*** QA RECORD WHEN COMPLETED ***

RC 2 4 2
CFP GEN ENGR 7-6
OOR Admin Services
Letter Number NA
System Number NA
Number of Pages 1
Document Date
Calc. Reference #

# TROJAN CALCULATION COVER SHEET

Sheet 1 Cont'd on Sheet 2

Title REACTOR VESSEL WASTE CLASSIFICATION ANALYSIS

Trojan Nuclear Plant

Calculation No. RPC 97-018

Structure CONTAINMENT

Supersedes Calculation No. 0

System 62/64

Quality-Related Yes/No

Component REACTOR VESSEL

Status: X Final ___ Interim

References (PMR/DPMR, SPEER, MR, PSC, etc.)

| Affected Document No. | Has Been Changed by Identify Change Vehicle: (MR, DPMR, DCP, PCF, SPEER, PSC, etc.) | Or Revision has been Deferred by (Identify Memo, CTL, etc.) | Responsible Supervisor/Date (Deferrals Only) |
|---|---|---|---|
| | | | |

Calculation Objective TO DETERMINE THE PROPER WASTE CLASSIFICATION OF THE REACTOR VESSEL WITH ALL ITS INTERNAL SUB-COMPONENTS AS ONE UNIT.

Revision Description

| Rev. No. | Preparer | Date | Verified By | Date | Approved By | Date |
|---|---|---|---|---|---|---|
| 0 | Michael Murdock | 5/27/97 | [signature] | 5/28/97 | [signature] | 5/28/97 |
| | | | | | | |
| | | | | | | |
| | | | | | | |

**PORTLAND GENERAL ELECTRIC CALCULATION SHEET**

Calculation No. RPC 97-018 Revision 0 Sheet 2 of 8

Preparer Michael Murdock MM Date 5/27/97

Verifier JK Huey Date 5/28/97

Table of Contents

## PORTLAND GENERAL ELECTRIC CALCULATION SHEET

Calculation No. RPC 97-018 Revision 0 Sheet 3 of 8

Preparer Michael Murdock Date 5/27/97

Verifier [signature] Date 5/28/97

### REACTOR VESSEL WASTE CLASSIFICATION ANALYSIS

**OBJECTION**

The objective of this calculation is to determine the proper waste classification of the Reactor Vessel with all its internal sub-components as one unit.

**ACCEPTANCE CRITERIA:**

None

**SUMMARY:**

The purpose of this calculation is to aid in the planning for the eventual shipping for burial of the reactor vessel. This calculation can be used as a guideline when an actual shipping date is determined. The activities will need to be decay corrected when a date is known.

The approach to determine the waste class will be in accordance with 10CFR 61 and the US Ecology burial license.

The activity is obtained from the activation analysis done as part of the Site Characterization in support of the Trojan Decommissioning Plan and RPC 96-008, "Reactor Vessel and Internals Surface Area Contamination". The calculation will be done under the guidelines of RP 310, Rev 2.

**ASSUMPTIONS, DESIGN INPUTS, AND METHODOLOGY:**

Both the Activation and corrosion activity represent activity decay corrected to 11/97.

In accordance with section 3.3 of the Branch Technical Position, the reactor vessel and core internals is considered to be one component containing neutron activated metals incorporating radioactive in its design thus allowing concentrating averaging over the displaced volume of the material.

The displaced volume is the mass of the metals only as identified in Table of TLG's calculation titled "RVAIR - Weight and C. G." as received under PGE letter No. 102-97L. This does not include any closure plates, impact limiters, or shielding.

The majority of the activation is located in the region between the upper and lower core plates including the vessel cladding and walls.

The surface contamination is considered to be distributed over all the reactor vessels' internal surfaces. Although <1% of the total activity, the surface contamination will be considered in this calculation.

The contribution from the Incore Flux Thimbles, currently in place in the vessel, has not been included in the overall total activity but is assumed to be less than .05% of the total activity.

**PORTLAND GENERAL ELECTRIC    CALCULATION SHEET**

Calculation No. RPC 97-018  Revision 0  Sheet 4 of 8

Preparer Michael Murdock  Date 5/27/97

Verifier SR Huey  Date 5/28/97

Displaced Volume of Metal = 2622.2 $ft^3$ or 74,266.368 cc ( reference TLG's "Weight and C. G. Calculation)

Material Weight = Displaced Volume x 7.86 gms/cc or 583,733,653 gms

Activation Activity = 2.007 curies ( See Att. 1 , taken from table 4.7.30 of the Site Characterization Report)

Corrosion or Surface area Activity = 155.2 curies ( from Table 1 of RPC 96-008 )

**RESULTS:**

The reactor vessel and sub-components are Class C waste.

Table 1 results = .328 of Class C limit

Table 2 results = .303 of Class C limit

SNM = 1.55 gms of Pu

Total Pu = 5.25 curies

**REFERENCES:**

1. RPC 96-008, Reactor Vessel & Internals Surface Area Activity
2. RVAIR 102-97L, PGE Tracking No. of TLG's RVAIR Weight and C.G. Calculation
3. RP 310 Rev 2, Determination of Radioactive Material Shipping and Waste Classifications.
4. Table 4.7.30 of the Trojan Site Characterization Report

**BODY OF CALCULATION:**

1. RP Form 73 & 78 applicable pages of RP 310
2. Attachment 1, QPRO Spreadsheet of Table 4.7.30

Form RP 73
Page 4 of 5

Shipment Number: N/A
Package Number: Reactor Vessel

12. Determination of Concentration

Material Volume = 7.43E+07 cc (D)

Material Weight = 5.84E+08 gm (E)

| Isotope | A<br>Ship Date<br>Activity<br>(mCi) | B<br>Conc<br>Isotope<br><5 yr T 1/2<br>(uCi/cc)<br>(Ax1E3)/D | C<br>Conc<br>Isotope<br>>5 yr T 1/2<br>(uCi/cc)<br>(Ax1E3)/D | F<br>Conc.<br>Isotope<br>(nCi/gm)<br>(Ax1E6)/E | G<br>Conc<br>Factor<br>(gm/mCi) | H<br>Gram<br>of<br>Isotope<br>(AxG) |
|---|---|---|---|---|---|---|
| H-3 | 8.11E+01 | | 1.09E-03 | | | |
| H-3 (A) | 4.46E+05 | | 6.00E+00 | | | |
| C-14 | 1.15E+02 | | 1.55E-03 | | | |
| C-14 (A) | 2.18E+05 | | 2.93E+00 | | | |
| Mn-54 | 8.57E+01 | 1.15E-03 | | | | |
| Mn-54 (A) | 2.16E+06 | 2.91E+01 | | | | |
| Fe-55 | 2.77E+04 | 3.73E-01 | | | | |
| Fe-55 (A) | 6.97E+08 | 9.39E+03 | | | | |
| Co-60 | 9.92E+04 | | 1.34E+00 | | | |
| Co-60 (A) | 1.15E+09 | | 1.55E+04 | | | |
| Nb-94 (A) | 3.29E+03 | | 4.42E-02 | | | |
| Ni-59 (A) | 9.53E+05 | | 1.28E+01 | | | |
| Sb-125 | 1.67E+03 | 2.25E-02 | | | | |
| Sb-125 (A) | 4.03E+02 | 5.42E-03 | | | | |
| Eu152 (A) | 1.49E+04 | | 2.00E-01 | | | |
| Ce-144 | 4.64E+01 | 6.25E-04 | | | | |
| Ni-63 | 2.00E+04 | | 2.69E-01 | | | |
| Ni-63 (A) | 1.57E+08 | | 2.12E+03 | | | |
| Sr-90 | 9.24E+02 | | 1.24E-02 | | | |
| Tc-99 (A) | 7.06E+02 | | 9.50E-03 | | | |
| TRU | 1.81E+02 | | 2.43E-03 | 3.09E-01 | | |
| Pu-238 | 8.35E+01 | | 1.12E-03 | 1.43E-01 | 5.8E-05 | 4.84E-03 |
| Pu-239/40 | 9.33E+01 | | 1.26E-03 | 1.60E-01 | 1.60E-02 | 1.49E+00 |
| Pu-241 | 5.07E+03 | | 6.83E-02 | 8.69E+00 | 9.60E-06 | 4.87E-02 |
| Cm-242 | 1.56E-02 | 2.10E-07 | | 2.67E-05 | | |
| Totals | 2.01E+09<br>Sum A<br>(mCi) | 9.42E+03<br>Sum B<br>(uCi/cc) | 1.76E+04<br>Sum C<br>(uCi/cc) | | | 1.55<br>Sum H<br>(gms. of Pu) |

Prepared by: Michael Murdock MM
Checked by: SKN

I. Michael Murdock Verify that this is a true computer generated copy of RP 310, Rev 2

Shipment Number N/A
Package Number Reactor Vessel

13. Evaluation of results for Burial Site Limitations and possible DOT/NRC Form 741 Shipment.

a. Determine total grams of SNM:

| 1.55 | + | 0 | + | 0 | = | 1.55 |
|---|---|---|---|---|---|---|
| Grams of Pu | | Grams of U-235 | | Grams of U-233 | | Grams of SNM |

If SNM > or = 1 gram, notify Radwaste Supervisor.

b. Burial Site TRU is TRU with T-1/2 >5 yrs except Pu-238, Pu-239/240, Pu-241 and Cm-242 from Column F.

Burial Site TRU: 3.09E-01 (nCi/gm) Limit <10 nCi/gm

c. Waste Class TRU is TRU with T-1/2 >5 yrs and PU-238 and Pu-239/240 from Column F.

Waste Class TRU:

| 3.09E-01 | + | 1.43E-01 | + | 1.60E-01 | = | 6.12E-01 |
|---|---|---|---|---|---|---|
| (TRU) | | (Pu-238) | | (Pu-239/240) | | (nCi/gm) |

14. Sum all isotopes with half life <5 yrs:

9.42E+03 (uCi/cc) Limit <7 UCi/cc

15. Sum All isotopes with half life >5 yrs:

1.76E+04 (uCi/cc) Limit <1 uCi/cc

Note: If any of the above limits are exceeded, notify the Radwaste Specialist

Prepared by Michael Murdock

Checked by [signature]



I. Michael Murdock Verify that this is a true computer generated copy of RP 310, Rev 2

FORM RP 78
DETERMINATION OF
WASTE CLASSIFICATION

Shipment No. N/A
Package No. Reactor Vessel

| | A | Class A | | Class B | | Class C | |
|---|---|---|---|---|---|---|---|
| | | B | | C | | D | |
| Table 1 Isotopes | Isotopic Conc. (uCi/cc) | Limit (uCi/cc) | Quotient (A/B) | Limit (uCi/cc) | Quotient (A/C) | Limit (uCi/cc) | Quotient (A/D) |
| C-14 | 1.55E-03 | 8.00E-01 | 1.94E-03 | (b) | | 8.00E+00 | 1.94E-04 |
| C-14 (A) | 2.93E+00 | 8.00E+00 | 3.67E-01 | (b) | | 8.00E+01 | 3.67E-02 |
| Tc-99 | 9.50E-03 | 3.00E-01 | 3.17E-02 | (b) | | 3.00E+00 | 3.17E-03 |
| Ni 59 (A) | 1.28E+01 | 2.20E+01 | 5.83E-01 | (b) | | 2.20E+02 | 5.83E-02 |
| Nb-94 (A) | 4.42E-02 | 2.00E-02 | 2.21E+00 | (b) | | 2.00E-01 | 2.21E-01 |
| TRU (a,c) | 6.12E-01 | 1.00E+01 | 6.12E-02 | (b) | | 1.00E+02 | 6.12E-03 |
| Pu-241 (a) | 8.69E+00 | 3.50E+02 | 2.48E-02 | (b) | | 3.50E+03 | 2.48E-03 |
| Cm-242 (a) | 2.67E-05 | 2.00E+03 | 1.34E-08 | (b) | | 2.00E+04 | 1.34E-09 |
| | Sum of Table 1 Quotients | | 3.28E+00 | | | | 3.28E-01 |

NOTE: TRU with T-1/2 >5 yrs = all TRU except Pu-238, Pu-239/240, Pu-241 and Cm-242.

| | A | Class A | | Class B | | Class C | |
|---|---|---|---|---|---|---|---|
| | | B | | C | | D | |
| Table 2 Isotopes | Isotopic Conc. (uCi/cc) | Limit (uCi/cc) | Quotient (A/B) | Limit (uCi/cc) | Quotient (A/C) | Limit (uCi/cc) | Quotient (A/D) |
| H-3 | 6.00E+00 | 4.00E+01 | 1.50E-01 | (d) | N/A | (d) | N/A |
| Co-60 | 1.55E+04 | 7.00E+02 | 2.21E+01 | (d) | N/A | (d) | N/A |
| Ni-63 | 2.69E-01 | 3.50E+00 | 7.69E-02 | 7.00E+01 | | 7.00E+02 | 3.85E-04 |
| Ni- 63 (A) | 2.12E+03 | 3.50E+01 | | 7.00E+02 | | 7.00E+03 | 3.03E-01 |
| Sr-90 | 1.24E-02 | 4.00E-02 | 3.11E-01 | 1.50E+02 | | 7.00E+03 | 1.78E-06 |
| Cs-137 | | 1.00E+00 | | 4.40E+01 | | 4.60E+03 | |
| Nuclides w/ (T-1/2 ,<5 yr) | 9.42E+03 | 7.00E+02 | 1.35E+01 | (d) | N/A | (d) | N/A |
| | Sum of Table 2 Quotients | | 3.61E+01 | | | | 3.03E-01 |

a - Units are in nCi/gm

b - If Class A limit is exceeded, the waste is Class C or greater.

c - TRU = TRU with T-1/2 >5 yrs + Pu-238 + Pu-239/240.

d - No Limit.

NOTE: It is not necessary to list on the manifest any nuclide whose Class A quotient is less than 0.01 except C-14, Tc-99, I-129, and H-3.

Prepared by Michael Murdock

Checked by [signature]



I Michael Murdock Verify that this is a true computer generated copy of RP 310, Rev 2

**PORTLAND GENERAL ELECTRIC** **CALCUALTION SHEET**

Calculation No. RPC 97-018 Revision 0 Sheet 8 of 8

Preparer Michael Murdock Date 05/27/97

Verifier Date 5/28/97

**Table 1**

**REACTOR VESSEL**

**CURIE CONTENT FIVE YEARS AFTER SHUTDOWN (ACTIVATION ONLY)**

| SUB-COMPONENTS | H-3 | C-14 | Sb-125 | Mn-54 | Eu-152 | Fe-55 | Co-60 | Ni-59 | Ni-63 | Nb-94 | Tc-99 | Totals |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Core Baffle | 2.336E+02 | 1.186E+02 | 3.320E-01 | 1.632E+03 | 7.631E-03 | 3.732E+05 | 7.248E+05 | 5.329E+02 | 8.688E+04 | 2.225E+00 | 5.091E-01 | 1.187E+06 |
| Core Formers | 8.755E+01 | 7.010E+01 | 5.108E-02 | 2.110E+02 | 1.180E-01 | 2.293E+05 | 2.412E+05 | 2.614E+02 | 4.781E+04 | 5.682E-01 | 7.650E-02 | 5.189E+05 |
| Lower Core Barrel | 5.465E+01 | 1.183E+01 | 1.579E-03 | 1.633E+02 | 7.867E+00 | 3.733E+04 | 8.555E+04 | 6.519E+01 | 9.156E+03 | 2.409E-01 | 6.102E-02 | 1.323E+05 |
| Upper Core Barrel | 5.151E-04 | 1.115E-04 | 1.488E-08 | 1.539E-03 | 7.415E-05 | 3.518E-01 | 8.063E-01 | 6.144E-04 | 8.629E-02 | 2.270E-06 | 5.762E-07 | 1.247E+00 |
| Thermal pads | 1.303E+01 | 2.832E+00 | 2.003E-04 | 3.919E+01 | 2.997E+00 | 8.933E+03 | 2.058E+04 | 1.567E+01 | 2.193E+03 | [illegible]E-02 | 1.469E-02 | 3.178E+04 |
| Vessel Clad | 5.786E+00 | 1.265E+00 | 6.640E-05 | 6.790E+00 | 5.358E-01 | 4.132E+03 | 6.165E+03 | 7.574E+00 | 1.021E+03 | [illegible]4E-02 | 2.584E-03 | 1.134E+04 |
| Vessel Wall | 4.198E+00 | 6.222E-03 | 1.332E-02 | 8.041E+00 | 1.142E+00 | 1.813E+03 | 3.928E+02 | 1.338E-01 | 1.877E[illegible]01 | 1.751E-03 | 4.445E-03 | 2.238E+03 |
| Lower Core Plate | 2.918E+01 | 9.392E+00 | 4.323E-03 | 8.017E+01 | 3.715E-01 | 3.022E+04 | 5.143E+04 | 4.757E+01 | 7.130E[illegible] | 1.351E-01 | 2.934E-02 | 8.895E+04 |
| Lower Core Sup Col | 8.758E+00 | 1.829E+00 | 6.465E-05 | 4.417E+00 | 4.546E-01 | 6.052E+03 | 7.419E+03 | 1.136E+01 | 1.564E+03 | 1.456E-02 | 1.733E-03 | 1.500E+04 |
| Lower Core Sup | 2.018E-02 | 4.356E-03 | 1.630E-10 | 5.103E-02 | 7.771E-03 | 1.386E+01 | 2.919E+01 | 2.490E-02 | 3.422E+00 | 7.907E-05 | 2.010E-05 | 4.658E+01 |
| Below Low Core Sup | 7.563E-06 | 1.492E-06 | 0.000E+00 | 3.390E-06 | 2.207E-06 | 4.940E-03 | 6.062E-03 | 9.576E-06 | 1.239E-03 | 1.168E-08 | 1.396E-09 | 1.227E-02 |
| Upper Core Plate | 7.512E+00 | 1.673E+00 | 1.291E-04 | 1.555E+01 | 1.037E+00 | 5.377E+03 | 9.988E+03 | 9.612E+00 | 1.323E+03 | 2.586E-02 | 5.851E-03 | 1.672E+04 |
| Upper Core Sup Col | 1.399E+00 | 2.744E-01 | 6.527E-07 | 6.787E-01 | 3.272E-01 | 9.077E+02 | 1.129E+03 | 1.754E+00 | 2.276E+02 | 2.207E-03 | 2.700E-04 | 2.269E+03 |
| Totals | 4.457E+02 | 2.178E+02 | 4.028E-01 | 2.161E+03 | 1.487E+01 | 6.973E+05 | 1.149E+06 | 9.532E+02 | 1.573E+05 | 3.286E+00 | 7.056E-01 | 2.007E+06 |

NOTE: 1. The activity is from the Site Characterization Report Table 4.7.30

2. Nuclides omitted were; Ar-39, Ca-41, Ca-45, Sn-119m, and Te-125m due to such a small contirbution and not a factor in waste classification

**ATTACHMENT III**

->Proj. File
-> Lw Reg File

US Ecology, Inc.
P.O. Box 638
Richland, Washington 99352
509/377-2411

**USEcology**

an *American Ecology* company

March 12, 1996

Gary Robertson
Head of Waste Management Section
State of Washington
Department of Health
Division of Radiation Protection
Airdustrial Center, Building 5
PO Box 47827
Olympia, Washington 98504-7827

Dear Mr. Robertson:

US Ecology and Portland General Electric would like to take this opportunity to address some of the issues pertaining to the disposal of the Trojan Reactor Vessel at the Low-Level Radioactive Waste Disposal Site near Richland Washington. We are specifically requesting the Department's concurrence on the waste classification of the reactor vessel.

The reactor vessel package will consist of the vessel internals and the reactor pressure vessel as one component. The vessel will be certified as an NRC Type B package. We believe this package meets the requirement for classification as a Class C Stable waste form. Void spaces will be filled to the maximum extent possible with a concrete based grout. These items are discussed in detail below in addition to issues of concern raised at our January 25 presentation.

## Stability

Washington Administrative Code 246-250-050 specifies the stability requirements for radioactive waste disposal. Additionally, NUREG-0782 (Draft Environmental Impact Statement on 10CFR61, Volume 2) provides the same guidance for stability for disposal purposes. Specifically section 5.5.2.4 of the NUREG states that stable waste forms must maintain their physical dimensions and consistency under the conditions of compressive load, radiation, and biodegradation expected to be encountered in disposal. This is to preclude slumping, collapse or other failing of the trench cap; the need for active long-term maintenance and the ability to predict long-term performance. Under the section "Form of the Waste as Generated", activated steel from nuclear reactors is given as one of the examples of waste that meets stability.

With regard to the Trojan Reactor Vessel, the physical dimension and consistency will be maintained under the compressive load of shallow land burial due to the inherent construction of the vessel. The carbon and stainless steel along with the concrete grout will form a solid structure with more than adequate strength to prevent any deformation. Radiation effects as far as maintaining physical dimensions and consistency, are not a concern since the vessel was designed to withstand much higher radiation fields when the reactor was in operation. Biodegradation is also not a concern because the vessel is steel. Since the proposed disposal method meets criteria for stability and void reduction, due to the waste being irradiated metal and void spaces being filled with a concrete grout, we do not believe a specific topical report should be necessary.

## Niobium 94

Total Nb-94 contained in the reactor vessel under this proposal is 3.290 curies. The breakdown of the individual subcomponents within the vessel are presented in the table below.

| Subcomponent | Activity (Ci) | % of Total Activity |
|---|---|---|
| Core Baffle | 2.230 | 68% |
| Core Former | 0.568 | 17% |
| Lower Core Plate | 0.135 | 4% |
| Remainder | 0.360 | 11% |
| TOTAL | 3.290 | 100% |

Under the proposal where the reactor vessel internals are segmented and shipped in individual liners the total niobium content would be 0.360 curies. It should be noted that Nb-94 is generally considered to be an external dose hazard. Due to the thickness of the reactor vessel walls and external shielding combined with the shielding provided by the grout used to fill the void spaces, external dose rates on the outside of the vessel will be less than 200 mr/hr. The combined shielding is necessary due to the package containing an estimated 1 $E^6$ curies of Co-60 at time of shipment. Given that the package will adequately reduce the radiation levels from the 1 $E^6$ curies of Co-60, it will be more than sufficient in minimizing intruder dose in the future from Nb-94.

## Disposal Trench

To provide for the segregation of class A wastes, any future trenches must maintain total separation of stable and unstable waste forms between trenches. Presently, the open Class C trenches do not have sufficient space to efficiently dispose of the reactor vessel. Trench 12 will be constructed as the new stable trench which will have the space to accept the reactor vessel. A portion of the trench will be constructed with a ramp to allow the heavy haul trailer access to the trench bottom. The ramp will have a slope of about 4-6% slope with no area greater than 10% slope.

**Source Term**

An individual pathway analysis, to determine the impact of the increase in the site's source term with the disposal of this component, has been completed. This analysis, entitled "Trojan Reactor Vessel Dose Analysis" is enclosed as Attachment 1 to this letter.

**Voids**

The grouting process of the Trojan reactor vessel will be the same process used on the Trojan steam generators. The steam generators were filled with a low density grout in containment through several fill and vent connections. Approximately 2 days later the generators were moved out of containment and inspected for additional voids resulting from settling. An approximately 5% void was refilled using the same grout prior to final closure of the generators.

**License Variance**

A License variance request will be submitted to exceed the possession limit of section 6.A. (60,000 curies) of the site license WN-I019-2. The variance will be temporary to cover the reactor vessel shipment only. The variance request will be for approximately 2,100,000 curies.

**Waste Classification**

Attached is a final curie content and waste classification for the vessel. As you can see from the attached calculation sheets provided by PGE, the waste will be Class C. This information is based on the latest data available but is subject to revision once the vessel is removed and can be better characterized. Any revision to the final curie content will most likely be in the downward direction since the present calculations assume the most conservative assumptions.

Your timely review and comments with this matter would greatly be appreciated. Please contact me at (800) 567-2372 if I can be of any further assistance.

Sincerely,

Arthur J. Palmer III, CHP
Chief Radiological Control
& Safety Officer

# TROJAN REACTOR VESSEL DOSE ANALYSIS

# TABLE OF CONTENTS

# 1. GROUND-WATER PATHWAY DOSE ANALYSIS

## 1.1 PURPOSE

This calculation examines the expected dose via the ground-water pathway attributed to the disposal of the Trojan Reactor Vessel as a single component at the US Ecology Low-Level Radioactive Waste Disposal Facility in Richland, Washington.

## 1.2 APPROACH

The calculation follows the methodology used in the Dose Analysis for the 1996 Closure Plan. The models used in the analyses are described in that plan. A simplified description of the analysis is as follows.

The metal reactor vessel is buried intact in the trench. Infiltrating water (from precipitation) that migrates through the cap comes in contact with the reactor vessel, and leaches radionuclides from the vessel. The leachate migrates downward to the water table, where it is diluted in the uppermost aquifer. A hypothetical well withdraws the water for a subsistence farming family, that irrigates a vegetable ç rden and pastureland for a dairy cow. The primary dose for this pathway is via inges' ..

## 1.3 DATA

The data used in this analysis are presented in Table 1; support for this data is presented in other calculations and in the Closure Plan, and referenced in Table 1.

### 1.3.1 Source Term

The activity inventory for the analysis comes from Portland General Electric. The reactor vessel activity is presented for November 1, 1997, 5 years post shutdown. The activity is separated into surface contamination and activation, with the activation totaling $2.01 \times 10^6$ Curies (Ci), and surface contamination totaling 357.9 Curies.

The development of the source term for the analysis follows the method used in the Closure Plan. Only long-lived, high activity isotopes are expected to remain in any significant quantity after migration through the vadose zone. The cut-off for isotopes was for half-lives equal or greater than 0.1 times the travel time through the vadose zone, and activities greater than 1 Curie. Additionally, Sr-90, an isotope which is easily uptaken by humans, was included.

Ten radionuclides were selected from the activity inventory for the source term for the ground-water pathway analysis. Several other isotopes were included in the analysis for comparison with the Closure Plan. The source term for the reactor vessel is presented in Table 2.

### 1.3.2 Assumptions

The assumptions used in this analysis are as follow.

The peak concentrations of leached constituents in the leachate are assumed to reach the ground-water at the same time, and to reach the well at the same time. This assumption is conservative because it maximizes the exposure value to the hypothetical individual.

There is no time delay associated with the leaching. This assumption does not account for the gradual leaching and removal of radionuclides from the reactor itself. The assumption is that the activity is distributed uniformly, and can be leached uniformly from the reactor vessel. The assumption is conservative because the radionuclides will leach from the vessel slowly, thereby decreasing the amount of radionuclide available for transport.

Solubility and distribution coefficients from the Closure Plan will be used. The leaching concentration of radionuclides from the metal reactor vessel are likely to be lower than the solubility of the radionuclides, but since no other values could be located, these conservatively larger values will be used.

The assumptions described above represent an upper bounding condition on the expected dose attributed to the disposal of the Trojan reactor vessel via the ground-water pathway.

## 1.4 ANALYSIS

The transport of the source term isotopes through the vadose zone was modeled using the TRANSS program. Two infiltration rates were modeled, 0.2 inches/year and 0.05 inches/year. These two cases simulate the infiltration of the area surrounding the Facility (natural conditions, as if the final cap was completely ineffectual), and the conditions over the Facility with the final cap in place and functioning as designed, respectively. These infiltration rates are the result of natural and expected precipitation at the Facility.

The maximum concentration for each radionuclide was selected for the input to the aquifer. Mixing and dilution occurs in the aquifer. The flow through the aquifer is greater than the recharge rate from infiltrating water, therefore dilution of the leachate occurs. The dilution factor is dependent on the infiltration rate, and is calculated to be about 0.003 for 0.2 inches/year, and 0.0007 for 0.05 inches/year. Multiplying the leachate at the ground-water table by the dilution factor yields the expected concentration in the hypothetical well.

Well concentrations less than $1 \times 10^{-50}$ pCi/L are eliminated from further analysis. This left seven radionuclides for dose analysis. The isotopes for dose analysis are listed in Table 3.

### 1.5 SUMMARY

The well concentration becomes the input to the PRESTO-II computer code for dose analysis. The results of the dose analysis are presented in Table 4. These output data are presented on the fourth page of the PRESTO-II output, under the selected individual dose equivalent (in mrem/year). The doses are small, and are less than 1 mrem/year to any organ.

The PRESTO-II model did not identify dose factors for Nb-94 in its internal database. The dose for Nb-94 was calculated by the equations upon which the PRESTO-II model is based. The niobium dose is checked in another calculation. The dose calculated from Nb-94 was far less than 0.01 mrem/year.

## 2. DIRECT EXPOSURE DOSE ANALYSIS

### 2.1 INTRODUCTION

This calculation examines the direct gamma exposure possible from the disposal at the Richland Facility of the intact Trojan Reactor Vessel. The reactor vessel is to be shielded by a soil cover. The exposure of an individual while outside man-made structures was examined.

### 2.2 CONCEPTUAL MODEL

The conceptual model for the direct exposure calculation is that an intruder could be exposed to direct gamma radiation from the waste buried in the trench. For this analysis, consideration is given to the following potential scenarios:

1. At closure, a Facility operator/maintenance person may receive direct exposure while standing directly over the trench;

2. After closure, a person standing at the Facility boundary may receive direct exposure from a capped trench;

3. At some future date after closure, an inadvertent intruder may receive direct exposure by intermittently passing over the Facility area and receive direct exposure via this pathway; and

4. At some future date after closure, an inadvertent intruder may receive direct exposure by living within a structure constructed into the waste and/or cover (e.g. the basement scenario).

The Facility is located in the area identified as the Central Plateau. The findings of the Hanford Future Site Uses Working Group has identified this area as the location for waste storage from cleanup activities from the rest of the Hanford Reservation. One

pathway, direct exposure via a basement construction in the waste, was eliminated as a result of the assumption of post-closure institutional control to be exercised outside of the Facility. The Central Plateau region, in which the Facility is located, is anticipated to be the waste storage area for the cleanup activities to be performed at the Hanford reservation. As such, the Facility area will have restricted access as a result of the larger-scale storage area. Therefore, the basement construction scenario was eliminated on the basis of the following assumptions.

- Institutional control for the Central Plateau area will eliminate long-term access (e.g. no residential or commercial construction) to the Facility; and
- The elimination of on-site construction will eliminate the possibility of people living over the Facility.

Therefore, the bounding (highest exposure) case for the above scenarios is the first case involving the Facility operator/maintenance person (receptor).

The conceptual model for this pathway considers the radiation source to be a large solid mass, since the source is the waste in the burial trenches. The receptor is assumed to be standing on top of the trench cap, which serves as a shield to direct radiation. The effects of cover thickness on the direct exposure dose rate were examined. The analysis performed for the Facility was undertaken to provide conservative radiological impacts to a hypothetical maximally exposed individual. The assumptions and data are considered extremely conservative for the conditions at the Facility, and are discussed in the following sections.

## 2.3 APPROACH

### 2.3.1 Selection and Justification of Model

The computer program MICROSHIELD (version 4.1) was used for this analysis. The MICROSHIELD program allows a user to input a source geometry and shield thickness, density and material type. The data and calculation methods used in MICROSHIELD are documented and the program is widely accepted.

### 2.3.2 Assumptions

The following assumptions were used for modeling the direct exposure pathway for the disposal of the Trojan Reactor vessel at the Richland Facility:

1. The reactor vessel will be disposed as a single, intact unit.
2. The reactor vessel activity is shielded by steel and concrete to reduce surface dose rates to less than 200 mrem/hour. Side cladding thicknesses were estimated to reduce the reactor surface activity to levels below this limit. These thicknesses were then included as side and end cladding of the reactor vessel in the disposal modeling.

3. A total thickness of 16.5 feet of soil will cover the reactor vessel. Sensitivity analyses were performed that examined the exposure and dose for changes in soil cover thickness in two foot increments.

4. Radionuclide activities of the reactor waste as of 5 years after shutdown will be used. The activation activity is $2.01 \times 10^6$ Curies, compared to the 357.9 Curies of surface contamination. The contribution to the dose from the surface contamination are negligible, and was not included in the analysis.

5. The radiation exposure is to the maximally exposed individual, and is based on a person who is standing on the trench cap in the center of the Facility. An hourly exposure rate and dose was calculated.

6. The trench caps will be maintained such that major soil erosion is repaired. Because of the Facility's location and the soil conditions, the soil erosion potential is assumed to be minimal.

## 2.4 DATA

### 2.4.1 Areal Geometry

The cap area is $3.64 \times 10^6$ square feet (rectangular shape). This area includes the plan view area of the trenches, the areas between the trenches, and the areas to the sides of the trenches. This area is the cap area used in HELP model studies performed for Closure Plan.

The source geometry that most closely matches the Facility conditions is that of a cylindrical source with side and end cladding, and with end shields geometry. In this situation, the waste is isolated by a horizontal shield (e.g. the cap). The reactor vessel volume is 7,951 $ft^3$ ($2.25 \times 10^8$ $cm^3$), with a void volume of 5295.3 $ft^3$ and a displaced volume of 2655.7 $ft^3$. Estimated dimensions are about 16 foot diameter and 40 feet high. The total weight of the reactor vessel was estimated to be $5.91 \times 10^8$ g, which yields an average density of 2.6 $g/cm^3$ for the source material.

### 2.4.2 Cladding and Cover Thickness

The thickness of the waste is equivalent to the reactor vessel dimensions. The previous Closure Plan[1] indicates that 8 feet of soil will be placed over the new trenches, in which the reactor vessel will be placed. The soil cap design will be added on top of the trenches and soil cover. The cap is about 8.5 feet thick. This yields a total cover thickness over the source of about 16.5 feet. Additional soil backfill (reportedly 1 to 10

[1]US Ecology, 1990, Site Stabilization and Closure Plan for Low-Level Radioactive Waste Management Facility, Richland, Washington.

ft) may be placed over the trenches prior to the cap placement, to level the Facility. This additional material was not considered in this calculation, as its depth is unknown.

The side cladding thicknesses were estimated to produce a design exposure rate of 200 mrem/hour. The disposal design identified eight inches of steel shielding. Concrete cladding was added to reduce the exposure rate. A sensitivity analysis was performed using MICROSHIELD for different concrete thicknesses; a concrete thickness of six inches yielded an exposure of 162 mrad/hour. Therefore, eight inches of steel and six inches of concrete were used as the side and end cladding for the trench disposal exposure.

### 2.4.3 Waste and Cover Density

The model requires the shield material be defined. The shield material is the earthen cover. The side and end cladding (steel and concrete) is included as part of the shielding described above. MICROSHIELD contains attenuation information for concrete, but not for soil. The shield (cover) was simulated as concrete, but the density was reduced to match the cover soil density of about 94 pounds per cubic foot (lb/ft$^3$) (1.5 g/cm$^3$). The waste was simulated as iron, but the density was reduced to match the density of the source (about 2.6 g/cm$^3$).

### 2.4.4 Source Analysis

The total activation activity in the Reactor Vessel is $2.01 \times 10^6$ curies, and consists of 11 isotopes, listed in Table 5.

### 2.4.5 Data Summary

The following summarizes the data used in the analysis of the direct exposure pathway.

1. Soil used for trench backfill and trench cap material is sand with a density of 94 lb/ft$^3$ (1.5 g/cm$^3$). This is an average dry unit weight for a silica sand[2], and matches soil densities measured at the Facility[3].

2. The trench area is $3.64 \times 10^6$ ft$^2$.

3. The waste in the trenches was modeled as a large cylinder that had a radius of 8 feet and a thickness of 40 feet.

---

[2] Lambe, T.W., and R.V. Whitman, 1969, Soil Mechanics, John Wiley & Sons, New York, p. 31, Table 3.2

[3] Bergeron, M.P., et al, 1987, Geohydrology of a Low-Level Radioactive Waste Disposal Facility, Richland, Washington, Battelle, Pacific Northwest Lab., p. 39.

4. The waste has an average density of about 160 $lb/ft^3$ (2.6 $g/cm^3$). The average waste density is calculated from reactor vessel density, volume and void ratios.

5. The source term waste activity (summed over 11 isotopes) is about 2.01 million Ci. The waste is assumed to be disposed of intact within the trench.

6. The trench cap is modeled as 16.5 feet thick above the waste.

## 2.5 MODEL SIMULATIONS

The model was run for a reference value of 16.5 ft of shield (TROJAN4.MS4 and TROJAN4.ASC). A sensitivity analysis of shield thickness was performed for values of 0 to 16 feet in 2 feet increments to examine the results of possible cap removal.

## 2.6 SUMMARY

The results of this analysis indicate that the direct exposure route is not a pathway that requires additional analysis. Reported average external gamma exposure rates for areas around the Hanford Reservation range from 0.15 mR/day to a maximum of 1.8 mR/day (0.0063 mR/hr to 0.075 mR/hr). The trench backfill and cover design reduce the amount of radiation to below background levels.

The dose rate for an individual standing over the intact 16.5-foot trench cap was estimated to be $2.3 \times 10^{-15}$ mrad/hr ($5.5 \times 10^{-14}$ mrad/day) or an exposure rate of $2.6 \times 10^{-15}$ mR/hr ($6.2 \times 10^{-14}$ mR/day). This value was compared to the minimum average gamma exposure rate for 1971-1972 at the Hanford Reservation 100 Area of 0.15 mR/day (ERDA, 1975) and determined to be insignificant.

# TABLE 1

## DATA

| Parameter | Value |
|---|---|
| Trench depth | 45 ft |
| Vadose zone thickness | 265 ft |
| Vadose zone porosity | 0.30 |
| Saturated zone thickness | 100 ft |
| Saturated zone porosity | 0.10 |
| Infiltration (surrounding area) | 0.2 in/yr |
| Infiltration (through cap) | 0.05 in/yr |
| Trench, intertrench area | $3.64 \times 10^6$ square feet |

## TABLE 2

## SOURCE TERM

| Isotope | Activity | Half-Life |
|---|---|---|
| Am-241 | 2.54E-01 | 458 |
| C-14 | 2.18E+02 | 5,730 |
| Ni-59 | 9.53E+02 | 8.00E+04 |
| Ni-63 | 1.57E+05 | 92 |
| Nb-94 | 3.29E+00 | 2.00E+04 |
| Pu-238 | 1.92E-01 | 86.4 |
| Pu-239/Pu-240 | 2.15E-01 | 24,390 / 6,600 |
| Pu-242 | 1.08E-03 | 3.79E+05 |
| Sr-90 | 2.13E+00 | 27.7 |
| Tc-99 | 7.06E-01 | 2.12E+05 |

TABLE 3

RADIONUCLIDES FOR DOSE ANALYSIS

| Isotope | Infiltration = 0.2 in/yr Hypothetical Well Concentration (pCi/L) | Infiltration = 0.05 in/yr Hypothetical Well Concentration (pCi/L) |
|---|---|---|
| C-14 | 3.90E+01 | 5.65E+00 |
| Nb-94 | 7.2 E-14 | 4.03E-16 |
| Ni-59 | 3.21E-03 | 5.73E-05 |
| Pu-239/ Pu-240 | 1.78E-07 | 2.29E-09 |
| Pu-242 | 5.49E-07 | 4.96E-08 |
| Sr-90 | 2.44E-15 | 1.19E-17 |
| Tc-99 | 2.65E-01 | 5.80E-02 |

## TABLE 4

## SUMMARY OF RESULTS

| Organ | PRICH22.PRS<br>Infiltration Rate<br>0.2 in/yr | PRICH23.PRS<br>Infiltration Rate<br>0.05 in/yr |
|---|---|---|
| Body | 0.261 | 0.038 |
| Red Marrow | 0.555 | 0.080 |
| Thyroid | 0.160 | 0.024 |

Note: C-14 is major contributing radionuclide

## TABLE 5

## DIRECT EXPOSURE ACTIVITY

| Isotope | Activity (Curies) |
|---|---|
| H-3 | 6.55E+02 |
| C-14 | 2.18E+02 |
| Sb-125 | 4.04E-01 |
| Mn-54 | 2.16E+03 |
| Eu-152 | 2.54E+01 |
| Fe-55 | 6.97E+05 |
| Co-60 | 1.15E+06 |
| Ni-59 | 9.53E+02 |
| Ni-63 | 1.57E+05 |
| Nb-94 | 3.29E+00 |
| Tc-99 | 7.06E-01 |
| **TOTAL** | **2.01E+06** |

ATTACHMENT IV

-> RW Reg. File

RECEIVED

STATE OF WASHINGTON
DEPARTMENT OF HEALTH
DIVISION OF RADIATION PROTECTION
Airdustrial Center, Bldg. 5 • P.O. Box 47827 • Olympia, Washington 98504-7827

June 10, 1996

Art Palmer, Chief
Radiological & Safety Officer
US Ecology, Inc.
120 Franklin Road
Oak Ridge, Tennessee 37830

Dear Mr. Palmer:

This is in response to your letters dated March 12 and April 17, 1996, requesting the department's review on the waste classification for the Trojan reactor vessel.

We have reviewed your submittals and have determined that the waste classification of the Trojan waste appears to be consistent with the Nuclear Regulatory Commission's January 17, 1995 Final Branch Technical position on Concentration Averaging and Encapsulation. As a result, the classification of this waste appears to conform to your state of Washington radioactive Materials license WN-I019-2, and WAC 246-249-040. Please be advised, however, it is the generator's responsibility to ensure compliance with waste classification and waste form. It is requested that if any of the data used for waste calculations change significantly, that the revised numbers be submitted to the department.

If you should have any questions, do not hesitate to contact me.

Sincerely,

Mikel J. Elsen

Mikel J. Elsen
Radiation Health Physicist

cc: WDOH - Richland, WA

American Ecology Corporation
120 Franklin Road
Oak Ridge, TN 37830

Tel: 615/483-8768
Fax: 615/483-8078

# American Ecology

April 17, 1996

Mr. Mikel J. Elsen
Radiation Health Physicist
Washington Department of Health
P.O. Box 47827
Olympia, Washington 98504-7827

Dear Mr. Elsen:

This is provided in response to you letter of April 2, 1996 transmitting your comments regarding the disposal of the Trojan Reactor Vessel. Each of your comments is reprinted below followed by our response.

Comment 1: *"When does PGE expect to find out if the NRC will issue a C of C on the reactor vessel?"*

Response: PGE met with the U.S. NRC on January 31, 1996 to discuss the proposed shipment of the reactor vessel and internals. PGE proposed several alternatives for shipment of the package. The NRC's preference was to license it as a Type B package. PGE discussed the requirements for a Type B package and the ability to meet the requirements. Overall, it appears that the NRC will license the package if properly designed and the shipment is completed in a well controlled manner. Initial conceptional design meetings will be held with the NRC in May. The final Safety Analysis Report will be submitted late 1996. It is expected the Certificate of Compliance will be issued late 1997.

Comment 2: *"Please supply the waste classification calculation. It was not submitted with your request. Will the activation analysis for the vessel with internals be analytically verified with any sampling? A sample may be able to be compared to the activation analysis to verify the waste classification results.*

Response: The waste classification calculation was inadvertently omitted from the copies distributed. A copy is attached with this submittal. US Ecology regrets the error and apologizes for any inconvenience it may have caused.

The two approaches historically used to characterize routinely generated activated metals components included; (1) direct sampling of individual components coupled with radiochemical analysis of samples, and (2) the use of activation analysis computer programs. Both approaches rely on the gross radioactivity method in the BTP and have only been employed to define Part 61 scaling factors. These scaling factors are then used in conjunction with radiation levels to quantify the base radionuclide Co-60 to which the scaling factors are applied. The direct sample approach to scaling factor determination peaked in usage about 1987, and has not been used at all since 1992. This is based on its time and cost, the difficulty with the representative sampling of routine components, and the uncertainties in the analytical results. Ni-59 was always scaled from measured Ni-63 and Nb-94 concentrations were always defined as LLDs.

The direct sampling method has never been used to determine scaling factors for reactor vessel internals. This is due to the difficulty of obtaining representative samples from internals and the very high Co-60 concentrations in components which approach Class C limits which prevents accurate radiochemical analysis of the samples. The concentrations of significant radionuclides in internals components vary as a function of the base metal's nickel and contaminant content and the integrated flux in the base metal. Empirical data, in the form of piece specific radiation surveys from Yankee Rowe internals, indicate that concentrations varied by five orders of magnitude from component to component. Additionally, the concentration variations in a single component were found to be one to two orders of magnitude. Some direct sampling analysis has been performed at PNL on nonfuel bearing components under a government contract, and the reported comparisons between sample results and activation analysis results were good.

The waste classification analysis for Trojan is based on a detailed one dimensional neutron transport and point neutron activation analysis and the material properties of the component parts. These calculations were performed using TLG Services, Inc. FISSPEC and O2FLUX computer codes and ANISN and ORIGEN computer codes obtained through the Oak Ridge National Laboratory's Radiation Shielding Information Center. Ancillary calculations were performed using TLG's ANISNOUT and O2READ computer codes.

The one dimensional neutron transport model was normalized with data obtained from a Westinghouse Electric Corporation report on a reactor vessel radiation surveillance specimen which was removed from the plant in 1990.

Based on the above, PGE does not intend to obtain samples from the Reactor Vessel or Internals. The waste characterization will be based on the activation analysis and radiation surveys. The radiation surveys will be used to quantify Co-60 content and the activation analysis based scaling factors will be applied to Co-60 quantities. This is fully consistent with the NRC BTP gross radioactivity method of characterization and was employed during the Shoreham and Yankee Rowe projects.

Comment 3: *"Is the Pu-241 that is used in your pathway analysis decayed into Am-241? Our results indicate 2.5 Ci where the proposal shows 0.25 Ci of Am-241. The difference could be in the Pu-241 initial activity (e.g., 11.7 Ci). Please show how the value for Am-241 was arrived at."*

Response: The value of 0.25 Ci of Am-241 documented in our waste classification report was based on the fractional percentages of the sample results taken from a S/G tube in 1994. The results were not decayed to November 1997 as were the other activation analysis results.

The isotope Pu-241 has a short half life (13.2 years) relative to the travel time through the vadose zone. Combined with a large distribution coefficient, the isotope will not migrate an appreciable distance from the reactor vessel, but will decay to Am-241 and U-237. Because the decay chain is almost exclusively to Am-241, this will yield a maximum activity of Am-241 of 0.571 curies at approximately 60 years. The decay product of 0.571 curies of Am-241 from Pu-241 compares to the 0.254 curies of Am-241 identified in the source term from the reactor, and used in the ground-water pathway analysis. The change from 0.254 to 0.571 curies represents and increase of about 2 times.

Am-241 has a larger distribution coefficient value and longer half life than Pu-241. The estimated leachate concentration at the water table for Am-241 was $3.69 \times 10^{-123}$ and $5.41 \times 10^{-125}$ ρCi/L for recharge rates of 0.2 in/yr

and 0.05 in/yr, respectively. Because Am-241 is adsorption-controlled, multiplying the increased activity ratio times the concentration at the water table should yield an approximation of the concentration at the water table with the new source value. These values are $7.38 \times 10^{-123}$ and $1.08 \times 10^{-122}$ ρCi/L for the two recharge rates. These concentrations are negligible.

With regard to waste class calculation, at the time of shipment, the sum of fractions is estimated to be 0.335 due to 11.7 curies of Pu-241 and 0.25 curies of Am-241. The activity shift still maintains the waste as Class C. Over time the Pu-241 activity decreases and the Am-241 increases. At 60 years the Pu-241 activity is <0.1 times the original and the Am-241 activity reaches a peak of approximately 0.571 curies. The sum of the fractions at that time would actually decrease.

Comment 4: *"In the pathway analysis it is assumed that the package's exterior dose is only 200 mr/hr. With one of the steam generators, a 430 mr/hr hot spot was found and enclosed before shipping. With this in mind, perhaps the PA should use 1 R/hr as an initial exposure before taking any shielding into account. This dose rate is the most conservative and should not impact the outcome of the analysis."*

Response: It is expected that the reactor vessel and internals will be shielded to less than 200 mR/hr to meet Department of Transportation shipping requirements. However, for purposes of evaluating this possibility, the PA results may be scaled directly to the increase in package dose rates. Specifically, the 1 R/hr case may be evaluated by extrapolation linearly and directly from the PA results for the 200 mR/hr cases. That is, multiplying the results of the 200 mR/hr cases by a factor of 5 yields the results for the associated 1 R/hr cases. A table of the base case and the sensitivity analysis cases for the 200/hr package and the and the associated 1 R/hr package are provided below.

| Soil Cover Thickness (ft) | Dose Rate In Air (mRad/hr) @ 200 mR/hr | Dose Rate In Air (mRad/hr) @ 1 R/hr |
|---|---|---|
| 0.5 | $3.416 \times 10^{1}$ | $1.705 \times 10^{2}$ |
| 2.5 | $3.628 \times 10^{-1}$ | $1.184 \times 10^{0}$ |
| 4.5 | $3.654 \times 10^{-3}$ | $1.827 \times 10^{-2}$ |
| 6.5 | $3.552 \times 10^{-5}$ | $1.776 \times 10^{-4}$ |
| 8.5 | $3.361 \times 10^{-7}$ | $1.681 \times 10^{-6}$ |
| 10.5 | $3.120 \times 10^{-9}$ | $1.560 \times 10^{-8}$ |
| 12.5 | $3.862 \times 10^{-11}$ | $1.931 \times 10^{-10}$ |
| 14.5 | $3.583 \times 10^{-13}$ | $1.792 \times 10^{-12}$ |
| 16.5 | $3.314 \times 10^{-15}$ | $1.657 \times 10^{-14}$ |

Comment 5: *"Will any shielding be welded onto the vessel? Is this weight used in the waste classification process?"*

Response: Shielding may be welded onto the reactor vessel to reduce radiation levels on the exterior of the package as necessary. However, the weight of the shielding will not be used in the waste classification process.

Comment 6: *"Since this is not a routine shipment, procedures for the handling and disposal of this waste should be developed and submitted to the department for review. Additionally, the proposed trench set-up (e.g. where the ramp(s), the final vessel placement, and backfilling should be addressed. If the proposed trench is different that what is contained in the March 6, 1991 Comprehensive Facility Utilization Plan, Document 200-DOC-001, Rev. 3, the department must approve the change."*

Response: At this time, we are requesting the Department's concurrence with the waste classification of the PGE reactor vessel. We presently anticipate that the reactor vessel will de disposed of in Trench 12. This is consistent with the proposed use of Trench 12 in the CFUP of March 6, 1991. Waste placement, ramp location and backfilling are not addressed in the CFUP. Only trench configurations (i.e., maximum dimensions and slopes) are addressed.

A number of large components have been disposed at the Richland site without special site handling and disposal procedures. These have included the Trojan steam generators and pressurizer as well as the Pathfinder reactor vessel. Since the dose rate on the reactor vessel is expected to be less than 200 mR/hr, we believe that this unit will be able to be appropriately disposed within the existing site procedure framework. As the project progresses if it becomes necessary to develop additional site procedures, these will be forwarded to the Department for review and approval.

Comment 7: *"It is the department's opinion that the basement construction scenario be examined in the pathway analysis."*

Response: The basement scenario has, in effect, been run as a part of the sensitivity analysis performed for the Direct Exposure Dose Analysis. This sensitivity analysis examined the effect of varying the trench cap thickness from 0.5 to 16.5 feet in two-foot thick increments. The results of these analysis are presented in response to Comment 4. Since radium is not present in the source term, the reactor vessel disposal does not need to be evaluated for radon gas.

Comment 8: *"What is the process for verifying that the void spaces are filled with grout?"*

Response: The grouting process will be developed to provide the best engineering assurance that the package is being completely filled. This will include filling the vessel until a positive vent of Low Density Cellular Concrete is obtained from each vent connection. Any small voids that form during the filling process, will fill by gravity flow and by the thermal expansion driving force as the grout heats up during cure. Any voids created at the top/vents due to the above will be filled. A final visual inspection will be completed from each top vent to ensure the package is filled. In addition, a time-weighted average density and total weight of LDCC constituents will be used to estimate the volume of injected grout and verify it is greater than or equal to the available internal free volume. This process will ensure the void spaces are filled with grout.

Comment 9: *"You have indicated that the vessel will have 357.9 curies of surface contamination. How will the contamination on the vessel be handled so that it will not be spread?"*

Response: The 357.9 curies of surface contamination represents contamination contained within the reactor vessel on reactor vessel walls and internals. The outside of the reactor vessel will be below Department of Transportation shipping limits for surface contamination. We expect the outside of the package to be essentially free from removable radioactive contamination.

We appreciate the Department's review of our request and trust these replies appropriately respond to your comments. PGE is presently incurring unrecoverable costs associated with this project; consequently, we appreciate your continued timely consideration and evaluation of this proposal.

Sincerely,

Arthur J. Palmer

Arthur J. Palmer, CHP
Chief Radiological Control
and Safety Officer

Att.

PORTLAND GENERAL ELECTRIC CALCULATION SHEET

Calculation No. N/A Revision 0 Sheet 1 Cont'd on Sht

Preparer Michael Murdock Date 2/13/96

Verifier [signature] Date 2/15/96

# REACTOR VESSEL WASTE CLASSIFICATION ANALYSIS

## INTRODUCTION

The intent of this analysis is to determine the waste classification of the Trojan Reactor Vessel with its sub-components (internals) as one complete package. The approach will be to use the radioactivity, as analyzed for and, identified in the Trojan Decommissioning Plan along with the most current revision of the Radiation Protection Manual Procedure, RPMP 4, "Determination of Radioactive Material Shipping And Waste Classifications". The anslysis will be performed combining both the surface contamination activity and the neutron activated activity to calculate over the envelope volume of the package.

## REVIEW CRITERIA

The analysis will be reviewed and checked for accuracy and regulatory conformance but will not be documented or follow the same format as an approved PGE calculation.

## RESULTS

→ The vessel results in being (Class C) waste.

→ The package contains 3.56 gms of plutonium.

→ Table 1 results - .335

→ Table 2 results - .299

# PORTLAND GENERAL ELECTRIC CALCULATION SHEET

Calculation No. N/A Revision 0 Sheet 2 Cont'd on Sht

Preparer Michael Murdock Date 2/13/96

Verifier SR Huey Date 2/15/96

## COMPONENT ASSUMPTIONS

→ The reactor vessel and core internals is considered to be one component containing neutron activated metals incorporating radioactivity in its design thus allowing concentration averaging over the displaced volume of the material.

→ The "envelope volume" is considered to be the reactor vessel (including the head) and the reactor core internals minus the void space in accordance with section 3.3 of the BTP. (Mass of metal only)

→ The majority of the activation is located in the region between the upper and lower core plates including the vessel cladding and walls.

→ The surface contamination (although <1% of the total activity) is considered to be distributed over all the reactor vessel internal surfaces.

→ The activity contribution of the Incore Flux Thimbles, currently in place in the vessel, has not been calculated as of yet or been included in the total source term but is assumed to be less than .05% of the total activity.

## COMPONENT DIMENSIONS

→ Burial/Envelope Volume = 7951 $ft^3$
(Does not include any additional steel shielding or penetration closures)

→ Void Volume = 5295.3 $ft^3$
(With internals)

→ Displaced Volume = 2655.7 $ft^3$ or (75,201,049 cc)
(Envelope volume minus major void volumes)

→ Material Weight = 591,080,249 gms
[Displaced Volume x 7.86 gms/cc]

PORTLAND GENERAL ELECTRIC CALCULATION SHEET

Calculation No. N/A Revision 0 Sheet 1 Cont'd on Sht

Preparer Michael Murdock Date 2/6/96

Verifier SkHuey Date 2/15/96

REACTOR VESSEL ACTIVITY
(5 years Post Shutdown, 11/1/97)

| NUCLIDE | SURFACE CONTAMINATION (Curies) | ACTIVATION (Curies) |
|---|---|---|
| H-3 | 1.87E-01 | 6.55E+02 |
| C-14 | 2.65E-01 | 2.18E+02 |
| Sb-125 | 3.86E+00 | 4.04E-01 |
| Ce-144 | 1.08E-01 | |
| Mn-54 | 1.98E-01 | 2.16E+03 |
| Eu-152 | | 2.54E+01 |
| Fe-55 | 6.39E+01 | 6.97E+05 |
| Co-60 | 2.29E+02 | 1.15E+06 |
| Ni-59 | | 9.53E+02 |
| Ni-63 | 4.60E+01 | 1.57E+05 |
| Nb-94 | | 3.29E+00 |
| Sr-90 | 2.13E+00 | |
| Tc-99 | | 7.06E-01 |
| Pu-238 | 1.92E-01 | |
| Pu-239/240 | 2.15E-01 | |
| Pu-241 | 1.17E+01 | |
| Cm-242 | 3.62E-05 | |
| Cm-243 | 8.29E-02 | |
| Cm-244 | 7.87E-02 | |
| Am-241 | 2.54E-01 | |
| Pu-242 | 1.08E-03 | |
| TOTAL | 357.9 Curies | 2.01E+06 Curies |

Shipment Number N/A
Package Number RV & Int

12. Determination of Concentration

Material Volume 7.52E+07 cc (D)

Material Weight 5.91E+08 gm (E)

| Isotope | A Ship Date Activity (mCi) | B Conc Isotope (uCi/cc) (Ax1E3)/D <5 yr T 1/2 | C Conc Isotope (uCi/cc) (Ax1E3)/D >5 yr T 1/2 | F Conc. Isotope (nCi/gm) (Ax1E6)/E | G Conc Factor (gm/mCi) | H Gram of Isotope (AxG) |
|---|---|---|---|---|---|---|
| H-3 | 1.87E+02 | | 2.49E-03 | | | |
| H-3 (A) | 6.55E+05 | | 8.71E+00 | | | |
| C-14 | 2.65E+02 | | 3.52E-03 | | | |
| C-14 (A) | 2.18E+05 | | 2.90E+00 | | | |
| Mn-54 | 1.98E+02 | 2.63E-03 | | | | |
| Mn-54 (A) | 2.16E+06 | 2.87E+01 | | | | |
| Fe-55 | 6.39E+04 | 8.50E-01 | | | | |
| Fe-55 (A) | 6.97E+08 | 9.27E+03 | | | | |
| Co-60 | 2.29E+05 | | 3.05E+00 | | | |
| Co-60 (A) | 1.15E+09 | | 1.53E+04 | | | |
| Nb-94 (A) | 3.29E+03 | | 4.37E-02 | | | |
| Ni-59 (A) | 9.53E+05 | | 1.27E+01 | | | |
| Sb-125 | 3.86E+03 | 5.13E-02 | | | | |
| Sb-125 (A) | 4.04E+02 | 5.37E-03 | | | | |
| Eu152 (A) | 2.54E+04 | | 3.38E-01 | | | |
| Ce-144 | 1.08E+02 | 1.44E-03 | | | | |
| Fe-55 | 6.39E+04 | 8.50E-01 | | | | |
| Ni-63 | 4.60E+04 | | 6.12E-01 | | | |
| Ni-63 (A) | 1.57E+08 | | 2.09E+03 | | | |
| Sr-90 | 2.13E+03 | | 2.83E-02 | | | |
| Tc-99 (A) | 7.06E+02 | | 9.39E-03 | | | |
| I-129 | | | | | | |
| TRU | 4.16E+02 | | 5.53E-03 | 7.03E-01 | | |
| Pu-238 | 1.92E+02 | | 2.55E-03 | 3.25E-01 | 5.8E-05 | 1.11E-02 |
| Pu-239/40 | 2.15E+02 | | 2.86E-03 | 3.64E-01 | 1.60E-02 | 3.44E+00 |
| Pu-241 | 1.17E+04 | | 1.56E-01 | 1.98E+01 | 9.60E-06 | 1.12E-01 |
| Cm-242 | 3.62E-02 | 4.81E-07 | | 6.12E-05 | | |
| Pu-242 | 1.08E+00 | | 1.44E-05 | 1.83E-03 | | |
| | | 9.30E+03 Sum B (uCi/cc) | 1.74E+04 Sum C (uCi/cc) | | | 3.56 Sum H (gms. of Pu) |



I. *[signature]* Verify that this is a true computer generated copy of RPMP 4, Rev 17.

Shipment Number N/A
Package Number RV & Int

13. Evaluation of results for Burial Site Limitations and possible DOT/NRC Form 741 Shipment.

a. Determine total grams of SNM:

| 3.56 | + | 0 | + | 0 | = | 3.56 |
|---|---|---|---|---|---|---|
| Grams of Pu | | Grams of U-235 | | Grams of U-233 | | Grams of SNM |

If SNM > or = 1 gram, notify Radwaste Supervisor.

b. Burial Site TRU is TRU with T-1/2 >5 yrs except Pu-238, Pu-239/240, Pu-241 and Cm-242 from Column F.

Burial Site TRU: 7.03E-01 (nCi/gm) Limit <10 nCi/gm

c. Waste Class TRU is TRU with T-1/2 >5 yrs and PU-238 and Pu-239/240 from Column F.

Waste Class TRU:

| 7.03E-01 | + | 3.25E-01 | + | 3.64E-01 | = | 1.39E+00 |
|---|---|---|---|---|---|---|
| (TRU) | | (Pu-238) | | (Pu-239/240) | | (nCi/gm) |

14. Sum all isotopes with half life <5 yrs:

9.30E+03 (uCi/cc) Limit <7 UCi/cc

15. Sum All isotopes with half life >5 yrs:

1.74E+04 (uCi/cc) Limit <1 uCi/cc

Note: If any of the above limits are exceeded, notify the Radwaste Supervisor.

Prepared by Michael Murdock

Checked by [signature]



I. Michael Murdock Verify that this is a true computer generated copy of RPMP 4, Rev 17.

FORM RP 78

WORK SHEET FOR DETERMINATION OF WASTE CLASSIFICATION

| | | Class A | | Class B | | Class C | |
|---|---|---|---|---|---|---|---|
| | A | B | | C | | D | |
| Table 1 Isotopes | Isotopic Conc. (uCi/cc) | Limit (uCi/cc) | Quotient (A/B) | Limit (uCi/cc) | Quotient (A/C) | Limit (uCi/cc) | Quotient (A/D) |
| C-14 | 2.90E+00 | 8.00E+00 | 3.63E-01 | (b) | | 8.00E+01 | 3.63E-02 |
| Tc-99 | 9.39E-03 | 3.00E-01 | 3.13E-02 | (b) | | 3.00E+00 | 3.13E-03 |
| Ni 59 | 1.27E+01 | 2.20E+01 | 5.77E-01 | (b) | | 2.20E+02 | 5.77E-02 |
| Nb-94 | 4.37E-02 | 2.00E-02 | 2.19E+00 | (b) | | 2.00E-01 | 2.19E-01 |
| TRU (a,c) | 1.39E+00 | 1.00E+01 | 1.39E-01 | | | 1.00E+02 | 1.39E-02 |
| Pu-241 (a) | 1.98E+01 | 3.50E+02 | 5.66E-02 | (b) | | 3.50E+03 | 5.66E-03 |
| Cm-242 (a) | 6.12E-05 | 2.00E+03 | 3.06E-08 | (b) | | 2.00E+04 | 3.06E-09 |
| | Sum of Table 1 Quotients | | 3.35E+00 | | | | 3.35E-01 |

NOTE: TRU with T-1/2 >5 yrs = all TRU except Pu-238, Pu-239/240, Pu-241 and Cm-242.

| | | Class A | | Class B | | Class C | |
|---|---|---|---|---|---|---|---|
| | A | B | | C | | D | |
| Table 2 Isotopes | Isotopic Conc. (uCi/cc) | Limit (uCi/cc) | Quotient (A/B) | Limit (uCi/cc) | Quotient (A/C) | Limit (uCi/cc) | Quotient (A/D) |
| H-3 | 8.71E+00 | 4.00E+01 | 2.18E-01 | (d) | N/A | (d) | N/A |
| Co-60 | 1.53E+04 | 7.00E+02 | 2.19E+01 | (d) | N/A | (d) | N/A |
| Ni-63 | 2.09E+03 | 3.50E+01 | 5.97E+01 | 7.00E+02 | | 7.00E+03 | 2.99E-01 |
| Sr-90 | 2.83E-02 | 4.00E-02 | 7.08E-01 | 1.50E+02 | | 7.00E+03 | 4.04E-06 |
| Cs-137 | | 1.00E+00 | | 4.40E+01 | | 4.60E+03 | 0.00E+00 |
| Nuclides w/ (T-1/2 .5 yr) | | 7.00E+02 | 0.00E+00 | (d) | N/A | (d) | N/A |
| | Sum of Table 2 Quotients | | 8.25E+01 | | | | 2.99E-01 |

a - Units are in nCi/gm

b - If Class A limit is exceeded, the waste is Class C or greater.

c - TRU = TRU with T-1/2 >5 yrs + Pu-238 + Pu-239/240.

d - No Limit.

NOTE: It is not necessary to list on the manifest any nuclide whose Class A quotient is less than 0.01 except C-14, Tc-99, I-129 and H-3.

Prepared by Michael Murdock

Checked by [signature]



I Michael Murdock Verify that this is a true computer generated copy of RPMP 4, Rev 17