PGE

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Trojan Nuclear Plant
Docket 50-344
License NPF-1

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

PGE-1006-2001, Trojan Nuclear Plant
Annual Radiological Environmental Monitoring Report for 2001

This letter transmits Portland General Electric Company's Trojan Nuclear Plant Annual Radiological Environmental Monitoring Report for the calendar year 2001. This report is submitted in accordance with Trojan Permanently Defueled Technical Specification (PDTS) 5.8.1.2 and Sections IV.B.2, IV.B.3, and IV.C of Appendix I to Title 10 CFR 50. A copy of PGE-1021, "Offsite Dose Calculation Manual," is attached to the report as Appendix A, pursuant to PDTS 5.7.2.3.2

Sincerely,

Stephen M. Quennoz

Stephen M. Quennoz
Vice President, Power Supply/Generation

Enclosure

c: Director, NRC Region IV, DNMS
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IE25

PGE-1006-2001

# TROJAN NUCLEAR PLANT
# RADIOLOGICAL ENVIRONMENTAL
# MONITORING REPORT

January through December 2001

April 2002

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, 2001, through December 31, 2001.

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 2001.

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

Fifteen (15) sampling locations were used in the Radiological Environmental Monitoring Program from January 1, 2001, through December 31, 2001. 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 (now known as the Oregon Office of Energy) on September 22, 1994, for approval. The revisions to the program were approved on December 12, 1994.

## 2.2 SAMPLING PROCEDURES

### 2.2.1 AIR PARTICULATE

Air particulate sampling is performed weekly. The samples are gathered with a low-volume air sampling device which is designed to draw a constant flow rate regardless of the pressure drop across the filter. The sampling devices are set to maintain one cfm. The sample pump, metering devices, and timer are in a weatherproof housing. The filter is located in a sample housing that is connected to an air inlet about one meter above the ground. Glass fiber filters are used to collect particulate matter.

The glass fiber filter is removed from the air sampler and placed in a two-inch plastic petri dish. Air flow readings and other data required to compute the levels of radioactivity are recorded and submitted to the analysis laboratory along with the samples.

### 2.2.2 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. Two TLD cards are placed in each container.[1] The time of collection, the exposure period, and any abnormal conditions such as moisture in the holders, damage done

[1] The practice of placing two TLDs in each container was discontinued beginning in 2002. A second TLD does not provide any additional useful information.

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.

### 2.2.3 WELL WATER

Well water is collected quarterly from the tap that leads off the pump.[2] The line is purged for about five minutes prior to collection. Sixty milliliters are drawn from the one-gallon sample for tritium analysis. The remainder of the sample is put in a one-gallon polyethylene bottle and acidified with concentrated HCl. The bottles are securely sealed and labeled, and collection data forms are prepared specifying site, date collected, volume, and sample type.

### 2.2.4 DRINKING WATER

Four-week composite samples of municipal drinking water are collected for Rainier (Sample Location 8) and St. Helens (Sample Location 9) at their respective intake structures on the Columbia River. Rainier is downstream of the Trojan Nuclear Plant while St. Helens is upstream. At each location, 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 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.

### 2.2.5 SHORELINE SOIL

Shoreline soil samples of about one quart in volume are taken twice a year.[2] The samples are

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[2] Well water samples to have been collected in June 2001 and September 2001 and a shoreline soil sample to have been collected in September 2001 were not collected as required by the ODCM. ODCM Control 3.3.1 requires well water samples be collected at least once per 92 days and shorline soil samples be collected at least once per 184 days. These sample collection failures were identified in January 2002 and attributed to human error. The responsible technician believed the samples had been collected. Other samples scheduled at the same time were properly collected, but no evidence could be found that the missing samples had been collected. As a compensatory measure, a shoreline soil sample was collected in January 2002 and the sample results are included in this report. A well water sample had already been collected in December 2001 prior to the discovery of the missing samples. Therefore, no additional well water sample was collected since it would provide no useful information regarding the missed collection periods.

Several measures were taken to prevent recurrence. The sample collection failures were reviewed with both the technician and the engineer responsible for the Radiological Environmental Monitoring Program. The 2002 sample calendar was verified to have the correct collection dates for the well water and shoreline soil samples (as well as other samples). A new practice of initialing the calendar once the sample is collected was also initiated. Due to a delay of 6 weeks or more between the time of collection and the reporting of sample results, a back-end verification using sample results was not instituted since it would not provide a timely check.

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 | Air Particulate | Well Water | Surface Water | Shore Soil |
| 1 - Trojan North Building | 300 | WNW | Q | | | | |
| 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 | | | MC | |
| 9 - St. Helens (Municipal Water Supply) | 16,898 | SSE | Q | | | MC | |
| 10- Columbia River | 116,510* | E | | | | | S/A** |
| 11- Prescott Water Supply | 1,287 | NNW | | | Q*** | | |
| 12- Meteorology Tower | 805 | S | | W | | | |
| 13- N Site Boundary at Columbia River | 800 | NNW | Q | W | | | |
| 14- S Site Boundary | 1,332 | S | Q | | | | |

TABLE 2-1

SAMPLING LOCATIONS AND FREQUENCY BY TYPE

| Sample Location | Radial | | Sample | | | | |
|---|---|---|---|---|---|---|---|
| | Distance (meters) | Direction | TLD | Air Particulate | Well Water | Surface Water | Shore Soil |
| 15- E Fenceline | 93 | E | Q | | | | |

LEGEND:
W Weekly.
MC Monthly Composite.
Q Quarterly.
S/A Semi-Annually.
* Columbia River Distance refers to meters measured from mouth.
** For 2001, samples were collected in March 2001 and January 2002. The latter sample was collected as a compensatory measure due to the failure to collect the sample scheduled for September 2001.
*** For 2001, samples were collected in March 2001 and December 2001. The samples scheduled for June 2001 and September 2001 were not collected.

# 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 3.7], or those of the U. S. Environmental Protection Agency (EPA).

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 ($2\sigma$). 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 AIR PARTICULATES

Gross beta concentrations are measured with low background, window-type (0.85 $mg/cm^2$ in thickness), proportional counting systems. The LLD for gross beta measurements is less than or equal to 0.01 $pCi/m^3$ assuming a collected air volume of 285 $m^3$/week.

Gamma isotopic analyses are performed with germanium detectors. The LLD requirements for gamma scans are given in Table 3-1.

## 3.3 DRINKING AND WELL 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 the low-background beta counters mentioned in Section 3.2, 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.

## 3.4 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 scan are given in Table 3-1.

## 3.5 AMBIENT RADIATION MEASUREMENTS

Quarterly ambient gamma radiation measurements are made using TLDs supplied by a vendor. Each environmental dosimeter is composed of a $CaF_2$:Dy (TLD-200) element and a LiF:Mg,Ti (TLD-100) element, both of which are 0.035 inches thick. The $CaF_2$:Dy element is shielded by 80 mg/cm$^2$ ABS plastic, 0.010 inches of tantalum and 0.002 inches of lead. The LiF:Mg,Ti element is shielded by 80 mg/cm$^2$ ABS plastic only.

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

## 3.6 QUALITY CONTROL

A large number of the analyses performed by the analysis laboratory are for quality control purposes. The analysis laboratory participates in Environmental Protection Agency (EPA) and Department of Energy (DOE) interlaboratory comparison programs for environmental measurements. Reports of quality control analyses are presented monthly to PGE.

Results of EPA and DOE interlaboratory comparisons for 2001 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 failed 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 for the year 2001.

## 3.7 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.

5. National Environmental Research Center, Environmental Protection Agency; Handbook of Radiochemical Analytical Methods. Program Element 1HA 325. Office of Research and Development, Las Vegas, NV 89114.

TABLE 3-1

PROGRAM ANALYSES AND LOWER LIMIT OF DETECTION

| Program Analysis | Lower Limit of Detection (LLD)[a] |
|---|---|
| Air Particulate-gross beta | 0.01 $pCi/m^3$ |
| Air Particulate-gamma scan | 0.05 $pCi/m^3$ Cs-134<br>0.06 $pCi/m^3$ Cs-137 |
| Water-gross beta | 1 pCi/liter |
| Water-tritium | 1000 pCi/liter |
| Water-gamma scan | 15 pCi/liter Mn-54<br>15 pCi/liter Co-60<br>15 pCi/liter Cs-134<br>18 pCi/liter Cs-137 |
| Shoreline Soil-gamma scan (dry) | 0.15 pCi/g Cs-134<br>0.18 pCi/g Cs-137 |
| Direct Radiation | 0.04 mR/day or less |

[a] LLD is defined in Section 3.1

TABLE 3-2

EPA AND DOE INTERLABORATORY COMPARISON PROGRAM RESULTS

June 2001 Results

| Analysis | EPA/DOE Value (Bq/Filter) | Eberline Value (Bq/Filter) | Ratio | Evaluation |
|---|---|---|---|---|
| Air Filter | | | | |
| Gross Beta | 2.580 | 2.830 | 1.097 | Pass |
| Mn-54 | 6.520 | 7.715 | 1.183 | Pass |
| Co-60 | 19.440 | 22.240 | 1.144 | Pass |
| Cs-134 | 2.830 | 3.010 | 1.064 | Pass |
| Cs-137 | 8.760 | 10.579 | 1.208 | Pass |

| Analysis | EPA/DOE Value (Bq/Kg) | Eberline Value (Bq/Kg) | Ratio | Evaluation |
|---|---|---|---|---|
| Soil | | | | |
| Cs-137 | 1740.000 | 2003.580 | 1.151 | Pass |

| Analysis | EPA/DOE Value (Bq/L) | Eberline Value (Bq/L) | Ratio | Evaluation |
|---|---|---|---|---|
| Water | | | | |
| Gross Beta | 1297.000 | 1212.000 | 0.934 | Pass |
| H-3 | 79.300 | 60.800 | 0.767 | Pass |
| Co-60 | 98.200 | 103.500 | 1.054 | Pass |
| Cs-137 | 73.000 | 78.130 | 1.070 | Pass |

TABLE 3-2

EPA AND DOE INTERLABORATORY COMPARISON PROGRAM RESULTS

December 2001 Results

| Analysis | EPA/DOE Value (Bq/Filter) | Eberline Value (Bq/Filter) | Ratio | Evaluation |
|---|---|---|---|---|
| Air Filter | | | | |
| Gross Beta | 12.770 | 12.400 | 0.971 | Pass |
| Mn-54 | 81.150 | 99.800 | 1.230 | Pass |
| Co-60 | 17.500 | 20.400 | 1.166 | Pass |
| Cs-134 | 12.950 | 13.700 | 1.058 | Pass |
| Cs-137 | 17.100 | 21.200 | 1.240 | Pass |

| Analysis | EPA/DOE Value (Bq/Kg) | Eberline Value (Bq/Kg) | Ratio | Evaluation |
|---|---|---|---|---|
| Soil | | | | |
| Cs-137 | 612.330 | 694.000 | 1.133 | Pass |

| Analysis | EPA/DOE Value (Bq/L) | Eberline Value (Bq/L) | Ratio | Evaluation |
|---|---|---|---|---|
| Water | | | | |
| Gross Beta | 7970.000 | 7590.000 | 0.952 | Pass |
| H-3 | 207.000 | 228.000 | 1.101 | Pass |
| Co-60 | 209.000 | 216.000 | 1.033 | Pass |
| Cs-137 | 45.133 | 48.400 | 1.072 | Pass |

## 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

| Nuclide Analyzed | Number Processed | Within 2 Sigma | Between 2-3 Sigma | Over 3 Sigma |
|---|---|---|---|---|
| Beta | 44 | 31 | 11 | 2 |
| H-3 | 44 | 34 | 10 | 0 |
| Co-60 | 38 | 26 | 12 | 0 |
| Cs-137 | 63 | 51 | 12 | 0 |

# 4.0 RESULTS AND DISCUSSION

## 4.1 SAMPLES FROM THE TERRESTRIAL ENVIRONMENT

### 4.1.1 AIR PARTICULATES

The gross beta air particulate data obtained during 2001 were comparable to the data obtained during the years 1982 through 2000 (except May 1986) and the preoperational period. Gross beta concentrations for air particulates for sampling periods in 2001 remained generally at low levels.

Average concentrations with their average standard deviations for the years 2001 and before are presented in Table 4-1 for both onsite and offsite locations. Due to revisions of the Radiological Environmental Monitoring Program, air samples were only collected at onsite locations during 2001.

In October 1980, the People's Republic of China tested a nuclear device in the atmosphere. For this reason, the increased average concentrations in 1981 were due to increased fallout levels from the October 1980 Chinese test and not from operation of the Trojan Nuclear Plant. The larger average standard deviation for the 1986 data was due to the increased gross beta activity for May 1986 resulting from the Chernobyl reactor accident near Kiev, Ukraine.

For 2001, the measurement of gamma emitting radionuclides in monthly composites of air particulate filters resulted in no detectable activity.

Data for these air monitoring samples are listed in Chapter 5, Tables 5-1, 5-2, and 5-3.

### 4.1.2 WELL WATER

Tritium levels were below the sensitivity requirements of the program. Gamma emitting radionuclides were not detected in well water samples. The data are presented in Chapter 5, Table 5-4.

### 4.1.3 AMBIENT RADIATION LEVELS

Gamma radiation levels (mR/day) for dosimeter measurements at locations in the environs around the Trojan Nuclear Plant during 2001 are shown in Chapter 5, Table 5-5. All of the dosimeter measurements obtained within the Controlled Area showed no increase in ambient radiation levels.

Trojan onsite measurements during 2001 were less than, but not significantly different from, the control locations. Average gamma radiation levels for the years prior to, and including, 2001 are presented in Table 4-2 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 5-6.

Table 4-3 presents the annual average of the gross beta activity for the two water sample sites from 1980 through 2001. 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 5-7.

## 4.3 SUMMARY OF RESULTS

Table 4-4 presents a summary of the radioactivity analysis results for each medium or pathway sampled during 2001 for the Radiological Environmental Monitoring Program. The format of Table 4-4 is that which is required by ODCM Control 4.1.1.

A review of Table 4-4 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 air particulate sample measurements, the gross beta annual mean concentrations were less than, but not significantly different from, the five year (1990-1994) mean concentration for the control location.

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.

For the radioactivity measurements in drinking water, the annual mean for the gross beta determination was higher (though not significantly) for the upstream or control location (St. Helens) than it was for the downstream location (Rainier).

As is shown by Table 4-4, 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 GROSS BETA CONCENTRATIONS
FOR AIR PARTICULATES
($10^{-2}$ pCi/m$^3$)

| Year | Trojan Site | Oregon | Washington |
|---|---|---|---|
| Preop | 2±2 | 2±2 | 3±2 |
| 1976 | 2±6 | 3±8 | 2±4 |
| 1977 | 3±4 | 4±4 | 5±2 |
| 1978 | 2±2 | 2±1 | 2±1 |
| 1979 | 1±1 | 1±1 | 1±1 |
| 1980 | 3±4 | 3±4 | 2±4 |
| 1981 | 11±2 | 11±4 | 11±1 |
| 1982 | 2±5 | 2±7 | 2±6 |
| 1983 | 2±2 | 2±2 | 2±2 |
| 1984 | 2±2 | 2±2 | 2±2 |
| 1985 | 2±2 | 2±1 | 2±1 |
| 1986 | 3±7 | 3±6 | 3±7 |
| 1987 | 1±1 | 1±1 | 1±1 |
| 1988 | 1±1 | 1±1 | 1±1 |
| 1989 | 2±2 | 2±2 | 2±2 |
| 1990 | 2±1 | 2±1 | 2±1 |
| 1991 | 2±1 | 2±1 | 2±1 |
| 1992 | 2±1 | 2±1 | 2±1 |
| 1993 | 3±2 | 3±2 | 3±2 |
| 1994 | 3±2 | 3±1 | 3±1 |
| 1995 | 2±1 | * | * |
| 1996 | 2±1 | * | * |
| 1997 | 2±1 | * | * |
| 1998 | 2±1 | * | * |
| 1999 | 2±1 | * | * |
| 2000 | 2±2 | * | * |
| 2001 | 2±2 | * | * |

* Due to revisions of the Radiological Environmental Monitoring Program, air samples are no longer collected at offsite locations.

TABLE 4-2

AVERAGE AMBIENT GAMMA RADIATION LEVELS
mR/Day

| Year | Trojan Site | Oregon | Washington |
|---|---|---|---|
| 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 | * |

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

TABLE 4-3

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

| Year | No. 8 -Rainier (Downstream) | No. 9 - St. Helens (Upstream) |
|---|---|---|
| 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 | 1.7±0.5 | 2.7±0.8 |
| 1998 | 2±1 | 3±1 |
| 1999 | 2±1 | 3±1 |
| 2000 | 2±1 | 3±1 |
| 2001 | 2±1 | 3±1 |

TABLE 4-4

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY

Trojan Nuclear Plant, Columbia County, Oregon, Docket 50-344, Reporting Period: January 1-December 31, 2001

| Medium or Pathway Sampled (Unit of Measurement) | Type and Total Number of Analyses Performed | Lower Limit of Detection (LLD) | All Indicator Locations Mean(f)[a] Range | Location with Highest Annual Mean: Name, Distance, and Direction | Location with Highest Annual Mean: Mean(f)[a] Range | Control Locations Mean(f)[a] Range | Number of Reportable Events |
|---|---|---|---|---|---|---|---|
| Air Particulates ($pCi/m^3$) | Gross ß-102 | 0.010 | 0.018(102/102)<br>0.011-0.043 | 13 - Trojan North Site Boundary - 800 Meters NNW | 0.018(51/51)<br>0.011-0.043 | 0.025(256/256)[c]<br>0.007-0.107 | N/A[b] |
| | | | | 12 - Meteorology Tower - 805 Meters S | 0.018 (51/51)<br>0.011-0.042 | | |
| | γ-scan-26 | Table 3-1 | <LLD | - | <LLD | <LLD[c] | 0 |
| Well Water (pCi/liter) | Tritium-2 | 1000 | <LLD | - | <LLD | N/A[b] | 0 |
| | γ-scan-2 | Table 3-1 | <LLD | - | <LLD | N/A[b] | 0 |

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

[b] N/A - Not applicable.

[c] Based on measurements taken at a control location in Portland, Oregon, from 1990 through 1994.

TABLE 4-4

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY

Trojan Nuclear Plant, Columbia County, Oregon, Docket 50-344, Reporting Period: January 1-December 31, 2001

| Medium or Pathway Sampled (Unit of Measurement) | Type and Total Number of Analyses Performed | Lower Limit of Detection (LLD) | All Indicator Locations Mean(f)[a] Range | Location with Highest Annual Mean: Name, Distance, and Direction | Location with Highest Annual Mean: Mean(f)[a] Range | Control Locations Mean(f)[a] Range | Number of Reportable Events |
|---|---|---|---|---|---|---|---|
| Ambient Radiation (mR/day) | γ-exposure-90 | 0.04 | 0.08(72/74)<br>0.04-0.12 | 14 - S Site Boundary 1332 Meters S | 0.10(6/6)<br>0.08-0.12 | 0.10(16/16)<br>0.05-0.13 | N/A[b] |
| Drinking Water (pCi/liter) | Gross ß-26 | 1 | 2.6(25/26)<br>1.2-5.1 | 8 - Rainier<br>3.8 mi - NW | 1.8(12/13)<br>1.2-2.9 | 3.5(13/13)<br>2.8-5.1 | N/A[b] |
| | Tritium-26 | 1000 | <LLD | - | <LLD | <LLD | 0 |
| | γ-scan-26 | Table 3-1 | <LLD | - | <LLD | <LLD | 0 |
| Shoreline Soil (pCi/g - dry) | γ-scan-2 | Table 3-1 | <LLD | - | <LLD | N/A[b] | N/A[b] |

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

[b] N/A - Not applicable.

[c] Based on measurements taken at a control location in Portland, Oregon, from 1990 through 1994.

# 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. |
| pCi/$m^3$ | A reporting unit used with air particulate data which refers to the radioactivity content expressed in picocuries per unit volume of air expressed in cubic meters passed through the glass fiber filter. Note that the volumes are not corrected to standard conditions. |
| 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 2$\sigma$ (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 4.3-1 of Trojan's Offsite Dose Calculation Manual. |

TABLE 5-1

GROSS BETA IN AIR PARTICULATE FILTERS

| | Location 12 | | Location 13 | |
|---|---|---|---|---|
| Collection Date | Volume ($m^3$) | Gross ß ($pCi/m^3$) | Volume ($m^3$) | Gross ß ($pCi/m^3$) |
| 1/9/01 | 285 | 0.042±0.003 | 285 | 0.035±0.003 |
| 1/16/01 | 295 | 0.020±0.002 | 295 | 0.019±0.002 |
| 1/23/01 | 285 | 0.042±0.003 | 285 | 0.041±0.003 |
| 1/30/01 | 280 | 0.028±0.002 | 280 | 0.032±0.003 |
| 2/6/01 | 285 | 0.013±0.002 | 285 | 0.013±0.002 |
| 2/13/01 | 290 | 0.022±0.004 | 290 | 0.018±0.003 |
| 2/20/01 | 280 | 0.028±0.005 | 245 | 0.027±0.005 |
| 2/27/01 | 295 | 0.016±0.003 | 295 | 0.017±0.003 |
| 3/6/01 | 270 | 0.020±0.004 | 275 | 0.018±0.003 |
| 3/13/01 | 290 | 0.021±0.004 | 290 | 0.021±0.004 |
| 3/20/01 | 290 | 0.014±0.003 | 285 | 0.014±0.003 |
| 3/27/01 | 290 | 0.016±0.003 | 290 | 0.017±0.003 |
| 4/3/01 | 280 | 0.013±0.003 | 280 | 0.013±0.003 |
| 4/8/01 | 285 | 0.016±0.003 | 280 | 0.014±0.003 |
| 4/17/01 | 285 | 0.018±0.003 | 290 | 0.015±0.003 |
| 4/24/01 | 280 | 0.016±0.003 | 285 | 0.015±0.003 |
| 5/1/01 | 290 | 0.014±0.003 | 290 | 0.014±0.003 |
| 5/8/01 | 285 | 0.016±0.003 | 285 | 0.017±0.003 |
| 5/15/01 | 290 | 0.018±0.003 | 290 | 0.017±0.003 |
| 5/22/01 | 280 | 0.012±0.003 | 280 | 0.014±0.003 |
| 5/29/01 | 285 | 0.020±0.004 | 285 | 0.018±0.003 |
| 6/5/01 | 285 | 0.012±0.002 | 285 | 0.011±0.002 |
| 6/12/01 | 285 | 0.012±0.003 | 285 | 0.012±0.003 |
| 6/19/01 | 280 | 0.013±0.003 | 280 | 0.011±0.002 |
| 6/26/01 | 285 | 0.014±0.003 | 285 | 0.014±0.003 |
| 7/3/01 | 280 | 0.015±0.003 | 285 | 0.016±0.003 |
| 7/10/01 | 290 | 0.013±0.003 | 285 | 0.017±0.003 |
| 7/17/01 | 280 | 0.011±0.002 | 285 | 0.015±0.002 |
| 7/24/01 | 285 | 0.011±0.002 | 285 | 0.011±0.002 |

TABLE 5-1

GROSS BETA IN AIR PARTICULATE FILTERS

| Collection Date | Location 12 Volume ($m^3$) | Location 12 Gross ß ($pCi/m^3$) | Location 13 Volume ($m^3$) | Location 13 Gross ß ($pCi/m^3$) |
|---|---|---|---|---|
| 7/31/01 | 280 | 0.012±0.003 | 275 | 0.014±0.003 |
| 8/7/01 | 290 | 0.014±0.003 | 290 | 0.013±0.003 |
| 8/14/01 | 175 | 0.019±0.004 | 290 | 0.016±0.003 |
| 8/21/01 | 280 | 0.013±0.002 | 275 | 0.012±0.002 |
| 8/28/01 | 280 | 0.015±0.002 | 285 | 0.017±0.002 |
| 9/4/01 | 290 | 0.013±0.002 | 290 | 0.012±0.002 |
| 9/11/01 | 270 | 0.016±0.002 | 280 | 0.015±0.002 |
| 9/18/01 | 290 | 0.014±0.002 | | * |
| 9/25/01 | 285 | 0.016±0.002 | 245 | 0.018±0.002 |
| 10/2/01 | 280 | 0.018±0.002 | 285 | 0.019±0.002 |
| 10/9/01 | 250 | 0.027±0.002 | 250 | 0.028±0.002 |
| 10/16/01 | 285 | 0.016±0.002 | 285 | 0.014±0.002 |
| 10/23/01 | 290 | 0.013±0.002 | 290 | 0.015±0.002 |
| 10/30/01 | 290 | 0.015±0.002 | 285 | 0.016±0.002 |
| 11/6/01 | 280 | 0.016±0.002 | 285 | 0.014±0.002 |
| 11/13/01 | 290 | 0.042±0.003 | 290 | 0.043±0.003 |
| 11/20/01 | 290 | 0.014±0.002 | 285 | 0.015±0.002 |
| 11/27/01 | 285 | 0.013±0.002 | 285 | 0.013±0.002 |
| 12/4/01 | 285 | 0.012±0.002 | 285 | 0.011±0.002 |
| 12/11/01 | 280 | 0.012±0.002 | 275 | 0.012±0.002 |
| 12/18/01 | 285 | 0.012±0.002 | 290 | 0.011±0.002 |
| 12/26/01 | | * | 290 | 0.029±0.002 |
| 1/2/02 | 370 | 0.040±0.002 | 600 | 0.021±0.001 |

* Air monitor not running. Power cut off.

TABLE 5-2

SUMMARY - GROSS BETA IN AIR SAMPLES

| | pCi/m$^3$ | | |
|---|---|---|---|
| | Mean | Maximum | Minimum |
| Trojan Onsite Stations | | | |
| Location 12 | 0.018±0.020 | 0.042 | 0.011 |
| Location 13 | 0.018±0.019 | 0.043 | 0.011 |

TABLE 5-3

GAMMA EMITTERS: CONCENTRATIONS IN AIR PARTICULATE FILTERS
(Monthly Composites)

| Collection Dates | ($pCi/m^3$) Location 12 | Location 13 |
|---|---|---|
| 1/16/01-2/06/01 | <LLD | <LLD |
| 2/13/01-3/06/01 | <LLD | <LLD |
| 3/13/01-4/03/01 | <LLD | <LLD |
| 4/08/01-5/01/01 | <LLD | <LLD |
| 5/08/01-5/29/01 | <LLD | <LLD |
| 6/05/01-6/26/01 | <LLD | <LLD |
| 7/03/01-7/24/01 | <LLD | <LLD |
| 7/31/01-8/21/01 | <LLD | <LLD |
| 8/28/01-9/18/01 | <LLD | <LLD |
| 9/25/01-10/16/01 | <LLD | <LLD |
| 10/23/01-11/13/01 | <LLD | <LLD |
| 11/20/01-12/11/01 | <LLD | <LLD |
| 12/18/01-1/05/02 | <LLD | <LLD |

LLD: 0.05 $pCi/m^3$ Cs-134
0.06 $pCi/m^3$ Cs-137

## TABLE 5-4

## RADIOACTIVITY IN WELL WATER

| Collection Date[a] | pCi/l Location 11 Tritium | Gamma Emitters |
|---|---|---|
| 3/06/01 | < 1000 | < LLD |
| 12/04/01 | < 1000 | < LLD |

LLD: 15 pCi/l Mn-54, Co-60, Cs-134
18 pCi/l Cs-137
1000 pCi/l H-3

[a] No samples collected in June 2001 or September 2001 - see Section 2.2.3.

TABLE 5-5

AMBIENT GAMMA RADIATION LEVELS

mR/Day[a]

| Location | First Quarter 12/28/00-3/29/01 | Second Quarter 3/29/01-6/28/01 | Third Quarter 6/28/01-10/01/01 | Fourth Quarter 10/01/01-1/02/02 |
|---|---|---|---|---|
| 1 | 0.09 | 0.11 | 0.08 | 0.11 |
| 2 | 0.05 | 0.05 | <LLD | 0.05 |
| 3 | 0.07 | 0.08 | 0.04 | 0.06 |
| 4 | 0.08 | 0.07 | 0.05 | 0.10 |
| 5 | 0.10 | 0.10 | 0.07 | 0.10 |
| 6 | [b] | 0.11 | 0.08 | 0.10 |
| 7 | 0.07 | 0.08 | 0.05 | 0.09 |
| 8 | 0.11 | 0.10 | 0.08 | 0.12 |
| 9 | 0.11 | 0.11 | 0.07 | 0.12 |
| 13 | 0.08 | [b] | 0.04 | 0.10 |
| 14 | 0.09 | [b] | 0.10 | 0.12 |
| 15 | 0.08 | 0.08 | 0.05 | 0.06 |

LLD: 0.04 mR/day

[a] Two TLDs were placed at each location. Reported value is the average of the two TLDs. Beginning in 2002, only a single TLD is place at each location.

[b] TLD lost.

TABLE 5-6

RADIOACTIVITY IN DRINKING WATER

| Location 8 - Rainier Municipal Water Supply | | | | Location 9 - St. Helens Municipal Water Supply | | | |
|---|---|---|---|---|---|---|---|
| | pCi/l | | | | pCi/l | | |
| Collection Dates | Gross Beta | Tritium | Gamma Emitters | Collection Dates | Gross Beta | Tritium | Gamma Emitters |
| 1/09/01-2/06/01 | 1.2±0.8 | <1000 | <LLD | 1/09/01-2/06/01 | 3.3±1.0 | <1000 | <LLD |
| 2/06/01-3/06/01 | 2.9±1.0 | <1000 | <LLD | 2/06/01-3/06/01 | 5.1±1.2 | <1000 | <LLD |
| 3/06/01-4/03/01 | <1.3 | <1000 | <LLD | 3/06/01-4/03/01 | 2.9±1.0 | <1000 | <LLD |
| 4/03/01-5/01/01 | 2.7±1.0 | <1000 | <LLD | 4/03/01-5/01/01 | 3.6±1.0 | <1000 | <LLD |
| 5/01/01-5/29/01 | 1.2±0.8 | <1000 | <LLD | 5/01/01-5/29/01 | 3.0±0.9 | <1000 | <LLD |
| 5/29/01-6/26/01 | 1.2±0.8 | <1000 | <LLD | 5/29/01-6/26/01 | 2.8±0.9 | <1000 | <LLD |
| 6/26/01-7/24/01 | 1.9±0.9 | <1000 | <LLD | 6/26/01-7/24/01 | 3.0±0.9 | <1000 | <LLD |
| 7/24/01-8/21/01 | 1.6±0.9 | <1000 | <LLD | 7/24/01-8/21/01 | 2.9±1.0 | <1000 | <LLD |
| 8/21/01-9/18/01 | 1.4±0.8 | <1000 | <LLD | 8/21/01-9/18/01 | 3.1±1.0 | <1000 | <LLD |
| 9/18/01-10/16/01 | 1.9±0.8 | <1000 | <LLD | 9/18/01-10/16/01 | 4.9±1.3 | <1000 | <LLD |
| 10/16/01-11/13/01 | 1.4±0.8 | <1000 | <LLD | 10/16/01-11/13/01 | 3.5±1.1 | <1000 | <LLD |
| 11/13/01-12/11/01 | 2.1±0.8 | <1000 | <LLD | 11/13/01-12/11/01 | 3.0±1.0 | <1000 | <LLD |
| 12/11/01-1/08/02 | 1.6±0.8 | <1000 | <LLD | 12/11/01-1/08/02 | 3.8±1.2 | <1000 | <LLD |

LLD: 15 pCi/l Mn-54, Co-60, Cs-134
18 pCi/l Cs-137
1000 pCi/l H-3

## TABLE 5-7

## RADIOACTIVITY IN SHORELINE SOIL
(Semiannual Collections)

pCi/g (dry)

Location 10

| Collection Date | Gamma Emitters |
|---|---|
| 03/13/01 | < LLD |
| 01/10/02[a] | <LLD |

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

[a] Compensatory sample - see Section 2.2.5.

TROJAN NUCLEAR PLANT
RADIOLOGICAL ENVIRONMENTAL
MONITORING REPORT

# APPENDIX A

## PGE-1021, OFFSITE DOSE CALCULATION MANUAL
## AMENDMENT 21

PGE-1021

# OFFSITE DOSE CALCULATION MANUAL

Amendment 21

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 Trojan Technical Specifications. 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 and liquid effluents, and for the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints. 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 Trojan Technical Specifications. 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 Trojan Technical Specifications. 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 Trojan Technical Specifications 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.

## 2.5 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 gaseous and liquid effluent monitoring Alarm/Trip Setpoints, 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 Trojan Technical Specifications and (2) descriptions of the information that should be included in the Annual Radiological Environmental Monitoring and annual Radioactive Effluent Release Reports required by Trojan Technical Specifications.

## 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.

## 2.13 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 |

# 3.0 CONTROLS AND SURVEILLANCE REQUIREMENTS

## CONTROLS

Controls and ACTION requirements shall be applicable during the conditions specified for each Control.

Adherence to the requirements of the Control and/or associated ACTION within the specified time interval shall constitute compliance with the Control. In the event the Control is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

## SURVEILLANCE REQUIREMENTS

Surveillance Requirements shall be applicable during the conditions specified for individual Controls unless otherwise stated in an individual Surveillance Requirement.

Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined above, shall constitute noncompliance with the OPERABILITY requirements for a Control. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

Entry into a specified applicability condition shall not be made unless the Surveillance Requirement(s) associated with the Control have been performed within the stated surveillance interval or as otherwise specified.

## CONTROL 3.1.1 - LIQUID

The radioactive liquid effluent monitoring instrumentation channels shown in Table 3.1.1-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Control 3.2.1.1 are not exceeded.

### APPLICABILITY

As shown in Table 3.1.1-1.

### ACTION

a. With a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint less conservative than a value which will ensure the limits of Control 3.2.1.1 are met, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels operable, take the ACTION shown in Table 3.1.1-1. 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

a. The setpoints shall be determined in accordance with procedures as described in Section 7 of the ODCM and shall be recorded (formerly Surveillance Requirement 4.1.1.1).

b. Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL

SURVEILLANCE REQUIREMENTS (Continued)

CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 3.1.1-2 (formerly Surveillance Requirement 4.1.1.2).

BASIS

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases. The alarm/trip setpoints for these instruments shall be calculated in accordance Section 7 of the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR 50.

## TABLE 3.1.1-1

| Radioactive Liquid Effluent Monitoring Instrumentation | | | |
|---|---|---|---|
| **Instrument** | **Minimum Channels Operable** | **Applicability** | **Action** |
| 1. Gross Radioactivity Monitors Providing Automatic Termination of Release | | | |
| a. Liquid Radwaste Effluent Line Process Radiation Monitor | 1 | * | 1 |
| 2. Flow Rate Measurement Devices | | | |
| a. Liquid Radwaste Effluent Line Flow | 1 | * | 2 |
| b. Discharge Structure Flow Recorder | 1 | * | 2 |

* 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 may continue provided that prior to initiating a release:

1. At least two independent samples are analyzed in accordance with Table 3.2.1.1-1, and

2. A technically qualified member of the Facility Staff independently verifies the release rate calculations and discharge valving, and

3. Corrective action is initiated to return the channels to operable status; or

Otherwise suspend release of radioactive effluents via this pathway.

ACTION 2 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.

TABLE 3.1.1-2

| Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements | | | | |
|---|---|---|---|---|
| **Instrument** | **Channel Check** | **Source Check** | **Channel Calibration** | **Channel Functional Test** |
| 1. Gross Beta or Gamma Radioactivity Monitors Providing Alarm and Automatic Isolation | | | | |
| a. Liquid Radwaste Effluent Line Process Radiation Monitor | D* | P | R(1) | Q(1) |
| 2. Flow Rate Monitors | | | | |
| a. Liquid Radwaste Effluent Line Flow | D* | N/A | R** | N/A |
| b. Discharge Structure Flow Recorder | D* | N/A | R** | N/A |

* During releases via this pathway.

** Applies only to flow indication loop.

(1) The Channel Functional Test shall, where applicable, include verification that automatic isolation of the affected pathway and/or control room annunciator occurs if:

a) Instrument indicates above alarm trip setpoint.

b) Instrument indicates a downscale failure.

c) Controls not in OPERATE mode, or instrument de-energized.

## CONTROL 3.1.2 - GASEOUS

The radioactive gaseous effluent monitoring instrumentation channels shown in Table 3.1.2-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Control 3.2.2.1 are not exceeded.

### APPLICABILITY

As shown in Table 3.1.2-1.

### ACTION

a. With a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint less conservative than the value shown in Table 3.1.2-1, adjust the setpoint to within the limit without delay or declare the channel inoperable.

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation 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

a. The setpoints shall be determined in accordance with procedures as described in Section 7 of the ODCM and shall be recorded (formerly Surveillance Requirement 4.1.2.1).

b. Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL

SURVEILLANCE REQUIREMENTS (Continued)

CALIBRATION, and CHANNEL FUNCTIONAL TEST operations at the frequencies shown in Table 3.1.2-2 (formerly Surveillance Requirement 4.1.2.2).

BASIS

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases, however, in no case will the limits of 10 CFR 20 be exceeded. The alarm/trip setpoints for these instruments shall be calculated in accordance with Section 7 of the ODCM. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR 50.

| Radioactive Gaseous Effluent Monitoring Instrumentation | | | | |
|---|---|---|---|---|
| **Instrument** | **Minimum Channels Operable** | **Applicable Modes** | **Alarm Trip Setpoint** | **Action** |
| 1. Auxiliary Building Ventilation Monitoring System | | | | |
| a. Noble Gas Activity Monitor (PRM-2C) | 1 | * | ** | 1 |
| b. Particulate Composite Sampler | 1 | * | N/A | 2 |
| c. Sampler Flow Rate Measuring Device for Composite Sampler | 1 | * | N/A | 3 |
| 2. Condensate Demineralizer Building Effluent Monitoring | | | | |
| a. Particulate Composite Sampler | 1 | * | N/A | 4 |
| b. Sampler Flow Rate Measuring Device for Composite Sampler | 1 | * | N/A | 4 |

**Table Notation**

* During releases via this pathway.

** Set to assure the limits of 3.2.2.1 are met.

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 grab samples are taken at least once per 7 days and these samples are analyzed for gross activity within 24 hours.

ACTION 2 With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided grab samples are taken and analyzed once per 7 days when plant release rates are anticipated to be stable. If transient release rates are anticipated, a sufficient number of additional samples will be taken to document release rates.

ACTION 3 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 24 hours.

ACTION 4 With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, effluent releases via this pathway will be stopped.

| Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements | | | | |
|---|---|---|---|---|
| **Instrument** | **Channel Check** | **Source Check** | **Channel Calibration** | **Channel Functional Test** |
| 1. Auxiliary Building Ventilation Monitoring System | | | | |
| a. Noble Gas Activity Monitor (PRM-2C) | D* | M | R | Q(1) |
| b. Particulate Composite Sampler | W(2) | N/A | N/A | N/A |
| c. Sampler Flow Rate Measuring Device for Composite Sampler | W* | N/A | N/A | N/A |

| Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements | | | | |
|---|---|---|---|---|
| Instrument | Channel Check | Source Check | Channel Calibration | Channel Functional Test |
| 2. Condensate Demineralizer Building | | | | |
| a. Particulate Composite Sampler | W(2) | N/A | N/A | N/A |
| b. Sampler Flow Rate Measuring Device for Composite Sampler | W* | N/A | N/A | N/A |

* During or prior to releases via this pathway.

(1) The CHANNEL FUNCTIONAL TEST shall, where applicable, include verification that automatic isolation of the affected pathway and/or control room annunciator occurs if:

a. Instrument indicates above alarm trip setpoint

b. Instrument indicates a downscale failure

c. Controls not in OPERATE mode

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

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).

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).

| 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) ($\mu$Ci/ml)[a]** |
| Batch[d] Waste Release Tanks | P | P | Grab Sample/ Principal Gamma Emitters[e] | $5x10^{-7}$[b] |
| | P | M | Composite[c]/ Tritium | $1x10^{-5}$ |
| | P | M | Composite[c]/ Gross Alpha | $1x10^{-7}$ |
| | P | Q | Composite[c]/ Sr-90 | $5x10^{-8}$ |
| | P | Q | Composite[c]/ Fe-55 | $1x10^{-6}$ |

## 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 $\Delta t$. $\Delta t$ 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.

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 and dissolved gases, 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).

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 $\leq$ 3 mrem to the total body. The design objective for any organ will be limited to $\leq$ 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 and noble gases 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.

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 and dissolved gases 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).

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 Intervenors 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 and noble gases 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.

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 values:

a. The dose rate limit for noble gases shall be ≤ 500 mrem/yr to the total body and ≤ 3000 mrem/yr to the skin, and

b. The dose rate limit for tritium and 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 limits, immediately decrease the release rate to comply with the limit(s) given in Control 3.2.2.1.

SURVEILLANCE REQUIREMENTS

a. The release rate of noble gases in gaseous effluents shall be such that

$$2.0\ Q_{Tv}\ \overline{K_v} \le 1 \ and\ 0.33\ Q_{Tv}\ (\overline{L_v} + 1.1\ \overline{N_v}) \le 1.$$

For Kr-85, the limiting release rate is ≤ 0.176 Ci/sec (formerly Surveillance Requirement 4.2.2.1.1).

b. The release rate of tritium and radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be such that $0.67\ Q_v\ \overline{R_i} \le 1$ 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).

SURVEILLANCE REQUIREMENTS (Continued)

c. The above release rates are determined in accordance with the methodology and parameters in Section 6 of the ODCM (formerly Surveillance Requirement 4.2.2.1.3).

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 specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to ≤ 500 mrem/year to the total body or to ≤ 3000 mrem/year to the skin. These release rate limits also restrict 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

| 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) (μCi/ml)[a]** |
| A. Fuel and Auxiliary Building Ventilation Exhaust | M[cd] | M[cd] | Grab Sample/Noble Gas (Kr-85)[e] | $1x10^{-4}$[b] |
| | | M[cd] | Composite/Principal Gamma Emitters[e] | $1x10^{-11}$[b] |
| | | M[cd] | Composite/Gross Alpha | $1x10^{-11}$[b] |
| | | M[cd] | Composite/Tritium | $1x10^{-6}$[b] |
| | | Q[cd] | Composite/Sr-90 | $1x10^{-11}$[b] |
| B. Condensate Demineralizer Building Exhaust | M[cd] | M[cd] | Composite/Principle Gamma Emitters[e] | $1x10^{-11}$[b] |
| | | M[cd] | Composite/Gross Alpha | $1x10^{-11}$[b] |
| | | Q[cd] | Composite/Sr-90 | $1x10^{-11}$[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: Kr-85 for gaseous emissions and 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.

CONTROL 3.2.2.2 - DOSE, NOBLE GASES

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

During any calendar quarter to ≤ 5 mrad for gamma radiation and ≤ 10 mrad for beta radiation.

APPLICABILITY

At all times.

ACTION

With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days upon determination, pursuant to Control 4.1.3, a Special 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 releases to below the design objectives.

SURVEILLANCE REQUIREMENTS

a. Release Rate Calculations: The average release rate of noble gases from the site during any calendar quarter shall be such that:

$$50\ Q_{Tv}\ \overline{N_v} < 1 \qquad \text{or} \qquad 25\ Q_{Tv}\ \overline{M_v} < 1$$

For Kr-85, the limiting release rate is < 1.54E-3 Ci/sec (formerly Surveillance Requirement 4.2.2.2.1).

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 4.2.2.2.2).

BASIS

This Control is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR 50. As outlined in Section II.B of Appendix I, 10 CFR 50, the design objective air dose in areas at or beyond the SITE BOUNDARY due to noble gases released in gaseous effluents will be limited during any calendar year to ≤ 10 mrad for gamma radiation and ≤ 20 mrad for beta radiation. As outlined in Section IV.A to Appendix I to 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 material in gaseous effluents will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I, which requires that calculational procedures be 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 radioactive noble gases 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 the air doses at the SITE BOUNDARY will be based upon the historical average atmospheric conditions.

CONTROL 3.2.2.3 - DOSE, TRITIUM AND RADIONUCLIDES IN PARTICULATE FORM

The dose to a MEMBER OF THE PUBLIC from tritium and 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 tritium and 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).

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 tritium and 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 tritium and 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.

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.

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 tritium and 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. The gaseous effluent air doses due to gaseous effluent releases to areas at or beyond the SITE BOUNDARY when averaged over a calendar quarter shall not exceed 2.5 mrad for gamma radiation and 5.0 mrad for beta radiation.

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.

SURVEILLANCE REQUIREMENTS

a. The average release rate of noble gases from the site during any calendar quarter shall be such that:

$$100\ Q_{Tv}\ \overline{N_v} < 1 \qquad \text{or} \qquad 50\ Q_{Tv}\ \overline{M_v} < 1$$

For Kr-85, the limiting release rate is < 7.69E-4 Ci/sec (formerly Surveillance Requirement 4.2.2.4.1).

b. The average release rate of tritium and 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).

c. 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 4.2.2.4.3).

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.

BASIS (Continued)

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 (20 mrad/yr beta air dose; 10 mrad/yr gamma air dose; 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.

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 mrems to the total body or any organ, except thyroid, which shall be limited to less than or equal to 75 mrems.

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, 3.2.2.2, 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

a. Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Control 3.2.1.2, 3.2.2.2, 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

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.

## 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, spent resins, and boric acid solutions) 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.

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

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:

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

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, 3.2.2.2 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 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. Ground Water | 1 | At least once per 92 days. | Gamma isotopic and tritium analysis of each sample. |
| b. Surface Water including Drinking Water | 2 | Composite samples over 31-day period from Columbia River (1 upstream and 1 downstream) | Gross beta and gamma isotopic analysis of each sample. Tritium analysis of composite sample at least once per 92 days. |
| c. Sediment from Shoreline | 1 | At least once per 184 days. | Gamma isotopic of each sample. |
| 3. AIRBORNE<br>Particulates | 2 | Continuous operation of sampler with sample collection as required by dust loading but at least once per 7 days. | Particulate sampler. Analyze for gross beta radioactivity ≥ 24 hours following filter change. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days. |

* Sample locations are identified in Table 3.3.1-4

**TABLE 3.3.1-2**

| Reporting Levels for Radioactivity Concentrations in Environmental Samples | | |
|---|---|---|
| **Analysis** | **Airborne Particulates ($pCi/m^3$)** | **Water (pCi/l)** |
| H-3 | | $5x10^4$<br>$2x10^4$ (a) |
| Mn-54 | | $5x10^2$ |
| Co-60 | | $2x10^2$ |
| Cs-134 | 4 | 20 |
| Cs-137 | 6 | 20 |

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

TABLE 3.3.1-3

| Maximum Values for the Lower Limits of Detection (LLD)[a] | | | |
|---|---|---|---|
| **Analysis** | **Airborne Particulate ($pCi/m^3$)** | **Sediment (pCi/kg, dry)** | **Water (pCi/l)** |
| gross beta | $1x10^{-2}$ | | 4 (1[b]) |
| H-3 | | | 2000 (1000[b]) |
| Mn-54 | | | 15 |
| Co-60 | | | 15 |
| Cs-134 | $5x10^{-2}$ | 150 | 15 |
| Cs-137 | $6x10^{-2}$ | 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) |
| $\lambda$ | = | the radioactive decay constant for the particular radionuclide |
| $\Delta t$ | = | 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 <u>a priori</u> (before the fact) limit representing the capability of a measurement system and not as an <u>a posteriori</u> (after the fact) limit for a particular measurement.

b - LLD for drinking water.

TABLE 3.3.1-4

| Sampling Locations and Frequency by Type | | | | | | | |
|---|---|---|---|---|---|---|---|
| | Radial | | Sample | | | | |
| Sample Location | Distance (meters) | Direction | TLD | Air Particulate | Well Water | Surface Water | Shore Soil |
| 1 - Trojan North Building | 300 | WNW | Q | | | | |
| 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 | | | MC | |
| 9 - St. Helens (Municipal Water Supply) | 16,898 | SSE | Q | | | MC | |
| 10 - Columbia River | 116,510* | E | | | | | S/A |
| 11 - Prescott Water Supply | 1,287 | NNW | | | Q | | |
| 12 - Meteorology Tower | 805 | S | | W | | | |
| 13 - N Site Boundary at Columbia River | 800 | NNW | Q | W | | | |
| 14 - S Site Boundary | 1,332 | S | Q | | | | |
| 15 - E Fenceline | 93 | E | Q | | | | |

LEGEND:
W - Weekly
MC - Monthly Composite
* - Columbia River mileage refers to meters measured from mouth
Q - Quarterly
S/A - Semi-annually

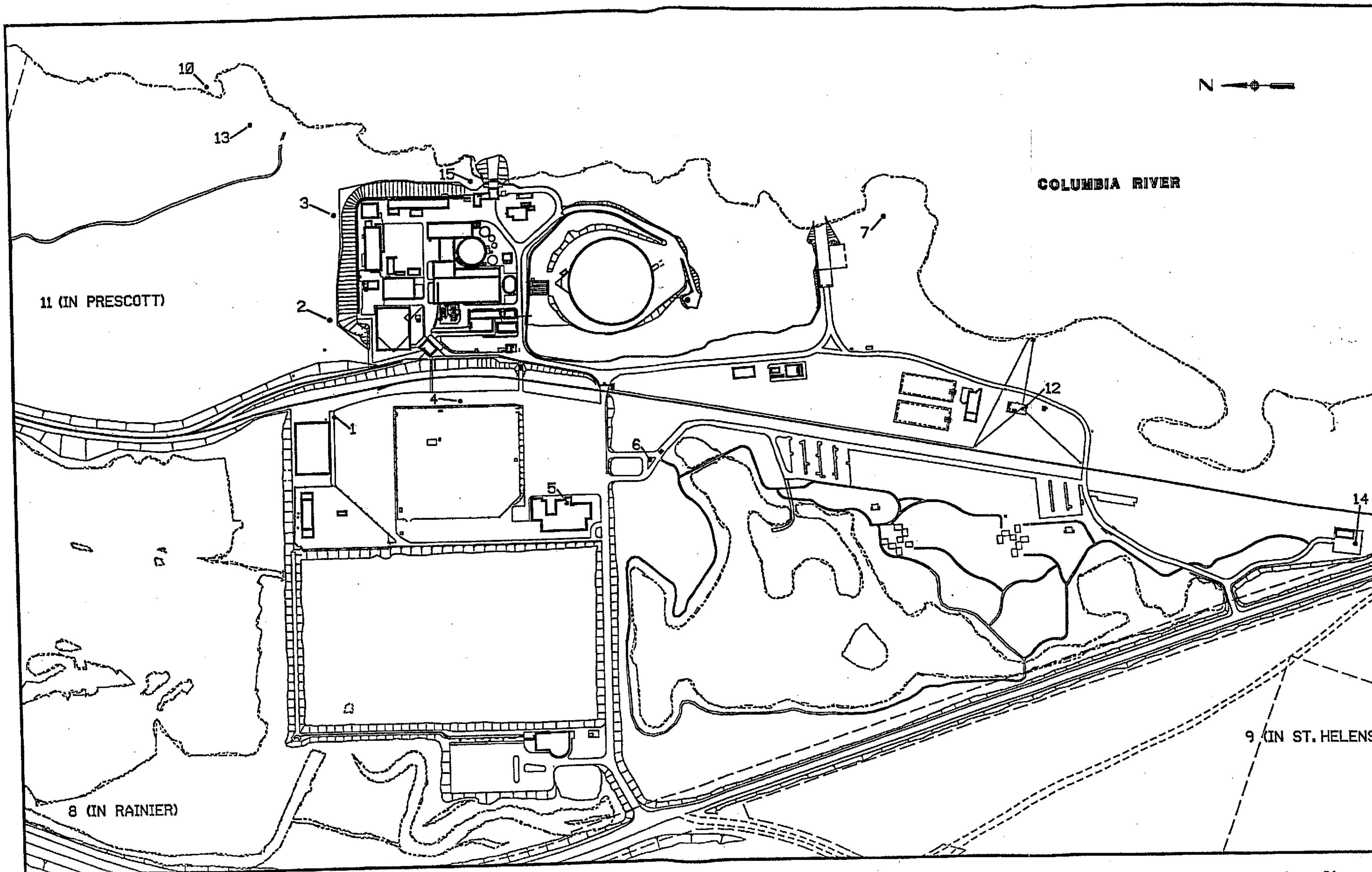


**Figure 3.3.1-1**
**Sampling Locations**

## 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 operation 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 Trojan Technical Specifications.

## 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 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 DSAR, Table 3.5-1. In addition, the maximum noble gas gamma air and beta air doses shall be evaluated in areas at the SITE BOUNDARY. 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.

4.1.3 SPECIAL REPORTS (Continued)

a. Radioactive Effluents, Controls 3.2.1.2, 3.2.2.2, 3.2.2.3, and 3.2.2.5.

b. Radiological Environmental Monitoring, Control 3.3.1.

## 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

4.2 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.

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* 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.

## 4.3 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 indicated 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] | Location With Highest Annual Mean Name Distance and Direction | Mean (f)[b] Range[b] | 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 Table3.3.1-3 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):

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

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 $\mu$Ci/ml

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

$C_{i\ell}$ = the average concentration of radionuclide i, during time period $\Delta T_\ell$, in $\mu$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

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

5.1 INTRODUCTION (Continued)

The term $F_\ell$, the near field average dilution factor, is determined as follows for time period $\Delta T_\ell$ :

$$F_\ell = \frac{\text{liquid radioactive waste discharge volume}}{\text{total plant discharge volume x plant dilution factor}}$$

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:

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

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):

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

where

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

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

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 cfs average river flow ÷ 1000 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)

## 5.2 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:

$$f = \sum_i \frac{A_i}{ECV_i} \tag{5-3}$$

where

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

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

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

$$C = \frac{f \times F_e}{F_p} \tag{5-4}$$

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 $\leq 1.0$.

## 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.

## 5.3 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 and dissolved gases, exceeds 2.5 Ci/quarter.

The intermediate surveillance value of 2.5 Ci/quarter, excluding tritium and dissolved gases, 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 and dissolved gases):

Total Body

$$D_{TB} = \sum_{\ell} \frac{\Delta T_{\ell}}{F_{pd} V_{\ell}} \sum_{i} A_{TB_i} \times Q_{i_\ell} \tag{5-5}$$

Maximum Organ

$$D_{MO} = \sum_{\ell} \frac{\Delta T_{\ell}}{F_{pd} V_{\ell}} \sum_{i} A_{MO_i} \times Q_{i_\ell} \tag{5-6}$$

where

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

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

5.3 CONTROL 3.2.1.2 (Continued)

$\Delta T_\ell$ = the $\ell^{th}$ 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)

$V_\ell$ = volume of total plant discharge flow for time $\Delta T_\ell$, 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 $\mu$Ci/ml, see Table 5-1 for values

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

$Q_{i\ell}$ = activity released of nuclide i, over time period $\Delta T_\ell$, $\mu$Ci

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.

## 5.4 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 and dissolved gases, 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.

A flow diagram of the liquid radwaste treatment system, as applicable to Control 3.2.1.3, is shown in DSAR Figure 5.3-1.

## 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<br>Adult Ingestion Dose Factors<br>(mrem/hr per $\mu$Ci/ml) | | |
|---|---|---|
| Nuclide | Total Body $A_{TB_i}$ | Maximum Organ $A_{MO_i}$ |
| 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 noble gas dose rate contributions may be determined using the following general equations (adapted from NUREG-0133, Sections 5.2.1 and 5.3.1):

Gamma air dose rate, $D_a^{\gamma}$, mrad/yr

$$D_a^{\gamma} = 1000 \sum_i N_i \times Q_{TV_i} \tag{6-1}$$

Beta air dose rate, $D_a^{\beta}$, mrad/yr

$$D_a^{\beta} = 1000 \sum_i M_i \times Q_{TV_i} \tag{6-2}$$

Skin dose rate, $D_s$, mrem/yr

$$D_s = 1000 \sum_i (L_i + 1.1 \times N_i) \times Q_{TV_i} \tag{6-3}$$

Total body dose rate, $D_{TB}$, mrem/yr

$$D_{TB} = 1000 \sum_i K_i \times Q_{TV_i} \tag{6-4}$$

where

$K_i$ = total body dose factor due to gamma emissions for nuclide i, rem/yr per Ci/sec

$L_i$ = skin dose factor due to beta emissions for nuclide i, rem/yr per Ci/sec

## 6.1 INTRODUCTION (Continued)

$M_i$ = air dose factor due to beta emissions for nuclide i, rad/yr per Ci/sec

$N_i$ = air dose factor due to gamma emissions for nuclide i, rad/yr per Ci/sec (note that these are "air" rads not "tissue" rads)

$Q_{TV_i}$ = noble gas activity release rate of nuclide i, Ci/sec

1000 = constant, mrad/rad or mrem/rem

1.1 = constant, the average ratio of tissue to air energy absorption coefficients with the units of rem/"air" rad

Derivation of $K_i$, $L_i$, $M_i$, and $N_i$ are presented in Appendices A and C. These values are listed in Table A-1.

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

$$D_{IPC} = 1000 \sum_i R_i \times Q_i \tag{6-5}$$

where

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

$R_i$ = dose factor for nuclides other than noble gases 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.

## 6.2 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 the lowest of the three values computed as follows (based on Equations 5-3, 5-4, and 5-5):

Noble Gases

$$Q_{TV} \le \frac{1}{2.0\overline{K}_v} \qquad (6\text{-}6)$$

$$Q_{TV} \le \frac{1}{0.33\ (\overline{L}_v + 1.1\ \overline{N}_v)} \qquad (6\text{-}7)$$

Tritium, and particulates ($T_{1/2}$ > 8 days)

$$Q_v \le \frac{1}{0.67\overline{R}_i} \qquad (6\text{-}8)$$

where

$Q_{TV} = \sum_i Q_{TV_i}$ = total noble gas release rate, Ci/sec

$Q_V = \sum_i Q_{V_i}$ = total tritium, and particulate Ci/sec ($T_{1/2} > 8$ days)

$Q_{TV_i}$ = noble gas release rate for nuclide i, Ci/sec

$Q_{V_i}$ = particulate release rate for nuclide i, Ci/sec

6.2 CONTROL 3.2.2.1 (Continued)

$\overline{K}_v = (1/Q_{TV}) \sum_i Q_{TV_i} K_i$

$\overline{L}_v = (1/Q_{TV}) \sum_i Q_{TV_i} L_i$

$\overline{N}_V = (1/Q_{TV}) \sum_i Q_{TV_i} N_i$

1.1 = constant, the average ratio of tissue to air energy absorption coefficients with the units of rem/"air" rad

$\overline{R}_i = (1/Q_V) \sum_i Q_{V_i} R_i$

$K_i$ = gamma total body dose factor for nuclide i, rem/yr per Ci/sec

$L_i$ = beta skin dose factor for nuclide i, rem/yr per Ci/sec

$N_i$ = gamma air dose factor for nuclide i, rad/yr per Ci/sec
(NOTE: these are "air" rads not "tissue" rads.)

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

Since Kr-85 is the only remaining noble gas, Equation 6-6 and Equation 6-7 can be solved. The noble gas release rate to determine compliance with Control 3.2.2.1 is:

$$Q_{TV} \leq 0.176 \text{ Ci/sec}$$

Nuclides which require analysis of monthly or quarterly composite samples (e.g., H-3, Sr-90) are not considered in the calculation required by Surveillance Requirement b. 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 b. of Control 3.2.2.1.

## 6.3 CONTROL 3.2.2.2

This section will be used to demonstrate compliance with Control 3.2.2.2 at least once per 31 days.

Noble gas release rates shall be evaluated at least once per 31 days using the more limiting of Equations 6-9 and 6-10, together with the dose factors in Table A-1:

$$50\ Q_{TV}\ \overline{N}_V < 1 \qquad (6\text{-}9)$$

or

$$25\ Q_{TV}\ \overline{M}_V < 1 \qquad (6\text{-}10)$$

where

$Q_{TV} = \Sigma\ Q_{TV_i} =$ total noble gas release rate, Ci/sec

$Q_{TV_i} =$ noble gas release rate for nuclide i, Ci/sec

$\overline{N}_V = \left(\frac{1}{Q_{TV}}\right) \Sigma\ Q_{TV_i} N_i$

$\overline{M}_V = \left(\frac{1}{Q_{TV}}\right) \Sigma\ Q_{TV_i} M_i$

$N_i =$ gamma air dose factor for nuclide i, rad/yr per Ci/sec

$M_i =$ beta air dose factor for nuclide i, rad/yr per Ci/sec

Since Kr-85 is the only remaining noble gas, Equations 6-9 and 6-10 can be solved. The noble gas release rate to demonstrate compliance with Control 3.2.2.2 is:

$$Q_{TV} < 1.54\text{E-}3 \text{ Ci/sec}$$

## 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), and tritium releases to determine compliance with Control 3.2.2.3 as follows:

$$100\ Q_V\ \overline{R}_i < 1 \qquad (6\text{-}11)$$

where

$\overline{R_i}$ = $1/Q_v\ \Sigma\ Q_{v_i}\ R_i$

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

$Q_v$ = $\Sigma_i\ Q_v$ = total tritium and particulate release rate, Ci/sec

$Q_{v_i}$ = cumulative quarterly release rate of each, particulate, and tritium nuclide i, Ci/sec

Nuclides which require analysis of monthly or quarterly composite samples (e.g., H-3, 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.

### 6.4.2 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.

## 6.5 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.

Flow diagrams of the ventilation exhaust treatment system, as applicable to Control 3.2.2.4, are shown in Figure 6-1.

### 6.5.1 NOBLE GASES

The noble gas release rate limits for Control 3.2.2.4 will be determined using the equations listed below. The allowable release rate is the lower of the two values calculated by Equations 6-12 and 6-13:

$$Q_{TV} < \frac{1}{100\ \overline{N}_V} \tag{6-12}$$

$$Q_{TV} < \frac{1}{50\ \overline{M}_V} \tag{6-13}$$

where all parameters have been previously defined.

Since Kr-85 is the only remaining noble gas, Equations 6-12 and 6-13 can be solved. The release rate to demonstrate compliance with Control 3.2.2.4 is: $Q_{TV} < 7.69E\text{-}4$ Ci/sec.

### 6.5.2 PARTICULATES AND TRITIUM

The particulate and tritium 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 6-14:

$$Q_V < \frac{1}{200\ \overline{R}_i} \tag{6-14}$$

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

Surveillance Requirement a. 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, 3.2.2.2, 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, 3.2.2.2, or 3.2.2.3.

### 6.6.2 METHODOLOGY

Dose calculations for the three effluent categories of Controls 3.2.1.2, 3.2.2.2, and 3.2.2.3 are to be performed in accordance with the methodology of Sections 5.3, 6.3, 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.

## 6.7 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.

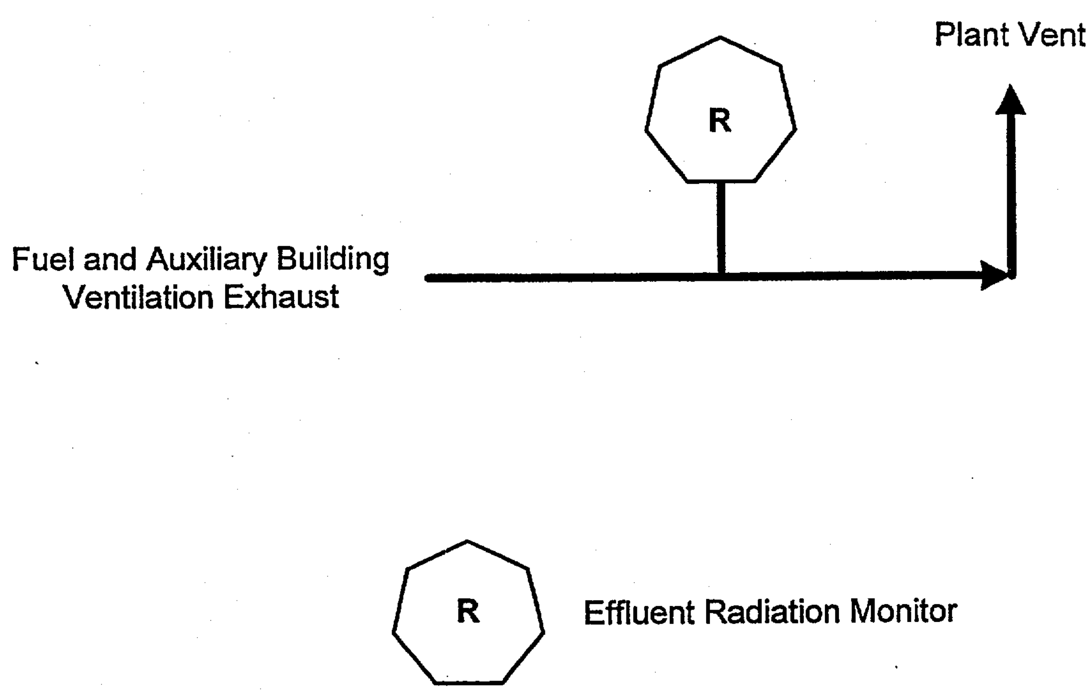


**Figure 6-1 Ventilation Exhaust Treatment System**

## 7.0 EFFLUENT MONITOR SETPOINT CALCULATIONS

### 7.1 LIQUID EFFLUENT MONITORS

This section will be used to ensure compliance with Control 3.2.1.1.

Alarm/trip setpoints (High Alarm) will be used for effluent radiation monitors as described below.

#### 7.1.1 LIQUID RADWASTE DISCHARGE MONITOR (PRM-9)

The setpoint for liquid radwaste discharge is dependent on the actual values for radwaste discharge flow rate, plant dilution flow rate, and isotopic composition of the effluents; thus, a variable radiation monitor setpoint must be utilized.

Prior to discharge, an isotopic analysis of the batch release will be made for principal gamma emitters to determine the required Dilution Ratio (DR) as follows:

$$DR = \Sigma_i \frac{A_i}{ECV_i} \tag{7-1}$$

where

$A_i$ = concentration of nuclide i in batch release, µCi/ml

$ECV_i$ = 10 CFR 20 effluent concentration value for nuclide i, µCi/ml.

The maximum tank discharge flow rate ($F_d$) is then determined as follows:

$$F_d = \frac{0.7 \times V_m}{DR} \tag{7-2}$$

where

$F_d$ = maximum tank discharge flow rate, gpm

7.1.1 LIQUID RADWASTE DISCHARGE MONITOR (PRM-9) (Continued)

$V_m$ = minimum expected plant dilution flow rate, gpm

0.7 = conservatism factor

The radiation monitor setpoint will then be calculated as follows:

$$\text{High Setpoint} = \frac{\Sigma_i\, A_i \times \epsilon_i}{0.7} \qquad (7\text{-}3)$$

where

High Setpoint = radiation monitor High Alarm setpoint, net cpm

$A_i$ = concentration of nuclide i in batch release, $\mu$Ci/ml

$\epsilon_i$ = radiation monitor efficiency for nuclide i, cpm per $\mu$Ci/ml

For batch releases of very low activity, the calculated radiation monitor setpoint may be statistically insignificant when compared to PRM background radiation fluctuations. In these cases, the radiation monitor setpoint may be calculated as follows:

$$\text{High Setpoint} = 3.0 \times \Delta\,\text{PRM} \qquad (7\text{-}4)$$

where

$\Delta$ PRM = PRM background reading fluctuation.

In addition, during low flow conditions, PRM-9 readings increase during the discharge due to increasing background radiation. If this effect causes the PRM to alarm, the High Setpoint can be reset.

## 7.0 EFFLUENT MONITOR SETPOINT CALCULATIONS

### 7.2 GASEOUS EFFLUENT MONITORS

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

#### 7.2.1 SETPOINT CALCULATIONS FOR NOBLE GAS EFFLUENT CHANNELS (PRM 2C)

Maximum setpoint values for noble gas effluent channels will be based on Kr-85. The limiting release rate for Kr-85 is the minimum of the release rates calculated by Equations 7-5 and 7-6.

$$Q_i \leq \frac{1}{2K_i} \tag{7-5}$$

$$Q_i \leq \frac{1}{0.33(L_i + 1.1\ N_i)} \tag{7-6}$$

where

$Q_i$ = limiting release rate for nuclide i, in Ci/sec

$K_i$ = total body dose factor for nuclide i, in rem/yr per Ci/sec

$L_i$ = beta skin dose factor for nuclide i, in rem/yr per Ci/sec

$N_i$ = gamma air dose factor for nuclide, in rad/yr per Ci/sec
(NOTE: these are "air" rads, not "tissue" rads)

1.1 = constant, the average ratio of tissue to air energy absorption coefficients with the units of rem/"air" rad

For Kr-85, $Q_i \leq$ 0.176 Ci/sec.

Radiation monitor setpoints are calculated using Equation 7-7.

$$\text{Setpoint} \leq \frac{Q_i \times \epsilon \times 60 \times 10^6}{F \times 2.83E4} \tag{7-7}$$

## 7.2.1 SETPOINT CALCULATIONS FOR NOBLE GAS EFFLUENT CHANNELS (PRM 2C) (Continued)

where

Setpoint = High alarm setpoint value, net cpm

60 = constant, sec/min

$10^6$ = constant, µCi/Ci

2.83E4 = constant, cc/ft$^3$

F = Fuel and Auxiliary Building Vent Exhaust effluent flow rate, cfm
= 105,000 cfm

$Q_i$ = release rate of Kr-85, Ci/sec
= 0.176 Ci/sec

$\epsilon$ = detector efficiency for Kr-85, cpm per µCi/cc
= 6.8E+7 cpm per µCi/cc

The setpoint for PRM 2C is 2.4E+5 cpm.

## 7.2.2 SETPOINT CALCULATIONS FOR PARTICULATE CHANNEL (PRM 2A)

Limiting release rates for particulates are calculated using Equation 7-8.

$$Q_p \le \frac{1-Q_{SR}\ R_{SR}}{0.67\ R_{LIM}} \qquad (7\text{-}8)$$

### 7.2.2 SETPOINT CALCULATIONS FOR PARTICULATE CHANNEL (PRM 2A) (Continued)

where

$Q_p$ = limiting release rate for particulates (excluding Sr-90), Ci/sec

$Q_{SR}$ = maximum quarterly release rate of Sr-90, Ci/sec from Table 7-1

$R_{SR}$ = particulate dose factor for Sr-90, rem/yr per Ci/sec

$R_{LIM}$ = largest particulate dose factor (excluding Sr-90) rem/yr per Ci/sec

The limiting release rate from Equation 7-8 will then be applied to Equation 7-7 with the appropriate detector efficiencies and effluent flow rates to obtain the particulate channel setpoints.

Normally, the setpoints would be based on the nuclide with the most restrictive dose factor (Sr-90). However, the dose factor for Sr-90 is so much higher than the dose factors for other particulates in relation to actual releases that to assume that all particulates are Sr-90 for setpoint calculations would be unnecessarily restrictive. The contribution of Sr-90 was weighted by the maximum quarterly release yet observed. Basing the setpoints on the next most restrictive nuclide, Cs-134, Equation 7-8 can be solved and the limiting release rate applied to Equation 7-7 to obtain the setpoint for PRM 2A.

For Cs-134, $Q_p \leq 2.1$ E-7 Ci/sec

## 7.2.2 SETPOINT CALCULATIONS FOR PARTICULATE CHANNEL (PRM 2A) (Continued)

The radiation monitor setpoint for PRM 2A can be calculated by solving Equation 7-7, where

$F$ = effluent flow rate, cfm
= 105,000 cfm (Fuel and Auxiliary Building vent exhaust)

$Q_i$ = release rate of Cs-134, Ci/sec
= 2.1E-7 Ci/sec

$\epsilon$ = detector efficiency for Cs-134, cpm per $\mu$Ci/cc
= 1.10E+12 cpm per $\mu$Ci/cc

and all other terms have been previously defined.

The setpoint for PRM 2A is 4.7E+3 cpm. No other PRMs have a particulate channel setpoint.

## 7.2.3 CONDENSATE DEMINERALIZER BUILDING EFFLUENT MONITORING

To ensure compliance with Control 3.2.2.1, samples will be analyzed monthly. Due to the limited effluent volume discharged from the building, it is not necessary to set release limits for this pathway.

TABLE 7-1

| Historical Particulate Releases | | | |
|---|---|---|---|
| **Year** | **Quarter** | **Curies Released** | |
| | | **Sr-90** | **Total** |
| 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 |

**TABLE 7-2**

| Effluent Pathway Flow Rates Used For Method 1 Setpoint Calculations | |
|---|---|
| **Effluent Monitor** | **Flow Rate (cfm)** |
| PRM-2: Fuel and Auxiliary Building Vent Exhaust | 105,000 |

**TABLE 7-3**

| Particulate Channel Detector Efficiencies (PRM 2A) | |
|---|---|
| **Nuclide** | **Detector Efficiency (cpm/$\mu$Ci/cc)** |
| Co-60 | 8.20E+11 |
| Sr-90 | 1.40E+12 |
| Cs-134 | 1.10E+12 |
| Cs-137 | 1.50E+12 |

# 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.

## 8.2.2 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.

## 8.3 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.

## 8.4 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

## 8.5 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

### DERIVATION OF NOBLE GAS DOSE FACTORS (K, L, M, N)

The noble gas dose factors were derived using the maximum annual average site boundary $\chi/Q$ for batch releases as follows:

$$K_i = DFB_i \times \chi/Q \times 10^{12} \times 10^{-3} \qquad \text{(A-1)}$$

$$L_i = DFS_i \times \chi/Q \times 10^{12} \times 10^{-3} \qquad \text{(A-2)}$$

$$M_i = DF_i^{\beta}i \times \chi/Q \times 10^{12} \times 10^{-3} \qquad \text{(A-3)}$$

$$N_i = DF_i^{\gamma}i \times \chi/Q \times 10^{12} \times 10^{-3} \qquad \text{(A-4)}$$

where

$K_i$ = gamma total body dose factor for nuclide i, rem/yr per Ci/sec

$L_i$ = beta skin dose factor for nuclide i, rem/yr per Ci/sec

$M_i$ = beta air dose factor for nuclide i, rad/yr per Ci/sec

$N_i$ = gamma air dose factor for nuclide i, rad/yr per Ci/sec

$DFB_i$ = Regulatory Guide 1.109 (Rev. 1, 10/77) total body dose factor, mrem/yr per pCi/$m^3$

$DFS_i$ = Regulatory Guide 1.109 (Rev. 1, 10/77) skin dose factor, mrem/yr per pCi/$m^3$

$DF_i^{\beta}$ = Regulatory Guide 1.109 (Rev. 1, 10/77) beta air dose factor, mrad/yr per pCi/$m^3$

$DF_i^{\gamma}$ = Regulatory Guide 1.109 (Rev. 1, 10/77) gamma air dose factor, mrad/yr per $pCi/m^3$

$\chi/Q$ = $1.3 \times 10^{-5}$ $sec/m^3$ (historical average maximum continuous site boundary $\chi/Q$, see Appendix C)

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

$10^{-3}$ = constant, rem/mrem or rad/mrad

The values of $K_i$, $L_i$, $M_i$, $N_i$, and $DFB_i$, $DFS_i$ $DF_i^{\beta}$, $DF_i^{\gamma}$ ,are given in Table A-1.

## TABLE A-1

| Noble Gas Dose Factors | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | Regulatory Guide 1.109 (Rev. 1, 10/77) Dose Factors | | | | | | | |
| Nuclide | $DF_i^\beta$ β-air (mrad/yr) (pCi/m³ | $DFS_i$ β-skin (mrem/yr) (pCi/m³) | $DF_i^\gamma$ γ-air (mrad/yr) (pCi/m³) | $DFB_i$ γ-body (mrem/yr) (pCi/m³) | $K_i$ γ-body (rem/yr) (Ci/sec) | $L_i$ β-skin (rem/yr) (Ci/sec) | $M_i$ β-air (rad/yr) (Ci/sec) | $N_i$ γ-air (rad/yr) (Ci/sec) |
| Kr-85 | 2.0E-3 | 1.3E-3 | 1.7E-5 | 1.6E-5 | 2.1E-1 | 1.7E+1 | 2.6E+1 | 2.2E-1 |

## 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 breath the air at the nearest residence with the highest $\chi/Q$ 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:

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

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 $Ci/m^3$

$R_i^G$ = ground plane pathway dose factor for nuclide i, mrem/yr per $Ci/m^2$-sec

$R_i^V$ = vegetable ingestion pathway dose factor for nuclide i, mrem/yr per $Ci/m^2$-sec

$R_i^M$ = meat ingestion pathway dose factor for nuclide i, mrem/yr per$Ci/m^2$-sec

$R_i^C$ = cow or goat milk ingestion pathway dose factor for nuclide i, mrem/yr per $Ci/m^2$-sec

$\chi/Q_c$ = atmospheric dispersion factor for continuous releases at the site boundary, $sec/m^3$

$D/Q_c$ = 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.

<u>Inhalation Pathway Dose Factor $R_i^I$</u>

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

where

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

(BR) = breathing rate of the receptor of child age group = 3700 $m^3$/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/m*$^2$-sec)

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

where

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

8760 = constant, hr/yr

$\lambda_i$ = decay constant for nuclide i, sec$^{-1}$

t = exposure time, 4.73 x $10^8$ sec (15 yr)

(DFG$_i$) = ground plane total body dose conversion factor for nuclide i, mrem/hr per pCi/m$^2$ (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/m*$^2$-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:

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

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, $8.6 \times 10^4$ 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, $5.2 \times 10^6$ 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)

fe = 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)

$\lambda_w$ = decay constant for removal of activity on leaf and plant surfaces by weathering, $5.73 \times 10^{-7}$ 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.

The concentration of tritium in vegetation is based on the airborne concentration rather than the deposition. Therefore, the $R_i^V$ for tritium is based on $\chi/Q$:

$$R^V_{H-3} = (10^{12})(10^3)\left[U^L_a f_L + U^S_a f_g\right](DFL_i)\,[0.75(0.5/H)] \qquad \text{(B-5)}$$

where

$10^3$ = constant, gm/kg

H = absolute humidity of the atmosphere, 8 gm/m$^3$ (Trojan Appendix I Evaluation, 5/76)

0.75 = fraction of total plant mass that is water

0.5 = ratio of the specific activity of the plant mass water to the specific activity of the atmospheric water

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

and all other terms have been defined previously.

Grass-Cow-Meat Pathway Factor $R^M_i$ (mrem/yr per Ci/m²-sec)

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

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)

$t_f$ = transport time from pasture to receptor, 1.73 x $10^6$ 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 x $10^6$ 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/m² (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/m² (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15).

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

and all other terms have been defined previously.

The concentration of tritium in meat is based on its airborne concentration rather than the deposition. Therefore, the $R_i^M$ for tritium is based on $\chi$/Q:

$$R_{H-3}^{M} = (10^{12})(10^{3})F_f Q_F U_{ap}(DFL_i)\ [0.75(0.5/H)] \tag{B-7}$$

where all terms have been defined previously.

<u>Grass-Goat-Milk Pathway Factor $R_i^C$ *(mrem/yr per Ci/m²-sec)*</u>

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

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/m$^2$ (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, $1.73 \times 10^5$ sec (2 days) (Regulatory Guide 1.109, Rev. 1, 10/77, Table E-15)

$t_h$ = transport time from pasture, to harvest, to goat, $7.78 \times 10^6$ 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.

<u>Grass-Cow-Milk Pathway Factor $R_i^C$ *(mrem/yr per Ci/m²-sec)*</u>

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

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/m$^2$ (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, $1.73 \times 10^5$ 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 x $10^6$ 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.

The concentration of tritium in milk is based on the airborne concentration rather than the deposition. Therefore, the $R_i^C$ for tritium is based on $\chi$/Q:

$$R_{H-3}^{C} = (10^{12})(10^3)F_m Q_F U_{ap}(DFL_i)\ [0.75(0.5/H)] \qquad \text{(B-10)}$$

where all parameters have been defined previously.

## TABLE B-1

| Dose Factors for Controlling Exposure Location | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | Child Dose Factors | | | | Stable Element Transfer Coefficients | | Dose Parameters | | | | | |
| | | | | | | Meat | Milk | Inhalation | Ground | Vegetable | Meat | Milk |
| Nuclide | $\lambda_i$ ($sec^{-1}$) | $DFA_i$ (mrem)/(pCi) | $DFG_i$ (mrem/hr)/($pCi/m^2$) | $DFL_i$ (mrem)/(pCi) | Retention Factor (r) | $F_f$ (day)/(kg) | $F_m$ (day)/(l) | $R_i^I$ (mrem/yr)/($Ci/m^3$) | $R_i^G$ ($m^2$-mrem/yr)/(Ci/sec) | $R_i^V$ ($m^2$-mrem/yr)/(Ci/sec) | $R_i^M$ ($m^2$-mrem/yr)/(Ci/sec) | $R_i^C$ ($m^2$-mrem/yr)/(Ci/sec) |
| H-3 | 1.79E-09 | 3.04E-07 | 0.0 | 2.03E-07 | 0.2 | 1.2E-02 | 1.7E-01 | 1.12E+09 | 0.0 | 4.01E+09[a] | 2.34E+08[a] | 2.7E+10 |
| 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] $\frac{mrem/yr}{Ci/m^3}$

TABLE B-2

| Particulate Dose Factors | |
|---|---|
| **Nuclide** | **$R_i$ Composite Dose Factor at Site Boundary (rem/yr) (Ci/sec)** |
| H-3 | 1.07E+02 |
| 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.

| 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 | 6.10E-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 |

| 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 ($\chi/Q$, sec/m$^3$). The methodology contained in this Appendix assumes quarterly average meteorology and ground-level release sector average $\chi/Q$ (Meteorology and Atomic Energy Equation 3.144). The following equation is used to calculate the adjusted $\chi/Q$ values for specific locations within the site boundary:

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

where

$f$ = wind frequency, percent

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

$x$ = downwind distance, meters

$n$ = number of cardinal compass sectors = 16

$\sigma_z$ = vertical dispersion parameter, Pasquill Class E

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

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 ($\chi/Q$) at the location of interest to the $\chi/Q$ at the site boundary:

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

where

| | | |
|---|---|---|
| $D^A$ | = | dose at the location of interest (mrem) |
| $(\chi/Q)^A$ | = | adjusted $\chi/Q$ at the location of interest (from Equation D-1) (sec/$m^3$) |
| $(\chi/Q)^{SB}$ | = | $\chi/Q$ 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. Based upon the historical data, a batch release correction factor of 3.8 will be used and a continuous release correction factor of 4.0 will be used.

## TABLE D-1

| Correction Factor<br>for Persons Utilizing Unrestricted Areas<br>Within the Site Exclusion Area Boundary | | | |
|---|---|---|---|
| **QUARTER** | **YEAR** | **BATCH**<br>$\frac{(X/Q)^{A}}{(X/Q)^{SB}}$ | **CONTINUOUS**<br>$\frac{(X/Q)^{A}}{(X/Q)^{SB}}$ |
| 3 | 1992 | 2.5 | 2.8 |
| 4 | 1992 | 2.1 | 1.9 |
| 1 | 1993 | 1.5 | 1.5 |
| 2 | 1993 | 1.7 | 2.4 |
| 3 | 1993 | 3.8 | 4.0 |
| 4 | 1993 | * | 1.6 |
| **MAXIMUM** | | **3.8** | **4.0** |
| **MINIMUM** | | **1.5** | **1.5** |
| **AVERAGE** | | **2.3** | **2.4** |

* No batch releases during the quarter.

## 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.

| 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 |

| 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 |

| 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 |

| 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