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Annual Radiological Environmental Monitoring Report for 2004

This letter transmits Portland General Electric Company's Trojan Nuclear Plant Annual Radiological Environmental Monitoring Report for the Calendar Year 2004. This report is submitted in accordance with PGE-1021, "Offsite Dose Calculation Manual (ODCM)," and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to Title 10 CFR 50. A copy of PGE-1021 is attached to the report as Appendix A.

Sincerely,

Stephen M. Quennoz
Vice President, Generation

Enclosure

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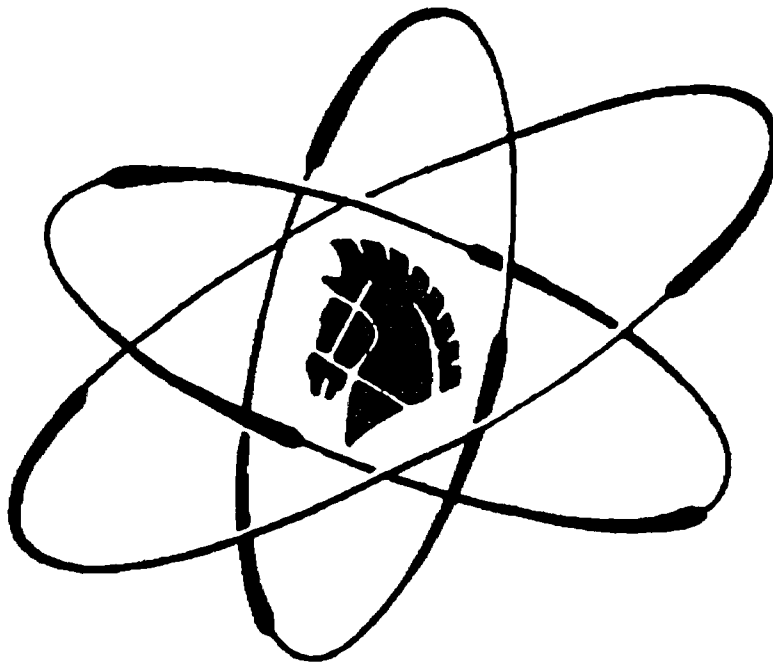
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Trojan Nuclear Plant

Radiological Environmental Monitoring Report 2004

PORTLAND GENERAL ELECTRIC COMPANY



TROJAN NUCLEAR PLANT
RADIOLOGICAL ENVIRONMENTAL
MONITORING REPORT

January through December 2004

February 2005

Prepared by

PORTLAND GENERAL ELECTRIC COMPANY

With Analyses By
Eberline Services
ALBUQUERQUE, NEW MEXICO

TROJAN NUCLEAR PLANT
RADIOLOGICAL ENVIRONMENTAL
MONITORING REPORT

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ABSTRACT

This report presents the data obtained through the analyses of environmental samples collected through the Portland General Electric Trojan Nuclear Plant Radiological Environmental Monitoring Program for the period January 1, 2004, through December 31, 2004.

Most of the radionuclide analyses on the environmental samples resulted in non-detectable values for radionuclides that could be released from the Trojan Nuclear Plant. In no case did radioactivity that could be attributed to the Trojan Nuclear Plant exceed the Reporting Levels of the Offsite Dose Calculation Manual (ODCM) for Trojan.

1.0 INTRODUCTION

The Trojan Nuclear Plant, an 1130 megawatt-electric pressurized water reactor, first achieved criticality on December 15, 1975. On January 27, 1993, Portland General Electric notified the Nuclear Regulatory Commission of their decision to permanently shut down the Trojan Nuclear Plant. This report presents the analytical data from the Radiological Environmental Monitoring Program with appropriate interpretation for 2004.

The analytical contractor during this period has been Eberline Services, Albuquerque, New Mexico. In comparing data obtained during this period with those from previous periods, care should be taken to ensure that differences in procedures among the contractors are considered.

Information concerning the Radiological Environmental Monitoring Program prior to this period may be found in earlier reports.

2.0 SAMPLING AND PROGRAM PROCEDURES

2.1 SAMPLING LOCATIONS

Thirteen (13) sampling locations were used in the Radiological Environmental Monitoring Program from January 1, 2004, through December 31, 2004. These sampling locations are shown in Figure 2-1. Table 2-1 includes a listing of the sites, their distance from Trojan, and the type and frequency of sample collection.

During 1994 a review of the environmental sample results from 1977 through 1993 was conducted. In general, the review confirmed that radioactivity attributable to Trojan Nuclear Plant during power operations was not detected in the environmental samples. Therefore, since the production of radioactivity had ceased when the reactor was permanently shut down, and from that point forward, the radioactivity in both liquid and gaseous effluents continued to decrease, it was evident that the environmental sampling requirements could be reduced. Therefore, revisions to the Radiological Environmental Monitoring Program were submitted to the Oregon Department of Energy on September 22, 1994, for approval. The revisions to the program were approved on December 12, 1994.

During 2002 a review of the environmental sample results from 1993 through 2001 was conducted. The review confirmed that sample radioactivity concentrations are at environmental background levels. Radiological effluent release data for the same period showed that the radioactivity in liquid and gaseous effluents has consistently remained at low levels (relative to pre-shutdown plant operations). Based on these facts, further reductions in environmental sampling requirements were discussed with the Oregon State Health Division. Revisions to the Radiological Environmental Monitoring Program to implement these reductions were submitted to the Oregon Office of Energy on June 19, 2002, for approval. The revisions to the program were approved on August 1, 2002, and implemented shortly thereafter.

2.2 SAMPLING PROCEDURES

2.2.1 AMBIENT RADIATION MEASUREMENTS USING TLDs

Thermoluminescent dosimeters (TLDs) are placed for field exposure and collected on a quarterly frequency. The TLDs are placed about one meter above ground level in plastic containers. A single TLD badge is placed in each container. The time of collection, the exposure period, and any abnormal conditions such as moisture in the holders, damage done by animals, etc., are recorded when the TLDs are retrieved. Care is taken to minimize exposure to the TLDs between collection and delivery to the laboratory. Trip TLDs are carried with the field TLDs during transport to and from the field.

2.2.2 DRINKING WATER

Four-week composite samples of municipal drinking water are collected for Rainier at its intake structure on the Columbia River. Rainier is downstream of the Trojan Nuclear Plant. A compositing sampler takes a sample every two hours and aliquots of this four-week composite are sent for analysis. From these aliquots, 60 milliliters or more are sent for tritium analysis and two one-gallon polyethylene bottles are acidified with concentrated HCL and sent for the other analyses. The bottles are securely sealed and labeled, and collection data forms are prepared specifying site, date collected, volume, and sample type.

2.2.3 SHORELINE SOIL

Shoreline soil samples of about one quart in volume are taken twice a year. The samples are taken from a one square foot area at a depth of between one and four inches. Vegetation and large rocks are removed from the sample before it is placed in a plastic container. The containers are securely sealed and labeled. The sample site identification number, date collected, and volume obtained are recorded on the collection data forms.

TABLE 2-1

SAMPLING LOCATIONS AND FREQUENCY BY TYPE

Sample Location	Radial		Sample		
	Distance (meters)	Direction	TLD	Surface Water	Shore Soil
1 - Trojan North Building	300	WNW	Q	MC	S/A
2 - NW Fenceline	210	NW	Q		
3 - N Fenceline	191	N	Q		
4 - Switchyard	191	WSW	Q		
5 - Training Building	354	SW	Q		
6 - Park Entrance	354	SSW	Q		
7 - South End Cooling Tower	640	SE	Q		
8 - Rainier	6,115	NW	Q		
9 - St. Helens (Municipal Water Supply)	16,898	SSE	Q		
10- Columbia River	116,510*	E			
13- N Site Boundary at Columbia River	800	NNW	Q		
14- S Site Boundary	1,332	S	Q		
15- E Fenceline	93	E	Q		

LEGEND:

MC Monthly Composite.

Q Quarterly.

S/A Semi-Annually.

* Columbia River Distance refers to meters measured from mouth.

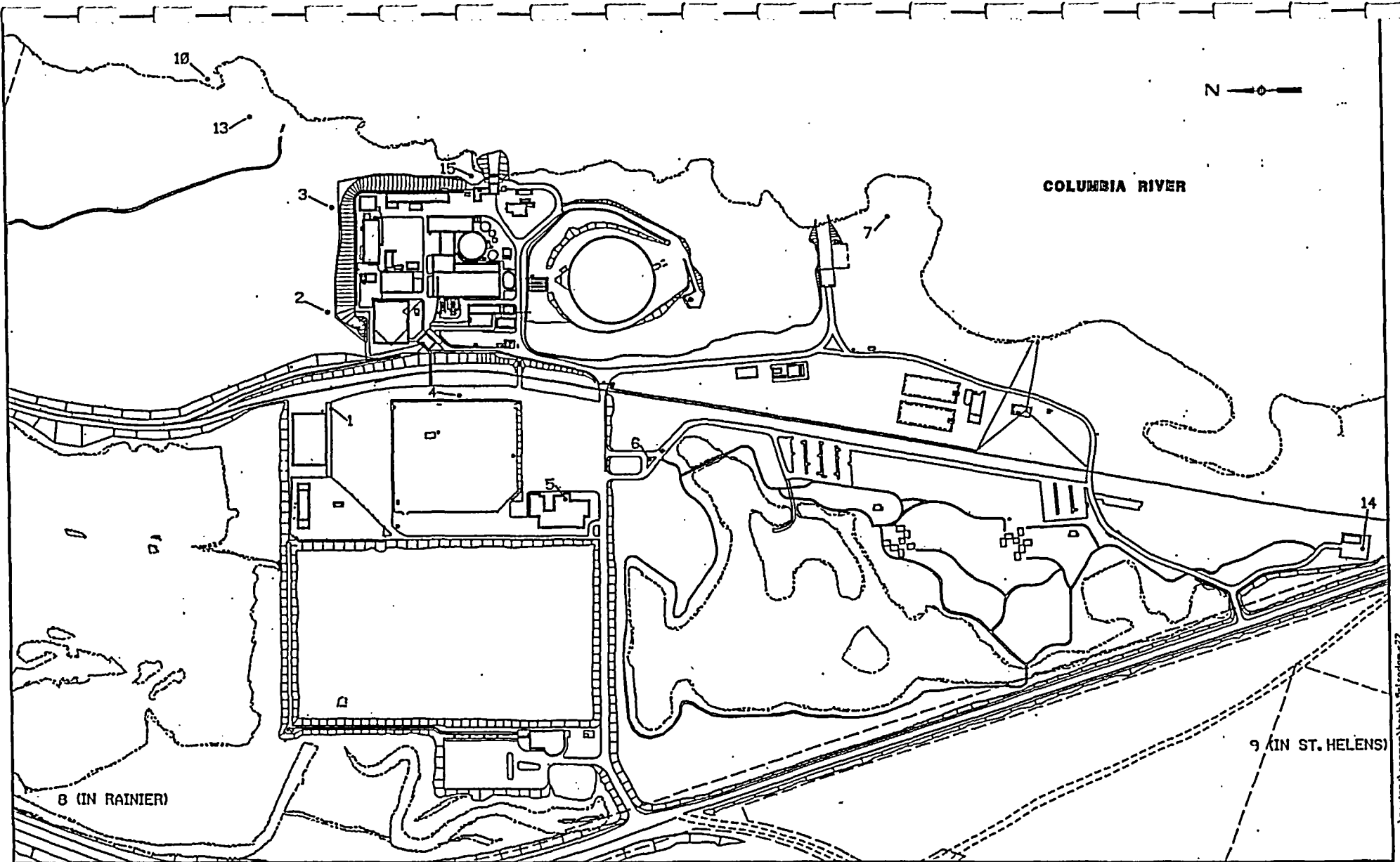


FIGURE 2-1

SAMPLING LOCATIONS

3.0 ANALYTICAL PROCEDURES AND COUNTING METHODS

Samples are analyzed for the various radioactive components by standard radiochemical methods. These methods are equal to, and in most cases, identical with, those of the U.S. Department of Energy [Health and Safety Laboratory (HASL) Procedures Manual, HASL-300, see references, Section 3.6].

Analyses of individual sample types, general methods, and routine analytical sensitivities are discussed below. The analytical program and sensitivity requirements are given in Table 3-1.

3.1 ANALYTICAL DETECTION LIMITS AND UNCERTAINTY

In environmental radiological analyses the dominant known uncertainty is usually the sample count rate. This uncertainty is calculated by standard methods (HASL-300), and is reported at the 95 percent confidence level (2σ). The lower limit of detection (LLD) is defined as the smallest concentration of radioactive material in a sample that will yield a net indication, above system background, that will be detected with 95 percent probability with only five percent probability of falsely concluding that a blank observation represents a real signal. Analytical data for samples for which concentrations are less than or equal to the LLD are preceded by the symbol "<" unless otherwise specified.

3.2 DRINKING WATER

Gross beta analysis of water samples is performed by evaporation of a measured aliquot of the sample, digestion, planchetting of the processed sample and radiometric assay by low-background beta counters, with an LLD of 1 pCi/liter. Tritium analysis is performed on water samples to the required LLD of 1000 pCi/liter by liquid scintillation counting. Gamma isotopic analysis is performed using germanium detectors. The LLD requirements for gamma scans are given in Table 3-1.

3.3 SHORELINE SOIL

Samples are oven-dried and results reported based on dry weight. Gamma emitters are measured with germanium detectors. The LLD requirements for gamma scans are given in Table 3-1.

3.4 AMBIENT RADIATION MEASUREMENTS

Quarterly ambient gamma radiation measurements are made using TLDs supplied by a vendor. Each environmental dosimeter consists of four elements: 2CaF₂ and 2LiF TLD elements with a photon energy response range of 40 to 1200 keV.

Environmental dosimeters retrieved from the field are sent to the vendor for processing on a quarterly basis.

3.5 QUALITY CONTROL

A large number of the analyses performed by the analysis laboratory are for quality control purposes. The analysis laboratory participates in the Department of Energy (DOE) interlaboratory comparison program for environmental measurements. Reports of quality control analyses are presented semi-annually. Results of the DOE interlaboratory comparisons for 2004 are given in Table 3-2. Only the results for those types of analyses performed for PGE are included. In those cases where the laboratory fails the performance evaluation study, the laboratory performs an investigation to determine the cause and corrective action as required. Table 3-3 summarizes the environmental duplicates results performed by the analysis laboratory for the year 2004.

3.6 REFERENCES FOR ANALYTICAL PROCEDURES

1. American Public Health Association, American Water Works Association and Water Pollution Control Federation (1971): Standard Methods for the Examination of Water and Wastewater. Thirteenth edition, pp 583-632; 12th edition, pp 325-352. APHA, 1740 Broadway, New York, NY 10019.
2. Department of Health, Education and Welfare, Public Health Service: Radioassay Procedures for Environmental Samples. National Center for Radiological Health (1967), Sec. 1, pp 36-115.
3. Atomic Energy Commission: Regulatory Guide 4.3 (September 1973).
4. Health and Safety Laboratory, Atomic Energy Commission: HASL Procedures Manual (now known as EML of the Department of Energy). HASL, 376 Hudson Street, New York, NY 10014.

TABLE 3-1

PROGRAM ANALYSES AND LOWER LIMIT OF DETECTION

<u>Program Analysis</u>	<u>Lower Limit of Detection (LLD)^[a]</u>
Water-gross beta	1 pCi/liter
Water-tritium	1000 pCi/liter
Water-gamma scan	15 pCi/liter Co-60 15 pCi/liter Cs-134 18 pCi/liter Cs-137
Shoreline Soil-gamma scan (dry)	0.15 pCi/g Cs-134 0.18 pCi/g Cs-137
Direct Radiation	0.04 mR/day or less ^[b]

^[a] LLD is defined in Section 3.1

^[b] Minimum reportable exposure rate, including background, is 10 mR/qtr.

TABLE 3-2

DOE INTERLABORATORY COMPARISON PROGRAM RESULTS

June 2004 Results

<u>Soil</u> <u>Analysis</u>	<u>DOE</u> <u>Value (Bq/Kg)</u>	<u>Eberline</u> <u>Value (Bq/Kg)</u>	<u>Ratio</u>	<u>Evaluation</u>
Cs-137	1323.000	1140.000	0.862	Pass
<u>Water</u> <u>Analysis</u>	<u>DOE</u> <u>Value (Bq/L)</u>	<u>Eberline</u> <u>Value (Bq/L)</u>	<u>Ratio</u>	<u>Evaluation</u>
Gross Beta	1170.000	1170.000	1.000	Pass
H-3	186.600	197.000	1.056	Pass
Co-60	163.200	142.000	0.870	Pass
Cs-137	51.950	45.000	0.866	Pass

December 2004 Results

The DOE Inter laboratory Comparison Program was terminated following June 2004.

TABLE 3-3

QUALITY CONTROL ANALYSES SUMMARY

The table below summarizes results of samples run for process quality control purposes during the subject year. Only the results for those types of analysis performed for PGE are included. These listings are in addition to such measurements as detector backgrounds, check source values, radiometric-gravimetric comparisons, system calibrations, etc. Detailed listings of each measurement are maintained at the analysis laboratory and are available for inspection if required.

Environmental Duplicates
(number processed/relative error ratio >3)

Sample Matrix	Nuclide Analyzed	Month												Totals
		Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	
Water														
	Gross Beta	--	--	--	7/0	8/0	3/0	--	3/1	6/0	8/1	4/0	3/0	42/2
	Tritium	2/0	4/0	1/0	2/0	3/0	3/0	5/0	--	4/0	3/0	3/0	4/0	34/0
	Co-60	14/0	6/0	17/0	5/0	5/0	--	--	--	1/0	1/0	1/0	1/0	51/0
	Cs-134	5/0	--	12/0	5/0	5/0	5/0	21/0	--	--	--	--	1/0	54/0
	Cs-137	14/0	6/0	17/0	5/0	5/0	5/0	21/0	5/0	1/0	1/0	2/0	1/0	83/0
Soil	Co-60	18/0	8/0	--	--	--	--	--	--	--	--	--	--	26/0
	Cs-134	--	--	4/0	--	--	--	--	--	--	1/0	--	--	5/0
	Cs-137	18/0	8/0	4/0	--	--	--	--	--	1/0	1/0	--	--	32/0

$$\text{Relative Error Ratio} = \frac{|x_{\text{orig}} - x_{\text{dup}}|}{\sqrt{\sigma_{\text{orig}}^2 + \sigma_{\text{dup}}^2}}$$

4.0 RESULTS AND DISCUSSION

4.1 SAMPLES FROM THE TERRESTRIAL ENVIRONMENT

4.1.1 AMBIENT RADIATION LEVELS

Gamma radiation levels (mR/day) for dosimeter measurements at locations in the environs around the Trojan Nuclear Plant during 2004 are shown in Chapter 5, Table 5-1. All of the dosimeter measurements obtained within the Controlled Area showed no increase in ambient radiation levels, with the exception of locations directly affected by shine from loaded Casks in the Trojan Independent Spent Fuel Storage Installation (ISFSI). Data from these locations are listed but not included in the statistical data summary.

Trojan onsite measurements during 2004 were less than, but not significantly different from, the control locations. Average gamma radiation levels for the years prior to, and including, 2004 are presented in Table 4-1 for both onsite and control locations.

4.2 SAMPLES FROM THE AQUATIC ENVIRONMENT

4.2.1 DRINKING WATER SAMPLES

No radioactivity attributable to operation of the Trojan Nuclear Plant was detected in any of the water samples. The data are presented in Chapter 5, Table 5-2.

Table 4-2 presents the annual average of the gross beta activity for the two water sample sites from 1980 through 2004. These samples were not collected prior to 1980. The annual average values do not differ significantly over the years.

4.2.2 SHORELINE SOIL

None of the shoreline soil samples showed detectable levels of gamma emitters. The data are presented in Chapter 5, Table 5-3.

4.3 SUMMARY OF RESULTS

Table 4-3 presents a summary of the radioactivity analysis results for each medium or pathway sampled during 2004 for the Radiological Environmental Monitoring Program. The format of Table 4-3 is that which is required by ODCM Control 4.1.1.

A review of Table 4-3 shows that none of the radioactivity measurements, averaged over a quarter year period, was larger than the Reporting Levels defined by ODCM Control 3.3.1.

For the ambient radiation measurements, the mean value for the control locations was not significantly different than the mean values for the Trojan onsite locations. Control locations are Rainier and St. Helens, Oregon. Data from locations directly affected by shine from loaded Casks in the Trojan ISFSI are not included in the summary of results.

For the radioactivity measurements in drinking water, the annual mean for the gross beta determination was not significantly different from prior years.

As is shown by Table 4-3, there is no indication that the operations of the Trojan Nuclear Plant had a radiological impact on the environs around the Plant.

TABLE 4-1

AVERAGE AMBIENT GAMMA RADIATION LEVELS
mR/Day

<u>Year</u>	<u>Trojan Site</u>	<u>Oregon</u>	<u>Washington</u>
1976	0.13	0.14	0.13
1977	0.13	0.15	0.14
1978	0.11	0.13	0.13
1979	0.11±0.02	0.14±0.02	0.13±0.03
1980	0.11±0.02	0.14±0.02	0.12±0.01
1981	0.11±0.03	0.14±0.02	0.12±0.02
1982	0.14±0.03	0.16±0.02	0.15±0.02
1983	0.12±0.02	0.14±0.02	0.13±0.01
1984	0.12±0.03	0.13±0.02	0.12±0.02
1985	0.12±0.03	0.14±0.02	0.12±0.02
1986	0.12±0.03	0.14±0.03	0.12±0.02
1987	0.13±0.03	0.15±0.03	0.12±0.02
1988	0.12±0.02	0.14±0.02	0.12±0.02
1989	0.11±0.02	0.14±0.02	0.12±0.02
1990	0.11±0.02	0.13±0.03	0.11±0.02
1991	0.11±0.02	0.13±0.02	0.13±0.02
1992	0.10±0.03	0.13±0.03	0.12±0.02
1993	0.10±0.03	0.12±0.03	0.10±0.03
1994	0.19±0.03	0.22±0.03	0.20±0.03
1995	0.08±0.02	0.11±0.01	*
1996	0.09±0.02	0.11±0.02	*
1997	0.08±0.02	0.09±0.02	*
1998	0.08±0.02	0.09±0.01	*
1999	0.07±0.02	0.09±0.01	*
2000	0.07±0.02	0.09±0.01	*
2001	0.08±0.02	0.10±0.02	*
2002	0.08±0.02	0.10±0.02	*
2003	0.09±0.02	0.10±0.03	*
2004	0.10±0.04	0.10±0.02	*

* Due to revisions of the Radiological Environmental Monitoring Program, ambient gamma radiation levels are no longer measured in the state of Washington.

TABLE 4-2

AVERAGE GROSS BETA CONCENTRATIONS
FOR DRINKING WATER FROM COLUMBIA RIVER
(Units: pCi/l)

<u>Year</u>	<u>No. 8 -Rainier (Downstream)</u>	<u>No. 9 - St. Helens (Upstream)</u>
1980	2±2	2±1
1981	2±1	3±1
1982	3±2	4±2
1983	3±2	4±2
1984	3±2	4±2
1985	3±2	4±1
1986	3±2	3±2
1987	3±2	4±1
1988	4±2	6±3
1989	3±2	4±2
1990	2±3	5±3
1991	3±3	1±2
1992	2±1	3±1
1993	2±1	3±1
1994	2±1	3±1
1995	2±0.4	3±1
1996	2±0.4	3±1
1997	2±1	3±1
1998	2±1	3±1
1999	2±1	3±1
2000	2±1	3±1
2001	2±1	3±1
2002	2±2	3±2
2003	2±2	*
2004	2±1	*

* Due to revisions of the Radiological Environmental Monitoring Program, drinking water samples are no longer collected at St. Helens.

TABLE 4-3

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY

Trojan Nuclear Plant, Columbia County, Oregon, Docket 50-344, Reporting Period: January 1-December 31, 2004

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection (LLD) ^[a]	All Indicator Locations Mean(f) ^[b] Range	Location with Highest Annual Mean		Control Locations Mean(f) ^[b] Range	Number of Reportable Events
				Name, Distance, and Direction	Mean(f) ^[b] Range		
Ambient Radiation ^[d] (mR/day)	γ-exposure-36	0.04	0.10(36/36) 0.04-0.21	13 - N Site Boundary 800 meters NNW	0.16(4/4) 0.11-0.21	0.10(8/8) 0.08-0.13	N/A ^[c]
Drinking Water (pCi/liter)	Gross β-13	1	1.5(13/13) 1.2-2.0	8 - Rainier 3.8 mi - NW	1.5(13/13) 1.2-2.0	N/A ^[c]	N/A ^[c]
	Tritium-13	1000	<LLD	-	<LLD	<LLD	0
	γ-scan-13	Table 3-1	<LLD	-	<LLD	<LLD	0
Shoreline Soil (pCi/g - dry)	γ-scan-2	Table 3-1	<LLD	-	<LLD	N/A ^[c]	N/A ^[c]

^[a] Nominal Lower Limit of Detection (LLD) as defined in Section 3.1.^[b] Mean and range based upon detectable measurements only. The fraction of detectable measurements at specified locations is indicated in parentheses (f).^[c] N/A - Not applicable.^[d] Locations directly affected by shine from loaded Casks in the Trojan ISFSI are not included in data summary.

5.0 COMMENTS ON AND TERMS USED IN DATA TABLES

Dry Weight	A reporting unit used for shoreline soil in which the amount of sample is taken to be the weight of the sample after removal of moisture by drying in an oven at about 110°C for about 15 hours.
Gamma Emitters or Gamma Isotopic	Samples were analyzed by high resolution germanium gamma spectrometry. The resulting spectrum is analyzed by a computer program which scans about 50 to 2000 KeV and lists the energy peaks of any nuclides present in concentrations exceeding the sensitivity limits set for that particular experiment.
Error Terms	Figures following "±" are error terms based on counting uncertainties at the 2σ (95 percent confidence) level unless otherwise specified. Values preceded by the "<" symbol were below the stated concentration as defined by the notation associated with Table 3.3.1-3 of Trojan's Offsite Dose Calculation Manual.

TABLE 5-1

AMBIENT GAMMA RADIATION LEVELS

mR/Day

<u>Location</u>	<u>First Quarter</u> <u>12/31/03-3/31/04</u>	<u>Second Quarter</u> <u>3/31/04-6/30/04</u>	<u>Third Quarter</u> <u>6/30/04-9/30/04</u>	<u>Fourth Quarter</u> <u>9/30/04-12/29/04</u>
1	0.09	0.12	0.08	0.10
2*	0.13	0.11	0.18	0.20
3*	1.70	2.08	2.00	2.42
4	0.10	0.07	0.05	0.09
5	0.09	0.07	0.07	0.12
6	0.12	0.10	0.09	0.11
7	0.09	0.05	0.04	0.07
8	0.11	0.08	0.09	0.13
9	0.12	0.08	0.09	0.12
13	0.16	0.11	0.16	0.21
14	0.12	0.10	0.09	0.11
15*	0.16	0.19	0.22	0.28

* Location directly affected by shine from loaded Casks in the Trojan ISFSI.

TABLE 5-2

RADIOACTIVITY IN DRINKING WATER

<u>Location 8 - Rainier Municipal Water Supply</u>			
<u>pCi/l</u>			
<u>Collection Dates</u>	<u>Gross Beta</u>	<u>Tritium</u>	<u>Gamma Emitters</u>
12/09/03-1/06/04	1.5±0.4	<LLD	<LLD
1/06/04-2/03/04	1.3±0.4	<LLD	<LLD
2/03/04-3/02/04	1.7±0.4	<LLD	<LLD
3/02/04-3/30/04	1.7±0.4	<LLD	<LLD
3/30/04-4/26/04	2.0±0.5	<LLD	<LLD
4/26/04-5/25/04	1.3±0.4	<LLD	<LLD
5/25/04-6/22/04	1.2±0.4	<LLD	<LLD
6/22/04-7/20/04	1.7±0.4	<LLD	<LLD
7/20/04-8/17/04	1.3±0.2	<LLD	<LLD
8/17/04-9/14/04	1.4±0.3	<LLD	<LLD
9/14/04-10/12/04	1.4±0.3	<LLD	<LLD
10/12/04-11/09/04	1.7±0.3	<LLD	<LLD
11/09/04-12/07/04	1.9±0.3	<LLD	<LLD

LLD: 15 pCi/l Co-60, Cs-134
 18 pCi/l Cs-137
 1000 pCi/l H-3

TABLE 5-3

RADIOACTIVITY IN SHORELINE SOIL
(Semiannual Collections)

pCi/g (dry)

<u>Location 10</u>		
Collection		Gamma
<u>Date</u>		<u>Emitters</u>
3/02/04		< LLD
9/15/04		< LLD

LLD: 0.15 pCi/g Cs-134
0.18 pCi/g Cs-137

**TROJAN NUCLEAR PLANT
RADIOLOGICAL ENVIRONMENTAL
MONITORING REPORT**

APPENDIX A

**PGE-1021, OFFSITE DOSE CALCULATION MANUAL
AMENDMENT 24**

PGE-1021

OFFSITE DOSE CALCULATION MANUAL

Amendment 24

Portland General Electric Company
121 SW Salmon Street
Portland, OR 97204

PGE-1021
TROJAN NUCLEAR PLANT
OFFSITE DOSE CALCULATION MANUAL

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1.0 INTRODUCTION

The Offsite Dose Calculation Manual (ODCM) contains the Radioactive Effluent Controls Program required by PGE-8010, Trojan Nuclear Quality Assurance (QA) Program. This Program includes the Radiological Effluent Controls and their associated Surveillance Requirements, plus the methodology and parameters to be used for the calculation of offsite doses resulting from radioactive gaseous (with no spent nuclear fuel remaining on site, the term "gaseous" is used herein to represent radioactive particulates that can become airborne effluent constituents) and liquid effluents, and for the calculation of liquid effluent monitoring alarm/trip setpoints if such monitoring instruments are used. The implementation of this Program ensures compliance with the requirements of 10 CFR 50.36a, Subpart D of 10 CFR 20, Appendix I of 10 CFR 50, and 40 CFR 190. The dose calculation methodology is based on plant-specific applications of the dose models contained in Regulatory Guide 1.109 (Rev. 1, 10/77) and/or the simplified models presented in NUREG 0133 (10/78).

The ODCM contains the Radiological Environmental Monitoring Program required by PGE-8010, Trojan Nuclear QA Program. This Program consists of monitoring stations and sampling programs designed to confirm the dose estimates made under the Radiological Effluent Controls Program and to meet the requirements of Appendix I to 10 CFR 50. The Radiological Environmental Monitoring Program of the ODCM also includes requirements to participate in an interlaboratory comparison program.

The ODCM contains the Process Control Program (PCP) for solid radioactive wastes which is required by PGE-8010, Trojan Nuclear QA Program. The ODCM also contains administrative controls regarding the content of the annual Radiological Environmental Monitoring Report and the annual Radioactive Effluent Release Report which are required by PGE-8010, Trojan Nuclear QA Program, and administrative controls regarding major changes to radioactive waste treatment systems.

2.0 DEFINITIONS

The defined terms in this section appear in capitalized type and are applicable throughout these controls.

2.1 ACTION

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

2.2 CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY, including alarm and/or trip functions.

2.3 CHANNEL CALIBRATION

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

2.4 CHANNEL CHECK

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

2.0 DEFINITIONS

2.5 FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 2.1.

2.6 LIQUID RADWASTE TREATMENT SYSTEM

LIQUID RADWASTE TREATMENT SYSTEM is the system used to reduce radioactive materials in liquid effluents by filtering and demineralizing the radioactive wastes for the purpose of reducing the total radioactivity prior to release to the environment.

2.7 MEMBER(S) OF THE PUBLIC

MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

2.8 OFFSITE DOSE CALCULATION MANUAL

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of liquid effluent monitoring Alarm/Trip Setpoints if such monitoring instruments are used, and in the conduct of the Radiological Environmental Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by PGE-8010, Trojan Nuclear QA Program, and (2) descriptions of the information that should be included in the Annual Radiological Environmental Monitoring and annual Radioactive Effluent Release Reports required by PGE-8010, Trojan Nuclear QA Program.

2.0 DEFINITIONS

2.9 OPERABLE - OPERABILITY

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

2.10 SITE BOUNDARY

The SITE BOUNDARY shall be that line beyond which the land is neither owned, nor leased, nor otherwise controlled by the licensee.

2.11 SOLIDIFICATION

SOLIDIFICATION shall be the conversion of radioactive wastes from liquid systems to a form that meets shipping and burial ground requirements.

2.12 SOURCE CHECK

A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to either an installed detector check source or to a background radiation level if background exceeds the installed check source strength.

2.0 DEFINITIONS

2.13 UNRESTRICTED AREA

An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

2.14 VENTILATION EXHAUST TREATMENT SYSTEMS

The VENTILATION EXHAUST TREATMENT SYSTEMS are those systems designed and installed to reduce the gaseous radioactive material in particulate form in effluents by passing the ventilation exhaust from the Fuel and Auxiliary Buildings through HEPA filters prior to release to the environment. Such systems are not considered to have any effect on noble gas effluents.

TABLE 2.1

Frequency Notation	
Notation	Frequency
D	At least once per 24 hours
W	At least once per 7 days
M	At least once per 31 days
Q	At least once per 92 days
R	At least once per 18 months
P	Completed prior to each release
N/A	Not applicable

3.1 RADIOACTIVE EFFLUENT INSTRUMENTATION

CONTROL 3.1.1 - LIQUID

The Discharge Structure Flow Recorder shown in Table 3.1.1-1 shall be OPERABLE to ensure that the limits of Control 3.2.1.1 are not exceeded.

Radioactive liquid effluent monitoring of grab samples prior to release will ensure the limits of Control 3.2.1.1 are met.

APPLICABILITY

As shown in Table 3.1.1-1.

ACTION

- a. With the Discharge Structure Flow Recorder not OPERABLE, take the ACTION shown in Table 3.1.1-1. Restore the instrument to OPERABLE status within 30 days or identify the cause of the inoperable channels in the annual Radioactive Effluent Release Report in lieu of any other report.

SURVEILLANCE REQUIREMENTS

- a. The Discharge Structure Flow Recorder shall be demonstrated OPERABLE by performance of a CHANNEL CHECK and CHANNEL CALIBRATION at the frequencies shown in Table 3.1.1-2 (formerly Surveillance Requirement 4.1.1.2).

BASIS

The OPERABILITY and use of the Discharge Structure Flow Recorder is consistent with the requirements of General Design Criteria 60, 63, and 64 of Appendix A to 10 CFR 50.

TABLE 3.1.1-1

Radioactive Liquid Effluent Monitoring Instrumentation			
Instrument	Minimum Channels Operable	Applicability	Action
1. Flow Rate Measurement Device			
a. Discharge Structure Flow Recorder	1	*	1

* During releases via this pathway

Table Notation

ACTION 1 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided the flow rate is estimated at least once per 4 hours during actual releases.

Liquid Radwaste Effluent Line Flow: Flowrate shall be determined by either estimating the flowrate during discharge or by using a flowrate previously calculated by testing of the LRW System.

TABLE 3.1.1-2

Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements		
Instrument	Channel Check	Channel Calibration
1. Flow Rate Monitor		
a. Discharge Structure Flow Recorder	D*	R**

* During releases via this pathway.

** Applies only to flow indication loop.

Liquid Radwaste Effluent Line Flow: There is no flow instrumentation in the effluent line. Flowrate shall be determined by either estimating the flowrate during discharge or by using a flowrate previously calculated by testing of the LRW System.

3.1 RADIOACTIVE EFFLUENT INSTRUMENTATION

CONTROL 3.1.2 – GASEOUS

The radioactive gaseous effluent sampling equipment channels shown in Table 3.1.2-1 shall be OPERABLE to ensure that the limits of Control 3.2.2.1 are not exceeded.

APPLICABILITY

As shown in Table 3.1.2-1.

ACTION

With less than the minimum number of radioactive gaseous effluent sampling equipment channels operable, take the ACTION shown in Table 3.1.2-1. With the inoperable channels not returned to OPERABLE status within 30 days, identify the cause of the inoperable channels in the annual Radioactive Effluent Release Report in lieu of any other report.

SURVEILLANCE REQUIREMENTS

Each radioactive gaseous effluent sampling equipment channel shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at the frequencies shown in Table 3.1.2-2.

BASIS

The radioactive gaseous effluent sampling equipment is provided to measure the releases of radioactive materials in gaseous effluents during actual or potential releases. However, in no case will the limits of 10 CFR 20 be exceeded. The OPERABILITY and use of this equipment is consistent with the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR 50.

TABLE 3.1.2-1

Sheet 1 of 1

Radioactive Gaseous Effluent Sampling Equipment			
Instrument	Minimum Channels Operable	Applicable Modes	Action
1. Auxiliary Building Ventilation Monitoring System			
a. Particulate Composite Sampler	1	*	1
b. Sampler Flow Rate Measuring Device for Composite Sampler	1	*	1
2. Condensate Demineralizer Building Effluent Monitoring			
a. Particulate Composite Sampler	1	*	1
b. Sampler Flow Rate Measuring Device for Composite Sampler	1	*	1

*During releases via this pathway.

ACTION 1 With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway will be stopped.

TABLE 3.1.2-2

Sheet 1 of 1

Radioactive Gaseous Effluent Sampling Equipment Surveillance Requirements	
Instrument	Channel Check
1. Auxiliary Building Ventilation Monitoring System	
a. Particulate Composite Sampler	W(1)
b. Sampler Flow Rate Measuring Device for Composite Sampler	W*
2. Condensate Demineralizer Building	
a. Particulate Composite Sampler	W(1)
b. Sampler Flow Rate Measuring Device for Composite Sampler	W*

* During or prior to releases via this pathway.

(1) CHANNEL CHECK consists of verification of sampler flow through the sampler.

3.2 RADIOACTIVE EFFLUENTS

CONTROL 3.2.1.1 - LIQUID EFFLUENTS CONCENTRATION

The concentration of radioactive material released in liquid effluents from the site to UNRESTRICTED AREAS shall be limited to the concentrations specified in 10 CFR 20, Appendix B, Table 2, Column 2.

APPLICABILITY

At all times.

ACTION

With the concentration of radioactive material released from the site to UNRESTRICTED AREAS exceeding the above limits, immediately restore concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

- a. Radioactive liquid wastes shall be sampled and analyzed according to the sampling and analysis program of Table 3.2.1.1-1 (formerly Surveillance Requirement 4.2.1.1.1).
- b. The results of radioactive analysis shall be used with the calculational methods in Section 5 of the ODCM to assure that the concentration at the point of release is limited to the values in Control 3.2.1.1 (formerly Surveillance Requirement 4.2.1.1.2).

3.2 RADIOACTIVE EFFLUENTS

BASIS

This Control is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to UNRESTRICTED AREAS will be less than the concentration levels specified in Appendix B, Table 2, Column 2, to 10 CFR 20. This limitation provides reasonable assurance that the levels of radioactive materials in bodies of water in UNRESTRICTED AREAS will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, to a MEMBER OF THE PUBLIC and (2) restrictions authorized by 10 CFR 20.1301(e). This Control does not affect the requirement to comply with the annual limitations of 10 CFR 20.1301(a).

TABLE 3.2.1.1-1

Sheet 1 of 2

Radioactive Liquid Waste Sampling and Analysis Program				
Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Sample/ Activity Analysis	Lower Limit of Detection (LLD) ($\mu\text{Ci/ml}$) ^a
Batch ^d Waste Release Tanks	P	P	Grab Sample/ Principal Gamma Emitters ^e	5×10^{-7b}
	P	M	Composite ^e / Tritium	1×10^{-5}
	P	M	Composite ^e / Gross Alpha	1×10^{-7}
	P	Q	Composite ^e / Sr-90	5×10^{-8}
	P	Q	Composite ^e / Fe-55	1×10^{-6}

Table Notation

- a. The lower limit of detection (LLD) is defined in Table Notation a. of Table 3.3.1-3 of Control 3.3.1 with the exception of Δt . Δt in this case is the elapsed time between midpoint of sample collection and time of counting.
 - b. For certain radionuclides with low gamma yield or low energies, or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the LLD. Under these circumstances, the LLD may be increased provided that such an increase will not result in a discharge of that radionuclide which is greater than the effluent concentration value specified in 10 CFR 20, Appendix B, Table 2, Column 2, in the diluted stream.
 - c. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.
 - d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analysis, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.
 - e. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Co-60, Cs-134, and Cs-137. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level. When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the annual Radioactive Effluent Release Report.
-

3.2 RADIOACTIVE EFFLUENTS

CONTROL 3.2.1.2 - DOSE

The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released to UNRESTRICTED AREAS shall be limited:

During any calendar quarter to ≤ 1.5 mrem to the total body and to ≤ 2.5 mrem to any organ.

APPLICABILITY

At all times.

ACTION

With the calculated dose from the release of radioactive material in liquid effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days upon determination, pursuant to Control 4.1.3, a Special Report in lieu of any other report, which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to prevent recurrence and to reduce the releases to below the design objectives. This Special Report shall also include (1) the results of radiological analyses of the drinking water source, and (2) the radiological impact in finished drinking water supplies with regard to the requirements of 40 CFR 141, Safe Drinking Water Act.

SURVEILLANCE REQUIREMENTS

Dose Calculations. Cumulative doses due to liquid releases to UNRESTRICTED AREAS shall be determined in accordance with Section 5 of the ODCM at least once per 31 days when the cumulative liquid activity release, excluding tritium, exceeds 2.5 Ci/qtr. The cumulative liquid activity release will be determined at least once per 31 days (formerly Surveillance Requirement 4.2.1.2.1).

3.2 RADIOACTIVE EFFLUENTS

BASIS

This Control is provided to implement the requirements of Section II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. Section II.A of Appendix I, 10 CFR 50 specifies design objective dose to an individual defined as a MEMBER OF THE PUBLIC, from radioactive materials in liquid effluents released to UNRESTRICTED AREAS will be limited during any calendar year to ≤ 3 mrem to the total body. The design objective for any organ will be limited to ≤ 5 mrem during any calendar year in accordance with PGE Agreement with intervenors dated May 1972. Section IV.A to 10 CFR 50 specifies the limiting condition for operation as one-half the design objective annual exposure in one calendar quarter.

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 liquid effluents will be kept "as low as reasonably achievable." Also, for fresh water sites with drinking water supplies which can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141. The dose calculations that are in Section 5 of the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in Section 5 of the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents will be consistent with the methodology provided in Regulatory Guides 1.109 (Rev. 1) and 1.113 (Rev. 1). The intermediate surveillance value of 2.5 Ci/qtr excluding tritium is a release rate which has been shown during 7 years of Trojan operation to be significantly below the Control value of 1.5 mrem/qtr total body and 2.5 mrem/qtr to any organ. Refer to PGE-1015, Annual Operating Report of Trojan Nuclear Power Plant for 1977, 1978, 1979, 1980, 1981, 1982 and 1983.

3.2 RADIOACTIVE EFFLUENTS

CONTROL 3.2.1.3 - LIQUID WASTE TREATMENT

The LIQUID RADWASTE TREATMENT SYSTEM shall be maintained and used to reduce the radioactive materials in liquid wastes prior to their discharge when the liquid activity release, excluding tritium to UNRESTRICTED AREAS when averaged over a calendar quarter would exceed 1.25 Ci/qtr.

APPLICABILITY

At all times.

ACTION

With radioactive liquid waste being discharged without treatment and in excess of the above limits, the following information shall be provided in the annual Radioactive Effluent Release Report:

- a. Identification of equipment not OPERABLE and the reason for inoperability.
- b. Action(s) taken to restore the inoperable equipment to OPERABLE status.
- c. Summary description of action(s) taken to prevent a recurrence.

SURVEILLANCE REQUIREMENTS

Cumulative liquid activity releases excluding tritium and dissolved gases to UNRESTRICTED AREAS shall be determined at least once per 31 days (formerly Surveillance Requirement 4.2.1.3).

3.2 RADIOACTIVE EFFLUENTS

BASIS

This Control ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirements that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This Control implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR 50 and Design Objective Section II.D of Appendix I to 10 CFR 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as one quarter of the annual design objective set forth in Section II.A of Appendix I, 10 CFR 50, for liquid effluents (3 mrem/yr whole body; 5 mrem/yr maximum organ per PGE Agreement with Intervenor dated May 1972). The dose calculational procedures specified in Section 5 of the ODCM include sufficient factors of conservatism to ensure that the sum of both treated and untreated releases will not result in doses exceeding the design objectives. The surveillance value of 1.25 Ci/qtr excluding tritium is a release rate which has been shown during the first 7 years of Trojan operation to be significantly below the Control value of 0.75 mrem/qtr total body and 1.25 mrem/qtr to any organ.

3.2 RADIOACTIVE EFFLUENTS

CONTROL 3.2.2.1 - GASEOUS EFFLUENTS DOSE RATE

The dose rate to areas at or beyond the SITE BOUNDARY due to radioactive materials released in gaseous effluents from the site shall be limited to the following:

The dose rate limit for radionuclides in particulate form with half-lives greater than 8 days shall be ≤ 1500 mrem/yr to any organ.

APPLICABILITY

At all times.

ACTION

With dose rate(s) exceeding the above limit, immediately decrease the release rate to comply with the limit given in Control 3.2.2.1.

SURVEILLANCE REQUIREMENTS

- a. The release rate of radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be such that $0.67 Q_i \overline{R}_i \leq 1$ by using the results of the sampling and analysis program specified in Table 3.2.2.1-1 (formerly Surveillance Requirement 4.2.2.1.2).
- b. The above release rate is determined in accordance with the methodology and parameters in Section 6 of the ODCM (formerly Surveillance Requirement 4.2.2.1.3).

3.2 RADIOACTIVE EFFLUENTS

BASIS

This Control provides reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY in excess of the design objectives of Appendix I to 10 CFR Part 50. It provides operational flexibility for releasing gaseous effluents to satisfy the Section II.A and II.C design objectives of Appendix I to 10 CFR Part 50. For MEMBERS OF THE PUBLIC who may, at times, be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The release rate limit restricts the corresponding dose rate above background to any organ to ≤ 1500 mrem/year. This Control does not affect the requirement to comply with the annual limitations of 10 CFR 20.1301(a).

TABLE 3.2.2.1-1

Sheet 1 of 2

Radioactive Gaseous Waste Sampling and Analysis Program				
Gaseous Release Type	Minimum Sampling Frequency	Minimum Analysis Frequency	Type of Sample/ Activity Analysis	Lower Limit of Detection (LLD) ($\mu\text{Ci/ml}$) ^a
A. Fuel and Auxiliary Building Ventilation Exhaust	M^{cd}	M^{cd}	Composite/Principal Gamma Emitters ^c	$1 \times 10^{-11\text{b}}$
		M^{cd}	Composite/Gross Alpha	$1 \times 10^{-11\text{b}}$
		Q^{cd}	Composite/Sr-90	$1 \times 10^{-11\text{b}}$
B. Condensate Demineralizer Building Exhaust	M^{cd}	M^{cd}	Composite/Principle Gamma Emitters ^c	$1 \times 10^{-11\text{b}}$
		M^{cd}	Composite/Gross Alpha	$1 \times 10^{-11\text{b}}$
		Q^{cd}	Composite/Sr-90	$1 \times 10^{-11\text{b}}$

Table Notation

- a. The lower limit of detection (LLD) is defined in Table Notation a. of Table 3.3.1-3 of Control 3.3.1 with the exception of Δt . Δt in this case is the elapsed time between midpoint of sample collection and time of counting.
 - b. For certain radionuclides with low gamma yield or low energies, or for certain radionuclide mixtures, it may not be possible to measure radionuclides in concentrations near the LLD. Under these circumstances, the LLD may be increased provided that such an increase will not result in a discharge of that radionuclide which is greater than the effluent concentration value specified in 10 CFR 20, Appendix B, Table 2, Column 1, in plant effluents.
 - c. Analysis shall also be performed following occurrences which could alter the mixture of radionuclides.
 - d. Samples shall be taken and analyzed at the specified minimum frequency when there is a discharge through each release point.
 - e. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Co-60, Cs-134, 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 below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the annual Radioactive Effluent Release Report.
-

3.2 RADIOACTIVE EFFLUENTS

CONTROL 3.2.2.2 - DOSE, NOBLE GASES

This section is intentionally deleted.

3.2 RADIOACTIVE EFFLUENTS

CONTROL 3.2.2.3 - DOSE, RADIONUCLIDES IN PARTICULATE FORM

The dose to a MEMBER OF THE PUBLIC from radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at or beyond the SITE BOUNDARY shall be limited to the following:

During any calendar quarter to ≤ 2.5 mrem to any organ.

APPLICABILITY

At all times.

ACTION

With the calculated dose from the release of radioactive materials in particulate form, or radionuclides other than noble gases in gaseous effluents exceeding the above limit, prepare and submit to the Commission within 30 days upon determination, pursuant to Control 4.1.3, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions to be taken to prevent recurrence and to reduce releases to below the design objectives.

SURVEILLANCE REQUIREMENTS

- a. Release Rate Calculations: The average release rate of radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at or beyond the SITE BOUNDARY during any calendar quarter shall be such that:

$$100 Q_v \overline{R_i} < 1$$

(formerly Surveillance Requirement 4.2.2.3.1).

3.2 RADIOACTIVE EFFLUENTS

SURVEILLANCE REQUIREMENTS (Continued)

- b. The above release rates are determined in accordance with the methodology and parameters in Section 6 of the ODCM at least once per 31 days (formerly Surveillance Requirement 4.2.2.3.2).

BASIS

This Control is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR 50. As outlined in Section II.C of Appendix I, 10 CFR 50, the design objective dose to an individual from radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at or beyond the SITE BOUNDARY will be limited during any calendar year to ≤ 15 mrem to any organ. This value is further reduced to 5 mrem/year maximum organ dose per PGE Agreement with Intervenors, dated May 1972. As outlined in Section IV.A to Appendix I, 10 CFR 50, the limiting condition for operation is specified as one-half the design objective annual exposure in one calendar quarter.

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 effluents will be kept "as low as is reasonably achievable." The ODCM 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 an individual through appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in Section 6 of the ODCM for calculating the doses due to the actual release rates of particulates in gaseous effluents will be consistent with the methodology provided in Regulatory Guides 1.109 (Rev. 1) and 1.111 (Rev. 1). The ODCM equations provided for determining these doses will be based upon the historical average atmospheric conditions. The release rate specifications for radionuclides in particulate form with half-lives greater than 8 days are dependent on the existing radionuclide pathways to man, in areas at or beyond the SITE BOUNDARY.

3.2 RADIOACTIVE EFFLUENTS

BASIS (Continued)

The pathways which are examined in the development of these calculations are: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat-producing animals graze with consumption of milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

3.2 RADIOACTIVE EFFLUENTS

CONTROL 3.2.2.4 - VENTILATION EXHAUST TREATMENT

The VENTILATION EXHAUST TREATMENT SYSTEMS shall be maintained and used to reduce radioactive materials in gaseous waste prior to their discharge when the doses due to radionuclides in particulate form with half-lives greater than 8 days in gaseous effluent releases to areas at or beyond the SITE BOUNDARY when averaged over a calendar quarter would exceed 1.25 mrem to any organ.

APPLICABILITY

At all times.

ACTION

With gaseous waste being discharged for more than 31 days without treatment and in excess of the above limits, discuss in the annual Radioactive Effluent Release Report the following information:

- a. Identification of equipment not OPERABLE and the reason for inoperability.
- b. Action(s) taken to restore the inoperable equipment to OPERABLE status.
- c. Summary description of action(s) taken to prevent a recurrence.

3.2 RADIOACTIVE EFFLUENTS

SURVEILLANCE REQUIREMENTS

- a. The average release rate of radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released to areas at or beyond the SITE BOUNDARY during any calendar quarter shall be such that:

$$200 Q_v \overline{R_i} < 1$$

(formerly Surveillance Requirement 4.2.2.4.2).

- b. The above release rate is determined in accordance with the methodology and parameters in Section 6 of the ODCM at least once per 31 days (formerly Surveillance Requirement 4.2.2.4.3).

BASIS

This Control ensures that the ventilation exhaust treatment systems will be used whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This Control implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR 50 and design objective Section II.D of Appendix I to 10 CFR 50.

The specified limits governing the use of appropriate portions of the systems were specified as one quarter of the annual design objective set forth in Sections II.B and II.C of Appendix I, 10 CFR 50, for gaseous effluents (5 mrem/yr maximum organ dose per PGE Agreement with intervenors, dated May 1972).

The dose calculational procedures specified in Section 6 of the ODCM include sufficient factors of conservatism to ensure that the sum of both treated and untreated releases will not result in doses exceeding the design objectives.

3.2 RADIOACTIVE EFFLUENTS

CONTROL 3.2.2.5 - TOTAL DOSE

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

APPLICABILITY

At all times.

ACTION

With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Controls 3.2.1.2 or 3.2.2.3, calculations should be made to determine whether the above limits of Control 3.2.2.5 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days a Special Report pursuant to 10 CFR 20.2203(a)(4) and Control 4.1.3.

SURVEILLANCE REQUIREMENTS

Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Control 3.2.1.2 and 3.2.2.3 Surveillance Requirements and in accordance with the methodology and parameters in Section 6 of the ODCM (formerly Surveillance Requirement 4.2.2.5.1).

BASIS

This Control is provided to meet the dose limitations of 40 CFR 190 that have been incorporated into 10 CFR 20.1301(d). The Control requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel

3.2 RADIOACTIVE EFFLUENTS

BASIS (Continued)

cycle sources exceed 25 mrem to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem. It is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if radioactive effluents remain within twice the dose design objectives of Appendix I.

3.2 RADIOACTIVE EFFLUENTS

CONTROL 3.2.3.1 - SOLID RADIOACTIVE WASTE

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

APPLICABILITY

At all times.

ACTION

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

SURVEILLANCE REQUIREMENTS

The PROCESS CONTROL PROGRAM, as defined in the ODCM shall be used to verify that processed wet radioactive wastes (e.g., filter sludges and spent resins) meet the shipping and burial ground requirements with regard to solidification and dewatering (formerly Surveillance Requirement 4.2.3).

BASIS

This Control ensures that radioactive wastes that are transported from the site shall meet the solidification requirements specified by the burial ground license of the respective states to which the radioactive material will be shipped.

3.3 RADIOLOGICAL ENVIRONMENTAL MONITORING

CONTROL 3.3.1 - MONITORING PROGRAM

A radiological environmental monitoring program as specified in Table 3.3.1-1 shall be conducted in accordance with written procedures. (Reductions in the scope of this program shall be discussed with the Oregon State Health Division before implementing the reduction.)

APPLICABILITY

At all times.

ACTION

- a. With the radiological environmental monitoring program not being conducted as specified in Table 3.3.1-1, Table 3.3.1-4, and Figure 3.3.1-1, prepare and submit to the Commission, in the annual Radiological Environmental Monitoring Report, a description of the reasons for not conducting the program as required and the plans for preventing recurrence.

(Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or to malfunctions of equipment. If the latter, every effort shall be made to complete corrective action prior to the end of the next sampling period.)

- b. With the level of radioactivity in an environmental sampling medium at one or more of the locations specified in Table 3.3.1-1 exceeding the limits of Table 3.3.1-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days from receipt of analysis results for the affected calendar quarter, a Special Report which includes an evaluation of any release conditions, environmental factors or other aspects which caused

3.3 RADIOLOGICAL ENVIRONMENTAL MONITORING

ACTION (Continued)

the limits of Table 3.3.1-2 to be exceeded. When more than one of the radionuclides in Table 3.3.1-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{Concentration (1)}}{\text{Limit Level (1)}} + \frac{\text{Concentration (2)}}{\text{Limit Level (2)}} + \dots > 1.0$$

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

When Radionuclides other than those in Table 3.3.1-2 are detected and are the result of plant effluents, this report shall be submitted if the potential for annual dose to an individual is equal to or greater than the calendar year limits of Control 3.2.1.2 and 3.2.2.3.

SURVEILLANCE REQUIREMENTS

The radiological environmental monitoring samples shall be collected pursuant to Table 3.3.1-1 from the locations shown in Table 3.3.1-4 and Figure 3.3.1-1 and shall be analyzed pursuant to the requirements of Tables 3.3.1-1 and 3.3.1-3 (formerly Surveillance Requirement 4.3.1).

BASIS

In accordance with ODCM Control 3.3.1, the radiological environmental monitoring stations are listed in Table 3.3.1-4 with the radial distance presented in meters. The location of these stations with respect to the Trojan Nuclear Plant is shown in Figure 3.3.1-1.

The radiological monitoring program required by this Control provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides

3.3 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASIS (Continued)

which lead to the highest potential radiation exposures of individuals resulting from station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways.

The LLDs for drinking water meet the requirements of 40 CFR 141.

TABLE 3.3.1-1

Radiological Environmental Monitoring Program			
Exposure Pathway and/or Sample	Minimum* Number of Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
1. DIRECT RADIATION	12	At least once per 92 days.	Gamma dose measured by a single dosimeter at each location. At least once per 92 days.
2. WATERBORNE			
a. Surface Water including Drinking Water	1	Composite sample over 31-day period from Columbia River (downstream)	Gross beta and gamma isotopic analysis of each sample. Tritium analysis of composite sample at least once per 92 days.
b. Sediment from Shoreline	1	At least once per 184 days.	Gamma isotopic of each sample.

* Sample locations are identified in Table 3.3.1-4

TABLE 3.3.1-2

Reporting Levels for Radioactivity Concentrations in Environmental Samples	
Analysis	Water (pCi/l)
H-3	5×10^4 2×10^4 (a)
Co-60	2×10^2
Cs-134	20
Cs-137	20

(a) For drinking water samples. This is a 40 CFR 141 value.

TABLE 3.3.1-3

Sheet 1 of 2

Maximum Values for the Lower Limits of Detection (LLD) ^a		
Analysis	Sediment (pCi/kg, dry)	Water (pCi/l)
gross beta		4 (1 ^b)
H-3		2000 (1000 ^b)
Co-60		15
Cs-134	150	15
Cs-137	180	18

Table Notation

- a - The LLD is defined, for the purposes of these Controls, 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% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

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

$$LLD = \frac{4.66 s_b}{(E)(V)(2.22)(Y)(e^{-\lambda \Delta t})}$$

where

LLD	=	the "a priori" lower limit of detection as defined above (as pCi per unit mass or volume)
s_b	=	the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute)
E	=	the counting efficiency (as counts per disintegration)
V	=	the sample size (in units of mass or volume)
2.22	=	the number of transformation per minute per picocurie
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

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

- b - LLD for drinking water.

TABLE 3.3.1-4

Sampling Locations and Frequency by Type					
Sample Location	Radial		Sample		
	Distance (meters)	Direction	TLD	Surface Water	Shore Soil
1 - Trojan North Building	300	WNW	Q	MC	S/A
2 - NW Fenceline	210	NW	Q		
3 - N Fenceline	191	N	Q		
4 - Switchyard	191	WSW	Q		
5 - Training Building	354	SW	Q		
6 - Park Entrance	354	SSW	Q		
7 - South End Cooling Tower	640	SE	Q		
8 - Rainier	6,115	NW	Q		
9 - St. Helens (Municipal Water Supply)	16,898	SSE	Q		
10 - Columbia River	116,510*	E			
13 - N Site Boundary at Columbia River	800	NNW	Q		
14 - S Site Boundary	1,332	S	Q		
15 - E Fenceline	93	E	Q		

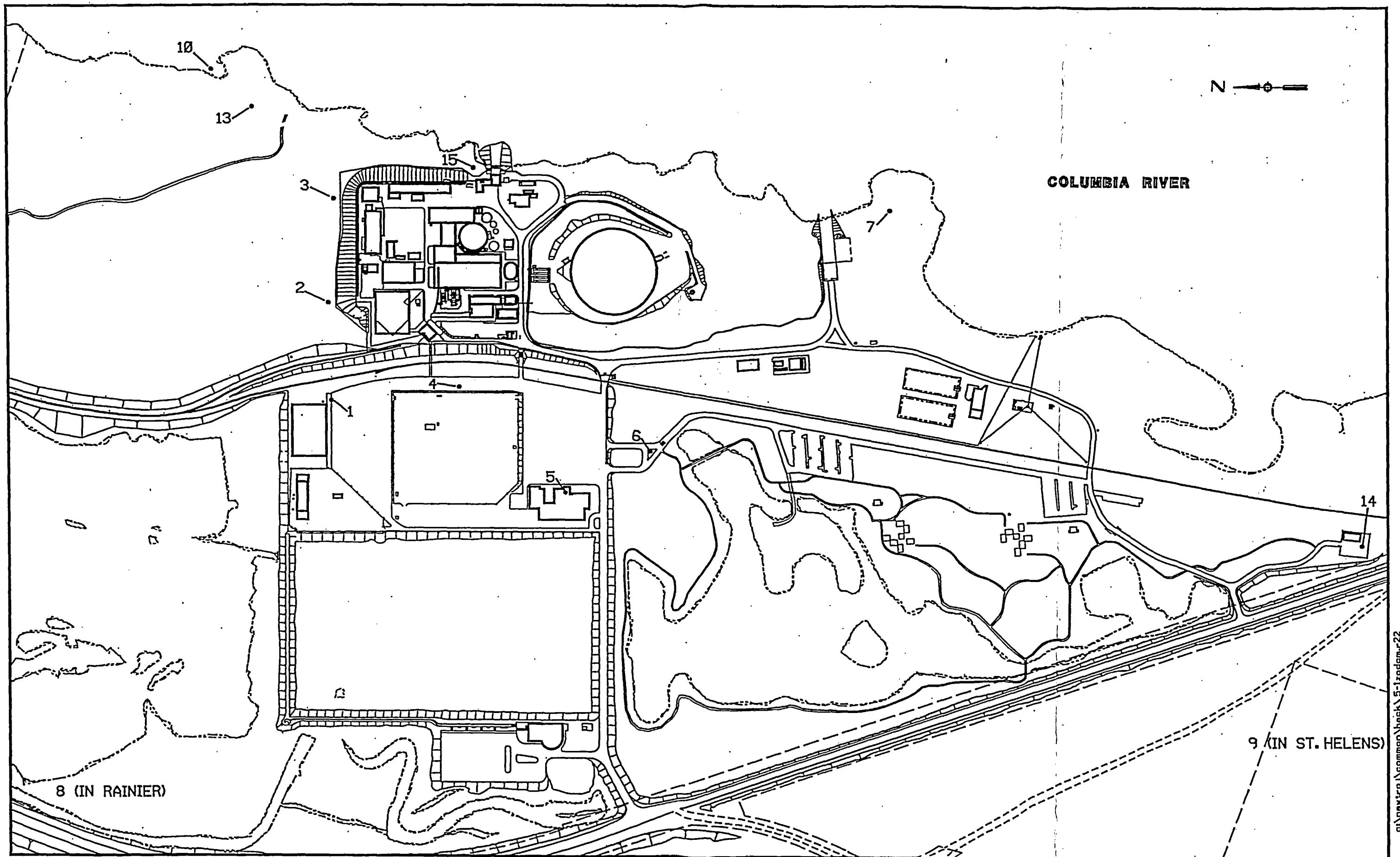
LEGEND:

Q - Quarterly

MC - Monthly Composite

S/A - Semi-annually

* - Columbia River mileage refers to meters measured from mouth



TROJAN NUCLEAR PLANT
 FIGURE 3.3.1-1 SAMPLING LOCATIONS
 3-37

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3.3 RADIOLOGICAL ENVIRONMENTAL MONITORING

CONTROL 3.3.2 - INTERLABORATORY COMPARISON PROGRAM

Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by NRC.

APPLICABILITY

At all times.

ACTION

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

SURVEILLANCE REQUIREMENTS

The results of analyses performed as part of the above required Interlaboratory Comparison Program shall be included in the annual Radiological Environmental Monitoring Report pursuant to Control 4.1.1 (formerly Surveillance Requirement 4.3.2).

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 a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

4.0 ADMINISTRATIVE CONTROLS

4.1 REPORTING REQUIREMENTS

4.1.1 ANNUAL RADIOLOGICAL ENVIRONMENTAL MONITORING REPORT

The Annual Radiological Environmental Monitoring Report shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant activities on the environment. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The Annual Radiological Environmental Monitoring Report shall include summarized and tabulated results in the format of Table 4.1.1-1 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion in the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program including sampling methods for each sample type, size and physical characteristics of each sample type, sample preparation methods, analytical methods, and measuring equipment used; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program required by Control 3.3.2.

Any changes to the ODCM made during the reporting period, shall be reported as provided in PGE-8010, Trojan Nuclear QA Program.

4.1.2 ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

The Annual Radioactive Effluent Release Report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21 (Rev. 1), "Measuring, Evaluating, and Reporting Radioactivity in Solid

4.0 ADMINISTRATIVE CONTROLS

Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants", with data summarized on a quarterly basis following the format of Appendix B thereof.

The Annual Radioactive Effluent Release Report shall include an assessment of the radiation doses from radioactive effluents to MEMBERS OF THE PUBLIC due to their activities in UNRESTRICTED AREAS during the report period. All assumptions used in making these assessments (e.g., specific activity, exposure time and location) shall be included in these reports.

The Annual Radioactive Effluent Release Report shall include a copy of all licensee event reports required by 10 CFR 50.73(a)(2)(viii).

The Annual Radioactive Effluent Release Report shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit as outlined in Regulatory Guide 1.21. Exceptions to Regulatory Guide 1.21 are documented in Table 1-1 of PGE-1061, Trojan Nuclear Plant Defueled Safety Analysis Report and License Termination Plan (PGE-1078). The assessment of radiation doses shall be performed in accordance with Sections 5 and 6 of the ODCM.

4.1.3 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 pursuant to the requirements of the applicable reference control.

- a. Radioactive Effluents, Controls 3.2.1.2, 3.2.2.3, and 3.2.2.5.
- b. Radiological Environmental Monitoring, Control 3.3.1.

4.0 ADMINISTRATIVE CONTROLS

4.2 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid)

Licensee initiated major changes* to the radioactive waste treatment systems (liquid, gaseous and solid):

- a. A summary description of the change including discussion of the equipment, components and processes involved shall be reported to the Commission. The change shall be reviewed and approved in accordance with plant procedures.
- 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. An 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 license application and amendments thereto;
 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 license application and amendments thereto;
 6. An estimate of the exposure to plant operating personnel as a result of the change; and

4.0 ADMINISTRATIVE CONTROLS

4.2 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and Solid) (Continued)

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

* Major changes to the radioactive waste treatment systems are permanent changes which would alter the capacity of handling radioactive wastes or differ in the method of treatment.

4.0 ADMINISTRATIVE CONTROLS

4.3 CHANGES TO THE ODCM

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(s) together with the appropriate analyses or evaluations justifying the change(s); and a determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, and 40 CFR 190, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

Changes to the ODCM shall become effective after review and approval by an Independent Safety Reviewer and the approval of the General Manager, Trojan or designee; and shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as part of, or concurrent with, the Radiological Environmental Monitoring Report for the period of the report in which any change in 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, and shall indicate the date (i.e., month and year) the change was implemented.

TABLE 4.1.1-1

Radiological Environmental Monitoring Program Summary							
Trojan Nuclear Plant, Columbia County, Oregon, Docket No. 50-344, Reporting Period _____							
Medium or Pathway Sampled (Units of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection ^a (LLD)	All Indicator Locations Mean(f) ^b Range ^b	<u>Location With Highest Annual Mean</u> Name Distance and Direction		Control Locations Mean(f) ^b Range ^b	Number of Reportable Events

a Nominal Lower Limit of Detection (LLD) as defined in table notation a of Table 3.3.1-3 of Control 3.3.1.

b Mean and range based upon detectable measurements only. The fraction of detectable measurements at specified locations is indicated in parentheses (*f*).

5.0 LIQUID EFFLUENT DOSE CALCULATIONS

5.1 INTRODUCTION

Cumulative quarterly dose contributions due to radioactive liquid effluents released to UNRESTRICTED AREAS will be determined at least once per 31 days when the cumulative liquid activity release, excluding tritium and dissolved gases, exceeds 2.5 Ci/quarter. These dose contributions will be calculated for all radionuclides identified in liquid effluents released to the UNRESTRICTED AREA using the following general equation (Reference: NUREG-0133, pg. 15):

$$D_j = \sum_i \left[A_{ij} \sum_{\ell=1}^m \Delta T_{\ell} C_{i\ell} F_{\ell} \right] \quad (5-1)$$

where

D_j = the cumulative quarterly dose commitment to any organ, j , from the liquid effluents for total time period $\sum_{\ell=1}^m \Delta T_{\ell}$, in mrem

A_{ij} = ingestion dose commitment factor to the total body or any organ j , for each identified nuclide i , listed in Table 5-1, in mrem/hr per $\mu\text{Ci/ml}$

ΔT_{ℓ} = the length of the ℓ^{th} time over which $C_{i\ell}$ and F_{ℓ} are averaged for all liquid releases, in hours

$C_{i\ell}$ = the average concentration of radionuclide i , during time period ΔT_{ℓ} , in $\mu\text{Ci/ml}$. The term $C_{i\ell}$ is the undiluted concentration of radioactive material in liquid waste determined in accordance with Table 3.2.1.1-1

F_{ℓ} = the near field average dilution factor for $C_{i\ell}$ during any liquid release.

5.0 LIQUID EFFLUENT DOSE CALCULATIONS

5.1 INTRODUCTION (Continued)

The term F_t , the near field average dilution factor, is determined as follows for time period ΔT_t :

$$F_t = \frac{\text{liquid radioactive waste discharge volume}}{\text{total plant discharge volume} \times \text{plant dilution factor}}$$

The plant dilution factor, F_{pd} , accounts for mixing effects of the dilution pipe. This value is determined in accordance with NUREG-0133, Page 16, as equal to:

$$F_{pd} = \frac{1000 \text{ cfs}}{\text{average total plant discharge}}$$

The average total plant discharge of 3,025 gpm is the historical average for the years 1996-1998, and as such is a representative average total plant discharge value.

The term A_{ij} , the ingestion dose factors for any organ, are tabulated in Table 5-1. For simplicity and conservatism, a single maximum organ dose factor for each nuclide was calculated using the critical organ for each nuclide. The following equation was used in calculating the ingestion dose factors (Reference: NUREG-0133, pg. 16):

$$A_{ij} = k_o \left[\frac{U_w}{D_w} + U_F BF_i \right] DF_i \quad (5-2)$$

where

A_{ij} = composite dose parameter for total body or maximum organ of an adult for nuclide i , in mrem/hr per $\mu\text{Ci/ml}$

k_o = conversion factor, $1.14 \times 10^5 = 10^6 \text{ pCi}/\mu\text{Ci} \times 10^3 \text{ ml/kg} \div 8760 \text{ hr/yr}$

5.0 LIQUID EFFLUENT DOSE CALCULATIONS

5.1 INTRODUCTION (Continued)

U_w = 730 kg/yr, adult maximum annual water consumption rate (from Regulatory Guide 1.109, Rev. 1, 10/77, Table E-5)

D_w = dilution factor from the near field concentration to the potable water intake,
 $230 = 230,000 \text{ cfs average river flow} \div 1000 \text{ cfs near field dilution flow}$

U_F = 21 kg/yr, adult maximum annual fish consumption rate (from Regulatory Guide 1.109, Rev. 1, 10/77, Table E-5)

BF_i = bioaccumulation factor for nuclide i, in fish, pCi/kg per pCi/l (from Regulatory Guide 1.109, Rev. 1, 10/77, Table A-1)

DF_i = dose conversion factor for nuclide i, for adults - total body or maximum organ in mrem/pCi ingested (from Regulatory Guide 1.109, Rev. 1, 10/77, Table E-11)

5.0 LIQUID EFFLUENT DOSE CALCULATIONS

5.2 CONTROL 3.2.1.1

This section will be used to demonstrate compliance with Control 3.2.1.1 by providing the calculational methods to use with the results of radioactive analysis required by Surveillance Requirement a. of Control 3.2.1.1.

Once the results of a radioactive analysis are obtained, the fractional ECV(f) should be calculated using the equation:

$$f = \sum_i \frac{A_i}{ECV_i} \quad (5-3)$$

where

A_i = concentration of nuclide i in sample, $\mu\text{Ci/ml}$

ECV_i = 10 CFR 20, Appendix B, Table 2, Column 2 effluent concentration value for nuclide i, $\mu\text{Ci/ml}$

The resulting fractional ECV must be adjusted for plant dilution using the following equation:

$$C = \frac{f \times F_e}{F_p} \quad (5-4)$$

where

C = fraction of Control 3.2.1.1 limit

F_e = liquid radioactive waste discharge flow rate prior to dilution, gpm

F_p = total plant dilution flow rate, gpm

Releases comply with Control 3.2.1.1 if the value of C is ≤ 1.0 .

5.0 LIQUID EFFLUENT DOSE CALCULATIONS

5.2 CONTROL 3.2.1.1 (Continued)

Nuclides which require analysis of monthly or quarterly composite samples (e.g., H-3, Fe-55, Sr-90) are not considered in the calculation required by Control 3.2.1.1 at the time of the release. When the results from these analyses are available, they will be used to confirm that those nuclides, averaged over the sample period, did not cause violation of Control 3.2.1.1.

5.0 LIQUID EFFLUENT DOSE CALCULATIONS

5.3 CONTROL 3.2.1.2

This section will be used to demonstrate compliance with Control 3.2.1.2 at least once per 31 days when the cumulative liquid activity release, excluding tritium, exceeds 2.5 Ci/quarter.

The intermediate surveillance value of 2.5 Ci/quarter, excluding tritium, is a release rate which has been shown during 7 yr of Trojan operation to be significantly below the Control value of 1.5 mrem/quarter total body and 2.5 mrem/quarter to any organ. This is demonstrated in Appendix E.

5.3.1 METHOD 1

The following plant-specific applications of Equation 5-1 will be used in Method 1 should the quarterly release exceed 2.5 Ci/quarter (excluding tritium):

Total Body

$$D_{TB} = \sum_t \frac{\Delta T_t}{F_{pd} V_t} \sum_i A_{TB,i} \times Q_{i,t} \quad (5-5)$$

Maximum Organ

$$D_{MO} = \sum_t \frac{\Delta T_t}{F_{pd} V_t} \sum_i A_{MO,i} \times Q_{i,t} \quad (5-6)$$

where

D_{TB} = cumulative quarterly total body dose incurred to date, mrem

D_{MO} = cumulative quarterly maximum organ dose incurred to date, mrem

5.0 LIQUID EFFLUENT DOSE CALCULATIONS

5.3 CONTROL 3.2.1.2 (Continued)

ΔT_ℓ = the ℓ^{th} time period in a calendar quarter over which the dose is evaluated, hr (i.e., dose for 7-day period has $\Delta T_\ell = 168$ hours)

V_ℓ = volume of total plant discharge flow for time ΔT_ℓ , ml

F_{pd} = plant-specific dilution factor, as defined in Equation 5-1, discussion of parameters follows equation and defines F_{pd}

$A_{TB,i}$ = total body dose parameter for nuclide i , mrem/hr per $\mu\text{Ci/ml}$, see Table 5-1 for values

$A_{MO,i}$ = maximum organ dose parameter for nuclide i , mrem/hr per $\mu\text{Ci/ml}$, see Table 2-1 for values

Q_{it} = activity released of nuclide i , over time period ΔT_ℓ , μCi

Nuclides which require analysis of monthly or quarterly composite samples (e.g., Fe-55, Sr-90) are not considered in the calculation required by the Surveillance Requirements of Control 3.2.1.2, at the time of release. When the results from these analyses are available, they will be used to confirm that those nuclides, averaged over the sample period, did not cause the total liquid release to exceed 2.5 Ci/quarter or the total calculated doses to exceed Control 3.2.1.2.

5.3.2 METHOD 2 (Optional)

Should the dose limits of Control 3.2.1.2 be exceeded using Method 1, a more accurate dose calculation may be made using the methodology in Regulatory Guide 1.109 (Rev. 1, 10/77) to demonstrate compliance.

5.0 LIQUID EFFLUENT DOSE CALCULATIONS

5.4 CONTROL 3.2.1.3

This section is used to demonstrate compliance with Control 3.2.1.3 at least once per 31 days.

The surveillance value of 1.25 Ci/quarter, excluding tritium, is a release rate which has been shown during the first 7 yr of Trojan operation to be significantly below the Control value of 0.75 mrem/quarter total body and 1.25 mrem/quarter to any organ. Should this surveillance value be exceeded, the radwaste treatment systems will be used.

5.0 LIQUID EFFLUENT DOSE CALCULATIONS

5.5 REPORTING REQUIREMENT 4.1.2

This section describes the method that will be used to calculate doses from liquid effluents, as required by ODCM Reporting Requirement 4.1.2 (Annual Radioactive Effluent Release Report).

5.5.1 GENERAL METHODOLOGY

The models of Regulatory Guide 1.109 (Rev. 1, 1977) will be utilized, incorporating Trojan site-specific modeling parameters, to compute doses from liquid effluents for this Control. In addition to the four principal Regulatory Guide 1.109 liquid effluent dose pathways, a PGE-developed swimming immersion dose pathway has been added to include radiation exposure to swimmers in the Columbia River. The PGE computer codes utilized in these calculations are documented, validated and controlled in accordance with written, quality-related procedures.

5.5.2 PLANT/SITE-SPECIFIC ASSUMPTIONS

Hydrologic dilution factors will be based on actual river flow rates and effluent flow rates during the reporting period. Drinking water and agricultural exposure pathways will assume dilution into the full river flow. Other exposure pathways will assume dilution into the plant mixing zone, which is defined as that portion of the river from the Oregon shore to a point 300 ft from the end of the active region of the diffuser pipe.

TABLE 5-1

Liquid Effluent Adult Ingestion Dose Factors (mrem/hr per $\mu\text{Ci/ml}$)		
Nuclide	Total Body $A_{TB,}$	Maximum Organ $A_{MO,}$
H-3	2.7E-1	2.7E-1
Na-22	4.2E+3	4.2E+3
Fe-55	1.1E+2	6.6E+2
Co-60	5.7E+2	4.9E+3
Sr-90	1.4E+5	5.5E+5
Sb-125	1.2E+0	5.5E+1
Cs-134	5.8E+5	7.1E+5
Cs-137	3.5E+5	5.3E+5

Note: Zero in this table is <1.0 except H-3.

6.0 GASEOUS EFFLUENT DOSE CALCULATIONS

6.1 INTRODUCTION

The particulate ($T_{1/2} > 8$ days) dose contributions may be determined using the following general equation:

$$D_{IPC} = 1000 \sum_i R_i \times Q_i \quad (6-1)$$

where

D_{IPC} = dose rate at controlling exposure location, mrem/yr

R_i = dose factor at the site boundary for critical organ and age group, rem/yr per Ci/sec

Q_i = Particulate activity release rate of nuclide i, Ci/sec

Derivation of R_i values is presented in Appendix B. These values are listed in Table B-2.

6.0 GASEOUS EFFLUENT DOSE CALCULATIONS

6.2 CONTROL 3.2.2.1

This section, together with Section 7, will be used to demonstrate compliance with Control 3.2.2.1.

Allowable release rates for batch and continuous releases will be computed such that the dose rate limits of Control 3.2.2.1 are not exceeded. The allowable release rate is computed as follows (based on Equations 5-3, 5-4, and 5-5):

Particulates ($T_{1/2} > 8$ days)

$$Q_v \leq \frac{1}{0.67 \overline{R}_i} \quad (6-2)$$

where

$$Q_v = \sum_i Q_{v_i} = \text{total particulate Ci/sec } (T_{1/2} > 8 \text{ days})$$

$$Q_{v_i} = \text{particulate release rate for nuclide } i, \text{ Ci/sec}$$

$$\overline{R}_i = 1/Q_v \sum Q_{v_i} R_i$$

$$R_i = \text{particulate dose factor for nuclide } i, \text{ rem/yr per Ci/sec}$$

Nuclides which require analysis of monthly or quarterly composite samples (e.g., Sr-90) are not considered in the calculation required by Surveillance Requirement a. of Control 3.2.2.1 at the time of the release. When the results from these analyses are available, they will be used to confirm that those nuclides, averaged over the sample period, did not cause violation of Surveillance Requirement a. of Control 3.2.2.1.

6.0 GASEOUS EFFLUENT DOSE CALCULATIONS

6.3 CONTROL 3.2.2.2

This section is intentionally deleted.

6.4 CONTROL 3.2.2.3

This section will be used to demonstrate compliance with Control 3.2.2.3 at least once per 31 days.

6.4.1 METHOD 1

Utilize the actual particulate ($T_{1/2} > 8$ days) releases to determine compliance with Control 3.2.2.3 as follows:

$$100 Q_v \overline{R}_i < 1 \quad (6-3)$$

where

$$\overline{R}_i = 1/Q_v \sum Q_{vi} R_i$$

$$R_i = \text{dose factor for nuclide } i, \text{ rem/yr per Ci/sec from Table B-2}$$

$$Q_v = \sum_i Q_{vi} = \text{total particulate release rate, Ci/sec}$$

$$Q_{vi} = \text{cumulative quarterly release rate of each particulate nuclide } i, \text{ Ci/sec}$$

Nuclides which require analysis of monthly or quarterly composite samples (e.g., Sr-90) are not considered in the calculation required by Surveillance Requirement a. of Control 3.2.2.3 every 31 days. When the results of these analyses are available, they will be used to confirm that those nuclides, averaged over the sample period, did not cause violation of Surveillance Requirement a. of Control 3.2.2.3.

6.0 GASEOUS EFFLUENT DOSE CALCULATIONS

6.4.2 METHOD 2 (Optional)

Should the dose limits of ODCM Control 3.2.2.3 be exceeded using Method 1, a more accurate dose calculation may be made using the methodology specified in Section 6.7 to demonstrate compliance.

6.0 GASEOUS EFFLUENT DOSE CALCULATIONS

6.5 CONTROL 3.2.2.4

This section will be used to demonstrate compliance with Control 3.2.2.4 at least once per 31 days.

The particulate release rate limits for Control 3.2.2.4 will be determined using the equation listed below. The allowable release rate is calculated by Equation 6-4:

$$Q_v < \frac{1}{200R_i} \quad (6-4)$$

where all parameters have been previously defined.

6.6 CONTROL 3.2.2.5 - TOTAL DOSE

This section describes the methods to be used to determine compliance with Control 3.2.2.5, which requires that the annual (calendar year) dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be less than or equal to 25 mrem to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

6.6.1 SURVEILLANCE REQUIREMENTS

The Surveillance Requirement of Control 3.2.2.5 requires that cumulative dose contributions from liquid effluents and from gaseous effluents shall be determined in accordance with Control 3.2.1.2 and 3.2.2.3 Surveillance Requirements, and in accordance with the methodology and parameters in the ODCM.

These calculations are to be performed in order to determine whether entry into the ACTION statement of Control 3.2.2.5 is required. The ACTION statement is entered when the calculated doses from the releases of radioactive materials in liquid or gaseous effluents exceed twice the limits of Controls 3.2.1.2 or 3.2.2.3.

6.0 GASEOUS EFFLUENT DOSE CALCULATIONS

6.6.2 METHODOLOGY

Dose calculations for the effluent categories of Controls 3.2.1.2 and 3.2.2.3 are to be performed in accordance with the methodology of Sections 5.3 and 6.4, respectively. If any one of these dose limits is exceeded by a factor of two or more, then a specific determination of the actual dose to the likely most exposed real member of the public shall be performed. This evaluation shall include a determination of the total dose from all effluent pathways plus direct radiation contributions from radwaste, etc.

Should the above total dose determination be required, realistic estimates of the specific receptor location and exposure pathways shall be developed in accordance with appropriate NRC guidance.

6.0 GASEOUS EFFLUENT DOSE CALCULATIONS

6.7 REPORTING REQUIREMENT 4.1.2

This section describes the method that will be used to calculate doses from gaseous effluents, as required by Reporting Requirement 4.1.2 (Annual Radioactive Effluent Release Report).

6.7.1 GENERAL METHODOLOGY

The models of Regulatory Guide 1.109 (Rev. 1, 1977) will be utilized, incorporating site-specific modeling parameters, to compute doses from gaseous effluents for this Control.

6.7.2 PLANT/SITE-SPECIFIC ASSUMPTIONS

Meteorological dispersion and deposition factors will be based on historical meteorological data from the Trojan meteorological monitoring system. Separate meteorological factors have been derived for continuous releases. The meteorological model described in Appendix C will be used.

The methodology described in Appendix D will be used to assess the radiation doses from radioactive effluents to individuals due to their activities in UNRESTRICTED AREAS during the reporting period. The results will be reported in the annual Radioactive Effluent Release Report.

7.1 EFFLUENT MONITOR SETPOINT CALCULATIONS

7.1 LIQUID EFFLUENT MONITORING

This section ensures compliance with Control 3.2.1.1.

Liquid radioactive waste will be batch discharged to the Columbia River through the Discharge and Dilution Structure (D&DS). The liquid waste will be stored in tanks. Prior to release to the river, the tank contents shall be sampled and analyzed in accordance with Table 3.2.1.1-1 and a technically qualified member of the Facility Staff approves the release rate calculation.

7.2 GASEOUS EFFLUENT MONITORS

This section will be used to ensure compliance with Control 3.2.2.1.

To ensure compliance with Control 3.2.2.1, Auxiliary/Fuel Building and Condensate Demineralizer Building samples will be analyzed monthly. Due to the limited effluent volume discharged from the buildings, it is not necessary to set release limits for these pathways.

TABLE 7-1

Historical Particulate Releases			
<u>Year</u>	<u>Quarter</u>	<u>Curies Released</u>	
		<u>Sr-90</u>	<u>Total</u>
1977	1	5.2E-4	1.3E-2
	2	8.2E-5	1.3E-2
	3	2.9E-5	1.2E-3
	4	2.8E-5	3.7E-4
1978	1	2.7E-4	3.6E-3
	2	7.9E-5	2.0E-3
	3	7.8E-5	5.4E-4
	4	5.2E-5	3.8E-4
1979	1	1.1E-5	5.3E-3
	2	3.8E-5	4.4E-3
	3	5.7E-5	8.4E-4
	4	1.2E-4	9.3E-3
1980	1	5.1E-6	1.4E-3
	2	5.4E-5	1.1E-2
	3	5.8E-5	6.9E-4
	4	4.6E-6	8.1E-4
1981	1	9.0E-6	1.5E-2
	2	1.4E-5	2.1E-2
	3	5.2E-6	9.7E-4
	4	1.1E-5	2.0E-3
1982	1	2.6E-5	1.8E-3
	2	4.5E-4	4.3E-3
	3	3.6E-5	8.7E-4
	4	3.7E-4	2.1E-3
1983	1	1.1E-4	2.4E-3
	2	3.6E-4	9.4E-4
	3	5.5E-5	4.6E-4
	4	5.5E-5	5.0E-4
1984	1	8.2E-5	2.1E-3
	2	8.7E-5	2.2E-3
	3	7.4E-5	5.2E-4
	4	5.1E-5	6.1E-4
1985	1	6.6E-5	2.0E-4
	2	1.0E-8	2.5E-5
	3	5.6E-5	2.0E-4
	4	4.9E-7	1.9E-6

8.0 TROJAN PROCESS CONTROL PROGRAM FOR SOLID RADIOACTIVE WASTE

This chapter will be used to ensure compliance with Control 3.2.3.1 and the waste form requirements of 10 CFR 61.56.

8.1 PURPOSE

To verify that processed radioactive wastes to be shipped offsite for burial meet the shipping and burial ground requirements for solidification and dewatering.

8.2 PROCESS CONTROL PROGRAM FOR STABILIZING RADIOACTIVE WASTE BY SOLIDIFICATION

8.2.1 SCOPE

This section pertains to radioactive waste containing a total specific activity which exceeds the concentration limits for Class A waste as defined in 10 CFR 61 or requires a change in waste form to meet specific disposal site requirements. These wastes must be stabilized by solidification and contain no freestanding liquids (as defined by applicable regulations or license conditions) prior to shipment offsite for burial, or else be packaged in a high-integrity container in accordance with Section 8.3.

8.2.2 PROGRAM ELEMENTS

For the disposal of radioactive waste requiring solidification, PGE shall implement the following steps:

- (1) An NRC-approved contract vendor solidification service shall be utilized. The contract vendor solidification service may consist of solidification by the contractor or supply of materials, procedures, and process control program for PGE solidification.

8.0 TROJAN PROCESS CONTROL PROGRAM FOR SOLID RADIOACTIVE WASTE

8.2.2 PROGRAM ELEMENTS (Continued)

- (2) This vendor service shall include transmittal to PGE of copies of their solidification procedure and process control program prior to performing the solidification.
- (3) The process parameters included in the process control program may include, but are not limited to, waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.
- (4) The vendor solidification procedure and process control program shall be incorporated into a Plant Operating Manual procedure that will be effective during the solidification process. This procedure will identify all plant interfaces with the vendor's equipment (e.g., flush water, fire protection shielding requirements, etc.), as well as identify the actions to be taken if excess free liquids are observed. This procedure shall require at least one representative test specimen from at least every tenth batch of waste processed to ensure solidification. The procedure should also include the actions to be taken if the test specimen fails to solidify.
- (5) This procedure shall be reviewed per plant procedures for adequacy in meeting applicable State, Federal, Department of Transportation, and burial ground regulatory requirements and approved by the General Manager, Trojan or designee prior to its implementation. This review shall ensure that the stability requirements of 10 CFR 61.56(b) for wastes exceeding Class A concentrations are met by the vendor solidification program.

8.0 TROJAN PROCESS CONTROL PROGRAM FOR SOLID RADIOACTIVE WASTE

8.3 PROCESS CONTROL PROGRAM FOR RADIOACTIVE WASTE PACKAGED IN HIGH-INTEGRITY CONTAINERS

8.3.1 SCOPE

This section pertains to radioactive waste containing a specific activity which exceeds the concentration limits for Class A waste as defined in 10 CFR 61 or requires the stability afforded by a high integrity container to be acceptable for a specific disposal site. These wastes must be stabilized by packaging in dewatered form in a high-integrity container which meets burial ground and regulatory requirements, or else be solidified in accordance with Section 8.2.

8.3.2 PROGRAM ELEMENTS

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

- (1) A contract vendor high-integrity container shall be used.
- (2) The container shall be demonstrated to have been authorized by the State of Washington prior to acceptance for use by PGE. This shall include provision by the vendor to PGE of documentation reflecting this authorization.
- (3) The material placed in the high-integrity container shall meet all applicable burial ground and regulatory waste form requirements for waste which is packaged in this manner.
- (4) The above criteria shall be met by following plant procedures which will be reviewed and approved by the General Manager, Trojan or designee in accordance with plant administrative procedures prior to implementation at the time of packaging and disposal.

8.0 TROJAN PROCESS CONTROL PROGRAM FOR SOLID RADIOACTIVE WASTE

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

8.4.1 SCOPE

This section pertains to bead-type spent radioactive demineralizer resin and other wet wastes, such as absorbed oils, which do not exceed the concentration limits for Class A waste as defined in 10 CFR 61, but may have specific waste form and/or packaging requirements in the license conditions for a specific disposal site.

8.4.2 PROGRAM ELEMENTS

- (1) The dewatered resin or wet wastes must meet the requirements of 10 CFR 61.56 or those of the burial ground (whichever is more restrictive) for freestanding, noncorrosive liquid.
- (2) For bead resins, the preceding criterion will be met by following approved Plant Operating Manual procedures for dewatering resin.
- (3) Liquid waste other than oil must be solidified or packaged in sufficient absorbent material to absorb twice the volume of liquid. Oil must be solidified.

8.5 SUPPORTING DOCUMENTS

The following types of procedures are used in support of the process control program. Vendor procedures are retained and maintained by Radiation Protection:

- (1) PGE-Trojan and Vendor Procedures⁽¹⁾:

Radioactive Waste Shipment Procedures

Radioactive Waste Packaging Procedures

8.0 TROJAN PROCESS CONTROL PROGRAM FOR SOLID RADIOACTIVE WASTE

8.5 SUPPORTING DOCUMENTS (Continued)

Radioactive Waste Classification Procedures

10 CFR 61 Sampling Program Procedures

- (1) Vendor procedures incorporated into Trojan procedures are specifically referenced in the procedures using them.

8.6 PROGRAM CHANGES

Changes to the PCP shall be documented and records of reviews performed shall be retained.

This documentation shall contain sufficient information to support the change(s) and appropriate analyses or evaluations justifying the change(s); and a determination that the change(s) maintain the overall conformance of the solidified waste product to the existing requirements of Federal, State, or other applicable regulations.

Changes to the PCP shall be effective after review and approval by an Independent Safety Reviewer and the approval of the General Manager, Trojan or designee.

APPENDIX A

This appendix intentionally deleted.

APPENDIX B

DERIVATION OF PARTICULATE DOSE FACTORS

DOSE FACTOR R_i

The term R_i is based on the combination of: (a) inhalation, ground plane, vegetable ingestion, meat ingestion and milk ingestion pathways which are present at the location of maximum potential dose (i.e., the controlling exposure location), (b) annual average continuous release meteorology at the controlling exposure location, (c) the most restrictive age group (child), and (d) the critical organ for each nuclide.

Determination of the Site Boundary as the Controlling Exposure Location

The controlling exposure location is that offsite location where the combination of existing pathways and annual average meteorology would indicate the maximum potential dose. That is, the controlling exposure individual is assumed to breathe the air at the nearest residence with the highest χ/Q value, to reside at the nearest residence with the highest D/Q value, and to obtain all the individual's vegetables, meat, and milk from the production locations with the highest D/Q values. To be conservative, it is assumed that the controlling exposure location for all pathways is the site boundary.

The meteorology at the site boundary is discussed in Appendix C and is listed in Table C-1.

The following general equation is used to calculate R_i values:

$$R_i = 10^{-3} [(R_i^I \times \chi/Q_c) + (R_i^G \times D/Q_c) + (R_i^V \times D/Q_c) + (R_i^M \times D/Q_c) \times (R_i^C \times D/Q_c)] \quad (B-1)$$

where

R_i = total dose factor for nuclide i , rem/yr per Ci/sec

R_i^I = inhalation pathway dose factor for nuclide i, mrem/yr per Ci/m³

R_i^G = ground plane pathway dose factor for nuclide i, mrem/yr per Ci/m²-sec

R_i^V = vegetable ingestion pathway dose factor for nuclide i, mrem/yr per Ci/m²-sec

R_i^M = meat ingestion pathway dose factor for nuclide i, mrem/yr per Ci/m²-sec

R_i^C = cow or goat milk ingestion pathway dose factor for nuclide i, mrem/yr per Ci/m²-sec

χ/Q_c = atmospheric dispersion factor for continuous releases at the site boundary, sec/m³

D/Q_c = atmospheric deposition factor for continuous releases at the site boundary, m⁻²

10^{-3} = constant, rem/mrem.

The dose factors, $R_i^I, R_i^G, R_i^V, R_i^M, R_i^C$, were derived as follows and are listed in Table B-1.

Inhalation Pathway Dose Factor R_i^I

$$R_i^I = 10^{12} (BR) (DFA_i) \quad (B-2)$$

where

10^{12} = constant, pCi/Ci

(BR) = breathing rate of the receptor of child age group = 3700 m³/yr (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-5)

(DFA_i) = maximum organ inhalation dose factor for the receptor for nuclide *i*, in mrem/pCi (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-9). The total body is considered as an organ in the selection of the DFA_i.

Ground Plane Pathway Factor R_i^G (mrem/yr per Ci/m² - sec)

$$R_i^G = (10^{12}) (8760) (SF) (DFG_i) [(1 - e^{-\lambda_i t}) / \lambda_i] \quad (B-3)$$

where

10¹² = constant, pCi/Ci

8760 = constant, hr/yr

λ_i = decay constant for nuclide *i*, sec⁻¹

t = exposure time, 4.73 x 10⁸ sec (15 yr)

(DFG_i) = ground plane total body dose conversion factor for nuclide *i*, mrem/hr per pCi/m² (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-6)

SF = shielding factor for residential structures, 0.7 (from Regulatory Guide 1.109, Appendix C).

Vegetation Pathway Factor R_i^V (mrem/yr per Ci/m² - sec)

Man is considered to consume two types of vegetation, fresh leafy vegetables and produce. The vegetation dose factor combines these two pathways using the following equation:

$$R_i^V = 10^{12} \left[\frac{(r)f_e}{Y_v(\lambda_i + \lambda_w)} \right] (DFL_i) [U_a^L f_L e^{-\lambda_i t} + U_a^S f_S e^{-\lambda_i t}] \quad (B-4)$$

where

- U_a^L = consumption rate of fresh leafy vegetation by the child receptor, 26 kg/yr (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-5)
- U_a^S = consumption rate of produce and stored vegetation by the child receptor, 520 kg/yr (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-5)
- f_L = fraction of the annual intake of fresh leafy vegetation grown locally, 1.0 (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)
- f_g = fraction of the annual intake of produce and stored vegetation grown locally, 0.76 (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)
- t_L = average time between harvest of leafy vegetation and its consumption, 8.6×10^4 sec (1 day) (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)
- t_h = average time between harvest of stored vegetation and its consumption, 5.2×10^6 sec (60 day) (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)
- Y_v = vegetation area density, 2.0 kg/m^2 (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)
- f_e = fraction of radionuclides that is elemental iodine, 0.5 for radioiodines, 1.0 otherwise (Regulatory Guide 1.109, Revision 1, 10/77, Appendix C)
- r = fraction of deposited activity retained on vegetation, 0.2 for particulates (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)
- λ_w = decay constant for removal of activity on leaf and plant surfaces by weathering, $5.73 \times 10^{-7} \text{ sec}^{-1}$ (corresponding to a 14-day half life) (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

(DFL_i) = maximum organ ingestion dose factor for nuclide i, in mrem/pCi
(Regulatory Guide 1.109, Rev. 1, 10/77, Table E-13)

10^{12} = constant, pCi/Ci

and all other terms have been defined previously.

Grass-Cow-Meat Pathway Factor $R_i^M = (\text{mrem/yr per Ci/m}^2 - \text{sec})$

$$R_i^M = 10^{12} \frac{Q_F(U_{ap})}{\lambda_i + \lambda_w} (F_f)(r)(f_e)(DFL_i) \left[\frac{f_p f_s}{Y_p} + \frac{(1 - f_p f_s)e^{-\lambda_i t_h}}{Y_s} \right] e^{-\lambda_i t_r} \quad (\text{B-6})$$

where

F_f = stable element transfer coefficient for meat, in days/kg (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-1)

U_{ap} = child receptor's meat consumption rate, 41 kg/yr (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-5)

t_r = transport time from pasture to receptor, 1.73×10^6 sec (20 days) (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

t_h = transport time from crop field to receptor, 7.78×10^6 sec (90 days) (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

Y_s = agricultural productivity by unit area (stored food), 2.0 kg/m^2 (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

f_p = fraction of year that cow is on pasture, 0.5 (Regulatory Guide 1.109, Rev. 0, Page 1.109-26)

f_s = fraction of cow feed that is pasture grass while cow is on pasture, 1.0

Q_F = cows' consumption rate of feed, 50 kg/day (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-3)

Y_p = agricultural productivity by unit area (pasture), 0.7 kg/m^2 (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15).

$$10^{12} = \text{constant, pCi/Ci}$$

and all other terms have been defined previously.

Grass-Goat-Milk Pathway Factor R_i^C (mrem/yr per Ci/m² - sec)

$$R_i^C = 10^{12} \frac{Q_F(U_{ap})}{\lambda_i + \lambda_w} (F_m)(r)(f_c)(DFL_i) \left[\frac{f_p f_s}{Y_p} + \frac{(1 - f_p f_s)e^{-\lambda_i t_h}}{Y_s} \right] e^{-\lambda_i t_r} \quad (B-8)$$

where

Q_F = goat's consumption rate of feed, 6 kg/day (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-3)

U_{ap} = child receptor's milk consumption rate, 330 l/yr (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-5)

Y_s = agricultural productivity by unit area of stored feed, 2.0 kg/m² (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

F_m = stable element transfer coefficient for milk, in days/l (Regulatory Guide 1.109, Rev. 1, 10/77, Tables E-1 and E-2)

r = fraction of deposited activity retained on goat's feed grass, 0.2 for particulates (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

t_f = transport time from pasture to goat, to milk, to receptor, 1.73×10^5 sec (2 days)
(Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

t_h = transport time from pasture, to harvest, to goat, 7.78×10^6 sec (90 days)
(Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

10^{12} = constant, pCi/Ci

and all other terms have been previously defined.

Grass-Cow-Milk Pathway Factor R_i^c (mrem/yr per Ci/m² - sec)

$$R_i^c = 10^{12} \frac{Q_F(U_{ap})}{\lambda_i + \lambda_w} (F_m)(r)(f_e)(DFL_i) \left[\frac{f_p f_s}{Y_p} + \frac{(1 - f_p f_s)e^{-\lambda_i t_h}}{Y_s} \right] e^{-\lambda_i t_f} \quad (B-9)$$

where

Q_F = cow's consumption rate of feed, 50 kg/day (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-3)

U_{ap} = child receptor's milk consumption rate, 330 l/yr (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-5)

Y_s = agricultural productivity by unit area of stored feed, 2.0 kg/m² (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

F_m = stable element transfer coefficient for milk, in days/l (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-1)

r = fraction of deposited activity retained on cow's feed grass, 0.2 for particulates
(Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

t_f = transport time from pasture to cow, to milk, to receptor, 1.73×10^5 sec (2 days)
(Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

t_h = transport time from pasture, to harvest, to cow, 7.78×10^6 sec (90 days)
(Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

10^{12} = constant, pCi/Ci

and all other terms have been previously defined.

TABLE B-1

Dose Factors for Controlling Exposure Location												
Nuclide	λ_i (sec^{-1})	Child Dose Factors			Retention Factor (r)	Stable Element Transfer Coefficients						
						Dose Parameters						
		DFA _i (mrem) (pCi)	DFG _i (mrem/hr) (pCi/m ²)	DFL _i (mrem) (pCi)		Meat F _f (day) (kg)	Milk F _m (day) (l)	Inhalation R _i ^I (mrem/yr) (Ci/m ³)	Ground R _i ^G (m ² -mrem/yr) (Ci/sec)	Vegetable R _i ^V (m ² -mrem/yr) (Ci/sec)	Meat R _i ^M (m ² -mrem/yr) (Ci/sec)	Milk R _i ^C (m ² -mrem/yr) (Ci/sec)
Fe-55	8.45E-09	3.00E-05	0.0	1.15E-05	0.2	4.0E-02	1.3E-04	1.11E+11	0.0	7.99E+14	3.03E+14	8.04E+12
Co-60	4.18E-09	1.91E-03	1.70E-08	2.93E-05	0.2	1.3E-02	1.0E-03	7.07E+12	2.15E+16	2.10E+15	2.57E+14	1.60E+14
Sr-90	7.94E-10	2.73E-02	0.0	1.70E-02	0.2	6.0E-04	1.4E-02	1.01E+14	0.0	1.24E+18	7.01E+15	1.32E+18
Cs-134	1.07E-08	2.74E-04	1.20E-08	3.84E-04	0.2	4.0E-03	3.0E-01	1.01E+12	6.82E+15	2.63E+16	1.00E+15	6.14E+17
Cs-137	7.30E-10	2.45E-04	4.20E-09	3.27E-04	0.2	4.0E-03	3.0E-01	9.07E+11	1.03E+16	2.39E+16	8.99E+14	5.43E+17

[a] $\frac{\text{mrem/yr}}{\text{Ci/m}^3}$

TABLE B-2

Particulate Dose Factors	
<u>Nuclide</u>	R_I Composite Dose Factor at Site Boundary <u>(rem/yr)</u> (Ci/sec)
Fe-55	7.31E+04
Co-60	1.64E+06
Sr-90	9.26E+07
Cs-134	7.02E+06
Cs-137	6.53E+06

APPENDIX C

METEOROLOGY

QUARTERLY AVERAGE METEOROLOGY

Meteorology data required for the compilation of the radioactive release reports in Control 4.1.2 was calculated at the end of each calendar quarter from 1976 until 1993 using the NRC Computer code XOQDOQ and the methodology of Regulatory Guide 1.111 (Rev. 1, 7/77).

The maximum site boundary X/Q and D/Q values for continuous releases for each quarter from 1976 until 1993 are presented in Table C-1. Table C-1 includes the calculated average X/Q and D/Q values for the entire time period. These average values are to be used for the compilation of radioactive release reports in ODCM Control 4.1.2.

TABLE C-1

Sheet 1 of 2

Historical Meteorological Data Continuous Release					
Quarter	Year	Direction	X/Q	D/Q	PDF
1	1976	N	6.10E-05	5.70E-07	0.85
2	1976	N	3.60E-05	2.20E-07	0.85
3	1976	N	1.80E-05	1.00E-07	0.85
4	1976	N	3.70E-05	2.00E-07	0.85
1	1977	N	5.40E-05	3.00E-07	0.85
2	1977	N	3.80E-05	1.50E-07	0.85
3	1977	N	6.20E-06	2.90E-08	0.92
4	1977	N	8.60E-06	6.00E-08	0.92
1	1978	NNW	1.20E-05	6.50E-08	0.92
2	1978	N	1.10E-05	3.30E-08	0.92
3	1978	N	6.70E-06	3.10E-08	0.93
4	1978	NNW	1.30E-05	5.10E-08	0.92
1	1979	NNW	1.10E-05	7.00E-08	0.92
2	1979	N	5.90E-06	2.00E-08	0.92
3	1979	ESE	4.80E-06	2.10E-08	0.91
4	1979	N	1.40E-05	8.10E-08	0.92
1	1980	NNW	1.20E-05	6.10E-08	0.92
2	1980	N	8.50E-06	2.60E-08	0.92
3	1980	ESE	4.50E-06	1.90E-08	0.91
4	1980	N	1.40E-05	6.90E-08	0.92
1	1981	N	1.40E-05	5.30E-08	0.92
2	1981	N	9.60E-06	3.60E-08	0.92
3	1981	N	2.90E-06	1.80E-08	0.92
4	1981	N	8.80E-06	7.20E-08	0.92
1	1982	N	9.20E-06	7.10E-08	0.92
2	1982	NNW	5.70E-06	3.60E-08	0.92
3	1982	ESE	6.30E-06	2.60E-08	0.91
4	1982	N	1.10E-05	6.00E-08	0.92
1	1983	N	1.40E-05	7.90E-08	0.92
2	1983	N	6.50E-06	2.30E-08	0.92
3	1983	N	7.00E-06	2.20E-08	0.92
4	1983	N	1.30E-05	7.30E-08	0.92
1	1984	NNW	1.20E-05	8.00E-08	0.92
2	1984	N	1.10E-05	4.10E-08	0.92
3	1984	N	1.0E-06	1.50E-08	0.92
4	1984	N	1.50E-05	8.20E-08	0.92
1	1985	NNW	1.20E-05	6.40E-08	0.92
2	1985	N	6.00E-06	1.90E-08	0.92
3	1985	ENE	7.20E-06	9.30E-08	0.92
4	1985	N	1.40E-05	7.00E-08	0.92
1	1986	N	1.30E-05	5.50E-08	0.92
2	1986	N	9.00E-06	2.80E-08	0.92
3	1986	ENE	7.30E-06	1.00E-08	0.92
4	1986	N	1.10E-05	5.60E-08	0.92

TABLE C-1

Sheet 2 of 2

Historical Meteorological Data Continuous Release					
Quarter	Year	Direction	X/Q	D/Q	PDF
1	1987	N	1.40E-05	6.20E-08	0.92
2	1987	N	5.80E-06	2.00E-08	0.92
3	1987	ESE	4.80E-06	1.50E-08	0.91
4	1987	N	1.40E-05	5.10E-08	0.92
1	1988	N	1.30E-05	6.10E-08	0.92
2	1988	N	9.00E-06	2.50E-08	0.92
3	1988	ESE	1.10E-05	2.90E-08	0.91
4	1988	N	1.90E-05	6.50E-08	0.92
1	1989	NNW	1.50E-05	8.80E-08	0.92
2	1989	E	9.10E-06	1.60E-08	0.92
3	1989	ESE	7.80E-06	2.70E-08	0.91
4	1989	N	1.30E-05	6.10E-08	0.92
1	1990	N	1.20E-05	6.00E-08	0.92
2	1990	N	9.40E-06	3.10E-08	0.92
3	1990	ESE	5.10E-06	2.10E-08	0.91
4	1990	N	1.60E-05	8.50E-08	0.92
1	1991	NNW	1.20E-05	7.70E-08	0.92
2	1991	N	6.50E-06	2.60E-08	0.92
3	1991	ESE	4.80E-06	2.30E-08	0.91
4	1991	NNW	1.20E-05	7.70E-08	0.92
1	1992	NNW	1.40E-05	7.10E-08	0.92
2	1992	E	6.30E-06	2.00E-08	0.92
3	1992	ESE	4.50E-06	2.00E-08	0.91
4	1992	N	1.30E-05	8.00E-08	0.92
1	1993	NNW	1.50E-05	6.80E-08	0.92
2	1993	NNW	1.10E-05	4.50E-08	0.92
3	1993	ESE	6.70E-06	2.20E-08	0.91
4	1993	NNW	9.50E-06	3.40E-08	0.91
AVERAGE			1.25E-05	6.44E-08	0.91

APPENDIX D

METHODOLOGY FOR DETERMINING DOSES TO PERSONS UTILIZING UNRESTRICTED AREAS WITHIN THE SITE EXCLUSION AREA BOUNDARY

Noble gas doses are directly proportional to the atmospheric dispersion factor (χ/Q , sec/m^3). The methodology contained in this Appendix assumes quarterly average meteorology and ground-level release sector average χ/Q (Meteorology and Atomic Energy Equation 3.144). The following equation is used to calculate the adjusted χ/Q values for specific locations within the site boundary:

$$\chi/Q^A (\text{sec}/\text{m}^3) = \left[\frac{2}{\pi} \right]^{1/2} \left[\frac{(0.01) (f)}{\sigma_z \bar{\mu} (2\pi \frac{x}{n})} \right] [\text{OF}] \quad \text{Equation (D-1)}$$

where

f = wind frequency, percent

$\bar{\mu}$ = mean wind speed, meters/sec

x = downwind distance, meters

n = number of cardinal compass sectors = 16

σ_z = vertical dispersion parameter, Pasquill Class E

OF = occupancy factor = $\frac{\text{hours of annual occupancy}}{8760}$

NOTE: Occupancy factors for both recreational and occupational cases should be considered.

To obtain doses at these locations, multiply the calculated doses at the site boundary by the ratio of the adjusted atmospheric dispersion factor (χ/Q) at the location of interest to the χ/Q at the site boundary:

$$D^A = \left[\frac{(\chi/Q)^A}{(\chi/Q)^{SB}} \right] [D^{SB}] \quad \text{Equation (D-2)}$$

where

D^A = dose at the location of interest (mrem)

$(\chi/Q)^A$ = adjusted χ/Q at the location of interest (from Equation D-1) (sec/m^3)

$(\chi/Q)^{SB}$ = χ/Q at the site boundary (sec/m^3)

D^{SB} = dose at site boundary (mrem).

The methodology described in this Appendix was used to determine doses to persons utilizing unrestricted areas within the site exclusion area boundary beginning in the third quarter of 1992 and continuing through the fourth quarter of 1993. Table D-1 summarizes the correction factors calculated during this time period.

TABLE D-1

Correction Factor for Persons Utilizing Unrestricted Areas Within the Site Exclusion Area Boundary		
QUARTER	YEAR	CONTINUOUS $\frac{(\chi/Q)^A}{(\chi/Q)^{SB}}$
3	1992	2.8
4	1992	1.9
1	1993	1.5
2	1993	2.4
3	1993	4.0
4	1993	1.6
MAXIMUM		4.0
MINIMUM		1.5
AVERAGE		2.4

APPENDIX E

BASIS FOR CURIE RELEASE VALUES UTILIZED IN LIQUID EFFLUENT SURVEILLANCE REQUIREMENTS

This appendix demonstrates that Control 3.2.1.2, Radioactive Effluents (Liquid) Dose, based on the total curies released (excluding tritium and dissolved gases), results in offsite doses significantly below the Control limits of 1.5 mrem total body/calendar quarter and 2.5 mrem to maximum organ/calendar quarter.

Table E-1 presents the maximum calculated dose due to liquid effluents from Trojan for the period 1976-1985 (reference PGE-1015 dated March 1977 and PGE-1015 for 1978 through 1985).

Columns 4 and 5 of Table E-1 show the offsite dose which would have resulted if Trojan had released 2.5 Ci in each of the quarters during 1976-1985. It can be seen that in no case would 2.5 mrem to maximum organ or 1.5 mrem total body have been exceeded. In fact, during the quarter with the highest mrem/Ci released factor, the offsite doses were <10 percent of the Control limits of 1.5 mrem total body and 2.5 mrem maximum organ.

TABLE E-1

Sheet 1 of 4

Calculated Aquatic Dose Due to Liquid Releases						
Year	Qtr.	Curies Released ^[a] During Calendar Qtr.	Max. Organ Dose ^[b] (mrem)	Max. Body Dose ^[b] (mrem)	Max. Organ Dose mrem/2.5 Ci Released	Total Body Dose mrem/2.5 Ci Released
1976	1	2.6E-1	1.6E-3	5.6E-4	1.5E-2	5.4E-3
	2	9.4E-1	2.4E-3	2.9E-4	6.4E-3	7.7E-4
	3	8.9E-1	2.4E-3	6.1E-4	6.7E-3	1.7E-3
	4	6.2E-1	3.8E-3	5.5E-4	1.5E-2	2.2E-3
1977	1	4.5E-1	9.5E-3	9.3E-4	5.3E-2	5.2E-3
	2	2.65E+0	7.4E-3	1.9E-3	7.0E-3	1.8E-3
	3	1.0E+0	4.9E-3	1.4E-3	1.2E-2	3.5E-3
	4	9.2E-2	9.9E-4	3.3E-4	2.7E-2	9.0E-3
1978	1	2.79E-1	3.0E-3	7.0E-4	2.7E-2	6.3E-3
	2	1.65E-1	1.0E-3	2.2E-4	1.5E-2	3.3E-3
	3	7.82E-2	1.5E-3	4.1E-4	4.8E-2	1.3E-2
	4	1.85E-1	3.6E-3	6.1E-4	4.9E-2	8.2E-3

TABLE E-1

Sheet 2 of 4

Calculated Aquatic Dose Due to Liquid Releases						
Year	Qtr.	Curies Released ^[a] During Calendar Qtr.	Max. Organ Dose ^[b] (mrem)	Max. Body Dose ^[b] (mrem)	Max. Organ Dose mrem/2.5 Ci Released	Total Body Dose mrem/2.5 Ci Released
1979	1	1.41E-1	2.2E-3	3.0E-4	3.9E-2	5.3E-3
	2	4.56E-2	3.3E-4	9.6E-5	1.8E-2	5.3E-3
	3	4.10E-2	6.3E-4	1.9E-4	3.8E-2	1.2E-2
	4	3.27E-1	7.6E-3	3.9E-4	5.8E-2	3.0E-3
1980	1	1.27E-1	2.4E-3	2.1E-4	4.7E-2	4.1E-3
	2	3.81E-1	[c]	[c]	[c]	[c]
	3	1.01E-1	[c]	[c]	[c]	[c]
	4	1.78E-1	1.4E-3	6.1E-4	2.0E-2	8.6E-3
1981	1	2.65E-1	1.1E-2	1.1E-3	1.0E-1	1.0E-2
	2	3.18E-1	[c]	[c]	[c]	[c]
	3	2.18E-1	5.8E-3	2.1E-3	6.7E-2	2.4E-2
	4	1.93E-1	7.2E-3	7.0E-3	9.3E-2	9.1E-2

TABLE E-1

Sheet 3 of 4

Calculated Aquatic Dose Due to Liquid Releases						
Year	Qtr.	Curies Released ^[a] During Calendar Qtr.	Max. Organ Dose ^[b] (mrem)	Max. Body Dose ^[b] (mrem)	Max. Organ Dose mrem/2.5 Ci Released	Total Body Dose mrem/2.5 Ci Released
1982	1	2.98E-1	1.0E-2	5.0E-3	8.4E-2	4.2E-2
	2	2.15E-1	5.8E-3	6.7E-4	6.7E-2	7.8E-3
	3	2.64E-1	1.9E-2	6.9E-4	1.8E-1	6.5E-3
	4	7.89E-2	5.3E-3	3.2E-4	1.7E-1	1.0E-2
1983	1	4.63E-2	1.3E-3	3.3E-4	7.0E-2	1.8E-2
	2	9.81E-2	1.5E-3	1.3E-4	3.8E-2	3.3E-3
	3	1.07E-1	1.9E-3	1.7E-3	4.4E-2	4.0E-2
	4	5.89E-2	1.2E-3	1.1E-3	5.1E-2	4.7E-2
1984	1	6.18E-2	1.3E-3	1.3E-3	5.3E-2	5.3E-2
	2	1.07E-1	1.1E-3	2.3E-4	2.6E-2	5.4E-3
	3	6.96E-2	2.8E-3	2.9E-4	1.0E-1	1.0E-2
	4	1.11E-1	4.4E-3	2.0E-4	9.9E-2	4.5E-3

TABLE E-1

Sheet 4 of 4

Calculated Aquatic Dose Due to Liquid Releases						
Year	Qtr.	Curies Released ^[a] During Calendar Qtr.	Max. Organ Dose ^[b] (mrem)	Max. Body Dose ^[b] (mrem)	Max. Organ Dose mrem/2.5 Ci Released	Total Body Dose mrem/2.5 Ci Released
1985	1	7.57E-2	1.1E-3	6.5E-4	3.6E-2	2.2E-2
	2	1.54E-1	1.4E-3	1.4E-3	2.3E-2	2.3E-2
	3	8.70E-2	1.6E-3	1.6E-3	4.6E-2	4.6E-2
	4	1.48E-1	1.6E-3	1.6E-3	2.7E-2	2.7E-2

[a] Excluding tritium and dissolved gases

[b] Including tritium and dissolved gases

[c] Doses during these quarters resulted from a release path no longer available at Trojan

APPENDIX F

QUALITY ASSURANCE REQUIREMENTS FOR THE ENVIRONMENTAL AND EFFLUENT MONITORING PROGRAM

Environmental and effluent monitoring is a quality-related activity. The Trojan Quality Assurance Program is applicable to the following areas:

- ◆ Organization
- ◆ QA program and training
- ◆ Design control
- ◆ Procurement
- ◆ Instructions, procedures and drawings
- ◆ Document control
- ◆ Purchased material, equipment and services
- ◆ Identification of materials
- ◆ Testing
- ◆ M & TE
- ◆ Handling, storage and shipping
- ◆ Inspection, test and operating status
- ◆ Nonconforming materials
- ◆ Corrective actions
- ◆ QA records
- ◆ Audits