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PG&E Letter HBL-20-005

10 CFR 50, Appendix I
10 CFR 50.36a

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 2019

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, 2019. This report is required by Appendix B, Section 6.3 of the Humboldt Bay Quality Assurance Plan.

Enclosure 2 contains Revision 29 to the "SAFSTOR/Decommissioning Offsite Dose Calculation Manual" as required by Section 4.2 of the "SAFSTOR/Decommissioning Offsite Dose Calculation Manual."

Enclosure 3 contains Revision 30 to the "SAFSTOR/Decommissioning Offsite Dose Calculation Manual" as required by Section 4.2 of the "SAFSTOR/Decommissioning Offsite Dose Calculation Manual."

Enclosure 4 contains Revision 31 to the "SAFSTOR/Decommissioning Offsite Dose Calculation Manual" as required by Section 4.2 of the "SAFSTOR/Decommissioning Offsite Dose Calculation Manual."

There are no new or revised regulatory commitments (as defined by NEI 99-04) made in this letter.

If you have any questions regarding this submittal, please contact Mr. Philippe Soenen at 805-459-3701.

Sincerely,

Loren D. Sharp

Enclosures

cc:	HBPP Humboldt Distribution
cc/enc:	John B. Hickman, NRC Project Manager
	Scott A. Morris, NRC Region IV Administrator

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

HUMBOLDT BAY POWER PLANT UNIT 3
ANNUAL RADIOACTIVE
EFFLUENT RELEASE REPORT

January 1 through December 31, 2019

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INTRODUCTION

This report summarizes gaseous and liquid radioactive effluent releases from Humboldt Bay Power Plant (HBPP) Unit 3 for the four quarters of 2019. The report includes calculated potential radiation doses 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 Offsite Dose Calculation Manual (ODCM) limits.

The HBPP Main Plant Stack, a ground level release path, and stack particulate airborne monitoring system (SPAMS), the real time effluent monitor, were shut down on October 14, 2015, and permanently removed from service to facilitate partial demolition of the Reactor Building.

The information is reported as required by Appendix B, Section 6.3 of the Humboldt Bay Quality Assurance Plan and Section 4.2 of the ODCM, 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 is not available. Previous HBPP 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, Iodine, and Noble Gasses

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 Kr-85. During 2008, the spent nuclear fuel was transferred from the spent fuel pool to the independent spent fuel storage installation (ISFSI), so there is now no source term for Kr-85.

Air Particulate Filter Composites – Sr-90 and Am-241

No modular high-efficiency particulate air (HEPA) ventilation units were used during the reporting period. No weekly sampling was required for monitoring effluents by the ODCM.

Gaseous Effluents – Tritium

Tritium sampling is not required by the HBPP ODCM. No tritium samples were collected during this reporting period.

Liquid Effluents

The last batch discharge of radioactive liquid effluent occurred on December 11, 2013. Subsequent radioactive liquid effluent batches were transported to US Ecology for offsite disposal under the 10 CFR 20.2002 exemption. There were no liquid shipments during this reporting period.

Average Energy

Calculations for the average energy of gaseous releases of fission and activation gases are not required for HBPP.

I. SUPPLEMENTAL INFORMATION

A. Regulatory Limits

1. Gaseous Effluents

a. Noble Gas Release Rate Limit

Noble gases are no longer an issue since the spent nuclear fuel has been relocated to the ISFSI.

b. Iodine Release Rate Limit

Due to the long decay time since the Unit was shut down, 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, an effluent flow rate and an annual average dispersion factor for the sector with the least favorable atmospheric dispersion. There were no operable effluent paths and no particulate samples were collected during the reporting period.

The applicable annual average dispersion factor for incidental releases is 6.59E-3 seconds per cubic meter.

2. Liquid Effluents

a. Concentration Limit

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

B. Effluent Concentration Limits

1. Gaseous Effluents

Effluent concentration limits for gaseous effluents are taken from 10 CFR 20, Appendix B, Table 2, Column 1.

2. Liquid Effluents

Effluent concentration limits for liquid effluents are taken from 10 CFR 20, Appendix B, Table 2, Column 2.

C. Measurements and Approximations of Total Radioactivity

1. Gaseous Effluents – Elevated Release

Elevated releases did not occur at HBPP during the reporting period.

2. Gaseous Effluents – Ground-level Release

a. Fission and Activation Gases

Fission and activation gases are no longer an issue since the spent fuel has been relocated to the ISFSI.

b. Iodines

Due to the long decay time since operation (shutdown July 2, 1976), no detectable releases of radioactive Iodine can be expected. Therefore, neither the Technical Specifications nor the ODCM require that these radionuclides be monitored.

c. Particulates

Radioactive particulates released from modular HEPA ventilation units are monitored by continuous sample collection on particulate filters when used. No areas involving elevated airborne radioactivity were identified, so no modular HEPA ventilation units were used during the reporting period.

3. Liquid Effluents

a. Batch Releases

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

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 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

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 released.

Section A of Table 1, 2A, and 2B have been omitted as fission and activation gases are neither expected or measured.

Table 2A is for reporting the quantities of each of these nuclides determined to be released from an elevated release point (there are none).

Table 2B presents the quantities of each of the nuclides determined to be released by ground level release points (there are none).

There were no “Batch Mode” gaseous releases during this report period.

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.

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

TABLE 1

GASEOUS EFFLUENTS – SUMMATION OF ALL RELEASES

B. Particulates

	Units	First Quarter	Second Quarter	Third Quarter	Fourth Quarter	Est. Total Error, %
1. Total release	Ci	N/A	N/A	N/A	N/A	N/A
2. Average release rate	μCi/sec	N/A	N/A	N/A	N/A	
3. Percent of applicable limit	%	N/A	N/A	N/A	N/A	
4. Applicable limit	μCi/cc	N/A	N/A	N/A	N/A	
5. Gross alpha radioactivity	Ci	N/A	N/A	N/A	N/A	

Table Notes:

N/A – There were no gaseous effluent releases during the reporting period.

	Units	First Quarter	Second Quarter	Third Quarter	Fourth Quarter
Stack Release Path	%	N/A	N/A	N/A	N/A
Incidental Release Path	%	N/A	N/A	N/A	N/A

No operating modular HEPA units after 6/7/16.

TABLE 2A

GASEOUS EFFLUENTS – ELEVATED RELEASE – PARTICULATES CONTINUOUS MODE - NUCLIDES RELEASED

Nuclides Released	Unit	Continuous Mode			
		First Quarter	Second Quarter	Third Quarter	Fourth Quarter
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

Table Notes:

N/A – There were no elevated gaseous effluents during the report period.

TABLE 2B

GASEOUS EFFLUENTS – GROUND-LEVEL RELEASES NUCLIDES RELEASED

Nuclides Released	Unit	Continuous Mode			
		First Quarter	Second Quarter	Third Quarter	Fourth Quarter
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
Americium-241	Ci	N/A	N/A	N/A	N/A
Total for period	Ci	N/A	N/A	N/A	N/A

Table Notes:

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

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	N/A	N/A	N/A	N/A	N/A
2. Average diluted concentration	μCi/ml	N/A	N/A	N/A	N/A	
3. Percent of applicable limit	%	N/A	N/A	N/A	N/A	
4. Applicable limit	μCi/ml	N/A	N/A	N/A	N/A	
B. Tritium						
1. Total release	Ci	N/A	N/A	N/A	N/A	N/A
2. Average diluted concentration	μCi/ml	N/A	N/A	N/A	N/A	
3. Percent of applicable limit	%	N/A	N/A	N/A	N/A	
4. Applicable limit	μCi/ml	N/A	N/A	N/A	N/A	
C. Gross Alpha Radioactivity						
1. Total release	Ci	N/A	N/A	N/A	N/A	N/A
D. Volume of waste released (prior to dilution)	Liters	N/A	N/A	N/A	N/A	N/A
E. Volume of dilution water	Liters	N/A	N/A	N/A	N/A	N/A

Table Notes:

There were no batch liquid effluent releases during the report period.

TABLE 4

LIQUID EFFLUENTS – NUCLIDES RELEASED

Nuclides Released	Unit	Batch 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
Cobalt-60	Ci	N/A	N/A	N/A	N/A
Americium-241	Ci	N/A	N/A	N/A	N/A
Nickel-63	Ci	N/A	N/A	N/A	N/A
Tritium	Ci	N/A	N/A	N/A	N/A
Total for period	Ci	N/A	N/A	N/A	N/A

Table Notes:

There were no batch liquid effluent releases during the report period.

III. SOLID RADIOACTIVE WASTE

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.	There were no spent resins, filter sludges, evaporator bottoms, etc. shipments during this reporting period.		
b. Dry compressible waste, soils, contaminated equipment, etc. (2)	Cubic Meter	250.6	1.00E1
	Ci	0.001	5.60E1
c. Irradiated components, control rods, etc.	There were no irradiated components, control rods, etc. shipments during this reporting period.		
d. Other (processed waste from HBPP via processor to burial)	There were no radioactive waste shipments to a processor for disposal during this reporting period.		

2. Estimate of major nuclide composition (by type of waste)	Unit	Nuclide	12 Month Period
a. Spent resins, filter sludges, evaporator bottoms, etc.	There were no spent resins, filter sludges, evaporator bottoms, etc. shipments during this reporting period.		
b. Dry compressible waste, soils, contaminated equipment, etc. (1)	%	H-3	42.0
	%	C-14	2.24
	%	Fe-55	1.63
	%	Co-60	5.24
	%	Ni-63	42.0
	%	Cs-137	6.8

TABLE 5 – Continued

2. Estimate of major nuclide composition (by type of waste)	Unit	Nuclide	12 Month Period
c. Irradiated components, control rods, etc.	There were no irradiated components, control rods, etc. shipments during this reporting period.		
d. Other (processed waste)	There were no radioactive waste shipments to a processor for disposal during this reporting period.		

3.a. Solid waste disposition from HBPP	Number of Shipments	Mode of Transportation	Destination
	13 [2]	Truck – NCF	US Ecology
3.b. Solid waste disposition via processor to disposal	N/A	N/A	via Toxco to Clive or WCS
B.1 Irradiated fuel shipments	None	N/A	N/A

Table Notes:

[1] Radionuclides contributing less than 0.1% to the total activity are not listed in Table 5.2.b. The value for I-129 is associated with propagation of the lower limit of detection, a requirement for reporting in waste manifests and was not identified by sample analysis.

[2] 13 shipments were made to US Ecology under a 10 CFR 20.2002 exemption.

IV. RADIOLOGICAL IMPACT ON MAN

A comparison of calculated doses from various paths has shown that the offsite doses are due to direct radiation. Maximum doses to individuals (for the maximally exposed organs and age groups) are summarized in Table 6. Doses from noble gases are not reported, as noble gas releases were neither expected nor measured. There are no airborne or liquid dose pathways from the adjacent ISFSI, and the direct radiation measurement locations for HBPP include the contribution from the ISFSI. Therefore, these doses comply with 40 CFR 190 as there are no other uranium fuel cycle facilities within 8 km of the HBPP and ISFSI.

A. Dose to the average individual in the population, based on the guidance of Regulatory Guide 1.109, from all receiving-water-related pathways is not calculated for 2019, because there were no batch liquid effluent releases during this report period. The last batch liquid effluent discharge occurred on December 11, 2013.

   With no batch liquid effluent discharge, doses continue to be well below the 10 CFR 50, Appendix I numerical guidelines for limiting effluents as low as is reasonably achievable (3 millirem per year (mrem/yr)) to the total body and 10 mrem/yr to any organ).

B. Total body dose to the average individual in the population from gaseous effluents to a distance of 50 miles from the site is 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 (determined with the same dispersion and deposition parameters as used to calculate maximum exposure) is not calculated, since there were no releases.

C. Total body dose (to the average individual in unrestricted areas from direct radiation from the facility) is based on thermoluminescent dosimeter (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 (teenage group). For this group, direct radiation would result in an exposure of 0.1 mrem/yr, calculated as follows:

   Specification 2.10 of the ODCM limits 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 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).

   Potential direct radiation exposure to an individual at the site boundary is highest at the north boundary of the site. Due to the possibility that an individual at the shoreline (fishing, bird watching, etc.) may use the path along the Coastal Trail, TLD Stations T8, T9, and T10 along the path have been historically used to estimate an annual radiation exposure. During 2019, these TLDs moved as the decommissioning progressed and the controlled area perimeter fence was relocated. TLD locations at the perimeter continue to conservatively represent the areas of the site accessible to the public. The ODCM calculation model for the direct radiation exposure pathway assumes a maximum occupancy factor of 67 hours per year, based on regulatory guidance for shoreline recreation for the teenage group.

T-1 was the highest perimeter monitoring point for 2019 at 62 mrem total. The average of all perimeter monitoring points (T1 through T-16) before background subtraction was 50.5 mrem with a variability from 49 mrem at location T-6 to 62 mrem at location T-1.

Total background dose for the year based on annual dose from offsite TLDs 1, 2, 14, 25, and 17 = (50.0 + 57.3 + 47.3 + 49.0 + 48.7)/5 = 50.5 mrem

Subtracting the yearly background dose from the maximum dose at T1:

62.0 mrem – 50.5 mrem = 11.5 mrem above background for the year

11.5 mrem corrected to the 67-hour occupancy: 11.5 x 67 hours/8760 hours per yr = 0.1 mrem additional at the fence line.

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. It is also well below the 25 mrem annual dose limitations in ODCM Specification 2.10 and 40 CFR 190.

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)	-	-	-	-	-
	-	-	-	-	-
Airborne Effluents					
Particulates (2)	-	-	-	-	-
	-	-	-	-	-
Direct Radiation (3)	0.06	0.03	0.03	0.03	0.1

Notes

1. Maximum total body and organ doses to individuals in unrestricted areas from receiving-water-related exposure pathways is not calculated since there were no batch liquid effluent releases during this report period. The last batch liquid effluent discharge occurred on December 11, 2013.
2. Maximum total body and organ dose to individuals in unrestricted areas from airborne effluent-related exposure pathways is not calculated since there were no airborne effluent releases during this report period. The plant stack was shut down in October 2015. Modular HEPA ventilation units were not used during the reporting period because no elevated airborne radioactivity areas were observed.
3. Total body dose (to the maximum individual in the population) is based on TLD results at locations near the site boundary, using the shoreline occupancy factors of Regulatory Guide 1.109 for the maximum potential individual (teenage group).

V. CHANGES TO THE OFFSITE DOSE CALCULATION MANUAL

As decommissioning proceeds at HBPP, system changes or removal may require changes to the ODCM. Changes were made to the ODCM during the reporting period. ODCM, Revisions 29, 30, and 31, are attached as Enclosures 2, 3, and 4, respectively, with a summary of changes that occurred during the reporting period.

VI. CHANGES TO THE PROCESS CONTROL PROGRAM

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

VII. CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

HBPP no longer performs batch liquid effluent discharges.

VIII. INOPERABLE EFFLUENT MONITORING INSTRUMENTATION

Liquid Effluent Monitoring

Effective December 23, 2013, HBPP no longer uses outfall canal dilution for liquid effluents. There were no batch liquid effluent releases during this report period.

Airborne Effluent Monitoring Instrumentation

No airborne radioactivity areas were identified in 2019, so no modular HEPA ventilation units were used during the reporting period.

SPAMS was removed from service on October 14, 2015.

IX. ERRATA

2018 Annual Radioactive Effluent Release Report Errata:

None

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
HUMBOLDT BAY POWER PLANT

SAFSTOR/Decommissioning Offsite Dose Calculation Manual
Revision 29

Summary of Changes Included in Revision 29 of the SAFSTOR/Decommissioning Offsite Dose Calculation Manual

Summary of Changes:

Page / Section	Change Date	Change	Reason
Page I-22 Figure 2-1	Rev. 29	Locations T-2, T-9, T-10, and T-11 moved during 2018.	Thermoluminescent dosimeter locations reflect changes in the perimeter fencing and areas that are no longer controlled to prevent pubic access.

(Procedure Classification - Quality Related)

INTRODUCTION

The SAFSTOR/DECOMMISSIONING 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 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 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 requirements for the REMP. 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. As of December 31, 2013, HBPP ceased liquid radioactive effluent discharges via the discharge canal to Humboldt Bay. The scope of the REMP and interlaboratory comparison program are the dosimeters and air samples required to evaluate the direct radiation and gaseous effluents from HBPP.

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 decommissioning period. Part II of the ODCM contains the calculational methods for gaseous and liquid effluents to preserve site specific data although the gaseous effluent pathway is limited to Modular HEPA Units on a selected basis and the liquid discharge pathway has been terminated.

The ODCM also contains the Process Control Program (PCP) for solid radioactive wastes, administrative controls regarding the content of the Annual Radiological Environmental Monitoring Program Report, administrative controls regarding the content of the Annual Radioactive Effluent Release Report, and administrative controls regarding major changes to radioactive waste treatment systems.

The ODCM shall become effective after approval by the HB Director. 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 required level of radioactive effluent control and not adversely impact the accuracy or reliability of effluent or dose 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 II - CALCULATIONAL METHODS AND PARAMETERS - (Continued)

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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 Deleted

1.4 Deleted

1.5 Deleted

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.

1.7 Deleted

1.8 FREQUENCY NOTATION

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

1.9 Deleted

1.10 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.

1.11 INSTANTANEOUS CONCENTRATION

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

1.12 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.

1.13 MODULAR HEPA VENTILATION UNIT

MODULAR HEPA VENTILATION UNIT consists of HEPA filter trains discharged to the environment and sampled in accordance with ANSI/HPS N13.1-1999.

1.14 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, 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 Monitoring Report and the Annual Radioactive Effluent Release Report. The ODCM also contains the Process Control Program (PCP) for solid radioactive wastes.

1.15 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).

1.16 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, disposal site(s) requirements, and other requirements governing the disposal of solid radioactive waste.

1.17 Deleted

1.18 RESTRICTED AREA

The RESTRICTED AREA is defined by 10CFR20.1003. The physical location(s) of the RESTRICTED AREA shall be defined in plant procedures.

1.19 SITE BOUNDARY

The SITE BOUNDARY shall be the boundary of the UNRESTRICTED AREA used in the offsite dose calculations for gaseous and liquid effluents. The SITE BOUNDARY is shown in Figure 1-1. Ingress and egress through the SITE BOUNDARY are controlled by the Company.

1.20 Deleted

1.21 Deleted

1.22 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.

1.23 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.”

1.24 VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing particulates from the gaseous exhaust stream prior to release to the environment.

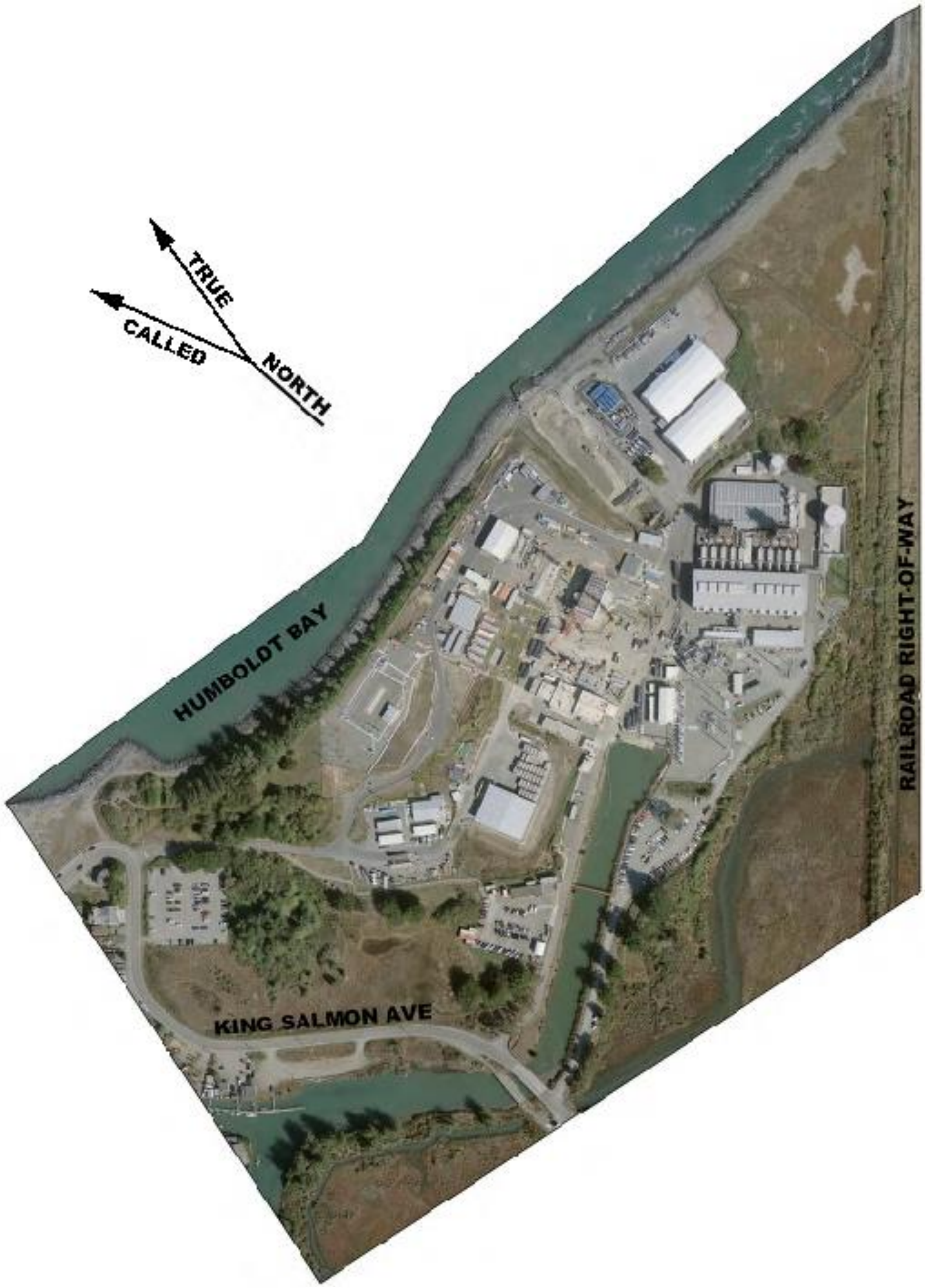
1.25 Deleted

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.

Figure 1-1
SITE BOUNDARY



2.0 SPECIFICATIONS

2.1 **Deleted; Table 2-1 - Deleted; Table 2.2 - Deleted**

2.2 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION[1]

LIMITING CONDITIONS

2.2.1 Deleted - plant stack is no longer in operation.

SURVEILLANCE REQUIREMENTS

2.2.2 Deleted

Table 2-3 - Deleted
Table 2-4 - Deleted

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

Deleted (See BASES Section 3.2 and Appendix A)

Table 2-5 (Deleted)

2.4 LIQUID EFFLUENT – DOSE Deleted - No longer applicable

2.5 Deleted - No longer applicable

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

SURVEILLANCE REQUIREMENTS

2.6.2 Deleted (see BASES section 3.5)

2.6.3 Deleted (see BASES section 3.5)

2.6.4 Radioactive particulates, with half-lives of greater than 8 days, in gaseous effluents released to the environment shall be sampled and analyzed in accordance with the sampling and analysis program of Table 2-6, and their concentrations shall be compared with the limits of 10CFR20, Appendix B, Table 2, Column 1. IF their concentrations exceed those limits, the calculational methods in Part II of the ODCM shall be used to determine whether or not the limits of Specification 2.6.1 have been exceeded. The actual sample period shall be used to determine the dose rate during the sample period.

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]
Modular HEPA Ventilation Discharge				
	Continuous[b,d]	W[b] Mixing Box Particulate Sample	Principal Gamma Emitters[e]	1×10^{-11}
	Continuous[b,d]	W[b] Mixing Box Particulate Sample	Gross Alpha	1×10^{-12}
	Continuous[b,d]	W[b] Mixing Box Particulate Sample	Gross Beta	6.7×10^{-12}
	Continuous[b,d]	Q Composite of Mixing Box Particulate Samples	Sr-90[g]	1×10^{-11}
	Continuous[b,d,h]	Q Composite of Mixing Box Particulate Samples	Am-241	1×10^{-12}
	Continuous[b,d,i]	Q Composite of Mixing Box Particulate Samples	Am-241	1×10^{-14}

Table 2-6 (Continued)

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),

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. NOTE: The LLDs are achievable with a reasonable count time assuming adequate effluent volume and sample volume. If the LLD is not achieved, initiate a condition report to document that the LLD was not achieved and indicate a probable cause (short runtime, equipment malfunction, etc.). RP Supervision will determine if additional calculations should be performed per Surveillance 2.6.4.

Table 2-6 (Continued)

Table Notation (Continued)

[b] Samples shall be changed at least once per 7 days (3 day extension permitted), assuming effluent pathway is in continuous use (typically > 40 hrs per week). Samples may be collected more frequently for short duration use of a Modular HEPA Ventilation Unit.

[c] Deleted

[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, and 2.8.

[e] The principal gamma emitters for which the LLD specification applies exclusively are 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.

[f] Deleted based on SPAMS no longer in service.

[g] Analysis specific to Sr-90 may be replaced by analysis for total radioactive Strontium.

[h] When release volume is less than or equal to 3.26×10^{11} ml (e.g., 1.15E+7 cubic feet).

[i] When release volume exceeds 3.26×10^{11} ml (e.g., 1.15E+7 cubic feet).

2.7 Deleted

2.8 GASEOUS EFFLUENTS: DOSE - RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITIONS

2.8.1 The dose to a MEMBER OF THE PUBLIC from the release of 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 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 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 dose calculation for the current calendar quarter and the current calendar year, for the release of radioactive materials in particulate form with half-lives greater than 8 days,
OR

Perform a BASELINE COMPARISON for gaseous effluent radioactivity (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.
OR

Perform a dose assessment, if weekly sampling indicates the effluent from modular HEPA units exceed 0.1 uCi of alpha emitters or Sr-90. The assessment of alpha and beta may be performed with appropriate compensation for naturally occurring nuclides.

As explained in Specification Bases section 3.8, neither routine surveillance nor dose calculations are required for Tritium in gaseous effluents.

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 disposal site(s) requirements.

APPLICABILITY: At all times.

ACTION:

With the provisions of the 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) meet the shipping, disposal site(s) requirements with regard to dewatering and off site vendor processes.

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 gaseous effluents exceeding twice the limits of Specification 2.8.1.a, or 2.8.1.b, calculations should be made, which include direct radiation contributions from Unit No. 3, 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 gaseous effluents shall be calculated in accordance with dose calculation methodology provided for Specification 2.8.1.

2.11 REMP MONITORING PROGRAM

LIMITING CONDITIONS

2.11.1 A radiological environmental monitoring program shall be provided to monitor the radiation and radionuclides in the environs of the facility. The 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 Program Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. A Special Report pursuant to Administrative Control 4.3, 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 Specification 2.8. Prepare and submit to the Commission within 30 days of obtaining analytical results from the affected sampling period which includes an evaluation of release conditions, environmental factors or other aspects which caused the dose limits to be exceeded. 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 Program Report.

SURVEILLANCE REQUIREMENTS

2.11.2 The radiological environmental monitoring samples shall be collected pursuant to Table 2-7 from the “Quality Related” locations given in Tables 2-7 and 2-10 and Figures, 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

	PROGRAM DESCRIPTION			PROGRAM BASIS
Exposure Pathway and/or Sample	**Number of Samples and Locations**[a]	**Sampling and Collection Frequency**	**Type of Analysis**	**ODCM Specs (QR)**
AIRBORNE	4 onsite locations, 1 offsite location	Continuous sampler operation with sample collection at least once per 7 days[1][c]	Gross alpha and gross beta radioactivity following filter change Gamma isotopic[b] analysis on quarterly composite (by station)	X
DIRECT RADIATION	Minimum of 8 onsite stations, at or within the SITE BOUNDARY fence line, 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
WATERBORNE	None	N/A	N/A	
INGESTION	None	N/A	N/A	
TERRESTRIAL	None	N/A	N/A	

Table Notations

QR - Quality Related

[1]Performed by HBPP

[3]Performed by a NVLAP accredited processor

[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 Program 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] Gamma isotopic analysis means the identification and quantification of gamma emitting radionuclides that may be attributable to the effluents from the facility.

[c] Continuous sampler operation may be limited to normal work hours to represent effluents from decommissioning activities. Count times may need to be adjusted to achieve the recommended LLDs in Table 2-9.

Table 2-8 (Deleted)

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

Analysis	Airborne Particulate (pCi/m^3)
Gross Beta	0.01
H-3	
Co-60	
Cs-137	0.06

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 Monitoring Program 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:

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

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.

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 Radiological Environmental Monitoring Program 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.

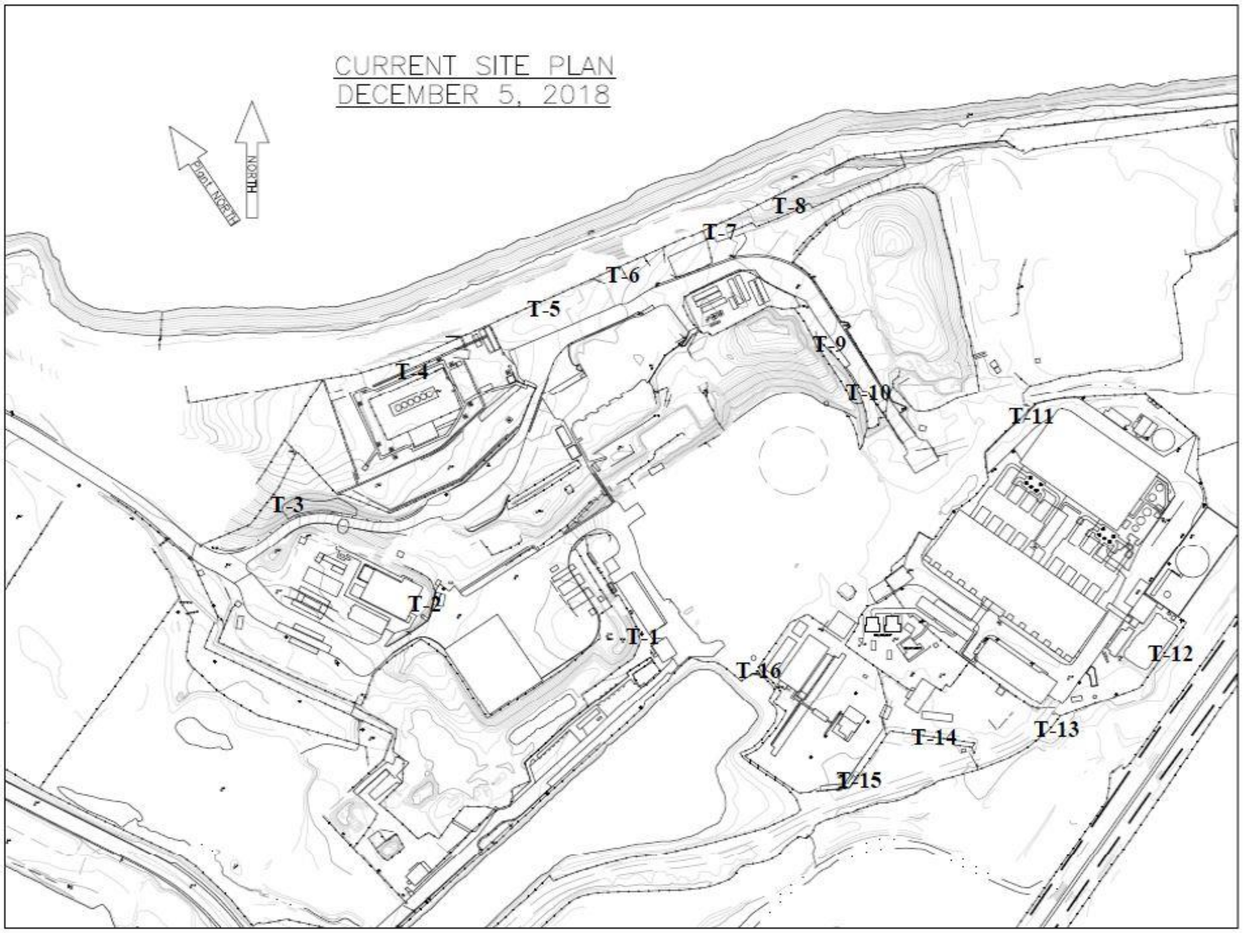
Table 2-10
DISTANCES AND DIRECTIONS TO ENVIRONMENTAL MONITORING STATIONS

Station No.	Code	Station Name	Radial Direction Sector	Radial Direction By Degrees	Radial Distance from Plant (Miles)
1	**Δ**	**King Salmon Picnic Area**	**W**	**270**	**0.3**
2	**Δ**	**180 Dinsmore Drive, Fortuna**	**SSE**	**158**	**9.4**
3	□	Humboldt Hill Road at Bret Harte Lane	SSE	158	0.9
14	**Δ**	**South Bay School Parking Lot**	**S**	**180**	**0.4**
17	**Δ**	**Control Set at Humboldt Substation, Eureka**	**NEE**	**61**	**5.8**
25	**Δ**	**Irving Drive, Humboldt Hill**	**SSE**	**175**	**1.3**

Table Notations

Code: Δ Dosimetry Station □ Air Particulate Station

Figure 2-1
HBPP Onsite TLD Locations

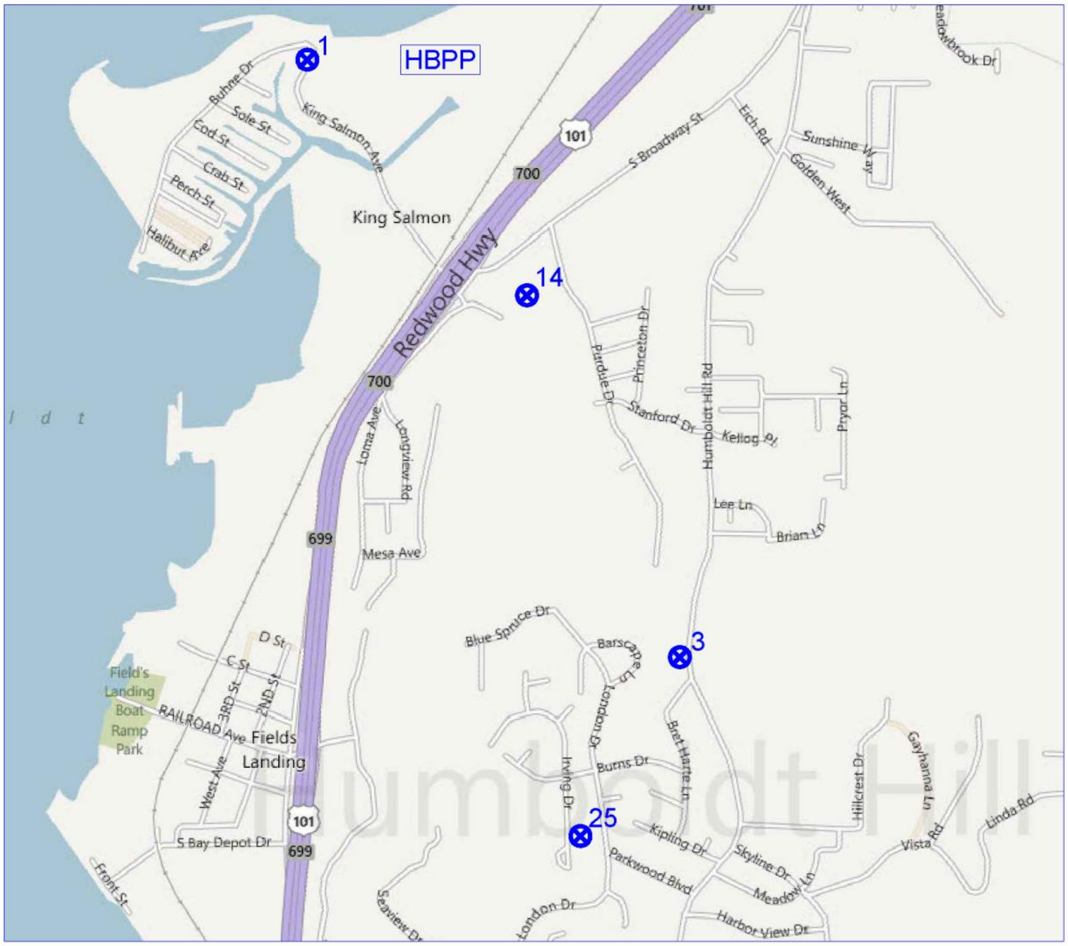


02/19

Monitoring locations T7, T10, T11, T13, T16, T2, T3, and T5 generally represent REMP Site Boundary direct exposure monitoring locations in the 8 primary compass points beginning with T-7 to representing north and moving clockwise.

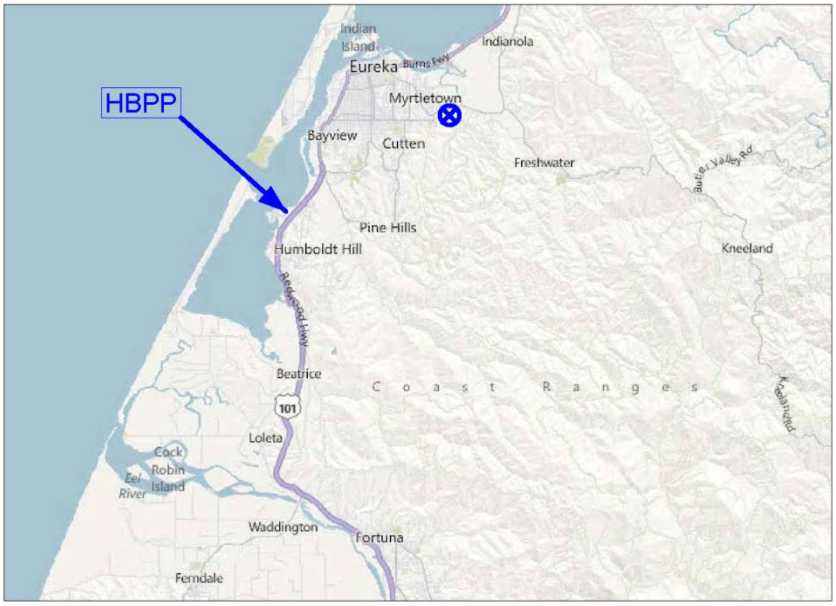
Figure 2-2 - Deleted

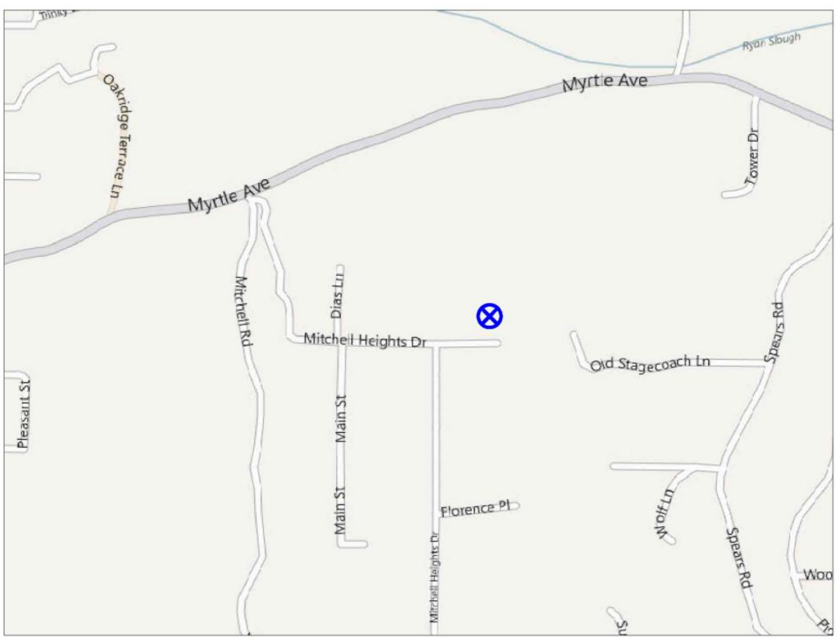
Figure 2-3
HBPP OFFSITE SAMPLING LOCATIONS - HUMBOLDT HILL



Station	GPS Coordinates (NAD83/NAVD88 CA. Zone 1)			Decimal Degrees	
	Easting	Northing	el.	Latitude	Longitude
1	5948026.52	2161183.79	11.38	40.74156	-124.21903
3	5951260.28	2155706.11	234.94	40.72676	-124.20274
14	5949876.83	2158864.39	18.65	40.73533	-124.20802
25	5950247.30	2154214.18	229.22	40.72260	-124.20626

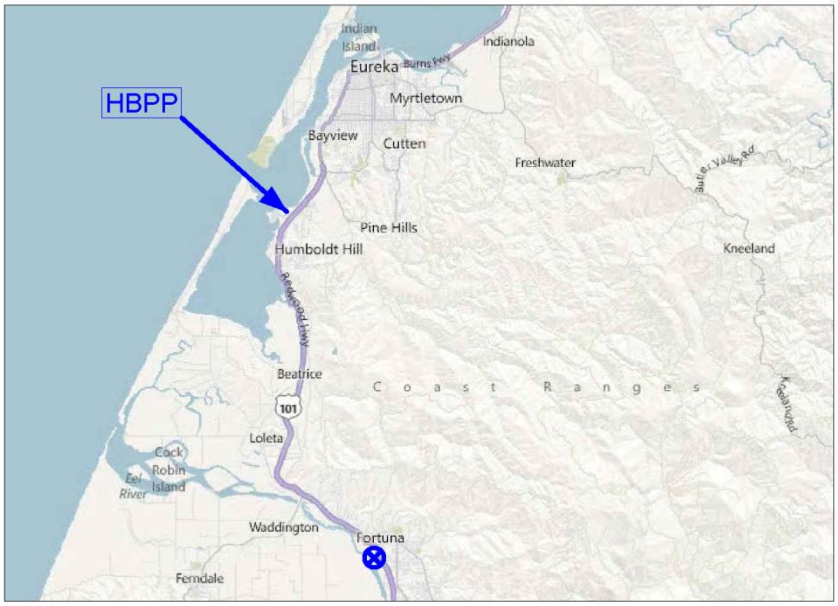
Figure 2-4
HBPP OFFSITE SAMPLING LOCATIONS - EUREKA

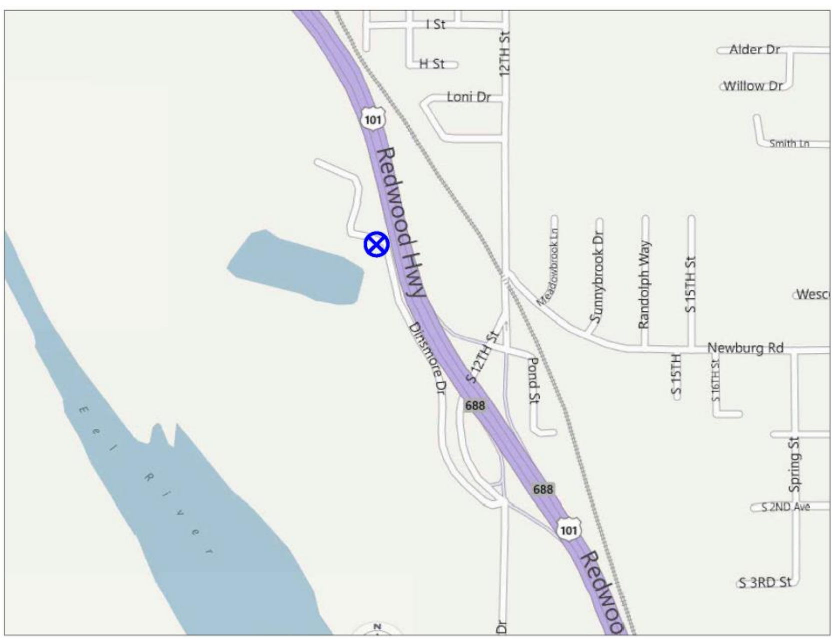




Station	GPS Coordinates (NAD83/NAVD88 CA. Zone 1)			Decimal Degrees	
	Easting	Northing	el.	Latitude	Longitude
17	5976549.55	2175490.19	164.85	40.78276	-124.11324

Figure 2-5
HBPP OFFSITE SAMPLING LOCATIONS - FORTUNA





Station	GPS Coordinates (NAD83/NAVD88 CA. Zone 1)			Decimal Degrees	
	Easting	Northing	el.	Latitude	Longitude
2	5962583.86	2105797.82	35.53	40.59057	-124.15746

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 Program Report.

SURVEILLANCE REQUIREMENTS

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

2.13 RADIOACTIVE WASTE INVENTORY

LIMITING CONDITIONS

2.13.1 Liquid Radioactive Waste In Outdoor Tanks

The radiological inventory of wastes in outdoor tanks that are not capable of retaining or treating tank overflows shall not exceed 0.25 Ci.

APPLICABILITY: At all times.

ACTION:

When the inventory exceeds the conditions as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Monitoring Program Report.

2.13.2 Deleted

SURVEILLANCE REQUIREMENTS

2.13.3 An inventory of the estimated liquid radioactive waste in outdoor tanks inventory shall be maintained to verify the 0.25 Ci limit is not exceeded.

OR

Provide overflow protection.

OR

Use process knowledge of typical concentration and tank volume to verify that the 0.25 Ci is not exceeded.

3.0 SPECIFICATION BASES

3.1 Radioactive Gaseous Effluent Monitoring Instrumentation Basis

Deleted – The plant stack ceased operation in 2015. Monitoring gaseous effluent is limited to sampling and analysis of Modular HEPA Units.

3.2 Liquid Effluent Concentration Basis

Deleted - Liquid effluents are no longer discharged to Humboldt Bay. Effective December 31, 2013, discharge of processed radioactive liquid effluents to Humboldt Bay was terminated. Any remaining or incidental radioactive liquid in concentrations exceeding 10 times 10 CFR 20, Appendix B, Table 2 Column 2 are manifested for disposal at a licensed disposal site. Sampling and manifesting requirements are consistent with the requirements of the receiving facility not subject to ODCM methodology.

3.3 Liquid Effluent Dose Basis

Deleted - Liquid effluents are no longer discharged to Humboldt Bay.

3.4 Liquid Effluent Treatment Basis

Deleted - Liquid effluents are no longer discharged to Humboldt Bay.

3.5 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).

Stack operation and monitoring ceased operation in 2015, so the reporting period for 2015 includes the dose contribution from the plant stack prior to ceasing operation. Modular HEPA Ventilation Units continue to be sampled as a gaseous effluent pathway.

Noble gas monitoring is not required because the spent fuel (noble gas source term) has been transferred to the ISFSI. Tritium monitoring is not required in gaseous effluents because the tritium source term was the spent fuel pool water which is now empty. Residual water in various plant drains and sumps contain low levels of tritium (generally at or below the drinking water standard (2E-5 uCi/ml or 20,000 pCi/L) and does not require monitoring as a gaseous plant effluent.

3.6 Deleted

Gaseous effluent monitoring is not required for noble gases because the spent fuel (noble gas source term) has been transferred to the ISFSI.

3.7 Deleted

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 basis for the dose calculation threshold of 0.1 uCi alpha emission or Sr-90 in a week assumes a continuous ground level release of 1.65E-13 uCi/sec and an X/Q of 6.59E-3 sec/m^3. The limiting inhalation dose is to a teen age member of the public at the site boundary at approximately 0.3 mrem/wk (15 mrem/yr) to the bone from alpha emitters. Compliance with this Specification has been established on a licensing basis by the SAFSTOR Environmental Report and NUREG-1166, “Final Environmental Statement for Decommissioning Humboldt Bay Power Plant.” These reports have demonstrated that routine release of Tritium and radioactive materials in particulate form (with half-lives greater than 8 days) in gaseous effluents during decommissioning will not cause the Specification to be exceeded. As long as routine releases do not exceed the baseline quantities evaluated in these reports, no further dose calculation is necessary.

The previously evaluated tritium source term was the spent fuel pool water, which is now empty. Residual water in various plant drains and sumps contain low levels of tritium (at or below the drinking water standard (2E-5 uCi/ml or 20,000 pCi/L) and does not require monitoring as a gaseous plant effluent.

3.9 Solid Radioactive Waste Basis

This Specification ensures that radioactive wastes that are transported from the site shall meet the disposal site(s) licensee and/or waste acceptance criteria for free standing liquids 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 is required to provide assurance that the baseline conditions established by the Environmental Report are not deteriorating and it supplements the SAFSTOR Environmental Report baseline

environmental conditions by conducting onsite and offsite environmental monitoring to evaluate routine conditions during decommissioning and to document any increased nuclide concentrations and/or radiation levels resulting from accidents during decommissioning.

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.

The LLD's required by Table 2-9 are considered optimum for routine environmental measurements in industrial laboratories. HBPP no longer includes water, milk, fish, food products, or sediment in its routine REMP sampling program. Sampling and analysis in support of the License Termination Plan is independent of the ODCM requirements.

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.

3.13 Radioactive Waste Inventory Basis

The requirements for limits on the accumulation of liquid radioactive waste in outdoor tanks were transferred from the license Technical Specifications.

4.0 ADMINISTRATIVE CONTROLS

4.1 Annual Radiological Environmental Monitoring Report

A report on the Decommissioning Radiological Environmental Monitoring Program shall be prepared annually in accordance with the NRC Branch Technical Position and submitted to the NRC by May 1 of each year.

The Annual Radiological Environmental Monitoring 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. The material provided shall be consistent with the objectives outlined in the ODCM, and in 10CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

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 Report Annual Summary, or equivalent. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in the next annual report.

d. A summary description of the Decommissioning 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. Deleted – water samples are not collected as a part of the REMP.

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

Table 4-1
RADIOLOGICAL ENVIRONMENTAL MONITORING REPORT ANNUAL 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							
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	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
Groundwater	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
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 REPORT ANNUAL 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 as 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].

Not Required - not required by the HBPP Offsite Dose Calculation Manual. 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. 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. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10CFR 50.36a and 10CFR Part 50, Appendix I, Section IV.B.I. Beginning in the reporting year 2014, liquid effluents shipped for processing or disposal at a regulated disposal site are included in the annual report.

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 gaseous monitoring instrumentation inoperable for greater than 30 days in accordance with Specification 2.2. Beginning the reporting year 2015, following cessation of the plant stack operation, the effluent monitoring instrumentation associated with Specification 2.2 ceased operation. Inoperability and lack of timely corrective action is only applicable to the period of plant stack operation. Anomalies associated with monitoring effluent from Modular HEPA Ventilation systems will be reported.

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 2.8 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 approved by the HB Director.

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.

4.5 Process Control Program Changes

a. Changes to the Process Control Program (PCP) shall be documented and records of reviews performed shall be retained as required for the duration of Decommissioning.

b. The following information shall be available for review:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

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

c. The change shall become effective after approval of the HB Director.

PART II - CALCULATIONAL METHODS AND PARAMETERS

1.0 UNRESTRICTED AREA EFFLUENT CONCENTRATIONS

1.1 LIQUID EFFLUENT UNRESTRICTED AREA CONCENTRATIONS

Specification 2.3.1 requires that the Radioactive Liquid Effluent Sample concentrations (RLES) are calculated 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). This requirement is defined by the following relationship.

$$\sum_i \frac{C_{i,\,Canal}}{10 \times ECL_i} \leq 1 \qquad (1\text{-}1)$$

where:

$C_{i\text{-}Canal}$ = The concentration of isotope " i " in the canal discharge point to Humboldt Bay.

ECL_i = Effluent Concentration Limit for radionuclide " i " from 10 CFR 20, Appendix B, Table 2, Column, 2 (μCi/ml)

1.1.1 If the outfall location is not at the furthermost portion of the canal from the entrance to the Bay the concentration of the isotope $C_{i\text{-}Canal}$ is equal to the concentration being discharged at the outfall.

1.2 UNRESTRICTED AREA GASEOUS EFFLUENT CONCENTRATIONS

1.2.1 Equation C-4 of Regulatory Guide 1.109 demonstrates how to calculate dose from inhalation:

The annual dose associated with inhalation of all radionuclides, to organ j *of an individual in age group* a*, is then:*

$D_{ja}(r,\theta) = R_a \Sigma x_i(r,\theta) DFA_{ija}$

where

D_{ja} *is the annual dose rate to organ* j *of an individual in age group* a

R_a *is the breathing rate for age group* a

$x_i(r,\theta)$ *is the* annual *average ground-level concentration of nuclide* i *in air in sector* θ *at distance* r, *in* pCi/m^3

DFA_{ija} *is the dose factor for nuclide* i *to organ* j *of age group* a

To calculate $x_i(r,\theta)$ the annual average ground-level concentration of nuclide i in air in sector θ at distance r, in pCi/m^3 the equation must be rearranged to:

$D_{ja}(r,\theta)/(\text{DFAija}\, R_a) = x_i(r,\theta)$

Assuming that:

Americium-241 is the primary nuclide

The maximally exposed group is the Teen based on breathing rates and DFAija

The DFAija to the bone of a Teen from Am-241 is 1.77 mrem/pCi

The DFAija are taken from: NRC NUREG/CR-4013, "*LADTAP-II Technical Reference and User Guide*"

The Teen breathing rate is 8000 m^3/year

Therefore the ground-level concentration of Am-241 in air in sector θ at distance r, in pCi/m^3 that will produce a dose rate of 1500 mrem/year to the bone of a Teen is:

(1500 mrem/year) / (1.77 mrem/pCi) / (8000 m^3/year) = 1.06E-1 pCi/ m^3

1.06E-1pCi/ m^3 =

(1.06E-1 pCi/m^3) / (1E6 pCi /µCi) / (1E6 ml/m^3) = 1.06E-13 µCi/ml

1.2.2 Quantity of radioactive material released

Equation C-3 of Regulatory Guide 1.109 demonstrates how to calculate the quantity of material that must be released to produce a given airborne concentration:

The annual average airborne concentration of radionuclide i at the location (r, θ) with respect to the release point may be determined as

$$x_i(r,\theta) = 3.17 \times 10^4\ Q_i(\chi/Q)^D(r,\theta)$$

where

$x_i(r,\theta)$	*is the annual average ground-level concentration of nuclide* i *in air in sector* θ *at distance* r, *in* pCi/m^3
3.17×10^4	*is the number of pCi/Ci divided by the number of sec/yr*
$(\chi/Q)^D(r,\theta)$	*is the annual average atmosphere dispersion factor, in sec/m^3.*
Q_i	*is the release rate of nuclide I to the atmosphere, in Ci/yr*

A value of 6.59E-3 sec/m^3 was used for the incidental release path atmosphere dispersion factor at the site boundary $(\chi/Q)^D(r,\theta)$ for releases from Modular HEPA Units. This is based on a release rate of 2000 cfm. (Ref: Safstor ODCM, Appendix B, 2.0) 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*.

To determine the release rate that will result in an average ground-level concentration the above equation must be rearranged to:

$$Q_i = x_i(r,\theta) / (3.17 \times 10^4(\chi/Q)^D(r,\theta))$$

Therefore the Modular HEPA Unit release rate of Am-241 required to equal the incidental ground-level concentration at the site boundary calculated above is:

1.06E-1 pCi/m^3 / ((3.17E4 (pCi/Ci)/ (sec/yr)) * (6.59E-3 sec/m^3)) =

5.07E-4 Ci/yr or 5.07E2 uCi/yr

1.2.3 Transmission Fraction

Deleted – no on line monitoring provided.

1.2.4 Effluent Concentration

The Modular HEPA Unit concentration that would result in a release rate of 5.07E-4 Ci/yr is equal to:

Total release (Curies/year) / Release rate (cc/year)

The average annual Modular HEPA Unit flow rate is 2,000 cfm

This results in a total volume of 2.98E13 cc/yr

This is based on (2000 ft^3/min * 525,600 minutes/yr * 28,317 cc/ft^3).

(5.07E-4 Ci * 1E6 µCi/Ci) / (2.98E13 cc/yr) = 1.70E-11 µCi/cc

Therefore an indicated Modular HEPA concentration of 1.70E-11 µCi/cc at 2000 cfm for one calendar year would result in a dose of 1500 mrem to a member of the public at the site boundary.

Two times the indicated release rate is equal to3.4E-11 µCi/cc.
Two hundred times the indicated release rate is equal to 3.4E-9 µCi/cc.

1.2.5 Relationship to EPA PAG

To compare the release rates calculated above the following assumptions were made:

Am-241 dose conversion factor in rem / cm^{-3} µCi hr, from EPA 400 = 5.3E8

Since no credit is taken for an elevated release point or an annual average χ/Q the same atmospheric dispersion factor is used in the calculations below.

Assuming that an unplanned release occurs at two times the ODCM release rate for one hour the total activity released is equal to:

3.4E-11 µCi/cc * 2000 ft^3/min * 28,317 cc/ft^3 * 60 min = 1.16E-1 µCi

(1.16E-1 µCi) * (5.3E8 rem / cm^{-3} uCi hr) * (6.59E-3 sec/m^3) / (1E6 cm^3/m^3) / (3600 sec/hour) = 1.13E-4 rem

This is much less than the EPA PAG of 1 Rem

Assuming that an unplanned release occurs at two hundred times the ODCM release rate for 15 minutes the total activity released is equal to:

3.4E-9 µCi/cc * 2000 ft^3/min * 28,317 cc/ft^3 * 15 min = 2.89E0 µCi

This results in a dose of:

(2.89E0 µCi) * (5.3E8 rem / cm^{-3} uCi hr) * (6.59E-3 sec/m^3) / (1E6 cm^3/m^3) / (3600 sec/hour) =

2.80E-3 rem

This is much less than the EPA PAG of 1 Rem.

1.2.6 Relationship to 10CFR20 Appendix B Table 2 Effluent Concentration limits

The 10CFR20 Appendix B Table 2 Effluent Concentration limit for Am-241 is 2E-14 µCi/ml.

The average annual ground-level concentration in air (x_i) in pCi/m^3 is equal to:

x_i = (3.17E4 (pCi/Ci)/ (sec/year)) * Q * (X/Q)

Where Q is equal to the quantity of radioactive material released in a year in Curies/year

ODCM Modular HEPA Unit incidental release X/Q = 6.59E-3 sec/ m^3

If x_i = 2E-14 µCi/ml then:

Q = (2E-14 µCi/ml * 1E6 ml/m^3 * 1E6 pCi/µCi) / ((3.17E4 (pCi/Ci)/ (sec/yr)*(6.59E-3 sec/ m^3))

Q = 9.57E-5 Ci/yr

The average annual Modular HEPA Unit volume based on the ODCM is 2.98E13 cc/yr.

This is based on (2000 cfm * 525,600 minutes/yr * 28,317 cc/cfm).

Therefore, the Modular HEPA Unit effluent concentration required to result in a fence-line concentration of 2E-14 µCi/ml is:

(9.57E-5 Ci/yr * 1E6 µCi/Ci) / (2.98E13 cc/yr * 1 cc/ml) = 3.2E-12 µCi/ml

1.2.7 Conversion Factor from Effluent Concentration to µCi/day

The release rate in µCi/day = Modular HEPA Unit concentration in µCi/cc * 2000 ft^3/min * 1440 minutes/day * 28317 cc/ ft^3

The release rate in µCi/day = Modular HEPA Unit concentration in µCi/cc * 8.16E10 cc/day

1.2.8 Conversion Factor from µCi/day to % of NUE
An NUE is equal to a release rate of 3000 mrem/year

%NUE = (Offsite dose rate / NUE threshold) * 100

%NUE = ((Conversion Factor * Release Rate) / NUE threshold) * 100

%NUE = ((Conversion Factor * 100) / NUE threshold) * Release Rate

The Conversion Factor is equal to (1.77E6 mrem/µCi) * (6.59E-3 sec/ m^3) * (8000 m^3/year) / (8.64E4 sec/day)

This is equal to1.08E3 mrem/year per µCi/day

1.2.9 Results

The 10CFR20 Appendix B Table 2 Effluent Concentration limit for Am-241 is 2E-14 µCi/ml. The Modular HEPA Unit effluent concentration that would result in a fence-line concentration of 2E-14 µCi/ml is 3.2E-12 µCi/ml.

3.2E-12 uCi/ml * 8.16E10 cc/day * 1ml/cc * 1.08E3 mrem-day/uCi-yr = 4.70E2 mrem/yr.
470 mrem/yr / 8760 hr/yr = 5.365E-2 mrem/hr

Assuming that an unplanned release occurs at two times the ODCM release rate for one hour the offsite dose corresponding to an NUE would be 1.07E-4 rem (0.107 mrem) which is much less than the EPA PAG.

Assuming that an unplanned release occurs at two hundred times the ODCM release rate for fifteen minutes the offsite dose corresponding to an Alert would be 2.675E-3 rem (2.7 mrem) which is much less than the EPA PAG.

Note that Am-241 is used in the example calculations and is expected to be limiting. Other alpha emitting isotopes such as Pu-238, Pu-239/240 and Cm-243/244 are evident in the contamination at HBPP. Since the Effluent Concentration Limits (ECLs), Derived Air Concentration (DAC) values and organ Dose Conversion Factors (DCFs) are similar, the Am-241 values may be assumed to be gross alpha with appropriate compensation for naturally occurring isotopes.

Other radionuclides (Co-60, Sr-90, Cs-137, etc.) are important in determining actual offsite dose and in demonstrating compliance with the ECL using the sum of the fractions rule. The example calculations are used similarly for each isotope in the mix with their respective ECL, DCF and exposure pathway (inhalation, ingestion, and submersion).

Although not relevant to the hypothetical offsite dose calculation in the ECL and NUE analysis above, assumed effluent concentrations are approximately 1 DAC, 2 DAC, and 200 DAC for Am-241 at the point of release. Airborne radioactivity control measures to control worker dose, also limits the potential offsite dose.

2.0 LIQUID EFFLUENT DOSE CALCULATIONS

2.1 MONTH (31 DAY PERIOD) Deleted

2.2 CALENDAR QUARTER - Deleted

2.3 CALENDAR YEAR - Deleted

2.4 LIQUID EFFLUENT DOSE CALCULATION METHODOLOGY

As of December 31, 2013, HBPP has ceased liquid radioactive effluent discharges via the discharge canal to Humboldt Bay. Any remaining processed liquid radioactive waste is transported offsite for land disposal at an authorized disposal facility. The following calculation methodology is preserved as a part of the ODCM for ease of reference to site specific parameters in the event of an accidental release of liquid radioactive effluent. No recurring liquid effluent dose calculations are expected for the remainder of decommissioning.

The equations specified in this section 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.

Equation (2) of Regulatory Guide 1.109 provides for the use of a site specific mixing ratio (i.e. reciprocal of the dilution factor) that describes the near term and near field mixing of the tidal flow from the Discharge Canal into Humboldt Bay. A two-dimensional numerical analysis, depth-averaged, finite element hydrodynamic model (reference 12.1) was developed by CH2MHILL and used to estimate the dispersion of the canal discharge in the Bay. The analysis indicated that an additional dilution factor of 80 for batch release applications or a dilution factor of 20 for continuous release applications can conservatively be used to describe the Bay dilution. A factor of 20 will be applied in this calculation to address any combination of release modes.

Since the intake canal contains a larger volume of water, use of the above dilution factors for effluent releases to the intake canal provides a simplified, conservative methodology for calculating annual dose from effluent releases to the intake canal.

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:

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

where:

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

$C_{i\text{-Bay diluted}}$ = The average diluted Bay concentration, pico-Curie/liter, for radionuclide, i. If the outfall to the canal is at the furthest most portion of the canal from the entrance to the Bay, this will be estimated by calculating the total activity released (e.g. effluent concentration $C_{i\text{ effluent}}$ in pCi/L times the discharge volume V_D in Liters) then dividing the total activity of the nuclide discharged during the period, pico-Curies, by the dilution volume (e.g. total discharged volume V_D plus total tidal flow V_{TD} during the period in liters), and by the Bay dilution factor of 20. The total annual tidal flow for the outfall canal is 2.47E+9 Liters/year (e.g., 6.77E+6 Liters/day). If Gross Alpha radioactivity is determined to be in the effluent , Pu-241 will be considered to be present at 3.25 times the amount of detected Gross Alpha radioactivity. Note that the resulting dose commitment is the annual dose rate (mrem per year) for a time frame with this average concentration. Doses (NOT dose rates) for periods shorter than a year must be proportionately reduced.

$$C_{i\text{ - Bay diluted}} = \frac{C_{i\text{ - Effluent}} \times V_D}{(V_D + V_{TD}) \times 20} \tag{2-2}$$

If the outfall is not located in the furthest most portion of the canal from the entrance to the Bay, no credit for tidal dilution of the canal will be taken and the diluted Bay concentration will be calculated using the following equation.

$$C_{i\text{ - Bay diluted}} = \frac{C_{i\text{ - Effluent}}}{20} \tag{2-3}$$

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.

Dose calculations can be performed using the above methodology for the current month, quarter, or year.

Table 2-1
Ingestion Dose Factors for Adult Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from NUREG/CR-4013 (LADTAP II input values)

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	5.99×10^{-8}	5.99×10^{-8}	5.99×10^{-8}	5.99×10^{-8}	5.99×10^{-8}
Co-60	No Data	2.14×10^{-6}	4.72×10^{-6}	No Data	No Data	4.02×10^{-5}
Ni-63	1.30×10^{-4}	9.01×10^{-6}	4.36×10^{-6}	No Data	No Data	1.88×10^{-6}
Sr-90	8.71×10^{-3}	No Data	1.75×10^{-4}	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}
Pu-241	1.57×10^{-5}	7.45×10^{-7}	3.32×10^{-7}	1.53×10^{-6}	No Data	1.40×10^{-6}
Am-241	7.55×10^{-4}	7.05×10^{-4}	5.41×10^{-5}	4.07×10^{-4}	No Data	7.42×10^{-5}
Gross α	7.55×10^{-4}	7.05×10^{-4}	5.41×10^{-5}	4.07×10^{-4}	No Data	7.42×10^{-5}

Table 2-2
Ingestion Dose Factors for Teen Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from NUREG/CR-4013 (LADTAP II input values)

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	6.04×10^{-8}	6.04×10^{-8}	6.04×10^{-8}	6.04×10^{-8}	6.04×10^{-8}
Co-60	No Data	2.81×10^{-6}	6.33×10^{-6}	No Data	No Data	3.66×10^{-5}
Ni-63	1.77×10^{-4}	1.25×10^{-5}	6.00×10^{-6}	No Data	No Data	1.99×10^{-6}
Sr-90	1.02×10^{-2}	No Data	2.04×10^{-4}	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}
Pu-241	1.75×10^{-5}	8.40×10^{-7}	3.69×10^{-7}	1.71×10^{-6}	No Data	1.48×10^{-6}
Am-241	7.98×10^{-4}	7.53×10^{-4}	5.75×10^{-5}	4.31×10^{-4}	No Data	7.87×10^{-5}
Gross α	7.98×10^{-4}	7.53×10^{-4}	5.75×10^{-5}	4.31×10^{-4}	No Data	7.87×10^{-5}

Table 2-3
Ingestion Dose Factors for Child Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from NUREG/CR-4013 (ladTAP II input values)

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	1.16×10^{-7}	1.16×10^{-7}	1.16×10^{-7}	1.16×10^{-7}	1.16×10^{-7}
Co-60	No Data	5.29×10^{-6}	1.56×10^{-5}	No Data	No Data	2.93×10^{-5}
Ni-63	5.38×10^{-4}	2.88×10^{-5}	1.83×10^{-5}	No Data	No Data	1.94×10^{-6}
Sr-90	2.56×10^{-2}	No Data	5.15×10^{-4}	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}
Pu-241	3.87×10^{-5}	1.58×10^{-6}	8.04×10^{-7}	2.96×10^{-6}	No Data	1.44×10^{-6}
Am-241	1.36×10^{-3}	1.17×10^{-3}	1.02×10^{-4}	6.23×10^{-4}	No Data	7.64×10^{-5}
Gross α	1.36×10^{-3}	1.17×10^{-3}	1.02×10^{-4}	6.23×10^{-4}	No Data	7.64×10^{-5}

Table 2-4
Bioaccumulation Factors for Saltwater Environment
(pCi/kg per pCi/liter)
Selected Nuclides from Regulatory Guide 1.109, Table A-1 and from NUREG/CR-4013

Element	Fish	Invertebrate
H	9.0×10^{-1}	9.3×10^{-1}
Co	1.0×10^{2}	1.0×10^{3}
Ni	1.0×10^{2}	2.5×10^{2}
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}
Pu	3.0	2.0×10^{2}
Am	2.5×10^{1}	1.0×10^{3}
Gross α	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 EFFLUENT TREATMENT

3.1 TREATMENT REQUIREMENTS

3.1.1 Deleted

3.1.2 Deleted

3.2 Deleted

4.0 GASEOUS EFFLUENT DOSE CALCULATIONS

4.1 DOSE RATE

4.1.1 Deleted

As explained in Specification Bases 3.7, Noble Gases are not required to be monitored, and the corresponding dose rate need not be calculated.

4.1.2 Tritium and Radioactive Particulates

There are no short-lived radioactive particulates in the effluent, so radioactive decay can be neglected. Meteorological parameters are assumed to be constant, and applied for the most conservative location. Therefore, the radioactive particulates dose rate calculation methodology is the same as the radioactive particulates dose calculation methodology. Refer to sections 4.3.3 through 4.3.8 for the appropriate equations.

As explained in Specification Bases 3.5, Tritium is not required to be monitored, and the corresponding dose rate need not be calculated. Nevertheless, if such a calculation is required, refer to sections 4.3.9 through 4.3.13 for the appropriate equations.

4.2 Deleted

4.3 DOSE - TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

4.3.1 Calendar Quarter

The methodology for calendar quarter calculations is the same as for the calendar year calculations provided by section 4.3.3, and discussed in section 4.3.2, with the exception that the resulting values for D (annual dose commitment, mrem/year) must be divided by 4 to convert them to quarterly dose commitment, mrem/quarter.

4.3.2 Calendar Year

The methods for calculating the dose due to 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.

The equations provided for determining the doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

4.3.3 Particulate Organ Dose Calculation Summation Methodology

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.

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. Radioactive decay may result in the dose from Transuranic radionuclides becoming significant. If Gross Alpha radioactivity is determined to be released, Pu-241 will be considered to be present at 3.25 times the amount of detected Gross Alpha radioactivity. The annual dose commitment will be calculated for any organ of an individual age group as follows:

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

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 Decommissioning releases range from about 0.002 mrem/year (fruit/vegetable consumption pathway) to less than 1 x 10^{-6} mrem/year (for direct radiation exposure from material deposited on the ground).

4.3.4 Particulate Inhalation Pathway Dose Calculation Methodology

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

where:

χ/Q = The atmospheric dispersion parameter, seconds/cubic meter.
= 1.0 x 10^{-5} seconds/cubic meter for releases from the 50 foot stack. Refer to Appendix B, 1.2.
= 6.59 x 10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

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

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

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. No values are provided for Transuranic radionuclides, as their dose contribution to this pathway is negligible.

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, 1.3.
= 5.39×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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

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

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 x 10^{-8} inverse square meters for releases from the 50 foot stack. Refer Appendix B, 1.3.
= 3.29 x 10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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 x 10^6 seconds.

4.3.7 Particulate Grass-Cow-Meat Pathway Dose Calculation Methodology

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

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.

F_f = 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 x 10^{-8} inverse square meters for releases from the 50 foot stack. Refer to Appendix B, 1.3.
= 3.29 x 10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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 x 10^6 seconds.

4.3.8 Particulate Vegetation Pathway Dose Calculation Methodology

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

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, 1.3.
= 3.29×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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. Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

4.3.10 Tritium Inhalation Pathway Dose Calculation Methodology

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

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, 1.2.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

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.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

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, 1.2.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

4.3.12 Tritium Grass-Cow-Meat Pathway Dose Calculation Methodology

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

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, 1.2.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

4.3.13 Tritium Vegetation Pathway Dose Calculation Methodology

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

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

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, 1.2.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

Table 4-1
Inhalation Dose Factors for Adult Age Group
(mrem/pico-Curie inhaled)
Selected Nuclides from Regulatory Guide 1.109, Table E-7 and from NUREG/CR-4013

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}
Pu-241	3.42×10^{-2}	8.69×10^{-3}	1.29×10^{-3}	5.93×10^{-3}	1.52×10^{-4}	8.65×10^{-7}
Gross α	1.68	1.13	7.75×10^{-2}	5.04×10^{-1}	1.82×10^{-1}	4.84×10^{-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 and from NUREG/CR-4013

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}
Pu-241	3.74×10^{-2}	9.56×10^{-3}	1.40×10^{-3}	6.47×10^{-3}	2.60×10^{-4}	9.17×10^{-7}
Gross α	1.77	1.20	8.05×10^{-2}	5.32×10^{-1}	3.12×10^{-1}	5.13×10^{-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 and from NUREG/CR-4013

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}
Pu-241	7.94×10^{-2}	1.75×10^{-2}	2.93×10^{-3}	1.10×10^{-2}	5.06×10^{-4}	8.90×10^{-7}
Gross α	2.97	1.84	1.28×10^{-1}	7.63×10^{-1}	6.08×10^{-1}	4.98×10^{-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 and from NUREG/CR-4013

Nuclide	Organ					
	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}
Pu-241	8.43×10^{-2}	1.85×10^{-2}	3.11×10^{-3}	1.15×10^{-2}	7.62×10^{-4}	8.97×10^{-7}
Gross α	3.15	1.95	1.34×10^{-1}	7.94×10^{-1}	9.03×10^{-1}	5.02×10^{-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

Nuclide	Total Skin	Total 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}

Values are not provided for Transuranic radionuclides, as their dose contribution to this pathway is negligible.

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 and from NUREG/CR-4013

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}
Pu-241	1.57×10^{-5}	7.45×10^{-7}	3.32×10^{-7}	1.53×10^{-6}	No Data	1.40×10^{-6}
Gross α	7.55×10^{-4}	7.05×10^{-4}	5.41×10^{-5}	4.07×10^{-4}	No Data	7.81×10^{-5}

Table 4-9
Ingestion Dose Factors for Teen Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-12 and from NUREG/CR-4013

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}
Pu-241	1.75×10^{-5}	8.40×10^{-7}	3.69×10^{-7}	1.71×10^{-6}	No Data	1.48×10^{-6}
Gross α	7.98×10^{-4}	7.53×10^{-4}	5.75×10^{-5}	4.31×10^{-4}	No Data	8.28×10^{-5}

Table 4-10
Ingestion Dose Factors for Child Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-13 and from NUREG/CR-4013

	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}
Pu-241	3.87×10^{-5}	1.58×10^{-6}	8.04×10^{-7}	2.96×10^{-6}	No Data	1.44×10^{-6}
Gross α	1.36×10^{-3}	1.17×10^{-3}	1.02×10^{-4}	6.23×10^{-4}	No Data	8.03×10^{-5}

Table 4-11
Ingestion Dose Factors for Infant Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-14 and from NUREG/CR-4013

	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}
Pu-241	4.25×10^{-5}	1.76×10^{-6}	8.82×10^{-7}	3.17×10^{-6}	No Data	1.45×10^{-6}
Gross α	1.46×10^{-3}	1.27×10^{-3}	1.09×10^{-4}	6.55×10^{-4}	No Data	8.10×10^{-5}

Table 4-12
Stable Element Transfer Data For Cow-Milk Pathway
(days/liter)
Selected Nuclides from Regulatory Guide 1.109, Table E-1 and from NUREG/CR-4013

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}
Pu	5.0×10^{-6}
Gross α	5.0×10^{-6}

Table 4-13
Stable Element Transfer Data For Cow-Meat Pathway
(days/kilo-gram)
Selected Nuclides from Regulatory Guide 1.109, Table E-1 and from NUREG/CR-4013

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}
Pu	2.0×10^{-4}
Gross α	2.0×10^{-4}

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. Deleted

b. Modular HEPA Ventilation Particulate releases, using equation (4-3).

c. Deleted. Tritium is no longer a gaseous effluent source term.

d. Liquid releases, No longer applicable.

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. Deleted

b. Modular HEPA Ventilation releases, using equation (4-3). Tritium is no longer a gaseous effluent source term.

c. Liquid releases, No longer applicable.

d. The potential direct radiation exposure to an individual at the site boundary based 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. Deleted

b. Modular HEPA Ventilation releases, using equation (4-3).

c. Liquid releases, No longer applicable.

d. The potential direct radiation exposure to an individual at the site boundary based 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

Deleted - Based on the status of decommissioning, HBPP no longer anticipates wastes exceeding a specific activity that is unacceptable to disposal site without solidification or exceeding Class A as defined in 10 CFR 61.

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

Deleted - HBPP no longer anticipates wastes exceeding a specific activity that is unacceptable to disposal site without solidification or exceeding Class A as defined in 10 CFR 61. HBPP no longer anticipates disposal of wastes requiring stabilization in a High Integrity Container (HIC).

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, filters and other wet wastes shipped for land burial which contain a total specific activity less than the disposal site(s) 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 disposal site(s) (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.

8.2.4 Contract vendor solidification or dewatering services are utilized in accordance with PG&E approved supplier list and procurement procedures.

8.2.5 Vendor services may be conducted off site in accordance with their facility license and procedures. Vendor services include written confirmation of acceptable disposal waste form.

8.2.6 Gross dewatering of resins and filters may be performed onsite to achieve transport requirements in preparation for additional processing to a final waste form by offsite vendor services.

8.2.7 On site activities, such as managing wet soils from decommissioning excavations and process water shall be performed utilizing approved procedures or work instructions to ensure compliance with transportation regulations, disposal facility license requirements and/or waste acceptance criteria.

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 Decommissioning period. All changes shall be reviewed and approved by the HB Director 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.

9.3 HBPP is allowed to modify or reduce environmental requirements in the ODCM provided HBPP considers the modification or reduction from a technical and decommissioning perspective. [CMT 10.1]

10.0 COMMITMENTS

10.1 HBPP does not intend to modify or reduce the environmental monitoring requirements as specified in the ODCM during the period of SAFSTOR and decommissioning activities. This applies to those environmental samples and analysis identified as either quality or non-quality samples. This commitment is to be incorporated into the next revision of the ODCM. NOTE: HBPP is allowed to modify or reduce environmental requirements in the ODCM provided HBPP considers the modification or reduction from a technical and decommissioning perspective.

11.0 RESPONSIBLE ORGANIZATION

Radiation Protection Manager

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. As of December 31, 2013, HBPP ceased processed liquid effluent to the discharge canal and processed liquid effluent will be transported for disposal at a regulated disposal site. Storm water and groundwater associated with excavations and groundwater inleakage to structures during decommissioning will typically be treated and released using the Ground Water Treatment System. The GWTS is an Active Treatment System (ATS) is designed to remove suspended solids in order to meet release criteria of the SWPP. The system will be limited to treating water containing soluble radionuclides less than 10 times the "new" 10 CFR 20, Appendix B, Table 2, Column 2 effluent concentration limits (ECLs) in order to ensure concentrations at the Site Boundary are maintained less than limiting condition 2.3.1.

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	8.60E-2	7.17E-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

Table A-3 below reflects the Gaseous Effluent Activity as a representation of the state of decommissioning during the calendar year 2013 relative to the Baseline above.

Table A-3
2013 Gaseous Effluent Activity

Type of Activity	Annual Release (Curies)	Monthly Average Release (Curies)
Particulate Gamma Emitters (total)	<1.5E-5	<1.3E-6
Strontium-90	<1E-6	<1E-7
Particulate Alpha Emitters (total)	<1E-6	<1E-7

Table A-3 data is summarized from the 2013 Annual Effluent Release Report and are listed as less than values because sampling results were the composite of LLD values. Tritium is no longer monitored due to a lack of significant source term.

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:

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

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

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

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 = 4.70×10^{-3} seconds/meter^3 (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 (1.2×10^{-4} meter^{-1})/ (6 x 6.07 meter) = 3.29×10^{-6} meter^{-2}.

APPENDIX C

Deleted

PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
HUMBOLDT BAY POWER PLANT

SAFSTOR/Decommissioning Offsite Dose Calculation Manual
Revision 30

Summary of Changes Included in Revision 30 of the SAFSTOR/Decommissioning Offsite Dose Calculation Manual

Summary of Changes:

Page / Section	Change Date	Change	Reason
Page I-22 Figure 2-1	Rev. 30	Locations T-5, T-6, T-7, T-8, and T-11 were moved during 2019.	Thermoluminescent dosimeter locations reflect changes in the perimeter fencing and areas that are no longer controlled to prevent pubic access.
Page I-18 Table 2-7	Rev. 30	Onsite Airborne Monitoring Locations reduced from 4 to 3	There is no source of airborne contamination, no perceived receptor or routinely exposed individual in this area, collecting air samples no longer provides useful information.
Page A-2 Appendix A, 1.1	Rev. 30	Removed the discussion about the Ground Water Treatment System. Added: The Ground Water Treatment System (GWTS) was removed from service in April 2019.	Storm water monitoring is provided through the Storm Water Pollution Prevention Plan and installed engineered features as a part of the final site restoration for that purpose.

(Procedure Classification - Quality Related)

INTRODUCTION

The SAFSTOR/DECOMMISSIONING 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 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 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 requirements for the REMP. 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. As of December 31, 2013, HBPP ceased liquid radioactive effluent discharges via the discharge canal to Humboldt Bay. The scope of the REMP and interlaboratory comparison program are the dosimeters and air samples required to evaluate the direct radiation and gaseous effluents from HBPP.

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 decommissioning period. Part II of the ODCM contains the calculational methods for gaseous and liquid effluents to preserve site specific data although the gaseous effluent pathway is limited to Modular HEPA Units on a selected basis and the liquid discharge pathway has been terminated.

The ODCM also contains the Process Control Program (PCP) for solid radioactive wastes, administrative controls regarding the content of the Annual Radiological Environmental Monitoring Program Report, administrative controls regarding the content of the Annual Radioactive Effluent Release Report, and administrative controls regarding major changes to radioactive waste treatment systems.

The ODCM shall become effective after approval by the HB Director. 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 required level of radioactive effluent control and not adversely impact the accuracy or reliability of effluent or dose 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 Deleted

1.4 Deleted

1.5 Deleted

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.

1.7 Deleted

1.8 FREQUENCY NOTATION

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

1.9 Deleted

1.10 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.

1.11 INSTANTANEOUS CONCENTRATION

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

1.12 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.

1.13 MODULAR HEPA VENTILATION UNIT

MODULAR HEPA VENTILATION UNIT consists of HEPA filter trains discharged to the environment and sampled in accordance with ANSI/HPS N13.1-1999.

1.14 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, 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 Monitoring Report and the Annual Radioactive Effluent Release Report. The ODCM also contains the Process Control Program (PCP) for solid radioactive wastes.

1.15 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).

1.16 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, disposal site(s) requirements, and other requirements governing the disposal of solid radioactive waste.

1.17 Deleted

1.18 RESTRICTED AREA

The RESTRICTED AREA is defined by 10CFR20.1003. The physical location(s) of the RESTRICTED AREA shall be defined in plant procedures.

1.19 SITE BOUNDARY

The SITE BOUNDARY shall be the boundary of the UNRESTRICTED AREA used in the offsite dose calculations for gaseous and liquid effluents. The SITE BOUNDARY is shown in Figure 1-1. Ingress and egress through the SITE BOUNDARY are controlled by the Company.

1.20 Deleted

1.21 Deleted

1.22 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.

1.23 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.”

1.24 VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing particulates from the gaseous exhaust stream prior to release to the environment.

1.25 Deleted

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.

Figure 1-1
SITE BOUNDARY

2.0 SPECIFICATIONS

2.1 **Deleted; Table 2-1 - Deleted; Table 2.2 - Deleted**

2.2 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION[1]

LIMITING CONDITIONS

2.2.1 Deleted - plant stack is no longer in operation.

SURVEILLANCE REQUIREMENTS

2.2.2 Deleted

Table 2-3 - Deleted
Table 2-4 - Deleted

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

Deleted (See BASES Section 3.2 and Appendix A)

Table 2-5 (Deleted)

2.4 LIQUID EFFLUENT – DOSE Deleted - No longer applicable

2.5 Deleted - No longer applicable

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

SURVEILLANCE REQUIREMENTS

2.6.2 Deleted (see BASES section 3.5)

2.6.3 Deleted (see BASES section 3.5)

2.6.4 Radioactive particulates, with half-lives of greater than 8 days, in gaseous effluents released to the environment shall be sampled and analyzed in accordance with the sampling and analysis program of Table 2-6, and their concentrations shall be compared with the limits of 10CFR20, Appendix B, Table 2, Column 1. IF their concentrations exceed those limits, the calculational methods in Part II of the ODCM shall be used to determine whether or not the limits of Specification 2.6.1 have been exceeded. The actual sample period shall be used to determine the dose rate during the sample period.

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]
Modular HEPA Ventilation Discharge				
	Continuous[b,d]	W[b] Mixing Box Particulate Sample	Principal Gamma Emitters[e]	1×10^{-11}
	Continuous[b,d]	W[b] Mixing Box Particulate Sample	Gross Alpha	1×10^{-12}
	Continuous[b,d]	W[b] Mixing Box Particulate Sample	Gross Beta	6.7×10^{-12}
	Continuous[b,d]	Q Composite of Mixing Box Particulate Samples	Sr-90[g]	1×10^{-11}
	Continuous[b,d,h]	Q Composite of Mixing Box Particulate Samples	Am-241	1×10^{-12}
	Continuous[b,d,i]	Q Composite of Mixing Box Particulate Samples	Am-241	1×10^{-14}

Table 2-6 (Continued)

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),

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. NOTE: The LLDs are achievable with a reasonable count time assuming adequate effluent volume and sample volume. If the LLD is not achieved, initiate a condition report to document that the LLD was not achieved and indicate a probable cause (short runtime, equipment malfunction, etc.). RP Supervision will determine if additional calculations should be performed per Surveillance 2.6.4.

Table 2-6 (Continued)

Table Notation (Continued)

[b] Samples shall be changed at least once per 7 days (3 day extension permitted), assuming effluent pathway is in continuous use (typically > 40 hrs per week). Samples may be collected more frequently for short duration use of a Modular HEPA Ventilation Unit.

[c] Deleted

[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, and 2.8.

[e] The principal gamma emitters for which the LLD specification applies exclusively are 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.

[f] Deleted based on SPAMS no longer in service.

[g] Analysis specific to Sr-90 may be replaced by analysis for total radioactive Strontium.

[h] When release volume is less than or equal to 3.26×10^{11} ml (e.g., 1.15E+7 cubic feet).

[i] When release volume exceeds 3.26×10^{11} ml (e.g., 1.15E+7 cubic feet).

2.7 Deleted

2.8 GASEOUS EFFLUENTS: DOSE - RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITIONS

2.8.1 The dose to a MEMBER OF THE PUBLIC from the release of 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.

<u>APPLICABILITY</u>: At all times.

<u>ACTION</u>:

With the calculated dose from the release of 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 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 dose calculation for the current calendar quarter and the current calendar year, for the release of radioactive materials in particulate form with half-lives greater than 8 days,
OR

Perform a BASELINE COMPARISON for gaseous effluent radioactivity (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.
OR

Perform a dose assessment, if weekly sampling indicates the effluent from modular HEPA units exceed 0.1 uCi of alpha emitters or Sr-90. The assessment of alpha and beta may be performed with appropriate compensation for naturally occurring nuclides.

As explained in Specification Bases section 3.8, neither routine surveillance nor dose calculations are required for Tritium in gaseous effluents.

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 disposal site(s) requirements.

APPLICABILITY: At all times.

ACTION:

With the provisions of the 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) meet the shipping, disposal site(s) requirements with regard to dewatering and off site vendor processes.

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 gaseous effluents exceeding twice the limits of Specification 2.8.1.a, or 2.8.1.b, calculations should be made, which include direct radiation contributions from Unit No. 3, 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 gaseous effluents shall be calculated in accordance with dose calculation methodology provided for Specification 2.8.1.

2.11 REMP MONITORING PROGRAM

LIMITING CONDITIONS

2.11.1 A radiological environmental monitoring program shall be provided to monitor the radiation and radionuclides in the environs of the facility. The 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 Program Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. A Special Report pursuant to Administrative Control 4.3, 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 Specification 2.8. Prepare and submit to the Commission within 30 days of obtaining analytical results from the affected sampling period which includes an evaluation of release conditions, environmental factors or other aspects which caused the dose limits to be exceeded. 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 Program Report.

SURVEILLANCE REQUIREMENTS

2.11.2 The radiological environmental monitoring samples shall be collected pursuant to Table 2-7 from the “Quality Related” locations given in Tables 2-7 and 2-10 and Figures, 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]	Sampling and Collection Frequency	Type of Analysis	PROGRAM BASIS ODCM Specs (QR)
AIRBORNE	3 onsite locations, 1 offsite location	Continuous sampler operation with sample collection at least once per 7 days[(1)(c)]	Gross alpha and gross beta radioactivity following filter change Gamma isotopic[(b)] analysis on quarterly composite (by station)	X
DIRECT RADIATION	Minimum of 8 onsite stations, at or within the SITE BOUNDARY fence line, 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
WATERBORNE	None	N/A	N/A	
INGESTION	None	N/A	N/A	
TERRESTRIAL	None	N/A	N/A	

08/19

Table Notations

QR - Quality Related (1)Performed by HBPP (3)Performed by a NVLAP accredited processor

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

(c) Continuous sampler operation may be limited to normal work hours to represent effluents from decommissioning activities. Count times may need to be adjusted to achieve the recommended LLDs in Table 2-9.

Table 2-8 (Deleted)

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

Analysis	Airborne Particulate (pCi/m^3)
Gross Beta	0.01
H-3	
Co-60	
Cs-137	0.06

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 Monitoring Program 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:

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

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.

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 Radiological Environmental Monitoring Program 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.

Table 2-10
DISTANCES AND DIRECTIONS TO ENVIRONMENTAL MONITORING STATIONS

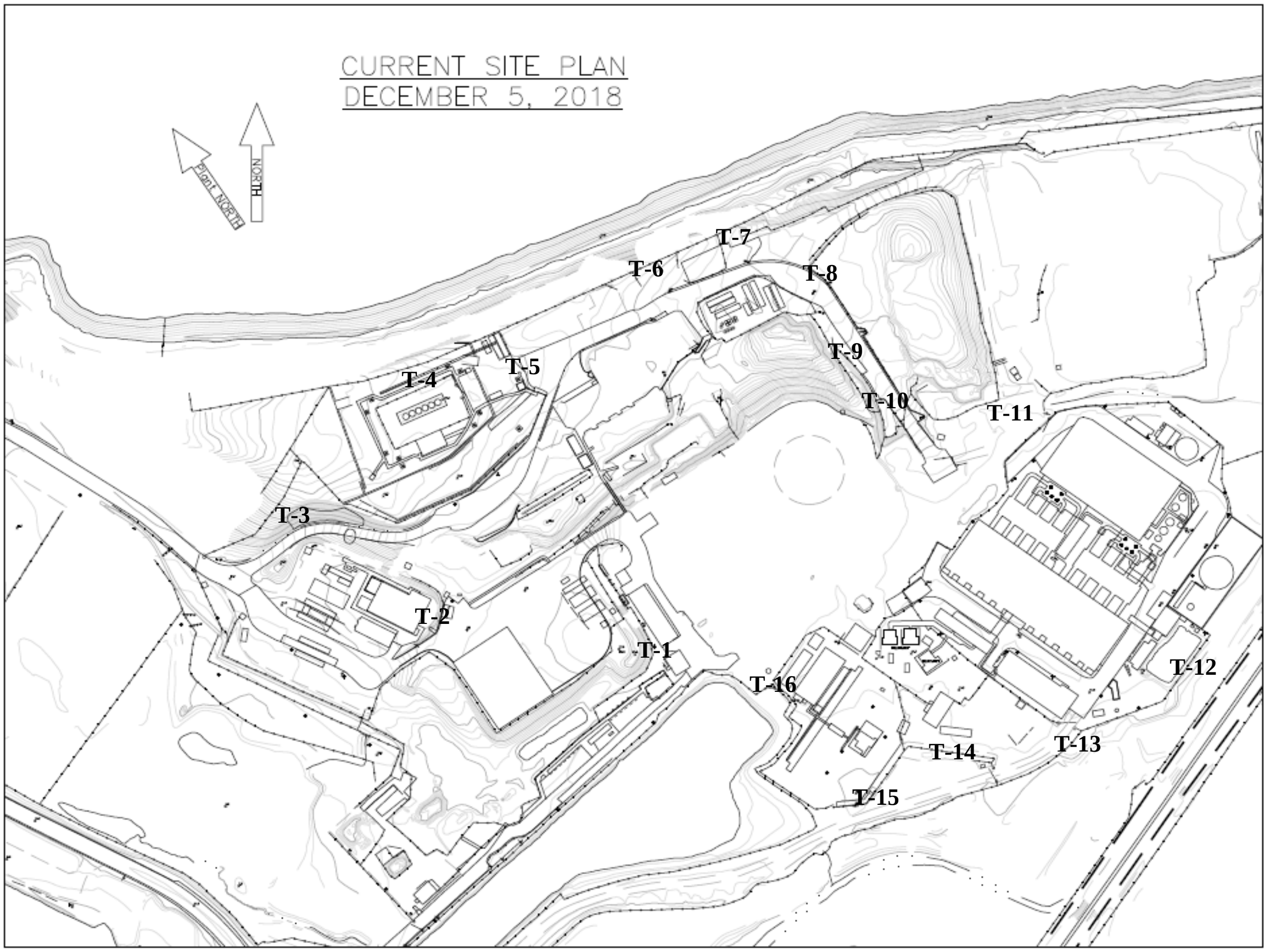
Station No.	Code	Station Name	Radial Direction Sector	By Degrees	Radial Distance from Plant (Miles)
1	**Δ**	**King Salmon Picnic Area**	**W**	**270**	**0.3**
2	**Δ**	**180 Dinsmore Drive, Fortuna**	**SSE**	**158**	**9.4**
3	☐	Humboldt Hill Road at Bret Harte Lane	SSE	158	0.9
14	**Δ**	**South Bay School Parking Lot**	**S**	**180**	**0.4**
17	**Δ**	**Control Set at Humboldt Substation, Eureka**	**NEE**	**61**	**5.8**
25	**Δ**	**Irving Drive, Humboldt Hill**	**SSE**	**175**	**1.3**

Table Notations

Code: Δ Dosimetry Station ☐ Air Particulate Station

Figure 2-1
HBPP Onsite TLD Locations

08/19

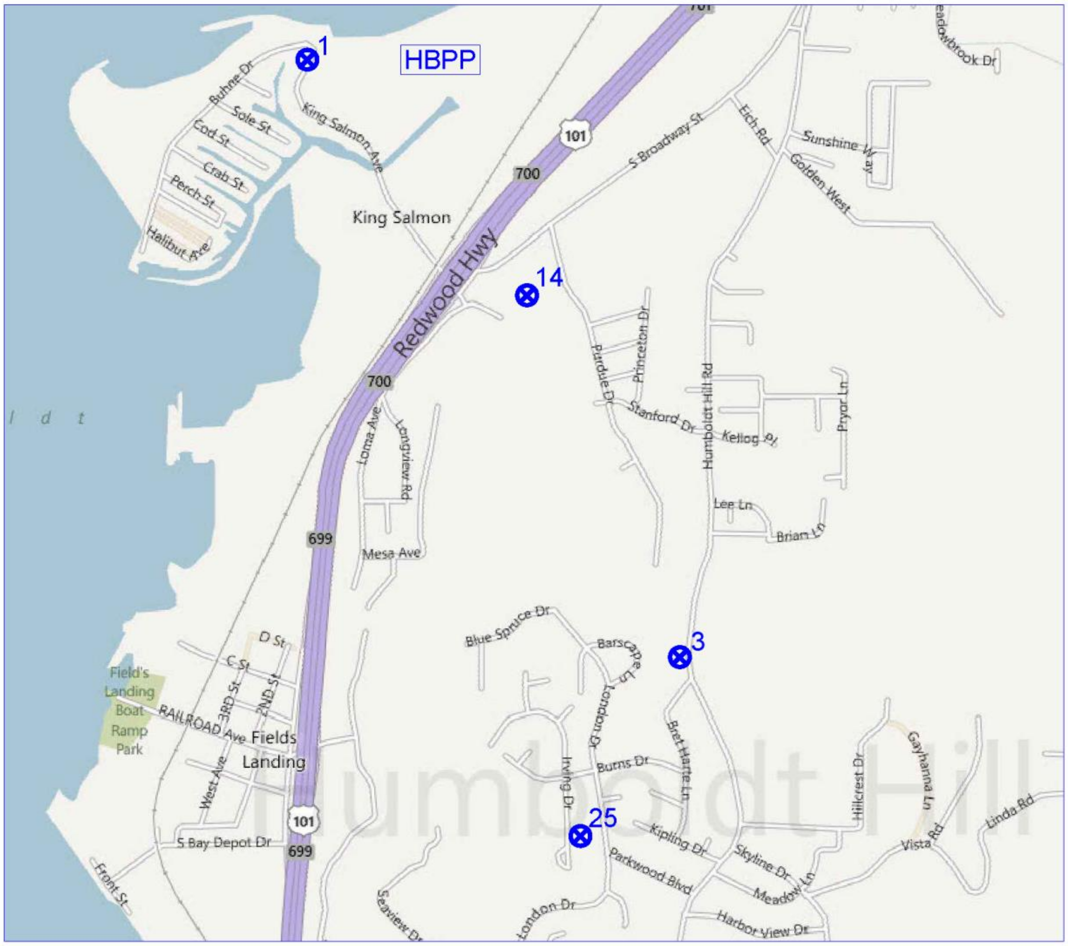


08/19

Monitoring locations T7, T10, T11, T13, T16, T2, T3, and T5 generally represent REMP Site Boundary direct exposure monitoring locations in the 8 primary compass points beginning with T-7 to representing north and moving clockwise.

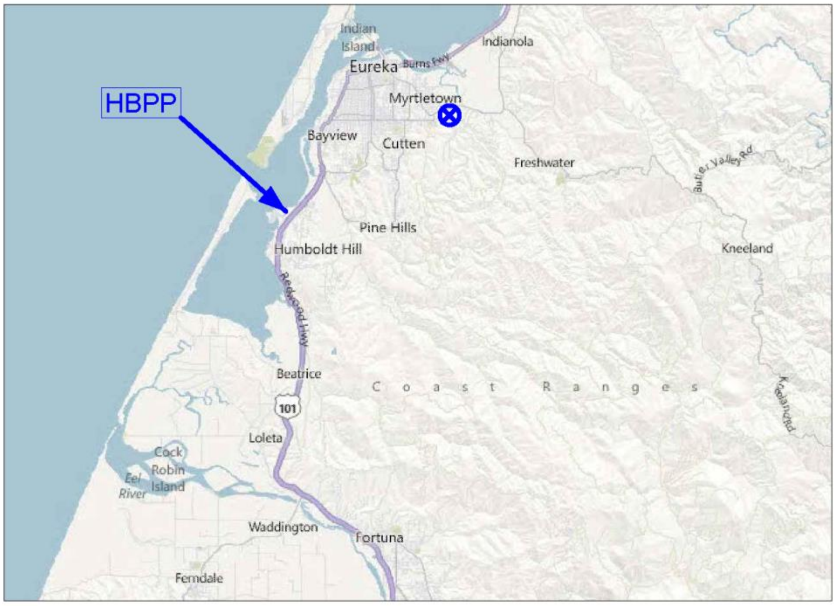
Figure 2-2 - Deleted

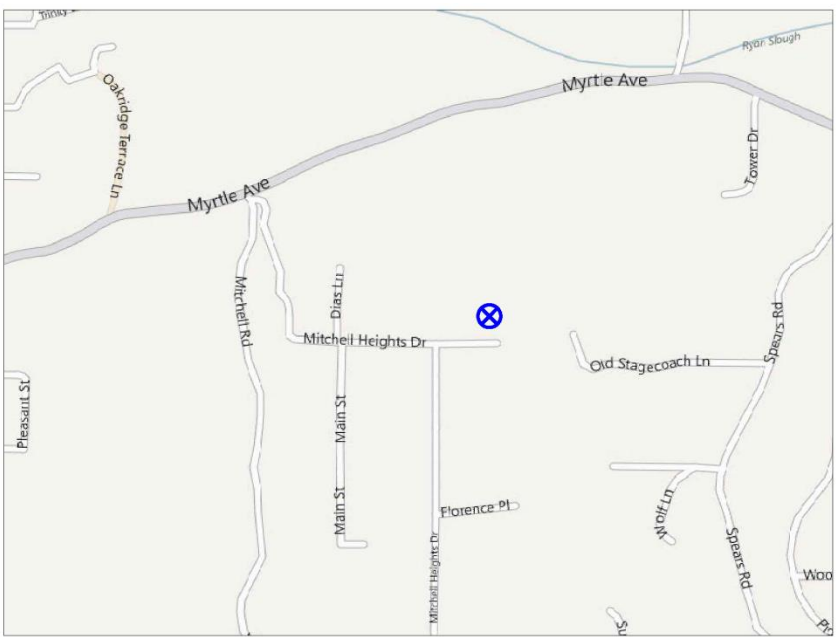
Figure 2-3
HBPP OFFSITE SAMPLING LOCATIONS - HUMBOLDT HILL



Station	GPS Coordinates (NAD83/NAVD88 CA. Zone 1)			Decimal Degrees	
	Easting	Northing	el.	Latitude	Longitude
1	5948026.52	2161183.79	11.38	40.74156	-124.21903
3	5951260.28	2155706.11	234.94	40.72676	-124.20274
14	5949876.83	2158864.39	18.65	40.73533	-124.20802
25	5950247.30	2154214.18	229.22	40.72260	-124.20626

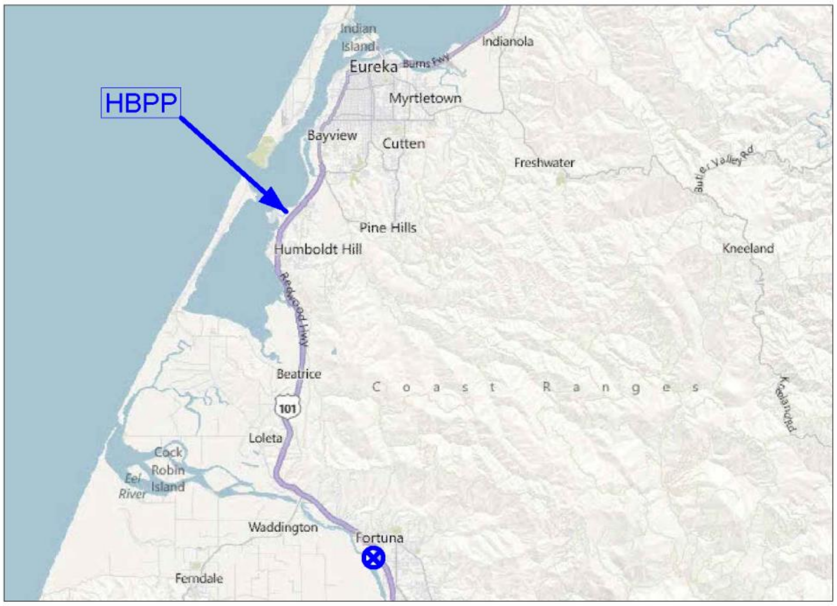
Figure 2-4
HBPP OFFSITE SAMPLING LOCATIONS - EUREKA

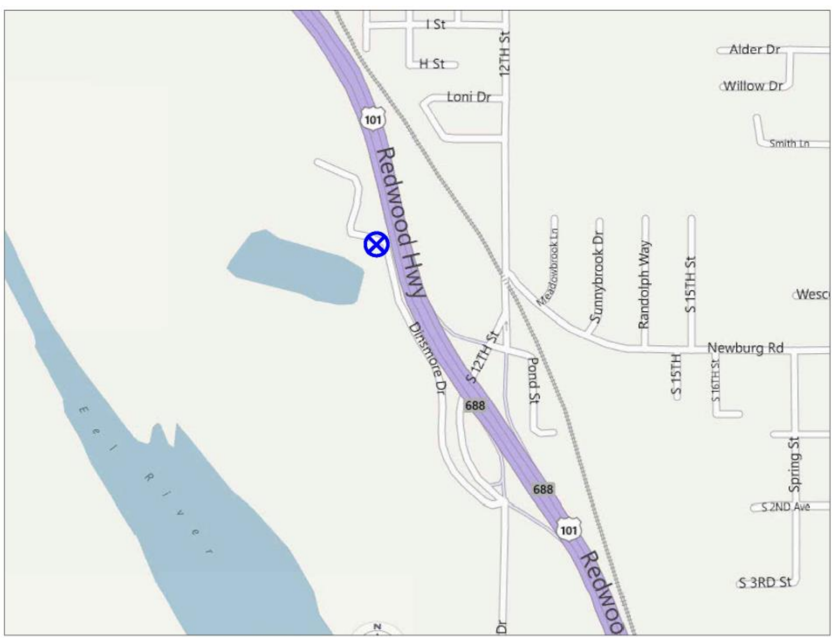




Station	GPS Coordinates (NAD83/NAVD88 CA. Zone 1)			Decimal Degrees	
	Easting	Northing	el.	Latitude	Longitude
17	5976549.55	2175490.19	164.85	40.78276	-124.11324

Figure 2-5
HBPP OFFSITE SAMPLING LOCATIONS - FORTUNA





Station	GPS Coordinates (NAD83/NAVD88 CA. Zone 1)			Decimal Degrees	
	Easting	Northing	el.	Latitude	Longitude
2	5962583.86	2105797.82	35.53	40.59057	-124.15746

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 Program Report.

SURVEILLANCE REQUIREMENTS

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

2.13 RADIOACTIVE WASTE INVENTORY

LIMITING CONDITIONS

2.13.1 Liquid Radioactive Waste In Outdoor Tanks

The radiological inventory of wastes in outdoor tanks that are not capable of retaining or treating tank overflows shall not exceed 0.25 Ci.

APPLICABILITY: At all times.

ACTION:

When the inventory exceeds the conditions as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Monitoring Program Report.

2.13.2 Deleted

SURVEILLANCE REQUIREMENTS

2.13.3 An inventory of the estimated liquid radioactive waste in outdoor tanks inventory shall be maintained to verify the 0.25 Ci limit is not exceeded.

OR

Provide overflow protection.

OR

Use process knowledge of typical concentration and tank volume to verify that the 0.25 Ci is not exceeded.

3.0 SPECIFICATION BASES

3.1 Radioactive Gaseous Effluent Monitoring Instrumentation Basis

Deleted – The plant stack ceased operation in 2015. Monitoring gaseous effluent is limited to sampling and analysis of Modular HEPA Units.

3.2 Liquid Effluent Concentration Basis

Deleted - Liquid effluents are no longer discharged to Humboldt Bay. Effective December 31, 2013, discharge of processed radioactive liquid effluents to Humboldt Bay was terminated. Any remaining or incidental radioactive liquid in concentrations exceeding 10 times 10 CFR 20, Appendix B, Table 2 Column 2 are manifested for disposal at a licensed disposal site. Sampling and manifesting requirements are consistent with the requirements of the receiving facility not subject to ODCM methodology.

3.3 Liquid Effluent Dose Basis

Deleted - Liquid effluents are no longer discharged to Humboldt Bay.

3.4 Liquid Effluent Treatment Basis

Deleted - Liquid effluents are no longer discharged to Humboldt Bay.

3.5 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).

Stack operation and monitoring ceased operation in 2015, so the reporting period for 2015 includes the dose contribution from the plant stack prior to ceasing operation. Modular HEPA Ventilation Units continue to be sampled as a gaseous effluent pathway.

Noble gas monitoring is not required because the spent fuel (noble gas source term) has been transferred to the ISFSI. Tritium monitoring is not required in gaseous effluents because the tritium source term was the spent fuel pool water which is now empty. Residual water in various plant drains and sumps contain low levels of tritium (generally at or below the drinking water standard (2E-5 uCi/ml or 20,000 pCi/L) and does not require monitoring as a gaseous plant effluent.

3.6 Deleted

Gaseous effluent monitoring is not required for noble gases because the spent fuel (noble gas source term) has been transferred to the ISFSI.

3.7 Deleted

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 basis for the dose calculation threshold of 0.1 uCi alpha emission or Sr-90 in a week assumes a continuous ground level release of 1.65E-13 uCi/sec and an X/Q of 6.59E-3 sec/m^3. The limiting inhalation dose is to a teen age member of the public at the site boundary at approximately 0.3 mrem/wk (15 mrem/yr) to the bone from alpha emitters. Compliance with this Specification has been established on a licensing basis by the SAFSTOR Environmental Report and NUREG-1166, “Final Environmental Statement for Decommissioning Humboldt Bay Power Plant.” These reports have demonstrated that routine release of Tritium and radioactive materials in particulate form (with half-lives greater than 8 days) in gaseous effluents during decommissioning will not cause the Specification to be exceeded. As long as routine releases do not exceed the baseline quantities evaluated in these reports, no further dose calculation is necessary.

The previously evaluated tritium source term was the spent fuel pool water, which is now empty. Residual water in various plant drains and sumps contain low levels of tritium (at or below the drinking water standard (2E-5 uCi/ml or 20,000 pCi/L) and does not require monitoring as a gaseous plant effluent.

3.9 Solid Radioactive Waste Basis

This Specification ensures that radioactive wastes that are transported from the site shall meet the disposal site(s) licensee and/or waste acceptance criteria for free standing liquids 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 is required to provide assurance that the baseline conditions established by the Environmental Report are not deteriorating and it supplements the SAFSTOR Environmental Report baseline

environmental conditions by conducting onsite and offsite environmental monitoring to evaluate routine conditions during decommissioning and to document any increased nuclide concentrations and/or radiation levels resulting from accidents during decommissioning.

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.

The LLD's required by Table 2-9 are considered optimum for routine environmental measurements in industrial laboratories. HBPP no longer includes water, milk, fish, food products, or sediment in its routine REMP sampling program. Sampling and analysis in support of the License Termination Plan is independent of the ODCM requirements.

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.

3.13 Radioactive Waste Inventory Basis

The requirements for limits on the accumulation of liquid radioactive waste in outdoor tanks were transferred from the license Technical Specifications.

4.0 ADMINISTRATIVE CONTROLS

4.1 Annual Radiological Environmental Monitoring Report

A report on the Decommissioning Radiological Environmental Monitoring Program shall be prepared annually in accordance with the NRC Branch Technical Position and submitted to the NRC by May 1 of each year.

The Annual Radiological Environmental Monitoring 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. The material provided shall be consistent with the objectives outlined in the ODCM, and in 10CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

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 Report Annual Summary, or equivalent. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in the next annual report.

d. A summary description of the Decommissioning 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. Deleted – water samples are not collected as a part of the REMP.

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

Table 4-1
RADIOLOGICAL ENVIRONMENTAL MONITORING REPORT ANNUAL 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							
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	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
Groundwater	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
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 REPORT ANNUAL 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 as 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].

Not Required - not required by the HBPP Offsite Dose Calculation Manual. 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. 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. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10CFR 50.36a and 10CFR Part 50, Appendix I, Section IV.B.I. Beginning in the reporting year 2014, liquid effluents shipped for processing or disposal at a regulated disposal site are included in the annual report.

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 gaseous monitoring instrumentation inoperable for greater than 30 days in accordance with Specification 2.2. Beginning the reporting year 2015, following cessation of the plant stack operation, the effluent monitoring instrumentation associated with Specification 2.2 ceased operation. Inoperability and lack of timely corrective action is only applicable to the period of plant stack operation. Anomalies associated with monitoring effluent from Modular HEPA Ventilation systems will be reported.

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 2.8 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 approved by the HB Director.

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.

4.5 Process Control Program Changes

a. Changes to the Process Control Program (PCP) shall be documented and records of reviews performed shall be retained as required for the duration of Decommissioning.

b. The following information shall be available for review:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

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

c. The change shall become effective after approval of the HB Director.

PART II - CALCULATIONAL METHODS AND PARAMETERS

1.0 UNRESTRICTED AREA EFFLUENT CONCENTRATIONS

1.1 LIQUID EFFLUENT UNRESTRICTED AREA CONCENTRATIONS

Specification 2.3.1 requires that the Radioactive Liquid Effluent Sample concentrations (RLES) are calculated 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). This requirement is defined by the following relationship.

$$\sum_i \frac{C_{i,\,Canal}}{10 \times ECL_i} \leq 1 \qquad (1\text{-}1)$$

where:

$C_{i\text{-}Canal}$ = The concentration of isotope “ i “ in the canal discharge point to Humboldt Bay.

ECL_i = Effluent Concentration Limit for radionuclide “ i “ from 10 CFR 20, Appendix B, Table 2, Column, 2 (µCi/ml)

1.1.1 If the outfall location is not at the furthermost portion of the canal from the entrance to the Bay the concentration of the isotope $C_{i\text{-}Canal}$ is equal to the concentration being discharged at the outfall.

1.2 UNRESTRICTED AREA GASEOUS EFFLUENT CONCENTRATIONS

1.2.1 Equation C-4 of Regulatory Guide 1.109 demonstrates how to calculate dose from inhalation:

The annual dose associated with inhalation of all radionuclides, to organ j *of an individual in age group* a*, is then:*

$$D_{ja}(r,\theta) = R_a \Sigma x_i(r,\theta) DFA_{ija}$$

where

D_{ja} *is the annual dose rate to organ* j *of an individual in age group* a

R_a *is the breathing rate for age group* a

$x_i(r,\theta)$ *is the* annual *average ground-level concentration of nuclide* i *in air in sector* θ *at distance* r*, in* pCi/m^3

DFA_{ija} *is the dose factor for nuclide* i *to organ* j *of age group* a

To calculate $x_i(r,\theta)$ the annual average ground-level concentration of nuclide i in air in sector θ at distance r, in pCi/m^3 the equation must be rearranged to:

$$D_{ja}(r,\theta)/(\text{DFAija } R_a) = x_i(r,\theta)$$

Assuming that:

Americium-241 is the primary nuclide

The maximally exposed group is the Teen based on breathing rates and DFAija

The DFAija to the bone of a Teen from Am-241 is 1.77 mrem/pCi

The DFAija are taken from: NRC NUREG/CR-4013, "*LADTAP-II Technical Reference and User Guide*"

The Teen breathing rate is 8000 m^3/year

Therefore the ground-level concentration of Am-241 in air in sector θ at distance r, in pCi/m^3 that will produce a dose rate of 1500 mrem/year to the bone of a Teen is:

(1500 mrem/year) / (1.77 mrem/pCi) / (8000 m^3/year) = 1.06E-1 pCi/ m^3

1.06E-1pCi/ m^3 =

(1.06E-1 pCi/m^3) / (1E6 pCi /µCi) / (1E6 ml/m^3) = 1.06E-13 µCi/ml

1.2.2 Quantity of radioactive material released

Equation C-3 of Regulatory Guide 1.109 demonstrates how to calculate the quantity of material that must be released to produce a given airborne concentration:

The annual average airborne concentration of radionuclide i at the location (r, θ) with respect to the release point may be determined as

$$x_i(r,\theta) = 3.17 \times 10^4\, Q_i(\chi/Q)^D(r,\theta)$$

where

$x_i(r,\theta)$ *is the annual average ground-level concentration of nuclide* i *in air in sector* θ *at distance* r, *in* pCi/m^3

3.17×10^4 *is the number of pCi/Ci divided by the number of sec/yr*

$(\chi/Q)^D(r,\theta)$ *is the annual average atmosphere dispersion factor, in sec/m^3.*

Q_i *is the release rate of nuclide I to the atmosphere, in Ci/yr*

A value of 6.59E-3 sec/m^3 was used for the incidental release path atmosphere dispersion factor at the site boundary $(\chi/Q)^D(r,\theta)$ for releases from Modular HEPA Units. This is based on a release rate of 2000 cfm. (Ref: Safstor ODCM, Appendix B, 2.0) 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*.

To determine the release rate that will result in an average ground-level concentration the above equation must be rearranged to:

$$Q_i = x_i(r,\theta) / (3.17 \times 10^4(\chi/Q)^D(r,\theta))$$

Therefore the Modular HEPA Unit release rate of Am-241 required to equal the incidental ground-level concentration at the site boundary calculated above is:

1.06E-1 pCi/m^3 / ((3.17E4 (pCi/Ci)/ (sec/yr)) * (6.59E-3 sec/m^3)) =

5.07E-4 Ci/yr or 5.07E2 uCi/yr

1.2.3 Transmission Fraction

Deleted – no on line monitoring provided.

1.2.4 Effluent Concentration

The Modular HEPA Unit concentration that would result in a release rate of 5.07E-4 Ci/yr is equal to:

Total release (Curies/year) / Release rate (cc/year)

The average annual Modular HEPA Unit flow rate is 2,000 cfm

This results in a total volume of 2.98E13 cc/yr

This is based on (2000 ft^3/min * 525,600 minutes/yr * 28,317 cc/ft^3).

(5.07E-4 Ci * 1E6 µCi/Ci) / (2.98E13 cc/yr) = 1.70E-11 µCi/cc

Therefore an indicated Modular HEPA concentration of 1.70E-11 µCi/cc at 2000 cfm for one calendar year would result in a dose of 1500 mrem to a member of the public at the site boundary.

Two times the indicated release rate is equal to3.4E-11 µCi/cc.
Two hundred times the indicated release rate is equal to 3.4E-9 µCi/cc.

1.2.5 Relationship to EPA PAG

To compare the release rates calculated above the following assumptions were made:

Am-241 dose conversion factor in rem / cm^{-3} µCi hr, from EPA 400 = 5.3E8

Since no credit is taken for an elevated release point or an annual average χ/Q the same atmospheric dispersion factor is used in the calculations below.

Assuming that an unplanned release occurs at two times the ODCM release rate for one hour the total activity released is equal to:

3.4E-11 µCi/cc * 2000 ft^3/min * 28,317 cc/ft^3 * 60 min = 1.16E-1 µCi

(1.16E-1 µCi) * (5.3E8 rem / cm^{-3} uCi hr) * (6.59E-3 sec/m^3) / (1E6 cm^3/m^3) / (3600 sec/hour) = 1.13E-4 rem

This is much less than the EPA PAG of 1 Rem

Assuming that an unplanned release occurs at two hundred times the ODCM release rate for 15 minutes the total activity released is equal to:

3.4E-9 µCi/cc * 2000 ft^3/min * 28,317 cc/ft^3 * 15 min = 2.89E0 µCi

This results in a dose of:

(2.89E0 µCi) * (5.3E8 rem / cm^{-3} uCi hr) * (6.59E-3 sec/m^3) / (1E6 cm^3/m^3) / (3600 sec/hour) =

2.80E-3 rem

This is much less than the EPA PAG of 1 Rem.

1.2.6 Relationship to 10CFR20 Appendix B Table 2 Effluent Concentration limits

The 10CFR20 Appendix B Table 2 Effluent Concentration limit for Am-241 is 2E-14 µCi/ml.

The average annual ground-level concentration in air (x_i) in pCi/m^3 is equal to:

x_i = (3.17E4 (pCi/Ci)/ (sec/year)) * Q * (X/Q)

Where Q is equal to the quantity of radioactive material released in a year in Curies/year

ODCM Modular HEPA Unit incidental release X/Q = 6.59E-3 sec/ m^3

If x_i = 2E-14 µCi/ml then:

Q = (2E-14 µCi/ml * 1E6 ml/m^3 * 1E6 pCi/µCi) / ((3.17E4 (pCi/Ci)/ (sec/yr)*(6.59E-3 sec/ m^3))

Q = 9.57E-5 Ci/yr

The average annual Modular HEPA Unit volume based on the ODCM is 2.98E13 cc/yr.

This is based on (2000 cfm * 525,600 minutes/yr * 28,317 cc/cfm).

Therefore, the Modular HEPA Unit effluent concentration required to result in a fence-line concentration of 2E-14 µCi/ml is:

(9.57E-5 Ci/yr * 1E6 µCi/Ci) / (2.98E13 cc/yr * 1 cc/ml) = 3.2E-12 µCi/ml

1.2.7 Conversion Factor from Effluent Concentration to µCi/day

The release rate in µCi/day = Modular HEPA Unit concentration in µCi/cc * 2000 ft^3/min * 1440 minutes/day * 28317 cc/ ft^3

The release rate in µCi/day = Modular HEPA Unit concentration in µCi/cc * 8.16E10 cc/day

1.2.8 Conversion Factor from µCi/day to % of NUE
An NUE is equal to a release rate of 3000 mrem/year

%NUE = (Offsite dose rate / NUE threshold) * 100

%NUE = ((Conversion Factor * Release Rate) / NUE threshold) * 100

%NUE = ((Conversion Factor * 100) / NUE threshold) * Release Rate

The Conversion Factor is equal to (1.77E6 mrem/µCi) * (6.59E-3 sec/ m^3) * (8000 m^3/year) / (8.64E4 sec/day)

This is equal to1.08E3 mrem/year per µCi/day

1.2.9 Results

The 10CFR20 Appendix B Table 2 Effluent Concentration limit for Am-241 is 2E-14 µCi/ml. The Modular HEPA Unit effluent concentration that would result in a fence-line concentration of 2E-14 µCi/ml is 3.2E-12 µCi/ml.

3.2E-12 uCi/ml * 8.16E10 cc/day * 1ml/cc * 1.08E3 mrem-day/uCi-yr = 4.70E2 mrem/yr.
470 mrem/yr / 8760 hr/yr = 5.365E-2 mrem/hr

Assuming that an unplanned release occurs at two times the ODCM release rate for one hour the offsite dose corresponding to an NUE would be 1.07E-4 rem (0.107 mrem) which is much less than the EPA PAG.

Assuming that an unplanned release occurs at two hundred times the ODCM release rate for fifteen minutes the offsite dose corresponding to an Alert would be 2.675E-3 rem (2.7 mrem) which is much less than the EPA PAG.

Note that Am-241 is used in the example calculations and is expected to be limiting. Other alpha emitting isotopes such as Pu-238, Pu-239/240 and Cm-243/244 are evident in the contamination at HBPP. Since the Effluent Concentration Limits (ECLs), Derived Air Concentration (DAC) values and organ Dose Conversion Factors (DCFs) are similar, the Am-241 values may be assumed to be gross alpha with appropriate compensation for naturally occurring isotopes.

Other radionuclides (Co-60, Sr-90, Cs-137, etc.) are important in determining actual offsite dose and in demonstrating compliance with the ECL using the sum of the fractions rule. The example calculations are used similarly for each isotope in the mix with their respective ECL, DCF and exposure pathway (inhalation, ingestion, and submersion).

Although not relevant to the hypothetical offsite dose calculation in the ECL and NUE analysis above, assumed effluent concentrations are approximately 1 DAC, 2 DAC, and 200 DAC for Am-241 at the point of release. Airborne radioactivity control measures to control worker dose, also limits the potential offsite dose.

2.0 LIQUID EFFLUENT DOSE CALCULATIONS

2.1 MONTH (31 DAY PERIOD) Deleted

2.2 CALENDAR QUARTER - Deleted

2.3 CALENDAR YEAR - Deleted

2.4 LIQUID EFFLUENT DOSE CALCULATION METHODOLOGY

As of December 31, 2013, HBPP has ceased liquid radioactive effluent discharges via the discharge canal to Humboldt Bay. Any remaining processed liquid radioactive waste is transported offsite for land disposal at an authorized disposal facility. The following calculation methodology is preserved as a part of the ODCM for ease of reference to site specific parameters in the event of an accidental release of liquid radioactive effluent. No recurring liquid effluent dose calculations are expected for the remainder of decommissioning.

The equations specified in this section 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.

Equation (2) of Regulatory Guide 1.109 provides for the use of a site specific mixing ratio (i.e. reciprocal of the dilution factor) that describes the near term and near field mixing of the tidal flow from the Discharge Canal into Humboldt Bay. A two-dimensional numerical analysis, depth-averaged, finite element hydrodynamic model (reference 12.1) was developed by CH2MHILL and used to estimate the dispersion of the canal discharge in the Bay. The analysis indicated that an additional dilution factor of 80 for batch release applications or a dilution factor of 20 for continuous release applications can conservatively be used to describe the Bay dilution. A factor of 20 will be applied in this calculation to address any combination of release modes.

Since the intake canal contains a larger volume of water, use of the above dilution factors for effluent releases to the intake canal provides a simplified, conservative methodology for calculating annual dose from effluent releases to the intake canal.

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:

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

where:

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

$C_{i\text{-Bay diluted}}$ = The average diluted Bay concentration, pico-Curie/liter, for radionuclide, i. If the outfall to the canal is at the furthest most portion of the canal from the entrance to the Bay, this will be estimated by calculating the total activity released (e.g. effluent concentration $C_{i\text{ effluent}}$ in pCi/L times the discharge volume V_D in Liters) then dividing the total activity of the nuclide discharged during the period, pico-Curies, by the dilution volume (e.g. total discharged volume V_D plus total tidal flow V_{TD} during the period in liters), and by the Bay dilution factor of 20. The total annual tidal flow for the outfall canal is 2.47E+9 Liters/year (e.g., 6.77E+6 Liters/day). If Gross Alpha radioactivity is determined to be in the effluent , Pu-241 will be considered to be present at 3.25 times the amount of detected Gross Alpha radioactivity. Note that the resulting dose commitment is the annual dose rate (mrem per year) for a time frame with this average concentration. Doses (NOT dose rates) for periods shorter than a year must be proportionately reduced.

$$C_{i\text{ - Bay diluted}} = \frac{C_{i\text{ - Effluent}} \times V_D}{(V_D + V_{TD}) \times 20} \tag{2-2}$$

If the outfall is not located in the furthest most portion of the canal from the entrance to the Bay, no credit for tidal dilution of the canal will be taken and the diluted Bay concentration will be calculated using the following equation.

$$C_{i\text{ - Bay diluted}} = \frac{C_{i\text{ - Effluent}}}{20} \tag{2-3}$$

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.

Dose calculations can be performed using the above methodology for the current month, quarter, or year.

Table 2-1
Ingestion Dose Factors for Adult Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from NUREG/CR-4013 (LADTAP II input values)

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	5.99×10^{-8}	5.99×10^{-8}	5.99×10^{-8}	5.99×10^{-8}	5.99×10^{-8}
Co-60	No Data	2.14×10^{-6}	4.72×10^{-6}	No Data	No Data	4.02×10^{-5}
Ni-63	1.30×10^{-4}	9.01×10^{-6}	4.36×10^{-6}	No Data	No Data	1.88×10^{-6}
Sr-90	8.71×10^{-3}	No Data	1.75×10^{-4}	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}
Pu-241	1.57×10^{-5}	7.45×10^{-7}	3.32×10^{-7}	1.53×10^{-6}	No Data	1.40×10^{-6}
Am-241	7.55×10^{-4}	7.05×10^{-4}	5.41×10^{-5}	4.07×10^{-4}	No Data	7.42×10^{-5}
Gross α	7.55×10^{-4}	7.05×10^{-4}	5.41×10^{-5}	4.07×10^{-4}	No Data	7.42×10^{-5}

Table 2-2
Ingestion Dose Factors for Teen Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from NUREG/CR-4013 (LADTAP II input values)

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	6.04×10^{-8}	6.04×10^{-8}	6.04×10^{-8}	6.04×10^{-8}	6.04×10^{-8}
Co-60	No Data	2.81×10^{-6}	6.33×10^{-6}	No Data	No Data	3.66×10^{-5}
Ni-63	1.77×10^{-4}	1.25×10^{-5}	6.00×10^{-6}	No Data	No Data	1.99×10^{-6}
Sr-90	1.02×10^{-2}	No Data	2.04×10^{-4}	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}
Pu-241	1.75×10^{-5}	8.40×10^{-7}	3.69×10^{-7}	1.71×10^{-6}	No Data	1.48×10^{-6}
Am-241	7.98×10^{-4}	7.53×10^{-4}	5.75×10^{-5}	4.31×10^{-4}	No Data	7.87×10^{-5}
Gross α	7.98×10^{-4}	7.53×10^{-4}	5.75×10^{-5}	4.31×10^{-4}	No Data	7.87×10^{-5}

Table 2-3
Ingestion Dose Factors for Child Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from NUREG/CR-4013 (ladTAP II input values)

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	1.16×10^{-7}	1.16×10^{-7}	1.16×10^{-7}	1.16×10^{-7}	1.16×10^{-7}
Co-60	No Data	5.29×10^{-6}	1.56×10^{-5}	No Data	No Data	2.93×10^{-5}
Ni-63	5.38×10^{-4}	2.88×10^{-5}	1.83×10^{-5}	No Data	No Data	1.94×10^{-6}
Sr-90	2.56×10^{-2}	No Data	5.15×10^{-4}	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}
Pu-241	3.87×10^{-5}	1.58×10^{-6}	8.04×10^{-7}	2.96×10^{-6}	No Data	1.44×10^{-6}
Am-241	1.36×10^{-3}	1.17×10^{-3}	1.02×10^{-4}	6.23×10^{-4}	No Data	7.64×10^{-5}
Gross α	1.36×10^{-3}	1.17×10^{-3}	1.02×10^{-4}	6.23×10^{-4}	No Data	7.64×10^{-5}

Table 2-4
Bioaccumulation Factors for Saltwater Environment
(pCi/kg per pCi/liter)
Selected Nuclides from Regulatory Guide 1.109, Table A-1 and from NUREG/CR-4013

Element	Fish	Invertebrate
H	9.0×10^{-1}	9.3×10^{-1}
Co	1.0×10^{2}	1.0×10^{3}
Ni	1.0×10^{2}	2.5×10^{2}
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}
Pu	3.0	2.0×10^{2}
Am	2.5×10^{1}	1.0×10^{3}
Gross α	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 EFFLUENT TREATMENT

3.1 TREATMENT REQUIREMENTS

3.1.1 Deleted

3.1.2 Deleted

3.2 Deleted

4.0 GASEOUS EFFLUENT DOSE CALCULATIONS

4.1 DOSE RATE

4.1.1 Deleted

As explained in Specification Bases 3.7, Noble Gases are not required to be monitored, and the corresponding dose rate need not be calculated.

4.1.2 Tritium and Radioactive Particulates

There are no short-lived radioactive particulates in the effluent, so radioactive decay can be neglected. Meteorological parameters are assumed to be constant, and applied for the most conservative location. Therefore, the radioactive particulates dose rate calculation methodology is the same as the radioactive particulates dose calculation methodology. Refer to sections 4.3.3 through 4.3.8 for the appropriate equations.

As explained in Specification Bases 3.5, Tritium is not required to be monitored, and the corresponding dose rate need not be calculated. Nevertheless, if such a calculation is required, refer to sections 4.3.9 through 4.3.13 for the appropriate equations.

4.2 Deleted

4.3 DOSE - TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

4.3.1 Calendar Quarter

The methodology for calendar quarter calculations is the same as for the calendar year calculations provided by section 4.3.3, and discussed in section 4.3.2, with the exception that the resulting values for D (annual dose commitment, mrem/year) must be divided by 4 to convert them to quarterly dose commitment, mrem/quarter.

4.3.2 Calendar Year

The methods for calculating the dose due to 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.

The equations provided for determining the doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

4.3.3 Particulate Organ Dose Calculation Summation Methodology

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.

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. Radioactive decay may result in the dose from Transuranic radionuclides becoming significant. If Gross Alpha radioactivity is determined to be released, Pu-241 will be considered to be present at 3.25 times the amount of detected Gross Alpha radioactivity. The annual dose commitment will be calculated for any organ of an individual age group as follows:

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

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 Decommissioning releases range from about 0.002 mrem/year (fruit/vegetable consumption pathway) to less than 1 x 10^{-6} mrem/year (for direct radiation exposure from material deposited on the ground).

4.3.4 Particulate Inhalation Pathway Dose Calculation Methodology

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

where:

χ/Q = The atmospheric dispersion parameter, seconds/cubic meter.
= 1.0 x 10^{-5} seconds/cubic meter for releases from the 50 foot stack. Refer to Appendix B, 1.2.
= 6.59 x 10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

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

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

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. No values are provided for Transuranic radionuclides, as their dose contribution to this pathway is negligible.

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, 1.3.
= 5.39×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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, 1.3.
= 3.29×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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

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

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.

F_f = 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, 1.3.
= 3.29×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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

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

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, 1.3.
= 3.29×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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:

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

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 1×10^{-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. Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

4.3.10 Tritium Inhalation Pathway Dose Calculation Methodology

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

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, 1.2.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

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.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

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:

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

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, 1.2.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

4.3.12 Tritium Grass-Cow-Meat Pathway Dose Calculation Methodology

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

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, 1.2.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

4.3.13 Tritium Vegetation Pathway Dose Calculation Methodology

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

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

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 x 10^{-5} seconds/cubic meter for releases from the 50 foot stack. Refer to Appendix B, 1.2.
	=	6.59 x 10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

Table 4-1
Inhalation Dose Factors for Adult Age Group
(mrem/pico-Curie inhaled)
Selected Nuclides from Regulatory Guide 1.109, Table E-7 and from NUREG/CR-4013

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}
Pu-241	3.42×10^{-2}	8.69×10^{-3}	1.29×10^{-3}	5.93×10^{-3}	1.52×10^{-4}	8.65×10^{-7}
Gross α	1.68	1.13	7.75×10^{-2}	5.04×10^{-1}	1.82×10^{-1}	4.84×10^{-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 and from NUREG/CR-4013

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}
Pu-241	3.74×10^{-2}	9.56×10^{-3}	1.40×10^{-3}	6.47×10^{-3}	2.60×10^{-4}	9.17×10^{-7}
Gross α	1.77	1.20	8.05×10^{-2}	5.32×10^{-1}	3.12×10^{-1}	5.13×10^{-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 and from NUREG/CR-4013

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}
Pu-241	7.94×10^{-2}	1.75×10^{-2}	2.93×10^{-3}	1.10×10^{-2}	5.06×10^{-4}	8.90×10^{-7}
Gross α	2.97	1.84	1.28×10^{-1}	7.63×10^{-1}	6.08×10^{-1}	4.98×10^{-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 and from NUREG/CR-4013

Nuclide	Organ					
	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}
Pu-241	8.43×10^{-2}	1.85×10^{-2}	3.11×10^{-3}	1.15×10^{-2}	7.62×10^{-4}	8.97×10^{-7}
Gross α	3.15	1.95	1.34×10^{-1}	7.94×10^{-1}	9.03×10^{-1}	5.02×10^{-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

Nuclide	Total Skin	Total 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}

Values are not provided for Transuranic radionuclides, as their dose contribution to this pathway is negligible.

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 and from NUREG/CR-4013

	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}
Pu-241	1.57×10^{-5}	7.45×10^{-7}	3.32×10^{-7}	1.53×10^{-6}	No Data	1.40×10^{-6}
Gross α	7.55×10^{-4}	7.05×10^{-4}	5.41×10^{-5}	4.07×10^{-4}	No Data	7.81×10^{-5}

Table 4-9
Ingestion Dose Factors for Teen Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-12 and from NUREG/CR-4013

	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}
Pu-241	1.75×10^{-5}	8.40×10^{-7}	3.69×10^{-7}	1.71×10^{-6}	No Data	1.48×10^{-6}
Gross α	7.98×10^{-4}	7.53×10^{-4}	5.75×10^{-5}	4.31×10^{-4}	No Data	8.28×10^{-5}

Table 4-10
Ingestion Dose Factors for Child Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-13 and from NUREG/CR-4013

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}
Pu-241	3.87×10^{-5}	1.58×10^{-6}	8.04×10^{-7}	2.96×10^{-6}	No Data	1.44×10^{-6}
Gross α	1.36×10^{-3}	1.17×10^{-3}	1.02×10^{-4}	6.23×10^{-4}	No Data	8.03×10^{-5}

Table 4-11
Ingestion Dose Factors for Infant Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-14 and from NUREG/CR-4013

Nuclide	Organ					
	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}
Pu-241	4.25×10^{-5}	1.76×10^{-6}	8.82×10^{-7}	3.17×10^{-6}	No Data	1.45×10^{-6}
Gross α	1.46×10^{-3}	1.27×10^{-3}	1.09×10^{-4}	6.55×10^{-4}	No Data	8.10×10^{-5}

Table 4-12
Stable Element Transfer Data For Cow-Milk Pathway
(days/liter)
Selected Nuclides from Regulatory Guide 1.109, Table E-1 and from NUREG/CR-4013

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}
Pu	5.0×10^{-6}
Gross α	5.0×10^{-6}

Table 4-13
Stable Element Transfer Data For Cow-Meat Pathway
(days/kilo-gram)
Selected Nuclides from Regulatory Guide 1.109, Table E-1 and from NUREG/CR-4013

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}
Pu	2.0×10^{-4}
Gross α	2.0×10^{-4}

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. Deleted

b. Modular HEPA Ventilation Particulate releases, using equation (4-3).

c. Deleted. Tritium is no longer a gaseous effluent source term.

d. Liquid releases, No longer applicable.

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. Deleted

b. Modular HEPA Ventilation releases, using equation (4-3). Tritium is no longer a gaseous effluent source term.

c. Liquid releases, No longer applicable.

d. The potential direct radiation exposure to an individual at the site boundary based 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. Deleted

b. Modular HEPA Ventilation releases, using equation (4-3).

c. Liquid releases, No longer applicable.

d. The potential direct radiation exposure to an individual at the site boundary based 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

Deleted - Based on the status of decommissioning, HBPP no longer anticipates wastes exceeding a specific activity that is unacceptable to disposal site without solidification or exceeding Class A as defined in 10 CFR 61.

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

Deleted - HBPP no longer anticipates wastes exceeding a specific activity that is unacceptable to disposal site without solidification or exceeding Class A as defined in 10 CFR 61. HBPP no longer anticipates disposal of wastes requiring stabilization in a High Integrity Container (HIC).

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, filters and other wet wastes shipped for land burial which contain a total specific activity less than the disposal site(s) 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 disposal site(s) (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.

8.2.4 Contract vendor solidification or dewatering services are utilized in accordance with PG&E approved supplier list and procurement procedures.

8.2.5 Vendor services may be conducted off site in accordance with their facility license and procedures. Vendor services include written confirmation of acceptable disposal waste form.

8.2.6 Gross dewatering of resins and filters may be performed onsite to achieve transport requirements in preparation for additional processing to a final waste form by offsite vendor services.

8.2.7 On site activities, such as managing wet soils from decommissioning excavations and process water shall be performed utilizing approved procedures or work instructions to ensure compliance with transportation regulations, disposal facility license requirements and/or waste acceptance criteria.

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 Decommissioning period. All changes shall be reviewed and approved by the HB Director 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.

9.3 HBPP is allowed to modify or reduce environmental requirements in the ODCM provided HBPP considers the modification or reduction from a technical and decommissioning perspective. [CMT 10.1]

10.0 COMMITMENTS

10.1 HBPP does not intend to modify or reduce the environmental monitoring requirements as specified in the ODCM during the period of SAFSTOR and decommissioning activities. This applies to those environmental samples and analysis identified as either quality or non-quality samples. This commitment is to be incorporated into the next revision of the ODCM. NOTE: HBPP is allowed to modify or reduce environmental requirements in the ODCM provided HBPP considers the modification or reduction from a technical and decommissioning perspective.

11.0 RESPONSIBLE ORGANIZATION

Radiation Protection Manager

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. As of December 31, 2013, HBPP ceased processed liquid effluent to the discharge canal and processed liquid effluent will be transported for disposal at a regulated disposal site. The Ground Water Treatment System (GWTS) was removed from service in April 2019. 08/19

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	8.60E-2	7.17E-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

Table A-3 below reflects the Gaseous Effluent Activity as a representation of the state of decommissioning during the calendar year 2013 relative to the Baseline above.

Table A-3
2013 Gaseous Effluent Activity

Type of Activity	Annual Release (Curies)	Monthly Average Release (Curies)
Particulate Gamma Emitters (total)	<1.5E-5	<1.3E-6
Strontium-90	<1E-6	<1E-7
Particulate Alpha Emitters (total)	<1E-6	<1E-7

Table A-3 data is summarized from the 2013 Annual Effluent Release Report and are listed as less than values because sampling results were the composite of LLD values. Tritium is no longer monitored due to a lack of significant source term.

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:

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

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

$$X/Q = \frac{1}{\bar{U}_{10}\pi\Sigma_y\,\sigma_z} \qquad (3)$$

where:

$\bar{U}_{10}$ = 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 = 4.70×10^{-3} seconds/meter^3 (1)

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

$X/Q \quad = \quad 6.59 \times 10^{-3}$ 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 (1.4×10^{-4} meter^{-1})/ (6 x 4.33 meter) = 5.39×10^{-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

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PACIFIC GAS AND ELECTRIC COMPANY
NUCLEAR POWER GENERATION
HUMBOLDT BAY POWER PLANT

SAFSTOR/Decommissioning Offsite Dose Calculation Manual
Revision 31

Summary of Changes Included in Revision 31 of the SAFSTOR/Decommissioning Offsite Dose Calculation Manual

Summary of Changes:

Page / Section	Change Date	Change	Reason
Page ii, Introduction	Rev. 31	Added information on p-ii summarizing the status of gaseous effluents, the interlaboratory comparison program and process control program.	The change prepares the reviewer for the status of decommissioning. Although the effluent performance specifications outlined in the Offsite Dose Calculation Manual (ODCM) remain unchanged, the methods for demonstrating performance are changed.
Page I-18 Table 2-7	Rev 31	Onsite Airborne Monitoring Locations footnote (d) added to indicate that alpha, beta and gamma sample analysis will be performed by an offsite laboratory quarterly. Delete phrase: “following filter change”.	Maintaining onsite sample analysis for alpha and beta analysis requires keeping sources for calibration and response checks on instrumentation. The contaminant of concern has historically been Co-60, Cs-137, Sr-90 and Am-241. Each of these contaminants has a relatively long half-life (5.27 to 432 years). If present, the air sample quality is not diminished by quarterly assay for beta and alpha contaminants along with the composite gamma analysis.
Page I-23 Table 2-10, Figure 2-3	Rev 31	Delete Humboldt Hill Road air sample location 3	Onsite air samples are an indication of the environmental impact from the site. With no stack and no effluent pathway, the offsite location provides no value added.

Page / Section	Change Date	Change	Reason
I-26 2.12	Rev. 31	No change to the specification, however, the method of implementing a REMP Interlaboratory Comparison Program will change.	An interlaboratory comparison program can be maintained via the quarterly assessment of Radiological Environmental Monitoring Program (REMP) air samples by an independent laboratory. Instead of an independent laboratory sending blind samples to Humboldt Bay Power Plant (HBPP) for analysis. A quarterly assessment of the samples generated by HBPP will be sent to an independent laboratory for independent assay for gross beta, gross alpha, and gamma analysis in accordance with the requirements of ODCM Table 2-7. The off-site analysis will be performed to the lower limit of detection required by Table 2-9.

(Procedure Classification - Quality Related)

INTRODUCTION

The SAFSTOR/DECOMMISSIONING 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 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 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 requirements for the REMP. 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. As of December 31, 2013, HBPP ceased liquid radioactive effluent discharges via the discharge canal to Humboldt Bay. The Main Plant stack was shut down in October 2015. As of June 2016, use of modular HEPA units to control elevated airborne radioactivity and effluents was no longer required. Onsite analysis capability for the remaining REMP samples was available into October 2019 and remains available through an offsite laboratory. The interlaboratory comparison program requirement is satisfied via use of a reputable laboratory. The scope of the REMP and interlaboratory comparison program are the dosimeters and air samples required to evaluate the direct radiation and gaseous effluents from HBPP. 11/19

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 decommissioning period. Part II of the ODCM contains the calculational methods for gaseous and liquid effluents to preserve site specific data although the gaseous effluent pathway and the liquid discharge pathway has been terminated. 11/19

The ODCM also contains the Process Control Program (PCP) for solid radioactive wastes, administrative controls regarding the content of the Annual Radiological Environmental Monitoring Program Report, administrative controls regarding the content of the Annual Radioactive Effluent Release Report, and administrative controls regarding major changes to radioactive waste treatment systems. Since there are no remaining liquid process systems onsite, the requirement for a Process Control Program is effectively reduced to ensuring the receiving disposal site acceptance criteria is satisfied. 11/19

The ODCM shall become effective after approval by the HB Director. 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 required level of radioactive effluent control and not adversely impact the accuracy or reliability of effluent or dose 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 Deleted

1.4 Deleted

1.5 Deleted

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.

1.7 Deleted

1.8 FREQUENCY NOTATION

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

1.9 Deleted

1.10 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.

1.11 INSTANTANEOUS CONCENTRATION

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

1.12 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.

1.13 MODULAR HEPA VENTILATION UNIT

MODULAR HEPA VENTILATION UNIT consists of HEPA filter trains discharged to the environment and sampled in accordance with ANSI/HPS N13.1-1999.

1.14 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, 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 Monitoring Report and the Annual Radioactive Effluent Release Report. The ODCM also contains the Process Control Program (PCP) for solid radioactive wastes.

1.15 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).

1.16 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, disposal site(s) requirements, and other requirements governing the disposal of solid radioactive waste.

1.17 Deleted

1.18 RESTRICTED AREA

The RESTRICTED AREA is defined by 10CFR20.1003. The physical location(s) of the RESTRICTED AREA shall be defined in plant procedures.

1.19 SITE BOUNDARY

The SITE BOUNDARY shall be the boundary of the UNRESTRICTED AREA used in the offsite dose calculations for gaseous and liquid effluents. The SITE BOUNDARY is shown in Figure 1-1. Ingress and egress through the SITE BOUNDARY are controlled by the Company.

1.20 Deleted

1.21 Deleted

1.22 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.

1.23 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.”

1.24 VENTILATION EXHAUST TREATMENT SYSTEM

A VENTILATION EXHAUST TREATMENT SYSTEM is any system designed and installed to reduce radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal absorbers and/or HEPA filters for the purpose of removing particulates from the gaseous exhaust stream prior to release to the environment.

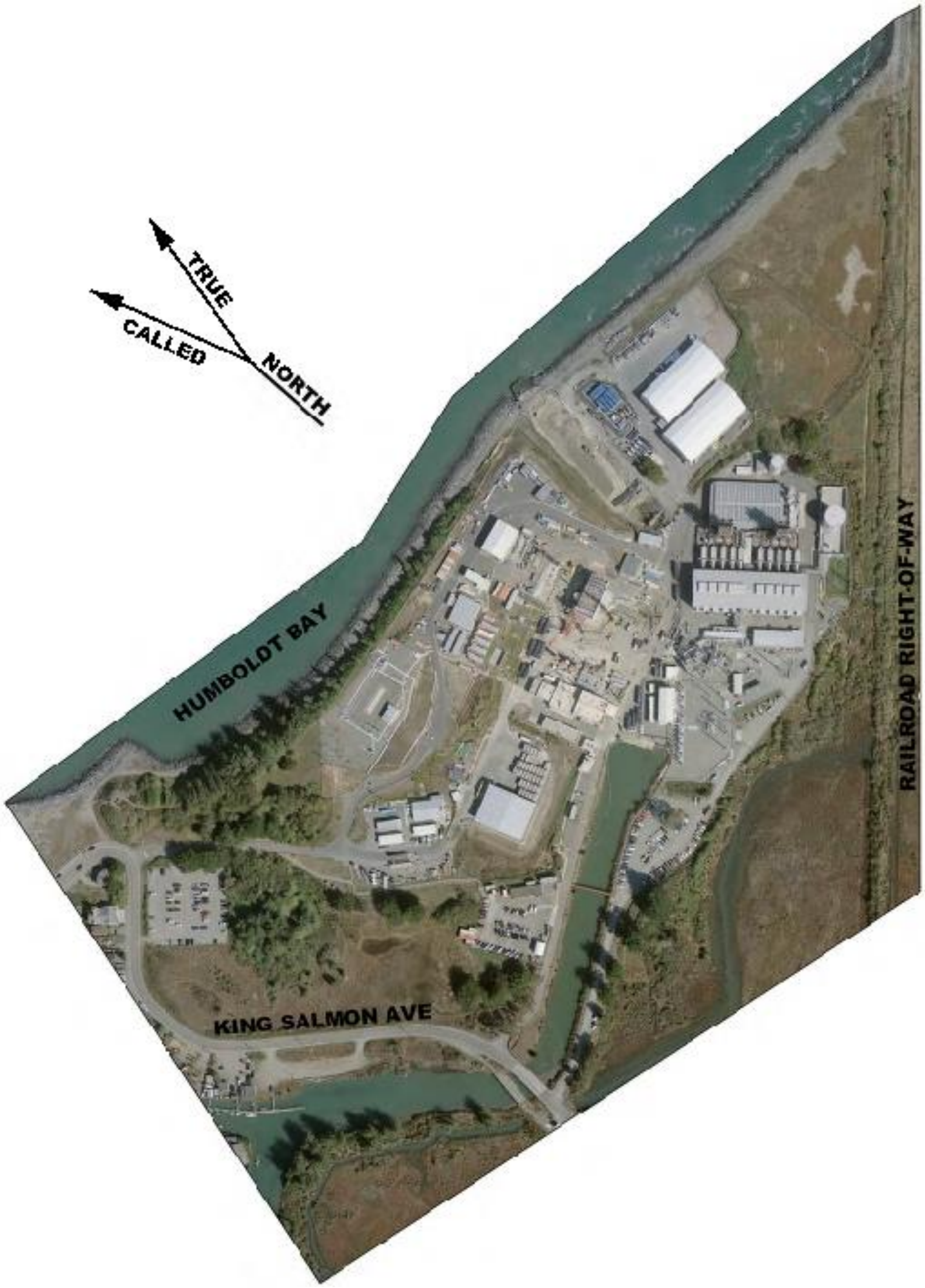
1.25 Deleted

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.

Figure 1-1
SITE BOUNDARY



2.0 SPECIFICATIONS

2.1 **Deleted; Table 2-1 - Deleted; Table 2.2 - Deleted**

2.2 RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION[1]

LIMITING CONDITIONS

2.2.1 Deleted - plant stack is no longer in operation.

SURVEILLANCE REQUIREMENTS

2.2.2 Deleted

Table 2-3 - Deleted
Table 2-4 - Deleted

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

Deleted (See BASES Section 3.2 and Appendix A)

Table 2-5 (Deleted)

2.4 LIQUID EFFLUENT – DOSE Deleted - No longer applicable

2.5 Deleted - No longer applicable

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

SURVEILLANCE REQUIREMENTS

2.6.2 Deleted (see BASES section 3.5)

2.6.3 Deleted (see BASES section 3.5)

2.6.4 Radioactive particulates, with half-lives of greater than 8 days, in gaseous effluents released to the environment shall be sampled and analyzed in accordance with the sampling and analysis program of Table 2-6, and their concentrations shall be compared with the limits of 10CFR20, Appendix B, Table 2, Column 1. IF their concentrations exceed those limits, the calculational methods in Part II of the ODCM shall be used to determine whether or not the limits of Specification 2.6.1 have been exceeded. The actual sample period shall be used to determine the dose rate during the sample period.

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]
Modular HEPA Ventilation Discharge				
	Continuous[b,d]	W[b] Mixing Box Particulate Sample	Principal Gamma Emitters[e]	1×10^{-11}
	Continuous[b,d]	W[b] Mixing Box Particulate Sample	Gross Alpha	1×10^{-12}
	Continuous[b,d]	W[b] Mixing Box Particulate Sample	Gross Beta	6.7×10^{-12}
	Continuous[b,d]	Q Composite of Mixing Box Particulate Samples	Sr-90[g]	1×10^{-11}
	Continuous[b,d,h]	Q Composite of Mixing Box Particulate Samples	Am-241	1×10^{-12}
	Continuous[b,d,i]	Q Composite of Mixing Box Particulate Samples	Am-241	1×10^{-14}

Table 2-6 (Continued)

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),

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. NOTE: The LLDs are achievable with a reasonable count time assuming adequate effluent volume and sample volume. If the LLD is not achieved, initiate a condition report to document that the LLD was not achieved and indicate a probable cause (short runtime, equipment malfunction, etc.). RP Supervision will determine if additional calculations should be performed per Surveillance 2.6.4.

Table 2-6 (Continued)

Table Notation (Continued)

[b] Samples shall be changed at least once per 7 days (3 day extension permitted), assuming effluent pathway is in continuous use (typically > 40 hrs per week). Samples may be collected more frequently for short duration use of a Modular HEPA Ventilation Unit.

[c] Deleted

[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, and 2.8.

[e] The principal gamma emitters for which the LLD specification applies exclusively are 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.

[f] Deleted based on SPAMS no longer in service.

[g] Analysis specific to Sr-90 may be replaced by analysis for total radioactive Strontium.

[h] When release volume is less than or equal to 3.26×10^{11} ml (e.g., 1.15E+7 cubic feet).

[i] When release volume exceeds 3.26×10^{11} ml (e.g., 1.15E+7 cubic feet).

2.7 Deleted

2.8 GASEOUS EFFLUENTS: DOSE - RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITIONS

2.8.1 The dose to a MEMBER OF THE PUBLIC from the release of 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 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 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 dose calculation for the current calendar quarter and the current calendar year, for the release of radioactive materials in particulate form with half-lives greater than 8 days,
OR

Perform a BASELINE COMPARISON for gaseous effluent radioactivity (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.
OR

Perform a dose assessment, if weekly sampling indicates the effluent from modular HEPA units exceed 0.1 uCi of alpha emitters or Sr-90. The assessment of alpha and beta may be performed with appropriate compensation for naturally occurring nuclides.

As explained in Specification Bases section 3.8, neither routine surveillance nor dose calculations are required for Tritium in gaseous effluents.

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 disposal site(s) requirements.

APPLICABILITY: At all times.

ACTION:

With the provisions of the 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) meet the shipping, disposal site(s) requirements with regard to dewatering and off site vendor processes.

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 gaseous effluents exceeding twice the limits of Specification 2.8.1.a, or 2.8.1.b, calculations should be made, which include direct radiation contributions from Unit No. 3, 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 gaseous effluents shall be calculated in accordance with dose calculation methodology provided for Specification 2.8.1.

2.11 REMP MONITORING PROGRAM

LIMITING CONDITIONS

2.11.1 A radiological environmental monitoring program shall be provided to monitor the radiation and radionuclides in the environs of the facility. The 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 Program Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. A Special Report pursuant to Administrative Control 4.3, 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 Specification 2.8. Prepare and submit to the Commission within 30 days of obtaining analytical results from the affected sampling period which includes an evaluation of release conditions, environmental factors or other aspects which caused the dose limits to be exceeded. 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 Program Report.

SURVEILLANCE REQUIREMENTS

2.11.2 The radiological environmental monitoring samples shall be collected pursuant to Table 2-7 from the "Quality Related" locations given in Tables 2-7 and 2-10 and Figures, 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)]	Sampling and Collection Frequency	Type of Analysis	PROGRAM BASIS ODCM Specs (QR)
AIRBORNE	2 onsite locations, 1 offsite location	Continuous sampler operation with sample collection at least once per 7 days[(1)(c)]	Gross alpha and gross beta radioactivity [(d)] Gamma isotopic[(b)] analysis on quarterly composite (by station)	X
DIRECT RADIATION	Minimum of 8 onsite stations, at or within the SITE BOUNDARY fence line, 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
WATERBORNE	None	N/A	N/A	
INGESTION	None	N/A	N/A	
TERRESTRIAL	None	N/A	N/A	

Table Notations

QR - Quality Related (1)Performed by HBPP (3)Performed by a NVLAP accredited processor

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

(c) Continuous sampler operation may be limited to normal work hours to represent effluents from decommissioning activities. Count times may need to be adjusted to achieve the recommended LLDs in Table 2-9.

(d) In the absence of onsite analysis capability, the weekly samples are sent to an offsite laboratory for gross alpha, gross beta analysis quarterly and composite gamma isotopic analysis. 11/19

Table 2-8 (Deleted)

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

Analysis	Airborne Particulate (pCi/m^3)
Gross Beta	0.01
H-3	
Co-60	
Cs-137	0.06

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 Monitoring Program 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:

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

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.

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 Radiological Environmental Monitoring Program 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.

Table 2-10
DISTANCES AND DIRECTIONS TO ENVIRONMENTAL MONITORING STATIONS

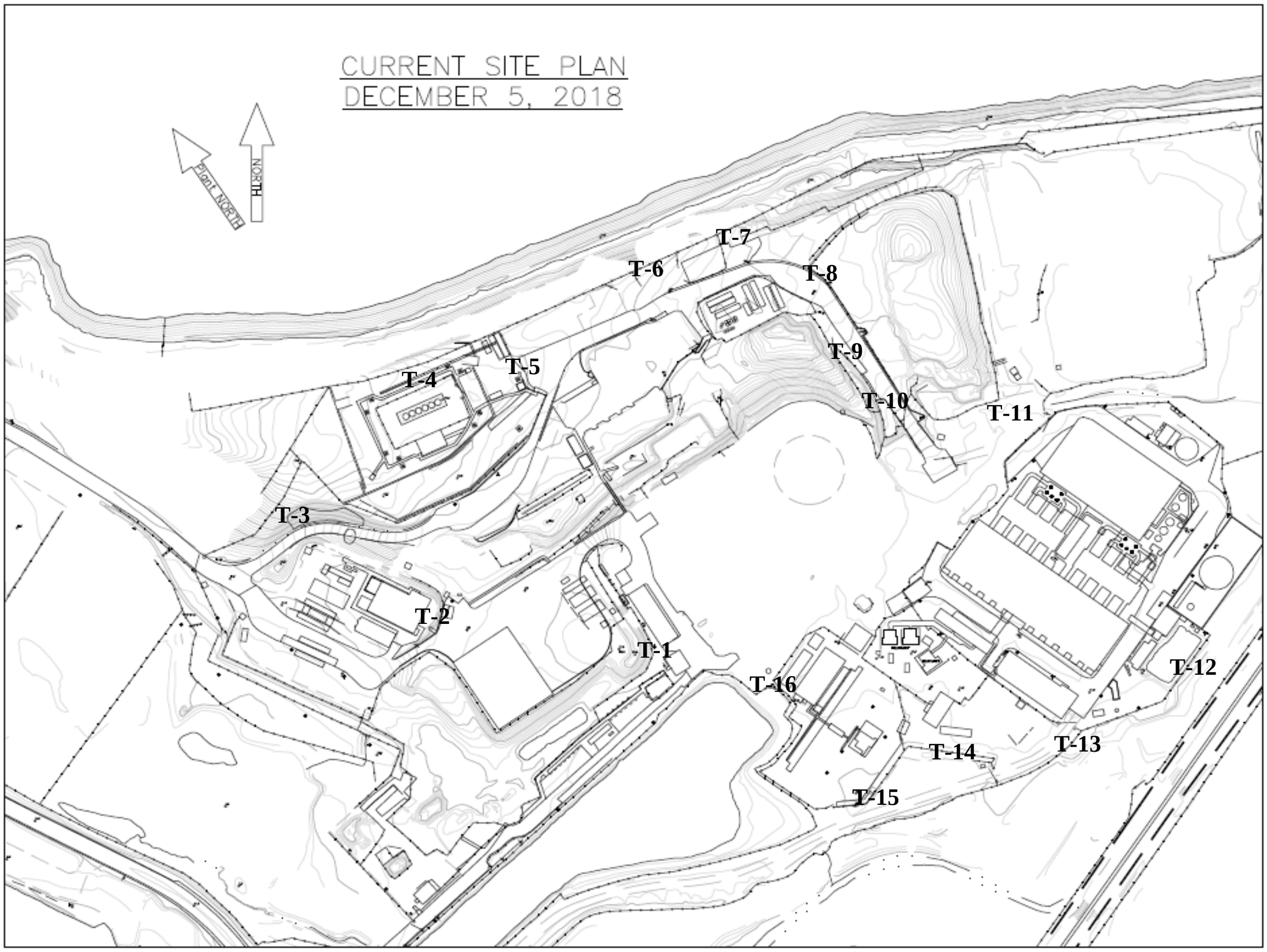
Station No.	Code	Station Name	Radial Direction Sector	Radial Direction By Degrees	Radial Distance from Plant (Miles)
1	Δ	King Salmon Picnic Area	W	270	0.3
2	Δ	180 Dinsmore Drive, Fortuna	SSE	158	9.4
14	Δ	South Bay School Parking Lot	S	180	0.4
17	Δ	Control Set at Humboldt Substation, Eureka	NEE	61	5.8
25	Δ	Irving Drive, Humboldt Hill	SSE	175	1.3

11/19

Table Notations

Code: Δ Dosimetry Station 11/19

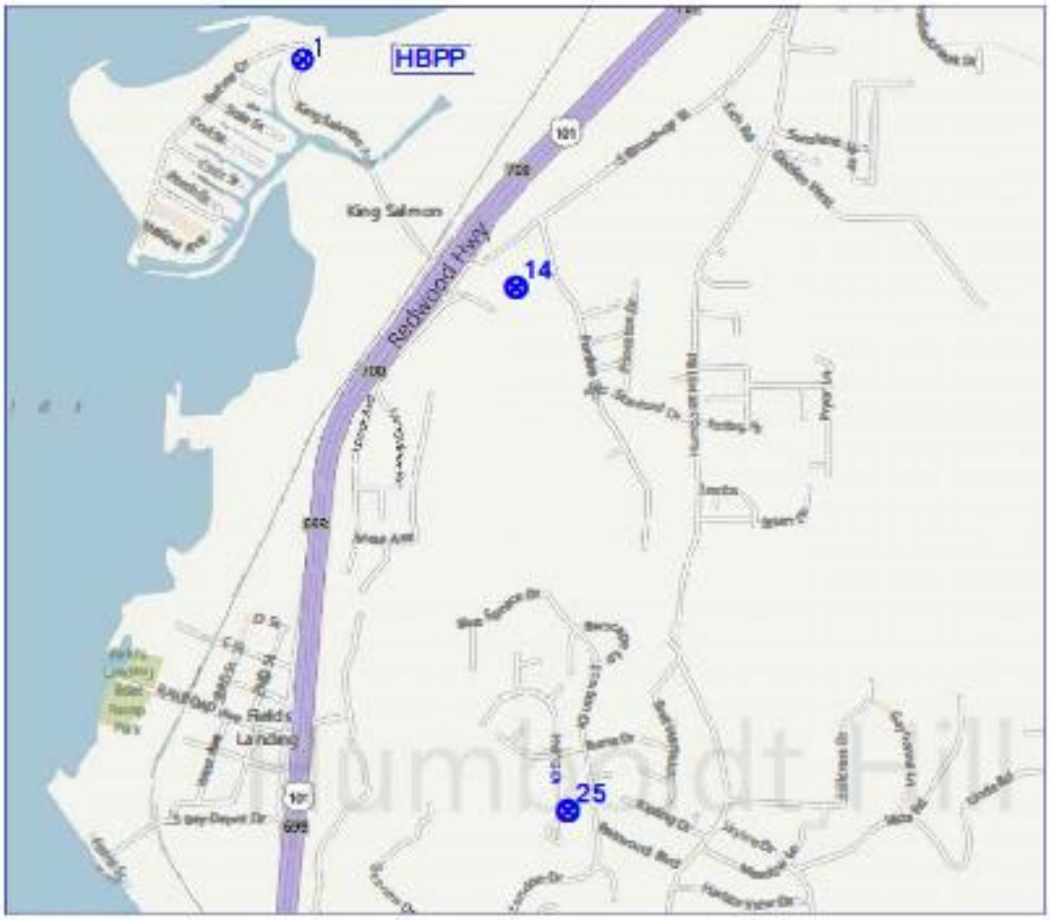
Figure 2-1
HBPP Onsite TLD Locations



Monitoring locations T7, T10, T11, T13, T16, T2, T3, and T5 generally represent REMP Site Boundary direct exposure monitoring locations in the 8 primary compass points beginning with T-7 to representing north and moving clockwise.

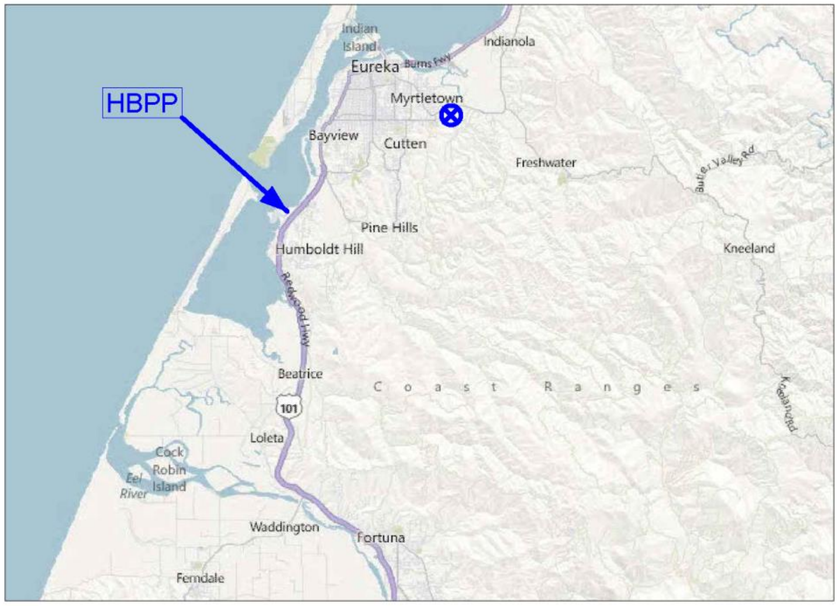
Figure 2-2 - Deleted

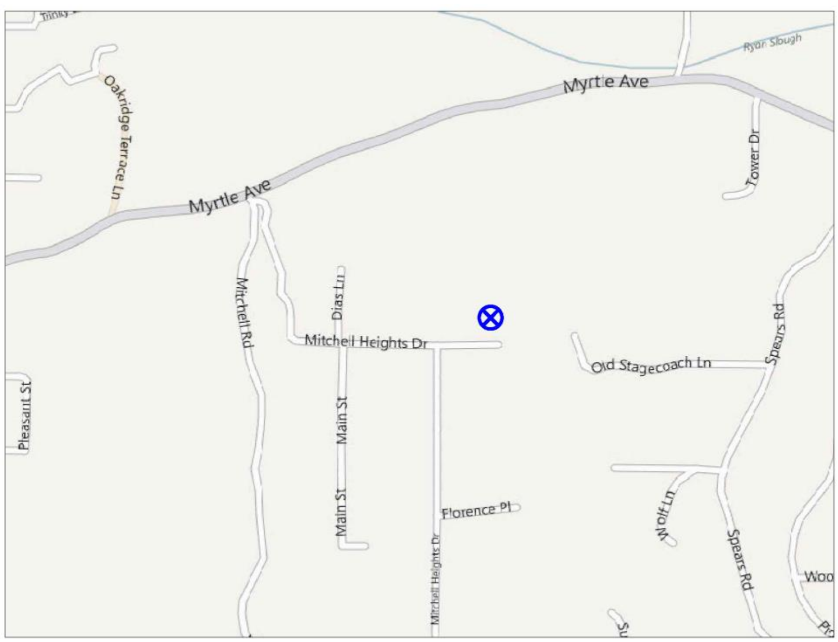
Figure 2-3
HBPP OFFSITE SAMPLING LOCATIONS - HUMBOLDT HILL



Station	GPS Coordinates (NAD83/NAVD88 CA. Zone 1)			Decimal Degrees	
	Easting	Northing	el.	Latitude	Longitude
1	5948026.52	2161183.79	11.38	40.74156	-124.21903
14	5949876.83	2158864.39	18.65	40.73533	-124.20802
25	5950247.30	2154214.18	229.22	40.72260	-124.20626

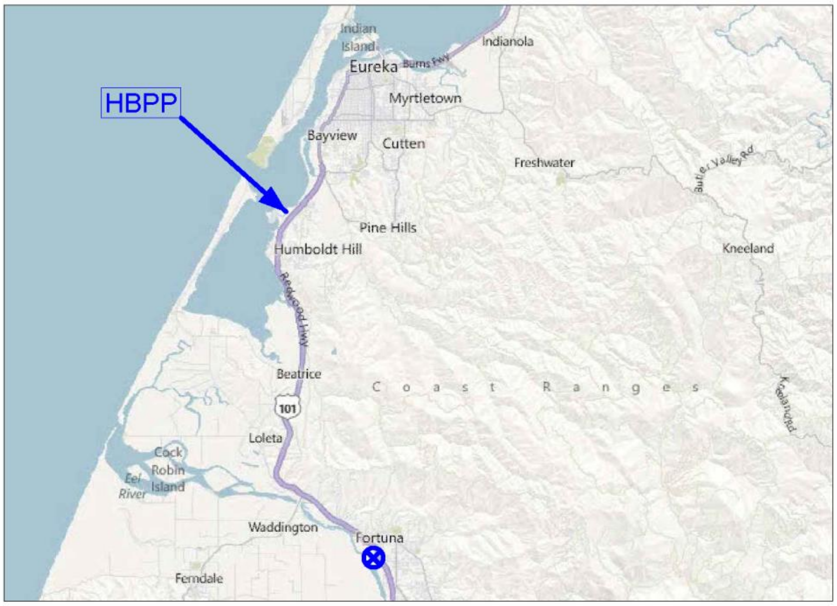
Figure 2-4
HBPP OFFSITE SAMPLING LOCATIONS - EUREKA

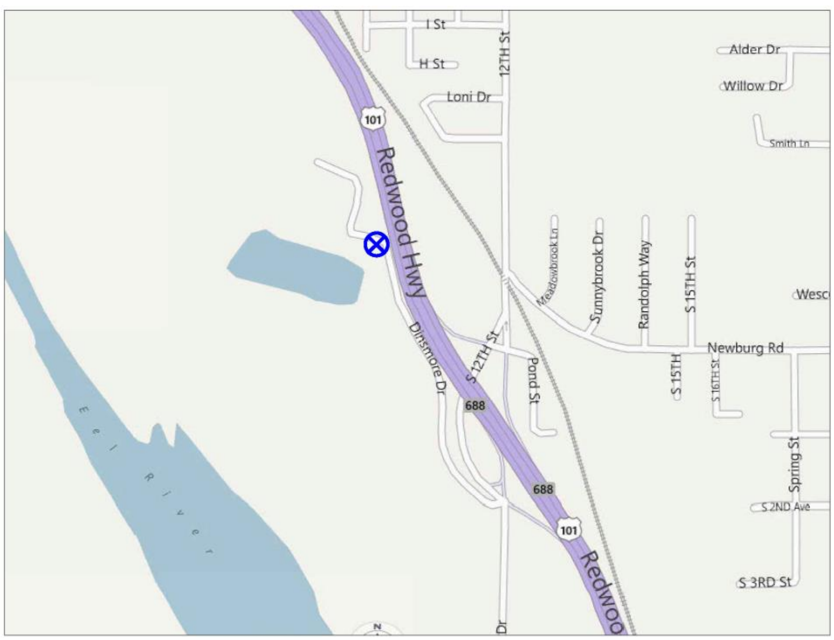




Station	GPS Coordinates (NAD83/NAVD88 CA. Zone 1)			Decimal Degrees	
	Easting	Northing	el.	Latitude	Longitude
17	5976549.55	2175490.19	164.85	40.78276	-124.11324

Figure 2-5
HBPP OFFSITE SAMPLING LOCATIONS - FORTUNA





Station	GPS Coordinates (NAD83/NAVD88 CA. Zone 1)			Decimal Degrees	
	Easting	Northing	el.	Latitude	Longitude
2	5962583.86	2105797.82	35.53	40.59057	-124.15746

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 Program Report.

SURVEILLANCE REQUIREMENTS

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

2.13 RADIOACTIVE WASTE INVENTORY

LIMITING CONDITIONS

2.13.1 Liquid Radioactive Waste In Outdoor Tanks

The radiological inventory of wastes in outdoor tanks that are not capable of retaining or treating tank overflows shall not exceed 0.25 Ci.

APPLICABILITY: At all times.

ACTION:

When the inventory exceeds the conditions as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Monitoring Program Report.

2.13.2 Deleted

SURVEILLANCE REQUIREMENTS

2.13.3 An inventory of the estimated liquid radioactive waste in outdoor tanks inventory shall be maintained to verify the 0.25 Ci limit is not exceeded.

OR

Provide overflow protection.

OR

Use process knowledge of typical concentration and tank volume to verify that the 0.25 Ci is not exceeded.

3.0 SPECIFICATION BASES

3.1 Radioactive Gaseous Effluent Monitoring Instrumentation Basis

Deleted – The plant stack ceased operation in 2015. Monitoring gaseous effluent is limited to sampling and analysis of Modular HEPA Units.

3.2 Liquid Effluent Concentration Basis

Deleted - Liquid effluents are no longer discharged to Humboldt Bay. Effective December 31, 2013, discharge of processed radioactive liquid effluents to Humboldt Bay was terminated. Any remaining or incidental radioactive liquid in concentrations exceeding 10 times 10 CFR 20, Appendix B, Table 2 Column 2 are manifested for disposal at a licensed disposal site. Sampling and manifesting requirements are consistent with the requirements of the receiving facility not subject to ODCM methodology.

3.3 Liquid Effluent Dose Basis

Deleted - Liquid effluents are no longer discharged to Humboldt Bay.

3.4 Liquid Effluent Treatment Basis

Deleted - Liquid effluents are no longer discharged to Humboldt Bay.

3.5 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).

Stack operation and monitoring ceased operation in 2015, so the reporting period for 2015 includes the dose contribution from the plant stack prior to ceasing operation. Modular HEPA Ventilation Units continue to be sampled as a gaseous effluent pathway.

Noble gas monitoring is not required because the spent fuel (noble gas source term) has been transferred to the ISFSI. Tritium monitoring is not required in gaseous effluents because the tritium source term was the spent fuel pool water which is now empty. Residual water in various plant drains and sumps contain low levels of tritium (generally at or below the drinking water standard (2E-5 uCi/ml or 20,000 pCi/L) and does not require monitoring as a gaseous plant effluent.

3.6 Deleted

Gaseous effluent monitoring is not required for noble gases because the spent fuel (noble gas source term) has been transferred to the ISFSI.

3.7 Deleted

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 basis for the dose calculation threshold of 0.1 uCi alpha emission or Sr-90 in a week assumes a continuous ground level release of 1.65E-13 uCi/sec and an X/Q of 6.59E-3 sec/m^3. The limiting inhalation dose is to a teen age member of the public at the site boundary at approximately 0.3 mrem/wk (15 mrem/yr) to the bone from alpha emitters. Compliance with this Specification has been established on a licensing basis by the SAFSTOR Environmental Report and NUREG-1166, “Final Environmental Statement for Decommissioning Humboldt Bay Power Plant.” These reports have demonstrated that routine release of Tritium and radioactive materials in particulate form (with half-lives greater than 8 days) in gaseous effluents during decommissioning will not cause the Specification to be exceeded. As long as routine releases do not exceed the baseline quantities evaluated in these reports, no further dose calculation is necessary.

The previously evaluated tritium source term was the spent fuel pool water, which is now empty. Residual water in various plant drains and sumps contain low levels of tritium (at or below the drinking water standard (2E-5 uCi/ml or 20,000 pCi/L) and does not require monitoring as a gaseous plant effluent.

3.9 Solid Radioactive Waste Basis

This Specification ensures that radioactive wastes that are transported from the site shall meet the disposal site(s) licensee and/or waste acceptance criteria for free standing liquids 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 is required to provide assurance that the baseline conditions established by the Environmental Report are not deteriorating and it supplements the SAFSTOR Environmental Report baseline

environmental conditions by conducting onsite and offsite environmental monitoring to evaluate routine conditions during decommissioning and to document any increased nuclide concentrations and/or radiation levels resulting from accidents during decommissioning.

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.

The LLD's required by Table 2-9 are considered optimum for routine environmental measurements in industrial laboratories. HBPP no longer includes water, milk, fish, food products, or sediment in its routine REMP sampling program. Sampling and analysis in support of the License Termination Plan is independent of the ODCM requirements.

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.

3.13 Radioactive Waste Inventory Basis

The requirements for limits on the accumulation of liquid radioactive waste in outdoor tanks were transferred from the license Technical Specifications.

4.0 ADMINISTRATIVE CONTROLS

4.1 Annual Radiological Environmental Monitoring Report

A report on the Decommissioning Radiological Environmental Monitoring Program shall be prepared annually in accordance with the NRC Branch Technical Position and submitted to the NRC by May 1 of each year.

The Annual Radiological Environmental Monitoring 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. The material provided shall be consistent with the objectives outlined in the ODCM, and in 10CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

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 Report Annual Summary, or equivalent. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in the next annual report.

d. A summary description of the Decommissioning 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. Deleted – water samples are not collected as a part of the REMP.

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

Table 4-1
RADIOLOGICAL ENVIRONMENTAL MONITORING REPORT ANNUAL 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							
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	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
Groundwater	Not Required	N/A	N/A	N/A	N/A	Not Required	N/A
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 REPORT ANNUAL 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 as 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].

Not Required - not required by the HBPP Offsite Dose Calculation Manual. 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. 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. The material provided shall be consistent with the objectives outlined in the ODCM and in conformance with 10CFR 50.36a and 10CFR Part 50, Appendix I, Section IV.B.I. Beginning in the reporting year 2014, liquid effluents shipped for processing or disposal at a regulated disposal site are included in the annual report.

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 gaseous monitoring instrumentation inoperable for greater than 30 days in accordance with Specification 2.2. Beginning the reporting year 2015, following cessation of the plant stack operation, the effluent monitoring instrumentation associated with Specification 2.2 ceased operation. Inoperability and lack of timely corrective action is only applicable to the period of plant stack operation. Anomalies associated with monitoring effluent from Modular HEPA Ventilation systems will be reported.

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 2.8 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 approved by the HB Director.

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.

4.5 Process Control Program Changes

a. Changes to the Process Control Program (PCP) shall be documented and records of reviews performed shall be retained as required for the duration of Decommissioning.

b. The following information shall be available for review:

1. Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

2. A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

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

c. The change shall become effective after approval of the HB Director.

PART II - CALCULATIONAL METHODS AND PARAMETERS

1.0 UNRESTRICTED AREA EFFLUENT CONCENTRATIONS

1.1 LIQUID EFFLUENT UNRESTRICTED AREA CONCENTRATIONS

Specification 2.3.1 requires that the Radioactive Liquid Effluent Sample concentrations (RLES) are calculated 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). This requirement is defined by the following relationship.

$$\sum_i \frac{C_{i,\,Canal}}{10 \times ECL_i} \leq 1 \qquad (1\text{-}1)$$

where:

$C_{i\text{-}Canal}$ = The concentration of isotope " i " in the canal discharge point to Humboldt Bay.

ECL_i = Effluent Concentration Limit for radionuclide " i " from 10 CFR 20, Appendix B, Table 2, Column, 2 (μCi/ml)

1.1.1 If the outfall location is not at the furthermost portion of the canal from the entrance to the Bay the concentration of the isotope $C_{i\text{-}Canal}$ is equal to the concentration being discharged at the outfall.

1.2 UNRESTRICTED AREA GASEOUS EFFLUENT CONCENTRATIONS

1.2.1 Equation C-4 of Regulatory Guide 1.109 demonstrates how to calculate dose from inhalation:

The annual dose associated with inhalation of all radionuclides, to organ j *of an individual in age group* a*, is then:*

$D_{ja}(r,\theta) = R_a \Sigma x_i(r,\theta) DFA_{ija}$

where

D_{ja} *is the annual dose rate to organ* j *of an individual in age group* a

R_a *is the breathing rate for age group* a

$x_i(r,\theta)$ *is the* annual *average ground-level concentration of nuclide* i *in air in sector* θ *at distance* r*, in* pCi/m^3

DFA_{ija} *is the dose factor for nuclide* i *to organ* j *of age group* a

To calculate $x_i(r,\theta)$ the annual average ground-level concentration of nuclide i in air in sector θ at distance r, in pCi/m^3 the equation must be rearranged to:

$D_{ja}(r,\theta)/(DFAija\ R_a) = x_i(r,\theta)$

Assuming that:

Americium-241 is the primary nuclide

The maximally exposed group is the Teen based on breathing rates and DFAija

The DFAija to the bone of a Teen from Am-241 is 1.77 mrem/pCi

The DFAija are taken from: NRC NUREG/CR-4013, "*LADTAP-II Technical Reference and User Guide*"

The Teen breathing rate is 8000 m^3/year

Therefore the ground-level concentration of Am-241 in air in sector θ at distance r, in pCi/m^3 that will produce a dose rate of 1500 mrem/year to the bone of a Teen is:

$(1500 \text{ mrem/year}) / (1.77 \text{ mrem/pCi}) / (8000 \text{ m}^3\text{/year}) = 1.06E\text{-}1 \text{ pCi/ m}^3$

$1.06E\text{-}1 \text{pCi/ m}^3 =$

$(1.06E\text{-}1 \text{ pCi/m}^3) / (1E6 \text{ pCi /}\mu\text{Ci}) / (1E6 \text{ ml/m}^3) = 1.06E\text{-}13 \ \mu\text{Ci/ml}$

1.2.2 Quantity of radioactive material released

Equation C-3 of Regulatory Guide 1.109 demonstrates how to calculate the quantity of material that must be released to produce a given airborne concentration:

The annual average airborne concentration of radionuclide i at the location (r, θ) with respect to the release point may be determined as

$$x_i(r,\theta) = 3.17 \times 10^4 \, Q_i(\chi/Q)^D(r,\theta)$$

where

$x_i(r,\theta)$ *is the annual average ground-level concentration of nuclide* i *in air in sector* θ *at distance* r, *in* pCi/m^3

3.17×10^4 *is the number of pCi/Ci divided by the number of sec/yr*

$(\chi/Q)^D(r,\theta)$ *is the annual average atmosphere dispersion factor, in sec/m^3.*

Q_i *is the release rate of nuclide I to the atmosphere, in Ci/yr*

A value of 6.59E-3 sec/m^3 was used for the incidental release path atmosphere dispersion factor at the site boundary $(\chi/Q)^D(r,\theta)$ for releases from Modular HEPA Units. This is based on a release rate of 2000 cfm. (Ref: Safstor ODCM, Appendix B, 2.0) 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*.

To determine the release rate that will result in an average ground-level concentration the above equation must be rearranged to:

$$Q_i = x_i(r,\theta) / (3.17 \times 10^4 (\chi/Q)^D(r,\theta))$$

Therefore the Modular HEPA Unit release rate of Am-241 required to equal the incidental ground-level concentration at the site boundary calculated above is:

1.06E-1 pCi/m^3 / ((3.17E4 (pCi/Ci)/ (sec/yr)) * (6.59E-3 sec/m^3)) =

5.07E-4 Ci/yr or 5.07E2 uCi/yr

1.2.3 Transmission Fraction

Deleted – no on line monitoring provided.

1.2.4 Effluent Concentration

The Modular HEPA Unit concentration that would result in a release rate of 5.07E-4 Ci/yr is equal to:

Total release (Curies/year) / Release rate (cc/year)

The average annual Modular HEPA Unit flow rate is 2,000 cfm

This results in a total volume of 2.98E13 cc/yr

This is based on (2000 ft^3/min * 525,600 minutes/yr * 28,317 cc/ft^3).

(5.07E-4 Ci * 1E6 µCi/Ci) / (2.98E13 cc/yr) = 1.70E-11 µCi/cc

Therefore an indicated Modular HEPA concentration of 1.70E-11 µCi/cc at 2000 cfm for one calendar year would result in a dose of 1500 mrem to a member of the public at the site boundary.

Two times the indicated release rate is equal to3.4E-11 µCi/cc.
Two hundred times the indicated release rate is equal to 3.4E-9 µCi/cc.

1.2.5 Relationship to EPA PAG

To compare the release rates calculated above the following assumptions were made:

Am-241 dose conversion factor in rem / cm^{-3} µCi hr, from EPA 400 = 5.3E8

Since no credit is taken for an elevated release point or an annual average χ/Q the same atmospheric dispersion factor is used in the calculations below.

Assuming that an unplanned release occurs at two times the ODCM release rate for one hour the total activity released is equal to:

3.4E-11 µCi/cc * 2000 ft^3/min * 28,317 cc/ft^3 * 60 min = 1.16E-1 µCi

(1.16E-1 µCi) * (5.3E8 rem / cm^{-3} uCi hr) * (6.59E-3 sec/m^3) / (1E6 cm^3/m^3) / (3600 sec/hour) = 1.13E-4 rem

This is much less than the EPA PAG of 1 Rem

Assuming that an unplanned release occurs at two hundred times the ODCM release rate for 15 minutes the total activity released is equal to:

3.4E-9 µCi/cc * 2000 ft^3/min * 28,317 cc/ft^3 * 15 min = 2.89E0 µCi

This results in a dose of:

(2.89E0 µCi) * (5.3E8 rem / cm^{-3} uCi hr) * (6.59E-3 sec/m^3) / (1E6 cm^3/m^3) / (3600 sec/hour) =

2.80E-3 rem

This is much less than the EPA PAG of 1 Rem.

1.2.6 Relationship to 10CFR20 Appendix B Table 2 Effluent Concentration limits

The 10CFR20 Appendix B Table 2 Effluent Concentration limit for Am-241 is 2E-14 µCi/ml.

The average annual ground-level concentration in air (x_i) in pCi/m^3 is equal to:

x_i = (3.17E4 (pCi/Ci)/ (sec/year)) * Q * (X/Q)

Where Q is equal to the quantity of radioactive material released in a year in Curies/year

ODCM Modular HEPA Unit incidental release X/Q = 6.59E-3 sec/ m^3

If x_i = 2E-14 µCi/ml then:

Q = (2E-14 µCi/ml * 1E6 ml/m^3 * 1E6 pCi/µCi) / ((3.17E4 (pCi/Ci)/ (sec/yr)*(6.59E-3 sec/ m^3))

Q = 9.57E-5 Ci/yr

The average annual Modular HEPA Unit volume based on the ODCM is 2.98E13 cc/yr.

This is based on (2000 cfm * 525,600 minutes/yr * 28,317 cc/cfm).

Therefore, the Modular HEPA Unit effluent concentration required to result in a fence-line concentration of 2E-14 µCi/ml is:

(9.57E-5 Ci/yr * 1E6 µCi/Ci) / (2.98E13 cc/yr * 1 cc/ml) = 3.2E-12 µCi/ml

1.2.7 Conversion Factor from Effluent Concentration to µCi/day

The release rate in µCi/day = Modular HEPA Unit concentration in µCi/cc * 2000 ft^3/min * 1440 minutes/day * 28317 cc/ ft^3

The release rate in µCi/day = Modular HEPA Unit concentration in µCi/cc * 8.16E10 cc/day

1.2.8 Conversion Factor from µCi/day to % of NUE
An NUE is equal to a release rate of 3000 mrem/year

%NUE = (Offsite dose rate / NUE threshold) * 100

%NUE = ((Conversion Factor * Release Rate) / NUE threshold) * 100

%NUE = ((Conversion Factor * 100) / NUE threshold) * Release Rate

The Conversion Factor is equal to (1.77E6 mrem/µCi) * (6.59E-3 sec/ m^3) * (8000 m^3/year) / (8.64E4 sec/day)

This is equal to1.08E3 mrem/year per µCi/day

1.2.9 Results

The 10CFR20 Appendix B Table 2 Effluent Concentration limit for Am-241 is 2E-14 µCi/ml. The Modular HEPA Unit effluent concentration that would result in a fence-line concentration of 2E-14 µCi/ml is 3.2E-12 µCi/ml.

3.2E-12 uCi/ml * 8.16E10 cc/day * 1ml/cc * 1.08E3 mrem-day/uCi-yr = 4.70E2 mrem/yr.
470 mrem/yr / 8760 hr/yr = 5.365E-2 mrem/hr

Assuming that an unplanned release occurs at two times the ODCM release rate for one hour the offsite dose corresponding to an NUE would be 1.07E-4 rem (0.107 mrem) which is much less than the EPA PAG.

Assuming that an unplanned release occurs at two hundred times the ODCM release rate for fifteen minutes the offsite dose corresponding to an Alert would be 2.675E-3 rem (2.7 mrem) which is much less than the EPA PAG.

Note that Am-241 is used in the example calculations and is expected to be limiting. Other alpha emitting isotopes such as Pu-238, Pu-239/240 and Cm-243/244 are evident in the contamination at HBPP. Since the Effluent Concentration Limits (ECLs), Derived Air Concentration (DAC) values and organ Dose Conversion Factors (DCFs) are similar, the Am-241 values may be assumed to be gross alpha with appropriate compensation for naturally occurring isotopes.

Other radionuclides (Co-60, Sr-90, Cs-137, etc.) are important in determining actual offsite dose and in demonstrating compliance with the ECL using the sum of the fractions rule. The example calculations are used similarly for each isotope in the mix with their respective ECL, DCF and exposure pathway (inhalation, ingestion, and submersion).

Although not relevant to the hypothetical offsite dose calculation in the ECL and NUE analysis above, assumed effluent concentrations are approximately 1 DAC, 2 DAC, and 200 DAC for Am-241 at the point of release. Airborne radioactivity control measures to control worker dose, also limits the potential offsite dose.

2.0 LIQUID EFFLUENT DOSE CALCULATIONS

2.1 MONTH (31 DAY PERIOD) Deleted

2.2 CALENDAR QUARTER - Deleted

2.3 CALENDAR YEAR - Deleted

2.4 LIQUID EFFLUENT DOSE CALCULATION METHODOLOGY

As of December 31, 2013, HBPP has ceased liquid radioactive effluent discharges via the discharge canal to Humboldt Bay. Any remaining processed liquid radioactive waste is transported offsite for land disposal at an authorized disposal facility. The following calculation methodology is preserved as a part of the ODCM for ease of reference to site specific parameters in the event of an accidental release of liquid radioactive effluent. No recurring liquid effluent dose calculations are expected for the remainder of decommissioning.

The equations specified in this section 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.

Equation (2) of Regulatory Guide 1.109 provides for the use of a site specific mixing ratio (i.e. reciprocal of the dilution factor) that describes the near term and near field mixing of the tidal flow from the Discharge Canal into Humboldt Bay. A two-dimensional numerical analysis, depth-averaged, finite element hydrodynamic model (reference 12.1) was developed by CH2MHILL and used to estimate the dispersion of the canal discharge in the Bay. The analysis indicated that an additional dilution factor of 80 for batch release applications or a dilution factor of 20 for continuous release applications can conservatively be used to describe the Bay dilution. A factor of 20 will be applied in this calculation to address any combination of release modes.

Since the intake canal contains a larger volume of water, use of the above dilution factors for effluent releases to the intake canal provides a simplified, conservative methodology for calculating annual dose from effluent releases to the intake canal.

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:

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

where:

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

$C_{i\text{-Bay diluted}}$ = The average diluted Bay concentration, pico-Curie/liter, for radionuclide, i. If the outfall to the canal is at the furthest most portion of the canal from the entrance to the Bay, this will be estimated by calculating the total activity released (e.g. effluent concentration $C_{i\text{ effluent}}$ in pCi/L times the discharge volume V_D in Liters) then dividing the total activity of the nuclide discharged during the period, pico-Curies, by the dilution volume (e.g. total discharged volume V_D plus total tidal flow V_{TD} during the period in liters), and by the Bay dilution factor of 20. The total annual tidal flow for the outfall canal is 2.47E+9 Liters/year (e.g., 6.77E+6 Liters/day). If Gross Alpha radioactivity is determined to be in the effluent , Pu-241 will be considered to be present at 3.25 times the amount of detected Gross Alpha radioactivity. Note that the resulting dose commitment is the annual dose rate (mrem per year) for a time frame with this average concentration. Doses (NOT dose rates) for periods shorter than a year must be proportionately reduced.

$$C_{i\text{ - Bay diluted}} = \frac{C_{i\text{ - Effluent}} \times V_D}{(V_D + V_{TD}) \times 20} \quad (2\text{-}2)$$

If the outfall is not located in the furthest most portion of the canal from the entrance to the Bay, no credit for tidal dilution of the canal will be taken and the diluted Bay concentration will be calculated using the following equation.

$$C_{i\text{ - Bay diluted}} = \frac{C_{i\text{ - Effluent}}}{20} \quad (2\text{-}3)$$

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.

Dose calculations can be performed using the above methodology for the current month, quarter, or year.

Table 2-1
Ingestion Dose Factors for Adult Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from NUREG/CR-4013 (LADTAP II input values)

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	5.99×10^{-8}	5.99×10^{-8}	5.99×10^{-8}	5.99×10^{-8}	5.99×10^{-8}
Co-60	No Data	2.14×10^{-6}	4.72×10^{-6}	No Data	No Data	4.02×10^{-5}
Ni-63	1.30×10^{-4}	9.01×10^{-6}	4.36×10^{-6}	No Data	No Data	1.88×10^{-6}
Sr-90	8.71×10^{-3}	No Data	1.75×10^{-4}	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}
Pu-241	1.57×10^{-5}	7.45×10^{-7}	3.32×10^{-7}	1.53×10^{-6}	No Data	1.40×10^{-6}
Am-241	7.55×10^{-4}	7.05×10^{-4}	5.41×10^{-5}	4.07×10^{-4}	No Data	7.42×10^{-5}
Gross α	7.55×10^{-4}	7.05×10^{-4}	5.41×10^{-5}	4.07×10^{-4}	No Data	7.42×10^{-5}

Table 2-2
Ingestion Dose Factors for Teen Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from NUREG/CR-4013 (LADTAP II input values)

Nuclide	Organ					
	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	6.04×10^{-8}	6.04×10^{-8}	6.04×10^{-8}	6.04×10^{-8}	6.04×10^{-8}
Co-60	No Data	2.81×10^{-6}	6.33×10^{-6}	No Data	No Data	3.66×10^{-5}
Ni-63	1.77×10^{-4}	1.25×10^{-5}	6.00×10^{-6}	No Data	No Data	1.99×10^{-6}
Sr-90	1.02×10^{-2}	No Data	2.04×10^{-4}	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}
Pu-241	1.75×10^{-5}	8.40×10^{-7}	3.69×10^{-7}	1.71×10^{-6}	No Data	1.48×10^{-6}
Am-241	7.98×10^{-4}	7.53×10^{-4}	5.75×10^{-5}	4.31×10^{-4}	No Data	7.87×10^{-5}
Gross α	7.98×10^{-4}	7.53×10^{-4}	5.75×10^{-5}	4.31×10^{-4}	No Data	7.87×10^{-5}

Table 2-3
Ingestion Dose Factors for Child Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from NUREG/CR-4013 (ladTAP II input values)

	Organ					
Nuclide	Bone	Liver	Total Body	Kidney	Lung	GI-LLI
H-3	No Data	1.16×10^{-7}	1.16×10^{-7}	1.16×10^{-7}	1.16×10^{-7}	1.16×10^{-7}
Co-60	No Data	5.29×10^{-6}	1.56×10^{-5}	No Data	No Data	2.93×10^{-5}
Ni-63	5.38×10^{-4}	2.88×10^{-5}	1.83×10^{-5}	No Data	No Data	1.94×10^{-6}
Sr-90	2.56×10^{-2}	No Data	5.15×10^{-4}	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}
Pu-241	3.87×10^{-5}	1.58×10^{-6}	8.04×10^{-7}	2.96×10^{-6}	No Data	1.44×10^{-6}
Am-241	1.36×10^{-3}	1.17×10^{-3}	1.02×10^{-4}	6.23×10^{-4}	No Data	7.64×10^{-5}
Gross α	1.36×10^{-3}	1.17×10^{-3}	1.02×10^{-4}	6.23×10^{-4}	No Data	7.64×10^{-5}

Table 2-4
Bioaccumulation Factors for Saltwater Environment
(pCi/kg per pCi/liter)
Selected Nuclides from Regulatory Guide 1.109, Table A-1 and from NUREG/CR-4013

Element	Fish	Invertebrate
H	9.0×10^{-1}	9.3×10^{-1}
Co	1.0×10^{2}	1.0×10^{3}
Ni	1.0×10^{2}	2.5×10^{2}
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}
Pu	3.0	2.0×10^{2}
Am	2.5×10^{1}	1.0×10^{3}
Gross α	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 EFFLUENT TREATMENT

3.1 TREATMENT REQUIREMENTS

3.1.1 Deleted

3.1.2 Deleted

3.2 Deleted

4.0 GASEOUS EFFLUENT DOSE CALCULATIONS

4.1 DOSE RATE

4.1.1 Deleted

As explained in Specification Bases 3.7, Noble Gases are not required to be monitored, and the corresponding dose rate need not be calculated.

4.1.2 Tritium and Radioactive Particulates

There are no short-lived radioactive particulates in the effluent, so radioactive decay can be neglected. Meteorological parameters are assumed to be constant, and applied for the most conservative location. Therefore, the radioactive particulates dose rate calculation methodology is the same as the radioactive particulates dose calculation methodology. Refer to sections 4.3.3 through 4.3.8 for the appropriate equations.

As explained in Specification Bases 3.5, Tritium is not required to be monitored, and the corresponding dose rate need not be calculated. Nevertheless, if such a calculation is required, refer to sections 4.3.9 through 4.3.13 for the appropriate equations.

4.2 Deleted

4.3 DOSE - TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

4.3.1 Calendar Quarter

The methodology for calendar quarter calculations is the same as for the calendar year calculations provided by section 4.3.3, and discussed in section 4.3.2, with the exception that the resulting values for D (annual dose commitment, mrem/year) must be divided by 4 to convert them to quarterly dose commitment, mrem/quarter.

4.3.2 Calendar Year

The methods for calculating the dose due to 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.

The equations provided for determining the doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

4.3.3 Particulate Organ Dose Calculation Summation Methodology

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.

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. Radioactive decay may result in the dose from Transuranic radionuclides becoming significant. If Gross Alpha radioactivity is determined to be released, Pu-241 will be considered to be present at 3.25 times the amount of detected Gross Alpha radioactivity. The annual dose commitment will be calculated for any organ of an individual age group as follows:

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

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 Decommissioning releases range from about 0.002 mrem/year (fruit/vegetable consumption pathway) to less than 1 x 10^{-6} mrem/year (for direct radiation exposure from material deposited on the ground).

4.3.4 Particulate Inhalation Pathway Dose Calculation Methodology

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

where:

χ/Q = The atmospheric dispersion parameter, seconds/cubic meter.
= 1.0 x 10^{-5} seconds/cubic meter for releases from the 50 foot stack. Refer to Appendix B, 1.2.
= 6.59 x 10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

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. No values are provided for Transuranic radionuclides, as their dose contribution to this pathway is negligible.

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, 1.3.
= 5.39×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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

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

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, 1.3.
= 3.29×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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

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

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.

F_f = 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, 1.3.
= 3.29×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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

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

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, 1.3.
= 3.29×10^{-6} inverse square meters for releases other than from the 50 foot stack. Refer to Appendix B, 2.2.

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:

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

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. Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

4.3.10 Tritium Inhalation Pathway Dose Calculation Methodology

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

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, 1.2.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

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.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

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, 1.2.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

4.3.12 Tritium Grass-Cow-Meat Pathway Dose Calculation Methodology

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

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, 1.2.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

4.3.13 Tritium Vegetation Pathway Dose Calculation Methodology

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

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

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, 1.2.
= 6.59×10^{-3} seconds per cubic meter for releases other than from the 50 foot stack. Refer to Appendix B, 2.1.

Once the spent fuel pool is emptied, this source term and exposure pathway is no longer evaluated.

Table 4-1
Inhalation Dose Factors for Adult Age Group
(mrem/pico-Curie inhaled)
Selected Nuclides from Regulatory Guide 1.109, Table E-7 and from NUREG/CR-4013

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}
Pu-241	3.42×10^{-2}	8.69×10^{-3}	1.29×10^{-3}	5.93×10^{-3}	1.52×10^{-4}	8.65×10^{-7}
Gross α	1.68	1.13	7.75×10^{-2}	5.04×10^{-1}	1.82×10^{-1}	4.84×10^{-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 and from NUREG/CR-4013

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}
Pu-241	3.74×10^{-2}	9.56×10^{-3}	1.40×10^{-3}	6.47×10^{-3}	2.60×10^{-4}	9.17×10^{-7}
Gross α	1.77	1.20	8.05×10^{-2}	5.32×10^{-1}	3.12×10^{-1}	5.13×10^{-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 and from NUREG/CR-4013

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}
Pu-241	7.94×10^{-2}	1.75×10^{-2}	2.93×10^{-3}	1.10×10^{-2}	5.06×10^{-4}	8.90×10^{-7}
Gross α	2.97	1.84	1.28×10^{-1}	7.63×10^{-1}	6.08×10^{-1}	4.98×10^{-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 and from NUREG/CR-4013

Nuclide	Organ					
	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}
Pu-241	8.43×10^{-2}	1.85×10^{-2}	3.11×10^{-3}	1.15×10^{-2}	7.62×10^{-4}	8.97×10^{-7}
Gross α	3.15	1.95	1.34×10^{-1}	7.94×10^{-1}	9.03×10^{-1}	5.02×10^{-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

Nuclide	Total	
	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}

Values are not provided for Transuranic radionuclides, as their dose contribution to this pathway is negligible.

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 and from NUREG/CR-4013

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}
Pu-241	1.57×10^{-5}	7.45×10^{-7}	3.32×10^{-7}	1.53×10^{-6}	No Data	1.40×10^{-6}
Gross α	7.55×10^{-4}	7.05×10^{-4}	5.41×10^{-5}	4.07×10^{-4}	No Data	7.81×10^{-5}

Table 4-9
Ingestion Dose Factors for Teen Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-12 and from NUREG/CR-4013

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}
Pu-241	1.75×10^{-5}	8.40×10^{-7}	3.69×10^{-7}	1.71×10^{-6}	No Data	1.48×10^{-6}
Gross α	7.98×10^{-4}	7.53×10^{-4}	5.75×10^{-5}	4.31×10^{-4}	No Data	8.28×10^{-5}

Table 4-10
Ingestion Dose Factors for Child Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-13 and from NUREG/CR-4013

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}
Pu-241	3.87×10^{-5}	1.58×10^{-6}	8.04×10^{-7}	2.96×10^{-6}	No Data	1.44×10^{-6}
Gross α	1.36×10^{-3}	1.17×10^{-3}	1.02×10^{-4}	6.23×10^{-4}	No Data	8.03×10^{-5}

Table 4-11
Ingestion Dose Factors for Infant Age Group
(mrem/pico-Curie ingested)
Selected Nuclides from Regulatory Guide 1.109, Table E-14 and from NUREG/CR-4013

Nuclide	Organ					
	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}
Pu-241	4.25×10^{-5}	1.76×10^{-6}	8.82×10^{-7}	3.17×10^{-6}	No Data	1.45×10^{-6}
Gross α	1.46×10^{-3}	1.27×10^{-3}	1.09×10^{-4}	6.55×10^{-4}	No Data	8.10×10^{-5}

Table 4-12
Stable Element Transfer Data For Cow-Milk Pathway
(days/liter)
Selected Nuclides from Regulatory Guide 1.109, Table E-1 and from NUREG/CR-4013

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}
Pu	5.0×10^{-6}
Gross α	5.0×10^{-6}

Table 4-13
Stable Element Transfer Data For Cow-Meat Pathway
(days/kilo-gram)
Selected Nuclides from Regulatory Guide 1.109, Table E-1 and from NUREG/CR-4013

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}
Pu	2.0×10^{-4}
Gross α	2.0×10^{-4}

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. Deleted

b. Modular HEPA Ventilation Particulate releases, using equation (4-3).

c. Deleted. Tritium is no longer a gaseous effluent source term.

d. Liquid releases, No longer applicable.

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. Deleted

b. Modular HEPA Ventilation releases, using equation (4-3). Tritium is no longer a gaseous effluent source term.

c. Liquid releases, No longer applicable.

d. The potential direct radiation exposure to an individual at the site boundary based 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. Deleted

b. Modular HEPA Ventilation releases, using equation (4-3).

c. Liquid releases, No longer applicable.

d. The potential direct radiation exposure to an individual at the site boundary based 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

Deleted - Based on the status of decommissioning, HBPP no longer anticipates wastes exceeding a specific activity that is unacceptable to disposal site without solidification or exceeding Class A as defined in 10 CFR 61.

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

Deleted - HBPP no longer anticipates wastes exceeding a specific activity that is unacceptable to disposal site without solidification or exceeding Class A as defined in 10 CFR 61. HBPP no longer anticipates disposal of wastes requiring stabilization in a High Integrity Container (HIC).

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, filters and other wet wastes shipped for land burial which contain a total specific activity less than the disposal site(s) 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 disposal site(s) (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.

8.2.4 Contract vendor solidification or dewatering services are utilized in accordance with PG&E approved supplier list and procurement procedures.

8.2.5 Vendor services may be conducted off site in accordance with their facility license and procedures. Vendor services include written confirmation of acceptable disposal waste form.

8.2.6 Gross dewatering of resins and filters may be performed onsite to achieve transport requirements in preparation for additional processing to a final waste form by offsite vendor services.

8.2.7 On site activities, such as managing wet soils from decommissioning excavations and process water shall be performed utilizing approved procedures or work instructions to ensure compliance with transportation regulations, disposal facility license requirements and/or waste acceptance criteria.

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 Decommissioning period. All changes shall be reviewed and approved by the HB Director 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.

9.3 HBPP is allowed to modify or reduce environmental requirements in the ODCM provided HBPP considers the modification or reduction from a technical and decommissioning perspective. [CMT 10.1]

10.0 COMMITMENTS

10.1 HBPP does not intend to modify or reduce the environmental monitoring requirements as specified in the ODCM during the period of SAFSTOR and decommissioning activities. This applies to those environmental samples and analysis identified as either quality or non-quality samples. This commitment is to be incorporated into the next revision of the ODCM. NOTE: HBPP is allowed to modify or reduce environmental requirements in the ODCM provided HBPP considers the modification or reduction from a technical and decommissioning perspective.

11.0 RESPONSIBLE ORGANIZATION

Radiation Protection Manager

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. As of December 31, 2013, HBPP ceased processed liquid effluent to the discharge canal and processed liquid effluent will be transported for disposal at a regulated disposal site. The Ground Water Treatment System (GWTS) was removed from service in April 2019.

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	8.60E-2	7.17E-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

Table A-3 below reflects the Gaseous Effluent Activity as a representation of the state of decommissioning during the calendar year 2013 relative to the Baseline above.

Table A-3
2013 Gaseous Effluent Activity

Type of Activity	Annual Release (Curies)	Monthly Average Release (Curies)
Particulate Gamma Emitters (total)	<1.5E-5	<1.3E-6
Strontium-90	<1E-6	<1E-7
Particulate Alpha Emitters (total)	<1E-6	<1E-7

Table A-3 data is summarized from the 2013 Annual Effluent Release Report and are listed as less than values because sampling results were the composite of LLD values. Tritium is no longer monitored due to a lack of significant source term.

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:

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

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

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

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 = 4.70×10^{-3} seconds/meter^3 (1)

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

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

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 x 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 x 10^{-4} $meter^{-1}$)/ (6 x 6.07 meter) = 3.29 x 10^{-6} $meter^{-2}$.

APPENDIX C

Deleted