack flow rates, based on an NRC model named "XOQDOQ". The annual average D/Q value used in the ODCM (3.00×10^{-8}) is based on a stack flow of 25,000 cfm.

2.0 BASIS FOR DISPERSION/DEPOSITION VALUES - INCIDENTAL RELEASE PATHS

2.1 The atmospheric dispersion factor (X/Q) for incidental releases is 6.59×10^{-3} seconds/cubic meter, calculated as described below

2.1.1 This factor is based on the atmospheric models of Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. These models are intended to estimate meteorological dispersion for "real time" conditions (i.e., hourly), rather than "annual average" conditions. The applicable guidance is section 1.3.1 (Releases Through Vents or Other Building Penetrations), as it applies to all releases from points lower than 2.5 times the height of adjacent structures. This calculation generally follows the guidance for the use of equations 1, 2 and 3 of Regulatory Guide 1.145.

2.1.2 The assumed distance from the emission point to the potential receptor for this calculation is 150 meters. This is the approximate distance to publicly accessible areas from the structure with the most significant potential for airborne radioactivity (i.e. from the center of the Refueling Building to the trail at the edge of the bluff).

2.1.3 The meteorological conditions assumed for this calculation are for stable "fumigation" conditions (Pasquill stability class G), with a wind speed of 1 meters/second.

2.1.4 The applicable equations from Reg. Guide 1.145 are as follows:

$$X/Q = \frac{1}{\overline{U}_{10}\left(\pi\sigma_y\sigma_z + A/2\right)} \tag{1}$$

$$X/Q = \frac{1}{\overline{U}_{10}\left(3\pi\sigma_y\sigma_z\right)} \tag{2}$$

$$X/Q = \frac{1}{\overline{U}_{10}\pi\Sigma_y\sigma_z} \tag{3}$$

where:

$\overline{U}_{10}$ = wind speed at 10 meters above grade, equal to 1 meter/second.

σ_y = lateral plume spread, equal to 4.33 meters for Pasquill Class G at a distance of 150 meters.

σ_z = vertical plume spread, equal to 1.86 meters for Pasquill Class G at a distance of 150 meters.

A = vertical cross-sectional area of structures, equal to 375 meters^2, based on the Refueling Building dimensions (about 36 feet high, about 112 feet long).

Σ_y = lateral plume spread (including meander and building wake), meters, equal to $6\sigma_y$ (for distances less than 800 meters, wind speeds below 2 meters/second, and stability class G).

2.1.5 With these values, the results for equations 1, 2, and 3 are as follows:

$$X/Q = 4.70 \times 10^{-3} \text{ seconds/meter}^3 \tag{1}$$

$X/Q = 1.32 \times 10^{-2}$ seconds/meter^3 (2)

$X/Q = 6.59 \times 10^{-3}$ seconds/meter^3 (3)

Per the Reg. Guide, the higher value of equations 1 and 2 is to be compared with the value for equation 3, and the lower value of that comparison should be used. with this logic, the resulting value for X/Q is 6.59×10^{-3} seconds/meter^3.

2.2 The atmospheric deposition factor (D/Q) for incidental releases is 5.39×10^{-6} meter^{-2} for the Particulate Ground Plane Pathway, and is 3.29×10^{-6} meter^{-2} for all other deposition related pathways. The factors are calculated as described below

2.2.1 These factors are based on the atmospheric models of Regulatory Guide 1.111, *Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-water-cooled Reactors.* The applicable guidance is section C.3.b (Dry Deposition), and Figure 6 (Relative Deposition for Ground-level Releases). To determine the atmospheric deposition across a downwind sector, the value from Figure 6 is to be multiplied by the fraction of the release transported into the sector, and divided by the sector cross-wind arc length at the distance being considered. For this calculation, the deposited contamination will be assumed to be evenly distributed across the width of the plume, rather than across an arbitrary angular sector.

2.2.2 Two factors are necessary because the nearest location (along the bay) is not a credible location for farming. For the purposes of estimating offsite doses from incidental releases, the nearest "farm" will be assumed to be beyond the railroad tracks, Southeast of the plant.

2.2.3 For the Particulate Ground Plane Pathway, the assumed distance from the emission point to the potential receptor for this calculation is 150 meters. This is the approximate distance to publicly accessible areas from the structure with the most significant potential for airborne radioactivity (i.e. from the center of the Refueling Building to the trail at the edge of the bluff). At this distance, Figure 6 provides a Relative Deposition Rate value of 1.4×10^{-4} meter^{-1}. The plume width assumed for this calculation is the same as was used in equation 3 of section 2.1.4 (above), so that the plume width is approximately $6\sigma_y$. For σ_y equal to 4.33 meters (Pasquill Class G at a distance of 150 meters), D/Q is $(1.4 \times 10^{-4}$ meter$^{-1})/(6 \times 4.33$ meter$) = 5.39 \times 10^{-6}$ meter^{-2}.

2.2.4 For the pathways involving farming or ranching, the assumed distance from the emission point to the potential receptor for this calculation is 220 meters. This is the approximate distance to publicly accessible "grazing" areas from the structure with the most significant potential for airborne radioactivity (i.e. from the center of the Refueling Building to the other side of the railroad). At this distance,

Figure 6 provides a Relative Deposition Rate value of 1.2×10^{-4} $meter^{-1}$. The plume width assumed for this calculation is the same as was used in equation 3 of section 2.1.4 (above), with the plume width of approximately $6\sigma_y$., but at a greater distance. For σ_y equal to 6.07 meters (Pasquill Class G at a distance of 220 meters), D/Q is $(1.2 \times 10^{-4}\ meter^{-1})/(6 \times 6.07\ meter) = 3.29 \times 10^{-6}\ meter^{-2}$.

APPENDIX C

Kr-85 MONITOR CALIBRATION

1.0 Kr-85 MONITOR CALIBRATION

1.1 The original calibration factor was based on the manufacturer's calibration. This calibration was re-examined after a test was performed to determine the effects of sample line pressure drop on the calibration of the stack sampler/monitor. This section documents the results of that test and review.[1]

1.1.1 The two detector chambers were found to have essentially identical reduced pressures. The pressures in chambers 'A' and 'B' were -2.176 and -2.203 in. Hg. (relative to atmospheric pressure), respectively.

1.1.2 The effect of changing the stack particulate sample filter from 'dirty' to 'clean' was small, with a pressure drop difference of 0.009 in. Hg. The chamber 'A' pressure was measured (relative to atmospheric pressure) at -2.167 for the 'clean' filter condition and at -2.176 in. Hg. for the 'dirty' filter conditions.

1.1.3 The true system flowrate was found to differ slightly from the flowrate indicated on the Photohelic Gauge when the system was set up in the then normal S.T.P. flow calibration configuration (flow calibrator inlet at atmospheric pressure), but the calibration was accurate at the normal system conditions (approximately 2 in. Hg. vacuum). The test pressure/flow measurement results are summarized below:

Chamber 'A' Vacuum (in. Hg.)	Photohelic Gauge Indicated Flowrate (cfm)	C-812 Air Flow Calibrator Flowrate (cfm)
0.275	2.2	1.9
1.001	2.2	2.15
1.995	2.2	2.2
3.020	2.2	2.2

1.2 The Kr-85 monitoring system was originally calibrated with Kr-85 gas standards. The standard certificate concentrations were given for the gas at 'STP' (Standard Temperature & Pressure), but the calibration was performed at 'ambient' conditions, without any correction. According to the vendor of the radioactive standard gas, STP conditions are 760 mm Hg. and 0 °C (273 °K). The system calibration conditions were 'Ambient' temperature (recorded as 70 °F, or 294 °K) and 'Atmospheric' pressure (exact barometric pressure not recorded), at Indianapolis, IN. Since the elevation was about 800', the absolute atmospheric pressure could have ranged from about 29.0 to 29.6 in. Hg. Assuming that the absolute pressure was 29.3 in. Hg. (744 mm Hg.), the concentration of the gas in the chambers at the actual calibration conditions would have been lower by a factor of 0.909 due to the lower pressure and higher temperature:

$$\left(\frac{744}{760}\right)\left(\frac{273}{294}\right) = 0.909$$

[1] After the Technical Review Group meeting of 4/14/93, a test procedure was developed to determine the effects of sample line pressure drop. The test was performed on 5/18/93.

1.3 The following table summarizes the original calibration results, with the assumption that the absolute pressure for the calibration was 29.3 in. Hg.:

Gas Concentration at STP (µCi/cc)	Gas Concentration at Original Calibration Conditions (µCi/cc)	Detector 'A' Net Countrate (cpm)	Detector 'B' Net Countrat e (cpm)	Detector 'A' Calibration Factor (µCi/cc per cpm)	Detector 'B' Calibration Factor (µCi/cc per cpm)
1.84E-6	1.67E-6	6.08E1	6.42E1	2.75E-8	2.61E-8
1.66E-5	1.51E-5	4.88E2	4.98E2	3.09E-8	3.03E-8
1.67E-4	1.52E-4	5.10E3	5.39E3	2.98E-8	2.82E-8
1.67E-3	1.52E-3	5.26E4	5.46E4	2.89E-8	2.78E-8
1.09E-2	9.91E-3	3.36E5	3.38E5	2.95E-8	2.93E-8

1.4 The effect of the sample line pressure drop (see section 6.1.1) is to reduce the density of the gas in the detector chambers relative to the density of the gas leaving the stack, thereby making the system read lower than it would if the gas in the chambers was at atmospheric pressure. The correction factor for this effect is about 1.08 (29.92/27.74). If this correction is applied to the average of the 10 measurements above, the resulting calibration would be 1.08 x 2.88E-8 = 3.11E-8 µC/cc per cpm. This is essentially the same value as the one originally established (3.1E-8), so the error produced by neglecting the sample line pressure drop effectively canceled out the error resulting from incorrectly interpreting the original calibration.

1.5 The flow control system calibration (S.T.P. 3.16.7) was revised so that the Photohelic Gauge metering system flowrate is checked at the operating absolute pressure condition.