

SECTION 11TABLE OF CONTENTS

<u>Section</u>	<u>Title</u>	<u>Page</u>
11.0	RADIOACTIVE WASTE MANAGEMENT	11.1-1
11.1	SOURCE TERMS	11.1-1
11.1.1	<u>Activity in the Core</u>	11.1-1
11.1.2	<u>Activity in Reactor Coolant</u>	11.1-3
11.1.2.1	Fission Product Activity	11.1-3
11.1.2.2	Tritium Activity	11.1-4
11.1.2.2.1	Ternary Fission	11.1-5
11.1.2.2.2	Boron Activation	11.1-5
11.1.2.2.3	Lithium Activation	11.1-6
11.1.2.3	Corrosion Product Activity	11.1-7
11.1.2.4	Nitrogen-16 Activity	11.1-7
11.1.2.5	Argon-41 Activity	11.1-7
11.1.3	<u>Estimates of Gaseous Activity Released from Liquid or Steam Leakage</u>	11.1-8
11.1.3.1	Leakage Rates	11.1-8
11.1.3.1.1	Containment Vessel and Auxiliary Building	11.1-8
11.1.3.1.2	Turbine Building	11.1-8
11.1.3.2	Activity Levels in Leakage	11.1-8
11.1.3.3	Evolution of Gases from Leakage	11.1-10
11.1.3.4	Transport and Environmental Releases of Evolved Gases	11.1-11
11.1.3.5	Measurement and Control of Leakage	11.1-11
11.2	LIQUID WASTE SYSTEMS	11.2-1
11.2.1	<u>Design Objectives</u>	11.2-1
11.2.2	<u>System Description</u>	11.2-1
11.2.2.1	Process Arrangement	11.2-1
11.2.2.1.1	Clean Liquid Radwaste System	11.2-1
11.2.2.1.2	Miscellaneous Liquid Radwaste System	11.2-1
11.2.2.2	Waste Sources	11.2-1
11.2.2.2.1	Clean Liquid Radwaste System	11.2-1
11.2.2.2.2	Miscellaneous Liquid Radwaste System	11.2-3
11.2.2.2.2.1	Non-Detergent Wastes	11.2-3
11.2.2.2.2.2	Detergent Waste	11.2-7
11.2.2.2.2.3	Tritium	11.2-8
11.2.2.3	Equipment Decontamination Factors	11.2-8
11.2.2.3.1	Evaporator	11.2-8
11.2.2.3.2	Demineralizers (Radwaste System)	11.2-11
11.2.2.3.3	Demineralizers (Secondary System)	11.2-14
11.2.2.3.4	Filters	11.2-14
11.2.2.3.5	Degasifier	11.2-14
11.2.2.3.6	Waste Demineralizer System	11.2-15

TABLE OF CONTENTS (CONTINUED)

<u>Section</u>	<u>Title</u>	<u>Page</u>
11.2.3	<u>Operating Procedures</u>	11.2-15
11.2.3.1	Clean Liquid Radioactive Waste System	11.2-15
11.2.3.2	Miscellaneous Liquid Radioactive Waste System	11.2-16
11.2.4	<u>Performance Tests</u>	11.2-18
11.2.5	<u>Estimated Releases</u>	11.2-18
11.2.5.1	Clean Liquid Radioactive Waste System (Excluding Tritium)	11.2-18
11.2.5.1.1	Maximum Isotopic Specific Activities in Releases	11.2-18
11.2.5.1.2	Average Isotopic Specific Activities in Releases	11.2-19
11.2.5.1.3	Total Annual Activity Released	11.2-19
11.2.5.2	Miscellaneous Liquid Radioactive Waste System (Excluding Tritium)	11.2-20
11.2.5.2.1	Maximum Specific Activities in Releases	11.2-20
11.2.5.2.2	Average Specific Activities in Release Due to Non-Detergent Wastes	11.2-21
11.2.5.2.3	Average Specific Activities in Releases Due to Detergent Wastes	11.2-21
11.2.5.2.4	Average Specific Activities in Releases Due to Secondary System Leakage	11.2-21
11.2.5.2.5	Total Annual Activity Released Due to Non-Detergent Waste	11.2-22
11.2.5.2.6	Total Annual Activity Released Due to Detergent Waste	11.2-22
11.2.5.2.7	Total Annual Activity Released Due to Secondary System Leakage	11.2-23
11.2.5.3	Total Annual Releases of Tritium From All Sources	11.2-23
11.2.5.4	Summary of All Activity Released Each Year in Liquid Effluent	11.2-23
11.2.6	<u>Release Points</u>	11.2-24
11.2.7	<u>Dilution Factors</u>	11.2-24
11.2.7.1	Prior to Discharge	11.2-24
11.2.7.2	Further Dilution in Lake Erie	11.2-24
11.2.8	<u>Estimated Doses From Liquid Effluents</u>	11.2-26
11.3	GASEOUS WASTE SYSTEM	11.3-1
11.3.1	<u>Design Objectives</u>	11.3-1
11.3.2	<u>System Description</u>	11.3-1
11.3.2.1	Process Arrangement	11.3-1
11.3.2.2	Waste Sources	11.3-1
11.3.2.2.1	Un aerated Hydrogen-Containing Gas	11.3-1
11.3.2.2.2	Aerated Gases	11.3-3
11.3.2.2.3	Cover Gases	11.3-5
11.3.2.2.4	Building Ventilation and Purge Gases	11.3-6
11.3.2.2.5	Steam Jet Air Ejector Vent	11.3-15
11.3.2.2.6	Gaseous Decay Products	11.3-16
11.3.2.2.7	Airborne Particulate Decay Products	11.3-18
11.3.2.3	Equipment Decontamination Factors	11.3-18

TABLE OF CONTENTS (CONTINUED)

<u>Section</u>	<u>Title</u>	<u>Page</u>
11.3.3	<u>Operating Procedures</u>	11.3-20
11.3.4	<u>Performance Tests</u>	11.3-21
11.3.5	Estimated Releases	11.3-21
11.3.5.1	Annual Releases	11.3-21
11.3.5.2	Site Boundary Concentrations	11.3-23
11.3.5.2.1	Maximum Concentration	11.3-23
11.3.5.2.2	Average Concentration	11.3-24
11.3.6	<u>Release Points</u>	11.3-25
11.3.7	<u>Dilution Factors</u>	11.3-25
11.3.8	<u>Estimated Doses from Gaseous Effluents</u>	11.3-26
11.4	PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS	11.4-1
11.4.1	<u>Design Objectives</u>	11.4-1
11.4.2	<u>Continuous Monitoring</u>	11.4-1
11.4.2.1	General Design	11.4-1
11.4.2.2	Specific Design	11.4-3
11.4.2.2.1	Reactor Coolant Letdown	11.4-3
11.4.2.2.2	Clean and Miscellaneous Radioactive Waste Systems	11.4-3
11.4.2.2.3	Component Cooling Water System	11.4-3
11.4.2.2.4	Station Vent Monitoring System	11.4-3
11.4.2.2.5	Containment Vessel Monitor	11.4-4
11.4.2.2.6	Waste Gas Monitor	11.4-4
11.4.2.2.7	Vacuum System Discharge	11.4-4
11.4.2.2.8	Main Steam Lines	11.4-4
11.4.2.2.9	Service Water System	11.4-4
11.4.2.2.10	Intake Forebay	11.4-4
11.4.2.2.11	Station Effluent	11.4-5
11.4.2.2.12	Unit Storm Sewer Outlet	11.4-5
11.4.2.2.13	Deleted	
11.4.3	<u>Sampling</u>	11.4-5
11.4.3.1	Description	11.4-5
11.4.3.2	Operation	11.4-5
11.4.3.2.1	Miscellaneous Waste Monitor Tank, Detergent Waste Drain Tank and Clean Waste Monitor Tanks	11.4-5
11.4.3.2.2	Waste Gas Decay Tanks	11.4-6
11.4.3.2.3	Station Vent	11.4-6
11.4.3.2.4	Cooling Tower Blowdown	11.4-6
11.4.4	Calibration and Maintenance	11.4-6

TABLE OF CONTENTS (CONTINUED)

<u>Section</u>	<u>Title</u>	<u>Page</u>
11.5	SOLID WASTE SYSTEM	11.5-1
11.5.1	<u>Design Objectives</u>	11.5-1
11.5.2	<u>System Inputs</u>	11.5-1
11.5.3	<u>System Description</u>	11.5-5
11.5.4	Packaging	11.5-6
11.5.4.1	Shielded Cask with Disposable Liner	11.5-6
11.5.4.2	Shielded Container	11.5-7
11.5.4.3	Unshielded Storage Containers	11.5-7
11.5.5	<u>Storage Facilities</u>	11.5-7
11.5.6	Deleted	
11.5.7	<u>Shipment</u>	11.5-7
11.6	OFFSITE RADIOLOGICAL MONITORING PROGRAM	11.6-1
11.6.1	<u>Background Radiation</u>	11.6-1
11.6.2	<u>Critical Pathways</u>	11.6-1
11.6.3	<u>Sampling Media, Locations and Frequency</u>	11.6-1
11.6.4	<u>Analytical Sensitivity</u>	11.6-2
11.6.5	Data Analysis and Presentation	11.6-2
11.6.6	Program Statistical Sensitivity	11.6-3
11.7	REFERENCES	11.7-1
APPENDIX 11A -	Offsite Radiation Exposure Due to Operation of the Davis-Besse Nuclear Power Station	11A-1

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
11.1-1	Escape Rate Coefficients	11.1-12
11.1-2	Fission Product Activity in the Fuel	11.1-13
11.1-3	Fission Product Activity in the Fuel Rod Gaps	11.1-14
11.1-4	Calculated Reactor Coolant Activity for 0.1% Failed Fuel	11.1-15
11.1-5	Calculated Reactor Coolant Activity for 1% Failed Fuel	11.1-16
11.1-6	Tritium Production	11.1-17
11.1-7	Corrosion Product Activity in Reactor Coolant	11.1-18
11.1-8	Nitrogen-16 Activity in Reactor Coolant	11.1-19
11.1-9	Total System Leakage Rates	11.1-20
11.1-10	Radionuclide Specific Activities in Leakage	11.1-21
11.1-11	Total Evolution Rate of Radioactive Gases From System Leakage	11.1-23
11.2-1	Clean Liquid Radioactive Waste System Equipment List	11.2-28
11.2-2	Miscellaneous Liquid Radioactive Waste System Equipment List	11.2-33
11.2-3	Clean Liquid Waste Quantities	11.2-37
11.2-4	Average and Maximum Radionuclide Concentrations	11.2-38
11.2-5	Miscellaneous Liquid Radioactive Wastes	11.2-40
11.2-6	Maximum Radionuclide Concentrations in Non-Detergent Miscellaneous Liquid Waste	11.2-41
11.2-7	Average Radionuclide Concentrations in Non-Detergent Miscellaneous Liquid Waste	11.2-42

LIST OF TABLES (CONTINUED)

<u>Table</u>	<u>Title</u>	<u>Page</u>
11.2-8	Maximum Crud Activity Levels in Sluice Waste	11.2-43
11.2-9	Maximum and Average Radionuclide Concentrations in Liquid Secondary System Leakage	11.2-44
11.2-10	Decontamination Factors For Demineralizers	11.2-45
11.2-11	Clean Liquid Radwaste Recycle Flow Paths	11.2-46
11.2-12	Maximum Specific Activities at Significant Points in the Processing Cycle for Clean Liquid Radioactive Waste	11.2-47
11.2-13	Average Specific Activities at Significant Points in the Processing Cycle for Clean Liquid Radioactive Waste	11.2-49
11.2-14	Total Annual Activity Released to Lake From Clean Liquid Radioactive Waste System	11.2-51
11.2-15	Maximum Specific Activities at Significant Points in the Processing Cycle for Non-Detergent Miscellaneous Liquid Radioactive Waste	11.2-53
11.2-16	Average Specific Activities at Significant Points in the Processing Cycle for Non-Detergent Miscellaneous Liquid Radioactive Waste	11.2-54
11.2-17	Average Specific Activities at Significant Points in the Processing Cycle for Non-Detergent Miscellaneous Liquid Radioactive Waste	11.2-55
11.2-18	Average Specific Activities and Total Activities in Secondary System Liquid Waste Released to Lake	11.2-56
11.2-19	Total Annual Activity Released to Lake Due to Non-Detergent Miscellaneous Liquid Waste	11.2-57
11.2-20	Summary of Annual Releases of Activity in Liquid Effluents	11.2-58
11.2-21	Summary of Maximum Individual Doses Due to Liquid Releases	11.2-60
11.2-22	Summary of Whole Body Population Dose Due to Liquid Releases	11.2-61

LIST OF TABLES (CONTINUED)

<u>Table</u>	<u>Title</u>	<u>Page</u>
11.3-1	Gaseous Radioactive Waste System Equipment List	11.3-28
11.3-2	Maximum Specific Activity From Degasifier	11.3-30
11.3-3	Decay Scheme	11.3-31
11.3-4	Activity Releases In Aerated Gases	11.3-33
11.3-5	Activity Releases in Cover Gases	11.3-34
11.3-6	Activity Concentrations and Releases Due to Containment Building Purge	11.3-35
11.3-7	Activity Concentrations and Releases Due to Containment Building Ventilation During Refueling	11.3-36
11.3-8	Activity Concentrations and Releases Due to Auxiliary Building Ventilation (Excluding Refueling Areas)	11.3-37
11.3-9	Activity Concentrations and Releases Due to Auxiliary Building Ventilation in Refueling Areas	11.3-38
11.3-10	Activity Concentrations and Releases Due to Turbine Building Ventilation	11.3-39
11.3-11	Activity Releases From Steam Jet Air Ejector Vent	11.3-41
11.3-12	Annual Environmental Releases of Gaseous Xenon Daughter Activity Due to Several Sources Containing Significant Parent Isotope Inventories	11.3-42
11.3-13	Activities in Decay Tank	11.3-43
11.3-14	Estimated Annual Releases From Gaseous Radwaste System	11.3-44
11.3-15	Annual Average Radionuclide Concentrations at Site Boundary Due to Various Waste Gas Sources	11.3-48
11.3-16	Maximum Radionuclide Concentrations at Site Boundary Due to Various Waste Gas Sources	11.3-52
11.3-17	X/Q Values for Short-Term Releases Resulting from Accident Situation	11.3-56

LIST OF TABLES (CONTINUED)

<u>Table</u>	<u>Title</u>	<u>Page</u>
11.3-18	X/Q Values for Long-Term Releases for 24 hrs, 30 Days, and One Year	11.3-57
11.3-19	Summary of Maximum Individual Doses Due to Gaseous Releases	11.3-58
11.3-20	External Population Exposure Due to Gaseous Releases	11.3-59
11.4-1	Liquid, Gas and Airborne Radiation Monitors	11.4-7
11.4-2	Specific Activities at Sample Points in the Liquid and Gaseous Systems	11.4-18
11.4-3	Parameters to be Monitored and Sample Frequency	11.4-20
11.5-1	Estimated Total Activity of Solid Waste	11.5-9
11.5-2	Quantities of Solid Wastes Per Year	11.5-10
11.6-1	Background Radioactivity and Radiation	11.6-4
11.6-2	Radiological Environmental Monitoring Program	11.6-5
11.6-3	Environmental Monitoring Program Analytical Program	11.6-8

LIST OF FIGURES

<u>Figure</u>	<u>Title</u>
11.2-1	DELETED
11.2-1a	DELETED
11.2-2	Clean Liquid Radwaste System
11.2-2a	DELETED
11.2-3	Miscellaneous Liquid Radwaste
11.2-3a	DELETED
11.2-4	Clean Liquid Radioactive Waste System Flow Schematic – Normal Processing Cycle
11.2-5	Miscellaneous Liquid Radioactive Waste System Flow Schematic - Normal Processing Cycle
11.2-6	Release Points of Radioactivity
11.2-7	Ohio Fishing District No. 1
11.3-1	Gaseous Radwaste System
11.3-1a	DELETED
11.3-2	Gaseous Radioactive Waste System Flow Schematic
11.6-1	DELETED
11.6-2	DELETED

SECTION 11

11.0 RADIOACTIVE WASTE MANAGEMENT

The following information pertaining to radwaste source terms, releases and doses was written as part of the initial safety analysis and is now considered to be historical design information. At a number of places in the section, reference is made to an equilibrium refueling cycle of 277 effective full power days. Beginning with Cycle 5, the refueling cycles have been extended. This results in an increase in the quantities of some isotopes and reduction of others. Also, beginning with Cycle 13, the reload batch's cladding was changed from zircaloy to M5. These changes do not significantly affect plant releases or pose any difficulty in operating within current regulatory requirements.

All operations associated with radioactive materials are based on criteria and procedures ensuring that the operations can be performed in accordance with 10CFR20 and the "as low as reasonably achievable" standard set forth in 10CFR50. These criteria ensure protection of station personnel and the general public against excessive exposure due to the radiation from radioactive liquids, gases, and solids. This chapter provides information about radioactive source calculations and the systems which reduce the amount of radioactive material released from the station or which mitigate the consequences of such releases to ensure the radioactivity levels are as low as reasonably achievable.

11.1 SOURCE TERMS

All radioactive waste originates from fission products produced in the fuel and activation products produced in the reactor coolant system. Although essentially all fission product activity produced by the fuel is contained by the fuel cladding, minute quantities will enter the reactor coolant. These fission products, together with the activation products produced in the reactor coolant system, are transported by the reactor coolant through much of the station and are the source of radioactivity contained by other systems. This section describes the methods used to calculate radioactivity levels in the core, reactor coolant system, secondary coolant system, and auxiliary systems and discusses the leakage paths from these systems.

11.1.1 Activity in the Core

The fission products in the fuel and the fuel rod gap are calculated by a digital computer code that solves the rate equations for fission product buildup in the fuel and their transport from the fuel to the fuel rod gap. The code considers 178 radionuclides in 70 decay chains, with a maximum chain length of 5 radionuclides. The code is capable of computing activity for 100 selected times. For each time, t , that calculations are performed, the core power, thermal flux, and the fraction of power produced from any two fissile materials can be changed.

The general rate equation for the inventory of a radioactive nuclide, N , in the fuel is

$$\frac{dN_f}{dt} = RY + F\lambda'N_f' - \sigma\phi N_f - \lambda N_f - \alpha N_f$$

where N_f = inventory of a radioactive nuclide in the fuel, atoms

R = fission rate, fissions/sec

Y = independent fission yield of N_f

$\lambda'N_f'$ = activity of precursor in fuel, disintegrations/sec

F = fraction of precursor which decays to N_f

$\sigma\phi$ = neutron capture rate in N_f , sec^{-1}

λ = decay constant of N_f , sec^{-1}

α = escape rate coefficient of N_f , sec^{-1}

The escape rate coefficient, α , represents the fraction of activity in the fuel that is released, per unit time, from the fuel matrix. Values of these coefficients have been derived from experimental data for most elements and are reported in the literature (refs. 1-5). The experiments from which these data were obtained utilized purposely defected fuel elements in pressurized water loops. Those tests have been performed for a variety of fuel conditions. In selecting the coefficient values to be used in the activity calculations, a comparison of the experimental conditions and the design reactor operating conditions was made. Parameters considered in the comparison included: operating pressure and temperature of the fuel, enrichment of the fuel, fuel temperature, thermal output of the fuel (in kW/ft), fuel burnup, total length of the fuel, clad material, and clad thickness. The values of the escape rate coefficients for elements of lesser importance were estimated based on the chemical similarity with elements for which the escape rate coefficients were experimentally determined.

The escape rate coefficients used in this chapter are given in Table 11.1-1.

The general rate equation for the inventory of a radioactive nuclide, N, in the fuel rod gap is:

$$\frac{dN_G}{dt} = \alpha N_f + F\lambda'N_G' - \lambda N_G$$

where N_G = inventory of a radioactive nuclide in the fuel rod gap, atoms

α = escape rate coefficient, sec^{-1}

N_f = inventory of same radioactive nuclide in the fuel, atoms

F = fraction of precursor which decays to N_G

$\lambda'N_G'$ = activity of precursor in fuel rod gap, disintegrations/sec

λ = decay constant of N_G , sec^{-1}

The fission product inventory in the fuel and the fuel rod gap was calculated by the code described above using the following assumptions.

- a. Core full-power operation at 2772 MWt for all 277 (EFP) day equilibrium cycles. (One-third of the fuel has been irradiated for three cycles, one-third irradiated for two cycles, and the remaining one-third for one cycle.)

- b. Operation with no defective fuel assemblies so the maximum core fission product inventory is calculated.

The resulting fuel and fuel rod gap fission product inventories are given in Tables 11.1-2 and 11.1-3.

11.1.2 Activity in Reactor Coolant

11.1.2.1 Fission Product Activity

The fission product activity in the reactor coolant is calculated by a digital computer code that solves the rate equations for fission product buildup in the fuel, leakage from the fuel to the reactor coolant, and removal from the reactor coolant. The code considers 178 radionuclides in 70 decay chains, with a maximum chain length of 5 radionuclides. The activity can be calculated for up to 100 times and for each time, the core power, thermal flux, fraction of power produced from any two fissile materials, fraction of defective fuel rods, and purification system parameters can be changed. The calculation assumes that the activity that leaks from the fuel in defective fuel rods immediately reaches the reactor coolant.

The general rate equation for the inventory of a radioactive nuclide, N , in the reactor coolant is:

$$\frac{dN_C}{dt} = \alpha N_f^* + F \lambda' N_C' - \lambda N_C - \beta N_C - \gamma N_C$$

where N_C = inventory of a radioactive nuclide in the reactor coolant, atoms

α = escape rate coefficient, sec^{-1}

N_f^* = inventory of same radioactive nuclide in the fuel of defective fuel rods, atoms ($N_f^* = N_f \times \text{fraction defective fuel rods}$)

N_f = inventory of same radioactive nuclide in the fuel, atoms

F = fraction of precursor which decays to N_C

$\lambda' N_C'$ = Activity of precursor in reactor coolant, disintegrations/sec

λ = decay constant of N_C , sec^{-1}

β = removal rate coefficient of N_C by purification, sec^{-1}

γ = removal rate coefficient of N_C by plate-out, sec^{-1}

The general rate equation for the inventory of a radioactive nuclide, N , in the coolant purification system due to removal of N from the reactor coolant is:

$$\frac{dN_P}{dt} = \beta N_C + F \lambda' N_P' - \lambda N_P - \eta N_P$$

where N_P = inventory of a radioactive nuclide in the purification system, atoms

β = purification system removal rate coefficient of N_C , sec^{-1}

N_C = inventory of same radioactive nuclide in the reactor coolant, atoms

F = fraction of precursor which decays to N_P

$\lambda'N_P$ = activity of precursor in the purification system disintegrations/s

λ = decay constant of N_P , sec^{-1}

η = removal rate coefficient of N_P from the purification system, sec^{-1}

The fission product activity in the reactor coolant was calculated by the code described above using the following assumptions:

- a. Core full-power operation at 2772 MWt for all 277 (EFP) day equilibrium cycles. (One-third of an equilibrium cycle core has been irradiated three cycles, one-third for two cycles, and one-third for one cycle.)
- b. Operation during an equilibrium cycle with 0.1% of all fuel rods leaking fission products from the time of initial fuel loading. The leaking fuel rods are randomly distributed throughout the entire core. To define an upper design limit, the fission product activity has also been calculated assuming 1% failed fuel.
- c. Continuous reactor coolant purification through the purification demineralizer at an average rate of one reactor coolant system volume per day with a zero removal efficiency for krypton, xenon, tellurium, molybdenum, cesium, yttrium, and tritium and a 99% removal efficiency for all other nuclides.
- d. All nuclides, except krypton, xenon, and tritium are removed with a 99.9% efficiency during the processing of the reactor coolant bleed. Krypton and xenon are removed with a 99.9% efficiency during the processing of the reactor coolant bleed for the first 267 days of a 277-day equilibrium cycle. After that time, when the deborating demineralizers are used for bleed stream processing, the removal of krypton and xenon ceases unless the reactor coolant bleed evaporators are kept in service to de-gas the bleed to prevent the buildup of krypton and xenon in the reactor coolant.

The resulting reactor coolant activities in $\mu\text{Ci/cc}$ of hot reactor coolant (0.713 gm/cc) for 0.1% failed fuel and 1.0% failed fuel are given in Tables 11.1-4 and 11.1-5, respectively.

11.1.2.2 Tritium Activity

Tritium is generated by several mechanisms, but only three are significant; ternary fission in the fuel with subsequent transport to the reactor coolant, neutron activation of boron in the reactor coolant passing through the core, and neutron activation of lithium in the reactor coolant passing through the core.

11.1.2.2.1 Ternary Fission

Tritium is produced in the fuel by ternary fission. The rate equation for the tritium inventory in the reactor coolant resulting from ternary fission is:

$$\frac{dT_1}{dt} = YfPD - (\lambda + k)T_1$$

where T_1 = inventory of tritium from ternary fission in the reactor coolant system, atoms

Y = tritium fission yield, 8.7×10^{-5} tritium atom/fission (ref. 6)

f = fissions per second per unit thermal power

P = core thermal power

D = fraction of tritium escaping from the fuel and entering coolant

λ = tritium decay constant, sec^{-1}

k = loss rate from system, sec^{-1}

The fraction of tritium escaping from the fuel (D) includes both the fraction that diffuses through the Zircaloy cladding and the fraction that leaks from defective fuel rods. Based on work done at ORNL (ref. 7), one % of the tritium produced in non-defective fuel rods is assumed to diffuse through the Zircaloy cladding and enter the reactor coolant.

To calculate a maximum reactor coolant activity, the loss rate from the system (k) is assumed to be zero.

11.1.2.2.2 Boron Activation

Tritium is produced from the neutron activation of natural boron (19.6% ^{10}B) dissolved in the reactor coolant primarily by the reaction $^{10}\text{B} (n, 2\alpha)^3\text{H}$. The rate equation for the tritium inventory in the reactor coolant resulting from boron activation is:

$$\frac{dT_2}{dt} = V_C \sigma_1 \phi_1 N_B - (\lambda + k)T_2$$

where T_2 = inventory of tritium from boron activation, atoms

V_C = water volume in the active core region, cc

σ_1 = reaction cross section for the $^{10}\text{B} (n, 2\alpha)^3\text{H}$ reaction, cm^2

ϕ_1 = neutron flux, above 1.2 Mev threshold energy, $\text{n/cm}^2/\text{sec}$

N_B = boron concentration as a function of time ($N_B = N_{BO} - R_B t$), atoms/cc

N_{BO} = initial boron concentration, atoms/cc

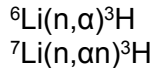
R_B = boron removal rate, atoms/cc/sec

λ = tritium decay constant, sec⁻¹

k = loss rate from system (assumed to be zero), sec⁻¹

11.1.2.2.3 Lithium Activation

Tritium is produced from the lithium in the reactor coolant by the reactions:



Lithium (99.9% ⁷Li) is added to the reactor coolant for pH control, and ⁷Li is also produced by the activation of ¹⁰B by the reaction ¹⁰B(n,α)⁷Li. The lithium concentration is assumed to be a constant 2 ppm since lithium is removed from the reactor coolant system during power operation to maintain the lithium concentration in the range 0.2 - 2.0 ppm. The rate equation for the tritium inventory in the reactor coolant resulting from lithium activation is:

$$\frac{dT_3}{dt} = V_C \sigma_2 \phi_2 X_1 N_L + V_C \sigma_3 \phi_3 X_2 N_L - (\lambda + k) T_3$$

where T_3 = inventory of tritium from lithium activation, atoms

V_C = water volume in the active core region, cc

σ_2 = reaction cross-section for the ⁶Li(n,α)³H reaction, cm²

ϕ_2 = thermal neutron flux, n/cm²/sec

σ_3 = reaction cross-section for the ⁷Li(n,αn)³H reaction, cm²

ϕ_3 = neutron flux above 2.8 Mev threshold energy, n/cm²/sec

N_L = lithium concentration, atoms/cc

X_1 = fraction of N_L which is ⁶Li

X_2 = fraction of N_L which is ⁷Li

λ = tritium decay constant, sec⁻¹

k = loss rate from system (assumed to be zero), sec⁻¹

Even though ⁷Li is produced from boron activation, it is assumed that the lithium in the reactor coolant remains at 99.9% ⁷Li (the isotopic composition of lithium added for pH control).

Using the above equations, the calculated amount of tritium produced in or entering the reactor coolant during the initial cycle and equilibrium cycles is given in Table 11.1-6. This quantity of tritium becomes uniformly distributed in the reactor coolant system, the clean liquid radwaste

system, and the boron recovery system. During refueling operations, the tritium is further diluted when the refueling canal is filled from the borated water storage tank.

11.1.2.3 Corrosion Product Activity

Corrosion products suspended in the reactor coolant may become activated when passing through the core. The most common corrosion products are ^{51}Cr , ^{54}Mn , ^{55}Fe , ^{59}Fe , ^{58}Co , ^{60}Co , and ^{95}Zr . The specific corrosion product activity is dependent on many factors, including the type of reactor plant and the materials of construction.

The corrosion product transport process is complex and stochastic, and calculational methods to accurately predict corrosion product activity have not been successfully correlated with operational data (ref. 8). Without operational data for the specific reactor system, analytical predictions of the corrosion product activity levels are at best rough approximations. Therefore, in the absence of an acceptable calculation method, the best alternative is to extrapolate data from an operating reactor (ref. 9) to the power level of this station.

The corrosion product activities in the reactor coolant in $\mu\text{Ci/cc}$ of hot reactor coolant (0.713 gm/cc) are given in Table 11.1-7.

11.1.2.4 Nitrogen-16 Activity

Nitrogen-16 is a concern only during reactor operation because of its short half life, 7.1 seconds. Nitrogen-16 is produced by the reaction $^{16}\text{O} (n,p) ^{16}\text{N}$ with an effective threshold at 10.7 Mev. The ^{16}N activity is computed with a digital computer code which calculates the equilibrium ^{16}N activity at any point in the reactor coolant system. The code accounts for the energy-dependent microscopic cross-section over the range 10.7 to 20 Mev and the variation of coolant density throughout the reactor coolant system. The code input parameters include reactor coolant velocities and densities at several locations in the system, physical dimensions of the core and the reactor coolant system, and the neutron flux at 10.7 Mev with the appropriate radial, axial, and energy slopes at various core locations. This input enables the code to divide the core into various regions and calculate the ^{16}N production from each region.

The Nitrogen-16 activity in $\mu\text{Ci/cc}$ of hot reactor coolant (0.713 gm/cc) is given in Table 11.1-8 for various points in the reactor coolant system.

11.1.2.5 Argon-41 Activity

Argon-41 is produced by Argon-40 activation. Argon-40 enters the reactor coolant in the makeup water from argon that may be present in the storage tank's nitrogen cover gas. The equilibrium ^{41}Ar activity in the reactor coolant is calculated by assuming a constant production rate from ^{40}Ar , and assuming decay is the only loss. The ^{40}Ar concentration in the reactor coolant is assumed to be constant at a value corresponding to the makeup water being saturated with atmospheric nitrogen (which is assumed to be 1.185 volume % ^{40}Ar) (ref. 10).

The equilibrium reactor coolant ^{41}Ar activity is 0.079 $\mu\text{Ci/cc}$ of hot reactor coolant.

11.1.3 Estimates of Gaseous Activity Released from Liquid or Steam Leakage

11.1.3.1 Leakage Rates

11.1.3.1.1 Containment Vessel and Auxiliary Building

No leakage is assumed from non-diaphragm process valve bonnet flanges, diaphragm valves, canned pumps (without seal), diaphragm compressors, small flanges, the reactor closure head, steam generator manholes and covers, pressurizer manholes and heaters, and core flooding, waste gas decay, waste gas surge and makeup tank manholes. For potential sources considered, the leakage rates assumed are as follows:

<u>Source</u>	<u>Assumed Leakage ($\rho = 1 \text{ gm/ml}$)</u>
a. Process Valves (non-diaphragm)	3 cc/hr
b. Instrument Valves (non-diaphragm)	3 cc/hr
c. Flanges (large)	30 cc/hr
d. Makeup Pump Seals (each)	9 gpd
e. Decay Heat Pump Seals (each)	60 cc/hr
f. Reactor Coolant Pump Seals (each)	1 gph (steady-state operation @ 700 psi Δp per seal)

Using the individual leakage rates given above, the total leakage rates have been estimated and are given in Table 11.1-9. The overall total leakage is estimated at 5.06 gph for normal operation. (Note: a 5 gph leak rate is used in Table 11.2-5 to estimate total annual miscellaneous system leakage.) In compiling Table 11.1-9, the simplifying assumption has been made that all secondary system leakage is confined to the turbine building.

11.1.3.1.2 Turbine Building

A gross leakage rate of 10 gpm is assumed for the turbine building. (Note: all leakage rates are based on water with a density of 1 gm/ml.) A simplified breakdown of this figure, based on the form and contamination level of the leakage, is given in Table 11.1-9.

11.1.3.2 Activity Levels in Leakage

To simplify calculations, potentially radioactive leakage is divided into seven (7) categories:

1. Primary coolant upstream of purification demineralizer
2. Primary coolant downstream of purification demineralizer
3. Refueling canal water upstream of demineralizer
4. Refueling canal water downstream of demineralizer
5. Secondary system steam

6. Secondary system liquid upstream of condensate polishing demineralizer
7. Secondary system liquid downstream of condensate polishing demineralizers

The maximum and average activity levels in each of these seven classes are given in Table 11.1-10 for those radionuclides that would be expected to evolve from the leakage in a gaseous form (i.e., iodines and noble gases). The assumptions used in compiling this table are as follows:

- a. Primary Coolant Upstream of Purification Demineralizer: This contains radionuclide concentrations equal to the maximum levels attained in the primary system during an equilibrium fuel cycle (see Tables 11.1-4 and 11.1-5). All values are corrected to correspond to a coolant density of 1 gm/ml. Maximum and average specific activities are based, respectively, on a 1% and 0.1% failed fuel condition.
- b. Primary Coolant Downstream of Purification Demineralizer: This contains the same activity levels as in (a) except that iodine concentrations are reduced by a factor of 100 to take into account passage through the purification demineralizer.
- c. Refueling Canal Water Upstream of Demineralizer: Little data has been found regarding either the activity levels expected to be present in refueling canal water or the fuel leak rates associated with the core during a cold shutdown. Qualitatively, it is obvious that, as the fuel cools, small openings in the cladding would tend to close and internal pressure in the fuel pins would decrease. Both of these actions should have the effect of reducing leakage below levels assumed for power operation. Of course, the reduction of core radionuclide inventories through decay would also limit leakage in time. In any event, except for tritium, no operating report has been found which indicates that gaseous radionuclides normally present a serious problem during refueling. There was an instance at the Haddam Neck Plant where a significant amount of iodine was released (ref. 11). However, it is believed that this type of occurrence can be prevented if the reactor vessel is thoroughly purged prior to its disassembly for refueling.

Based on the above reasoning, it is assumed that there is no significant leakage of iodine or noble gases from the core after the reactor shuts down. This means that practically all contamination is the result of activity that was present in the reactor coolant system water when the canal was filled. Therefore, disregarding the effects of decay, the canal water activity levels are based on roughly 37,400 gallons of primary coolant being diluted in 360,000 gallons of water that, except for tritium, is essentially free of radioactivity. The first of these quantities represents the amount of coolant remaining in the reactor vessel after it has been drained for refueling while the second represents the amount of water from the borated water storage tank needed to fill the canal. The activity levels in the primary coolant are assumed to be the same as those specified in Part (a) of this section except that, due to pre-refueling degassing, no noble gases are present.

The preceding assumptions result in refueling canal water radionuclide concentrations of about 10% of those given in Part (a).

Davis-Besse Unit 1 Updated Final Safety Analysis Report

- d. Refueling Canal Water Downstream of Demineralizer: This contains the same activity levels as in (c) except they have been reduced by a factor of 100 to take into account passage through either the purification or spent fuel demineralizers.
- e. Secondary System Steam Leakage:
 - 1. Primary-to-secondary leakage is 100 gpd.
 - 2. Maximum and average radionuclide activity levels in primary coolant are equal to those specified in Part (a).
 - 3. Two-thirds of the total secondary flow passes through a condensate polishing demineralizer while one-third bypasses it.
 - 4. The condensate polishing demineralizers have a DF of 5 for all ions and particulates and 1 for all gases.
 - 5. The partition factor for all radionuclides in the steam generator is 1.
 - 6. All noble gases in primary-to-secondary leakage are removed in the steam jet air ejector. Noble gases created by the decay of other radionuclides are treated later.
 - 7. The iodine partition factor in the steam jet air ejector is 1000 (see Subsection 11.3.2.3)
 - 8. All leakage occurs between the steam generators and the turbine.
- f. Secondary System Liquid Leakage Upstream of Condensate Demineralizer: The assumptions used are the same as those of Part (e) except that all leakage occurs between the condenser and the condensate polishing demineralizers.
- g. Secondary System Liquid Leakage Downstream of Condensate Demineralizer: The assumptions used are the same as those of Part (e) except that all leakage occurs between the steam generators and the point where the flow from the condensate polishing demineralizers mixes with that portion of the flow which bypassed the demineralizers.

It is noted that leakage of spent fuel pool water is not being considered. This is possible because, based on the assumptions used in Part (c) of this section, the activity levels of iodines and noble gases contained in it should not be significant. There will be some mixing of this water with refueling canal water during refueling operations. However, this should not be enough to result in major contamination.

11.1.3.3 Evolution of Gases From Leakage

When process water leakage is exposed to the atmosphere, there is a tendency for dissolved iodines and/or noble gases present to come out of solution. The % evolution that is assumed for the isotopes of each of these elements is as follows:

Davis-Besse Unit 1 Updated Final Safety Analysis Report

Iodine:	100%	(steam leakage)
	10%	(hot leakage)
	0.1%	(cold leakage)
Noble Gases:	100%	

These partition factors are more conservative than the values accepted by the NRC for use in evaluating the environmental impact of nuclear power stations (ref. 12). Using them with the total leakage rates given in Subsection 11.1.3.1 and assuming that all gases that are going to escape do so as the leakage occurs, the evolution rates of iodine and noble gases are as given in Table 11.1-11.

11.1.3.4 Transport and Environmental Releases of Evolved Gases

As access to the containment vessel is generally not necessary during reactor operation, gaseous activity levels there are normally allowed to build up. Under these conditions, equilibrium (and maximum) activity inventories will probably be attained by the shorter-lived isotopes. This activity is released to the environment when the plant is shutdown using the containment purge system or through the EVS system, if needed.

Gases evolved from leakage in the auxiliary building are assumed to be immediately purged to the station vent by the ventilation system. This ventilation system produces an air flow in the fuel handling and radwaste areas of 20,000 and 47,160 CFM, respectively. As with the containment vessel, if too much iodine is present air can be passed through the EVS charcoal filters at a reduced rate of 8,000 or 16,000 CFM.

11.1.3.5 Measurement and Control of Leakage

Technical Specifications ensure that leakage is kept as low as practical for systems outside containment that could contain highly radioactive fluids during a serious transient or accident. USAR Section 5.2.4 discusses RCS leakage detection. Leakage considerations for other systems can be found in the appropriate USAR sections for the systems. Section 15.4.6.5 provides the consequences of leakage of an Engineered Safety Features system during the Maximum Hypothetical Accident.

In general, leakage is minimized through the use of good maintenance and design practices. Included in the latter category is the extensive use of canned pumps, diaphragm valves, and live-loaded packing for valves in the various systems. Both of these items, due to the nature of their construction, have a very small potential for leakage.

TABLE 11.1-1

Escape Rate Coefficients

<u>Element</u>	<u>Escape Rate Coefficient, sec⁻¹</u>
Kr, Xe	6.5×10^{-8}
Br, I, Cs, Rb	1.3×10^{-8}
Te, Se, Sb, Sn	1.0×10^{-9}
Mo, Nb	2.0×10^{-9}
Sr, Ba	1.0×10^{-11}
Y, La, Ce	1.6×10^{-12}

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.1-2

Fission Product Activity in the Fuel

Core Activity, Ci* for Indicated Days Irradiation in Equilibrium Cycle

Radio-nuclide	4	10	30	60	90	120	150	180	210	240	267	277
Kr-83m	8.6 (+6)	8.6 (+6)	8.6 (+6)	8.6 (+6)	8.6 (+6)	8.6 (+6)	8.6 (+6)	8.6 (+6)	8.6 (+6)	8.6 (+6)	8.6 (+6)	8.6 (+6)
Kr-85m	2.46 (+7)	2.46 (+7)	2.46 (+7)	2.46 (+7)	2.46 (+7)	2.46 (+7)	2.46 (+7)	2.46 (+7)	2.46 (+7)	2.46 (+7)	2.46 (+7)	2.46 (+7)
Kr-85	2.59 (+5)	2.65 (+5)	2.83 (+5)	3.11 (+5)	3.47 (+5)	3.65 (+5)	3.93 (+5)	4.2 (+5)	2.94 (+5)	4.73 (+5)	4.97 (+5)	5.06 (+5)
Kr-87	4.52 (+7)	4.52 (+7)	4.52 (+7)	4.52 (+7)	4.52 (+7)	4.52 (+7)	4.52 (+7)	4.52 (+7)	4.52 (+7)	4.52 (+7)	4.52 (+7)	4.52 (+7)
Kr-88	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)
Rb-88	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)	6.84 (+7)
Sr-89	4.84 (+7)	5.06 (+7)	5.82 (+7)	6.64 (+7)	7.19 (+7)	7.57 (+7)	7.82 (+7)	7.99 (+7)	8.11 (+7)	8.18 (+7)	8.24 (+7)	8.25 (+7)
Sr-90	1.98 (+6)	2.02 (+6)	2.17 (+6)	2.39 (+6)	2.6 (+6)	2.82 (+6)	3.02 (+6)	3.24 (+6)	3.46 (+6)	3.66 (+6)	3.86 (+6)	7.79 (+6)
Sr-91	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)
Sr-92	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)
Y-90	1.96 (+6)	2.0 (+6)	2.14 (+6)	2.36 (+6)	2.57 (+6)	2.79 (+6)	3.0 (+6)	3.21 (+6)	3.43 (+6)	3.63 (+6)	3.82 (+6)	3.9 (+6)
Y-91	6.25 (+7)	6.57 (+7)	7.5 (+7)	8.51 (+7)	9.22 (+7)	9.72 (+7)	1.01 (+8)	1.03 (+8)	1.05 (+8)	1.06 (+8)	1.07 (+8)	1.07 (+8)
Mo-99	8.9 (+7)	1.29 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+8)
Xe-131m	2.76 (+5)	2.91 (+5)	4.41 (+5)	5.57 (+5)	5.85 (+5)	5.91 (+5)	5.91 (+5)	5.91 (+5)	5.91 (+5)	5.91 (+5)	5.91 (+5)	5.91 (+5)
Xe-133m	1.88 (+6)	3.18 (+6)	3.45 (+6)	3.45 (+6)	3.45 (+6)	3.45 (+6)	3.45 (+6)	3.45 (+6)	3.45 (+6)	3.45 (+6)	3.45 (+6)	3.45 (+6)
Xe-133	5.3 (+7)	1.02 (+8)	1.4 (+8)	1.43 (+8)	1.43 (+8)	1.43 (+8)	1.43 (+8)	1.43 (+8)	1.43 (+8)	1.43 (+8)	1.43 (+8)	1.43 (+8)
Xe-135m	3.52 (+7)	3.52 (+7)	3.52 (+7)	3.52 (+7)	3.52 (+7)	3.52 (+7)	3.52 (+7)	3.52 (+7)	3.52 (+7)	3.52 (+7)	3.52 (+7)	3.52 (+7)
Xe-135	1.76 (+7)	1.76 (+7)	1.76 (+7)	1.76 (+7)	1.76 (+7)	1.76 (+7)	1.76 (+7)	1.76 (+7)	1.76 (+7)	1.76 (+7)	1.76 (+7)	1.76 (+7)
Xe-138	1.27 (+8)	1.27 (+8)	1.27 (+8)	1.27 (+8)	1.27 (+8)	1.27 (+8)	1.27 (+8)	1.27 (+8)	1.27 (+8)	1.27 (+8)	1.27 (+8)	1.27 (+8)
I-131	3.13 (+7)	4.88 (+7)	7.0 (+7)	7.12 (+7)	7.44 (+7)	7.44 (+7)	7.44 (+7)	7.44 (+7)	7.44 (+7)	7.44 (+7)	7.44 (+7)	7.44 (+7)
I-132	6.31 (+7)	9.64 (+7)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)	1.09 (+8)
I-133	1.38 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)
I-134	1.83 (+8)	1.83 (+8)	1.83 (+8)	1.83 (+8)	1.83 (+8)	1.83 (+8)	1.83 (+8)	1.83 (+8)	1.83 (+8)	1.83 (+8)	1.83 (+8)	1.83 (+8)
I-135	1.4 (+8)	1.4 (+8)	1.4 (+8)	1.4 (+8)	1.4 (+8)	1.4 (+8)	1.4 (+8)	1.4 (+8)	1.4 (+8)	1.4 (+8)	1.4 (+8)	1.4 (+8)
Cs-134	6.35 (+5)	6.46 (+5)	6.87 (+6)	7.54 (+5)	8.24 (+5)	9.01 (+5)	9.81 (+5)	1.06 (+6)	1.15 (+6)	1.24 (+6)	1.32 (+6)	1.35 (+6)
Cs-136	4.27 (+ 5)	5.73 (+6)	8.22 (+5)	9.26 (+5)	9.48 (+5)	9.52 (+5)	9.53 (+5)	9.53 (+5)	9.53 (+5)	9.53 (+5)	9.53 (+5)	9.53 (+5)
Cs-137	2.47 (+6)	2.53 (+6)	2.7 (+6)	2.97 (+6)	3.23 (+6)	3.5 (+6)	3.77 (+6)	4.03 (+6)	4.29 (+6)	4.56 (+6)	4.8 (+6)	4.89 (+6)
Cs-138	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)
Ba-137m	2.27 (+6)	2.32 (+6)	2.48 (+6)	2.73 (+6)	2.98 (+6)	3.23 (+6)	3.47 (+6)	3.71 (+6)	3.96 (+6)	4.2 (+6)	4.42 (+6)	4.5 (+6)
Ba-139	1.41 (+8)	1.41 (+8)	1.41 (+8)	1.41 (+6)	1.41 (+6)	1.41 (+6)	1.41 (+6)	1.41 (+6)	1.41 (+6)	1.41 (+6)	1.41 (+6)	1.41 (+6)
Ba-140	6.45 (+7)	8.65 (+7)	1.24 (+8)	1.4 (+8)	1.43 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)
La-140	5.66 (+7)	7.81 (+7)	1.21 (+8)	1.39 (+8)	1.43 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)	1.44 (+8)
Ce-144	5.09 (+7)	5.19 (+7)	5.56 (+7)	6.07 (+7)	6.56 (+7)	7.0 (+7)	7.41 (+7)	7.8 (+7)	8.15 (+7)	8.49 (+7)	8.77 (+7)	8.88 (+7)

* Note: In this table, a number A x 10^{-B} is expressed as A(-B)

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.1-3

Fission Product Activity in the Fuel Rod Gaps

Fuel Rod Gap Activity, Ci (Total Core)*, For Indicated Days Irradiation in Equilibrium Cycle

Radio-nuclide	4	10	30	60	90	120	150	180	210	240	267	277
Kr-83m	6.87 (+3)	6.87 (+3)	6.87 (+3)	6.87 (+3)	6.87 (+3)	6.87 (+3)	6.87 (+3)	6.87 (+3)	6.87 (+3)	6.87 (+3)	6.87 (+3)	6.87 (+3)
Kr-85m	3.63 (+4)	3.63 (+4)	3.63 (+4)	3.63 (+4)	3.63 (+4)	3.63 (+4)	3.63 (+4)	3.63 (+4)	3.63 (+4)	3.63 (+4)	3.63 (+4)	3.63 (+4)
Kr-85	1.6 (+5)	1.62 (+5)	1.74 (+5)	1.92 (+5)	2.13 (+5)	2.34 (+5)	2.56 (+5)	2.78 (+5)	3.02 (+5)	3.26 (+5)	3.47 (+5)	3.55 (+5)
Kr-87	1.99 (+4)	1.99 (+4)	1.99 (+4)	1.99 (+4)	1.99 (+4)	1.99 (+4)	1.99 (+4)	1.99 (+4)	1.99 (+4)	1.99 (+4)	1.99 (+4)	1.99 (+4)
Kr-88	6.39 (+4)	6.39 (+4)	6.39 (+4)	6.39 (+4)	6.39 (+4)	6.39 (+4)	6.39 (+4)	6.39 (+4)	6.39 (+4)	6.39 (+4)	6.39 (+4)	6.39 (+4)
Rb-88	6.53 (+4)	6.53 (+4)	6.53 (+4)	6.53 (+4)	6.53 (+4)	6.53 (+4)	6.53 (+4)	6.53 (+4)	6.53 (+4)	6.53 (+4)	6.53 (+4)	6.53 (+4)
Sr-89	4.49 (+3)	4.6 (+3)	5.02 (+3)	5.66 (+3)	6.24 (+3)	6.72 (+3)	7.12 (+3)	7.42 (+3)	7.66 (+3)	7.84 (+3)	7.96 (+3)	8.0 (+3)
Sr-90	3.95 (+2)	4.05 (+2)	4.42 (+2)	5.01 (+2)	5.67(+2)	6.36 (+2)	7.13 (+2)	7.94 (+2)	8.8 (+2)	9.71 (+2)	1.06 (+3)	1.09 (+3)
Sr-91	1.08 (+3)	1.08 (+3)	1.08 (+3)	1.08 (+3)	1.08 (+3)	1.08 (+3)	1.08 (+3)	1.08 (+3)	1.08 (+3)	1.08 (+3)	1.08 (+3)	1.08 (+3)
Sr-92	2.4 (+2)	2.4 (+2)	2.4 (+2)	2.4 (+2)	2.4 (+2)	2.4 (+2)	2.4 (+2)	2.4 (+2)	2.4 (+2)	2.4 (+2)	2.4 (+2)	2.4 (+2)
Y-90	3.92 (+2)	4.01 (+2)	4.36 (+2)	4.95 (+2)	5.59 (+2)	6.29 (+2)	7.04 (+2)	7.84 (+2)	8.7 (+2)	9.61 (+2)	1.05 (+3)	1.08 (+3)
Y-91	1.27 (+3)	1.31 (+3)	1.44 (+3)	1.61 (+3)	1.76 (+3)	1.89 (+3)	1.99 (+3)	2.08 (+3)	2.14 (+3)	2.19 (+3)	2.23 (+3)	2.24 (+3)
Mo-99	2.71 (+4)	7.05 (+4)	9.9 (+4)	9.96 (+4)	9.96 (+4)	9.96 (+4)	9.96 (+4)	9.96 (+4)	9.96 (+4)	9.96 (+4)	9.96 (+4)	9.96 (+4)
Xe-131m	2.1 (+4)	2.29 (+4)	3.39 (+4)	5.04 (+4)	5.71 (+4)	5.89 (+4)	5.92 (+4)	5.94 (+4)	5.94 (+4)	5.94 (+4)	5.94 (+4)	5.94 (+4)
Xe-133m	1.58 (+4)	5.06 (+4)	6.78 (+4)	6.78 (+4)	6.78 (+4)	6.78 (+4)	6.78 (+4)	6.78 (+4)	6.78 (+4)	6.78 (+4)	6.78 (+4)	6.78 (+4)
Xe-133	1.0 (+6)	2.48 (+6)	5.55 (+6)	6.09 (+6)	6.11 (+6)	6.11 (+6)	6.11 (+6)	6.11 (+6)	6.11 (+6)	6.11 (+6)	6.11 (+6)	6.11 (+6)
Xe-135m	2.2 (+4)	2.2 (+4)	2.2 (+4)	2.2 (+4)	2.2 (+4)	2.2 (+4)	2.2 (+4)	2.2 (+4)	2.2 (+4)	2.2 (+4)	2.2 (+4)	2.2 (+4)
Xe-135	1.21 (+5)	1.21 (+5)	1.21 (+5)	1.21 (+5)	1.21 (+5)	1.21 (+5)	1.21 (+5)	1.21 (+5)	1.21 (+5)	1.21 (+5)	1.21 (+5)	1.21 (+5)
Xe-138	1.22 (+4)	1.22 (+4)	1.22 (+4)	1.22 (+4)	1.22 (+4)	1.22 (+4)	1.22 (+4)	1.22 (+4)	1.22 (+4)	1.22 (+4)	1.22 (+4)	1.22 (+4)
I-131	2.25 (+5)	3.51 (+5)	7.51 (+5)	9.29 (+5)	9.52 (+5)	9.55 (+5)	9.55 (+5)	9.55 (+5)	9.55 (+5)	9.55 (+5)	9.55 (+5)	9.55 (+5)
I-132	2.0 (+4)	4.35 (+4)	6.07 (+4)	6.12 (+4)	6.12 (+4)	6.12 (+4)	6.12 (+4)	6.12 (+4)	6.12 (+4)	6.12 (+4)	6.12 (+4)	6.12 (+4)
I-133	1.66 (+5)	1.99 (+5)	2.0 (+5)	2.0 (+5)	2.0 (+5)	2.0 (+5)	2.0 (+5)	2.0 (+5)	2.0 (+5)	2.0 (+5)	2.0 (+5)	2.0 (+5)
I-134	1.14 (+4)	1.14 (+4)	1.14 (+4)	1.14 (+4)	1.14 (+4)	1.14 (+4)	1.14 (+4)	1.14 (+4)	1.14 (+4)	1.14 (+4)	1.14 (+4)	1.14 (+4)
I-135	6.32 (+4)	6.32 (+4)	6.32 (+4)	6.32 (+4)	6.32 (+4)	6.32 (+4)	6.32 (+4)	6.32 (+4)	6.32 (+4)	6.32 (+4)	6.32 (+4)	6.32 (+4)
Cs-134	3.05 (+5)	3.14 (+5)	3.48 (+5)	4.02 (+5)	4.61 (+5)	5.28 (+5)	6.0 (+5)	6.78 (+5)	7.63 (+5)	8.56 (+5)	9.45 (+5)	9.79 (+5)
Cs-136	6.46 (+3)	7.57 (+3)	1.26 (+4)	1.73 (+4)	1.89 (+4)	1.94 (+4)	1.95 (+4)	1.95 (+4)	1.95 (+4)	1.95 (+4)	1.95 (+4)	1.95 (+4)
Cs-137	5.22 (+5)	5.35 (+5)	5.8 (+5)	6.54 (+5)	7.33 (+5)	8.2 (+5)	9.11 (+5)	1.01 (+6)	1.1 (+6)	1.22 (+6)	1.32 (+6)	1.36 (+6)
Cs-138	1.75 (+4)	1.75 (+4)	1.75 (+4)	1.75 (+4)	1.75 (+4)	1.75 (+4)	1.75 (+4)	1.75 (+4)	1.75 (+4)	1.75 (+4)	1.75 (+4)	1.75 (+4)
Ba-137m	4.8 (+5)	4.93 (+5)	5.33 (+5)	6.02 (+5)	6.74 (+5)	7.54 (+5)	8.39 (+5)	9.27 (+5)	1.02 (+6)	1.12 (+6)	1.21 (+6)	1.25 (+6)
Ba-139	1.96 (+3)	1.96 (+3)	1.96 (+3)	1.96 (+3)	1.96 (+3)	1.96 (+3)	1.96 (+3)	1.96 (+3)	1.96 (+3)	1.96 (+3)	1.96 (+3)	1.96 (+3)
Ba-140	8.86 (+2)	1.06 (+3)	1.73 (+3)	2.32 (+3)	2.51 (+3)	2.57 (+3)	2.58 (+3)	2.58 (+3)	2.58 (+3)	2.58 (+3)	2.58 (+3)	2.58 (+3)
La-140	8.86 (+2)	1.01 (+3)	1.7 (+3)	2.34 (+3)	2.55 (+3)	2.6 (+3)	2.63 (+3)	2.63 (+3)	2.63 (+3)	2.63 (+3)	2.63 (+3)	2.63 (+3)
Ce-144	1.23 (+3)	1.25 (+3)	1.34 (+3)	1.47 (+3)	1.63 (+3)	1.79 (+3)	1.96 (+3)	2.12 (+3)	2.3 (+3)	2.47 (+3)	2.63 (+3)	2.69 (+3)

* Note: In this table, a number A x 10^{-B} is expressed as A(-B)

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.1-4

Calculated Reactor Coolant Activity for 0.1% Failed Fuel

Fission Product Activity in Reactor Coolant, $\mu\text{Ci/cc}^*$, For
Indicated Time in Equilibrium Cycle, Full Power Days

Radio-nuclide	4	10	30	60	90	120	150	180	210	240	267	277
Kr-83m	2.05 (-2)	2.05 (-2)	2.05 (-2)	2.05 (-2)	2.05 (-2)	2.05 (-2)	2.05 (-2)	2.05 (-2)	2.05 (-2)	2.05 (-2)	2.04 (-2)	2.05 (-2)
Kr-85m	0.109	0.109	0.109	0.109	0.109	0.108	0.108	0.108	0.108	0.108	0.108	0.109
Kr-85	6.74 (-3)	1.68 (-2)	5.03 (-2)	9.92 (-2)	0.144	0.182	0.211	0.224	0.221	0.184	0.106	0.132
Kr-87	5.96 (-2)	5.96 (-2)	5.96 (-2)	5.96 (-2)	5.96 (-2)	5.96 (-2)	5.96 (-2)	5.96 (-2)	5.96 (-2)	5.96 (-2)	5.93 (-2)	5.96 (-2)
Kr-88	0.191	0.191	0.191	0.191	0.191	0.191	0.191	0.191	0.191	0.191	0.190	0.191
Rb-88	0.192	0.192	0.192	0.192	0.192	0.192	0.192	0.192	0.192	0.192	0.192	0.192
Sr-89	2.27 (-4)	2.38 (-4)	2.58 (-4)	2.80 (-4)	2.94 (-4)	3.04 (-4)	3.10 (-4)	3.15 (-4)	3.18 (-4)	3.20 (-4)	3.21 (-4)	3.21 (-4)
Sr-90	5.19 (-6)	5.40 (-6)	5.78 (-6)	6.34 (-6)	6.90 (-6)	7.46 (-6)	8.02 (-6)	8.58 (-6)	9.13 (-6)	9.68 (-6)	1.02 (-5)	1.04 (-5)
Sr-91	2.03 (-3)	2.03 (-3)	2.03 (-3)	2.03 (-3)	2.03 (-3)	2.03 (-3)	2.03 (-3)	2.03 (-3)	2.03 (-3)	2.03 (-3)	2.03 (-3)	2.03 (-3)
Sr-92	6.21 (-4)	6.21 (-4)	6.21 (-4)	6.21 (-4)	6.21 (-4)	6.21 (-4)	6.21 (-4)	6.21 (-4)	6.21 (-4)	6.21 (-4)	6.21 (-4)	6.21 (-4)
Y-90	9.78 (-6)	3.58 (-5)	1.36 (-4)	2.86 (-4)	4.26 (-4)	5.52 (-4)	6.54 (-4)	7.11 (-4)	7.19 (-4)	6.16 (-4)	3.59 (-4)	1.72 (-4)
Y-91	2.24 (-4)	5.76 (-4)	1.58 (-3)	2.69 (-3)	3.40 (-3)	3.83 (-3)	4.03 (-3)	3.97 (-3)	3.73 (-3)	3.12 (-3)	1.98 (-3)	1.07 (-3)
Mo-99	7.86(-2)	0.209	0.292	0.293	0.292	0.292	0.290	0.288	0.285	0.277	0.257	0.209
Xe-131m	1.60(-2)	3.41(-2)	8.70(-2)	0.140	0.157	0.160	0.158	0.153	0.147	0.133	0.104	0.134
Xe-133m	4.67(-2)	0.150	0.200	0.200	0.200	0.199	0.198	0.197	0.195	0.191	0.179	0.202
Xe-133	1.84	6.82	16.2	17.7	17.6	17.5	17.4	17.1	16.8	16.0	14.0	17.1
Xe-135m	6.58(-2)	6.59 (-2)	6.59 (-2)	6.59 (-2)	6.59 (-2)	6.59 (-2)	6.58 (-2)	6.58 (-2)	6.57 (-2)	6.56 (-2)	6.51 (-2)	6.60 (-2)
Xe-135	0.360	0.361	0.361	0.361	0.361	0.361	0.360	0.360	0.359	0.357	0.352	0.362
Xe-138	3.64 (-2)	3.64 (-2)	3.64 (-2)	3.64 (-2)	3.64 (-2)	3.64 (-2)	3.64 (-2)	3.64 (-2)	3.64 (-2)	3.64 (-2)	3.64 (-2)	3.64 (-2)
I-131	8.58 (-2)	0.145	0.215	0.229	0.230	0.230	0.230	0.230	0.230	0.230	0.230	0.230
I-132	5.26 (-2)	0.114	0.159	0.160	0.160	0.160	0.160	0.160	0.160	0.160	0.160	0.160
I-133	0.250	0.269	0.269	0.269	0.269	0.269	0.269	0.269	0.269	0.269	0.269	0.269
I-134	3.26 (-2)	3.26 (-2)	3.26 (-2)	3.26 (-2)	3.26 (-2)	3.26 (-2)	3.26 (-2)	3.26 (-2)	3.26 (-2)	3.26 (-2)	3.26 (-2)	3.26 (-2)
I-135	0.135	0.135	0.135	0.135	0.135	0.135	0.135	0.135	0.135	0.135	0.135	0.135
Cs-134	8.42 (-3)	2.09 (-2)	6.18 (-2)	0.121	0.174	0.223	0.263	0.286	0.291	0.255	0.157	8.10 (-2)
Cs-136	4.38 (-3)	1.17 (-2)	3.29 (-2)	4.82 (-2)	5.22 (-2)	5.27 (-2)	5.21 (-2)	5.05 (-2)	4.85 (-2)	4.41 (-2)	3.46 (-2)	2.26 (-2)
Cs-137	2.59 (-2)	6.45 (-2)	0.193	0.380	0.554	0.710	0.834	0.900	0.904	0.773	0.459	0.230
Cs-138	5.23 (-2)	5.23 (-2)	5.23 (-2)	5.23 (-2)	5.23 (-2)	5.23 (-2)	5.23 (-2)	5.23 (-2)	5.23 (-2)	5.23 (-2)	5.22 (-2)	5.21 (-2)
Ba-137m	2.37 (-2)	5.93 (-2)	0.177	0.350	0.510	0.653	0.767	0.828	0.832	0.711	0.422	0.212
Ba-139	5.41 (-3)	5.41 (-3)	5.41 (-3)	5.41 (-3)	5.41 (-3)	5.41 (-3)	5.41 (-3)	5.41 (-3)	5.41 (-3)	5.41 (-3)	5.41 (-3)	5.41 (-3)
Ba-140	1.92 (-4)	2.52 (-4)	3.51 (-4)	3.91 (-4)	3.99 (-4)	4.01 (-4)	4.01 (-4)	4.01 (-4)	4.01 (-4)	4.01 (-4)	4.01 (-4)	4.01 (-4)
La-140	6.86 (-5)	9.44 (-5)	1.38 (-4)	1.56 (-4)	1.59 (-4)	1.60 (-4)	1.60 (-4)	1.60 (-4)	1.60 (-4)	1.60 (-4)	1.60 (-4)	1.60 (-4)
Ce-144	2.08 (-5)	2.16 (-5)	2.31 (-5)	2.53 (-5)	2.73 (-5)	2.91 (-5)	3.08 (-5)	3.25 (-5)	3.40 (-5)	3.54 (-5)	3.65 (-5)	3.69 (-5)

* Notes: 1 – Coolant Density (ρ) = 0.713 gm/cc

2 - In this table, a number A x 10^{-B} is expressed as A(-B)

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.1-5

Calculated Reactor Coolant Activity for 1% Failed Fuel

Fission Product Activity in Reactor Coolant, $\mu\text{Ci/cc}^*$, For
Indicated Time in Equilibrium Cycle, Full Power Days

Radio-nuclide	4	10	30	60	90	120	150	180	210	240	267	277
Kr-83m	0.205	0.205	0.205	0.205	0.205	0.205	0.205	0.205	0.205	0.205	0.204	0.205
Kr-85m	1.09	1.09	1.09	1.09	1.09	1.08	1.08	1.08	1.08	1.08	1.08	1.09
Kr-85	6.74 (-2)	0.168	0.503	0.992	1.44	1.82	2.11	2.24	2.21	1.84	1.06	1.32
Kr-87	0.596	0.596	0.596	0.596	0.596	0.596	0.596	0.596	0.596	0.596	0.593	0.596
Kr-88	1.91	1.91	1.91	1.91	1.91	1.91	1.91	1.91	1.91	1.91	1.90	1.91
Rb-88	1.92	1.92	1.92	1.92	1.92	1.92	1.92	1.92	1.92	1.92	1.92	1.92
Sr-89	2.27 (-3)	2.38 (-3)	2.58 (-3)	2.80 (-3)	2.94 (-3)	3.04 (-3)	3.10 (-3)	3.15 (-3)	3.18 (-3)	3.20 (-3)	3.21 (-3)	3.21 (-3)
Sr-90	5.19 (-5)	5.40 (-5)	5.78 (-5)	6.34 (-5)	6.90 (-5)	7.46 (-5)	8.02 (-5)	8.58 (-5)	9.13 (-5)	9.68 (-5)	1.02 (-5)	1.04 (-5)
Sr-91	2.03 (-2)	2.03 (-2)	2.03 (-2)	2.03 (-2)	2.03 (-2)	2.03 (-2)	2.03 (-2)	2.03 (-2)	2.03 (-2)	2.03 (-2)	2.03 (-2)	2.03 (-2)
Sr-92	6.21 (-3)	6.21 (-3)	6.21 (-3)	6.21 (-3)	6.21 (-3)	6.21 (-3)	6.21 (-3)	6.21 (-3)	6.21 (-3)	6.21 (-3)	6.21 (-3)	6.21 (-3)
Y-90	9.78 (-5)	3.58 (-4)	1.36 (-3)	2.86 (-3)	4.26 (-3)	5.52 (-3)	6.54 (-3)	7.11 (-3)	7.19 (-3)	6.16 (-3)	3.59 (-3)	1.72 (-3)
Y-91	2.24 (-3)	5.76 (-3)	1.58 (-2)	2.69 (-2)	3.40 (-2)	3.83 (-2)	4.03 (-2)	3.97 (-2)	3.73 (-2)	3.12 (-2)	1.98 (-2)	1.07 (-2)
Mo-99	0.786	2.09	2.92	2.93	2.92	2.92	2.90	2.88	2.85	2.77	2.57	2.09
Xe-131m	0.160	0.341	0.870	1.40	1.57	1.60	1.58	1.53	1.47	1.33	1.04	1.34
Xe-133m	0.467	1.50	2.00	2.00	2.00	1.99	1.98	1.97	1.95	1.91	1.79	2.02
Xe-133	1.84 (+1)	6.82 (+1)	1.62 (+2)	1.77 (+2)	1.76 (+2)	1.75 (+2)	1.74 (+2)	1.71 (+2)	1.68 (+2)	1.60 (+2)	1.40 (+2)	1.71 (+2)
Xe-135m	0.658	0.659	0.659	0.659	0.659	0.659	0.658	0.658	0.657	0.656	0.651	0.660
Xe-135	3.60	3.61	3.61	3.61	3.61	3.61	3.60	3.60	3.59	3.57	3.52	3.62
Xe-138	0.364	0.364	0.364	0.364	0.364	0.364	0.364	0.364	0.364	0.364	0.364	0.364
I-131	0.858	1.45	2.15	2.29	2.30	2.30	2.30	2.30	2.30	2.30	2.30	2.30
I-132	0.526	1.14	1.59	1.60	1.60	1.60	1.60	1.60	1.60	1.60	1.60	1.60
I-133	2.50	2.69	2.69	2.69	2.69	2.69	2.69	2.69	2.69	2.69	2.69	2.69
I-134	0.326	0.326	0.326	0.326	0.326	0.326	0.326	0.326	0.326	0.326	0.326	0.326
I-135	1.35	1.35	1.35	1.35	1.35	1.35	1.35	1.35	1.35	1.35	1.35	1.35
Cs-134	8.42 (-2)	0.209	0.618	1.21	1.74	2.23	2.63	2.86	2.91	2.55	1.57	0.810
Cs-136	4.38 (-2)	0.117	0.329	0.482	0.522	0.527	0.521	0.505	0.485	0.441	0.346	0.226
Cs-137	0.259	0.645	1.93	3.80	5.54	7.10	8.34	9.00	9.04	7.73	4.59	2.30
Cs-138	0.523	0.523	0.523	0.523	0.523	0.523	0.523	0.523	0.523	0.523	0.522	0.521
Ba-137m	0.237	0.593	1.77	3.50	5.10	6.53	7.67	8.28	8.32	7.11	4.22	2.12
Ba-139	5.41 (-2)	5.41 (-2)	5.41 (-2)	5.41 (-2)	5.41 (-2)	5.41 (-2)	5.41 (-2)	5.41 (-2)	5.41 (-2)	5.41 (-2)	5.41 (-2)	5.41 (-2)
Ba-140	1.92 (-3)	2.52 (-3)	3.51 (-3)	3.91 (-3)	3.99 (-3)	4.01 (-3)	4.01 (-3)	4.01 (-3)	4.01 (-3)	4.01 (-3)	4.01 (-3)	4.01 (-3)
La-140	6.86 (-4)	9.44 (-4)	1.38 (-3)	1.56 (-3)	1.59 (-3)	1.60 (-3)	1.60 (-3)	1.60 (-3)	1.60 (-3)	1.60 (-3)	1.60 (-3)	1.60 (-3)
Ce-144	2.08 (-4)	2.16 (-4)	2.31 (-4)	2.53 (-4)	2.73 (-4)	2.91 (-4)	3.08 (-4)	3.25 (-4)	3.40 (-4)	3.54 (-4)	3.65 (-4)	3.69 (-4)

* Notes: 1 – Coolant Density (ρ) = 0.713 gms/cc

2 – In this table, a number A x 10^{-B} is expressed as A(-B)

TABLE 11.1-6

Tritium Production

<u>Tritium Source</u>	<u>Production per cycle, Ci</u>	
	<u>Initial Cycle</u>	<u>Equilibrium Cycle</u>
Ternary fission	172	118
Boron Activation	235	132
Lithium Activation	141	96
Total	548	346

TABLE 11.1-7

Corrosion Product Activity
In Reactor Coolant

<u>Radionuclide</u>	<u>Activity, $\mu\text{Ci/cc}$</u>
^{51}Cr	2.63×10^{-3}
^{54}Mn	3.01×10^{-4}
^{55}Fe	1.04×10^{-2}
^{59}Fe	3.01×10^{-4}
^{58}Co	1.58×10^{-2}
^{60}Co	8.46×10^{-5}
^{95}Zr	2.07×10^{-2}

TABLE 11.1-8

Nitrogen-16 Activity
In Reactor Coolant

<u>Location</u>	<u>Activity, $\mu\text{Ci/cc}$</u>
Reactor Vessel Outlet	1.62×10^2
Steam Generator Inlet	1.3×10^2
Steam Generator Outlet	7.8×10^1
Reactor Vessel Inlet	7.3×10^1

TABLE 11.1-9

Total System Leakage Rates

	<u>Leakage Rates, cc/hr</u>		
	<u>During Normal Operation</u>	<u>During Cooldown</u>	<u>During Refueling</u>
<u>Auxiliary Building</u>			
Hot Primary Coolant Upstream of Purification Demineralizer	15	375	0
Hot Primary Coolant Downstream of Purification Demineralizer	9	9	0
Cold Primary Coolant Upstream of Purification Demineralizer	789	789	0
Cold Primary Coolant Downstream of Purification Demineralizer	2314	894	0
Refueling Canal Water Upstream of Demineralizer	0	0	1173
Refueling Canal Water Downstream of Demineralizer	0	0	894
<u>Containment Building</u>			
Hot Primary Coolant Upstream of Purification Demineralizer	663	711	-
Hot Primary Coolant Downstream of Purification Demineralizer	15,286	144	-
Cold Primary Coolant Upstream of Purification Demineralizer	36	36	-
Cold Primary Coolant Downstream of Purification Demineralizer	24	24	-
<u>Turbine Building</u>			
Secondary System Steam	1,135,626	-	-
Secondary System Liquid Upstream of Condensate Polishing Demineralizer	378,542	-	-
Secondary System Liquid Downstream of Condensate Polishing Demineralizer	757,083	-	-

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.1-10

Radionuclide Specific Activities in Leakage

Isotope	Specific Activity in Leakage, $\mu\text{Ci}/\text{ml}$							
	Primary Coolant Upstream of Purification Demineralizer		Primary Coolant Downstream of Purification Demineralizer		Refueling Canal Water Upstream of Demineralizer		Refueling Canal Water Downstream of Demineralizer	
	Max.	Avg.	Max.	Avg.	Max.	Avg.	Max.	Avg.
Ar-41	0.111	0.111	0.111	0.111	0	0	0	0
Kr-83m	0.288	2.88×10^{-2}	0.288	2.88×10^{-2}	0	0	0	0
Kr-85m	1.53	0.153	1.53	0.153	0	0	0	0
Kr-85	3.15	0.315	3.15	0.315	0	0	0	0
Kr-87	0.837	8.37×10^{-2}	0.837	8.37×10^{-2}	0	0	0	0
Kr-88	2.68	0.268	2.68	0.268	0	0	0	0
I-131	3.23	0.323	3.23×10^{-2}	3.23×10^{-3}	0.323	3.23×10^{-2}	3.23×10^{-3}	3.23×10^{-4}
I-132	2.25	0.225	2.25×10^{-2}	2.25×10^{-3}	0.225	2.25×10^{-2}	2.25×10^{-3}	2.25×10^{-4}
I-133	3.78	0.378	3.78×10^{-2}	3.78×10^{-3}	0.378	3.78×10^{-2}	3.78×10^{-3}	3.78×10^{-4}
I-134	0.458	4.58×10^{-2}	4.58×10^{-3}	4.58×10^{-4}	4.58×10^{-2}	4.58×10^{-3}	4.58×10^{-4}	4.58×10^{-5}
I-135	1.90	0.190	1.90×10^{-2}	1.90×10^{-3}	0.190	1.90×10^{-2}	1.90×10^{-3}	1.90×10^{-4}
Xe-131m	2.25	0.225	2.25	0.225	0	0	0	0
Xe-133m	2.84	0.284	2.84	0.284	0	0	0	0
Xe-133	249.	24.9	249.	24.9	0	0	0	0
Xe-135m	0.926	9.26×10^{-2}	0.926	9.26×10^{-2}	0	0	0	0
Xe-135	5.08	0.508	5.08	0.508	0	0	0	0
Xe-138	0.511	5.11×10^{-2}	0.511	5.11×10^{-2}	0	0	0	0

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.1-10 (Continued)

Radionuclide Specific Activities in Leakage

Isotope	Specific Activity in Leakage, $\mu\text{Ci}/\text{ml}$					
	Secondary System Steam		Secondary System Liquid Upstream of Condensate Polishing Demineralizer		Secondary System Liquid Downstream of Condensate Polishing Demineralizer	
	Max.	Avg.	Max.	Avg.	Max.	Avg.
Ar-41	3.28×10^{-7}	3.28×10^{-7}	0	0	0	0
Kr-83m	8.49×10^{-7}	8.49×10^{-8}	0	0	0	0
Kr-85m	4.25×10^{-6}	4.25×10^{-7}	0	0	0	0
Kr-85	9.30×10^{-6}	9.30×10^{-7}	0	0	0	0
Kr-87	2.47×10^{-6}	2.47×10^{-7}	0	0	0	0
Kr-88	7.92×10^{-6}	7.92×10^{-7}	0	0	0	0
I-131	1.79×10^{-6}	1.79×10^{-6}	1.79×10^{-6}	1.79×10^{-6}	8.36×10^{-6}	8.35×10^{-7}
I-132	1.05×10^{-6}	1.05×10^{-6}	1.05×10^{-6}	1.05×10^{-6}	4.90×10^{-6}	4.90×10^{-7}
I-133	2.05×10^{-6}	2.05×10^{-6}	2.05×10^{-6}	2.05×10^{-6}	9.57×10^{-6}	9.57×10^{-7}
I-134	1.68×10^{-6}	1.68×10^{-7}	1.68×10^{-6}	1.68×10^{-7}	7.84×10^{-7}	7.84×10^{-6}
I-135	9.85×10^{-6}	9.85×10^{-7}	9.85×10^{-6}	9.85×10^{-7}	4.60×10^{-6}	4.60×10^{-7}
Xe-131m	6.66×10^{-6}	6.66×10^{-7}	0	0	0	0
Xe-133m	8.42×10^{-6}	8.42×10^{-7}	0	0	0	0
Xe-133	7.29×10^{-6}	7.29×10^{-6}	0	0	0	0
Xe-135m	2.73×10^{-6}	2.73×10^{-7}	0	0	0	0
Xe-135	1.51×10^{-6}	1.51×10^{-6}	0	0	0	0
Xe-138	1.51×10^{-6}	1.51×10^{-7}				

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.1-11

Total Evolution Rate of Radioactive Gases From System Leakage

Radio-nuclide	Evolution Rate, $\mu\text{Ci/s}$									
	Containment Vessel				Auxiliary Building					
	Normal Operation		Cooldown		Normal Operation		Cooldown		Refueling	
	Max.	Avg.	Max.	Avg.	Max.	Avg.	Max.	Avg.	Max.	Avg.
Ar-41	0.494	0.494	2.83×10^{-2}	2.83×10^{-2}	9.65×10^{-2}	9.65×10^{-2}	6.38×10^{-2}	6.38×10^{-2}	--	--
Kr-83m	1.29	0.13	7.32×10^{-2}	7.32×10^{-3}	0.251	2.51×10^{-2}	0.166	1.66×10^{-2}	--	--
Kr-85m	6.81	0.69	0.389	3.89×10^{-2}	1.33	0.133	0.879	8.79×10^{-2}	--	--
Kr-85	14.01	1.41	0.801	8.01×10^{-2}	2.74	0.274	1.81	0.181	--	--
Kr-87	3.73	0.38	0.213	2.13×10^{-2}	0.728	7.28×10^{-2}	0.481	4.81×10^{-2}	--	--
Kr-88	11.92	1.20	0.682	6.82×10^{-2}	2.33	0.233	1.54	0.154	--	--
I-131	7.33×10^{-2}	7.33×10^{-3}	6.40×10^{-2}	6.40×10^{-3}	2.09×10^{-3}	2.09×10^{-4}	3.44×10^{-2}	3.44×10^{-3}	1.07×10^{-4}	1.07×10^{-5}
I-132	5.11×10^{-2}	5.11×10^{-3}	4.46×10^{-2}	4.46×10^{-3}	1.46×10^{-3}	1.46×10^{-4}	2.40×10^{-2}	2.40×10^{-3}	7.39×10^{-5}	7.39×10^{-6}
I-133	8.58×10^{-2}	8.58×10^{-3}	7.49×10^{-2}	7.49×10^{-3}	2.44×10^{-3}	2.44×10^{-4}	4.03×10^{-2}	4.03×10^{-3}	1.25×10^{-4}	1.25×10^{-5}
I-134	1.04×10^{-2}	1.04×10^{-3}	9.07×10^{-3}	9.07×10^{-4}	2.96×10^{-4}	2.96×10^{-5}	4.88×10^{-3}	4.88×10^{-4}	1.51×10^{-5}	1.51×10^{-6}
I-135	4.31×10^{-2}	4.31×10^{-3}	3.77×10^{-2}	3.77×10^{-3}	1.23×10^{-3}	1.23×10^{-4}	2.03×10^{-2}	2.03×10^{-3}	6.24×10^{-5}	6.24×10^{-6}
Xe-131m	10.01	1.01	0.572	5.72×10^{-2}	1.96	0.196	1.30	0.130	--	--
Xe-133m	12.63	1.27	0.722	7.22×10^{-2}	2.47	0.247	1.64	0.164	--	--
Xe-133	1107.29	110.73	63.288	6.339	216.29	21.63	142.97	14.30	--	--
Xe-135m	4.12	0.42	0.236	2.36×10^{-2}	0.805	8.05×10^{-2}	0.632	5.32×10^{-2}	--	--
Xe-135	22.60	2.26	1.30	0.130	4.42	0.442	2.92	0.292	--	--
Xe-138	2.28	0.23	0.130	1.30×10^{-2}	0.444	4.44×10^{-2}	0.294	2.94×10^{-2}	--	--

TABLE 11.1-11 (Continued)

Total Evolution Rate of Radioactive Gases From System Leakage

Radio-nuclide	Evolution Rate, $\mu\text{Ci/s}$	
	Turbine Building	
	Max.	Avg.
Ar-41	1.04×10^{-4}	1.04×10^{-4}
Kr-83m	2.68×10^{-4}	2.68×10^{-5}
Kr-85m	1.43×10^{-3}	1.43×10^{-4}
Kr-85	2.94×10^{-3}	2.94×10^{-4}
Kr-87	7.80×10^{-4}	7.80×10^{-5}
Kr-88	2.50×10^{-3}	2.50×10^{-4}
I-131	5.66×10^{-3}	5.66×10^{-4}
I-132	3.32×10^{-3}	3.32×10^{-4}
I-133	6.48×10^{-3}	6.48×10^{-4}
I-134	5.31×10^{-4}	5.31×10^{-5}
I-135	3.11×10^{-3}	3.11×10^{-4}
Xe-131m	2.11×10^{-3}	2.11×10^{-4}
Xe-133m	2.66×10^{-3}	2.66×10^{-4}
Xe-133	0.230	2.30×10^{-2}
Xe-135m	8.62×10^{-4}	8.62×10^{-5}
Xe-135	4.77×10^{-3}	4.77×10^{-4}
Xe-138	4.77×10^{-4}	4.77×10^{-5}

11.2 LIQUID WASTE SYSTEMS

11.2.1 Design Objectives

The systems handling liquid wastes are designed such that the estimated releases in liquid effluents comply with the following requirements of 10CFR20 and 10CFR50:

- a. The individual radionuclide concentrations in liquid effluents at the site boundary shall not exceed the limits for releases to unrestricted areas given in Appendix B of 10CFR20.
- b. The releases of radioactivity from the station shall comply with the as low as reasonably achievable standard set forth in 10CFR50.

Davis-Besse commits to the regulatory position of Regulatory Guide 4.15 (Revision 1, February 1979)

11.2.2 System Descriptions

11.2.2.1 Process Arrangement

11.2.2.1.1 Clean Liquid Radwaste System

The functional drawing for this system is shown in Figure 11.2-2. A list of major components, as well as design information for each, is contained in Table 11.2-1. Figure 11.2-4 is a schematic of the process cycle.

11.2.2.1.2 Miscellaneous Liquid Radwaste System

The functional drawing for this system is shown in Figure 11.2-3. A list of major components, as well as design information for each, is contained in Table 11.2-2.

11.2.2.2 Waste Sources

Radiation sources, activities, quantities and concentrations are dependent upon various factors such as, the fuel load (enrichment of fuel) and the operating cycle length. The current information is maintained in the ODCM and the source term analysis for the current operating cycle reload report.

11.2.2.2.1 Clean Liquid Radwaste System

The major source of waste for this system is reactor coolant letdown resulting from boron dilution operations or from coolant expansion during reactor startups. Other sources include leakage, drainage, and relief flows from valves and equipment containing reactor-grade liquid.

The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

With two exceptions, the individual sources are listed in Table 11.2-3 along with estimates and breakdowns of the quantities that can be expected from each in a year. The exceptions noted are the waste caused by relief valve operation and by leakage. It is possible to omit the first because it is not the result of what is considered a normal or expected occurrence. Leakage was not included because all of it is conservatively assumed to be processed through the less efficient miscellaneous liquid radioactive waste system.

Of the quantities listed in Table 11.2-3 all but two represent normal letdown through the makeup and purification system. The two exceptions are the draining of portions of the primary system for refueling and maintenance operations. Waste from the former will probably be sent through the spent fuel pool or purification demineralizer before it reaches the radwaste system. Therefore, its activity levels should be less than, or equal to, those attributed to letdown. The other waste quantity, resulting from maintenance operations, could reach the radwaste system without any prior demineralization and have radionuclide concentrations in it that are higher than those estimated for letdown wastes. However, because of the following considerations, it is not necessary to take this into account when evaluating the clean liquid radioactive waste system:

- a. The same coolant that is drained for maintenance operations is usually replaced when work is finished. It therefore, contributes little to the activity releases from the station.
- b. As maintenance operations normally occur during refueling, there should be approximately 10,300 ppm of boric acid in the coolant. Based on a final bottoms concentration of about 7%, this means that the evaporators would be operating with a concentration factor of less than 7 (assuming all feed is at the 10,300 ppm level). This would result in a higher than normal evaporator DF (see Subsection 11.2.2.3), which, if this waste were processed, would compensate for any lack of demineralization.

For evaluation purposes, all clean liquid waste is assumed to consist entirely of primary system letdown. As noted previously, this means that the radionuclide specific activities in the waste are the same as those in the reactor coolant which has been passed through a letdown cooler and a mixed-bed purification demineralizer. The cooler is assumed to reduce the coolant temperature from an operating value of about 570°F to 120°F or less. This results in a maximum decrease of 1.403 in liquid volume and a corresponding increase in specific activity. The demineralizer is assumed to have a DF of 1 for Cs, Y, Mo, Xe, Kr, crud and tritium, and 100 for all other radionuclides. The waste has also passed through a filter, but no credit is taken for it.

The average radionuclide concentrations expected in the reactor coolant are assumed to be equal to the maximum levels attained during an equilibrium cycle with 0.1% failed fuel. These values (found in Table 11.1-4) are listed, along with the corresponding average radionuclide concentrations downstream of the letdown cooler and in the clean liquid radwaste system feed, in Table 11.2-4. The 0.1% failed-fuel condition represents the expected average situation over an operating year.

The maximum radionuclide concentrations occurring in the reactor coolant are taken to be equal to the maximum levels attained during an equilibrium cycle with 1% failed fuel. Values for these were extracted from Table 11.1-5 and are given in Table 11.2-4, along with corresponding maximum concentrations downstream of the letdown cooler and in the feed to the clean liquid radioactive waste system. The 1% failed fuel condition represents an upper design limit beyond which reactor operation is unlikely.

The estimated corrosion product activity levels present are those given in Table 11.1-7, multiplied by 1.403 to correct for the change in density caused by decreasing temperature.

This concludes the historical information contained in this subsection.

11.2.2.2.2 Miscellaneous Liquid Radwaste System

11.2.2.2.2.1 Non-Detergent Wastes

The major sources of this class of wastes are:

- Miscellaneous system leakage
- Drainage from area washdown
- Sluicing of demineralizer resins
- Sampling and laboratory operations
- Condensate polishing demineralizer backwash
- (If there is a significant primary-secondary leak)

Assuming all non-detergent wastes are, at some time, exposed to the atmosphere, the various isotopes of xenon and krypton are assumed to have come out of solution and to no longer be present in significant amounts.

- a. Drainage from Area Washdowns, Sampling and Laboratory Operations, and Miscellaneous System Leakage

These three sources are grouped together because they can generally be categorized as consisting of some mixture of relatively clean water, processed or unprocessed reactor coolant, and/or spent fuel pool water. The possible contribution of the spent fuel pool is treated separately only because its activity levels are not derived directly from those in the primary system.

Although no estimates are included for the activity levels in the spent fuel pool water, they are normally kept relatively low to permit access to the pool area. Therefore, undiluted reactor coolant represents the upper limit of concentration found in waste from the four sources included in this group.

- b. Backflushing of Prefilter and Sluicing of Demineralizer Resins

This is the excess water produced during backflushing of the prefilter in the purification and makeup system and/or produced during handling of spent demineralizer resins. The former operation results in dislodged crud being added to the waste while the latter can contribute both crud and dissolved radioactive contamination. In both cases, the water used is assumed to be initially free of activity.

- c. Condensate Demineralizer Backwash

These wastes must be processed only if a primary-to-secondary leak produces significant contamination in the condensate polishing demineralizers' resin.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

Sources are listed in Table 11.2-5 along with the quantities of each that are expected to be produced each year.

Estimates of the maximum and average radionuclide concentrations in the waste from each individual source, or group of related sources, and in non-detergent waste as a whole are given in Tables 11.2-6 and 11.2-7, respectively.

Drainage from Area Washdowns, Sampling and Laboratory Operations, and Miscellaneous System Leakage:

- a. The maximum activity levels are set equal to the maximum values (corrected for temperature) attained in the primary system during a fuel cycle with 1% failed fuel.
- b. Estimates of the overall average activity levels, taking into account all three sources, are based on the following assumptions:

All the radioactivity attributed to these sources resides entirely in the waste created by miscellaneous system leakage, sampling, and laboratory operations. This assumption is justified on the basis that the contribution from area washdowns should contain only small amounts of activity. (This source is the result of cleaning operations in which relatively small quantities of contamination are diluted in larger volumes of wash water).

The entire 48,000 gallons of waste resulting from miscellaneous system leakage, sampling, and laboratory operations is unprocessed reactor coolant. The activity levels present are assumed to be the maximum values attained in the primary system during an equilibrium cycle with 0.1 % failed fuel. The designation of all 48,000 gallons as reactor coolant should be conservative since it assumes no leakage or sampling from systems containing less radioactivity.

Backflushing of Prefilter and Sluicing of Demineralizer Resin:

- a. Estimates of the dissolved radionuclide concentrations can be made based on equilibrium considerations. The situation is defined by the equilibrium coefficient, K_D :

$$K_D \text{ (trace ion distribution coef.)} = \frac{\text{conc. of ion in resin } (\mu\text{Ci/ml})}{\text{conc. of ion in liquid } (\mu\text{Ci/ml})}$$

Values for K_D have been determined experimentally for cesium and iodine for a range of conditions (ref. 13). While the numbers derived vary with the makeup of the liquid surrounding the resin and probably with the type of resins, 1000 is a reasonably conservative value.

To estimate the activity levels of the resins, the following assumptions are used:

The total activity of Cs-137 (the most significant cesium isotope) in the resins is the total that has leaked from the fuel during a complete cycle. For I-131 (the most significant iodine isotope), the activity present is that attributable to the maximum level found in the primary system volume during operation. It should be noted that these assumptions represent a situation which would not occur even under the most pessimistic operating conditions.

The activity specified above is distributed uniformly over the 100 ft³ of resins in purification demineralizers 1-1 and 1-2. Based on the resins being a chemically-equivalent mixture of cation and anion resins, the cesium is confined to about one third of the volume or about 33 ft³ of resin (this is based on the assumption that the cation portion of the resin has twice the capacity of the anion portion). The iodine will, therefore, be confined to about 67 ft³ of resin.

Using the preceding assumptions and the value for K_D noted earlier, the maximum concentrations of I-131 and Cs-137 in the sluice water are estimated to be:

Failed-Fuel Condition	Radio-nuclide	Est. Conc. in Sluice Water ($\mu\text{Ci/ml}$)	Max. Conc. in Reactor Coolant ($\mu\text{Ci/ml}$)
0.1%	I-131	0.040	0.323
	Cs-137	0.952	1.27
1%	I-131	0.40	3.23
	Cs-137	9.52	12.7

Included in the above table are the maximum concentrations of Cs-137 and I-131 that would be attained in the reactor coolant during cycles with 0.1% and 1% failed fuel. It is noted that these values are all higher than the corresponding concentrations found in the sluice water. There is no reason to believe that other radioactive dissolved species present in the waste would behave in a significantly different manner than those considered above. It is conservatively assumed therefore, that the concentrations of all dissolved radionuclides in the sluice wastes are the same as those in the primary system.

Estimating levels of insoluble corrosion products in the sluice water is more difficult. These particles are removed from process streams not only by the filters but also by demineralizers. If the filters are backwashed or the resins sluiced, some of the crud that has been collected is dislodged and mixes with the waste water created. It is not known how much becomes dislodged or how much settles out and is disposed with the resins. Thus, the following gross assumptions have been made in order to arrive at some kind of an estimate.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

The purification demineralizer filter and/or the resins of the purification demineralizers remove all crud that passes in series through them. During a simultaneous backwashing of the filter and sluicing of the resins, all the crud thus collected is released to the waste water.

The purification demineralizer filter and purification demineralizer can continuously process 45 gpm reactor coolant per day for a full 346-day operating cycle. The processed coolant contains the crud levels given in Table 11.1-7 (corrected for temperature).

Crud levels in the sluice water are based on the activity removed in a 346-day cycle being mixed with 500 ft³ of liquid. The 500 ft³ of liquid is assumed to be the minimum waste volume that would be considered for batch processing.

Based on the preceding assumptions, Table 11.2-8 lists conservative estimates of the maximum crud activity levels in the sluice waste.

- b. Estimates of average specific activities are based on the assumptions listed above except that the crud is assumed to be uniformly distributed over the entire volume of sluice waste (10,000 gal.).

Condensate Demineralizer Backwash

Estimates of waste activity levels have been made based on the following assumptions:

The primary-to-secondary leak is 100 gallons of reactor coolant per day.

All ionic activity in the leakage is absorbed on the 12 ft³Q of powdered resin contained in a single demineralizer (as with the regeneration waste, the corrosion products are not considered).

The subject demineralizer is backwashed after two weeks service. During this operation, 1000 ft³ of waste is produced.

The 1000 value for K_D used to estimate activity levels in the demineralizer sluice waste applies.

There is no carryover of resin particles.

Based on the above, the activity levels in the backwash waste are estimated to be about 0.0156 of those in reactor coolant. Maximum and average concentrations correspond to 1% and 0.1% failed fuel conditions, respectively.

Combined Sources:

Estimates of the maximum and average concentrations which are applicable to the non-detergent waste as a whole are based on the following assumptions.

The maximum specific activity of any radionuclide is considered to be the highest of the maxima estimated for the individual sources.

The overall average radionuclide concentrations are simply volume-weighted averages of the individual averages for various waste sources. For calculational purposes two different estimates were made. The first takes into account all waste sources except the condensate polishing demineralizer backwash. These values are used where higher average concentrations give more conservative results. Due to the low activity levels in the backwash and the large quantities involved, the inclusion of this waste would tend to lower the average. Also, if there were no significant primary-to-secondary leakage, the backwash waste would not even require processing. The second set of average values take into account all waste sources and is used primarily to estimate total station releases.

This ends the historical information contained in this subsection.

11.2.2.2.2 Detergent Waste

This type of waste comes from the hot showers (used to decontaminate personnel) and drains in the laboratory.

The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

Sources are listed in Table 11.2-5 along with the quantities that are expected from each in a year. Contamination present in these streams has probably resulted from operating personnel coming into contact with some form of transferable radioactivity. The amount of total contamination involved is small. It is assumed that the gross average specific activity in the detergent waste is of the order of magnitude of 10^{-5} $\mu\text{Ci/ml}$.

Secondary System Leakage

This waste results from liquid leakage from the secondary coolant system. An estimate of the total annual quantity resulting from this source is given in Table 11.2-5. All radioactivity present is assumed to come from the primary system by way of leaks in the steam generator. Estimates of the maximum and average activity levels for this class of waste are given in Table 11.2-9. These values are based on the assumptions used in Subsection 11.1.3.2, Parts (e), (f), and (g) and the following definitions:

Maximum Specific Activities: Equal to the maximum levels attained in leakage upstream of the condensate polish demineralizer.

Average Specific Activities: Equal to a volume-weighted average of the average specific activities of all liquid secondary leakage.

This ends the historical information contained in this subsection.

11.2.2.2.3 Tritium

Note: The Primary Water System has been abandoned in place.

The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

Tritium, in the form of tritiated water, is treated separately because, at this time, there is no feasible way of removing it from the waste through processing. For this reason, recycling of processed waste could result in the build up of tritium in primary water to the point where the population dose resulting from in-station exposure (ref. 14) would exceed the dose resulting from all the tritium being released to the environment. Such a buildup is retarded by the effects of radioactive decay and by the dilution of the primary system water during refueling operations. However, to conform to the philosophy of limiting doses to low levels, essentially all of the tritium present in the reactor coolant is assumed to be released from the station. For an equilibrium cycle, this amounts to about 346 Ci per year. The maximum rate of release is governed by the requirement to comply with the standards given in 10CFR20.

For calculational purposes, it is assumed that the tritium level in the primary system is maintained at, or below, the 0.4 $\mu\text{Ci/ml}$ level. This corresponds to a maximum concentration of 0.3 $\mu\text{Ci/ml}$ in the borated water storage tank.

This ends the historical information contained in this subsection.

11.2.2.3 Equipment Decontamination Factors

The decontamination factor (DF) for a piece of process equipment is defined as:

$$\text{DF} = \frac{\text{Radionuclide Concentration in Equipment Influent}}{\text{Radionuclide Concentration in Equipment Effluent}}$$

The DF's assigned to the various radwaste system components, as well as the basis for their selection, are given below.

11.2.2.3.1 Evaporator

The following information pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design. The ODCM provides information pertaining to the monitoring of waste streams and ensures the resultant offsite doses are in compliance with current regulatory requirements. The following DF information is considered historical.

In dealing with evaporators, two different decontamination factors are utilized. The first follows the format used above:

$$\text{DF}(t) = \frac{\text{Radionuclide Concentration in Evaporator Influent } (t)}{\text{Radionuclide Concentration in Evaporator Distillate } (t)}$$

where “(t)” denotes time dependence. This is the one that is used in the radwaste system analysis. The second definition is:

$$DF^*(t) = \frac{\text{Radionuclide Concentration in Evaporator Bottoms (t)}}{\text{Radionuclide Concentration in Evaporator Distillate (t)}}$$

This is the one used most often in describing the efficiency of an evaporator. One reason for this is that, for a given set of operating conditions, it should remain relatively constant. The relationship between these two definitions, assuming constant influent activities, is given by:

$$DF(t) = \frac{DF^*}{\text{Concentration Factor (t)}}$$

where:

$$\text{Concentration Factor} = \frac{\text{Bottoms Concentration (t)}}{\text{Influent Concentration (t)}}$$

or, if no bottoms have been discharged up to time “t”:

$$\text{Concentration Factor} = \frac{\text{Volume of Influent Processed Up to Time “t”}}{\text{Volume of Bottoms (t)}}$$

In establishing the efficiency of the Davis-Besse evaporators, no credit is taken for the removal of gases or tritium (i.e. $DF=1$). To obtain estimates for DF 's of other radionuclides, constant concentration factors are assumed. DF^* 's were determined from test data obtained at Rochester Gas and Electric's Robert Ginna Nuclear Generating Station (ref. 15). Using this approach, the following are the derivations of DF 's for the evaporative processing of both clean and miscellaneous (dirty) liquid wastes.

Clean Liquid Waste

a. Minimum DF

In the Ginna Tests run with reactor coolant and no chemical additions, a DF^* greater than 2×10^5 was achieved for Cs-137 and Na+. This value is, of course, applicable to all cesium isotopes. Since, with the exception of the iodines, the other ionic nuclides and corrosion products listed in Tables 11.2-4 and 11.1-7 can be expected to act reasonably similarly to cesium and sodium in an evaporative process, the 2×10^5 DF^* is also considered applicable to them. It should be noted, that, in an actual test with one of the Davis-Besse evaporators, a DF^* in excess of 10^6 was achieved for magnesium.

For I-131, the selection of a reasonable minimum value of DF^* was more difficult. The reason for this was that the values obtained in the Ginna Tests varied somewhat erratically with the pH of the evaporator bottoms. The approach taken was to adopt the use of the lowest DF^* reported for I-131, 3×10^3 . This value is, of course, the same for all iodine isotopes.

For clean liquid wastes, a maximum concentration factor of 100 is assumed. Based on a 950-gallon evaporator bottom volume, this amounts to the

Davis-Besse Unit 1 Updated Final Safety Analysis Report

processing of about 95,000 gal of feed. This corresponds to almost one clean waste receiver tank full of waste. The 100 concentration factor should be conservative, since it is not likely that this amount would be processed in a single batch. Even if it were, two evaporators would probably be used which, by doubling the bottom volume available, would have the effect of halving the concentration factor.

Since boric acid is being recovered from the clean liquid waste for reuse, the concentrate produced by the evaporator must average 7%. If feed runs out and the bottoms are not at this concentration, the evaporator can be shut down and the bottoms stored until more feed is available, or it can be run without feed until the 7% value is obtained. If this latter mode of operation is used, the DF can drop below acceptable limits, as the volume decreases and the concentration factor goes up. If this starts to occur, the operator can recycle the distillate back to the receiver tank and prevent any short-term deterioration in the evaporator DF. It should be noted that this procedure can be used at any time to limit the increase of the concentration factor.

Using the values for DF*'s and concentration factor just described, the following estimates of DF's were made for an evaporator handling clean liquid wastes:

Iodine.....	30
Crud.....	2000
Tritium.....	1
Gases.....	1
Others.....	2000

b. Average DF

The determination of average DF's is assumed to differ from that of minimum DF's only in the values of the concentration factor used. In this instance, the concentration factor is taken to be 50. (This would be the average in a system attaining a maximum value of 100.) The previously selected DF values of 2×10^5 and 3×10^3 for Cs-137 and I-131, respectively, are assumed. Although it should be possible to use a larger number for the average DF*'s, it is felt that there is not sufficient information available for making a reasonable estimate. Based on these assumptions and on the reasoning given in the preceding paragraphs, estimates of average DF's for an evaporator handling clean liquid waste are as follows:

Iodine.....	60
Crud.....	4000
Tritium.....	1
Gases.....	1
Others.....	4000

Miscellaneous Liquid Waste (abandoned in place)

a. Minimum DF

In the Ginna test, DF*'s of about 10^5 for Na+ and Cs-137 and 10^4 for I-131 were reported in a run made with general waste to which chemicals had been added

Davis-Besse Unit 1 Updated Final Safety Analysis Report

for pH control. These values are applicable to all cesium and iodine isotopes. As with the clean liquid waste, the 10^5 figure was applied to the other ionic radionuclides and corrosion products listed in Tables 11.2-6 and 11.2-7.

For miscellaneous waste, a concentration factor of 50 is assumed. While it is possible that larger factors could be achieved, this value should be conservative. Based on a bottom volume of 950 gallons, it represents the processing of about 47,500 gallons or more than 3-1/2 miscellaneous waste drain tanks full of waste. Operationally, it seems unlikely that there would be a need to process this much in a single batch.

If the solids content of the waste feed is low, the evaporator bottoms may not reach their normal disposal concentration before a run is terminated. In these circumstances, the unit can simply be operated without feed until the desired concentration, or some minimum volume, is attained. During this period, due to the decreasing bottom volume, the concentration factor can increase beyond acceptable limits. As with the case with clean liquid waste, when this happens, the operator can recycle the distillate back to the feed tank and prevent any deterioration of the DF's.

Using the values for DF*'s and concentration factors, just selected, the following estimates of DF's for an evaporator processing miscellaneous waste were made:

Iodine.....	200
Crud.....	2000
Tritium.....	1
Gases.....	1
Others.....	2000

b. Average DF

The determination of the average DF's is assumed to differ from that of the minimum DF's only in the choice of a concentration factor. In this instance, a concentration factor of 25 will be used. This would be the average in a system attaining a maximum value of 50. Although the DF*'s varied during the test run using the new concentration factor and the DF*'s previously selected, the following estimates were made of average DF's for the evaporator:

Iodine.....	400
Crud.....	4000
Tritium.....	1
Gases.....	1
Others.....	4000

This ends the historical information contained in this subsection.

11.2.2.3.2 Demineralizers (Radwaste System)

All radwaste system demineralizers are designed to provide desirable operating characteristics. In addition, the following features have been incorporated into the system to further ensure that all units perform adequately.

Connections are provided to take samples from all demineralizer influent and effluent lines.

Where redundant demineralizers are provided in the system design, the capability is generally available to operate them in series. This arrangement permits the downstream exchanger to polish the effluent from the other unit and to serve as a backstop in case the resin bed in the upstream demineralizer becomes saturated and breakthrough occurs.

The demineralizers used to polish the distillates from the evaporators are provided with recycle lines back to the evaporators. This permits the units to be flushed when they are being put into service after being shut down for any extended period. Flushing prevents the liquid initially contained in the demineralizer, which may be excessively contaminated, from being passed on to the monitor tanks.

The following information pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design. The ODCM provides information pertaining to the monitoring of waste streams and ensures the resultant offsite doses are in compliance with current regulatory requirements. The following DF information is considered historical.

Assuming that the above design features are used as needed and that proper operating procedures are also utilized, the following are the DF's that were assigned to the various types of demineralizers:

Mixed Bed Demineralizer.

The DF's assigned to mixed-bed radwaste demineralizers for individual ions are as follows:

For Each Ion in Table 11.1-4	
Prior Cumulative Demineralization DF Assumed*	DF for Subsequent Single Radwaste Mixed-Bed Demineralizer
1 (crud, gases, untreated streams)	100**
≥100 (treated streams, ionic radionuclides)	10

* includes DF associated with purification demineralizers

** value always used for concentrate demineralizer

An exception to the preceding table is the concentrate demineralizer which is assigned a DF 100 regardless of what comes before it.

a. Ionic Radionuclides

At present, there are no generally accepted values for ionic radionuclide DF's for use with mixed-bed demineralizers. The justification for using 100 as a maximum is based on values found in the literature. A selected list of these, applicable to Cs-137 and I-131, is given in Table 11.2-10. This table does not contain an entirely representative sampling as there are instances where DF's

lower than 100 have been reported. It is also noted that the operating conditions under which measurements were made varied considerably. All this makes arriving at a definitive conclusion very difficult, if not impossible. Still, there does seem to be ample evidence that, with proper design and operating conditions, DF's for cesium and iodine of at least 100 should be achievable. This value is also used for the other ionic radionuclides in Table 11.1-4 although information on the removal of these is limited and generally inconclusive.

The use of a DF of 10 after credit has already been taken for a cumulative DF of at least 100 is employed to take into account any effect the reduction of influent ion concentration might have on a demineralizers' efficiency. This is also the justification for always using a DF of 100 for the concentrate demineralizer. Even though prior credit may have been taken for demineralization, the feed to this unit is concentrated in the boric acid evaporators. This increases the concentrations of the impurity ions, making them easier to remove.

b. Crud and Gaseous Radionuclides

No credit is taken for the removal of crud or gases in the demineralizers (i.e. DF=1). However, as noted when discussing the DF's chosen for filters, demineralizers are efficient at removing crud.

Anion Demineralizers

a. Ionic Radionuclides

The only anion demineralizers in the system are the deborating units used to remove boric acid from the reactor coolant during the latter stages of a fuel cycle. No credit is taken for the removal of any radionuclide, although, from Table 11.2-10, it is seen that iodine levels should be reduced. These anion units are assumed to reduce the level of boron in their effluent to 1 ppm or less. Under certain conditions, higher concentrations may be allowed. The basis for these performance criteria is found in the results of tests published by one of the resin manufacturers (ref. 26).

b. Crud and Gaseous Radionuclides

No credit is taken for the removal of crud or gases (i.e. DF=1).

Cation Demineralizer

a. Ionic Radionuclides

Purification demineralizer 1-3 is normally utilized as a cation demineralizer. It plays no part in the analysis of radioactive releases from the station because no credit is taken for its use. Though it may also be used as a mixed bed demineralizer, it is normally used to reduce the concentration of lithium hydroxide as well as radioactivity levels of Cs-137. For this use, a DF of 20 for Cs-137 and one for all other radionuclides has been assumed. This conservative value of 20 for a cation bed is also supported by Table 11.2-10.

This ends the historical information contained in this subsection.

11.2.2.3.3 Demineralizers (Secondary System)

The following information pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design. The ODCM provides information pertaining to the monitoring of waste streams and ensures the resultant offsite doses are in compliance with current regulatory requirements. The following DF information is considered historical.

The only units considered in this category are the mixed-bed condensate polishing demineralizers. These differ from those of the previous section in that they use powdered rather than bead-type ion exchange resins. At this time, there appears to be little data available on the efficiencies of these units in removing radioactive impurities. The only reference located said that DF's of 2-50 for solubles and 100-1000 for insolubles had been measured when treating condensate (ref. 27). Based on this limited information, the following DF's were assigned:

Ions.....	5
Crud.....	5
Gases.....	1

This ends the historical information contained in this subsection.

11.2.2.3.4 Filters

The following information pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design. The ODCM provides information pertaining to the monitoring of waste streams and ensures the resultant offsite doses are in compliance with current regulatory requirements. The following DF information is considered historical.

A total DF of 10 is assumed for corrosion products (crud) for any number of passes through radwaste system filters. A DF of 1 is used for all other impurities.

Filtration ratings for the Davis-Besse filters are given in Tables 11.2-1 and 11.2-2. It has been reported that demineralizers are good filters and their efficiency for removing crud has been given as 89 to 99% (corresponding to DF's of approximately 10 to 100) (ref. 28). It should be very conservative to assume a DF of 10. A DF of 1 is assigned to the filters. Additional conservatism is added by the fact that no credit is taken for multiple passes through more than one unit.

This ends the historical information contained in this subsection.

11.2.2.3.5 Degasifier (No longer used)

The following information pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design. The ODCM provides information pertaining to the monitoring of waste streams and ensures the resultant offsite doses are in compliance with current regulatory requirements. The following DF information is considered historical.

The degasifier is assumed to have a DF of 10^5 for Xe and Kr and 1 for all other radionuclides (the possible removal of iodine is covered in Subsection 11.3.2.3). At present, there appears to be no data to support these or any other values. However, a theoretical DF of more than 10^6 has been calculated by the equipment vendor.

This ends the historical information contained in this subsection.

11.2.2.3.6 Waste Demineralizer System

Miscellaneous radioactive waste in the Miscellaneous Waste Drain Tank is processed through demineralizers, and discharged from the station from the Miscellaneous Waste Monitor Tank. The Demineralizer Skid consists of a solids collection filter, a booster pump, pre-filter, a post-filter and various demineralizers (with a reverse osmosis system) that can be lined up in a number of modes depending on the specific characteristics of the liquid waste to be processed. A typical system configuration consists of the filter, an activated carbon bed, zeolite, a cation bed, an anion bed, and a mixed resin bed. For a typical system configuration the nominal DF's are as follows:

The following information pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design. The ODCM provides information pertaining to the monitoring of waste streams and ensures the resultant offsite doses are in compliance with current regulatory requirements. The following DF information is considered historical.

Tritium and Noble Gases	1
Iodine	1000
Cesium	200
Others	700
Overall System	200

This ends the historical information contained in this subsection.

11.2.3 Operating Procedures

11.2.3.1 Clean Liquid Radioactive Waste System

Hydrogenated liquid waste pumped from the reactor coolant drain tank and letdown from the makeup and purification system are directed into a primary demineralizer filter and through a primary demineralizer into one of two clean waste receiver tanks. The clean waste receiver tank not being used for letdown and liquid waste collection can be filled with a clean boric acid solution for use during Reactor Coolant System cooldown. The piping and valves of the clean liquid radioactive waste system that are not heat traced are flushed with demineralized water after the concentrated boric acid is transferred into the clean waste receiver tank. Periodic or continuous recirculation of the contents of the Clean Waste Receiver Tank provides mixing.

When it is desired to process the Clean Waste Receiver Tank(s), the waste is fed to one, or both, of the boric acid evaporators. There, it is separated into its two reusable constituents: demineralized water and concentrated boric acid. The demineralized water (distillate) may be pumped through or around a polishing demineralizer and a clean waste monitor tank filter into one of the clean waste monitor tanks. Individual recirculation lines on these tanks ensure

uniform mixing. When the receiving clean waste monitor tank is full, it is isolated and a sample taken. Based on the analysis of this sample, the processed waste is either: (1) sent to the makeup system for eventual reuse; (2) recycled through all, or part, of the system for additional treatment; or (3) discharged to Lake Erie. The boric acid from the evaporator bottom may be pumped around or through the concentrate demineralizer into the concentrate storage tank. After the contents of this tank have been thoroughly mixed by recirculation, a sample may be taken and analyzed. The acid is then pumped either to one of the boric acid addition tanks, where it is stored for reuse, or to the miscellaneous liquid radioactive waste system, where it is further processed and discharged to Lake Erie. There is also a line that permits the direct addition of boric acid from the concentrate storage tank to the makeup tank if this is required. Gasses stripped from the waste in the boric acid evaporators are either sent to the waste gas system or discharged through the station vent.

When the concentration of boron in the primary system is low (such as near the end of core life), coolant letdown from the makeup and purification system may be diverted through one of the deborating demineralizers to remove boric acid, instead of being processed through an evaporator. From there, it is normally directed back to the primary system through the makeup tank.

If, at some point in the normal process cycle, sampling indicates that either treatment in the clean radwaste system has been inadequate or that no more is needed, alternate flow paths can be used. These include recycle lines which, by permitting the contents of the various system tanks to be redirected back through some portion of the cycle, are used to effectively increase the processing performed on the waste. (The most important of these lines are given in Table 11.2-11 along with the processing capacity of each.) There are also bypasses provided around all major components with the exception of the clean waste monitor tanks. These bypasses are used if a particular piece of equipment is not needed or is not available for service. The alternate processing to be performed is determined by the operator and is based primarily on the analysis of samples taken from the system tanks.

All waste tanks, except the Concentrates Storage Tank and the Miscellaneous Waste Drain Tank, are maintained above atmospheric pressure by a nitrogen blanketing system. When a tank is filled, excess cover gas is displaced to the gaseous radioactive waste system or to the station vent. When this tank is emptied, any replacement gas needed is recycled back from the waste gas system or supplied from the station nitrogen header.

All wastes being discharged to the environment pass two monitors (one required for normal system operation) that measure and record radiation levels and that control two downstream valves in the flow path. If either of these monitors detects excessive radiation, both downstream valves are closed and flow is terminated.

11.2.3.2 Miscellaneous Liquid Radioactive Waste System

During station operation, dirty and aerated wastes from various sources are collected in tanks as follows:

Davis-Besse Unit 1 Updated Final Safety Analysis Report

<u>Waste Sources</u>	<u>Collection Tank</u>
Laboratory and sampling Demineralizer sluice	
Area Washdowns	
Misc. system leakage	Miscellaneous waste drain Tank
Lab sinks	Detergent waste drain Tank
Hot Showers (decon)	
Condensate polishing demin. backwash (only if there is a significant steam generator leak)	Condensate demin. Holdup tanks (2)
Secondary sys. leakage (only if there is a significant steam generator leak)	Condensate demin. holdup tanks (2)

When a sufficient level is reached in the Miscellaneous Waste Drain Tank or the Detergent Waste Drain Tank, the contents are normally pumped through the Radwaste Demineralizer System for processing. The Radwaste Demineralizer System uses installed demineralizers with an integrated reverse osmosis system. Here suspended impurities are filtered and soluble ionic impurities are removed by ion exchange. However, if a sample is taken (after the tank has been recirculated to ensure thorough mixing), and analysis shows no significant radioactivity, processing through the Radwaste Demineralizer System is not required. The waste then flows through the demineralizer skid post filter, if using the Radwaste Demineralizer System, and into the Miscellaneous Waste Monitor Tank. Liquid from the Detergent Waste Drain Tank may also be released to the environment if sample results indicate acceptable levels for direct discharge.

When a sufficient level is reached in either one of the Condensate Demineralizer Holdup Tanks, the liquid contents of the tanks are directed to the Miscellaneous Liquid Radioactive Waste System for further processing unless sampling analysis shows no significant radioactivity. If no significant radioactivity is present, the contents may be released to the environment.

When the Miscellaneous Waste Monitor Tank is filled and after recirculation has ensured proper mixing, a sample is taken and analyzed. Based on the results of this sample, the waste is then either recycled through part or all of the system for additional processing, or diluted in a controlled manner and released from the station. Connections to the clean liquid radioactive waste system were provided for possible reuse of water from the miscellaneous waste system, but the pathway is not used due to contamination concerns. For flexibility, bypasses are provided around all major components with the exception of the monitor tank. These bypasses are used if a particular piece of equipment is not available for service or if, based on the results of sampling, the operator decides it is not needed.

The gas inlets and/or vents on the miscellaneous waste drain, detergent waste drain, and miscellaneous waste evaporator storage tanks are controlled to reduce the turnover of gases in the tank's vapor spaces. As the evolution of gaseous iodine from liquid waste is an equilibrium process, this tends to keep iodine liberation to a minimum. The miscellaneous waste monitor tank has a nitrogen feed arrangement that can be used as a purge to maintain a slight positive pressure.

The miscellaneous waste drain tank's nitrogen blanketing system has been disabled. This tank is equipped with a vacuum breaker to prevent the drawing of a vacuum when this tank is emptied.

All waste being discharged to the environment passes two redundant monitors (one is required for system operation) that measure and record the radiation level and that control two downstream valves in the flow path. If either of these monitors detects excessive radiation, both downstream valves are closed and flow is terminated.

11.2.4 Performance Tests

All pieces of processing equipment in these systems are provided with upstream and downstream sampling connections. Using the analysis of samples taken at these points, the actual operating DF of any component for any or all of the various radionuclides can be determined. In a more general way, the sampling done at various points in the waste processing cycle will give a good indication of how well the individual pieces of equipment are operating.

11.2.5 Estimated Releases

11.2.5.1 Clean Liquid Radioactive Waste System (Excluding Tritium)

11.2.5.1.1 Maximum Isotopic Specific Activities in Releases

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Estimates of the maximum specific activities in waste discharged to the environment from this system are based on the following assumptions:

- a. The waste feed processed by this system initially has maximum activities given in Table 11.2-4.
- b. DF's for the process cycle equipment are those given in Figure 11.2-4. As maximum specific activities are being estimated, the minimum evaporator DF is assumed.
- c. The minimum estimated specific activity for any radionuclide in undiluted processed waste is set at 10^{-8} $\mu\text{Ci/ml}$.
- d. No credit is taken for the effects of decay.
- e. Before being discharged, treated waste is diluted with uncontaminated water. Based on a diluting flow of 20,000 gpm, a treated waste release rate of 35 gpm and a mixing coefficient of 0.8 (see Subsection 11.2.7.1 for justification of this value), this results in an effective DF of 456 for all contaminants. It should be noted that 70 gpm is the normal release rate from this system. Lower rates can be used during abnormal situations such as the 1% failed fuel condition specified in assumption (a).

Based on the preceding assumptions, Table 11.2-12 was compiled and gives the maximum specific activities in the waste at significant points in its treatment up to and including release to the environment.

This ends the historical information contained in this section.

11.2.5.1.2 Average Isotopic Specific Activities in Releases

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Table 11.2-13 gives the average specific activities in the waste at several points during its treatment up to and including release to the environment. These estimates were based on the same assumptions that were used in the preceding section with the following differences:

- a. The waste feed processed initially has the average isotopic specific activities given in Table 11.2-4 (these correspond to 0.1% failed fuel).
- b. DF's for the process cycle equipment are those given in Figure 11.2-4. As average specific activities are being estimated, the average evaporator DF's are assumed.
- c. The processed waste is discharged into the dilution flow at a 70 gpm rate.

It should be noted that these "average" values take into account only those effluents produced when processed waste is actually being discharged. This flow amounts to only a portion of the total liquids that are released annually from the station. This means that the overall average of radionuclide specific activities in released liquids would be less than the values given in Table 11.2-13. These overall averages were not estimated because they do not provide any significant information about the adequacy of the clean liquid radioactive waste system or treating waste water.

This ends the historical information contained in this subsection.

11.2.5.1.3 Total Annual Activity Released

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

The total activity attributed to each radionuclide that is expected to be released in a year through this system is given in Table 11.2-14. These values are based on the following assumptions:

- a. The specific activities of the various radionuclides in undiluted processed waste are those given in Table 11.2-14

- b. The amount of processed waste from this system discharged annually from the station will be 258,000 gallons. A breakdown of that portion of the total waste that is released is given in Table 11.2-3

For operation of the reactor for the whole year with 1% failed fuel, the values in Table 11.2-14 (except for activation products) would be increased by a factor of 10. This is not an expected operating condition but represents an upper design condition.

This ends the historical information contained in this subsection.

11.2.5.2 Miscellaneous Liquid Radioactive Waste System (Excluding Tritium)

11.2.5.2.1 Maximum Specific Activities in Releases

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Estimates of the maximum specific activities in waste discharged to the environment from this system are based on the following assumptions:

- a. The waste processed by this system initially has the activity levels given in the last column of Table 11.2-6. These values correspond to estimates of the maximum concentration of each radionuclide that can occur. (Note: although secondary system leakage was not taken into account when Table 11.2-6 was compiled, it would have no effect on the choice of overall maximum activity levels.)
- b. DF's for the process cycle equipment are those given in Figure 11.2-5. As maximum specific activities are being estimated, the minimum evaporator DF is assumed. (Note: This assumption pertains to equipment originally installed in the plant. Due to efficiency reasons the waste evaporator has been replaced by the radwaste demineralizer.)
- c. The minimum estimated specific activity for an isotope in undiluted processed waste is set at 10^{-8} $\mu\text{Ci/ml}$.
- d. No credit is taken for the effects of radioactive decay.
- e. Before being discharged, treated waste is diluted with uncontaminated water. Based on a dilution flow of 20,000 gpm, a treated waste release rate of 16 gpm and a mixing coefficient of 0.8 (see Subsection 11.2.7.1 for justification of this value), an effective DF of 1000 results for all contaminants. It should be noted that 70 gpm is the normal release rate from this system. Lower rates can be used, as needed, during abnormal situations such as the very conservative condition specified in assumption (a).

Based on the preceding assumptions, Table 11.2-15 was compiled and gives the specific activities in the waste at significant points in its treatment up to, and including, release to the environment. It should be noted that the chance of the maximum levels of all isotopes being

present at one time is very remote. If the situation depicted in Table 11.2-15 were to occur, it is likely that additional processing would be used.

This ends the historical information contained in this subsection.

11.2.5.2.2 Average Specific Activities in Release Due to Non Detergent Wastes

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Table 11.2-16 and 11.2-17 give the average specific activities in the waste at several points during its treatment up to, and including, release to the environment. These values were arrived at by using the same assumptions as in Subsection 11.2.5.2.1 with the following exceptions:

- a. The waste shown in Table 11.2-16 will initially have the radionuclide activity levels given in Column (5) of Table 11.2-7.
- b. The waste shown in Table 11.2-17 will initially have the radionuclide activity levels given in the last column of Table 11.2-7.
- c. As average specific activities are being estimated, the average evaporator DF's are used. (evaporator is no longer used)
- d. Treated waste is discharged at a rate of 70 gpm.

11.2.5.2.3 Average Specific Activities in Releases Due to Detergent Wastes

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

No breakdown has been attempted for activity releases due to detergent wastes. However, based on an initial gross specific activity of about 10^{-5} $\mu\text{Ci/ml}$ (See Subsection 11.2.2.2.2), a release rate of 70 gpm, a dilution flow of 20,000 gpm, and a mixing coefficient of 0.8, the specific activity in the discharged waste should be less than 10^{-8} $\mu\text{Ci/ml}$.

This ends the historical information contained in this subsection.

11.2.5.2.4 Average Specific Activities in Releases Due to Secondary System Leakage

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Table 11.2-18 lists the average specific activities in the station liquid discharges to the environment that are due to secondary system leakage. These estimates were derived using the following assumptions:

- a. Untreated secondary system leakage (as radionuclide activity levels) is equal to the average values listed in Table 11.2-9.
- b. This waste receives no active treatment. It is simply diluted and released from the station. (Note: the capability exists to process part or all of this waste through the miscellaneous liquid radwaste system.)
- c. No credit is taken for the effects of radioactive decay.
- d. Dilution is based on a dilution flow of 20,000 gpm, a waste release rate of 70 gpm, and a mixing coefficient of 0.8.

This ends the historical information contained in this subsection.

11.2.5.2.5 Total Annual Activity Released Due to Non-Detergent Waste

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

The total activity attributed to each radionuclide that is released in a year through this system due to non-detergent waste is given in Table 11.2-19. These values are based on the following assumptions:

- a. The specific activities of the various radionuclides in undiluted processed waste are those given in Column 4 of Table 11.2-17.
- b. The amount of processed waste from this system discharged annually from the station will be approximately 200,000 gallons. A breakdown of this quantity is given in Table 11.2-5.

For operation of the reactor for the whole year with 1% failed fuel, the values in Table 11.2-19, except those for activation products, would be increased by approximately a factor of 10. This is not an expected operating condition but represents an upper design condition.

This ends the historical information contained in this subsection.

11.2.5.2.6 Total Annual Activity Released Due to Detergent Waste

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Based on an average gross specific activity of 10^{-5} $\mu\text{Ci}/\text{ml}$ and a total release of about 500,000 gallons, an estimated 1.9×10^{-2} Ci is discharged per year in detergent waste.

This ends the historical information contained in this subsection.

11.2.5.2.7 Total Annual Activity Released Due to Secondary System Leakage

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

The total activity attributed to each radionuclide that is released in a year is given in Table 11.2-18. These values are based on the following assumptions:

- a. The specific activities in undiluted secondary system leakage are the average values given in Table 11.2-9.
- b. The amount of leakage waste released from the station in a year is 2,102,400 gallons (5 gpm for 292 days).

If the reactor were operated an entire year (equilibrium cycle) with a 1% failed fuel condition, all values in Table 11.2-18, except those for activation products, would increase by a factor of about 10.

This ends the historical information contained in this subsection.

11.2.5.3 Total Annual Releases of Tritium from All Sources

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

As noted in Subsection 11.2.2.2.3, it is assumed that all tritium leaking into or produced in the reactor coolant is released to the environment. For an equilibrium fuel cycle with 0.1% failed fuel, this amounts to about 346 Ci per year (see Subsection 11.1.2.2). For a 1% failed fuel condition, estimated tritium releases increase to 443 Ci per year.

This ends the historical information contained in this subsection.

11.2.5.4 Summary of all Activity Released Each Year in Liquid Effluent

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Table 11.2-20 lists the total activity of each radionuclide that is released from the station each year in liquid effluents. Included is a breakdown of the contributions made from each of the several individual sources to these totals. Due to the relatively small quantity of activity involved and to the indeterminate nature of its composition, the contribution from detergent waste (5.87×10^{-3} Ci gross γ - β) was omitted.

This ends the historical information contained in this subsection.

11.2.6 Release Points

Processed liquid waste enters Lake Erie as shown on Figure 11.2-6.

11.2.7 Dilution Factors

11.2.7.1 Prior to Discharge

Before processed water is released to the environment it is mixed in a collection basin with the discharge from either the Service Water System, the dilution pump, a cooling tower make up pump or the cooling tower blowdown.

The following information pertaining to dilution factors was provided as part of the initial safety analysis. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following dilution factor information is considered historical.

The total dilution flow is approximately 20,000 gpm. The dilution process is assigned a mixing coefficient of 0.8. The 0.8 mixing factor means that the concentrations in the estimated station releases will be calculated on the basis that the waste being discharged is uniformly mixed with 16,000 gpm of dilution flow. This figure is supported by the following design features of the system:

- a. The discharge from the cooling tower makeup pump and cooling tower blowdown have inlets into the collection basin located as close together as possible and at right angles to each other. As this arrangement will promote turbulence, mixing among all basin influent streams should be good.
- b. Additional mixing occurs as the contents of the collection basin flow to the lake.

This ends the historical information contained in this subsection.

11.2.7.2 Further Dilution in Lake Erie

The following information pertaining to dilution factors was provided as part of the initial safety analysis. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following dilution factor information is considered historical.

A modified Fickian dispersion model was used to predict maximum radioactivity concentrations at distances over one kilometer from the point of discharge (ref 29). These dispersion estimates have been used to determine the potential radiological impact of plant operation at the following locations that utilize Lake Erie as a source of potable water: Camp Perry, Erie Industrial Park,

Davis-Besse Unit 1 Updated Final Safety Analysis Report

Port Clinton, Toledo, and Oregon. Maximum potential plume centerline concentrations as a function of distance were estimated using the following equation (ref. 29):

$$\bar{C}(X, 0, 0) = \frac{f Q}{2\pi[X(K_y K_z)]^{1/2}}$$

where:

$\bar{C}(X, 0, 0)$ = average centerline (maximum) concentration at a point (X) downstream from release point $\mu\text{Ci}/\text{cm}^3$
 Q = continuous release source term, $\mu\text{Ci}/\text{sec}$
 X = downstream distance, cm
 K_y = diffusivity in lateral direction, cm^2/sec
 K_z = diffusivity in vertical direction, cm^2/sec
 and f = frequency of flow in the direction of interest

From Okubo and Farlow (ref. 30) for Lake Erie drogue studies, K_y exceeds 3×10^4 (cm^2/sec) in all cases. The same values were observed in the Lake Michigan studies. For conservatism, a value of 1×10^4 (cm^2/sec) was used.

A conservative estimate of K_z ($1 \text{ cm}^2/\text{sec}$) (ref. 31) was also used. Thus the equation expressing the plume centerline concentration becomes:

$$\bar{C}(X, 0, 0) = \frac{1.59 \times 10^{-3}}{X} fQ$$

It is assumed that the warm water discharge will rise to the surface of the lake and be transported in the same direction as the surface current. Further, the surface currents are assumed to depend on the prevailing wind direction in the following way: winds out of the E, ESE, SE, or SSE are all presumed to induce a surface current towards Toledo and Oregon; winds out of the NNW, N, NNE, NE and ENE are all presumed to induce a surface current towards Camp Perry and Port Clinton; all other winds are presumed to induce surface currents carrying the discharge large distances off shore.

Based on the available meteorological information in Section 2.3, the value of the parameter " f ", after accounting for calms, is assumed to be 0.17 towards Toledo and 0.21 towards Camp Perry and Port Clinton.

Effective dilution factors have been obtained for the locations of greatest concern by taking the ratios of the annual discharge concentration for tritium and the concentrations predicted by the model. At the average expected release rate of 346 curies per year ($11 \mu\text{Ci}/\text{sec}$) the annual average discharge concentration for tritium is 1.087×10^{-5} micro mCi/cc . The predicted tritium concentrations at the given locations and the derived dilution factors are presented below:

	<u>Location</u>	<u>H-3 Concentration</u>	<u>Effective Dilution Factor</u>
1.	Erie Industrial Park (3.6 mi)	$6.38 \times 10^{-9} \mu\text{Ci}/\text{cc}$	1700
2.	Camp Perry (4.6 mi)	$5.00 \times 10^{-9} \mu\text{Ci}/\text{cc}$	2170
3.	Port Clinton (8.6 mi)	$2.67 \times 10^{-9} \mu\text{Ci}/\text{cc}$	4070
4.	Toledo & Oregon (12 mi)	$1.54 \times 10^{-9} \mu\text{Ci}/\text{cc}$	7080
5.	Carroll TS water (3.0 mi)	$6.19 \times 10^{-9} \mu\text{Ci}/\text{cc}$	1750

These derived effective dilution factors are utilized to predict the annual average radioactivity concentrations at the potable water intakes of the given locations.

These results have been compared to dilution predictions based on thermal analysis done by D.W. Pritchard and presented in Appendix 2 G, Volume I, of the Supplement to the Environmental Report for Davis-Besse (ref. 32). This comparison has revealed no significant discrepancies at distances greater than about 1 km.

Since the volume flow rate of the station discharge is very small in comparison to the volume of the near shore waters, near-field dilution will be rapid and recirculation is not significant. Eddy currents in the western Lake Erie Basin are not expected to affect predicted concentrations due to the substantial mixing that occurs in these patterns.

This ends the historical information contained in this subsection.

11.2.8 Estimated Doses from Liquid Effluents

The following information pertaining to estimated doses was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated dose information is considered historical.

A detailed evaluation of the radiological impact to be incurred by radioactive liquid releases from the station is contained in Appendix 11A. Only a summary of the results of that evaluation will be presented here. As explained in Appendix 11A, resulting exposure rates have been evaluated for maximum expected release rates based on 0.1% failed fuel cladding, and for design basis release rates based on 1.0% failed fuel cladding.

The exposure pathways evaluated include two modes of external exposure and two modes of internal exposure. All maximum individual exposure rates have been conservatively based on the annual average discharge concentrations as presented in Table 11A.4-1 of Appendix 11A. External exposure to the maximum individual has been evaluated for immersion (swimming) in the station discharge, and for sunbathing along the nearest shoreline. It was assumed that the maximum individual would swim no more than 200 hours per year in the station discharge and sunbathe no more than 200 hours per year along the nearest shoreline. Only gamma radiation was assumed to contribute to whole body radiation. Beta radiation from external sources was assumed to irradiate only the outermost surface layers of the human body. Population exposure was not evaluated for external exposure pathways.

Internal exposure was evaluated both for water ingestion and fish ingestion. Maximum individual exposure was based on an assumed water ingestion rate of 1.2 liters per day and an assumed fish ingestion rate of 50 grams per day of edible tissue. Activity concentration in the ingested water and fish were based upon the estimated annual average discharge concentrations. The NUS computer code DOSCAL performed the calculations necessary to obtain the resulting exposure rates to the whole body and all other significant body organs. Population exposure due to water and fish ingestion was also estimated, for the years 1970, 1990, and 2010.

Figure 11.2-7 illustrates the location of Ohio Fishing District 1 with respect to the discharge canal.

A dilution factor of 5000, used in predicting the population exposure from fish ingestion was obtained using the equation given in Subsection 11.2.7.2. It is applicable for a direction into which the surface current is carried 10 % of the time, at a distance of five miles. The equation used to obtain this dilution factor is presented below:

$$DF \text{ (dilution factor)} = \frac{1.087 \times 10^{-5}}{\left[\frac{1.59 \times 10^{-3} Qf}{x} \right]}$$

where:

1.087×10^{-5} = the discharge concentration of tritium for expected release rates, $\mu\text{Ci/cc}$

Q = the point source continuous release rate, $\mu\text{Ci/sec}$

x = the down-plume distance, cm

f = the frequency of flow in the direction of interest.

At 346 Ci/yr, Q is equal to 10.97 $\mu\text{Ci/sec}$. At a distance of five miles, x is equal to 8.045×10^5 centimeters. With f equal to 0.1, the computed value of DF is equal to just over 5000. This dilution factor is felt to be conservative due to the assumption of relatively high frequency of flow (10 percent) and a relatively short average distance (5 miles) in comparison to the dimensions of Ohio Fishing District 1.

The dilution factor of 4070 at a distance of 8.6 miles is based on a frequency of flow of 21 %. It is the difference in the assumed frequency of flow which causes the apparent discrepancy in the resulting dilution factors. The higher frequency of flow assumed to exist along the shoreline is due to the assumption that winds out of several directions might produce surface currents along the shore.

A summary of the maximum individual and population exposures found to result from liquid radioactivity releases is presented in Tables 11.2-21 and 11.2-22. A full explanation of the derivation of these results is contained in Appendix 11A.

This ends the historical information contained in this subsection.

TABLE 11.2-1

Clean Liquid Radioactive Waste System Equipment List

a. Clean Waste Receiver Tanks

Quantity	2
Type	Vertical Cylindrical Flat-Bottomed
Volume, gal	103,000
Design Pressure, psig	15
Design Temperature, °F	200
Material	304 SS
Design Code	ASME III, Class C

b. Clean Waste Monitor Tanks

Quantity	2
Type	Horizontal Cylindrical
Volume, gal	23,200
Design Pressure, psig	15
Design Temperature, °F	150
Material	304 SS
Design Code	ASME III, Class C

c. Concentrate Storage Tank

Quantity	1
Type	Vertical Cylindrical
Volume, gal	950
Design Pressure, psig	15
Design Temperature, °F	200
Material	304 SS
Design Code	ASME III, Class C

d. Spent Resin Storage Tank

Quantity	1
Type	Vertical Cylindrical
Volume, gal	6,000
Design Pressure, psig	75
Design Temperature, °F	200
Material	304 SS
Design Code	ASME III, Class 3

e. DELETED

|

TABLE 11.2-1 (Continued)

Clean Liquid Radioactive Waste System Equipment List

f. Clean Waste Receiver Tank Transfer Pumps

Quantity	2
Type	Centrifugal
Rated Capacity, gpm	140
Design Temperature, °F	150
Material	316 SS
Design Code	Manufacturer's Standard

g. Clean Waste Monitor Tank Transfer Pumps

Quantity	2
Type	Centrifugal
Rated Capacity, gpm	140
Design Temperature, °F	150
Material	316 SS
Design Code	Manufacturer's Standard

h. Concentrate Transfer Pumps

Quantity	1
Type	Centrifugal
Rated Capacity, gpm	50
Design Temperature, °F	200
Material	316 SS
Design Code	Manufacturer's Standard

i. Spent Resin Tank Overflow Pump

Quantity	1
Type	Centrifugal
Rated Capacity, gpm	100
Design Temperature, °F	120
Material	316 SS
Design Code	Manufacturer's Standard

j. Spent Resin Transfer Pump

Quantity	2
Type	Diaphragm, Air Powered
Rated Capacity, gpm	0 – 100
Design Temperature, °F	150

TABLE 11.2-1 (Continued)

Clean Liquid Radioactive Waste System Equipment List

k. Clean Waste Booster

Quantity	2
Type	Centrifugal, Canned
Rated Capacity, gpm	140
Design Temperature, °F	150
Material	316 SS
Design Code	Manufacturer's Standard

l. DELETED

m. Primary Demineralizers

Quantity	2
Type	Mixed Bed, Non-regenerative
Design Flow, gpm	140
Usable Resin Volume, ft ³	100
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Vessel Material	304 SS
Vessel Design Code	ASME III, Class 3

n. Deborating Demineralizers

Quantity	2
Type	Anion, Regenerative
Design Flow, gpm	140
Usable Resin Volume, ft ³	55
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Vessel Material	304 SS
Vessel Design Code	ASME III, Class 3

o. Clean Waste Polishing Demineralizers

Quantity	2
Type	Mixed Bed, Non-regenerative
Design Flow, gpm	40
Usable Resin Volume, ft ³	15
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Vessel Material	304 SS
Vessel Design Code	ASME III, Class 3

TABLE 11.2-1 (Continued)

Clean Liquid Radioactive Waste System Equipment List

p. Concentrate Demineralizer

Quantity	1
Type	Mixed Bed, Non-regenerative
Design Flow, gpm	40
Usable Resin Volume, ft ³	15
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Vessel Material	304 SS
Vessel Design Code	ASME III, Class 3

q. Primary Demineralizer Filters

Quantity	2
Type	Disposable Cartridge
Design Flow, gpm	140
Absolute Rating, micron	0.1 - 10
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Vessel Material	304 SS
Vessel Design Code	ASME III, Class 3

r. Clean Waste Monitor Tank Filters

Quantity	2
Type	Disposable Cartridge
Design Flow, gpm	40
Nominal Rating, micron	0.2-10
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Vessel Material	304 SS
Vessel Design Code	ASME III, Class 3

s. Spent Resin Tank Strainer

Quantity	1
Type	Metal, Back Flushable
Design Flow, gpm	100
Rating, micron	25
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Vessel Material	304 SS
Vessel Design Code	ASME III, Class 3

TABLE 11.2-1 (Continued)

Clean Liquid Radioactive Waste System Equipment List

t. Boric Acid Evaporators

Quantity	2
Type	Submerged Tube
Capacity, gpm	15
Normal Operating Temperature, °F	220
Vent Discharge Pressure, psig	7
Materials	304 and 316 SS (surfaces in contact with process fluids)
Design Codes	ASME III, Class C; ASME VIII; ANSI B31.7; ANSI B31.1 and TEMA-R

u. Degasifier (No longer used)

Quantity	1
Type	Packed Column
Capacity, gpm	140
Vent Discharge Pressure, psig	7
Materials	304 and 316 SS (surfaces in contact with process fluids)
Design Codes	ASME III, Class 3; ASME VIII; ANSI B31.1 and TEMA-R

v. Clean Waste Receiver Tank Filter

Quantity	1
Type	Disposable Cartridge
Design flow, gpm	140
Nominal Rating, micron	30
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	110
Vessel Material	340SS
Vessel Design Code	ASME VIII, Div I

TABLE 11.2-2

Miscellaneous Liquid Radioactive Waste System Equipment List

a. Miscellaneous Waste Drain Tank

Quantity	1
Type	Horizontal Cylindrical
Volume, gal	14,400
Design Pressure, psig	15
Design Temperature, °F	200
Material	304 SS
Design Code	ASME III, Class C

b. Detergent Waste Drain Tank

Quantity	1
Type	Vertical Cylindrical
Volume, gal	9,950
Design Pressure, psig	15
Design Temperature, °F	200
Material	304 SS
Design Code	ASME III, Class C

c. Miscellaneous Waste Monitor Tank

Quantity	1
Type	Vertical Cylindrical
Volume, gal (nominal)	9,800
Design Pressure, psig	15
Design Temperature, °F	150
Material	304 SS
Design Code	ASME III, Class C

d. Evaporator Storage Tank

Quantity	1
Type	Vertical Cylindrical
Volume, gal (nominal)	947
Design Pressure, psig	15
Design Temperature, °F	200
Material	304 SS
Design Code	ASME III, Class C

TABLE 11.2-2 (Continued)

Miscellaneous Liquid Radioactive Waste System Equipment List

e. Antifoam Tank (No longer used)

Quantity	1
Type	Vertical Cylindrical
Volume (usable), gal	50
Design Pressure, psig	15
Design Temperature, °F	110
Material	304 SS
Design Code	None

f. Miscellaneous Waste Drain Tank Pump

Quantity	1
Type	Centrifugal
Capacity, gpm	140
Design Temperature, °F	200
Material	316 SS
Design Code	Manufacturer's Standard

g. Detergent Waste Drain Tank Pump

Quantity	1
Type	Centrifugal
Capacity, gpm	140
Design Temperature, °F	200
Material	316 SS
Design Code	Manufacturer's Standards

h. Miscellaneous Waste Monitor Tank Pump

Quantity	1
Type	Centrifugal
Capacity, gpm	140
Design Temperature, °F	200
Material	316 SS
Design Code	Manufacturer's Standard

i. Evaporator Storage Tank Pump (No longer used)

Quantity	1
Type	Centrifugal, Canned
Capacity, gpm	20
Design Temperature, °F	200
Material	316 SS
Design Code	ASME III, Class 3

TABLE 11.2-2 (Continued)

Miscellaneous Liquid Radioactive Waste System Equipment List

j. Antifoam Tank Pump (No longer used)

Quantity	1
Type	Positive Displacement, Diaphragm
Capacity, gpm	0 - 10
Design Temperature, °F	110
Material	316 SS
Design Code	None

k. Waste Evaporator (abandoned in place)

Quantity	1
Type	Submerged Tube
Capacity, gpm	15
Vent Discharge Pressure, psig	7
Materials	304 and 316 SS (surfaces in contact with process fluids)
Design Codes	ASME III, Class C; ASME VIII; ANSI B31.7, ANSI B31.1 and TEMA-R

l. Waste Polishing Demineralizer

Quantity	1
Type	Mixed Bed, Non-regenerative
Design Flow, gpm	40
Usable Resin Volume, ft ³	14
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Vessel Material	304 SS
Vessel Design Code	ASME III, Class 3

m. Miscellaneous Waste Monitor Tank Filters

Quantity	2
Type	Disposable Element
Capacity, gpm	140
Micron Rating	≤10 absolute
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Vessel Material	304 SS
Vessel Design Code	ASME III, Class 3

TABLE 11.2-2 (Continued)

Miscellaneous Liquid Radioactive Waste System Equipment List

n. Evaporator Vent Charcoal Filter

Quantity	1
Type	Deep Bed
Capacity	200 ft ³
Capacity, scfm	50
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Vessel Material	304 SS
Vessel Design Code	ASME III, Class 3

o. Deleted

p. Demineralizer Skid Mechanical Post-Filter

Quantity	1
Filter Medium	Disposable Element
Vessel Material	304 SS
Pressure Rating	200 PSI
Vessel Design Code	ASME VIII

q. Demineralizer Skid Resin Filter Vessels

Quantity	5
Capacity	30 ft ³
Pressure/Temperature Rating	150 PSI at 130 °F
Vessel Material	304 SS
Vessel Design Code	ASME VIII

r. Demineralizer Skid Mechanical Pre-Filter

Quantity	1
Filter Medium	Disposable Element
Vessel Material	304 SS
Pressure Rating	150 PSI
Vessel Design Code	Manufacturer's Standard

s. Booster Pump

Quantity	1
Type	Centrifugal
Capacity, gpm	35
Design Temperature	150 F
Material	316 SS
Design Code	Manufacturer's Standard

TABLE 11.2-2 (Continued)

Miscellaneous Liquid Radioactive Waste System Equipment List

t. Reverse Osmosis Skid

Quantity	1
Filter Medium	Disposable Element
Vessel Material	304 SS
Pressure Rating	400 PSI
Vessel Design Code	Manufacturer's Standard

u. Solids Collection Filter Liner

Quantity	1
Filter Medium	Disposable Element
Vessel Material	304 SS / 316 SS
Pressure Rating	150 PSI
Vessel Design Code	ASME Section VIII Division 1 (non-stamped), B31.1

TABLE 11.2-3

Clean Liquid Waste Quantities (1)

<u>Waste Source</u>	<u>Waste Processed (Gal)</u>	<u>Processed Waste Discharged (Gal)</u>	<u>Assumptions and Comments</u>
Reactor Coolant System			
Startup Expansion	96,000	24,000	4 cold startups
Startup Dilution	146,000	39,000	2 cold startups at beginning of life, 1 cold startup at 100 and 200 full (ultimate) power days, respectively
	72,000		2 hot startups at peak xenon at 100 and 200 full power days, respectively
Shim Bleed Normal Cycle	195,000	195,000	Dilution from 1230 to 50 ppm boron
Power Transients	500,000		1-50% power transient per week
System Drain (refueling)	61,400		Drain to level of outlet nozzles
System Drain (maintenance)	84,000		Include drain of 1 steam generator
Totals	1,154,400	258,000	

- (1) The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.2-4

Average and Maximum Radionuclide Concentration (1)

Radio-nuclide	0.1% Failed Fuel (Average Conc.)			1.0% Failed Fuel (Maximum Conc.)		
	Max. Conc. in Reactor Coolant $\rho=0.713$ gm/ml $\mu\text{Ci/ml}$	Max. Conc. Downstream of Letdown Cooler $\rho=1$ gm/ml $\mu\text{Ci/ml}$	Max. Conc. in Clean Liquid Radwaste System Feed $\mu\text{Ci/ml}$	Max. Conc. in Reactor Coolant $\rho=0.713$ gm/ml $\mu\text{Ci/ml}$	Max. Conc. Downstream of Letdown Cooler $\rho=1$ gm/ml $\mu\text{Ci/ml}$	Max. Conc. in Clean Liquid Radwaste System Feed $\mu\text{Ci/ml}$
<u>Ionic Radionuclides</u>						
Rb-88	0.192	0.270	2.70×10^{-3}	1.92	2.70	2.70×10^{-2}
Sr-89	3.21×10^{-4}	4.51×10^{-4}	4.51×10^{-6}	3.21×10^{-3}	4.51×10^{-3}	4.51×10^{-5}
Sr-90	1.04×10^{-5}	1.46×10^{-5}	1.46×10^{-7}	1.04×10^{-4}	1.46×10^{-4}	1.46×10^{-6}
Sr-91	2.03×10^{-3}	2.85×10^{-3}	2.85×10^{-5}	2.03×10^{-2}	2.85×10^{-2}	2.85×10^{-4}
Sr-92	6.21×10^{-4}	8.72×10^{-4}	8.72×10^{-6}	6.21×10^{-3}	8.72×10^{-3}	8.72×10^{-5}
Y-90	7.19×10^{-4}	1.01×10^{-3}	1.01×10^{-3}	7.19×10^{-3}	1.01×10^{-2}	1.01×10^{-2}
Y-91	4.03×10^{-3}	5.66×10^{-3}	5.66×10^{-3}	4.03×10^{-2}	5.66×10^{-2}	5.66×10^{-2}
Mo-99	0.293	0.412	0.412	2.93	4.12	4.12×10^{-2}
I-131	0.230	0.323	3.23×10^{-3}	2.30	3.23	3.23×10^{-2}
I-132	0.160	0.225	2.25×10^{-3}	1.60	2.25	2.25×10^{-2}
I-133	0.269	0.378	3.78×10^{-3}	2.69	3.78	3.78×10^{-2}
I-134	3.26×10^{-2}	4.58×10^{-2}	4.58×10^{-4}	0.326	0.458	4.58×10^{-3}
I-135	0.135	0.190	1.90×10^{-3}	1.35	1.90	1.90×10^{-2}
Cs-134	0.291	0.409	0.409	2.91	4.09	4.09
Cs-136	5.27×10^{-2}	7.40×10^{-2}	7.40×10^{-2}	0.527	0.740	0.740
Cs-137	0.904	1.27	1.27	9.04	12.7	12.7
Cs-138	5.23×10^{-2}	7.34×10^{-2}	7.24×10^{-2}	0.523	0.734	0.734
Ba-137m	0.832	1.17	1.17	8.32	11.7	11.7
Ba-139	5.41×10^{-3}	7.60×10^{-3}	7.60×10^{-2}	5.41×10^{-2}	7.60×10^{-2}	7.60×10^{-4}
Ba-140	4.01×10^{-4}	5.63×10^{-4}	5.63×10^{-6}	4.01×10^{-3}	5.63×10^{-3}	5.63×10^{-5}
La-140	1.60×10^{-4}	2.15×10^{-4}	2.25×10^{-6}	1.60×10^{-3}	2.25×10^{-3}	2.25×10^{-5}
Ca-144	3.69×10^{-5}	5.18×10^{-5}	5.18×10^{-7}	3.69×10^{-4}	5.18×10^{-4}	5.18×10^{-6}

TABLE 11.2-4 (Continued)

Average and Maximum Radionuclide Concentrations

Radio-nuclide	0.1% Failed Fuel (Average Conc.)			1.0% Failed Fuel (Maximum Conc.)		
	Max. Conc. in Reactor Coolant $\rho=0.713$ gm/ml $\mu\text{Ci/ml}$	Max. Conc. Downstream of Letdown Cooler $\rho=1$ gm/ml $\mu\text{Ci/ml}$	Max. Conc. in Clean Liquid Radwaste System Feed $\mu\text{Ci/ml}$	Max. Conc. in Reactor Coolant $\rho=0.713$ gm/ml $\mu\text{Ci/ml}$	Max. Conc. Downstream of Letdown Cooler $\rho=1$ gm/ml $\mu\text{Ci/ml}$	Max. Conc. in Clean Liquid Radwaste System Feed $\mu\text{Ci/ml}$
<u>Gaseous Radionuclides</u>						
Ar-41	7.9×10^{-2}	0.111	0.111	7.9×10^{-2}	0.111	0.111
Kr-83m	2.05×10^{-2}	2.88×10^{-2}	2.88×10^{-2}	0.205	0.288	0.288
Kr-85m	0.109	0.153	0.153	1.09	1.53	1.53
Kr-85	0.224	0.315	0.315	2.24	3.15	3.15
Kr-87	5.96×10^{-2}	8.37×10^{-2}	8.37×10^{-2}	0.596	0.837	0.837
Kr-88	0.191	0.268	0.268	1.91	2.68	2.68
Xe-131m	0.160	0.225	0.225	1.60	2.25	2.25
Xe-133m	0.202	0.284	0.284	2.02	2.84	2.84
Xe-133	17.7	24.9	24.9	177.	249.	249.
Xe-135m	6.6×10^{-2}	9.26×10^{-2}	9.26×10^{-2}	0.66	0.926	0.926
Xe-135	0.362	0.508	0.508	3.62	5.08	5.08
Xe-138	3.64×10^{-2}	5.11×10^{-2}	5.11×10^{-2}	0.364	0.511	0.511

- (1) The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

TABLE 11.2-5

Miscellaneous Liquid Radioactive Wastes (1)

<u>Waste Source</u>	<u>Total Quantity (gal) Per Year</u>	<u>Assumptions and Comments</u>
<u>Non-Detergent Waste</u>		
Sampling Laboratory	3,000	12 samples per week at 5 gallons per sample
Demineralizer Sluice	10,000	Produced at a rate of 2 ft ³ /ft ³ resin; every demineralizer normally handling radioactive fluid is changed out once per refueling; includes 1,000 gallons for back-washing purification demineralizer filter
Regeneration Waste	38,000	20 ft ³ of waste produced per ft ³ if resin regenerated (this includes backwash and rinse contributions); 4 regenerations of 54 ft ³ deborating demineralizer occur per year (takes into account one (1) 50% power transient per week)
Area Washdowns	110,000	5 gpm hose, 1 hour per day
Miscellaneous System Leakage	45,000	5 gph leakage
Condensate Demineralizer	390,000	7481 gallons per backwash; 1 backwash per week for 52 weeks
<u>Detergent Waste</u>		
Hot showers, (decon) and detergent waste from laboratory drains	500,000	10 showers per day at 30 gallons per shower; 350,000 gallons per year of contaminated laundry per year
<u>Secondary System Leakage</u>	2,102,400	5 gpm for 292 days

- (1) The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.2-6

Maximum Radionuclide Concentrations in Non-Detergent Miscellaneous Liquid Waste (1)

Maximum Concentrations in Waste from Different Sources

Radionuclide	-1- Misc. Sys. Leakage, Drainage from Area Washdown Sampling and Laboratory Operations ($\mu\text{Ci/ml}$)	-2- Demineralizer Regeneration ($\mu\text{Ci/ml}$)	-3- Sluicing of Demin Resins ($\mu\text{Ci/ml}$)	-4- Condensate Demin. Backwash ($\mu\text{Ci/ml}$)	-5- Max. Conc. From Any Source ($\mu\text{Ci/ml}$)
<u>Ionic Radionuclides</u>					
Rb-88	2.70	0.675	2.70	4.22×10^{-2}	2.70
Sr-89	4.51×10^{-3}	1.13×10^{-3}	4.51×10^{-3}	7.04×10^{-5}	4.51×10^{-3}
Sr-90	1.46×10^{-4}	3.65×10^{-5}	1.46×10^{-4}	2.28×10^{-6}	1.46×10^{-4}
Sr-91	2.85×10^{-2}	7.13×10^{-3}	2.85×10^{-2}	4.45×10^{-4}	2.85×10^{-2}
Sr-92	8.72×10^{-3}	2.18×10^{-3}	8.72×10^{-3}	1.37×10^{-4}	8.72×10^{-3}
Y-90	1.01×10^{-2}	2.53×10^{-3}	1.01×10^{-2}	1.58×10^{-4}	1.01×10^{-2}
Y-91	5.66×10^{-2}	1.42×10^{-2}	5.66×10^{-2}	8.83×10^{-4}	5.66×10^{-2}
Mo-99	4.12	1.03	4.12	6.43×10^{-2}	4.12
I-131	3.23	0.808	3.23	5.04×10^{-2}	3.23
I-132	2.25	0.563	2.25	3.51×10^{-2}	2.25
I-133	3.78	0.945	3.78	5.90×10^{-2}	3.78
I-134	0.458	0.115	0.458	7.15×10^{-3}	0.458
I-135	1.90	0.475	1.90	2.97×10^{-2}	1.90
Cs-134	4.09	1.03	4.09	6.39×10^{-2}	4.09
Cs-136	0.740	0.185	0.740	1.16×10^{-2}	0.740
Cs-137	12.7	3.18	12.7	0.199	12.7
Cs-138	0.734	0.184	0.734	1.15×10^{-2}	0.734
Ba-137m	11.7	2.93	11.7	0.183	11.7
Ba-139	7.60×10^{-2}	1.90×10^{-2}	7.60×10^{-2}	1.19×10^{-3}	7.60×10^{-2}
Ba-140	5.63×10^{-3}	1.41×10^{-3}	5.63×10^{-3}	8.79×10^{-5}	5.63×10^{-3}
La-140	2.25×10^{-3}	5.63×10^{-4}	2.25×10^{-3}	3.51×10^{-5}	2.25×10^{-3}
Ce-144	5.18×10^{-4}	1.30×10^{-4}	5.18×10^{-4}	8.09×10^{-6}	5.18×10^{-4}
<u>Corrosion Products</u>					
Cr-51	3.69×10^{-3}	-	22.12	-	22.12
Mn-54	4.23×10^{-4}	-	2.54	-	2.54
Fe-55	1.46×10^{-2}	-	87.44	-	87.44
Fe-59	4.23×10^{-4}	-	2.54	-	2.54
Co-58	2.22×10^{-2}	-	132.84	-	132.84
Co-60	1.19×10^{-4}	-	0.72	-	0.72
Zr-95	2.91×10^{-2}	-	174.04	-	174.04

- (1) The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.2-7

Average Radionuclide Concentrations in Non-Detergent Miscellaneous Liquid Waste (1)

Average Concentrations In Waste from Different Sources

Radionuclide	-1- Misc. Sys. Leakage, Drainage from Area Washdowns, Laundry, Rinse, Sampling and Laboratory Operations ($\mu\text{Ci/ml}$)	-2- Demineralizer Regeneration ($\mu\text{Ci/ml}$)	-3- Sluicing of Demin. Resins ($\mu\text{Ci/ml}$)	-4- Condensate Demin. Backwash ($\mu\text{Ci/ml}$)	-5- Avg. Conc. Taking into Account Columns (1),(2), & (3) ($\mu\text{Ci/ml}$)	-6- Avg. Conc. Taking into Account Columns (1),(2), (3), & (4) ($\mu\text{Ci/ml}$)
<u>Ionic Radionuclides</u>						
Rb-88	7.20×10^{-2}	6.75×10^{-2}	0.270	4.22×10^{-3}	8.00×10^{-2}	3.22×10^{-2}
Sr-89	1.21×10^{-4}	1.13×10^{-4}	4.51×10^{-4}	7.04×10^{-6}	1.34×10^{-4}	5.37×10^{-5}
Sr-90	3.90×10^{-6}	3.65×10^{-6}	1.46×10^{-5}	2.28×10^{-7}	4.33×10^{-6}	1.74×10^{-6}
Sr-91	7.60×10^{-4}	7.13×10^{-4}	2.85×10^{-3}	4.45×10^{-5}	8.44×10^{-4}	3.40×10^{-4}
Sr-92	2.33×10^{-4}	2.18×10^{-4}	8.72×10^{-4}	1.37×10^{-5}	2.59×10^{-4}	1.04×10^{-4}
Y-90	2.70×10^{-4}	2.53×10^{-4}	1.01×10^{-3}	1.58×10^{-5}	3.00×10^{-4}	1.20×10^{-4}
Y-91	1.51×10^{-3}	1.42×10^{-3}	5.66×10^{-3}	8.83×10^{-5}	1.68×10^{-4}	6.74×10^{-4}
Mo-99	0.110	0.103	0.412	6.43×10^{-3}	0.122	4.91×10^{-2}
I-131	8.62×10^{-2}	8.08×10^{-2}	0.323	5.04×10^{-3}	9.57×10^{-2}	3.85×10^{-2}
I-132	6.00×10^{-2}	5.63×10^{-2}	0.225	3.51×10^{-3}	6.67×10^{-2}	2.68×10^{-2}
I-133	0.110	9.45×10^{-2}	0.378	5.90×10^{-3}	0.112	4.51×10^{-2}
I-134	1.23×10^{-2}	1.15×10^{-2}	4.58×10^{-2}	7.15×10^{-4}	1.26×10^{-2}	5.46×10^{-3}
I-135	5.07×10^{-2}	4.75×10^{-2}	0.190	2.97×10^{-3}	5.63×10^{-2}	2.27×10^{-2}
Cs-134	0.110	0.103	0.409	6.39×10^{-3}	0.122	4.87×10^{-2}
Cs-136	1.98×10^{-2}	1.85×10^{-2}	7.40×10^{-2}	1.16×10^{-3}	2.20×10^{-2}	8.82×10^{-3}
Cs-137	0.339	0.318	1.27	1.99×10^{-2}	0.376	0.152
Cs-138	1.96×10^{-2}	1.84×10^{-2}	7.34×10^{-2}	1.15×10^{-3}	2.18×10^{-2}	8.74×10^{-3}
Ba-137m	0.312	0.293	1.17	1.83×10^{-2}	0.347	0.140
Ba-139	2.03×10^{-3}	1.90×10^{-3}	7.60×10^{-3}	1.19×10^{-4}	2.25×10^{-3}	9.05×10^{-4}
Ba-140	1.51×10^{-4}	1.41×10^{-4}	5.63×10^{-4}	8.79×10^{-6}	1.67×10^{-4}	6.71×10^{-5}
La-140	6.00×10^{-5}	5.63×10^{-5}	2.25×10^{-4}	3.51×10^{-6}	6.67×10^{-5}	2.68×10^{-5}
Ce-144	1.39×10^{-5}	1.30×10^{-5}	5.18×10^{-5}	8.09×10^{-7}	1.54×10^{-5}	6.17×10^{-6}
<u>Corrosion Products</u>						
Cr-51	9.84×10^{-4}	-	8.28	-	0.364	0.135
Mn-54	1.13×10^{-4}	-	0.947	-	4.17×10^{-2}	1.54×10^{-2}
Fe-55	3.90×10^{-3}	-	32.71	-	1.44	0.531
Fe-59	1.13×10^{-4}	-	0.947	-	4.17×10^{-2}	1.54×10^{-2}
Co-58	5.92×10^{-3}	-	49.69	-	2.19	0.806
Co-60	3.18×10^{-5}	-	0.267	-	1.18×10^{-2}	4.33×10^{-3}
Zr-95	7.76×10^{-3}	-	65.10	-	2.87	1.06

- (1) The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

TABLE 11.2-8

Maximum Crud Activity Levels in Sluice Waste (1)

<u>Radionuclide</u>	<u>Concentration</u> <u>μCi/ml</u>
Cr-51.....	22.12
Mn-54.....	2.54
Fe-55.....	87.44
Fe-59.....	2.54
Co-58.....	132.84
Co-60.....	0.72
Zr-95.....	174.04

- (1) The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

TABLE 11.2-9

Maximum and Average Radionuclide Concentrations in Liquid Secondary System Leakage (1)

Radionuclide	Max. Conc. in Secondary System Leakage $\mu\text{Ci/ml}$	Avg. Conc. in Secondary System Leakage $\mu\text{Ci/ml}$
<u>Ionic Radionuclides</u>		
Rb-88	5.95×10^{-6}	3.84×10^{-7}
Sr-89	2.50×10^{-8}	1.62×10^{-9}
Sr-90	8.04×10^{-10}	5.19×10^{-11}
Sr-91	1.52×10^{-7}	9.80×10^{-9}
Sr-92	4.14×10^{-8}	2.67×10^{-9}
Y-90	5.56×10^{-8}	3.59×10^{-9}
Y-91	3.14×10^{-7}	2.03×10^{-8}
Mo-99	2.27×10^{-5}	1.47×10^{-6}
I-131	1.79×10^{-5}	1.16×10^{-6}
I-132	1.05×10^{-5}	6.77×10^{-7}
I-133	2.05×10^{-5}	1.33×10^{-6}
I-134	1.68×10^{-6}	1.09×10^{-7}
I-135	9.85×10^{-6}	6.35×10^{-7}
Cs-134	2.27×10^{-5}	1.47×10^{-6}
Cs-136	4.10×10^{-6}	2.65×10^{-7}
Cs-137	6.99×10^{-5}	4.51×10^{-6}
Cs-138	2.23×10^{-6}	1.44×10^{-7}
Ba-137m	5.66×10^{-6}	3.65×10^{-7}
Ba-139	3.18×10^{-7}	2.05×10^{-8}
Ba-140	3.11×10^{-8}	2.01×10^{-9}
La-140	1.23×10^{-8}	7.93×10^{-10}
Ce-144	2.87×10^{-9}	1.85×10^{-10}
Cr-151	2.05×10^{-8}	1.33×10^{-8}
Mn-54	2.34×10^{-9}	1.51×10^{-9}
Fe-55	8.04×10^{-8}	5.19×10^{-8}
Fe-59	2.34×10^{-9}	1.51×10^{-9}
Co-58	1.23×10^{-7}	7.93×10^{-8}
Co-60	6.54×10^{-10}	4.22×10^{-10}
Zr-95	1.61×10^{-7}	1.04×10^{-7}

- (1) The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.2-10

Decontamination Factors for Demineralizers (1)

<u>Radionuclide</u>	<u>D.F.</u>	<u>Resin</u>	<u>Remarks</u>	<u>Reference</u>
Cs-137	760,000	Cation, H Form	Lab tests with borated solution	16
I-131	714,000	Anion, OH Form	Lab tests with borated solution	
Cs-137	73,000	1:1 Mixed bed Li-borated Form	Lab study; test solution contained 100 ppm H ₃ Bo ₃ , 0.85 ppm Cs, and 2 ppm lithium	17
Cs-137	7	Mixed bed, K-borate Form	Tests run at Saxton Reactor	18
I-131	100			
Cs-137	1,000 (ave.)	Mixed Bed, Hydrogen-hydroxyl form	Lab study with high-purity water; experiments on in-pile test loop water generally confirm laboratory work with individual radioisotopes	19
I-131	1,000 (ave.)			
Cs-137	1,000	2:1 (volume ratio) mixture of cation and anion resins	Lab study done to determine effects of flow rate on DF's for Cs-137 and I-131 (DF's were found to decrease with increasing flow rate)	20
I-131	80-1,000			
Cs-137	1000	Cation, hydrogen form	Experimental study using waste generated at Hanford which had been steam-stripped	21
Cs-137	100	Cation, H Form	Lab study in which ion exchange was only one step in a process arrangement that also utilized coagulation and filtration	22
I-131	100	Anion, OH Form		
Cs-137	50	Cation	Operational study using system used to process low and intermediate level wastes at Idaho Chemical Processing Plant (ICPP); demineralizer studied used to polish evap. distillate	23
Cs-137	1	Mixed bed, Li-hydroxy1 form	Test using coolant water from the NRX Reactor at Chalk River, Ontario	24
I-131	1,000			
Cs-137	100	Cation	Lab study with test solution containing $1.1 \times 10^{-4}\text{M}$ NH ⁺ and $7 \times 10^{-5}\text{M}$ Na, DF applicable up to about 9000 column volumes	25

- (1) The following information pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design. The ODCM provides information pertaining to the monitoring of waste streams and ensures the resultant offsite doses are in compliance with current regulatory requirements. The following DF information is considered historical.

TABLE 11.2-11

Clean Liquid Radwaste Recycle Flow Paths

Originating Tank	Recycle Flow Path	Processing Rate (gpm)
Clean Waste Receiver Tank	Through the primary demineralizer filter and/or the primary demineralizer back to the originating or the other receiver tank.	140
Clean Waste Monitor Tank	Through the boric acid evaporator and/or polishing demineralizer and/or clean waste monitor tank filter back to the originating or the other monitor tank.	40
Clean Waste Monitor Tank	Through the primary demineralizer filter and/or the primary demineralizer and/or the deborating demineralizer back to the originating or the other monitor tank.	140
Clean Waste Monitor Tank	To one of the clean waste receiver tanks with the option of passing through the primary demineralizer filter and/or the primary demineralizer and/or the deborating demineralizer.	140
Concentrate Storage Tank	Through the concentrate demineralizer back to the tank.	40
Concentrate Storage Tank	To one of the clean waste receiver tanks or the boric acid evaporator. (Note: some dilution of the storage tank's contents may be necessary to prevent crystallization of concentrated boric acid in non-heat traced piping.)	40

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.2-12

Maximum Specific Activities at Significant Points in the Processing Cycle for Clean Liquid Radioactive Waste (1)

Radionuclide	Max. Sp. Act. In Waste ($\mu\text{Ci/ml}$)	Max. Sp. Act. In Waste After Degasifier ($\mu\text{Ci/ml}$)	Max. Sp. Act. In Waste After Primary Demin. Filter ($\mu\text{Ci/ml}$)	Max. Sp. Act. In Waste After Primary Demineralizer ($\mu\text{Ci/ml}$)	Max. Sp. Act. In Waste After Boric Acid Evaporator ($\mu\text{Ci/ml}$)	Max. Sp. Act. In Waste After Polishing Demineralizer ($\mu\text{Ci/ml}$)	Max. Sp. Act. In Waste After Clean Waste Monitor Tank Filter ($\mu\text{Ci/ml}$)	Max. Sp. Act. In Waste Discharged To Lake ($\mu\text{Ci/ml}$)
<u>Insoluble Corrosion Products</u>								
Cr-51	3.69×10^{-3}	3.69×10^{-3}	3.69×10^{-4}	3.69×10^{-4}	1.85×10^{-7}	1.85×10^{-7}	1.85×10^{-7}	4.06×10^{-10}
Mn-54	4.23×10^{-4}	4.23×10^{-4}	4.23×10^{-5}	4.23×10^{-5}	2.12×10^{-8}	2.12×10^{-8}	2.12×10^{-8}	4.65×10^{-11}
Fe-55	1.46×10^{-2}	1.46×10^{-2}	1.46×10^{-3}	1.46×10^{-3}	7.30×10^{-7}	7.30×10^{-7}	7.30×10^{-7}	1.61×10^{-9}
Fe-59	4.23×10^{-4}	4.23×10^{-4}	4.23×10^{-5}	4.23×10^{-5}	2.12×10^{-8}	2.12×10^{-8}	2.12×10^{-8}	4.65×10^{-11}
Co-58	2.22×10^{-2}	2.22×10^{-2}	2.22×10^{-3}	2.22×10^{-3}	1.11×10^{-6}	1.11×10^{-6}	1.11×10^{-6}	2.44×10^{-9}
Co-60	1.19×10^{-4}	1.19×10^{-4}	1.19×10^{-5}	1.19×10^{-5}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	2.20×10^{-11}
Zr-95	2.91×10^{-2}	2.91×10^{-2}	2.91×10^{-3}	2.91×10^{-3}	1.46×10^{-6}	1.46×10^{-6}	1.46×10^{-6}	3.21×10^{-9}
<u>Gaseous Radionuclides</u>								
Ar-41	0.111	1.11×10^{-6}	1.11×10^{-6}	1.11×10^{-6}	1.11×10^{-6}	1.11×10^{-6}	1.11×10^{-6}	2.44×10^{-9}
Kr-38m	0.288	2.88×10^{-6}	2.88×10^{-6}	2.88×10^{-6}	2.88×10^{-6}	2.88×10^{-6}	2.88×10^{-6}	6.32×10^{-9}
Kr-85m	1.53	1.53×10^{-5}	1.53×10^{-5}	1.53×10^{-5}	1.53×10^{-5}	1.53×10^{-5}	1.53×10^{-5}	3.36×10^{-8}
Kr-85	3.15	3.15×10^{-5}	3.15×10^{-5}	3.15×10^{-5}	3.15×10^{-5}	3.15×10^{-5}	3.15×10^{-5}	6.91×10^{-8}
Kr-87	0.837	8.37×10^{-6}	8.37×10^{-6}	8.37×10^{-6}	8.37×10^{-6}	8.37×10^{-6}	8.37×10^{-6}	1.84×10^{-8}
Kr-88	2.68	2.68×10^{-5}	2.68×10^{-5}	2.68×10^{-5}	2.68×10^{-5}	2.68×10^{-5}	2.68×10^{-5}	5.88×10^{-8}
Xe-131m	2.25	2.25×10^{-5}	2.25×10^{-5}	2.25×10^{-5}	2.25×10^{-5}	2.25×10^{-5}	2.25×10^{-5}	4.94×10^{-8}
Xe-133m	2.84	2.84×10^{-5}	2.84×10^{-5}	2.84×10^{-5}	2.84×10^{-5}	2.84×10^{-5}	2.84×10^{-5}	6.23×10^{-8}
Xe-133	249	2.49×10^{-3}	2.49×10^{-3}	2.49×10^{-3}	2.49×10^{-3}	2.49×10^{-3}	2.49×10^{-3}	5.47×10^{-6}
Xe-135m	0.926	9.26×10^{-6}	9.26×10^{-6}	9.26×10^{-6}	9.26×10^{-6}	9.26×10^{-6}	9.26×10^{-6}	2.04×10^{-8}
Xe-135	5.08	5.08×10^{-5}	5.08×10^{-5}	5.08×10^{-5}	5.08×10^{-5}	5.08×10^{-5}	5.08×10^{-5}	1.12×10^{-7}
Xe-138	0.511	5.11×10^{-6}	5.11×10^{-6}	5.11×10^{-6}	5.11×10^{-6}	5.11×10^{-6}	5.11×10^{-6}	1.13×10^{-8}

TABLE 11.2-12 (Continued)

Maximum Specific Activities at Significant Points in the Processing Cycle for Clean Liquid Radioactive Waste (1)

Radionuclide	Max. Sp. Act. In Waste	Max. Sp. Act. In Waste After Degasifier	Max. Sp. Act. In Waste After Primary Demin. Filter	Max. Sp. Act. In Waste After Primary Demineralizer	Max. Sp. Act. In Waste After Boric Acid Evaporator	Max. Sp. Act. In Waste After Polishing Demineralizer	Max. Sp. Act. In Waste After Clean Waste Monitor Tank Filter	Max. Sp. Act. In Waste Discharge To Lake
	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)
<u>Ionic Radionuclides</u>								
Rb-88	2.70×10^{-2}	2.70×10^{-2}	2.70×10^{-2}	2.70×10^{-3}	1.35×10^{-6}	1.35×10^{-7}	1.35×10^{-7}	2.97×10^{-10}
Sr-89	4.51×10^{-5}	4.51×10^{-5}	4.51×10^{-5}	4.51×10^{-6}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	2.20×10^{-11}
Sr-90	1.46×10^{-6}	1.46×10^{-6}	1.46×10^{-6}	1.46×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	2.20×10^{-11}
Sr-91	2.85×10^{-4}	2.85×10^{-4}	2.85×10^{-4}	2.85×10^{-5}	1.43×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	2.20×10^{-11}
Sr-92	8.72×10^{-5}	8.72×10^{-5}	8.72×10^{-5}	8.72×10^{-6}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	2.20×10^{-11}
Y-90	1.01×10^{-2}	1.01×10^{-2}	1.01×10^{-2}	1.01×10^{-4}	5.05×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	2.20×10^{-11}
Y-91	5.66×10^{-2}	5.66×10^{-2}	5.66×10^{-2}	5.66×10^{-4}	2.83×10^{-7}	2.83×10^{-8}	2.83×10^{-8}	6.21×10^{-11}
Mo-99	4.12	4.12	4.12	4.12×10^{-2}	2.06×10^{-5}	2.06×10^{-6}	2.06×10^{-6}	4.52×10^{-9}
I-131	3.23×10^{-2}	3.23×10^{-2}	3.23×10^{-2}	3.23×10^{-3}	1.08×10^{-4}	1.08×10^{-5}	1.08×10^{-5}	2.37×10^{-8}
I-132	2.25×10^{-2}	2.25×10^{-2}	2.25×10^{-2}	2.25×10^{-3}	7.50×10^{-5}	7.50×10^{-6}	7.50×10^{-6}	1.65×10^{-8}
I-133	3.78×10^{-2}	3.78×10^{-2}	3.78×10^{-2}	3.78×10^{-3}	1.26×10^{-4}	1.26×10^{-5}	1.26×10^{-5}	2.77×10^{-8}
I-134	4.58×10^{-3}	4.58×10^{-3}	4.58×10^{-3}	4.58×10^{-4}	1.53×10^{-5}	1.53×10^{-6}	1.53×10^{-6}	3.36×10^{-9}
I-135	1.90×10^{-2}	1.90×10^{-2}	1.90×10^{-2}	1.90×10^{-3}	6.34×10^{-5}	6.34×10^{-6}	6.34×10^{-6}	1.40×10^{-8}
Cs-134	4.09	4.09	4.09	4.09×10^{-2}	2.05×10^{-5}	2.05×10^{-6}	2.05×10^{-6}	4.50×10^{-9}
Cs-136	0.740	0.740	0.740	7.40×10^{-3}	3.70×10^{-6}	3.70×10^{-7}	3.70×10^{-7}	8.12×10^{-10}
Cs-137	12.7	12.7	12.7	0.127	6.35×10^{-5}	6.35×10^{-6}	6.35×10^{-6}	1.40×10^{-8}
Cs-138	0.734	0.734	0.734	7.34×10^{-3}	3.67×10^{-6}	3.67×10^{-7}	3.67×10^{-7}	8.05×10^{-10}
Ba-137m	11.7	11.7	11.7	0.117	5.85×10^{-5}	5.85×10^{-6}	5.85×10^{-6}	1.29×10^{-8}
Ba-139	7.60×10^{-4}	7.60×10^{-4}	7.60×10^{-4}	7.60×10^{-5}	3.80×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	2.20×10^{-11}
Ba-140	5.63×10^{-5}	5.63×10^{-5}	5.63×10^{-5}	5.63×10^{-6}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	2.20×10^{-11}
La-140	2.25×10^{-5}	2.25×10^{-5}	2.25×10^{-5}	2.25×10^{-6}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	2.20×10^{-11}
Ce-144	5.18×10^{-6}	5.18×10^{-6}	5.18×10^{-6}	5.18×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	2.20×10^{-11}

- (1) The following information pertaining to estimated doses was provided as part of the initial safety analysis. Operations of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated dose information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.2-13

Average Specific Activities at Significant Points in the Processing Cycle for Clean Liquid Radioactive Waste (1)

Radionuclide	Avg. Sp. Act. In Waste Feed	Avg. Sp. Act. In Waste After Degasifier	Avg. Sp. Act. In Waste After Primary Demin. Filter	Avg. Sp. Act. In Waste After Primary Demineralizer	Avg. Sp. Act. In Waste After Boric Acid Evaporator	Avg. Sp. Act. In Waste After Polishing Demineralizer	Avg. Sp. Act. In Waste After Clean Waste Monitor Tank Filter	Avg. Sp. Act. In Waste Discharged To Lake
	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)
<u>Insoluble Corrosion Products</u>								
Cr-51	3.69×10^{-3}	3.69×10^{-3}	3.69×10^{-4}	3.69×10^{-4}	9.23×10^{-8}	9.23×10^{-8}	9.23×10^{-8}	4.05×10^{-10}
Mn-54	4.23×10^{-4}	4.23×10^{-4}	4.23×10^{-5}	4.23×10^{-5}	1.06×10^{-8}	1.06×10^{-8}	1.06×10^{-8}	4.65×10^{-11}
Fe-55	1.46×10^{-2}	1.46×10^{-2}	1.46×10^{-3}	1.46×10^{-3}	3.65×10^{-7}	3.65×10^{-7}	3.65×10^{-7}	1.61×10^{-9}
Fe-59	4.23×10^{-4}	4.23×10^{-4}	4.23×10^{-5}	4.23×10^{-5}	1.06×10^{-8}	1.06×10^{-8}	1.06×10^{-8}	4.65×10^{-11}
Co-58	2.22×10^{-2}	2.22×10^{-2}	2.22×10^{-3}	2.22×10^{-3}	5.55×10^{-7}	5.55×10^{-7}	5.55×10^{-7}	2.44×10^{-9}
Co-60	1.19×10^{-4}	1.19×10^{-4}	1.19×10^{-5}	1.19×10^{-5}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Zr-95	2.91×10^{-2}	2.91×10^{-2}	2.91×10^{-3}	2.91×10^{-3}	7.28×10^{-7}	7.28×10^{-7}	7.28×10^{-7}	3.20×10^{-9}
<u>Gaseous Radionuclides</u>								
Ar-41	0.111	1.11×10^{-6}	1.11×10^{-6}	1.11×10^{-6}	1.11×10^{-6}	1.11×10^{-6}	1.11×10^{-6}	4.87×10^{-9}
Kr-38m	2.88×10^{-2}	2.88×10^{-7}	2.88×10^{-7}	2.88×10^{-7}	2.88×10^{-7}	2.88×10^{-7}	2.88×10^{-7}	1.27×10^{-9}
Kr-85m	0.153	1.53×10^{-6}	1.53×10^{-6}	1.53×10^{-6}	1.53×10^{-6}	1.53×10^{-6}	1.53×10^{-6}	6.72×10^{-9}
Kr-85	0.315	3.15×10^{-6}	3.15×10^{-6}	3.15×10^{-6}	3.15×10^{-6}	3.15×10^{-6}	3.15×10^{-6}	1.39×10^{-8}
Kr-87	8.37×10^{-2}	8.37×10^{-7}	8.37×10^{-7}	8.37×10^{-7}	8.37×10^{-7}	8.37×10^{-7}	8.37×10^{-7}	3.68×10^{-9}
Kr-88	0.268	2.68×10^{-6}	2.68×10^{-6}	2.68×10^{-6}	2.68×10^{-6}	2.68×10^{-6}	2.68×10^{-6}	1.18×10^{-8}
Xe-131m	0.225	2.25×10^{-6}	2.25×10^{-6}	2.25×10^{-6}	2.25×10^{-6}	2.25×10^{-6}	2.25×10^{-6}	9.87×10^{-9}
Xe-133m	0.284	2.84×10^{-6}	2.84×10^{-6}	2.84×10^{-6}	2.84×10^{-6}	2.84×10^{-6}	2.84×10^{-6}	1.25×10^{-8}
Xe-133	24.9	2.49×10^{-4}	2.49×10^{-4}	2.49×10^{-4}	2.49×10^{-4}	2.49×10^{-4}	2.49×10^{-4}	1.10×10^{-6}
Xe-135m	9.26×10^{-2}	9.26×10^{-7}	9.26×10^{-7}	9.26×10^{-7}	9.26×10^{-7}	9.26×10^{-7}	9.26×10^{-7}	4.07×10^{-9}
Xe-135	0.508	5.08×10^{-6}	5.08×10^{-6}	5.08×10^{-6}	5.08×10^{-6}	5.08×10^{-6}	5.08×10^{-6}	2.23×10^{-8}
Xe-138	5.11×10^{-2}	5.11×10^{-7}	5.11×10^{-7}	5.11×10^{-7}	5.11×10^{-7}	5.11×10^{-7}	5.11×10^{-7}	2.25×10^{-9}

TABLE 11.2-13 (Continued)

Average Specific Activities at Significant Points in the Processing Cycle for Clean Liquid Radioactive Waste (1)

Radionuclide	Avg. Sp. Act. In Waste Feed	Avg. Sp. Act. In Waste After Degasifier	Avg. Sp. Act. In Waste After Primary Demin. Filter	Avg. Sp. Act. In Waste After Primary Demineralizer	Avg. Sp. Act. In Waste After Boric Acid Evaporator	Avg. Sp. Act. In Waste After Polishing Demineralizer	Avg. Sp. Act. In Waste After Clean Waste Monitor Tank Filter	Avg. Sp. Act. In Waste Discharged To Lake
	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)	($\mu\text{Ci/ml}$)
<u>Ionic Radionuclides</u>								
Rb-88	2.70×10^{-3}	2.70×10^{-3}	2.70×10^{-3}	2.70×10^{-4}	6.75×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Sr-89	4.51×10^{-6}	4.51×10^{-6}	4.51×10^{-6}	4.51×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Sr-90	1.46×10^{-7}	1.46×10^{-7}	1.46×10^{-7}	1.46×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Sr-91	2.85×10^{-5}	2.85×10^{-5}	2.85×10^{-5}	2.85×10^{-6}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Sr-92	8.72×10^{-6}	8.72×10^{-6}	8.72×10^{-6}	8.72×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Y-90	1.01×10^{-3}	1.01×10^{-3}	1.01×10^{-3}	1.0×10^{-5}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Y-91	5.66×10^{-3}	5.66×10^{-3}	5.66×10^{-3}	5.66×10^{-5}	1.42×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Mo-99	0.412	0.412	0.412	4.12×10^{-3}	1.03×10^{-6}	1.03×10^{-7}	1.03×10^{-7}	4.52×10^{-10}
I-131	3.23×10^{-3}	3.23×10^{-3}	3.23×10^{-3}	3.23×10^{-4}	5.39×10^{-6}	5.39×10^{-7}	5.39×10^{-7}	2.37×10^{-9}
I-132	2.25×10^{-3}	2.25×10^{-3}	2.25×10^{-3}	2.25×10^{-4}	3.76×10^{-6}	3.75×10^{-7}	3.75×10^{-7}	1.65×10^{-9}
I-133	3.78×10^{-3}	3.78×10^{-3}	3.78×10^{-3}	3.78×10^{-4}	6.31×10^{-6}	6.30×10^{-7}	6.30×10^{-7}	2.77×10^{-9}
I-134	4.58×10^{-4}	4.58×10^{-4}	4.58×10^{-4}	4.58×10^{-5}	7.64×10^{-7}	7.64×10^{-8}	7.64×10^{-8}	3.36×10^{-10}
I-135	1.90×10^{-3}	1.90×10^{-3}	1.90×10^{-3}	1.90×10^{-4}	3.17×10^{-6}	3.17×10^{-7}	3.17×10^{-7}	1.40×10^{-9}
Cs-134	0.409	0.409	0.409	4.09×10^{-3}	1.03×10^{-6}	1.03×10^{-7}	1.03×10^{-7}	4.52×10^{-10}
Cs-136	7.40×10^{-2}	7.40×10^{-2}	7.40×10^{-2}	7.40×10^{-4}	1.85×10^{-7}	1.85×10^{-8}	1.85×10^{-8}	8.12×10^{-11}
Cs-137	1.27	1.27	1.27	1.27×10^{-2}	3.18×10^{-6}	3.18×10^{-7}	3.18×10^{-7}	1.40×10^{-9}
Cs-138	7.34×10^{-2}	7.34×10^{-2}	7.34×10^{-2}	7.34×10^{-4}	1.84×10^{-7}	1.84×10^{-8}	1.84×10^{-8}	8.08×10^{-11}
Ba-137m	1.17	1.17	1.17	1.17×10^{-2}	2.93×10^{-6}	2.93×10^{-7}	2.93×10^{-7}	1.29×10^{-9}
Ba-139	7.60×10^{-5}	7.60×10^{-5}	7.60×10^{-5}	7.60×10^{-6}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Ba-140	5.63×10^{-6}	5.63×10^{-6}	5.63×10^{-6}	5.63×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
La-140	2.25×10^{-6}	2.25×10^{-6}	2.25×10^{-6}	2.25×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Ce-144	5.18×10^{-7}	5.18×10^{-7}	5.18×10^{-7}	5.18×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}

- (1) The following information pertaining to estimated doses was provided as part of the initial safety analysis. Operations of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated dose information is considered historical.

TABLE 11.2-14

Total Annual Activity Released to Lake from Clean Liquid Radioactive Waste System (1)

<u>Radionuclide</u>	<u>Annual Release (Ci)</u>
<u>Insoluble Corrosion Products</u>	
Cr-51	9.02×10^{-5}
Mn-54	1.04×10^{-5}
Fe-55	3.57×10^{-4}
Fe-59	1.04×10^{-5}
Co-58	5.43×10^{-4}
Co-60	9.77×10^{-6}
Zr-95	7.11×10^{-4}
<u>Fission Products (gaseous)</u>	
Ar-41	1.09×10^{-3}
Kr-83m	2.82×10^{-4}
Kr-85m	1.50×10^{-3}
Kr-85	3.08×10^{-3}
Kr-87	8.18×10^{-4}
Kr-88	2.62×10^{-3}
Xe-131m	2.20×10^{-3}
Xe-133m	2.78×10^{-3}
Xe-133	0.244
Xe-135m	9.05×10^{-4}
Xe-135	4.97×10^{-3}
Xe-138	5.00×10^{-4}
<u>Fission Products (solid-ionic)</u>	
Rb-88	9.77×10^{-6}
Sr-89	9.77×10^{-6}
Sr-90	9.77×10^{-6}
Sr-91	9.77×10^{-6}
Sr-92	9.77×10^{-6}
Y-90	9.77×10^{-6}
Y-91	9.77×10^{-6}
Mo-99	1.01×10^{-4}
I-131	5.27×10^{-4}
I-132	3.67×10^{-4}
I-133	6.16×10^{-4}
I-134	7.47×10^{-5}
I-135	3.10×10^{-4}

TABLE 11.2-14 (Continued)

Total Annual Activity Released to Lake from Clean Liquid Radioactive Waste System (1)

<u>Radionuclide</u>	<u>Annual Release (Ci)</u>
Cs-134	1.01×10^{-4}
Cs-136	1.81×10^{-5}
Cs-137	3.11×10^{-4}
Cs-138	1.80×10^{-5}
Ba-137m	2.87×10^{-4}
Ba-139	9.77×10^{-6}
Ba-140	9.77×10^{-6}
La-140	9.77×10^{-6}
Ce-144	9.77×10^{-6}

Total Gaseous Activity = 0.265 Ci

Total Non-Gaseous Activity = 4.58×10^{-3} Ci

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.2-15

Maximum Specific Activities at Significant Points in the Processing Cycle for Non-Detergent Miscellaneous Liquid Radioactive Waste (2)

Radionuclide	Max. Sp. Act. in System Feed (Corrected for temp.) ($\mu\text{Ci/ml}$)	Max. Sp. Act. After Waste Evaporator ⁽¹⁾ ($\mu\text{Ci/ml}$)	Max. Sp. Act. After Waste Polishing Demineralizer ($\mu\text{Ci/ml}$)	Max. Sp. Act. After Misc. Waste Monitor Tank Filter ($\mu\text{Ci/ml}$)	Max. Sp. Act. in Waste Discharged to Lake ($\mu\text{Ci/ml}$)
<u>Ionic Radionuclides</u>					
Rb-88	2.70	2.70×10^{-3}	2.70×10^{-5}	2.70×10^{-5}	2.70×10^{-8}
Sr-89	4.51×10^{-3}	4.51×10^{-6}	4.57×10^{-8}	4.57×10^{-8}	4.51×10^{-11}
Sr-90	1.46×10^{-4}	1.46×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-11}
Sr-91	2.85×10^{-2}	2.85×10^{-5}	2.85×10^{-7}	2.85×10^{-7}	2.85×10^{-10}
Sr-92	8.72×10^{-3}	8.72×10^{-6}	8.72×10^{-8}	8.72×10^{-8}	8.72×10^{-11}
Y-90	1.01×10^{-2}	1.01×10^{-5}	1.01×10^{-7}	1.01×10^{-7}	1.01×10^{-10}
Y-91	5.66×10^{-2}	5.66×10^{-5}	5.66×10^{-7}	5.66×10^{-7}	5.66×10^{-10}
Mo-99	4.12	4.12×10^{-3}	4.12×10^{-5}	4.12×10^{-5}	4.12×10^{-8}
I-131	3.23	1.62×10^{-2}	1.62×10^{-4}	1.62×10^{-4}	1.62×10^{-7}
I-132	2.25	1.13×10^{-2}	1.13×10^{-4}	1.13×10^{-4}	1.13×10^{-7}
I-133	3.78	1.89×10^{-2}	1.89×10^{-4}	1.89×10^{-4}	1.89×10^{-7}
I-134	0.458	2.29×10^{-3}	2.29×10^{-5}	2.29×10^{-5}	2.29×10^{-8}
I-135	1.90	9.50×10^{-3}	9.50×10^{-5}	9.50×10^{-5}	9.50×10^{-8}
Cs-134	4.09	4.09×10^{-3}	4.09×10^{-5}	4.09×10^{-5}	4.09×10^{-8}
Cs-136	0.740	7.40×10^{-4}	7.40×10^{-6}	7.40×10^{-6}	7.40×10^{-9}
Cs-137	12.7	1.27×10^{-2}	1.27×10^{-4}	1.27×10^{-4}	1.27×10^{-7}
Cs-138	0.734	7.34×10^{-4}	7.34×10^{-6}	7.34×10^{-6}	7.34×10^{-9}
Ba-137m	11.7	1.17×10^{-2}	1.17×10^{-4}	1.17×10^{-4}	1.17×10^{-7}
Ba-139	7.60×10^{-2}	7.60×10^{-5}	7.60×10^{-7}	7.60×10^{-7}	7.60×10^{-10}
Ba-140	5.63×10^{-3}	5.63×10^{-6}	5.63×10^{-8}	5.63×10^{-8}	5.63×10^{-11}
La-140	2.25×10^{-3}	2.25×10^{-6}	2.25×10^{-8}	2.25×10^{-8}	2.25×10^{-11}
Ce-144	5.18×10^{-4}	5.18×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-11}
<u>Corrosion Products</u>					
Cr-51	22.12	2.22×10^{-2}	2.22×10^{-2}	2.22×10^{-3}	2.22×10^{-6}
Mn-54	2.54	2.54×10^{-3}	2.54×10^{-3}	2.54×10^{-4}	2.54×10^{-7}
Fe-55	87.44	8.75×10^{-2}	8.75×10^{-2}	8.75×10^{-3}	8.75×10^{-6}
Fe-59	2.54	2.54×10^{-3}	2.54×10^{-3}	2.54×10^{-4}	2.54×10^{-7}
Co-58	132.84	0.133	0.133	1.33×10^{-2}	1.33×10^{-2}
Co-60	0.72	7.12×10^{-4}	7.12×10^{-4}	7.12×10^{-5}	7.12×10^{-8}
Zr-95	174.04	0.175	0.175	1.75×10^{-2}	1.75×10^{-2}

(1) Waste Evaporator is no longer in use.

(2) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.2-16

Average Specific Activities at Significant Points in the Processing Cycle for Non-Detergent Miscellaneous Liquid Radioactive Waste * (2)

Radionuclide	Avg. Sp. Act. in System Feed (Corrected for temp.) ($\mu\text{Ci/ml}$)	Avg. Sp. Act. After Waste Evaporator ⁽¹⁾ ($\mu\text{Ci/ml}$)	Avg. Sp. Act. After Waste Polishing Demineralizer ($\mu\text{Ci/ml}$)	Avg. Sp. Act. After Misc. Waste Monitor Tank Filter ($\mu\text{Ci/ml}$)	Avg. Sp. Act. in Waste Discharged to Lake ($\mu\text{Ci/ml}$)
<u>Ionic Radionuclides</u>					
Rb-88	8.00×10^{-2}	4.00×10^{-5}	4.00×10^{-7}	4.00×10^{-7}	1.76×10^{-9}
Sr-89	1.34×10^{-4}	6.70×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Sr-90	4.33×10^{-6}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Sr-91	8.44×10^{-4}	4.22×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Sr-92	2.59×10^{-4}	1.30×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Y-90	3.00×10^{-4}	1.50×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Y-91	1.68×10^{-3}	8.40×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Mo-99	0.122	6.10×10^{-5}	6.10×10^{-7}	6.10×10^{-7}	2.68×10^{-9}
I-131	9.57×10^{-2}	2.40×10^{-4}	2.40×10^{-6}	2.40×10^{-6}	1.06×10^{-8}
I-132	6.67×10^{-2}	1.67×10^{-4}	1.67×10^{-6}	1.67×10^{-6}	7.33×10^{-9}
I-133	0.112	2.80×10^{-4}	2.80×10^{-6}	2.80×10^{-6}	1.23×10^{-8}
I-134	1.36×10^{-2}	3.40×10^{-5}	3.40×10^{-7}	3.40×10^{-7}	1.50×10^{-9}
I-135	5.63×10^{-2}	1.41×10^{-4}	1.41×10^{-6}	1.41×10^{-6}	6.19×10^{-9}
Cs-134	0.122	6.10×10^{-5}	6.10×10^{-7}	6.10×10^{-7}	2.68×10^{-9}
Cs-136	2.20×10^{-2}	1.10×10^{-5}	1.10×10^{-7}	1.10×10^{-7}	4.83×10^{-10}
Cs-137	0.376	1.88×10^{-4}	1.88×10^{-6}	1.88×10^{-6}	8.25×10^{-9}
Cs-138	2.18×10^{-2}	1.09×10^{-5}	1.09×10^{-7}	1.09×10^{-7}	4.79×10^{-10}
Ba-137m	0.347	1.74×10^{-4}	1.74×10^{-6}	1.74×10^{-6}	7.64×10^{-9}
Ba-139	2.25×10^{-3}	1.13×10^{-6}	1.13×10^{-8}	1.13×10^{-8}	4.96×10^{-11}
Ba-140	1.67×10^{-4}	8.35×10^{-3}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
La-140	6.67×10^{-5}	3.34×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Ce-144	1.54×10^{-5}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
<u>Corrosion Products</u>					
Cr-51	0.364	1.82×10^{-4}	1.82×10^{-4}	1.82×10^{-5}	7.99×10^{-8}
Mn-54	4.17×10^{-2}	2.09×10^{-5}	2.09×10^{-5}	2.09×10^{-6}	9.17×10^{-9}
Fe-55	1.44	7.20×10^{-4}	7.20×10^{-4}	7.20×10^{-5}	3.16×10^{-7}
Fe-59	4.17×10^{-2}	2.09×10^{-5}	2.09×10^{-5}	2.09×10^{-6}	9.17×10^{-9}
Co-58	2.19	1.10×10^{-3}	1.10×10^{-3}	1.10×10^{-4}	4.83×10^{-7}
Co-60	1.18×10^{-2}	5.90×10^{-6}	5.90×10^{-6}	5.90×10^{-7}	2.59×10^{-9}
Zr-95	2.87	1.44×10^{-3}	1.44×10^{-3}	1.44×10^{-4}	6.32×10^{-7}

* Excluding any contribution from the condensate polishing demineralizer backwash

(1) Waste Evaporator is no longer in use.

(2) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.2-17

Average Specific Activities at Significant Points in the Processing Cycle for Non-Detergent Miscellaneous Liquid Radioactive Waste * (2)

Radionuclide	Avg. Sp. Act. in System Feed (Corrected for temp.) ($\mu\text{Ci/ml}$)	Avg. Sp. Act. After Waste Evaporator ⁽¹⁾ ($\mu\text{Ci/ml}$)	Avg. Sp. Act. After Waste Polishing Demineralizer ($\mu\text{Ci/ml}$)	Avg. Sp. Act. After Misc. Waste Monitor Tank Filter ($\mu\text{Ci/ml}$)	Avg. Sp. Act. in Waste Discharged to Lake ($\mu\text{Ci/ml}$)
<u>Ionic Radionuclides</u>					
Rb-88	3.22×10^{-2}	1.61×10^{-5}	1.61×10^{-7}	1.61×10^{-7}	7.07×10^{-10}
Sr-89	5.37×10^{-5}	2.69×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Sr-90	1.74×10^{-6}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Sr-91	3.40×10^{-4}	1.70×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Sr-92	1.04×10^{-4}	5.20×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Y-90	1.20×10^{-4}	6.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Y-91	6.74×10^{-4}	3.37×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Mo-99	4.91×10^{-2}	2.46×10^{-5}	2.46×10^{-7}	2.46×10^{-7}	1.08×10^{-9}
I-131	3.85×10^{-2}	9.63×10^{-5}	9.63×10^{-7}	9.63×10^{-7}	4.23×10^{-9}
I-132	2.68×10^{-2}	6.70×10^{-5}	6.70×10^{-7}	6.70×10^{-7}	2.94×10^{-9}
I-133	4.57×10^{-2}	1.13×10^{-4}	1.13×10^{-6}	1.13×10^{-6}	4.96×10^{-9}
I-134	5.46×10^{-3}	1.37×10^{-5}	1.37×10^{-7}	1.37×10^{-7}	6.01×10^{-10}
I-135	2.27×10^{-2}	5.68×10^{-5}	5.68×10^{-7}	5.68×10^{-7}	2.50×10^{-9}
Cs-134	4.87×10^{-2}	2.44×10^{-5}	2.44×10^{-7}	2.44×10^{-7}	1.08×10^{-9}
Cs-136	8.82×10^{-3}	4.41×10^{-6}	4.41×10^{-8}	4.41×10^{-8}	1.94×10^{-10}
Cs-137	0.152	7.60×10^{-5}	7.60×10^{-7}	7.60×10^{-7}	3.34×10^{-9}
Cs-138	8.74×10^{-3}	4.37×10^{-6}	4.37×10^{-8}	4.37×10^{-8}	1.92×10^{-10}
Ba-137m	0.140	7.00×10^{-5}	7.00×10^{-7}	7.00×10^{-7}	3.08×10^{-9}
Ba-139	9.05×10^{-4}	4.53×10^{-7}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Ba-140	6.71×10^{-5}	3.36×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
La-140	2.68×10^{-5}	1.34×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
Ce-144	6.17×10^{-6}	1.00×10^{-8}	1.00×10^{-8}	1.00×10^{-8}	4.39×10^{-11}
<u>Corrosion Products</u>					
Cr-51	0.135	6.75×10^{-5}	6.75×10^{-5}	6.75×10^{-6}	2.97×10^{-8}
Mn-54	1.54×10^{-2}	7.70×10^{-6}	7.70×10^{-6}	7.70×10^{-7}	3.38×10^{-9}
Fe-55	0.531	2.66×10^{-4}	2.66×10^{-4}	2.66×10^{-5}	1.17×10^{-7}
Fe-59	1.54×10^{-2}	7.70×10^{-6}	7.70×10^{-6}	7.70×10^{-7}	3.38×10^{-9}
Co-58	0.805	4.03×10^{-4}	4.03×10^{-4}	4.03×10^{-5}	1.77×10^{-7}
Co-60	4.33×10^{-3}	2.17×10^{-6}	2.17×10^{-6}	2.17×10^{-7}	9.52×10^{-10}
Zr-95	1.06	5.30×10^{-4}	5.30×10^{-4}	5.30×10^{-5}	2.33×10^{-7}

* Including condensate polishing demineralizer backwash

(1) Waste Evaporator is no longer in use.

(2) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

TABLE 11.2-18

Average Specific Activities and Total Activities in Secondary System
Liquid Waste Released to Lake (1)

Radionuclide	Average Sp. Act. in Waste Discharged to Lake $\mu\text{Ci/ml}$	Total Act. in Waste Discharged to Lake Each Year Ci
<u>Ionic Radionuclides</u>		
Rb-88	1.69×10^{-9}	3.05×10^{-3}
Sr-89	7.11×10^{-12}	1.28×10^{-5}
Sr-90	2.28×10^{-13}	4.12×10^{-7}
Sr-91	4.30×10^{-11}	7.75×10^{-5}
Sr-92	1.18×10^{-11}	2.12×10^{-5}
Y-90	1.58×10^{-11}	2.85×10^{-5}
Y-91	8.91×10^{-11}	1.61×10^{-4}
Mo-99	6.45×10^{-9}	1.16×10^{-2}
I-131	5.09×10^{-9}	9.15×10^{-3}
I-132	2.97×10^{-9}	5.35×10^{-3}
I-133	5.84×10^{-9}	1.04×10^{-2}
I-134	4.79×10^{-10}	8.55×10^{-4}
I-135	2.79×10^{-9}	5.05×10^{-3}
Cs-134	6.45×10^{-9}	1.16×10^{-2}
Cs-136	1.17×10^{-9}	2.10×10^{-3}
Cs-137	1.98×10^{-8}	3.58×10^{-2}
Cs-138	6.32×10^{-10}	1.14×10^{-3}
Ba-137m	$1.61 \times 10_9$	2.90×10^{-3}
Ba-139	$9.00 \times 10_{11}$	1.63×10^{-4}
Ba-140	$8.82 \times 10_{12}$	1.59×10^{-5}
La-140	$3.48 \times 10_{12}$	6.30×10^{-6}
Ce-144	$8.12 \times 10_{13}$	1.47×10^{-6}
<u>Corrosion Products</u>		
Cr-51	5.84×10^{-11}	1.05×10^{-4}
Mn-54	6.63×10^{-12}	1.20×10^{-5}
Fe-55	2.28×10^{-10}	4.12×10^{-4}
Fe-59	6.63×10^{-12}	1.20×10^{-5}
Co-58	3.48×10^{-10}	6.30×10^{-4}
Co-60	1.86×10^{-12}	3.35×10^{-6}
Zr-95	4.57×10^{-10}	8.25×10^{-4}
Total		0.102

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

TABLE 11.2-19

Total Annual Activity Released to Lake Due to Non-Detergent Miscellaneous Liquid Waste (1)

<u>Radionuclide</u>	<u>Annual Release (Ci)</u>
<u>Insoluble Corrosion Products</u>	
Cr-51	1.58×10^{-2}
Mn-54	1.81×10^{-3}
Fe-55	6.23×10^{-2}
Fe-59	1.81×10^{-3}
Co-58	9.43×10^{-2}
Co-60	5.08×10^{-4}
Zr-95	0.124
<u>Fission Products (solid-ionic)</u>	
Rb-88	3.77×10^{-4}
Sr-89	2.34×10^{-5}
Sr-90	2.34×10^{-5}
Sr-91	2.34×10^{-5}
Sr-92	2.34×10^{-5}
Y-90	2.34×10^{-5}
Y-91	2.34×10^{-5}
Mo-99	5.76×10^{-4}
I-131	2.26×10^{-3}
I-132	1.57×10^{-3}
I-133	2.65×10^{-3}
I-134	3.21×10^{-4}
I-135	1.33×10^{-3}
Cs-134	5.71×10^{-4}
Cs-136	1.04×10^{-4}
Cs-137	1.78×10^{-3}
Cs-138	1.03×10^{-4}
Ba-137m	1.64×10^{-3}
Ba-139	2.34×10^{-5}
Ba-140	2.34×10^{-5}
La-140	2.34×10^{-5}
Ce-144	2.34×10^{-5}
Total	0.315

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

TABLE 11.2-20

Summary of Annual Releases of Activity in Liquid Effluents (1)

Radionuclide	<u>Annual Releases of Activity due to Individual Sources of Effluent, Ci</u>			Total Annual Release (Ci)
	Clean Liquid Waste	Non-Detergent Misc. Liquid Waste	Secondary System Leakage	
H-3				346
<u>Ionic Radionuclides</u>				
Rb-88	9.77×10^{-6}	3.77×10^{-4}	3.05×10^{-3}	3.44×10^{-3}
Sr-89	9.77×10^{-6}	2.34×10^{-5}	1.28×10^{-5}	4.60×10^{-5}
Sr-90	9.77×10^{-6}	2.34×10^{-5}	4.12×10^{-7}	3.36×10^{-5}
Sr-91	9.77×10^{-6}	2.34×10^{-5}	7.75×10^{-5}	1.11×10^{-4}
Sr-92	9.77×10^{-6}	2.34×10^{-5}	2.12×10^{-5}	5.44×10^{-5}
Y-90	9.77×10^{-6}	2.34×10^{-5}	2.85×10^{-5}	6.17×10^{-5}
Y-91	9.77×10^{-6}	2.34×10^{-5}	1.61×10^{-4}	1.95×10^{-4}
Mo-99	1.01×10^{-4}	5.76×10^{-4}	1.16×10^{-2}	1.23×10^{-2}
I-131	5.27×10^{-4}	2.26×10^{-3}	9.15×10^{-3}	1.20×10^{-2}
I-132	3.67×10^{-4}	1.57×10^{-3}	5.35×10^{-3}	7.29×10^{-3}
I-133	6.16×10^{-4}	2.65×10^{-3}	1.04×10^{-2}	1.37×10^{-2}
I-134	7.47×10^{-5}	3.21×10^{-4}	8.55×10^{-4}	1.26×10^{-3}
I-135	3.10×10^{-4}	1.33×10^{-3}	5.05×10^{-3}	6.69×10^{-3}
Cs-134	1.01×10^{-4}	5.71×10^{-4}	1.16×10^{-2}	1.23×10^{-2}
Cs-136	1.81×10^{-5}	1.04×10^{-4}	2.10×10^{-3}	2.23×10^{-3}
Cs-137	3.11×10^{-4}	1.78×10^{-3}	3.58×10^{-2}	3.79×10^{-2}
Cs-138	1.80×10^{-5}	1.03×10^{-4}	1.14×10^{-3}	1.27×10^{-3}
Ba-137m	2.87×10^{-4}	1.64×10^{-3}	3.35×10^{-2}	3.55×10^{-2}
Ba-139	9.77×10^{-6}	2.34×10^{-5}	1.63×10^{-4}	1.97×10^{-4}
Ba-140	9.77×10^{-6}	2.34×10^{-5}	1.59×10^{-5}	4.91×10^{-5}
La-140	9.77×10^{-6}	2.34×10^{-5}	6.30×10^{-6}	3.95×10^{-5}
Ce-144	9.77×10^{-6}	2.34×10^{-5}	1.47×10^{-6}	3.47×10^{-5}
<u>Gaseous Radionuclides</u>				
Ar-41	1.09×10^{-3}	0	0	1.09×10^{-3}
Kr-83m	2.82×10^{-4}	0	0	2.82×10^{-4}
Kr-85m	1.50×10^{-3}	0	0	1.50×10^{-3}
Kr-85	3.08×10^{-3}	0	0	3.08×10^{-3}
Kr-87	8.18×10^{-4}	0	0	8.18×10^{-4}
Kr-88	2.62×10^{-3}	0	0	2.62×10^{-3}
Xe-131m	2.20×10^{-3}	0	0	2.20×10^{-3}
Xe-133m	2.78×10^{-3}	0	0	2.78×10^{-3}
Xe-133	0.244	0	0	0.244
Xe-135m	9.05×10^{-4}	0	0	9.05×10^{-4}
Xe-135	4.97×10^{-3}	0	0	4.97×10^{-3}
Xe-138	5.00×10^{-4}	0	0	5.00×10^{-4}

TABLE 11.2-20 (Continued)

Summary of Annual Releases of Activity in Liquid Effluents (1)

<u>Annual Releases of Activity due to Individual Sources of Effluent, Ci</u>				
Radionuclide	Clean Liquid Waste	Non-Detergent Misc. Liquid Waste	Secondary System Leakage	Total Annual Release (Ci)
<u>Corrosion Products</u>				
Cr-51	9.02 x 10 ⁻⁵	1.58 x 10 ⁻²	1.05 x 10 ⁻⁴	1.60 x 10 ⁻²
Mn-54	1.04 x 10 ⁻⁵	1.81 x 10 ⁻³	1.20 x 10 ⁻⁵	1.84 x 10 ⁻³
Fe-55	3.57 x 10 ⁻⁴	6.23 x 10 ⁻²	4.12 x 10 ⁻⁴	6.31 x 10 ⁻²
Fe-59	1.04 x 10 ⁻⁵	1.81 x 10 ⁻³	1.20 x 10 ⁻⁵	1.84 x 10 ⁻³
Co-58	5.43 x 10 ⁻⁴	9.43 x 10 ⁻²	6.30 x 10 ⁻⁴	9.55 x 10 ⁻²
Co-60	9.77 x 10 ⁻⁶	5.08 x 10 ⁻⁴	3.35 x 10 ⁻⁶	5.22 x 10 ⁻⁴
Zr-95	7.11 x 10 ⁻⁴	0.124	8.25 x 10 ⁻⁴	0.126
Totals :				
Gaseous Activity	0.265	0	0	0.265
Non-Gaseous Activity	4.58 x 10 ⁻³	0.315	0.133	0.452
Tritium	-	-	-	346

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

Table 11.2-21

Summary of Maximum Individual Doses Due to Liquid Releases ¹

Doses Presented are those Calculated for Adults, mrem/yr

<u>Exposure Pathway</u>	<u>Dose from Maximum Expected Releases</u>				<u>Dose from Design Basis Releases</u>			
	Whole Body	Body Surface	Liver	Thyroid	Whole Body	Body Surface	Liver	Thyroid
Water ingestion	0.644	0.753	0.825	1.27	1.15	1.29	1.80	5.78
Fish ingestion	1.56	1.57	3.49	1.59	15.0	15.0	34.3	15.2
Swimming	0.0038	0.0515	0.0038	0.0038	0.0147	0.0775	0.0147	0.0147
Sunbathing	1.88	2.32	1.88	1.88	7.35	9.57	7.35	7.35
Exposure Totals	4.09	4.69	6.20	4.74	23.5	25.9	43.5	28.3

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

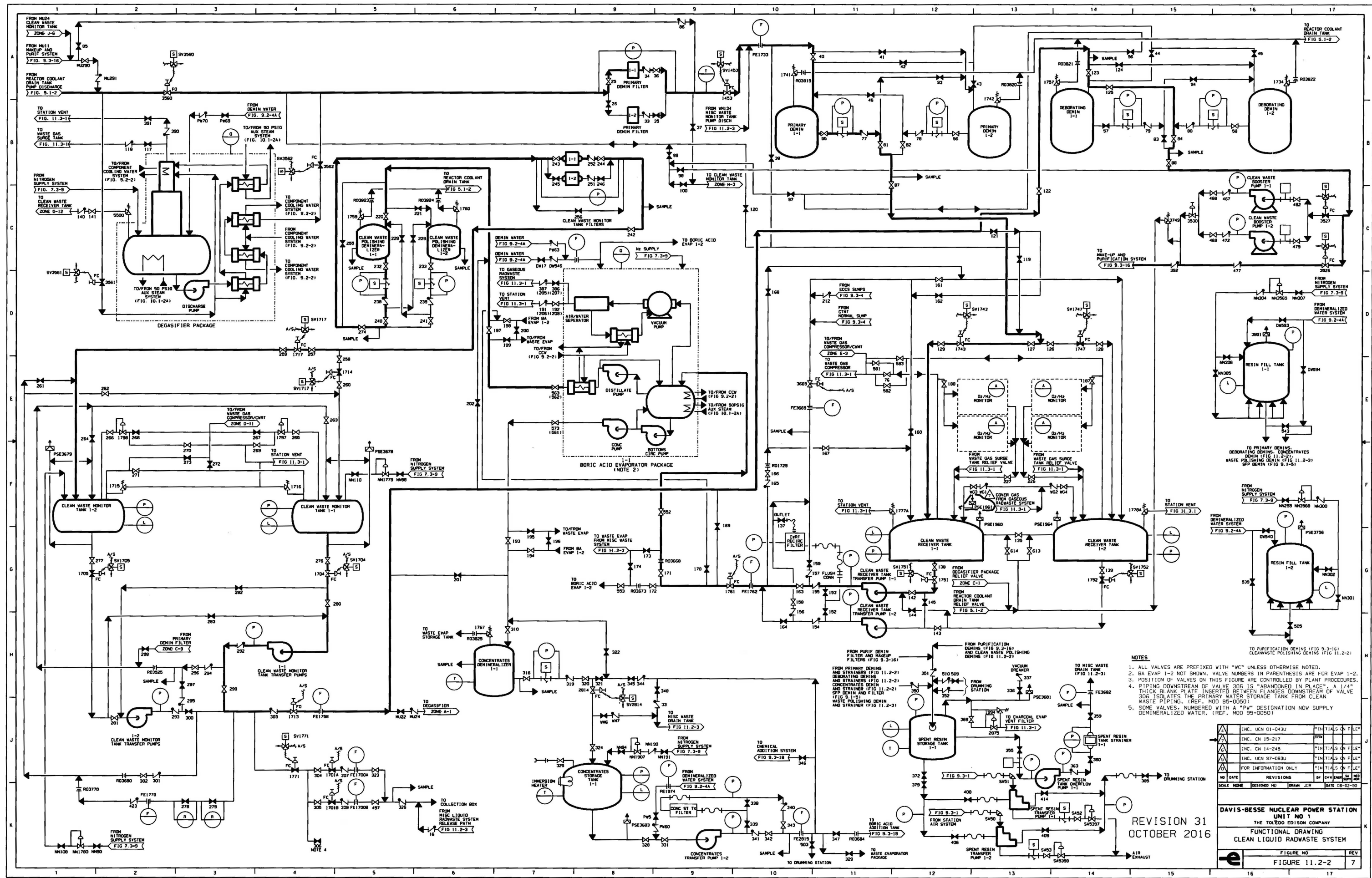
TABLE 11.2-22

Summary of Whole Body Population dose Due to Liquid Releases (1)

	Dose from Maximum Expected Releases, man-rem/year		
	<u>Year 1970</u>	<u>Year 1990</u>	<u>Year 2010</u>
Water Ingestion	0.050	0.072	0.100
Fish Ingestion	0.015	0.021	0.030
Totals*	0.064	0.093	0.129
	Dose from Design Basis Releases, man-rem/year		
	<u>Year 1970</u>	<u>Year 1990</u>	<u>Year 2010</u>
Water Ingestion	0.089	0.128	0.178
Fish Ingestion	0.140	0.204	0.283
Totals*	0.229	0.332	0.461

* Totals may not appear exact due to round-off of subtotals.

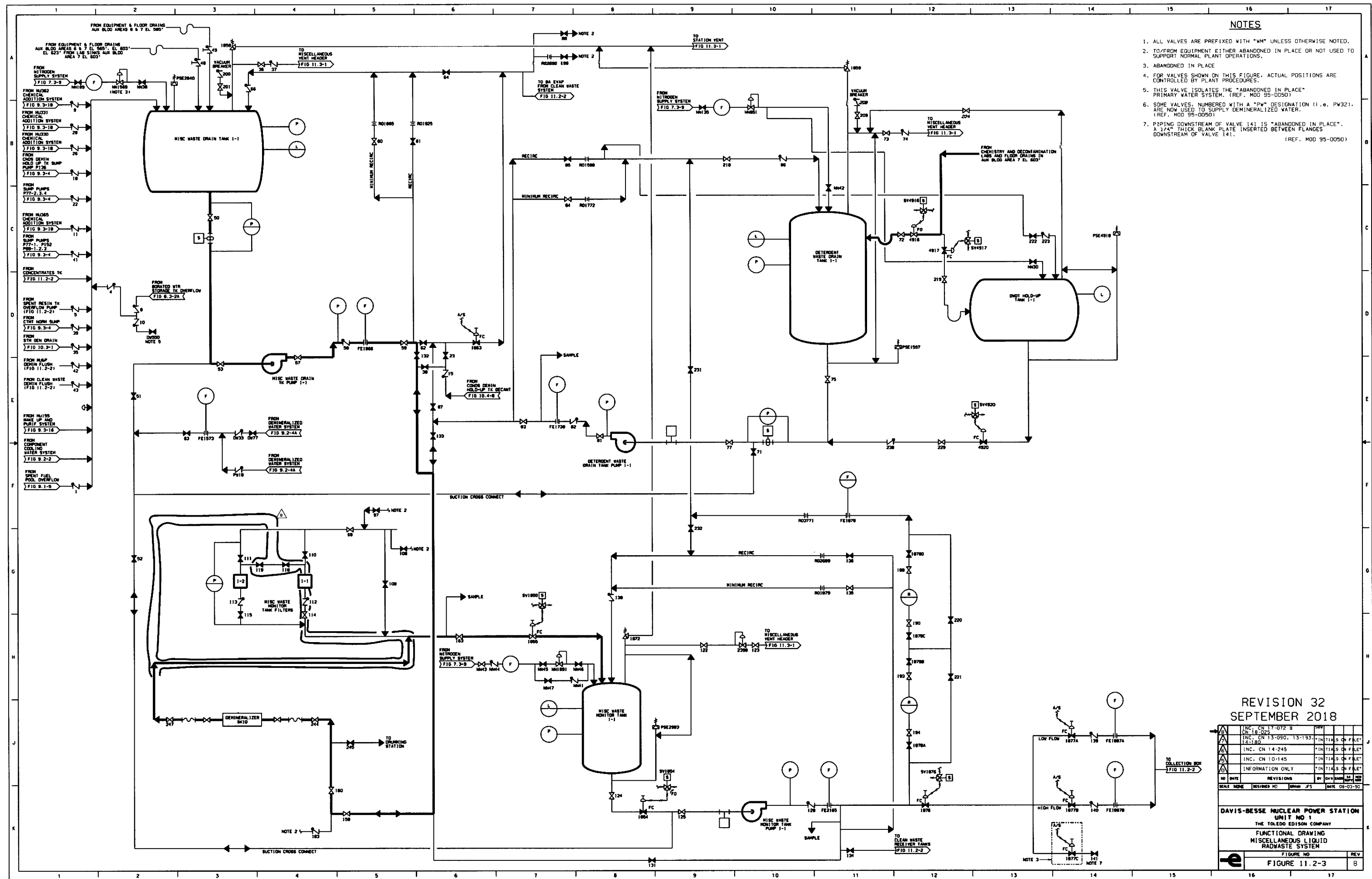
- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.



- NOTES**
1. ALL VALVES ARE PREFIXED WITH "WC" UNLESS OTHERWISE NOTED.
 2. BA EVAP 1-2 NOT SHOWN. VALVE NUMBERS IN PARENTHESIS ARE FOR EVAP 1-2.
 3. POSITION OF VALVES ON THIS FIGURE ARE CONTROLLED BY PLANT PROCEDURES.
 4. PIPING DOWNSTREAM OF VALVE 306 IS "ABANDONED IN PLACE". A 1/4" THICK BLANK PLATE INSERTED BETWEEN FLANGES DOWNSTREAM OF VALVE 306 ISOLATES THE PRIMARY WASTE STORAGE TANK FROM CLEAN WASTE PIPING. (REF. MOD 95-0050)
 5. SOME VALVES, NUMBERED WITH A "P" DESIGNATION NOW SUPPLY DEMINERALIZED WATER. (REF. MOD 95-0050)

REVISION 31
OCTOBER 2016

INC. UCN 01-043U	"INITIALS ON FILE"
INC. CN 15-217	SEM
INC. CN 14-245	"INITIALS ON FILE"
INC. UCN 97-063U	"INITIALS ON FILE"
FOR INFORMATION ONLY	"INITIALS ON FILE"
NO DATE	REVISIONS
SCALE NONE	DESIGNED MD
	DRAWN JR
	DATE 08-02-90
DAVIS-BESSE NUCLEAR POWER STATION	
UNIT NO 1	
THE TOLEDO EDISON COMPANY	
FUNCTIONAL DRAWING	
CLEAN LIQUID RADWASTE SYSTEM	
FIGURE NO	REV
FIGURE 11.2-2	7



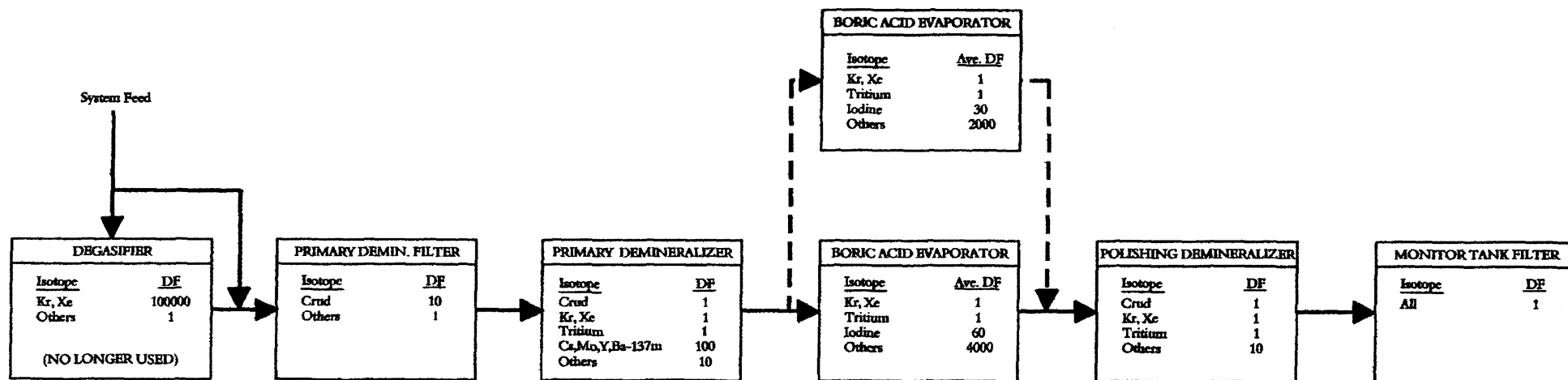
NOTES

1. ALL VALVES ARE PREFIXED WITH "WM" UNLESS OTHERWISE NOTED.
2. TO/FROM EQUIPMENT EITHER ABANDONED IN PLACE OR NOT USED TO SUPPORT NORMAL PLANT OPERATIONS.
3. ABANDONED IN PLACE
4. FOR VALVES SHOWN ON THIS FIGURE, ACTUAL POSITIONS ARE CONTROLLED BY PLANT PROCEDURES.
5. THIS VALVE ISOLATES THE "ABANDONED IN PLACE" PRIMARY WATER SYSTEM. (REF. MOD 95-0050)
6. SOME VALVES, NUMBERED WITH A "PW" DESIGNATION (I.E. PW32), ARE NOW USED TO SUPPLY DEMINERALIZED WATER. (REF. MOD 95-0050)
7. PIPING DOWNSTREAM OF VALVE 141 IS "ABANDONED IN PLACE". A 1/4" THICK BLANK PLATE INSERTED BETWEEN FLANGES DOWNSTREAM OF VALVE 141. (REF. MOD 95-0050)

REVISION 32
SEPTEMBER 2018

NO	DATE	REVISIONS	BY	CHKD	DATE
1	12-18-80	INC. CN 13-090, 13-193, 12-180
2	12-18-80	INC. CN 14-245
3	12-18-80	INC. CN 10-145
4	12-18-80	INFORMATION ONLY

DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO 1
THE TOLEDO EDISON COMPANY
FUNCTIONAL DRAWING
MISCELLANEOUS LIQUID
RADWASTE SYSTEM
FIGURE NO 11.2-3
REV 8

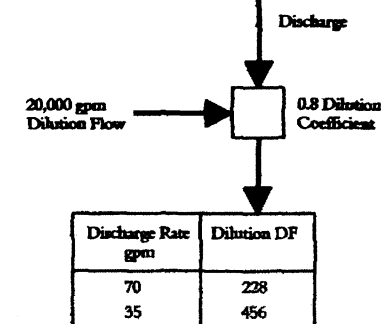


Maximum Accumulative System DF s (excluding the effects of dilution)*

Isotope	Accumulative DF (min.)	Accumulative DF (ave.)
Tritium	1	1
Kr, Xe	1.0×10^5	1.0×10^5
Crud	2.0×10^4	4.0×10^4
Iodine	3.0×10^3	6.0×10^3
Cs,Mo,y,Ba-137m	2.0×10^6	4.0×10^6
Kr, Xe	2.0×10^5	4.0×10^5

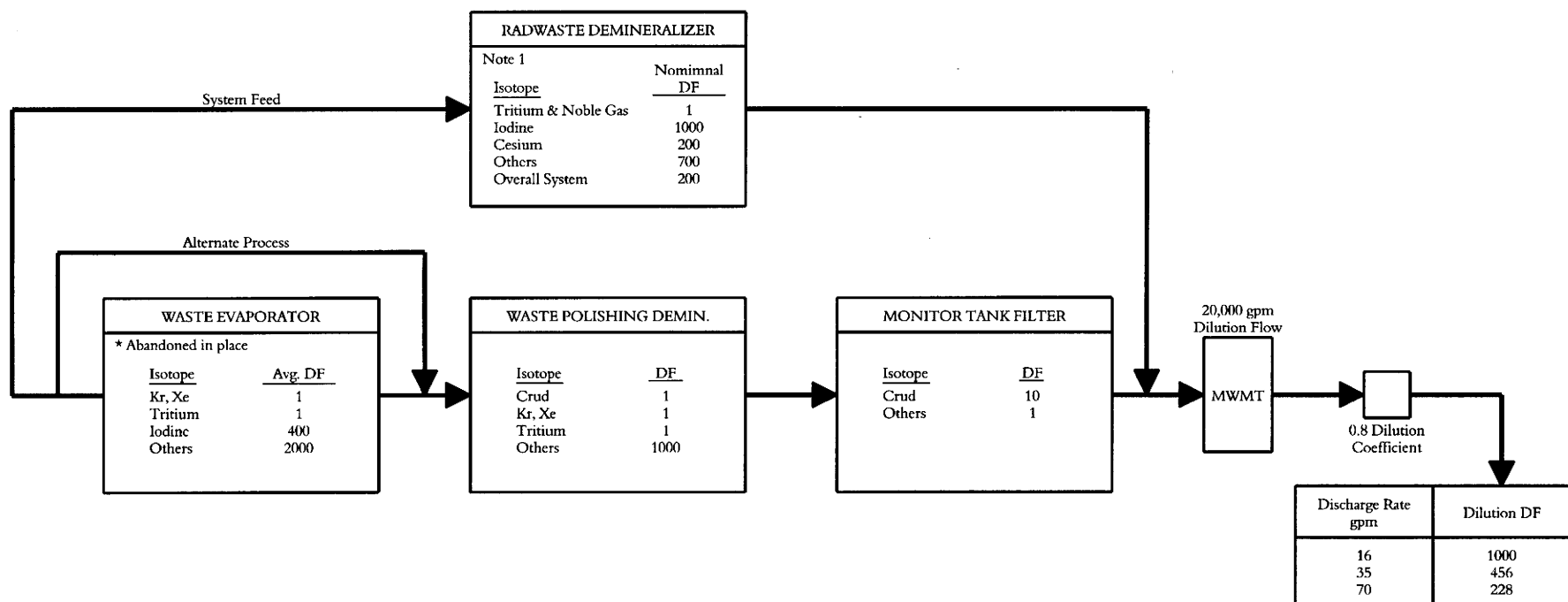
* In actual practice, there are many instances where it is not possible to take credit for the entire accumulative DF associated with this system. This results because a minimum limit of 1.0×10^{-8} uc/ml has been placed on the concentration of any individual radionuclide in undiluted waste.

1. The following information pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design. The ODCM provides information pertaining to the monitoring of waste streams and ensures the resultant offsite doses are in compliance with current regulatory requirements. The following DF information is considered historical.



DAVIS-BESSE NUCLEAR POWER STATION
CLEAN LIQUID RADIOACTIVE WASTE SYSTEM
FLOW SCHEMATIC-NORMAL PROCESSING CYCLE (1)
FIGURE 11.2-4

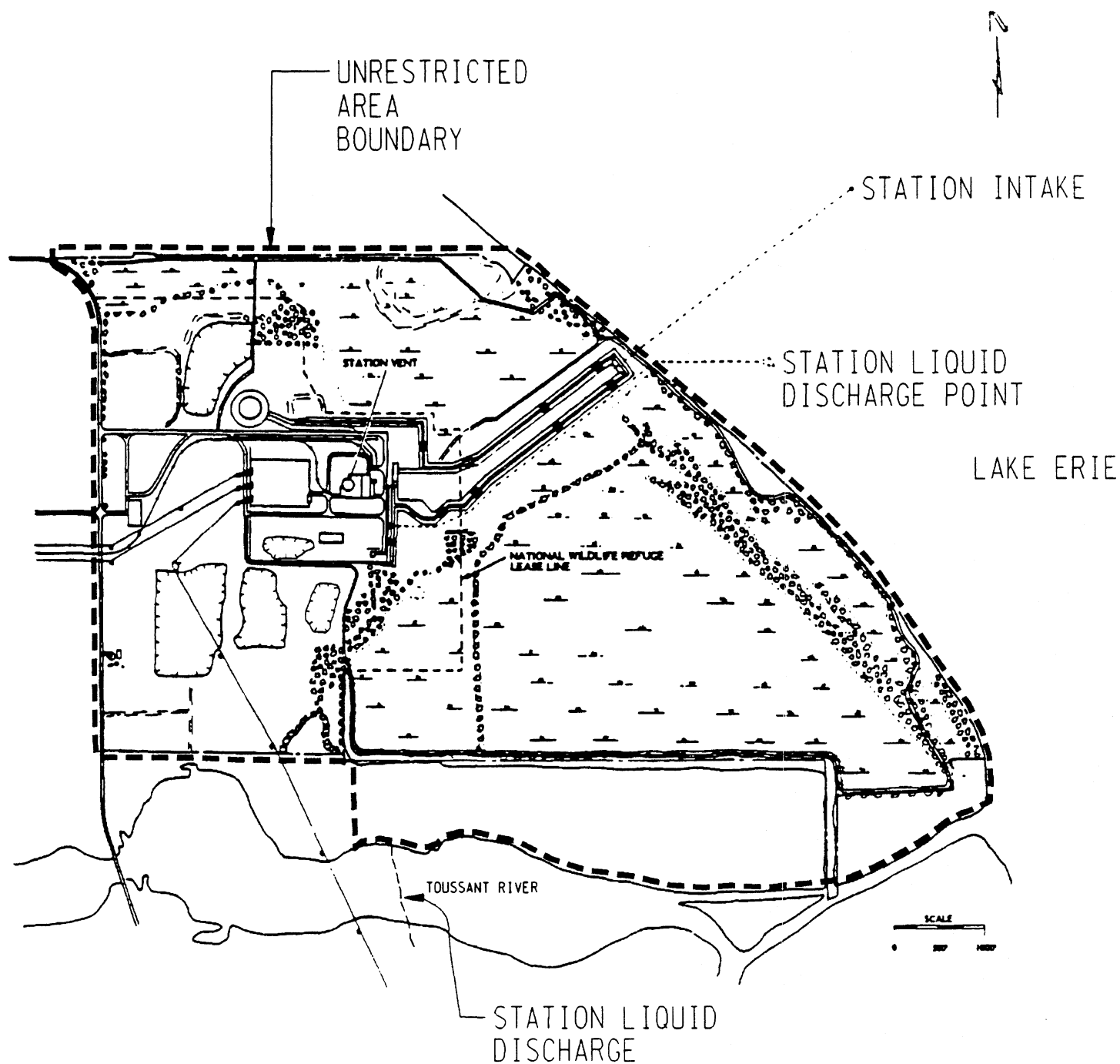
REVISION 26
JUNE 2008



1. Radwaste Demineralizer DF depends on demineralizer arrangement. Sampling of the MWMT and reprocessing are used to control system performance.
2. The following information pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design. The ODCM provides information pertaining to the monitoring of waste streams and ensures the resultant offsite doses are in compliance with current regulatory requirements. The following DF information is considered historical.

DAVIS-BESSE NUCLEAR POWER STATION
MISCELLANEOUS LIQUID RADIOACTIVE WASTE SYSTEM
FLOW SCHEMATIC-NORMAL PROCESSING CYCLE
FIGURE 11.2-5

REVISION 32
SEPTEMBER 2018



DAVIS-BESSE NUCLEAR POWER STATION
RELEASE POINTS OF RADIOACTIVITY
FIGURE 11.2-6

REVISION 22
NOVEMBER 2000

11.3 GASEOUS WASTE SYSTEM

11.3.1 Design Objectives

The system is designed so that estimated releases of gaseous effluents from the station comply with the following requirements of 10CFR20 and 10CFR50:

- a. The individual radionuclide concentrations in gaseous effluents at the site boundary shall not exceed the limits from releases to unrestricted areas given in Appendix B of 10CFR20.
- b. The releases of radioactivity from the station shall comply with the as low as reasonably achievable standard set forth in 10CFR50.

Davis-Besse commits to the regulatory position of Regulatory Guide 4.15 (Revision 1, February 1979).

11.3.2 System Description

11.3.2.1 Process Arrangement

The functional drawing for the portion of this system handling unaerated hydrogenated gases and some of the cover gases is shown in Figure 11.3-1. A list of major components as well as design information for each, is contained in Table 11.3-1. A flow schematic of the entire system is shown in Figure 11.3-2.

11.3.2.2 Waste Sources

Radiation sources, activities quantities and concentrations are dependent upon various factors such as, the fuel load (enrichment of fuel) and the operating cycle length. The current information is maintained in the ODCM and the source term analysis for the current cycle reload report.

11.3.2.2.1 Unaerated Hydrogen-Containing Gas

These gases are the primary concern of this system. They are present in relatively large quantities and contain most of the gaseous radioactivity that will be processed. In general, they contain minimal oxygen and often large amounts of hydrogen.

Sources include off-gas from the Reactor Coolant System in the Clean Waste Receiver Tanks, the gases vented from the boric acid evaporators (if aligned to the gaseous radwaste system) and reactor coolant drain tank. Significant quantities of waste can also be produced by such infrequent events as the venting of the makeup tank or pressurizer or the relieving of the quench tank relief valve. None of the components is specifically designed to withstand a hydrogen explosion for the following reasons:

- a. The condition under which a hydrogen explosion may take place will never arise. That condition is greater than 5% oxygen by volume in a hydrogen-rich atmosphere.
- b. There is no source input to the waste gas surge tank which contains sufficient amounts of oxygen. The major waste sources originate in primary reactor

Davis-Besse Unit 1 Updated Final Safety Analysis Report

coolant which has been calculated to contain much less than 1% oxygen (see Subsection 9.3.2).

- c. Oxygen/hydrogen monitors are installed on the waste gas surge tank to provide an additional measure of safety against the unlikely event of oxygen/hydrogen buildup. The concentration of oxygen shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume. An oxygen or hydrogen concentration above the specified limit will actuate a local and control room alarm.

Additional oxygen/hydrogen monitors are provided on each Clean Waste Receiver Tank.

- d. Should oxygen somehow contaminate one of the input sources, the condition will be alarmed and the operator will take steps to stop the introduction of oxygen.
- e. The waste gas surge tank is maintained at a net positive pressure with a nitrogen blanketing system to preclude the inleakage of air.

The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

A list of the various sources, along with the quantities of each that can be expected, is given in the table below. This table also contains the assumptions used in arriving at the values given.

<u>Gaseous Waste</u>		
<u>Waste Source</u>	<u>Quantity per year (STP)</u>	<u>Assumptions and Comments</u>
Off-gas from reactor coolant system	6,900 ft ³	Degas at 40 cc H ² per kg concentration. Assumes one 50% power transient per wk.
Off-gas from liquid sampling	120 ft ³	Degas at 40 cc H ² per kg concentration
Off-gas from makeup tank	900 ft ³	Vent once per year
Off-gas from pressurizer	60 ft ³	Vent once per year

The maximum continuous production of gases of this type is taken as that resulting from degassing primary coolant at 140 gpm. Based on a dissolved hydrogen concentration of 40 cc/kg, this results in a production rate of about 0.749 ft³/min. The maximum specific activity levels in this gas, for both a 0.1% and 0.1% failed fuel condition, are given in Table 11.3-2. These values were determined using the following assumptions:

- a. All the H₂, Xe, and Kr and 1/1000 of the iodines are stripped from primary coolant as it is processed through the degasifier.
- b. The concentration of dissolved hydrogen in the degasifier feed is 15 cc/kg of feed while the specific activities of Xe and Kr are equal to the maximum levels they

attain in reactor coolant during an equilibrium fuel cycle. The concentrations of the iodines are based on this latter assumption except that all coolant iodine levels are reduced by a factor of 100. This reduction takes into account the fact that normally the primary coolant processed by the degasifier has previously been passed through the purification demineralizer.

Because of decay, there may be some radioactive particulate matter in this class of gaseous waste. To estimate the maximum amount of such activity that could be present, the following simplifying assumptions were used.

- a. The amount of gases available to contribute decay products consist of all Xe, Kr and Iodine released from the fuel elements during a cycle with 0.1% failed fuel.
- b. All gases are assumed to have undergone at least 30 days of decay.

Table 11.3-3 lists each gaseous radionuclide, gives its decay path, and indicates the maximum amount of radioactive decay product, if any, that can be expected to form. It is noted that, even with the very conservative assumptions used, the only particulate matter that must be considered are small quantities of long-lived Rb-87 and Cs-135 (note: as these quantities should account for all possible Rb-87 and Cs-135 produced through decay each year, no release of these isotopes is assumed from any other gas source).

This ends the historical information contained in this subsection.

11.3.2.2.2 Aerated Gases

This classification covers those gases which, while consisting mostly of nitrogen, can be expected to contain significant amounts of oxygen but little radioactivity. This includes displaced cover and/or purge gases from the detergent waste drain tank, the condensate polishing demineralizer hold-up tank, and gasses stripped in the boric acid evaporators. (Note: The boric acid evaporators are a source only if they are aligned to the gaseous radwaste system. They are normally aligned to the station vent.) Due to the random nature of the sources, it is not possible to make a realistic estimate of the quantities of these gases that are produced over a given period.

The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

The small amounts of radioactivity present in these gases come primarily from the gases that evolve from the liquid wastes in the equipment being vented. As these liquids should have already had extensive contact with the atmosphere the xenon and krypton radionuclides are assumed to have already come out of solution and therefore, do not present a problem. However, the iodines are still present and must be considered as a potential radiological hazard.

Where possible, estimates of specific activities, activity release rates, total annual volume released and the total annual activity released have been made for each of the potential sources and are given in Table 11.3-4. These were based on the assumption listed below:

Davis-Besse Unit 1 Updated Final Safety Analysis Report

In all the tanks being considered, the ratio of the iodine concentration in the liquid phase to that in the vapor space is 1000 (i.e. the iodine partition factor is 1000, see Subsection 11.3.2.3 for justification).

Miscellaneous Waste Drain Tank

- a. The maximum iodine concentrations in liquid in this tank are the values given in the last column of Table 11.2-6.
- b. The average iodine concentrations in the liquid are those given in Column (5) of Table 11.2-7. (Note - condensate polishing demineralizer backwash waste does not pass through this tank).
- c. The total liquid passing through this tank in a year is approximately 210,000 gallons (see Table 11.2-5 for a breakdown of the contributing sources). Any liquid entering the tank displaces an equal amount of gas which is then vented.
- d. The rate of liquid addition is 160 gpm. This is the maximum design capacity of any of the radwaste system pumps.

Detergent Waste Drain Tank

The activity levels in detergent waste are too low to cause significant contamination in the gases with which it comes into contact (see Subsection 11.2.2.2.2).

Evaporator Storage Tank (No longer used)

- a. The maximum iodine concentrations in the liquid in this tank are 100 times the corresponding values specified above for the miscellaneous waste drain tank. The multiplier takes into account the concentrating effect of the evaporator.
- b. The average iodine concentrations in the liquid in this tank are 25 times the values listed in Column (6) of Table 11.2-7.
- c. The total liquid entering this tank annually (and displacing an equal amount of gas) is approximately 24,000 gallons. This corresponds to all the non-detergent miscellaneous and condensate polishing demineralizer backwash wastes concentrated by a factor of 25 in the waste evaporator.
- d. The rate of liquid addition to the tank is 50 gpm. This is the nominal rating of the waste evaporator concentrate discharge.

Condensate Polishing Demineralizer Hold-up Tanks

- a. The maximum and average iodine concentrations in backwash liquid waste are equal to the values given, respectively, in Column (4) of Table 11.2-6 and Column (4) of Table 11.2-7.
- b. The total waste entering these tanks each year is 390,000 gallons. (1 backwash per week).

Waste Evaporator (abandoned in place)

- a. An iodine partition factor of 1000 is applicable to the evaporator (see Subsection 11.3.2.3). The iodine isotopes are assumed to be evolved and to be vented as they enter the evaporator in the feed (i.e. 0.001 of the iodine in 15 gallons of waste is vented each minute).
- b. The evaporator processes waste at a 15 gpm rate.
- c. The maximum and average iodine concentrations in the evaporator feed are equal to the values given, respectively, in the last column of Table 11.2-6 and Column (6) of Table 11.2-7.
- d. The total amount of feed processed through the evaporator is approximately 596,000 gallons. This represents the total estimated quantity of non-detergent miscellaneous liquid waste and condensate polishing demineralizer backwash waste (it contains average iodine concentrations).

This ends the historical information contained in this section.

11.3.2.2.3 Cover Gases

Gases are used as inerting agents in the following radioactive waste system tanks:

Clean waste receiver tanks
Clean waste monitor tanks (optional)
Miscellaneous waste monitor tank (optional)

They consist almost entirely of nitrogen but can contain small amounts of hydrogen, noble gases and iodines which can be picked up from contact with liquid waste. From a radiological point of view, only iodine is considered to be present in sufficient quantities to warrant analysis.

The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

Estimates of isotopic specific activities, activity release rates, total annual volumes released and total annual activity released have been made for those gases that are vented from the tanks mentioned above and which, after processing, are slated to be released from the station. These are presented in Table 11.3-5. The assumptions used as a basis for these estimates are as follows:

In all tanks, the ratio of the iodine concentration in liquid to that in the vapor space is 1000 (i.e. the iodine partition factor is 1000; see Subsection 11.3.2.3 for justification).

Clean Waste Receiver Tanks

- a. The maximum iodine concentrations that occur in liquid contained in these tanks are those listed in Column (5) of Table 11.2-12. These values represent the

maximum reactor coolant specific activities after the coolant has been processed through the purification and primary demineralizers.

- b. The average iodine concentrations in the liquid that is contained in these tanks are those listed in Column (5) of Table 11.2-13. These are derived in the same manner as those in (a) except that the initial reactor coolant activities are average values.
- c. The amount of gas vented is set at 28,000 ft³. This number is relatively small because the cover gas system for these tanks normally operates as a closed cycle with no external releases. It represents about two tank volumes and takes into account the infrequent occurrences that might require that some gas be discharged.
- d. The rate of liquid addition to these tanks (and, therefore, the rate of gas displacement) is 560 gpm. This represents a very unlikely combination of flows from pumps and a relief valve.

Clean Waste Monitor Tank

As the contents of this tank will have been through the entire radwaste processing cycle, the activity levels in it are considered too low to produce significant contamination in the tanks cover gas.

Concentrate Storage Tank

- a. The maximum and average iodine concentrations in the liquid to be contained in this tank will be the same as those for the clean waste receiver tank. This corresponds to a concentration factor of 100 in the boric acid evaporator and a DF of 100 for iodine in the concentrates demineralizer.
- b. The total liquid passing through this tank is 6,000 ft³. This represents about six boric acid addition tanks, which should correspond to more than the total amount of boric acid that would be used during a fuel cycle.
- c. The rate of liquid addition of this tank is about 50 gpm. The nominal concentrate discharge rate of the evaporator is only 15 gpm, but two can be working at the same time.

Miscellaneous Waste Monitor Tank

Since the contents of this tank pass through the entire radwaste cycle, the activity levels in it are not considered large enough to produce significant contamination in the tank's cover gas.

This ends the historical information contained in this subsection.

11.3.2.2.4 Building Ventilation and Purge Gases

Note: The Primary Water System has been abandoned in place.

This class of gases consists almost entirely of air which contains relatively little radioactivity. It is produced as the result of purging the containment vessel and ventilating various areas of the

station. Any radioactive gases present are mainly those that have evolved from system leakage or when applicable, refueling canal or spent fuel pool water.

The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

Estimates of maximum and average radionuclide concentrations, activity release rates and total annual activity released have been made and are given in Tables 11.3-6 through 11.3-10. These values are based on the following assumptions:

- a. Containment Vessel Purge: (Per License Amendment 221 the containment purge isolation valves are closed with control power removed in Modes 1 through 4)

Maximum Concentrations:

1. Iodine, Krypton and Xenon:

Maximum concentrations are equal to containment vessel atmospheric concentrations at the beginning of purge.

Net free volume of containment vessel = $2.834 \times 10^6 \text{ ft}^3$

Introduction of radioactive gases into the containment vessel is due entirely to gases evolved from liquid leakage. Maximum evolution rates are given in Table 11.1-11.

Mixing within the containment vessel atmosphere is uniform and instantaneous.

Time between purges is 90 days (this amounts to about four (4) purges per year). At the beginning of the 90-day period, the containment vessel atmosphere is assumed to be essentially free of radioactivity.

Decay of airborne gaseous activity to radioactive daughter products is taken into account if the prospective parent nuclide is among those listed in Table 11.1-11. If more than one parent is present, the contribution of all is considered. In instances where a parent of a parent is present, the intermediate decay step is ignored when estimating daughter concentrations (i.e., the parent's parent is assumed to decay directly to the daughter).

Cs-138 and Rb-88 are assumed to be in equilibrium, respectively, with any Xe-138 and Kr-88 present in the purge gases. This is done because the effects of decay are taken into account after the gases have evolved from the leakage (see Subsection 11.3.2.2.7).

2. Tritium:

The estimate of the airborne tritium concentration is based on the containment atmosphere being saturated with water vapor derived from liquid leakage into the containment vessel. The saturation temperature is taken to be 120 °F, the maximum normal operating condition assumed for inside the containment. The tritium concentration in the leakage is taken to be 0.4 $\mu\text{Ci/ml}$. This is the arbitrary maximum level being assumed for the primary system (see Subsection 11.2.2.2.3).

Average Concentrations:

1. Iodine, Krypton and Xenon:

With the following exceptions, and additions, the assumptions used are the same as those used in estimating maximum concentrations:

The introduction of radioactive gases into the containment vessel atmosphere is equal to the average evolution rates given in Table 11.1-11.

Purge gas radionuclide concentrations are averaged over the time period covered by the purge.

Length of purge is 16 hours. (This allows time for maintenance in the containment vessel during operation.)

Containment vessel purge rate = 50,000 cfm.

2. Tritium:

The assumptions used are similar to those used for estimating iodine, krypton and xenon average concentrations with the following additions and/or exceptions:

Tritium, which is not listed in Table 11.1-11, is assumed to enter the containment vessel atmosphere entirely through the evaporation of tritiated water from system leakage. Because of the difficulty in estimating what portion of the leakage will evaporate, it is assumed that it all does as soon as it occurs.

Leakage rates into containment vessel are given in Table 11.1-9.

Tritium concentration in all leakage is 0.4 $\mu\text{Ci/ml}$. This represents the upper limit being assumed for the primary system.

Release Rates:

Maximum and average release rates are calculated by multiplying the purge rate by the maximum and average radionuclide concentrations (respectively) in the vented gases. As noted in Subsection 11.1.3.4, purge rate can vary according to

the number of fans used and whether or not charcoal filters are employed. Maximum and average release rates will be based on a 50,000 cfm purge flow.

Total Releases:

Total releases are found by multiplying the average rates times the length of the period over which a purge is assumed to occur (16 hours) times the number of purges per year (four (4)).

b. Containment Vessel Ventilation During Refueling:

As noted in Subsection 11.1.3.2, it is assumed that there is no significant leakage of iodine or noble gases from the core after the reactor shuts down. The only inventory of these isotopes available to contaminate the containment atmosphere is that which remained after the prerefueling purge and that which was contained in the primary coolant mixed with the refueling water.

To facilitate the analysis of this situation, the following simplified operating schedule is assumed:

-16 to 0 hours..... prerefueling purge

0 to 21 days..... refueling operations with refueling canal flooded
the entire time

Maximum Concentrations:

1. Iodine, Krypton and Xenon:

The maximum concentration of an iodine, xenon, or krypton isotope is equal to the sum of the maximum concentrations resulting from (1) activity present when refueling ventilation started and (2) activity that evolved from refueling canal water.

The maximum concentrations due to residual activity are equal to those found at the end of the prerefueling purge (i.e., at the beginning of the refueling ventilation period.) Assumptions used in calculating these end-of-purge concentrations are the same as those used for obtaining maximum concentrations in containment vessel purges.

Maximum iodine concentrations due to evolution from refueling canal water are set at 10^{-10} $\mu\text{Ci/ml}$ for each isotope. While there is no specific justification for using this value, it seems reasonable in view of (1) the slow rate of diffusion associated with low concentrations of iodines (see Subsection 11.3.2.3); (2) the fact that, before the head is removed from the reactor, the coolant that will be mixed with the refueling canal water can be partially decontaminated using the purification or spent fuel pool demineralizers and (3) the few significant problems with iodine reported during operating plant refuelings.

Due to prerefueling degassing, it is assumed that essentially all noble gases have been removed from the primary system prior to the removal

of the reactor head. There is no leakage from the fuel and the production of xenon from iodines present is considered in another section of the USAR. For these reasons, it is assumed that the evolution of xenon and krypton from refueling canal water does not make a significant contribution to maximum activity levels.

2. Tritium:

The maximum concentration of tritium in the containment vessel atmosphere is estimated as:

$$C_{\max} = \frac{EC_t}{F}$$

This equation assumes that equilibrium conditions have been reached.

The value for (E), the rate of evaporation from the refueling canal, is taken as 10.3 ft³/hr. The Meyer Equation (ref. 41) was used to calculate this value because it, among the several methods tried, gave the most conservative results. Assumptions used were:

Temperature of air blowing across refueling canal = 70 °F

Relative humidity of air blowing across refueling canal = 5%

Air speed across surface of refueling canal = 0.1 m/sec

Surface area of refueling canal = 1750 ft²

Temperature of refueling canal water = 120°F

(C_t), the refueling canal water tritium level, is taken to be 0.3 µCi/ml. This represents an upper limit which cannot be exceeded if the activity level in the primary system does not exceed 0.4 µCi/ml. (F), the containment vessel purge rate, is 50,000 cfm.

Average Concentrations:

1. Iodine, Krypton and Xenon:

The average concentrations of these radionuclides are equal to the sum of the individual average concentrations resulting from (1) activity present when refueling ventilation is started and (2) activity that evolved from refueling canal water.

The average concentrations due solely to residual activity are based on a 50,000 cfm ventilation flow rate; uniform and instantaneous mixing within the containment vessel atmosphere, and the initial activity levels present being equal to the average concentrations at the end of the prerefueling purge. The latter are based on the assumptions used in estimating average specific activities in containment vessel purge gases.

Iodine concentrations due to evolution from canal water are assumed to be directly proportional to iodine levels in the canal water. Therefore:

$$\text{Avg. Conc. (air)} = \text{Initial conc, (air)} \left[\frac{\text{Avg.conc.(water)}}{\text{InitialConc.(water)}} \right]$$

The initial iodine concentration in the air is set at 10^{-10} $\mu\text{Ci/cc}$ for each radionuclide (see preceding section on maximum concentrations). The initial concentrations in the canal water are equal to average values listed in Table 11.1-10. The average iodine concentrations in the canal water during the refueling period are based on 140 gpm of the discharge from the decay heat pump being recirculated back to the pump suction through a mixed-bed demineralizer with a DF of 100 for iodines. As with estimates of the maximum concentrations, the evolution of xenons and kryptons from the refueling canal water is not considered.

2. Tritium:

Assumed to be the same as maximum values.

Release Rates:

Maximum and average release rates are calculated by multiplying the ventilation rate by the maximum and average radionuclide concentrations (respectively) inside the containment vessel. As noted previously, all release rates are being based on a 50,000 cfm air flow.

Total Activity Releases:

Total activity releases are found by multiplying the total refueling time (3 weeks) by the average release rates.

c. Auxiliary Building Ventilation (excluding refueling areas):

The auxiliary building ventilation system is assumed to operate continuously. Its normal capacity is 47,160 SCFM if the EVS charcoal filters are not used and either 8,000 SCFM or 16,000 SCFM if they are.

Maximum Concentrations:

1. Iodine, Krypton and Xenon:

Estimates of these concentrations conservatively assume that all gaseous activity is removed by the ventilating air system as soon as it evolves from the leakage. The concentrations are then simply the maximum radionuclide evolution rates divided by the normal flow rate of the ventilation system. The maximum rate of evolution occurs during cooldown for iodine and during normal operation for krypton and xenon (see Table 11.1-11).

2. Tritium:

It is difficult to estimate the maximum concentration of tritium because of the problem of establishing an evaporation rate for liquid leakage (note: evaporation is assumed to be the primary way that tritium finds its way into the atmosphere). Therefore, for calculational purposes, it is conservatively assumed that all leakage evaporates immediately and that all tritium in it gets into the ventilation air flow. It is further assumed that the tritium level in all leakage is 0.4 $\mu\text{Ci/ml}$.

Average Concentrations:

1. Iodine, Krypton and Xenon:

These are calculated the same way as the maximum concentrations except that overall average activity release rates are used.

These rates are the time weighted averages of the averages given in Table 11.1-11 for individual operating situations. The lengths of time per year each of these situations is assumed to occur are as follows:

Normal operation: 292 days

Cooldown (decay heat removal system in use): 16 hr.

Refueling: 21 days

2. Tritium:

Same as maximum concentrations.

Release Rates:

Maximum and average release rates for iodine, krypton and xenon through the ventilation system are equal to the maximum and overall average evolution rates from leakage (see assumptions used in preceding paragraphs for explanations of the terms "maximum" and "overall average" evolution rates). Maximum and average tritium release rates are both equal to the maximum ventilation flow rate (47,160 scfm) times the maximum concentration levels.

Total Activity Releases:

Total releases are found by multiplying the average release rates by the period of time over which they are assumed to occur. The time periods involved are as given in the preceding section entitled "Average Concentrations."

d. Auxiliary Building Ventilation (refueling areas):

The refueling area ventilation system will probably be used only when access to this part of the auxiliary building is required. It has a normal capacity of 20,000 scfm which is reduced to 8,000 or 16,000 scfm if the EVS charcoal filters are utilized.

Maximum Concentrations:

1. Iodine, Krypton and Xenon:

As noted previously, it is being assumed that spent fuel pool water contains very low concentrations of these radionuclides. In view of this, it is further assumed that these radionuclides are not present in this ventilation air in significant amounts.

2. Tritium:

Note: The Primary Water System has been abandoned in place.

Tritium can get into the spent fuel pool water through contact with water from: (1) – refueling canal, (2) – the borated water storage tank. Conservatively, it will be assumed that the spent fuel pool water contains the maximum level of tritium being postulated for the primary system: 0.4 $\mu\text{Ci/ml}$. As in the containment vessel during refueling, the equilibrium concentration of tritium in the atmosphere is estimated as:

$$C_{\max} = \frac{EC_t}{F}$$

Using the Meyer Equation, the value of (E) is estimated as 10.31 ft^3/hr . Assumptions used in this calculation are:

Temperature of air blowing across spent fuel pool water = 70°F

Relative humidity of air blowing across pool water = 5%

Air speed across surface of pool water = 0.1 m/sec

Surface of pool = 1780 ft^2

Temperature of pool water = 120°F

Average Concentrations:

All average concentrations are equal to the corresponding maximum concentrations.

Release Rates:

As only tritium is involved, both the maximum and average activity release rates are found by multiplying the concentration of that isotope in the ventilation air times the maximum air flow (20,000 cfm).

Total Activity Releases:

Total releases are found by multiplying the release rates times the period over which they occur. For calculational purposes, it is assumed that access to the refueling area is required 100 days a year.

e. Turbine Building Ventilation:

The turbine building ventilation system is assumed to operate continuously and to have an overall capacity of about 285,000 cfm. It should be noted that the 285,000 cfm figure takes into account only those fans which appear to service general, rather than specialized, areas of the building.

Maximum Concentrations:

1. Iodine, Krypton and Xenon:

Estimates of these concentrations conservatively assume that all gaseous activity is removed by the ventilating air system as soon as it appears. Assuming that all activity is the result of evolution from secondary system leakage, the concentrations are simply the maximum radionuclide evolution rates (see Table 11.1-11) divided by the flow rate of the ventilation system.

2. Tritium:

The estimate of this concentration based on tritium entering the turbine building atmosphere at the same rate it enters the secondary system. This means that, assuming a 100 gpd primary-to-secondary leak, 10 gpm of secondary system leakage, a tritium concentration of 0.4 $\mu\text{Ci/ml}$ in the primary system, and no radioactive decay, the tritium level in the secondary system builds to about $2.78 \times 10^{-3} \mu\text{Ci/ml}$. If all secondary system leakage immediately vaporizes (a very conservative assumption since half of it is liquid), then 105.16 $\mu\text{Ci/minute}$ of tritium will mix with 285,000 cfm of air to give a maximum concentration of $1.31 \times 10^{-8} \mu\text{Ci/cc}$.

Average Concentrations:

1. Iodine, Krypton and Xenon:

These values are estimated the same way as the maximum concentrations except that the average, rather than the maximum, radionuclide evolution rates given in Table 11.1-11 are used.

2. Tritium:

Same as maximum concentration.

Release Rates:

The maximum and average activity release rates are found by multiplying the maximum and average concentrations of the various radionuclides in the ventilation air times the air flow (285,000 cfm).

Total Activity Releases:

Total releases are determined by multiplying average release rates times the period over which they occur. Since primary and secondary system activity levels are based on effective full power days (EFPD) operation, a release period of 292 days is used. (Note: this is conservative since an equilibrium cycle is anticipated to be only 277 EFPD).

This ends the historical information contained in this subsection.

11.3.2.2.5 Steam Jet Air Ejector Vent

If there is a significant steam generator leak, noble gases and iodines introduced into the secondary system are transported to the condenser where a portion of them are vented, through the steam jet air ejector, to the atmosphere.

The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

Table 11.3-11 contains estimates of release rates and total releases of radioactivity due to this path. The assumptions used in compiling this table are as follows:

Maximum Release Rates:

Based on the assumption given in Subsection 11.1.3.2, Part (d), with primary-to-secondary leakage having activity levels corresponding to 1 % failed fuel, all tritium released to the environment from the secondary system occurs in the turbine building ventilation air flow (see preceding section).

Average Release Rates:

Same assumptions as for maximum release rates except primary-to-secondary coolant activity levels correspond to 0.1 % failed fuel.

Total Annual Activity Releases:

These are the average release rates multiplied by the assuming EFPD operating cycle (292 days).

This ends the historical information contained in this subsection.

11.3.2.2.6 Gaseous Decay Products

The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

As shown in Table 11.3-3, several of the iodine isotopes decay to radioactive xenon isotopes. In estimating the environmental release of iodine in the gaseous form, this phenomenon was taken credit for, or after the iodines became airborne, no further decay as assumed. In either event, all iodine activity was accounted for. However, no consideration has previously been given the iodines remaining in liquids or affixed to ion exchange resins. As these isotopes decay, some of the xenons created would probably evolve into the atmosphere and eventually reach the environment. At present, there seems to be little data available to indicate that these decay products constitute a major radiological problem. This could be because the isotopes that would be produced in the largest amounts are Xe-135m and Xe-135. Due to their relatively short half-lives and low concentrations, these may decay before they can escape to the atmosphere. However, there has been at least one reported instance where significant, but not excessive, amounts of xenons were released during a maintenance operation on a demineralizer (ref. 33). Therefore, to be conservative, it has been assumed that all xenons available for release are released. Estimates of the total amounts of activity involved are given in Table 11.3-12 along with individual contributions from the several sources considered. The assumptions used in analyzing each source are as follows:

Refueling Canal Water:

- a. Total iodines in refueling canal water and associated systems are equal to that present in the 4000 ft³ of primary coolant remaining in the reactor vessel after it has been drained for refueling. The isotopic specific activities in this coolant are equal to the maximum levels attained in the primary system during an equilibrium fuel cycle with 0.1 % failed fuel. All values are corrected to correspond to a coolant density of 1 gm/ml (see Table 11.2-4).
- b. All xenon created through the decay of iodine during the 21 day refueling period immediately leaves the water and is purged from the containment vessel.

Purification Demineralizer:

- a. The total iodine initially on the demineralizer resins is defined by:

$$I_{\text{Total}} (\text{Ci}) = \frac{(\text{Primary System Letdown Rate, ml/sec})(\text{Sp. Act. in Letdown, } \mu\text{Ci/cc})}{(\lambda_i, \text{Sec}^{-1}) (10^4 \text{ Ci/ } \mu\text{Ci})}$$

Where:

$$\text{Letdown Rate} = 45 \text{ gpm average}$$

Davis-Besse Unit 1 Updated Final Safety Analysis Report

Sp. Act. in Letdown = average values in Table 11.2-4 for downstream of letdown cooler

λ = decay constant of the various iodine isotopes being considered

The above expression represents the equilibrium and maximum total activity that can be transferred to the resins from normal letdown which contains uniform activity levels (this assumes that no significant amounts of parent nuclides are also deposited on the resins).

- b. When the demineralizer is taken out of service, it is valved off and allowed to decay for at least 3 days. At the end of this period, all xenon present as well as amounts of xenon equivalent to all that could be produced from the iodines present are assumed released to the environment.

Miscellaneous System Leakage:

- a. The total iodine activity contained in miscellaneous system leakage is determined from the following assumptions:

The quantity of waste considered results from the leakage rates specified in Table 11.1-9 for containment vessel and auxiliary building.

Activity levels in leakage are the average values given in Table 11.1-10.

Total leakage is based on the following operating cycle:

Normal Operation	- 292 days
Cooldown	- 16 hours
Refueling	- 21 days

- b. The maximum amount of xenons that could result from all iodines in the leakage is released.

Clean Liquid Radwaste:

- a. Total reactor coolant letdown to clean liquid radioactive waste system is 1,154,400 gallons (see Table 11.2-3).
- b. Activity levels in letdown coolant are equal to the average values listed in Table 11.2-4 for clean liquid radwaste system feed.
- c. The maximum amount of xenons that could result from all iodines contained in coolant letdown to the clean liquid radwaste system is assumed released.

Primary-to-Secondary Leakage:

- a. Primary-to-secondary leakage amounts to 100 gpd. Total annual leakage, based on a 292 day operating period, is 29,200 gallons.
- b. Activity levels in leakage are equal to the values given in Table 11.2-4 for reactor coolant.

- c. The maximum amounts of xenons that could result from all the iodines contained in primary-to-secondary leakage are assumed released through steam jet air ejector.

This ends the historical information contained in this subsection.

11.3.2.2.7 Airborne Particulate Decay Products

The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

As shown in Table 11.3-3 some gaseous radionuclides decay to radionuclides. The releases of these daughter radionuclides to the environment were analyzed at appropriate places in the USAR in accordance with the following guidelines:

Long-Lived Radionuclides (Rb-87 and Cs-135):

The theoretical maximum quantity of Rb-87 and Cs-135 that can be produced in a year of reactor operation is assumed released with the unaerated hydrogen-containing waste. Its presence is not considered in any other releases.

Short-Lived Radionuclides (Rb-88 and Cs-138):

In instances where estimated releases took into account the decay of gases after they had escaped from liquids, it was assumed that Rb-88 and Cs-138 were present in the same concentrations as their parents, Kr-88 and Xe-138. If no decay was accounted for, the presence of daughter nuclides was not considered.

This ends the historical information contained in this subsection.

11.3.2.3 Equipment Decontamination Factors

The following information pertaining to decontamination factors (DF) was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design. The ODCM provides information pertaining to the monitoring of waste streams and ensures the resultant offsite doses are in compliance with current regulatory requirements. The following DF information is considered historical.

Charcoal Filter

With one exception, gas streams passed through a charcoal filter have their iodine concentrations reduced by 99.9% (DF=103). This efficiency is based on the conservative assumption that all iodine is in the hard-to-remove organic form. The exception is the gas vented from the miscellaneous waste evaporator storage tank, which experiences a DF of 104 for all forms of iodine (DF = 1 for all others).

There are experimental data available (refs. 34, 35) that indicate that DF's of 103 and 104 are

achievable even if only methyl iodide is present. These values are somewhat conservative since they were obtained using relatively shallow 10 cm deep absorbent beds. The charcoal filters used in the Davis-Besse radwaste system have at least 24 cm deep beds. The reason a higher DF can be used with the gas from the miscellaneous waste evaporator storage tank is that estimates indicate that it might contain relatively higher levels of iodine. It has been shown that the efficiency of filters tends to go up with the iodine concentration in the influent gas (ref. 36).

It is noted that smaller DF's are assigned to the larger charcoal filters used in the ventilation systems (see Subsection 6.2.3.2). This is necessary because in these units, the absorber beds are generally shallow and are susceptible to being bypassed. The filters being considered for the radwaste system are small, deep bed units, with very few, if any, potential paths for bypass leakage.

HEPA Filters

The only filter falling within this classification is the waste gas absolute filter. For calculational purposes, this unit is assumed to have a DF of 1 for all radionuclides in all forms. Actually, by definition, it has a minimum efficiency of 99.9% for 0.3 μ particles.

Iodine Partition Factors

In a gas-liquid system, the iodine partition factor is defined:

$$P(\text{partition factor}) = \frac{\text{gm iodine/liter of liquid}}{\text{gm iodine/liter of gas}}$$

Although this definition essentially describes an equilibrium situation, "P" is used to calculate the distribution of iodine in potentially non-equilibrated systems. The normal practice is to infer gaseous specific activity from the specific activity in the liquid phase.

The partition factor applied to radwaste should be quite large due to the very minute amounts of iodine involved. For this analysis, a value of 1000 is used for liquid-gas systems existing in partially filled tanks. This number is supported by the results of calculations made using a computer program similar to the one developed by L. F. Parsly of Oak Ridge (ref. 37). Actually, for the operating conditions and low concentrations of iodine expected in the radwaste system, much larger values of "P" were obtained. These larger numbers were not used to allow some margin for the uncertainty caused by the theoretical nature of the calculational method and by the fact that the possible effects of HIO volatility were not fully taken into account. Actually, the HIO volatility probably has little effect on values of "P" in the range of 1000. However, there are indications that it may become important when values exceed 10^4 (ref. 38).

A 1000 value for "P" is also applied to waste being processed through the degasifier and the evaporators. Due to the complexity of the systems involved, this represents more of a qualitative rather than a quantitative estimate. It is based primarily on the fact that, as previously mentioned, the quantities of iodine involved are extremely small and, therefore, do not tend to come out of solution easily (ref. 39). At this time, there is no known operating experience and/or calculational method available which could be used to support this or any other estimate.

Decay Period

Sufficient storage capacity is provided to allow for all unaerated hydrogenated waste gases to have 30 days of decay in an isolated tank prior to their release. Based on expected quantities of waste gas holdup periods of 60 days should actually be possible during station operation. Releases have been calculated based on a conservative 30-day decay.

However, the actual amount of activity released should be limited by the Offsite Dose Calculation Manual. Decay periods shorter than 30 days can be justified if tank activity levels are lower than the level calculated for 0.1% failed fuel and 30 days of decay.

This ends the historical information contained in this subsection.

11.3.3 Operating Procedures

Gasses from the containment vent header and the reactor coolant drain tank vent that may contain hydrogen and fission products flow to a header and then to the waste gas surge tank. When sufficient gas has been collected, a pressure switch starts one of two compressors and connects its suction to the surge tank and its discharge to one of three waste gas decay tanks. The particular components that are to be involved in this operation are preselected. The compressor runs until the pressure in the surge tank has been sufficiently reduced and then stops. When a decay tank is full (i.e., contains gas at 150 psig) or when the operator decides, it is valved off and another put in its place. A sample is then taken from the isolated tank and analyzed. If it shows a sufficiently low activity level, the stored gas can be released in a controlled manner through waste gas charcoal and HEPA filters to the station vent. If the analysis indicates significant contamination, the gases are allowed to decay until future sampling shows that they are suitable for release to the environment. Using two of the decay tanks, gases can be held for at least 60 days with release spread out over the next 30. For calculational purposes, a 30-day holdup period has been assumed.

To preclude inleakage of air, a nitrogen blanketing system can be put into service to maintain a minimum positive pressure in the waste gas surge tank. In addition, two oxygen monitors are installed on the waste gas surge tank to provide an additional measure of safety against the unlikely event of oxygen buildup. Two oxygen/hydrogen monitors are also installed on each of the two clean waste receiver tanks. Should the alarm setpoint be reached, the operator will take steps to stop the introduction of oxygen. If an oxygen monitor or an alarm is inoperable, oxygen samples shall be taken and analyzed every four hours when gases are being added to the waste gas surge tank and daily during other operations.

Another major function of this system is to handle the nitrogen cover gas that can be displaced from the clean waste receiver and clean waste monitor tanks as they are filled along with gasses stripped in the boric acid evaporators. (Note: The boric acid evaporators are a source only if they are aligned to the gaseous radwaste system. They are normally aligned to the station vent.) An effort is made to keep this gas separate from the hydrogen and fission product gases. This is done by forming the remaining compressor and decay tank into a separate processing chain. The compressor takes suction directly from a header common to both the receiver and monitor tanks. A pressure switch located in this header controls the off-on operation of the unit. Once compressed, the cover gas may be reused as needed, or after sampling vented to the atmosphere. If the cover gas becomes significantly radioactive, there may be some danger of it contaminating the processed waste in the monitor tanks. To preclude this, the vents on the monitor tanks can be adjusted such that only flow out is allowed. In this

type of arrangement, gas from the station nitrogen system is used for makeup when the tanks are emptied.

Because of the large inventory of cover gas present, it is possible for more of it to be displaced ($>10,000 \text{ ft}^3 @ 14.7 \text{ psia}$) than one decay tank can hold. If this happens, excess gas is directed to one of the other decay tanks normally used for high level waste. This is continued until: (1) - the filled tank, if sampling indicates it's safe, is vented through the filters to the atmosphere or (2) - the displacement of gas stops.

For flexibility, all compressors and all decay tanks in the system are interchangeable. Each is used at the discretion of the operator. There are also lines and valves available which allow the transfer of gas within the system. Normally, this is done by bleeding the gas from one of the decay tanks into the surge tank. From here, a compressor can direct it to any of the decay tanks, including the one it came from.

During this transfer, the gas can be cleaned by passing it through the charcoal and absolute filters. Finally, if it is known that the cover gas or the contents of the surge tank are not significantly contaminated, the compressors can bypass the decay tanks and discharge directly to the environment.

Regardless of how the equipment is utilized, all gases being vented to the atmosphere must pass two radiation monitors. If either of these detects excessive radiation, two downstream valves are closed and discharge stopped.

Gaseous wastes which should contain little or no radioactivity and/or which may contain oxygen are handled separately. This is to prevent unnecessary processing as well as preclude the formation of explosive mixtures. These low-level gases are simply collected, passed through a charcoal filter and then released through the station vent.

11.3.4 Performance Tests

All pieces of processing equipment in these systems are provided with upstream and downstream sampling connections. Using the analysis of samples taken at these points, the actual operating DF of any component for any or all of the various radionuclides can be determined.

11.3.5 Estimated Releases

11.3.5.1 Annual Releases

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Estimates of the radioactivity (by radionuclide) released each year from this system are given in Table 11.3-14 for each of the several sources of waste gas. These values were obtained using the following assumptions.

Hydrogen-Containing and Non-aerated Gases

- a. Each year a total of five gas decay tanks will be filled, isolated, allowed to decay and then vented to the atmosphere. Quantitatively, this could amount to up to 50,000 scf of gas, which is much more than station operation is expected to create.
- b. Except for Kr-85 and iodine, the activities of the radionuclides initially present in a decay tank when it is isolated are equal to the maximum levels attained in a primary system volume during an equilibrium fuel cycle in which 0.1% of the fuel has failed. This approximately corresponds to the total amount of each that can be found external to the fuel at any one time under the stated operating condition. The amount of iodine present is based on this same assumption except that the quantities are reduced by a factor of 9.09×10^4 . This reduction results from a DF for iodine of 100 in the purification demineralizer, a partition factor of 10^3 in the degasifier, a DF of 10 in the primary demineralizer, and a second partition factor of 10^3 in the boric acid evaporator. This amounts to all that is released to the primary coolant over a 277 day equilibrium fuel cycle.

A list of all initial activities is given in Table 11.3-13.

- c. After isolation, five of the filled gas decay tanks are assumed to decay for only 30 days. Based on expected quantities of waste gas, holdup periods of at least 60 days should actually be possible during station operation. A shorter decay period can be justified if tank activity levels are lower than the level calculated for 0.1% failed fuel and 30 days of decay.
- d. After venting of a tank starts, no further decay is assumed.
- e. The amounts of particulate radioactivity vented from the waste gas decay tanks are a maximum of 2.22×10^{-10} Ci of Rb-87 and 2.06×10^{-5} Ci of Cs-135 (these values were obtained from Table 11.3-3).
- f. Although most gases being vented to the environment pass through charcoal and HEPA filters, no credit for these are taken.

Aerated Gases

- a. The total amounts of radioactivity contained in the yearly production of gases from each significant source of these wastes are given in Table 11.3-4.
- b. All aerated gases are passed through deep bed charcoal filters before they are released to the environment. The DF's used for iodine are 10^4 for the gases from the miscellaneous waste evaporator storage tank and 10^3 for gases from the other sources.

Cover Gases

- a. The total amounts of radioactivity contained in the yearly production of gases from each significant source are given in Table 11.3-5.

- b. All gases are normally passed through a deep bed charcoal filter before being released to the environment. These filters have a DF of 10^3 for all isotopes of iodine.

Building Ventilation and Purge Gases

The total amounts of activity releases are equal to the total release values given in Tables 11.3-6 through 11.3-10 for the maximum purge and/or ventilation flow rates (i.e., 50,000 cfm for all instances except for 20,000 cfm in the case of the fuel handling area ventilation system). No use of charcoal filters is assumed.

Steam Jet Air Ejector Vent

The total radionuclide activity released are the values given in Table 11.3-11.

Gaseous Decay Products

Totals for activity released due to the source are given in table 11.3-12.

This ends the historical information contained in this subsection.

11.3.5.2 Site Boundary Concentrations

The following information pertaining to estimated releases was provided as part of the initial, safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical

Estimates of the maximum and average isotopic concentrations at the station site boundary resulting from each source of gas are given in Tables 11.3-16 and 11.3-15, respectively. These values are based on the following assumptions.

This ends the historical information contained in this subsection.

11.3.5.2.1 Maximum Concentration

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical

For all calculations of maximum site boundary concentrations, an 8-hour accident X/Q value of 4.47×10^{-4} sec/m³ is used (see Subsection 11.3.7).

Hydrogenated and Non-Aerated Cases

The contents of the decay tank which were allowed to decay only 30 days including all particulate decay products, are released uniformly over a 30 day period. The amounts of activity initially contained in this tank correspond to a 1% failed fuel condition (i.e., they are 10 times the levels listed in Table 11.3-3 and 11.3-14).

Aerated Gases

- a. The maximum activity vent rate for aerated gases from the various sources are given in Table 11.3-4.
- b. All gases are assumed to pass through a deep bed charcoal filter. The DF's used for iodine are 10^4 for gases vented from the miscellaneous waste evaporator storage tank and 10^3 for all other gases.

Cover Gases

- a. The maximum activity rate for the two sources of cover gas are given in Table 11.3-5.
- b. All gases are assumed to pass through a deep bed charcoal filter which has a DF of 1000 for all isotopes of iodine.

Building Ventilation and Purge Gases

The maximum activity release rates are the values given in Tables 11.3-6 through 11.3-10 for the maximum purge and/or ventilation flow rate.

Steam Jet Air Ejector Vent

The maximum activity release rates are the values given in Table 11.3-11 which correspond to a 1% failed fuel condition.

This ends the historical information contained in this subsection.

11.3.5.2.2 Average Concentration

The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

For all calculations of average site boundary concentrations a 30 day accident X/Q value of 1.29×10^{-5} s/m³ is used (see Subsection 11.3.7). This is a conservative choice since, due to the random nature of the releases and/or the period of time over which they occur, an annual X/Q would probably be more realistic.

Hydrogenated and Non-Aerated Gases

The activities contained in the five decay tanks that are vented annually (see Table 11.3-5 for releases of unaerated hydrogenated gases) are assumed released uniformly over a 150 day period (30 days per tank). The average vent rate takes into account only that period during which release is actually occurring (i.e. it is not average over the entire 365 day year).

Aerated Gases

- a. The average activity vent rate for aerated gases from the various sources are given in Table 11.3-4.
- b. All gases pass through a deep bed charcoal filter. The DF's are the same as given in Subsection 11.3.5.2.1.

Cover Gases

- a. The average activity vent rates for cover gases are given in Table 11.3-5.
- b. All gases pass through a deep bed charcoal filter, which has a DF of 10^3 for all isotopes of iodine.

Building Ventilation and Purge Gases

The average activity release rates are the values given in Tables 11.3-6 through 11.3-10 for the maximum purge and/or ventilation flowrate.

Steam Jet Air Ejector Vent

Average release rates are those given in Table 11.3-11 which correspond to a 0.1% failed fuel condition.

This ends the historical information contained in this subsection.

11.3.6 Release Points

The major release point for potentially radioactive gaseous waste, the station vent, is indicated on Figure 11.2-6.

11.3.7 Dilution Factors

The following information pertaining to dilution factors was provided as part of the initial safety analysis. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following dilution factor information is considered historical.

Dilution factors (X/Q) for gaseous releases from the station are defined as follows:

$$(X/Q) = \frac{(\text{Ground Level Activity Conc.})}{(\text{Release Rate})}$$

In this equation, (X/Q) is a function of: (1) the distance from the activity release point, (2) the period of time over which release is assumed to occur and (3) whether or not an accident or a routine release situation is being considered. Estimates of (X/Q) values for a number of distances and release situations were determined in Section 3.2. These are given in Table 11.3-17 and 11.3-18. Estimates were also made of annual average X/Q values for the site boundary (0.73 km) and the LPZ (3.2 km). These numbers, given below, were obtained using a short method utilizing joint frequency distributions of

Davis-Besse Unit 1 Updated Final Safety Analysis Report

<u>Distance</u>	<u>Annual Average X/Q, s/m³</u>
0.73 km	4.17×10^{-6}
3.2	3.60×10^{-7}

wind speed and direction by Pasquill stability types. These values are consistent with the values given in Table 11.3-18 and are considered to be reasonably precise.

Particulates

- a. The assumptions are the same as in Subsection 11.3.5.2.1 except that a 30 day X/Q is used.

This ends the historical information contained in this subsection.

11.3.8 Estimated Doses from Gaseous Effluents

The following information pertaining to estimated doses was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated dose information is considered historical.

A detailed evaluation of the radiological impact to be incurred by radioactive gaseous releases from the station is contained in Appendix 11A. Only a summary of the results of that evaluation will be presented here. As explained in Appendix 11A, resulting exposure rates have been evaluated for maximum expected release rates, based on 0.1% failed fuel cladding, as well as for design basis release rates, based on 1.0% failed fuel cladding.

The exposure pathways evaluated include two modes of external exposure and two modes of internal exposure. The potential external exposure has been estimated for immersion in a cloud of radioactive gas and for radiation received at a height of one meter above ground due to deposited particulate matter and halogens. External exposure has been evaluated for the maximum individual, at the site boundary, and for the entire population estimated to exist within fifty miles of the station, in the years 1970, 1990, and 2010.

Internal exposure due to inhalation of the released tritium, halogens, and particulates has been evaluated, based on the estimated annual average radioactivity concentrations in air at the site boundary. For tritium, it was assumed that the activity intake via inhalation alone would represent only one half of the total activity intake via inhalation and skin absorption combined. Internal exposure via food chain transport was also evaluated, specifically for the isotopes Sr-90, Cs-134, Cs-137, I-131, and I-133. The critical exposure pathway for food chain transport was considered to be the milk chain, whereby activity deposited on pasture is transferred to cow's milk which is subsequently ingested by man. Exposure to man via this pathway was evaluated at the site boundary and at the estimated location of the nearest actual cow. Population exposure was not evaluated for internal exposure pathways.

Where possible, the resulting exposure rates due to gaseous releases were evaluated for both adult and child receptors. Regarding external exposure, only gamma radiation has been assumed to contribute to whole body doses. Beta radiation from external sources was assumed to irradiate only the outermost external surfaces of the body. A summary of the resulting exposure rates, for adult and child, for maximum expected and design basis releases, is

Davis-Besse Unit 1 Updated Final Safety Analysis Report

presented in Tables 11.3-19 and 11.3-20. A detailed discussion of their evaluation and magnitude is contained in Appendix 11A.

This ends the historical information contained in this subsection.

TABLE 11.3-1

Gaseous Radioactive Waste System Equipment List

a. Waste Gas Surge Tank

Quantity	1
Type	Vertical Cylindrical
Volume, gal	7,675
Design Pressure, psig	15
Design Temperature, °F	200
Material	304 SS
Seismic Classification	Class 1
Design Code	ASME III, Class C

b. Waste Gas Decay Tank

Quantity	3
Type	Vertical Cylindrical
Volume, gal	7577
Design Pressure, psig	150
Design Temperature, °F	200
Material	304 SS
Seismic Classification	Class 1
Design Code	ASME III, Class C

c. Waste Gas Compressor

Quantity	2
Type	Diaphragm
Capacity (0 psig suction), scfm	30
Discharge Pressure, psig	150
Discharge Temperature (max), °F	120
Material (process contacting parts)	300 Series SS
Seismic Classification	Class 2
Design Code	ASME VIII, ASME Draft Pump And Valve Code, ANSI B31.1, ANSI B31.7, and TEMA-R

d. Waste Gas Absolute Filter

Quantity	1
Type	Disposable Cartridge
Capacity, scfm	50
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	200
Vessel Material	304 SS
Vessel Seismic Classification	Class 2
Vessel Design Code	ASME III, Class 3

TABLE 11.3-1 (Continued)

Gaseous Radioactive Waste System Equipment List

e. Waste Gas Charcoal Filter

Quantity	1
Type	Disposable Cartridge
Capacity, scfm	50
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	270
Vessel Material	304 SS
Vessel Seismic Classification	Class 2
Vessel Design Code	ASME III, Class 3

f. Evaporator Vent Charcoal Filter

Quantity	1
Type	Disposable Cartridge
Capacity, scfm	50
Vessel Design Pressure, psig	150
Vessel Design Temp., °F	270
Vessel Material	304 SS
Vessel Seismic Classification	Class 2
Vessel Design Code	ASME III, Class 3

TABLE 11.3-2

Maximum Specific Activity From Degasifier (1)

<u>Radionuclide</u>	<u>Maximum Sp. Act. in Gas Vented from Degasifier, $\mu\text{Ci/cc}$</u>	
	<u>0.1 % Failed Fuel</u>	<u>1% Failed Fuel</u>
Ar-41	7.41	7.41
Kr-83m	1.93	19.21
Kr-85m	10.20	102.00
Kr-85	21.00	210.00
Kr-87	5.59	55.81
Kr-88	17.87	178.67
Xe-131m	15.00	150.00
Xe-133m	18.94	189.34
Xe-133	1660.00	16,600.00
Xe-135m	6.18	61.74
Xe-135	33.87	338.67
Xe-138	3.41	34.07
I-131	0.216	2.16
I-132	0.151	1.50
I-133	0.253	2.52
I-134	3.06×10^{-2}	0.306
I-135	0.127	1.27

- (1) The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-3

Decay Scheme (1)

Radio-nuclide	Total Activity Released From Fuel Per Cycle -Ci-	Half-Life (t(1/2)) -hr-	Decay Scheme	Maximum Accumulation of Particulate Radioactivity Decay Products	
				Radionuclide	Ci
Kr-83m		1.86	100% \longrightarrow <u>Kr-83*</u>	none	-----
Kr-85m		4.36	77% \longrightarrow <u>Rb-85</u> 23% \longrightarrow <u>Kr-85</u> \longrightarrow <u>Rb-85</u>	none	-----
Kr-85		9.1×10^4	100% \longrightarrow <u>Rb-85</u>	none	-----
Kr-87	7.0×10^4	1.3	100% \longrightarrow <u>Rb-87</u> $\xrightarrow{t(1/2)=4.7 \times 10^{10}y}$ <u>Sr-87</u>	Rb-87	2.22×10^{-10}
Kr-88		2.8	100% \longrightarrow <u>Rb-88</u> $\xrightarrow{t(1/2)=17.8m}$ <u>Sr-88</u>	none	-----
Xe-131m		288	100% \longrightarrow <u>Xe-131</u>	none	-----
Xe-133m		55.2	100% \longrightarrow <u>Xe-133</u> \longrightarrow <u>Cs-133</u>	none	-----
Xe-133		126.5	100% \longrightarrow <u>Cs-133</u>	none	-----
Xe-135m	5.5×10^4	0.26	100% \longrightarrow <u>Xe-135</u> \longrightarrow <u>Cs-135</u> $\xrightarrow{t(1/2)=3.0 \times 10^6y}$ <u>Ba-135</u>	Cs-135	5.45×10^{-7}
Xe-135	2.5×10^4	9.13	100% \longrightarrow <u>Cs-135</u> $\xrightarrow{t(1/2)=3.0 \times 10^6y}$ <u>Ba-135</u>	Cs-135	8.69×10^{-6}
Xe-138		0.28	100% \longrightarrow <u>Cs-138</u> $\xrightarrow{t(1/2)=32.2m}$ <u>Ba-138</u>	none	-----
I-131		193.2	99.4% \longrightarrow <u>Xe-131</u> 0.6% \longrightarrow <u>Xe-131m</u> \longrightarrow <u>Xe-131</u>	none	-----

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-3 (Continued)

Decay Scheme (1)

Radio-nuclide	Total Activity Released from Fuel per Cycle -Ci-	Half-Life [t(1/2)] -hr-	Decay Scheme	Maximum Accumulation of Particulate Radioactivity Decay Products	
				Radionuclide	Ci
I-132		2.3	100% \longrightarrow <u>Xe-132</u>	none	-----
I-133		21	97.6% \longrightarrow <u>Xe-133</u> \longrightarrow Cs-133	none	-----
			2.4% \longrightarrow Xe-133m \longrightarrow <u>Xe-133</u>		
I-134		0.88	100% \longrightarrow <u>Xe-134</u>	none	-----
I-135	4.4 x 10 ⁴	6.7	70% \longrightarrow Xe-135 \longrightarrow Cs-135 $\xrightarrow{t(1/2)=3.0 \times 10^6 y}$ <u>Ba-135</u>	Cs-135	1.13 x 10 ⁻⁵
			30% \longrightarrow Xe-135m \longrightarrow <u>Xe-135</u> \longrightarrow <u>Cs-135</u> \longrightarrow <u>Ba-135</u>		

* Note: underline isotopes are stable

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-4

Activity Releases in Aerated Gases (3)

Source and Radionuclide	Total Volume of Gas Vented (1) (ft ³)	Max. Sp. Act. of Vented Gas (μCi/cc)	Max. Vent Rate (μCi/sec)	Average Sp. Act. of Vented Gas (μCi/cc)	Average Vent Rate (μCi/sec)	Total Act. Vented (Ci)
Mis. Waste Drain Tank						
I-131	30,614.6	3.23 x 10 ⁻³	32.61	9.57 x 10 ⁻⁵	0.97	8.26 x 10 ⁻²
I-132		2.25 x 10 ⁻³	22.72	6.67 x 10 ⁻⁵	0.68	5.76 x 10 ⁻²
I-133		3.78 x 10 ⁻³	38.16	1.12 x 10 ⁻⁴	1.14	9.67 x 10 ⁻²
I-134		4.58 x 10 ⁻⁴	4.63	1.36 x 10 ⁻⁵	0.14	1.18 x 10 ⁻²
I-135		1.90 x 10 ⁻³	19.18	5.63 x 10 ⁻⁵	0.57	4.86 x 10 ⁻²
Waste Evaporator ⁽²⁾						
I-131	82,614.6		3.06		3.65 x 10 ⁻²	9.01 x 10 ⁻²
I-132			2.13		2.54 x 10 ⁻²	6.27 x 10 ⁻²
					2.85 x 10 ⁻⁵	
					2.68 x 10 ⁻⁵	
					4.51 x 10 ⁻⁵	
					5.46 x 10 ⁻⁶	
					2.27 x 10 ⁻⁵	
I-133			3.58		4.27 x 10 ⁻²	0.106
I-134			0.434		5.17 x 10 ⁻³	1.28 x 10 ⁻²
I-135			1.80		2.15 x 10 ⁻²	5.32 x 10 ⁻²
Miscellaneous Waste Evaporator Storage Tank						
I-131	3,304.6	0.323	1018.91	9.63 x 10 ⁻⁴	3.04	9.02 x 10 ⁻²
I-132		0.225	709.77	6.70 x 10 ⁻⁴	2.12	6.27 x 10 ⁻²
I-133		0.378	1192.41	1.13 x 10 ⁻³	3.56	0.106
I-134		4.58 x 10 ⁻²	144.48	1.37 x 10 ⁻⁴	0.431	1.29 x 10 ⁻²
I-135		0.190	599.36	5.68 x 10 ⁻⁵	1.80	5.32 x 10 ⁻²
Condensate Demineralizer Hold-up Tank						
I-131	52,135.4			5.04 x 10 ⁻⁶	7.45 x 10 ⁻³	
I-132				3.51 x 10 ⁻⁶	5.19 x 10 ⁻³	
I-133				5.90 x 10 ⁻⁶	8.72 x 10 ⁻³	
I-134				7.15 x 10 ⁻⁷	1.06 x 10 ⁻³	
I-135				2.97 x 10 ⁻⁶	4.39 x 10 ⁻³	

(1) Total volume of gas will be lower due to changes in the Misc. Radwaste System, i.e., removal of Laundry System.

(2) Waste Evaporator is no longer used.

(3) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

TABLE 11.3-5

Activity Releases in Cover Gases (2)

Source	Radionuclide	Total Volume of Gas Vented (ft ³)	1% Failed Fuel		0.1% Failed Fuel		Total Activity Vented (Ci)
			Max Sp. Act. in Vented Gas (μCi/cc)	Max Vent Rate (μCi/sec)	Average Sp. Act. in Vented Gas (μCi/cc)	Average Vent Rate (μCi/sec)	
Clean Waste Receiver Tanks	I-131	28,000	3.23×10^{-3}	0.115	3.23×10^{-7}	1.15×10^{-2}	2.57×10^{-4}
	I-132		2.25×10^{-6}	7.95×10^{-2}	2.25×10^{-7}	7.95×10^{-3}	1.79×10^{-4}
	I-133		3.78×10^{-6}	0.134	3.78×10^{-7}	1.34×10^{-2}	3.00×10^{-4}
	I-134		4.58×10^{-7}	1.62×10^{-2}	4.58×10^{-8}	1.62×10^{-3}	3.64×10^{-5}
	I-135		1.90×10^{-6}	6.72×10^{-2}	1.90×10^{-7}	6.72×10^{-3}	1.51×10^{-4}
Concentrate ⁽¹⁾ Storage Tank	I-131	6,000	3.23×10^{-6}	1.02×10^{-2}	3.23×10^{-7}	1.02×10^{-3}	5.49×10^{-5}
	I-132		2.25×10^{-6}	7.10×10^{-3}	2.25×10^{-7}	7.10×10^{-4}	3.83×10^{-5}
	I-133		3.78×10^{-6}	1.20×10^{-2}	3.78×10^{-7}	1.20×10^{-3}	6.43×10^{-5}
	I-134		4.58×10^{-7}	1.45×10^{-3}	4.58×10^{-8}	1.45×10^{-4}	7.79×10^{-6}
	I-135		1.90×10^{-6}	6.00×10^{-3}	1.90×10^{-7}	6.00×10^{-4}	3.23×10^{-5}

⁽¹⁾ Cover gas is no longer used on Concentrates Storage Tank.

⁽²⁾ The following information pertaining to quantities and concentrations was provided as part of the initial safety analysis or added as changes to the systems were made. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity and concentration information is considered historical.

TABLE 11.3-6

Activity Concentrations and Releases Due to Containment Building Purge (1)
(Purge Rate 50,000 CFM)

Radionuclide	Maximum Concentration in Air Purged, $\mu\text{Ci/cc}$	Average Concentration in Air Purged, $\mu\text{Ci/cc}$	Maximum Activity Release Rates, $\mu\text{Ci/sec}$	Average Activity Release Rates, $\mu\text{Ci/sec}$	Total Activity Releases, Ci
H-3	3.16×10^{-5}	1.97×10^{-6}	7.46×10^2	46.49	10.72
Ar-41	5.80×10^{-8}	1.73×10^{-8}	1.37	0.41	9.38×10^{-2}
Kr-83m	1.58×10^{-7}	4.58×10^{-9}	3.72	0.11	2.49×10^{-2}
Kr-85m	1.91×10^{-6}	3.37×10^{-8}	44.90	0.794	0.183
Kr-85	1.34×10^{-3}	8.01×10^{-6}	3.15×10^4	1.39×10^2	43.52
Kr-87	3.11×10^{-7}	1.13×10^{-8}	7.33	0.267	6.14×10^{-2}
Kr-88	2.12×10^{-6}	4.90×10^{-8}	49.87	1.16	0.267
Rb-88	2.12×10^{-6}	4.90×10^{-8}	49.87	1.16	0.267
Xe-131m	1.84×10^{-4}	1.14×10^{-6}	4.35×10^3	26.75	6.17
Xe-133m	4.46×10^{-5}	3.13×10^{-7}	1.06×10^3	7.39	1.71
Xe-133	9.03×10^{-3}	5.80×10^{-5}	2.13×10^5	1.37×10^3	315.2
Xe-135m	7.42×10^{-8}	5.03×10^{-9}	1.75	0.119	2.74×10^{-2}
Xe-135	1.33×10^{-5}	1.59×10^{-7}	3.14×10^2	3.75	0.862
Xe-138	4.14×10^{-8}	2.92×10^{-9}	0.977	6.89×10^{-2}	1.59×10^{-2}
I-131	9.07×10^{-7}	5.69×10^{-9}	21.39	0.135	3.10×10^{-2}
I-132	7.86×10^{-9}	1.99×10^{-10}	0.186	4.69×10^{-3}	1.08×10^{-3}
I-133	1.15×10^{-7}	9.94×10^{-10}	2.70	2.35×10^{-2}	5.40×10^{-3}
I-134	5.83×10^{-10}	2.63×10^{-11}	1.38×10^{-2}	6.19×10^{-4}	1.43×10^{-4}
I-135	1.84×10^{-8}	2.57×10^{-10}	0.434	6.0×10^{-10}	1.40×10^{-3}
Cs-138	4.14×10^{-8}	2.97×10^{-9}	0.977	6.89×10^{-2}	1.59×10^{-2}

(1)The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

TABLE 11.3-7

Activity Concentrations and Releases Due to Containment Building Ventilation During Refueling (1)
(Ventilation Air Flow Rate 50,000 CFM)

Radionuclide	Maximum Concentration in Air Purged, $\mu\text{Ci/cc}$	Average Concentration in Air Purged, $\mu\text{Ci/cc}$	Maximum Activity Release Rates, $\mu\text{Ci/sec}$	Average Activity Release Rates, $\mu\text{Ci/sec}$	Total Activity Releases, Ci
H-3	1.02×10^{-6}	1.02×10^{-6}	24.07	24.07	43.68
Ar-41	1.54×10^{-8}	2.15×10^{-11}	0.364	5.06×10^{-4}	9.21×10^{-4}
Kr-83m	4.06×10^{-8}	5.71×10^{-12}	0.959	1.35×10^{-4}	2.45×10^{-4}
Kr-85m	2.51×10^{-7}	4.13×10^{-11}	5.93	9.75×10^{-4}	1.77×10^{-3}
Kr-85	5.94×10^{-7}	1.13×10^{-10}	14.02	2.67×10^{-3}	4.84×10^{-3}
Kr-87	1.05×10^{-7}	1.32×10^{-11}	2.48	3.12×10^{-4}	5.66×10^{-4}
Kr-88	4.08×10^{-7}	6.24×10^{-11}	9.63	1.48×10^{-3}	2.68×10^{-3}
Rb-88	4.08×10^{-7}	6.24×10^{-11}	9.63	1.48×10^{-3}	2.68×10^{-3}
Xe-131m	4.24×10^{-7}	8.01×10^{-11}	10.01	1.90×10^{-3}	3.43×10^{-3}
Xe-133m	5.29×10^{-7}	9.91×10^{-11}	12.49	2.34×10^{-3}	4.25×10^{-3}
Xe-133	4.67×10^{-5}	8.81×10^{-9}	1.11×10^{-3}	0.208	0.377
Xe-135m	4.97×10^{-8}	2.72×10^{-12}	1.18	6.42×10^{-5}	1.17×10^{-4}
Xe-135	8.97×10^{-7}	1.59×10^{-10}	21.17	3.76×10^{-3}	6.80×10^{-3}
Xe-138	2.90×10^{-8}	1.65×10^{-12}	0.685	3.90×10^{-5}	7.07×10^{-5}
I-131	3.20×10^{-9}	9.59×10^{-12}	7.56×10^{-2}	2.27×10^{-4}	4.11×10^{-4}
I-132	1.80×10^{-9}	9.53×10^{-13}	4.25×10^{-2}	2.25×10^{-5}	4.09×10^{-5}
I-133	3.63×10^{-9}	4.65×10^{-12}	8.57×10^{-2}	1.10×10^{-4}	2.00×10^{-4}
I-134	3.51×10^{-10}	3.28×10^{-13}	8.29×10^{-3}	7.74×10^{-6}	1.41×10^{-5}
I-135	1.77×10^{-9}	2.29×10^{-12}	4.18×10^{-2}	5.41×10^{-5}	9.81×10^{-5}
Cs-138	2.90×10^{-8}	1.65×10^{-12}	0.685	3.90×10^{-5}	7.07×10^{-5}

(1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

TABLE 11.3-8

Activity Concentrations and Releases Due to Auxiliary Building Ventilation (Excluding Refueling Areas) (1)
(Ventilation Air Flow Rate 47,160 CFM)

Radionuclide	Maximum Concentration in Air Purged, $\mu\text{Ci/cc}$	Average Concentration in Air Purged, $\mu\text{Ci/cc}$	Maximum Activity Release Rates, $\mu\text{Ci/sec}$	Average Activity Release Rates, $\mu\text{Ci/sec}$	Total Activity Releases, Ci
H-3	1.57×10^{-8}	1.57×10^{-8}	0.350	0.350	10.39
Ar-41	4.34×10^{-9}	4.05×10^{-9}	9.65×10^{-2}	8.99×10^{-2}	2.44
Kr-83m	1.13×10^{-8}	1.06×10^{-9}	0.251	2.34×10^{-2}	0.635
Kr-85m	5.98×10^{-8}	5.58×10^{-9}	1.33	0.124	3.37
Kr-85	1.24×10^{-7}	1.15×10^{-8}	2.74	0.256	6.93
Kr-87	3.28×10^{-8}	3.05×10^{-9}	0.728	6.78×10^{-2}	1.84
Kr-88	1.05×10^{-7}	9.77×10^{-9}	2.33	0.218	5.89
Rb-88					
Xe-131m	8.81×10^{-8}	8.22×10^{-9}	1.96	0.183	4.96
Xe-133m	1.11×10^{-7}	1.04×10^{-8}	2.47	0.230	6.25
Xe-133	9.72×10^{-6}	9.07×10^{-7}	216.29	20.17	546.53
Xe-135m	3.62×10^{-8}	3.38×10^{-9}	0.805	7.50×10^{-2}	2.04
Xe-135	2.00×10^{-7}	1.86×10^{-8}	4.42	0.414	11.17
Xe-138	2.00×10^{-8}	1.86×10^{-9}	0.444	4.14×10^{-2}	1.13
I-131	1.55×10^{-9}	9.11×10^{-12}	3.44×10^{-2}	2.02×10^{-4}	5.50×10^{-3}
I-132	1.08×10^{-9}	6.36×10^{-12}	2.40×10^{-2}	1.41×10^{-4}	3.84×10^{-3}
I-133	1.82×10^{-9}	1.07×10^{-11}	4.03×10^{-2}	2.37×10^{-4}	6.42×10^{-3}
I-134	2.20×10^{-10}	1.29×10^{-12}	4.88×10^{-3}	2.87×10^{-5}	7.78×10^{-4}
I-135	9.13×10^{-10}	5.36×10^{-12}	2.03×10^{-2}	1.19×10^{-4}	3.24×10^{-3}
Cs-138					

(1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

TABLE 11.3-9

Activity Concentrations and Releases Due to Auxiliary Building Ventilation in Refueling Areas (1)
(Ventilation Air Flow Rate 20,000 CFM)

Radionuclide	Maximum Concentration in Air Purged, $\mu\text{Ci/cc}$	Average Concentration in Air Purged, $\mu\text{Ci/cc}$	Maximum Activity Release Rates, $\mu\text{Ci/sec}$	Average Activity Release Rates, $\mu\text{Ci/sec}$	Total Activity Releases, Ci
H-3	3.44×10^{-6}	3.44×10^{-6}	32.46	32.46	280.29

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-10

Activity Concentrations and Releases Due to Turbine Building Ventilation (1)
(Ventilation Air Flow Rate 285,000 CFM)

Radionuclide	Maximum Concentration in Air Purged, $\mu\text{Ci/cc}$	Average Concentration in Air Purged, $\mu\text{Ci/cc}$	Maximum Activity Release Rates, $\mu\text{Ci/sec}$	Average Activity Release Rates, $\mu\text{Ci/sec}$	Total Activity Releases, Ci
H-3	1.31×10^{-8}	1.31×10^{-8}	1.76	1.76	44.22
Ar-41	7.70×10^{-13}	7.70×10^{-13}	1.04×10^{-4}	1.04×10^{-4}	2.61×10^{-3}
Cr-51	4.81×10^{-14}	4.81×10^{-14}	6.47×10^{-6}	6.47×10^{-6}	1.63×10^{-4}
Mn-54	5.49×10^{-15}	5.49×10^{-15}	7.38×10^{-7}	7.38×10^{-7}	1.86×10^{-5}
Fe-55	1.89×10^{-13}	1.89×10^{-13}	2.54×10^{-5}	2.54×10^{-5}	6.40×10^{-4}
Fe-59	5.49×10^{-15}	5.49×10^{-15}	7.38×10^{-7}	7.38×10^{-7}	1.86×10^{-5}
Co-58	2.89×10^{-13}	2.89×10^{-13}	3.89×10^{-5}	3.89×10^{-5}	9.80×10^{-4}
Co-60	1.54×10^{-15}	1.54×10^{-15}	2.07×10^{-7}	2.07×10^{-7}	5.20×10^{-6}
Kr-83m	1.99×10^{-12}	1.99×10^{-13}	2.68×10^{-4}	2.68×10^{-5}	6.75×10^{-4}
Kr-85m	1.06×10^{-11}	1.06×10^{-12}	1.43×10^{-3}	1.43×10^{-4}	3.59×10^{-3}
Kr-85	2.19×10^{-11}	2.19×10^{-12}	2.94×10^{-3}	2.94×10^{-4}	7.4×10^{-3}
Kr-87	5.78×10^{-12}	5.78×10^{-13}	7.77×10^{-4}	7.77×10^{-5}	1.96×10^{-3}
Kr-88	1.86×10^{-11}	1.86×10^{-12}	2.50×10^{-3}	2.50×10^{-4}	6.3×10^{-3}
Rb-87	-----	-----	-----	-----	-----
Rb-88	1.40×10^{-11}	1.40×10^{-12}	1.88×10^{-3}	1.88×10^{-4}	4.74×10^{-3}
Sr-89	5.84×10^{-14}	5.84×10^{-15}	7.85×10^{-6}	7.85×10^{-7}	1.98×10^{-5}
Sr-90	1.89×10^{-15}	1.89×10^{-16}	2.54×10^{-7}	2.54×10^{-8}	6.40×10^{-7}
Sr-91	3.54×10^{-13}	3.54×10^{-14}	4.76×10^{-5}	4.76×10^{-6}	1.20×10^{-4}
Sr-92	9.73×10^{-14}	9.73×10^{-14}	1.31×10^{-5}	1.31×10^{-6}	3.30×10^{-5}
Y-90	1.31×10^{-13}	1.31×10^{-14}	1.76×10^{-5}	1.76×10^{-6}	4.42×10^{-5}
Y-91	7.34×10^{-13}	7.34×10^{-14}	9.87×10^{-5}	9.87×10^{-6}	2.49×10^{-4}
Zr-95	3.78×10^{-13}	3.78×10^{-13}	5.08×10^{-5}	5.08×10^{-5}	1.28×10^{-3}
Mo-99	5.31×10^{-11}	5.31×10^{-12}	7.14×10^{-3}	7.14×10^{-4}	1.80×10^{-2}
Xe-131m	1.57×10^{-11}	1.57×10^{-12}	2.11×10^{-3}	2.11×10^{-4}	5.3×10^{-3}
Xe-133m	1.98×10^{-11}	1.98×10^{-12}	2.66×10^{-3}	2.66×10^{-4}	6.7×10^{-3}
Xe-133	1.71×10^{-9}	1.71×10^{-10}	0.230	2.30×10^{-2}	0.58
Xe-135m	6.40×10^{-12}	6.40×10^{-13}	8.61×10^{-4}	8.61×10^{-5}	2.17×10^{-3}
Xe-135	3.54×10^{-11}	3.54×10^{-12}	4.76×10^{-3}	4.76×10^{-4}	1.20×10^{-2}
Xe-138	3.54×10^{-12}	3.54×10^{-13}	4.76×10^{-4}	4.76×10^{-5}	1.2×10^{-3}

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-10 (Continued)

Activity Concentrations and Releases Due to Turbine Building Ventilation (1)
(Ventilation Air Flow Rate 285,000 CFM)

Radionuclide	Maximum Concentration in Air Purged, $\mu\text{Ci/cc}$	Average Concentration in Air Purged, $\mu\text{Ci/cc}$	Maximum Activity Release Rates, $\mu\text{Ci/sec}$	Average Activity Release Rates, $\mu\text{Ci/sec}$	Total Activity Releases, Ci
I-131	4.22×10^{-11}	4.22×10^{-12}	5.67×10^{-3}	5.67×10^{-4}	1.43×10^{-2}
I-132	2.42×10^{-11}	2.42×10^{-12}	3.26×10^{-3}	3.26×10^{-4}	8.21×10^{-3}
I-133	4.82×10^{-11}	4.82×10^{-12}	6.51×10^{-3}	6.51×10^{-4}	1.64×10^{-2}
I-134	3.95×10^{-12}	3.95×10^{-13}	5.32×10^{-4}	5.32×10^{-5}	1.34×10^{-3}
I-135	2.31×10^{-11}	2.31×10^{-12}	3.11×10^{-3}	3.10×10^{-4}	7.81×10^{-3}
Cs-134	5.31×10^{-11}	2.31×10^{-12}	7.14×10^{-3}	7.14×10^{-4}	1.80×10^{-2}
Cs-135	-----	-----	-----	-----	-----
Cs-136	9.58×10^{-12}	9.58×10^{-13}	1.29×10^{-3}	1.29×10^{-4}	3.25×10^{-3}
Cs-137	1.64×10^{-10}	1.64×10^{-11}	2.20×10^{-2}	2.20×10^{-3}	5.55×10^{-2}
Cs-138	5.22×10^{-12}	5.22×10^{-13}	7.02×10^{-4}	7.02×10^{-5}	1.77×10^{-3}
Ba-137m	1.33×10^{-11}	1.33×10^{-12}	1.79×10^{-3}	1.79×10^{-4}	4.50×10^{-3}
Ba-139	7.46×10^{-13}	7.46×10^{-14}	1.01×10^{-4}	1.01×10^{-5}	2.53×10^{-4}
Ba-140	7.28×10^{-14}	7.28×10^{-15}	9.80×10^{-6}	9.80×10^{-7}	2.47×10^{-5}
La-140	2.89×10^{-14}	2.89×10^{-15}	3.89×10^{-6}	3.89×10^{-7}	9.80×10^{-6}
Ce-144	6.72×10^{-15}	6.72×10^{-16}	9.04×10^{-7}	9.04×10^{-6}	2.28×10^{-6}

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

TABLE 11.3-11

Activity Releases From Steam Jet Air Ejector Vent (1)

Radionuclide	Maximum Release Rate From Steam Jet Air Ejector Vent, $\mu\text{Ci/sec}$	Average Release Rate From Steam Jet Air Ejector Vent, $\mu\text{Ci/sec}$	Total Annual Release From Steam Jet Air Ejector Vent, Ci
Ar-41	.49	.49	12.25
Kr-83m	1.30	0.13	3.27
Kr-85m	6.75	0.68	17.05
Kr-85	13.8	1.38	34.9
Kr-87	3.68	0.37	9.3
Kr-88	.8	1.18	29.8
Xe-131m	9.85	0.99	24.9
Xe-133m	12.5	1.25	31.4
Xe-133	1.09×10^3	1.09×10^2	2.75×10^3
Xe-135m	4.07	0.41	10.3
Xe-135	22.4	2.24	56.5
Xe-138	2.25	0.23	5.7
I-131	2.64×10^{-2}	2.64×10^{-3}	6.66×10^{-2}
I-132	1.53×10^{-2}	1.53×10^{-3}	3.86×10^{-2}
I-133	3.02×10^{-2}	3.02×10^{-3}	7.62×10^{-2}
I-134	2.48×10^{-3}	2.48×10^{-4}	6.3×10^{-3}
I-135	1.46×10^{-2}	1.46×10^{-3}	3.69×10^{-2}

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

TABLE 11.3-12

Annual Environmental Releases of Gaseous Xenon Daughter Activity Due to
Several Sources Containing Significant Parent Isotope Inventories (1)

Xenon Isotope	Purification Demineralizer	Releases of Gaseous Xenon, Ci/yr				Totals (Ci)
		Miscellaneous System Leakage	Clean Liquid Radwaste	Primary-to Secondary Leakage	Refueling Canal Water	
Xe-131m	3.634	1.614×10^{-2}	5.683×10^{-2}	0.144	2.798×10^{-2}	3.879
Xe-133m	0.626	4.247×10^{-2}	0.150	0.379	0.300	1.498
Xe-133	15.048	0.771	2.715	6.868	5.472	30.874
Xe-135m	8.331×10^{-2}	18.129	63.835	161.392	48.938	292.378
Xe-135	0.193	1.724	6.068	15.340	15.371	38.696

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-13

Activities in Decay Tank (3)

Radionuclide	Activity Initially in Decay (1) Tank, Ci	Activity After 30 Days Decay		Activity After 60 Days Decay		Activity After 90 Days Decay	
		Fraction of Original Activity Remaining	Total Activity Remaining (2), Ci	Fraction of Original Activity Remaining	Total Activity Remaining (2), Ci	Fraction of Original Activity Remaining	Total Activity Remaining (2), Ci
<u>Noble Gases</u>							
Ar-41	25.60	10 ⁻¹¹⁸	0	10 ⁻²³⁶	0		0
Kr-83m	6.65	10 ⁻¹¹⁶	0	10 ⁻²³⁶	0		0
Kr-85m	35.31	1.86 x 10 ⁻⁵⁰	0	10 ⁻¹⁰⁰	0		0
Kr-85	210.0	0.99462	208.88	0.98928	207.75	0.98350	206.54
Kr-87	19.31	10 ⁻¹⁶⁷	0	10 ⁻³³⁴	0		0
Kr-88	61.88	10 ⁻⁷⁹	0	10 ⁻¹⁵⁷	0		0
Xe-131m	51.84	0.17677	9.17	0.031249	1.62	5.55 x 10 ⁻³	0.29
Xe-133m	65.44	1.18 x 10 ⁻⁴	7.73 x 10 ⁻³	1.40 x 10 ⁻⁸	9.17 x 10 ⁻⁷	1.64 x 10 ⁻¹²	1.08 x 10 ⁻¹⁰
Xe-133	5733.83	0.01934	111.87	3.74 x 10 ⁻¹⁴	2.17	7.36 x 10 ⁻⁶	0.043
Xe-135m	21.39	10 ⁻⁸³⁶	0	10 ⁻¹⁶⁷²	0		0
Xe-135	117.27	2.75 x 10 ⁻²⁴	0	10 ⁻⁴⁸	0		0
Xe-138	11.80	10 ⁻⁷⁶⁵	0	10 ⁻¹⁵³⁰	0		0
<u>Halogens</u>							
I-131	8.20 x 10 ⁻⁴	0.0757	6.21 x 10 ⁻⁵	0.00573	4.70 x 10 ⁻⁶	4.33 x 10 ⁻⁴	3.56 x 10 ⁻⁷
I-132	5.71 x 10 ⁻⁴	10 ⁻⁹⁰	0	10 ⁻¹⁸⁰	0		0
I-133	9.59 x 10 ⁻⁴	3.87 x 10 ⁻¹¹	3.72 x 10 ⁻¹⁴	1.5 x 10 ⁻²¹	0		0
I-134	1.17 x 10 ⁻⁴	10 ⁻²⁴⁷	0	10 ⁻⁴⁹⁴	0		0
I-135	4.82 x 10 ⁻⁴	2.94 x 10 ⁻³³	0	8.64 x 10 ⁻⁶⁶	0		0
Total			328.96		211.52		206.88

(1) - Based on a 0.1% failed fuel condition; for 1% failed fuel, multiply all activities by 10.

(2) - Takes into account the presence of any parent nuclides.

(3) - The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-14
Estimated Annual Releases from Gaseous Radwaste System (2)

Radioactivity Released to Environment Annually Due to the Various Waste Gas Sources, Ci.

Radionuclide	Un-aerated Hydrogen- Containing Gases	Aerated Gases			Cover Gases		
		Misc. Waste Drain Tank Vent	Waste Evaporator Vent ⁽¹⁾	Evaporator Storage Tank Vent ⁽¹⁾	Condensate Demin. Backwash Tank Vent	Clean Waste Receiver Tank Vent	Concentrates Storage Tank Vent ⁽³⁾
H-3	-----	-----	-----	-----	-----	-----	-----
<u>Noble Gases</u>							
Ar-41	-----	-----	-----	-----	-----	-----	-----
Kr-83m	-----	-----	-----	-----	-----	-----	-----
Kr-85m	-----	-----	-----	-----	-----	-----	-----
Kr-85	208.88	-----	-----	-----	-----	-----	-----
Kr-87	-----	-----	-----	-----	-----	-----	-----
Kr-88	-----	-----	-----	-----	-----	-----	-----
Xe-131m	45.85	-----	-----	-----	-----	-----	-----
Xe-133m	3.87×10^{-2}	-----	-----	-----	-----	-----	-----
Xe-133	559.35	-----	-----	-----	-----	-----	-----
Xe-135m	-----	-----	-----	-----	-----	-----	-----
Xe-135	-----	-----	-----	-----	-----	-----	-----
Xe-138	-----	-----	-----	-----	-----	-----	-----
<u>Halogens</u>							
I-131	3.11×10^{-4}	8.26×10^{-5}	9.01×10^{-5}	9.01×10^{-6}	7.45×10^{-6}	2.57×10^{-7}	5.49×10^{-8}
I-132	-----	5.76×10^{-5}	6.27×10^{-5}	6.27×10^{-5}	5.19×10^{-6}	1.79×10^{-7}	3.83×10^{-8}
I-133	-----	9.67×10^{-5}	1.06×10^{-4}	1.06×10^{-5}	8.72×10^{-6}	3.00×10^{-7}	6.43×10^{-8}
I-134	-----	1.18×10^{-5}	1.28×10^{-5}	1.28×10^{-6}	1.06×10^{-6}	3.64×10^{-8}	7.79×10^{-9}
I-135	-----	4.86×10^{-5}	5.32×10^{-5}	5.32×10^{-6}	4.39×10^{-6}	1.51×10^{-7}	3.23×10^{-8}

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-14 (Continued)

Estimated Annual Releases from Gaseous Radwaste System (2)

Radioactivity Released to Environment Annually Due to the Various Waste Gas Sources, Ci.

Radionuclide	Unaerated Hydrogen- Containing Gases	Aerated Gases			Cover Gases		
		Misc. Waste Drain Tank Vent	Waste Evaporator Vent ⁽¹⁾	Evaporator Storage Tank Vent ⁽¹⁾	Condensate Demin. Backwash Tank Vent	Clean Waste Receiver Tank Vent	Concentrates Storage Tank Vent ⁽³⁾
<u>Particulates</u>							
Cr-51	----	----	----	----	----	----	----
Mn-54	----	----	----	----	----	----	----
Fe-55	----	----	----	----	----	----	----
Fe-59	----	----	----	----	----	----	----
Co-58	----	----	----	----	----	----	----
Co-60	----	----	----	----	----	----	----
Rb-87	2.22 x 10 ⁻¹⁰	----	----	----	----	----	----
Rb-88	----	----	----	----	----	----	----
Sr-89	----	----	----	----	----	----	----
Sr-90	----	----	----	----	----	----	----
Sr-91	----	----	----	----	----	----	----
Sr-92	----	----	----	----	----	----	----
Y-90	----	----	----	----	----	----	----
Y-91	----	----	----	----	----	----	----
Zr-95	----	----	----	----	----	----	----
Mo-99	----	----	----	----	----	----	----
Cs-134	----	----	----	----	----	----	----
Cs-135	2.06 x 10 ⁻⁵	----	----	----	----	----	----
Cs-136	----	----	----	----	----	----	----
Cs-137	----	----	----	----	----	----	----
Cs-138	----	----	----	----	----	----	----
Ba-137m	----	----	----	----	----	----	----
Ba-139	----	----	----	----	----	----	----
Ba-140	----	----	----	----	----	----	----
La-140	----	----	----	----	----	----	----
Ce-144	----	----	----	----	----	----	----

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-14 (Continued)

Estimated Annual Releases from Gaseous Radwaste System

Radioactivity Released to Environment Annually Due to the Various Waste Gas Source, Ci.

Building Ventilation and Purge Gases (3)

Radionuclide	Containment Vessel Purge	Containment Vessel Ventilation During Refueling	Auxiliary Bldg. Ventilation (excluding refueling area)	Auxiliary Bldg. Ventilation (refueling area)	Turbine Bldg. Ventilation	Steam Jet Air Ejector Vent	Decay Product Gases	Totals
H ₃	10.72	43.68	10.39	280.29	44.22	---	---	389.31*
<u>Noble Gases</u>								
Ar-41	9.38 x 10 ⁻²	9.21 x 10 ⁻⁴	2.44	---	2.61 x 10 ⁻³	12.25	---	14.79
Kr-83m	2.49 x 10 ⁻²	2.45 x 10 ⁻⁴	0.635	---	6.75 x 10 ⁻⁴	3.27	---	3.94
Kr-85m	0.183	1.77 x 10 ⁻³	3.37	---	3.59 x 10 ⁻³	17.05	---	20.61
Kr-85	43.52	4.84 x 10 ⁻³	6.93	---	7.41 x 10 ⁻³	34.9	---	294.25
Kr-87	6.14 x 10 ⁻²	5.66 x 10 ⁻⁴	1.84	---	1.96 x 10 ⁻³	9.3	---	11.21
Kr-88	0.267	2.68 x 10 ⁻³	5.89	---	6.3 x 10 ⁻³	29.8	---	35.97
Xe-131m	6.17	3.43 x 10 ⁻³	4.96	---	5.3 x 10 ⁻³	24.9	3.88	85.77
Xe-133m	1.702	4.25 x 10 ⁻³	6.25	---	6.7 x 10 ⁻³	31.4	1.50	40.90
Xe-133	315.2	0.377	546.53	---	0.58	2750.	30.88	4202.92
Xe-135m	2.74 x 10 ⁻²	1.17 x 10 ⁻⁴	2.04	---	2.17 x 10 ⁻³	10.3	292.38	304.75
Xe-135	0.862	6.80 x 10 ⁻³	11.17	---	1.2 x 10 ⁻²	56.5	38.70	107.26
Xe-138	1.59 x 10 ⁻²	7.07 x 10 ⁻⁵	1.13	---	1.2 x 10 ⁻³	5.7	---	6.85
<u>Halogens</u>								
I-131	3.10 x 10 ⁻²	4.11 x 10 ⁻⁴	5.50 x 10 ⁻³	---	1.43 x 10 ⁻²	6.66 x 10 ⁻²	---	0.119
I-132	1.08 x 10 ⁻³	4.09 x 10 ⁻⁵	3.84 x 10 ⁻³	---	8.21 x 10 ⁻³	3.86 x 10 ⁻²	---	5.20 x 10 ⁻²
I-133	5.40 x 10 ⁻³	2.00 x 10 ⁻⁴	6.42 x 10 ⁻³	---	1.64 x 10 ⁻²	7.62 x 10 ⁻²	---	0.105
I-134	1.43 x 10 ⁻⁴	1.41 x 10 ⁻⁵	7.78 x 10 ⁻⁴	---	1.34 x 10 ⁻³	6.3 x 10 ⁻³	---	8.61 x 10 ⁻³
I-135	1.40 x 10 ⁻³	9.81 x 10 ⁻⁵	3.24 x 10 ⁻³	---	7.81 x 10 ⁻³	3.69 x 10 ⁻²	---	4.96 x 10 ⁻²
<u>Particulates</u>								
Cr-51	---	---	---	---	1.63 x 10 ⁻⁴	---	---	1.63 x 10 ⁻⁴
Mn-54	---	---	---	---	1.86 x 10 ⁻⁵	---	---	1.86 x 10 ⁻⁵
Fe-55	---	---	---	---	6.40 x 10 ⁻⁴	---	---	6.40 x 10 ⁻⁴
Fe-59	---	---	---	---	1.86 x 10 ⁻⁵	---	---	1.86 x 10 ⁻⁵
Co-58	---	---	---	---	9.80 x 10 ⁻⁴	---	---	9.80 x 10 ⁻⁴
Co-60	---	---	---	---	5.20 x 10 ⁻⁶	---	---	5.20 x 10 ⁻⁶
Rb-87	---	---	---	---	---	---	---	2.22 x 10 ⁻¹⁰
Rb-88	0.267	2.68 x 10 ⁻³	---	---	4.74 x 10 ⁻³	---	---	0.275
Sr-89	---	---	---	---	1.98 x 10 ⁻⁵	---	---	1.98 x 10 ⁻⁵

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-14 (Continued)

Estimated Annual Releases from Gaseous Radwaste System

Radioactivity Released to Environment Annually Due to the Various Waste Gas Source, Ci.

Building Ventilation and Purge Gases (3)

Radionuclide	Containment Vessel Purge	Containment Vessel Ventilation During Refueling	Auxiliary Bldg. Ventilation (excluding refueling area)	Auxiliary Bldg. Ventilation (refueling area)	Turbine Bldg. Ventilation	Steam Jet Air Ejector Vent	Decay Product Gases	Totals
Sr-90	---	---	---	---	6.4×10^{-7}	---	---	6.4×10^{-7}
Sr-91	---	---	---	---	1.20×10^{-4}	---	---	1.20×10^{-4}
Sr-92	---	---	---	---	3.30×10^{-5}	---	---	3.30×10^{-5}
Y-90	---	---	---	---	4.42×10^{-5}	---	---	4.42×10^{-5}
Y-91	---	---	---	---	2.49×10^{-4}	---	---	2.49×10^{-4}
Zr-95	---	---	---	---	1.28×10^{-3}	---	---	1.28×10^{-3}
Mo-99	---	---	---	---	1.80×10^{-2}	---	---	1.80×10^{-2}
Cs-134	---	---	---	---	1.80×10^{-2}	---	---	1.80×10^{-2}
Cs-135	---	---	---	---	---	---	---	2.06×10^{-5}
Cs-136	---	---	---	---	3.25×10^{-3}	---	---	3.25×10^{-3}
Cs-137	---	---	---	---	5.55×10^{-2}	---	---	5.55×10^{-2}
Cs-138	1.59×10^{-2}	---	---	---	1.77×10^{-3}	---	---	1.77×10^{-2}
Ba-137m	---	---	---	---	4.50×10^{-3}	---	---	4.50×10^{-3}
Ba-139	---	---	---	---	2.53×10^{-4}	---	---	2.53×10^{-4}
Ba-140	---	---	---	---	2.47×10^{-5}	---	---	2.47×10^{-5}
La-140	---	---	---	---	9.80×10^{-6}	---	---	9.80×10^{-6}
Ce-144	---	---	---	---	2.28×10^{-6}	---	---	2.28×10^{-6}

* Due to the conservative nature of the assumptions used, this number is more representative of an upper limit for the total activity normally released. It should also be noted that the tritium released in gaseous effluent should reduce the amount that must be released in liquids in order to maintain a stable station tritium inventory (see Subsections 11.2.2.2.3 and 11.2.5.3).

(1) Waste Evaporator is no longer used.

(2) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

(3) Cover gas is no longer used on Concentrates Storage Tank.

TABLE 11.3-15

Annual Average Radionuclide Concentrations at Site Boundary Due to Various Waste Gas Sources (2)Annual Average Site Boundary Concentration, $\mu\text{Ci/cc}$

Radionuclide	Un aerated Hydrogenated Gases	Aerated Gases		Cover Gases		
		Misc. Waste Drain Tank Vent	Waste Evaporator Vent ⁽¹⁾	Evaporator Storage Tank Vent ⁽¹⁾	Clean Waste Receiver Tank Vent	Concentrates Storage Tank Vent ⁽³⁾
H ₃						
<u>Noble Gases</u>						
Ar-41	----	----	----	----	----	----
Kr-83m	----	----	----	----	----	----
Kr-85m	----	----	----	----	----	----
Kr-85	2.08×10^{-10}	----	----	----	----	----
Kr-87	----	----	----	----	----	----
Kr-88	----	----	----	----	----	----
Xe-131m	1.56×10^{-11}	----	----	----	----	----
Xe-133m	7.70×10^{-15}	----	----	----	----	----
Xe-133	1.20×10^{-10}	----	----	----	----	----
Xe-135m	----	----	----	----	----	----
Xe-135	----	----	----	----	----	----
Xe-138	----	----	----	----	----	----
<u>Halogens</u>						
I-131	8.06×10^{-17}	1.26×10^{-14}	4.71×10^{-16}	3.93×10^{-15}	1.49×10^{-16}	1.32×10^{-17}
I-132	----	8.78×10^{-15}	3.28×10^{-16}	2.74×10^{-15}	1.03×10^{-16}	9.16×10^{-18}
I-133	----	1.48×10^{-14}	5.51×10^{-16}	4.60×10^{-15}	1.73×10^{-16}	1.55×10^{-17}
I-134	----	1.81×10^{-15}	6.67×10^{-17}	5.56×10^{-16}	2.09×10^{-17}	1.88×10^{-18}
I-135	----	7.36×10^{-15}	2.78×10^{-16}	2.33×10^{-15}	8.67×10^{-17}	7.44×10^{-18}

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-15 (Continued)

Annual Average Radionuclide Concentrations at Site Boundary Due to Various Waste Gas Sources (2)

Annual Average Site Boundary Concentration, micro $\mu\text{Ci/cc}$

Radionuclide	Unaerated Hydrogenated Gases	Aerated Gases			Cover Gases	
		Misc. Waste Drain Tank Vent	Waste Evaporator Vent (1)	Evaporator Storage Tank Vent (1)	Clean Waste Receiver Tank Vent	Concentrates Storage Tank Vent (3)
<u>Particulates</u>						
Cr-51	----	----	----	----	----	----
Mn-54	----	----	----	----	----	----
Fe-55	----	----	----	----	----	----
Fe-59	----	----	----	----	----	----
Co-58	----	----	----	----	----	----
Co-60	----	----	----	----	----	----
Rb-87	1.11 x 10 ⁻²¹	----	----	----	----	----
Rb-88	----	----	----	----	----	----
Sr-89	----	----	----	----	----	----
Sr-90	----	----	----	----	----	----
Sr-91	----	----	----	----	----	----
Sr-92	----	----	----	----	----	----
Y-90	----	----	----	----	----	----
Y-91	----	----	----	----	----	----
Zr-95	----	----	----	----	----	----
Mo-99	----	----	----	----	----	----
Cs-134	----	----	----	----	----	----
Cs-135	1.03 x 10 ⁻¹⁶	----	----	----	----	----
Cs-136	----	----	----	----	----	----
Cs-137	----	----	----	----	----	----
Cs-138	----	----	----	----	----	----
Ba-137m	----	----	----	----	----	----
Ba-139	----	----	----	----	----	----
Ba-140	----	----	----	----	----	----
La-140	----	----	----	----	----	----
Ce-144	----	----	----	----	----	----

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-15 (Continued)

Annual Average Radionuclide Concentrations at Site Boundary Due to Various Waste Gas Sources (2)

Annual Average Site Boundary Concentration, $\mu\text{Ci/cc}$

Building Ventilation and Purge Gases

Radionuclide	Containment Vessel Purge	Containment Vessel Ventilation During Refueling	Auxiliary Bldg. Ventilation (excluding refueling area)	Auxiliary Bldg. Ventilation (refueling area)	Turbine Building Ventilation	Steam Jet Air Ejector Vent
H ₃	6.00×10^{-10}	3.11×10^{-10}	4.52×10^{-12}		2.28×10^{-11}	
<u>Noble Gases</u>						
Ar-41	5.29×10^{-12}	6.56×10^{-15}	1.16×10^{-12}	---	1.35×10^{-15}	6.33×10^{-12}
Kr-83m	1.42×10^{-12}	1.75×10^{-15}	3.02×10^{-13}	---	3.46×10^{-16}	1.68×10^{-12}
Kr-85m	1.03×10^{-11}	1.26×10^{-14}	1.60×10^{-12}	---	1.85×10^{-15}	8.78×10^{-12}
Kr-85	2.44×10^{-9}	3.45×10^{-14}	3.31×10^{-12}	---	3.80×10^{-15}	1.79×10^{-11}
Kr-87	3.45×10^{-12}	4.03×10^{-15}	8.75×10^{-13}	---	1.01×10^{-15}	4.78×10^{-12}
Kr-88	1.50×10^{-11}	1.91×10^{-14}	2.82×10^{-12}	---	3.23×10^{-15}	1.53×10^{-11}
Xe-131m	3.46×10^{-10}	2.46×10^{-14}	2.37×10^{-12}	---	2.73×10^{-15}	1.28×10^{-11}
Xe-133m	9.54×10^{-11}	3.02×10^{-14}	2.97×10^{-12}	---	3.44×10^{-15}	1.62×10^{-11}
Xe-133	1.77×10^{-8}	2.69×10^{-12}	2.61×10^{-10}	---	2.97×10^{-13}	1.41×10^{-9}
Xe-135m	1.54×10^{-12}	8.29×10^{-16}	9.68×10^{-13}	---	1.12×10^{-15}	5.29×10^{-12}
Xe-135	4.84×10^{-11}	4.86×10^{-14}	5.35×10^{-12}	---	6.15×10^{-15}	2.89×10^{-11}
Xe-138	8.89×10^{-13}	5.04×10^{-16}	5.35×10^{-13}	---	6.15×10^{-16}	2.97×10^{-12}
<u>Halogens</u>						
I-131	1.75×10^{-12}	2.93×10^{-15}	2.61×10^{-15}	---	7.32×10^{-15}	3.41×10^{-14}
I-132	6.06×10^{-14}	2.91×10^{-16}	1.82×10^{-15}	---	4.21×10^{-15}	1.98×10^{-14}
I-133	3.04×10^{-13}	1.42×10^{-15}	3.06×10^{-15}	---	8.40×10^{-15}	3.90×10^{-14}
I-134	7.99×10^{-15}	9.99×10^{-17}	3.71×10^{-16}	---	6.87×10^{-16}	3.20×10^{-15}
I-135	7.84×10^{-14}	6.98×10^{-16}	1.54×10^{-15}	---	4.00×10^{-15}	1.89×10^{-14}
Cr-51	---	---	---	---	8.35×10^{-17}	---
Mn-54	---	---	---	---	9.53×10^{-18}	---
Fe-55	---	---	---	---	3.28×10^{-16}	---
Fe-59	---	---	---	---	9.53×10^{-18}	---
Co-58	---	---	---	---	5.02×10^{-16}	---
Co-60	---	---	---	---	2.68×10^{-18}	---
Rb-87	---	---	---	---	---	---
Rb-88	1.50×10^{-11}	1.91×10^{-14}	---	---	2.43×10^{-15}	---
Sr-89	---	---	---	---	1.02×10^{-17}	---

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-15 (Continued)

Annual Average Radionuclide Concentrations at Site Boundary Due to Various Waste Gas Sources (2)

Annual Average Site Boundary Concentration, micro $\mu\text{Ci/cc}$

Building Ventilation and Purge Gases

Radionuclide	Containment Vessel Purge	Containment Vessel Ventilation During Refueling	Auxiliary Bldg. Ventilation (excluding refueling area)	Auxiliary Bldg. Ventilation (refueling area)	Turbine Building Ventilation	Steam Jet Air Ejector Vent
Sr-90	---	---	---	---	3.28×10^{-19}	---
Sr-91	---	---	---	---	6.15×10^{-17}	---
Sr-92	---	---	---	---	1.69×10^{-17}	---
Y-90	---	---	---	---	2.28×10^{-17}	---
Y-91	---	---	---	---	1.28×10^{-16}	---
Zr-95	---	---	---	---	6.56×10^{-16}	---
Mo-99	---	---	---	---	9.22×10^{-15}	---
Cs-134	---	---	---	---	9.22×10^{-15}	---
Cs-135	---	---	---	---	---	---
Cs-136	---	---	---	---	1.67×10^{-15}	---
Cs-137	---	---	---	---	2.84×10^{-14}	---
Cs-138	8.89×10^{-13}	5.04×10^{-16}	---	---	9.06×10^{-16}	---
Ba-137m	---	---	---	---	2.31×10^{-15}	---
Ba-139	---	---	---	---	1.31×10^{-16}	---
Ba-140	---	---	---	---	1.27×10^{-17}	---
La-140	---	---	---	---	5.02×10^{-18}	---
Ce-144	---	---	---	---	1.17×10^{-18}	---

- (1) Waste Evaporator is no longer used.
- (2) The following information pertaining to estimated doses was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.
- (2) Cover gas no longer used on Concentrates Storage Tank.

TABLE 11.3-16

Maximum Radionuclide Concentrations at Site Boundary Due to Various Waste Gas Sources (5)Maximum Site Boundary Concentrations, $\mu\text{Ci/cc}$

Radionuclide	Un aerated Hydrogenated Gases (1)	Aerated Gases		Cover Gases		
		Misc. Waste Drain Tank Vent	Waste Evaporator Vent (4)	Evaporator Storage Tank Vent (4)	Clean Waste Receiver Tank Vent	Concentrates Storage Tank Vent (6)
H ₃	----	----	----	----	----	----
<u>Noble Gases</u>						
Ar-41	----	----	----	----	----	----
Kr-83m	----	----	----	----	----	----
Kr-85m	----	----	----	----	----	----
Kr-85	3.61×10^{-7}	----	----	----	----	----
Kr-87	----	----	----	----	----	----
Kr-88	----	----	----	----	----	----
Xe-131m	1.59×10^{-8}	----	----	----	----	----
Xe-133m	1.34×10^{-11}	----	----	----	----	----
Xe-133	1.93×10^{-7}	----	----	----	----	----
Xe-135m	----	----	----	----	----	----
Xe-135	----	----	----	----	----	----
Xe-138	----	----	----	----	----	----
<u>Halogens</u>						
I-131	1.08×10^{-13}	1.46×10^{-11}	1.37×10^{-12}	4.56×10^{-11}	5.15×10^{-14}	4.56×10^{-15}
I-132	----	1.02×10^{-11}	9.53×10^{-13}	3.18×10^{-11}	3.56×10^{-14}	3.18×10^{-15}
I-133	6.42×10^{-23}	1.71×10^{-11}	1.61×10^{-12}	5.34×10^{-11}	5.99×10^{-14}	5.37×10^{-15}
I-134	----	2.07×10^{-12}	1.94×10^{-13}	6.46×10^{-12}	7.25×10^{-15}	6.49×10^{-16}
I-135	----	8.58×10^{-12}	8.05×10^{-13}	2.68×10^{-11}	3.01×10^{-14}	2.69×10^{-15}

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-16 (Continued)

Maximum Radionuclide Concentrations at Site Boundary Due to Various Waste Gas Sources (5)

Maximum Site Boundary Concentrations, μCi/cc						
Radionuclide	Unaerated Hydrogenated Gases (1)	Aerated Gases			Cover Gases	
		Misc. Waste Drain Tank Vent	Waste Evaporator Vent ⁽⁴⁾	Evaporator Storage Tank Vent ⁽⁴⁾	Clean Waste Receiver Tank Vent	Concentrates Storage Tank Vent ⁽⁶⁾
<u>Particulates</u>						
Cr-51	----	----	----	----	----	----
Mn-54	----	----	----	----	----	----
Fe-55	----	----	----	----	----	----
Fe-59	----	----	----	----	----	----
Co-58	----	----	----	----	----	----
Co-60	----	----	----	----	----	----
Rb-87	3.83 x 10 ⁻¹⁹	----	----	----	----	----
Rb-88	----	----	----	----	----	----
Sr-89	----	----	----	----	----	----
Sr-90	----	----	----	----	----	----
Sr-91	----	----	----	----	----	----
Sr-92	----	----	----	----	----	----
Y-90	----	----	----	----	----	----
Y-91	----	----	----	----	----	----
Zr-95	----	----	----	----	----	----
Mo-99	----	----	----	----	----	----
Cs-134	----	----	----	----	----	----
Cs-135	3.56 x 10 ⁻¹⁴	----	----	----	----	----
Cs-136	----	----	----	----	----	----
Cs-137	----	----	----	----	----	----
Cs-138	----	----	----	----	----	----
Ba-137m	----	----	----	----	----	----
Ba-139	----	----	----	----	----	----
Ba-140	----	----	----	----	----	----
La-140	----	----	----	----	----	----
Ce-144	----	----	----	----	----	----

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-16 (Continued)

Maximum Radionuclide Concentrations at Site Boundary Due to Various Waste Gas Sources (5)

Maximum Site Boundary Concentrations, $\mu\text{Ci/cc}$						
Building Ventilation and Purge Gases (2)						
Radionuclide	Containment Vessel Purge (3)	Containment Vessel Ventilation During Refueling	Auxiliary Bldg. Ventilation (excluding refueling area)	Auxiliary Bldg. Ventilation (refueling area)	Turbine Building Ventilation	Steam Jet Air Ejector
H-3	3.34×10^{-7}	1.08×10^{-8}	1.57×10^{-10}	1.46×10^{-8}	7.87×10^{-10}	---
<u>Noble Gases</u>						
Ar-41	6.13×10^{-10}	1.63×10^{-10}	4.32×10^{-11}	---	4.65×10^{-14}	2.20×10^{-10}
Kr-83m	1.67×10^{-9}	4.29×10^{-10}	1.13×10^{-10}	---	1.20×10^{-13}	5.82×10^{-10}
Kr-85m	2.01×10^{-8}	2.66×10^{-9}	5.95×10^{-10}	---	6.40×10^{-13}	3.02×10^{-9}
Kr-85	1.41×10^{-5}	6.27×10^{-9}	1.23×10^{-9}	---	1.32×10^{-12}	6.17×10^{-9}
Kr-87	3.28×10^{-9}	1.11×10^{-9}	3.26×10^{-10}	---	3.48×10^{-13}	1.65×10^{-9}
Kr-88	2.23×10^{-8}	4.31×10^{-9}	1.05×10^{-9}	---	1.12×10^{-12}	5.28×10^{-9}
Xe-131m	1.94×10^{-6}	4.48×10^{-9}	8.77×10^{-10}	---	9.44×10^{-13}	4.41×10^{-9}
Xe-133m	4.74×10^{-7}	5.59×10^{-9}	1.11×10^{-9}	---	1.19×10^{-12}	5.59×10^{-9}
Xe-133	9.53×10^{-5}	4.97×10^{-7}	9.67×10^{-8}	---	1.03×10^{-10}	4.88×10^{-7}
Xe-135m	7.83×10^{-10}	5.28×10^{-10}	3.60×10^{-10}	---	3.85×10^{-13}	1.82×10^{-9}
Xe-135	1.41×10^{-7}	9.47×10^{-9}	1.98×10^{-9}	---	2.13×10^{-12}	1.01×10^{-8}
Xe-138	4.37×10^{-10}	3.07×10^{-10}	1.99×10^{-10}	---	2.13×10^{-13}	1.01×10^{-9}
<u>Halogens</u>						
I-131	9.57×10^{-9}	3.38×10^{-11}	1.54×10^{-11}	---	2.54×10^{-12}	1.19×10^{-11}
I-132	8.32×10^{-11}	1.90×10^{-11}	1.08×10^{-11}	---	1.46×10^{-12}	6.84×10^{-12}
I-133	1.21×10^{-9}	3.84×10^{-11}	1.81×10^{-11}	---	2.91×10^{-12}	1.35×10^{-11}
I-134	6.17×10^{-12}	3.71×10^{-12}	2.19×10^{-12}	---	2.38×10^{-13}	1.11×10^{-12}
I-135	1.94×10^{-10}	1.87×10^{-11}	9.08×10^{-12}	---	1.40×10^{-12}	6.53×10^{-12}
<u>Particulates</u>						
Cr-51	---	---	---	---	2.90×10^{-15}	---
Mn-54	---	---	---	---	3.30×10^{-16}	---
Fe-55	---	---	---	---	1.14×10^{-14}	---
Fe-59	---	---	---	---	3.30×10^{-16}	---
Co-58	---	---	---	---	1.74×10^{-14}	---
Co-60	---	---	---	---	9.26×10^{-17}	---
Rb-87	---	---	---	---	---	---
Rb-88	2.23×10^{-8}	4.31×10^{-9}	---	---	8.41×10^{-13}	---
Sr-89	---	---	---	---	3.51×10^{-15}	---

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-16 (Continued)

Maximum Radionuclide Concentrations at Site Boundary Due to Various Waste Gas Sources (5)

Maximum Site Boundary Concentrations, $\mu\text{Ci/cc}$

Building Ventilation and Purge Gases (2)

Radionuclide	Containment Vessel Purge (3)	Containment Vessel Ventilation during Refueling	Auxiliary Bldg. Ventilation (excluding refueling area)	Auxiliary Bldg. Ventilation (refueling area)	Turbine Building Ventilation	Steam Jet Air Ejector
Sr-90	---	---	---	---	1.14×10^{-16}	---
Sr-91	---	---	---	---	2.13×10^{-14}	---
Sr-92	---	---	---	---	5.86×10^{-15}	---
Y-90	---	---	---	---	7.87	---
Y-91	---	---	---	---	4.42×10^{-14}	---
Zr-95	---	---	---	---	2.28×10^{-14}	---
Mo-99	---	---	---	---	3.20×10^{-12}	---
Cs-134	---	---	---	---	3.20×10^{-12}	---
Cs-135	---	---	---	---	---	---
Cs-136	---	---	---	---	5.77×10^{-13}	---
Cs-137	---	---	---	---	9.84×10^{-12}	---
Cs-138	4.37×10^{-10}	3.07×10^{-10}	---	---	3.14×10^{-13}	---
Ba-137m	---	---	---	---	8.01×10^{-13}	---
Ba-139	---	---	---	---	4.52×10^{-14}	---
Ba-140	---	---	---	---	4.39×10^{-15}	---
La-140	---	---	---	---	1.74×10^{-15}	---
Ce-144	---	---	---	---	4.05×10^{-16}	---

- Notes : (1) This is a controlled release, atmospheric conditions can be chosen to give more adequate dispersion.
- (2) All gases in this category can be passed through charcoal filters if iodine levels appear too high; also, flow rates can be reduced to lower site boundary concentrations.
- (3) Maximum levels occur only at the beginning of purge, at this time release rates can be reduced; also, as the purge is a controlled release, atmospheric conditions can be chosen to give more adequate dispersion.
- (4) Waste Evaporator is no longer used.
- (5) The following information pertaining to estimated doses was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated dose information is considered historical.
- (6) Concentrates storage tank is no longer on cover gas header.

TABLE 11.3-17

X/Q Values for Short-Term Releases Resulting from (1)
Accident Situation

<u>Time Period</u>	<u>X/Q, s/m³</u>	
	<u>Exclusion Boundary</u> (0.73 km)	<u>Low Population Zone</u> (3.2 km)
0 – 8 hours	4.47×10^{-4}	9.16×10^{-5}
8 – 24	2.50×10^{-4}	2.50×10^{-5}
1.4 days	6.75×10^{-5}	6.15×10^{-5}
4 – 30	1.29×10^{-5}	1.08×10^{-6}

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

TABLE 11.3-18

X/Q Values for Long-Term Releases for 24 Hrs, 30 days, and One Year (1)

<u>Distance (km)</u>	<u>X/Q, s/m³</u>		
	<u>24 Hrs</u>	<u>30 Days</u>	<u>Annual</u>
16	1.27×10^{-6}	4.70×10^{-8}	2.64×10^{-8}
32	4.88×10^{-7}	1.69×10^{-8}	9.16×10^{-9}
48	2.82×10^{-7}	9.10×10^{-9}	5.12×10^{-9}
64	1.98×10^{-7}	5.86×10^{-9}	3.31×10^{-9}
80	1.49×10^{-7}	4.24×10^{-9}	2.37×10^{-9}

- (1) The following information pertaining to estimated releases was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates cumulative releases, and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated release information is considered historical.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.3-19

Summary of Maximum Individual Doses Due to Gaseous Releases

Doses Presented are Based on Maximum Expected Releases (1)

Exposure Pathway	<u>Dose to Child, mr/yr*</u>			<u>Dose to Adult, mr/yr*</u>		
	W.B.	Body Surface	Thyroid	W.B.	Body Surface	Thyroid
Cloud immersion	0.593	1.30	0.593	0.593	1.30	0.593
External exposure from particulate deposition	0.588	1.194	0.588	0.588	1.194	0.588
Iodine inhalation	0.0004	0.0004	0.382	0.0004	0.0004	0.220
Tritium inhalation	0.096	0.113	0.125	0.096	0.113	0.125
Particulate inhalation	0.0012	0.0012	0.0012	0.0028	0.0028	0.0028
Food chain transport:						
At nearest cow	0.0195	0.0195	3.07	0.0060	0.0060	0.307
At sight boundary	0.102	0.102	16.2	0.0312	0.0312	1.64
Exposure Totals:						
Milk ingested from nearest cow:	1.33	2.66	4.77	1.29	2.62	1.84
Milk ingested from a cow located at the site boundary:	1.41	2.74	17.9	1.32	2.64	3.17

* Doses due to design basis releases are roughly a factor of ten higher in each case, except for exposure due to tritium which is unchanged.

- (1) The following information pertaining to estimated doses was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated dose information is considered historical.

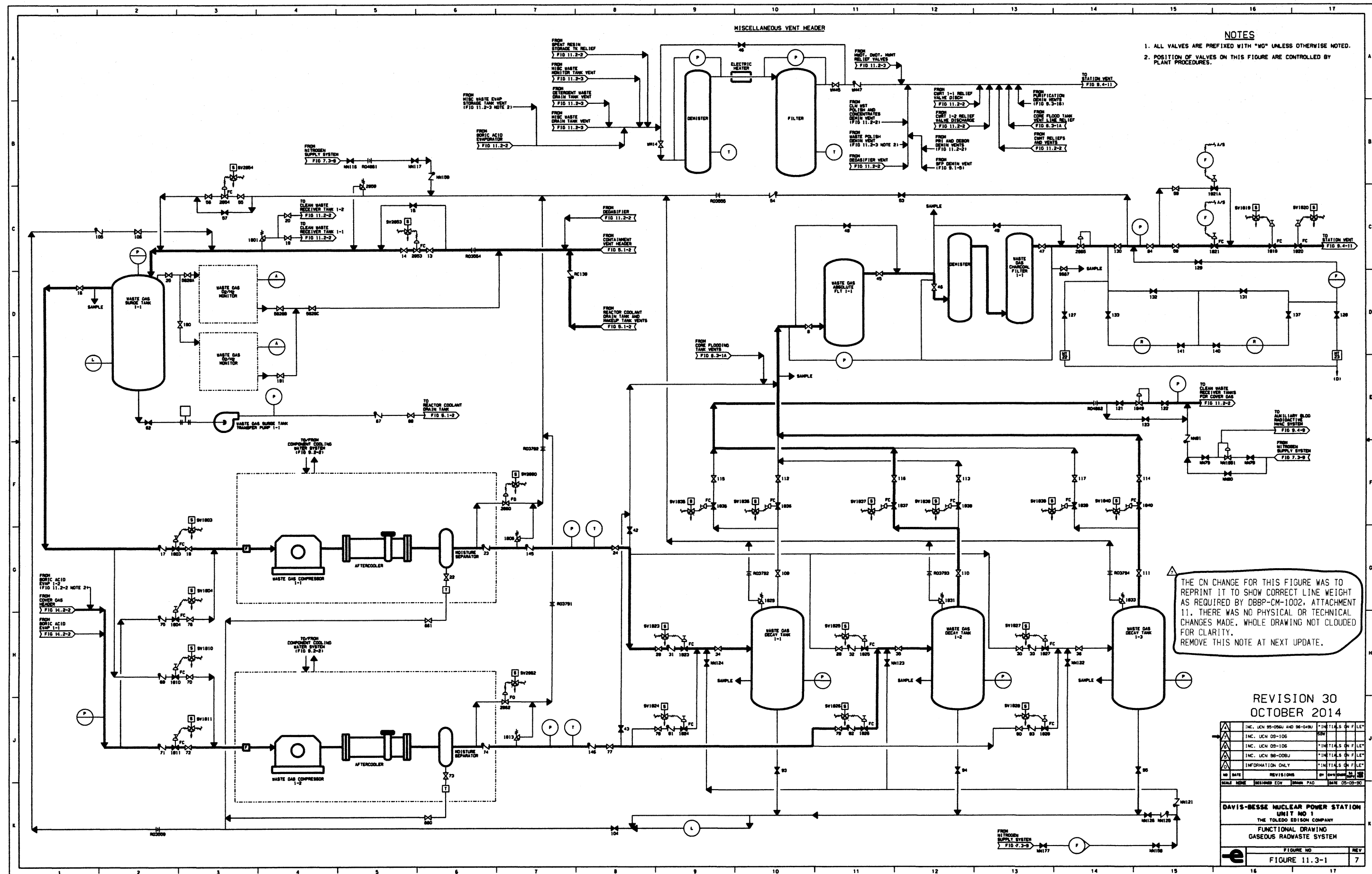
TABLE 11.3-20

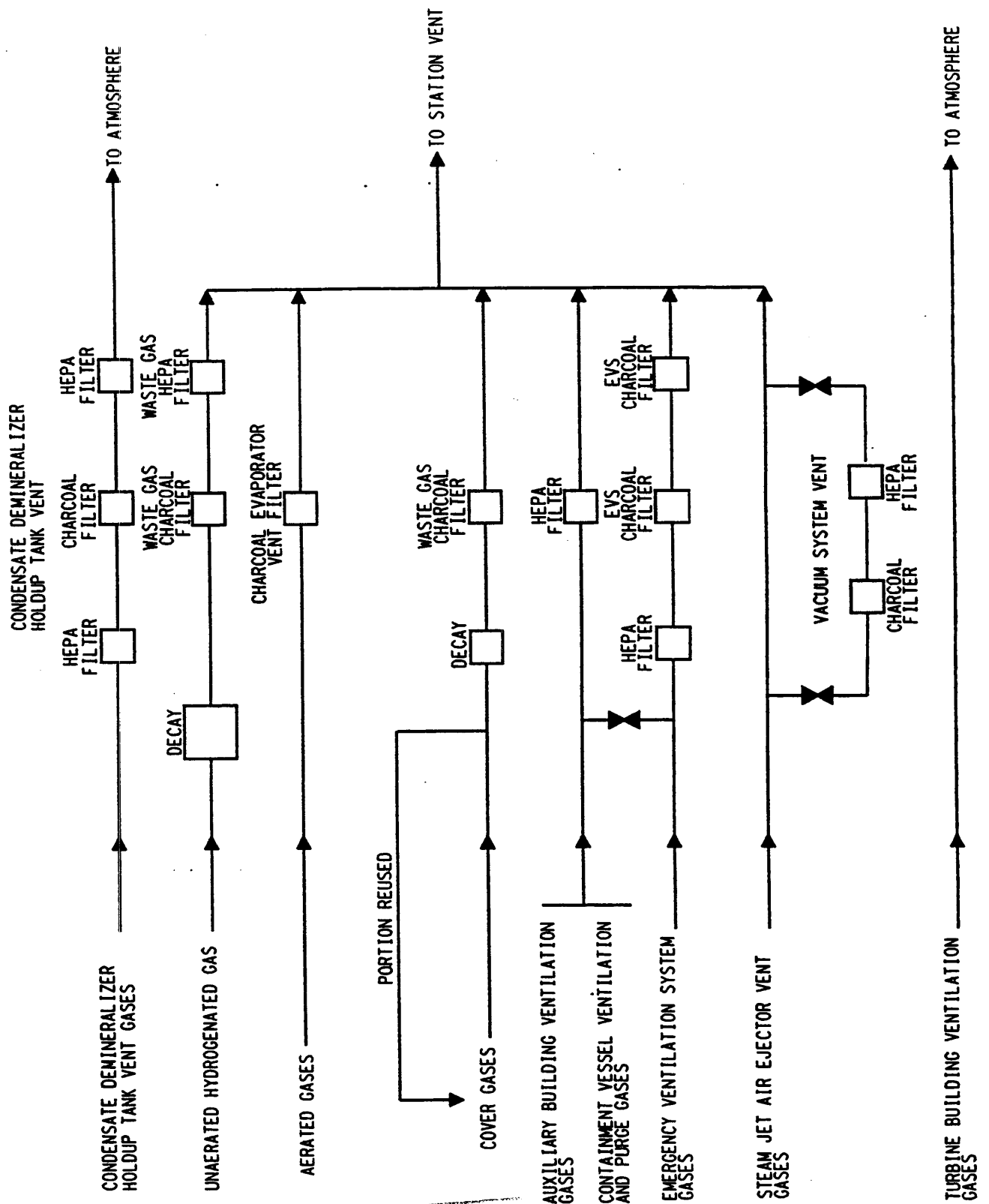
External Population Exposure Due to Gaseous Releases (1)

Year	Population Exposure Due to Cloud Immersion Man-rem/year				Avg. Per Capita Exposure Due to Cloud Immersion mrem/year*			
	Whole Body Exposure		Body Surface Exposure		Whole Body Exposure		Body Surface Exposure	
	M.E.R.	D.B.R.	M.E.R.	D.B.R.	M.E.R.	D.B.R.	M.E.R.	D.B.R.
1970	0.731	7.31	1.605	15.71	3.54-4	3.54-3	7.76-4	7.60-3
1990	0.958	9.58	2.103	20.58	3.19-4	3.19-3	7.00-4	6.85-3
2010	1.266	12.66	2.779	27.20	3.04-4	3.04-3	6.66-4	6.52-3
<u>Population Exposure Due to Particulate Deposition</u>								
	Whole Body Exposure		Body Surface Exposure		Whole Body Exposure		Body Surface Exposure	
	M.E.R.	D.B.R.	M.E.R.	D.B.R.	M.E.R.	D.B.R.	M.E.R.	D.B.R.
	<u>Whole Body Exposure</u>		<u>Body Surface Exposure</u>		<u>Whole Body Exposure</u>		<u>Body Surface Exposure</u>	
1970	0.725	7.25	1.474	14.74	3.51-4	3.51-3	7.15-4	7.15-3
1990	0.951	9.51	1.932	19.32	3.17-4	3.17-3	6.44-4	6.44-3
2010	1.256	12.56	2.552	25.52	3.02-4	3.02-3	6.13-4	6.13-3
<u>Total External Population Exposure</u>								
	Whole Body Exposure		Body Surface Exposure		Whole Body Exposure		Body Surface Exposure	
	M.E.R.	D.B.R.	M.E.R.	D.B.R.	M.E.R.	D.B.R.	M.E.R.	D.B.R.
	<u>Whole Body Exposure</u>		<u>Body Surface Exposure</u>		<u>Whole Body Exposure</u>		<u>Body Surface Exposure</u>	
1970	1.456	14.56	3.079	30.45	7.05-4	7.05-3	1.49-3	1.48-2
1990	1.909	19.09	4.035	39.90	6.36-4	6.36-3	1.34-3	1.33-2
2010	2.522	25.22	5.331	52.72	6.06-4	6.06-3	1.28-3	1.27-2

* NOTE: In this table, a number A x 10^{-B} is expressed as A-B.

- (1) The following information pertaining to estimated doses was provided as part of the initial safety analysis. Operation of the radwaste system must ensure compliance with current regulatory requirements. The ODCM provides the day-to-day methods for determining release rates, cumulative releases and for calculating the corresponding dose rates and cumulative quarterly and yearly doses. The following estimated dose information is considered historical.





DAVIS-BESSE NUCLEAR POWER STATION
 GASEOUS RADIOACTIVE WASTE SYSTEM FLOW SCHEMATIC
 FIGURE 11.3-2
 REVISION 20
 DECEMBER 1996

11.4 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS

11.4.1 Design Objectives

Davis-Besse commits to the regulatory position of Regulatory Guide 4.15 (Revision 1, February 1979).

The radiological monitoring system was designed to:

- a. Continuously detect and record the level of radioactivity in certain process streams and all station effluent streams to ensure compliance with the requirements of 10CFR20, 10CFR50, and Safety Guide Number 21. One exception is the station effluent discharge from the settling basins to the collection box to the lake which requires a grab sample to be obtained to determine the level of radioactivity. In lieu of continuously monitoring the discharge from the collection box to the lake, the settling basins are sampled and analyzed in accordance to the ODCM requirements. The methodology provided in the ODCM is acceptable for use in demonstrating compliance with concentration limits of 10 CFR 20, 10 CFR 50 and Safety Guide Number 21.
- b. Provide operating personnel with a continuous indication of the beta and gamma-emitting radioactivity in selected station areas during both normal operation and postulated occurrences.
- c. Provide alarm indication on all monitored points for activity level increases to a calculated maximum tolerance level and initiate protective functions, as required, on activity level increases above these tolerance values. All activity level measurements are acted upon; no discrimination is made between normal operational conditions and accident situations.

The radiological monitoring system provided to fulfill the above design objectives is outlined in Table 11.4-1.

Table 11.4-1 lists the process and effluent radioactivity monitors, specifying the detector types, measurement ranges, and sensitivities. Detector sensitivities are stated with respect to the predominant radionuclide in the system to be monitored. Liquid monitors are described first, then gaseous and airborne monitors.

11.4.2 Continuous Monitoring

11.4.2.1 General Design

Each channel of the process radioactivity monitoring system consists of remotely-located, interconnected subsystems. The detectors are located within the station adjacent to the monitored process. The control readout, recording, and power supply instrumentation is installed in the control cabinet room with alarm indication located in the main control room. Each monitor channel has, along with the indicators designated in Subsection 11.4.2.2, an indicator in the cabinet room. Each monitor channel has alarm functions along with complementary switches to initiate operational functions on high radioactivity levels.

The system is redundant through the measurement of the station effluents, both at their point of release and at their potential radioactivity source points within the station. A duplication of

instruments, to increase the level of confidence of measurements, is provided for certain station process effluent streams.

The monitor ranges were designed for all operational modes, including transients, accidents, and post-accident conditions. Instrument sensitivities are sufficient to provide resolution between signal and background measurements to a high degree of confidence for normal release rates.

The instrument ranges and setpoint values were determined using the criteria set down in 10CFR20 and 10CFR50. The alarm setpoints are adjustable and are set according to operational experience. These setpoints are placed at values consistent with requirements of the Radiation Protection Program, in order that significant changes in activity levels will be detected.

The subject radioactivity monitoring systems have been designed to meet all operational requirements with a maximum of reliability. The electronic circuits, with the exception of the detectors, are of solid state design and construction. Each channel has two methods for performance evaluation. The electronic circuitry is self-monitoring with an alarm initiation on malfunction or loss of measurement capability. Each detector's operational ability is periodically checked by exposure to a source of sufficient energy to provide an upscale response. Exposure to the check source is remotely controlled from the detector's control unit.

Each monitor channel initiating an operational function, in a Q-listed station process, to restrict the release of radionuclides above their regulatory limit was fabricated under a quality control program and designed and constructed to withstand destructive seismic forces as defined by the station maximum possible (larger) earthquake.

Each monitor channel performing essential functions is electrically and physically isolated from any other channel, draws its primary electrical power from an essential bus and is designed to comply to the IEEE Standard No. 279,1971. The monitor channels performing no essential functions are supplied from two non-essential uninterruptible AC busses. One exception is the Station Effluent Detector which is supplied from a single non-essential interruptible AC Bus.

The detector unit and local readout instruments are powered from their respective control unit located in the control cabinet room. One exception is the station effluent detector which is powered from local control unit.

The recorders and remote indicators are driven by an analog signal. Power for the recorders and remote indicators is taken from the bus supplying the panel on which they are located. Isolation, when required on the analog output circuits, is achieved by isolation transducers.

Alarm indication is achieved by the station computer, Fire Detection System, Radiation Monitoring System, or station annunciators located in the main control room. The computer and the Fire Detection System Network Display Unit (NDU) activate their respective printers and CRTs located in the main control room. The station annunciators are of the audible-visual type. The initiation of the alarm sequence is achieved through a set of electrical contacts in the respective monitor control unit. Power for all the modes of alarm initiation is taken from the nonessential uninterruptible ac bus.

The sample system for each off-line monitor was designed to provide representative samples to their respective detectors. For ventilation system particulate activity, where the velocity in the ventilation system is constant, the system was designed for isokinetic sampling. Each gas

monitor employing filter-type detectors has flow switches which indicate a high or low flow condition. The isokinetic nozzles were sized such that the velocity entering the port is the same as the velocity in the stream being sampled. In the case of the station vent where the velocity will vary, the system was designed for maximum sample flow. When the sampling switches to the accident range monitor, the flow through the accident range monitor is held constant by a manually adjusted metering valve.

11.4.2.2 Specific Design

11.4.2.2.1 Reactor Coolant Letdown

This system has a single, off-line detector continuously monitoring the reactor coolant letdown stream. The monitor functions as a failed fuel detector by measuring the Cs-138 concentration in this stream. The measurement is recorded in the cabinet room. The alarm initiation value is set at a value which will detect failed fuel.

11.4.2.2.2 Clean and Miscellaneous Radioactive Waste Systems

The discharge headers of the clean and the miscellaneous liquid radioactive waste systems each has two continuously operating radioactivity detecting systems. Both detectors measure gross radioactivity. The measurement from each detector is indicated on the radwaste panel (Location, Auxiliary Building, Elevation 545) and recorded in the cabinet room. Each detector initiates an alarm function and the closure of the discharge header valves when the measurement values reach a predetermined fraction of EC Value times the dilution factor. This is described in Subsection 11.2.7 for the discharge flow rate stated in Table 11.4-1.

11.4.2.2.3 Component Cooling Water System

Two detector systems, one on each of the main suction lines, monitor the activity of the component cooling water system by measuring gross activity. An alarm function and the closure of the atmospheric vent valve on the CCWS surge tank are initiated when the activity level as measured by either detector increases to a specific limit. The CCWS then relieves to the miscellaneous liquid radioactive waste system for over-pressure protection. When the detector is measuring gross activity or Cs-137, the alarm initiation and valve closure will be set according to operational experience consistent with regulatory requirements in order that significant changes in activity levels will be detected.

11.4.2.2.4 Station Vent Monitoring System

Two detection systems, each containing a normal and an accident range monitor, continuously monitor the unit vent for particulate, iodine, and gross gaseous radioactivity. Each normal range monitor consists of three detector units, passivated implanted planar silicon (PIPS) alpha-beta unit for the particulate channel, a charcoal cartridge scintillation unit for the iodine channel, and a low range PIPS beta unit for the noble gas channel. Each accident range monitor consists of a high range flow-through ionization chamber for the noble gas channel and two moveable shielded particulate and iodine samplers. The activity sensed by each of the normal range monitor's detectors is recorded in the cabinet room. Each accident range noble gas monitor output is recorded in the cabinet room. Detector setpoints are established so as not to exceed the offsite dose limits required by the Offsite Dose Calculation Manual (ODCM). Control Room isolation is also initiated when the activity level exceeds the setpoint.

11.4.2.2.5 Containment Vessel Monitor*

The containment vessel particulate and noble gas monitoring system continuously draws a sample from either of two sample lines originating from different points within the containment vessel. Each monitor has a fiberglass filter beta scintillation detector particulate channel and a beta sensitive noble gas channel.

Greater than normal particulate or noble gas activity in the containment can indicate an increase in reactor coolant leakage. An alarm is initiated if greater than normal particulate or noble gas activity concentration has been detected in the containment vessel atmosphere.

11.4.2.2.6 Waste Gas Monitor

The waste gas discharge line is continuously monitored by at least one of two off-line detectors, each measuring gross activity. The measurement is read out on the radwaste panel (Location: Auxiliary Building, Elevation 545) and recorded in the cabinet room. An alarm function and the closing of the waste discharge header valves are initiated prior to exceeding the setpoint established so as not to exceed the offsite dose rate limit as required by the ODCM.

11.4.2.2.7 Vacuum System Discharge

The Vacuum System discharge is continuously monitored for non-condensable gases from the condenser by at least one of two off-line detectors, each measuring gross activity. The measurement is used to detect a primary to secondary leak in the steam generators. The measurement is recorded in the cabinet room. The alarm setpoints for the gamma and beta scintillation detector are set according to operational experience consistent with regulatory requirements in order that significant changes in activity levels will be detected.

11.4.2.2.8 Main Steam Lines

Each main steam header has one continuously operating N₁₆ monitor to detect primary to secondary leakage.

11.4.2.2.9 Service Water System

A single off-line detector continuously monitors the service water stream before it enters into the station effluent stream. The instrument measures gross radioactivity. The alarm setpoint for the detector is set according to operational experience consistent with regulatory requirements in order that significant changes in activity levels will be detected.

11.4.2.2.10 Intake Forebay

A single detector continuously monitors the water taken into the station. The instrument measures gross radioactivity. The measurement is recorded in the cabinet room. The alarm setpoint for the detector is set according to operational experience consistent with regulatory requirements in order that significant changes in activity levels will be detected.

*Denotes an essential monitor

11.4.2.2.11 Station Effluent

The effluent released from the station is sampled and analyzed and or continuously monitored by required monitors in accordance to ODCM. The ODCM identifies "Required Monitors" for liquid effluents capable of initiating an alarm and an automatic termination of the release. The methodology provided in the Offsite Dose Calculation Manual (ODCM) demonstrates compliance with the regulatory requirements in order that significant changes in activity levels will be detected.

11.4.2.2.12 Unit Storm Sewer Outlet

A single off-line detector continuously monitors the station water's final discharge path to the environment. The instrument measures gross radioactivity. The measurement is recorded in the cabinet room. The alarm setpoint for the detector is set according to operational experience consistent with regulatory requirements in order that significant changes in activity level will be detected.

11.4.2.2.13 Deleted

11.4.3 Sampling

11.4.3.1 Description

The radioactive waste process systems, Miscellaneous Liquid Radwaste, Clean Liquid Radwaste, Gaseous Radwaste, are provided with grab sample points in order to better control the release of radioactivity. Prior to batch releases, grab samples are taken in order to set the maximum release rate. If this rate is exceeded, automatic closure of isolation valves will result.

The final effluent streams are provided with grab sampling capabilities to be used to confirm the inline process monitors' readings at any time.

The design basis and system description are given in Section 9.3.2.

11.4.3.2 Operation

The expected radionuclide composition of the various radwaste process system samples used for monitoring to discharge is shown in Table 11.4-2.

The normal parameters that will be monitored on these samples are shown in Table 11.4-3 together with the minimum sampling frequency.

11.4.3.2.1 Miscellaneous Waste Monitor Tank, Detergent Waste Drain Tank and Clean Waste Monitor Tanks

These tanks are all a part of the radioactive waste treatment system. The water in these tanks is locally grab sampled after appropriate mixing and purging of sample lines to ensure an accurate sample. The samples are tested for radioactivity to ensure compliance with Offsite Dose Calculation Manual (ODCM) limits for release to the environment.

The maximum expected radionuclide activities are shown in Table 11.4-2. The detergent waste drain tank gross activity is expected to be 10^{-5} $\mu\text{Ci/ml}$.

11.4.3.2.2 Waste Gas Decay Tanks

The contents of these tanks are sampled and tested prior to release. Samples of the gas from these sources are collected in a sample container at a location in the auxiliary building near the tanks. The sample container is installed in the line by its connectors and the upstream and downstream valves are opened. The valves in the sample line are then opened, one ahead of and one behind the container. A flow path is established between the gas area being sampled and the vent header. Gas is allowed to flow until the line and the sample container have been thoroughly purged. The two valves on both sides of the sample container are closed and the other valves returned to their original positions. The sample container is removed from the sampling station and transported to the laboratory for analysis.

The expected gaseous activity is shown in Table 11.4-2. These samples are tested to ensure compliance with ODCM limits for releases to the environment.

11.4.3.2.3 Station Vent

This sample will determine the concentration of radionuclides being released to the environment. Analysis at this point will enable calculations of concentrations and doses at the site boundary. The expected concentration is shown in Table 11.4-2. The sample is collected in a manner similar to that used for the waste gas tank sampling.

The station vent does not provide isokinetic sampling as required by NUREG 0737, Item II.F.1, Attachment 2. The NRC has granted Davis-Besse a variance (Log 1722 and Log 1-2748) from this requirement provided that no other sources enter the vent stack downstream of the HEPA filters and that the amount of material released is calculated using a correction factor for the flow.

11.4.3.2.4 Circulating Water

The sample is collected from the circulating water pump's discharge as a grab sample. Analysis for gross activity is performed to confirm the absence of radioactivity from this discharge point.

11.4.4 Calibration and Maintenance

All process and effluent radiation monitors are functionally checked and calibrated as required by the ODCM or Technical Specifications if applicable. Other process and effluent monitors are calibrated every 18 months with the exception of the Main Steam Line Radiation Monitors. The Main Steam Line Radiation Monitors are calibrated every refueling interval. Calibration is performed with National Institute of Standards & Technology (formerly National Bureau of Standards) traceable standards.

The calibration procedure exposes each detector to a known quantity of radioactivity in a constant reproducible geometry. The calibration equipment is portable to facilitate the calibration of the complete system from detector to indicator and/or recorder.

To reduce the required maintenance, an extensive pre-installation burn-in and calibration was performed to locate and eliminate potential malfunctions in the electronic circuitry.

The off-line sample systems, filters and pumps, are maintained on a periodic basis as determined by the manufacturer's recommendations and station operational experience.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.4-1

Liquid, Gas, and Airborne Radiation Monitors

LIQUID

<u>System Designation</u>	<u>Required Measurements</u>	<u>Sensitivity μCi/cc</u>	<u>Range μCi/cc (CPM)</u>	<u>Type Detector</u>	<u>Flowrates (gpm) (monitored process stream)/ (respective unit effluent)</u>
Reactor Coolant (Failed fuel detector) RE-1998	Cs-138 and gross gamma	1.42×10^{-6}	2.58×10^{-7} to 2.58×10^{-1} (10 to 10 ⁷)	Gamma Scintillation off-line (snow plow)	150/none
Misc. radwaste system outlet RE-1878A	gross radioactivity	2.59×10^{-7}	1.09×10^{-7} to ⁷ 1.09×10^{-1} (10 to 10 ⁷)	Gamma scintillation off-line	70/20000
RE-1878B	gross radioactivity	2.59×10^{-7}	1.09×10^{-7} to 1.09×10^{-1} (10 to 10 ⁷)	Gamma scintillation off-line	70/20000
Clean radwaste system outlet RE-1770A	gross radioactivity	6.81×10^{-7}	1.09×10^{-7} to 1.09×10^{-1} (10 to 10 ⁷)	Gamma scintillation off-line	70/20000
RE-1770B	gross radioactivity	6.81×10^{-7}	1.09×10^{-7} to 1.09×10^{-1} (10 to 10 ⁷)	Gamma scintillation off-line	70/20000
Component Cooling Return line RE-1412 RE-1413	gross radioactivity	1.23×10^{-6}	1.87×10^{-7} to 1.87×10^{-2} (10 to 10 ⁶)	Gamma Scintillation Off-line (snow plow)	7900/none
Main steam Line RE-600 RE-609	N-16	6.05×10^{-4}	1.1×10^{-4} to 1.1×10^{-2} (10 to 10 ⁷)	Gamma Scintillation off-line (snow plow)	5.9×10^6 lbs./hr/none
Service Water Outlet RE-8432	gross radioactivity	2.07×10^{-7}	1.09×10^{-7} to 1.09×10^{-1} (10 to 10 ⁷)	Gamma scintillation off-line	20000

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.4-1 (Continued)

Liquid, Gas, and Airborne Radiation Monitors

LIQUID

<u>System Designation</u>	<u>Temp. (°F) Norm/Max.</u>	<u>Press. (psig) Norm/Max.</u>	<u>Pipe Size, ANSI Rating & Material</u>	<u>Piping Design Code Classification</u>	<u>Quantity and Remarks</u>
Reactor coolant (Failed fuel Detector) RE-1998	120/200	150	3" – 150# Stn. stl.	ASME III Class 3	1 detector channel
Misc. radwaste system outlet RE-1878A	120/200	65/70	3" – 150# Stn. stl.	ASME III Class 3	2 detector channels, 1 per isotope
RE-1878B	120/200	65/70	3" – 150# Stn. stl.	ASME III Class 3	
Clean radwaste system outlet RE-1770A	120/200	90/95	3" – 150# Stn. stl.	ASME III Class 3	2 detector channels, 1 per isotope
RE-1770B	120/200	90/95	3" – 150# Stn. stl.	ASME III Class 3	
Component Cooling Return line RE-1412 RE-1413	110/150	10/17	20" – 150# Carb. stl.	ASME III Class 3	2 detector channels
Main steam line RE-600 RE-609	600	1155	36" – 600# Carb. stl.	ASME III Class 2	2 detector channels
Service water outlet RE-8432	48-100/150	5/50	30" – 150# Carb. stl.	ASME III Class 3	1 detector channel

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.4-1 (Continued)

Liquid, Gas, and Airborne Radiation Monitors

AIRBORNE AND GAS

<u>System Designation</u>	<u>Required Measurements</u>	<u>Sensitivity μCi/cc</u>	<u>Range μCi/cc (CPM)</u>	<u>Type Detector</u>	<u>Flowrates (gpm) (monitored process stream)/ (respective unit effluent)</u>
Collection box outlet to lake RE-8433	Abandoned				
Intake Forebay RE-8434	gross radioactivity	2.07×10^{-7}	1.09×10^{-7} to 1.09×10^{-1} (10 to 10 ⁷)	Gamma Scintillation off-line	Variable (1700 max)
Unit storm sewer outlet RE-4686	gross radioactivity	2.10×10^{-7}	1.09×10^{-7} to 1.09×10^{-1} (10 to 10 ⁷)	Gamma Scintillation off-line	
Condensate demineralizer backwash pump discharge RE-744	gross radioactivity	Deactivated in place		Gamma Scintillation off-line	125/2000

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.4-1 (Continued)

Liquid, Gas, and Airborne Radiation Monitors

LIQUID

<u>System Designation</u>	<u>Temp. (°F) Norm/Max.</u>	<u>Pressure (psig) Norm/Max</u>	<u>Pipe Size, ANSI Rating & Material</u>	<u>Piping Design Code Classification</u>	<u>Quantity and Remarks</u>
Collection box outlet to lake RE-8433	Abandoned				
Intake Forebay RE-8434	40-90/90	90/140	10" – 150# Carb. stl.	ANSI B31.1.0	1 detector channel
Unit storm sewer outlet RE-4686	32-81		N/A		1 detector channel
Condensate demineralizer backwash pump discharge RE-744	167/175	10/150	3" – 150# Carb. stl.	ANSI B31.1.0	1 detector channel

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.4-1 (Continued)

Liquid, Gas, and Airborne Radiation Monitors

AIRBORNE AND GAS

<u>System Designation</u>	<u>Required Measurements</u>	<u>Sensitivity μCi/cc</u>	<u>Range μCi/cc (CPM)</u>	<u>Type Detector</u>	<u>Flowrates (scfm) (monitored process stream)/ (respective unit effluent)</u>
Radioactive waste gas outlet RE-1822A	gross radioactivity	1.53×10^{-7}	6.45×10^{-8} to 6.45×10^{-2} (10 to 10^7)	Gamma Scintillation off-line	50/6.3 x 10^4
RE-1822B	gross radioactivity	3.10×10^{-6}	2.99×10^{-6} to 2.99 (10 to 10^7)	Beta Scintillation off-line	50/6.3 x 10^4
Fuel handling area, exhaust system RE-5403A	gross radioactivity	2.70×10^{-12}	1.83×10^{-12} to 1.83×10^{-6} (10 to 10^7)	Moving paper tape particulate filter-detector Beta Scintillation	2 x 10^4 /6.3 x 10^4
RE-5403B	I-131	1.00×10^{-11} (in 12 hours)	1.64×10^{-9} to 1.64×10^{-3} (10 to 10^7)	Fixed charcoal filter- detector Gamma Scintillation	2 x 10^4 /6.3 x 10^4
RE-5403C	gross radioactivity	^{**} 2.00×10^{-7}	2.76×10^{-7} to 2.76×10^{-1} (10 to 10^7)	Beta Scintillation	2 x 10^4 /6.3 x 10^4
Radwaste area, exhaust system RE-5405A	gross radioactivity	2.70×10^{-12}	1.83×10^{-12} to 1.83×10^{-6} (10 to 10^7)	Moving paper tape particulate filter-detector Beta Scintillation	4.4 x 10^4 / 6.3 x 10^4
RE-5405B	I-131	1.00×10^{-11} (in 12 hours)	1.64×10^{-9} to 1.64×10^{-3} (10 to 10^7)	Fixed charcoal filter- detector Gamma Scintillation	4.4 x 10^4 /6.3 x 10^4

** The sensitivity for the gas channel range is based upon a single isotope rather than a mixture of isotopes.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.4-1 (Continued)

Liquid, Gas, and Airborne Radiation Monitors

AIRBORNE AND GAS

<u>System Designation</u>	<u>Temp. (°F) Norm/Max.</u>	<u>Pressure (psig) Norm/Max</u>	<u>Pipe Size, ANSI Rating & Material</u>	<u>Piping Design Code Classification</u>	<u>Quantity and Remarks</u>
Radioactive waste gas outlet					
RE-1822A	150/200	10/10	1 1/2" – 150# Stn. stl.	ASME III Class 3	2 detector channels, 1 per isotope
RE-1822B	150/200	10/10	1 1/2" – 150# Stn. stl.	ASME III Class 3	
Fuel handling area, exhaust system					
RE-5403A	60/120	Atmos. plus	Rectangular duct 46" x 36"		1 off-line system, 3 detector channels per system
RE-5403B	60/120	Atmos. plus	Rectangular duct 46" x 36"		
RE-5403C	60/120	Atmos. plus	Rectangular duct 46" x 36"		
Radwaste area, exhaust system					
RE-5405A	60/120	Atmos. plus	Rectangular duct 46" x 36"		1 off-line system, 3 detector channels per system
RE-5405B	60/120	Atmos. plus	Rectangular duct 46" x 36"		

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.4-1 (Continued)
Liquid, Gas, and Airborne Radiation Monitors

AIRBORNE AND GAS

<u>System Designation</u>	<u>Required Measurements</u>	<u>Sensitivity μCi/cc</u>	<u>Range μCi/cc (CPM)</u>	<u>Type Detector</u>	<u>Flowrates (scfm) (monitored process stream)/ (respective unit effluent)</u>
RE-5405C	gross radioactivity	$2.00 \times 10^{-7**}$	2.76×10^{-7} to 2.76×10^{-1} (10 to 10^7)	Beta Scintillation	4.4×10^4 6.3×10^4
Station Vent					
RE-4598AAA RE-4598BAA	particulate gross radioactivity		2.7×10^{-11} to 1×10^{-4}	paper filter-alpha beta gamma silicon	6.3 to 7.1×10^4
RE-4598AAB RE-4598BAB	I-131 radioactivity		1×10^{-11} to 1×10^{-4}	charcoal cartridge- NaI:Crystal and PM tube	6.3 to 7.1×10^4
RE-4598AAC RE-4598BAC	noble gas gross radioactivity (Xe-133)		1×10^{-7} to 1×10^{-1}	Alpha beta gamma silicon	6.3 to 7.1×10^4
RE-4598ABC RE-4598BBC	noble gas gross radioactivity (Xe-133)		1×10^{-4} to 1×10^5	Ion chamber	6.3 to 7.1×10^4
Containment Vessel					
RE-4597AAA RE-4597BAA	particulate gross radio- activity		1×10^{-11} to 1×10^{-6}	Fiberglass filter-beta scintillation	N/A
RE-4597AAC RE-4597BAC	noble gas gross radioactivity (Xe-133)	2.62×10^{-6}	1×10^{-11} to 1×10^{-1}	Beta scintillation	N/A

** The sensitivity for the gas channel range is based upon a single isotope rather than a mixture of isotopes.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.4-1 (Continued)

Liquid, Gas, and Airborne Radiation Monitors

AIRBORNE AND GAS

<u>System Designation</u>	<u>Temp. (°F) Norm/Max.</u>	<u>Pressure (psig) Norm/Max</u>	<u>Pipe Size, ANSI Rating & Material</u>	<u>Piping Design Code Classification</u>	<u>Quantity and Remarks</u>
RE-5405C	60/120	Atmos. plus	Rectangular duct 46" x 36"		
Station Vent					
RE-4598AAA	50/120	0/1	84" dia		2 off-line systems
RE-4598BAA			carb. stl.		
RE-4598AAB	50/120	0/1	84" dia		
RE-4598BAB			carb. stl.		
RE-4598AAC	50/120	0/1	84" dia		
RE-4598BAC			carb. stl.		
RE-4598ABC	50/120	0/1	84" dia		
RE-4598BBC			carb. stl.		
Containment Vessel					
RE-4597AAA	100/140	0/10	1" – 150#	ASME III	
RE-4597BAA			stn. stl.	Class 2	
RE-4597AAC	100/140	0/10	1" – 150#	ASME III	
RE-4597BAC			stn. stl.	Class 2	

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.4-1 (Continued)

Liquid, Gas, and Airborne Radiation Monitors

AIRBORNE AND GAS

<u>System Designation</u>	<u>Required Measurements</u>	<u>Sensitivity μCi/cc</u>	<u>Range μCi/cc (CPM)</u>	<u>Type Detector</u>	<u>Flowrates (scfm) (monitored process stream)/ (respective unit effluent)</u>
Vacuum System Discharge					
RE-1003A	gross radioactivity	2.10×10^{-7}	1.09×10^{-7} to 1.09×10^{-1} (10 to 10^7)	Gamma Scintillation off-line	0 to 100 15 (norm.)
RE-1003B	gross radioactivity	3.27×10^{-7}	3.17×10^{-7} to 3.17×10^{-1} (10 to 10^7)	Beta Scintillation off-line	0 to 100 15 (norm.)
Containment purge exhaust filter					
RE-5052A	gross radioactivity	3.66×10^{-12}	3.05×10^{-12} to 3.05×10^{-6} (10 to 10^7)	Moving paper tape particulate filter-detector Beta Scintillation	5×10^4 Norm plus <u>8×10^3 LOCA</u> $*1.21 \times 10^5$
RE-5052B	I-131	7.40×10^{-10}	2.34×10^{-9} to 2.34×10^{-3} (10 to 10^7)	Fixed charcoal filter- detector Gamma Scintillation	5×10^4 Norm plus <u>8×10^3 LOCA</u> $*1.21 \times 10^5$
RE-5052C	gross radioactivity	2.84×10^{-7}	2.76×10^{-7} to 2.76×10^{-1} (10 to 10^7)	Beta Scintillation	5×10^4 Norm plus <u>8×10^3 LOCA</u> $*1.21 \times 10^5$
Control room, emergency ventilation fan discharge					
RE-5327A RE-5328A	gross radioactivity	2.21×10^{-12}	1.84×10^{-12} to 1.84×10^{-7} (10 to 10^6)	Moving paper tape particulate filter-detector Beta Scintillation	2900/each system

* When this system is employed, its respective effluent stream increases an equivalent amount.

** The sensitivity for the gas channel range is based upon a single isotope rather than a mixture of isotopes.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.4-1 (Continued)

Liquid, Gas, and Airborne Radiation Monitors

AIRBORNE AND GAS

<u>System Designation</u>	<u>Temp. (°F) Norm/Max.</u>	<u>Pressure (psig) Norm/Max</u>	<u>Pipe Size, ANSI Rating & Material</u>	<u>Piping Design Code Classification</u>	<u>Quantity and Remarks</u>
Vacuum system discharge RE-1003A	130/150	18	10" – 150# Carb. stl.	ANSI B31.1.0	2 detector channels
RE-1003B	130/150	18	10" – 150# Carb. stl.	ANSI B31.1.0	
Containment purge exhaust filter RE-5052A	60/120	Atmos. plus	48" dia		1 off-line system, 3 detector channels per system
RE-5052B	60/120	Atmos. plus	48" dia		
RE-5052C	60/120	Atmos. plus	48" dia		
Control room, emergency ventilation fan discharge RE-5327A RE-5328A	60/120	Atmos	Rectangular duct 34" x 12"		2 off-line systems, 3 detector channels per system

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.4-1 (Continued)

Liquid, Gas, and Airborne Radiation Monitors

AIRBORNE AND GAS

<u>System Designation</u>	<u>Required Measurements</u>	<u>Sensitivity μCi/cc</u>	<u>Range μCi/cc (CPM)</u>	<u>Type Detector</u>	<u>Flowrates (scfm) (monitored process stream)/ (respective unit effluent)</u>
RE-5327B RE-5328B	I-131	3.40×10^{-10}	1.08×10^{-9} to 1.08×10^{-4} (10 to 10 ⁶)	Fixed charcoal filter- detector Gamma Scintillation	2900/each system
RE-5327C RE-5328C	gross radioactivity	3.27×10^{-7}	3.17×10^{-7} to 3.17×10^{-2} (10 to 10 ⁶)	Beta Scintillation	2900/each system
<u>System Designation</u>	<u>Temp. (°F) Norm/Max.</u>	<u>Pressure (psig) Norm/Max</u>	<u>Pipe Size, ANSI Rating & Material</u>	<u>Piping Design Code Classification</u>	<u>Quantity and Remarks</u>
RE-5327B RE-5328B	60/120	Atmos	Rectangular duct 34" x 12"		
RE-5327C RE-5328C	60/120	Atmos	Rectangular duct 34" x 12"		

TABLE 11.4-2

Specific Activities At Sample Points in the Liquid & Gaseous Systems
(Average Activities with 0.1% Failed Fuel)

Radionuclide	Activities in Liquid Systems		Activities in Gaseous Systems		
	Clean Waste Monitor	Miscellaneous Waste	Radionuclide	Decay Tank	Station Vent
	Tank	Monitor Tank		(30 Days Decay)	
	(μCi/ml)	(μCi/ml)		(μCi/cc)	(μCi/cc)
<u>Fission Products (dissolved)</u>			<u>Noble Gases</u>		
Rb-88	1.00 x 10 ⁻⁸	4.00 x 10 ⁻⁷	Kr-85	1.50	4.25 x 10 ⁻⁷
Sr-89	1.00 x 10 ⁻⁸	1.00 x 10 ⁻⁸	Xe-131m	6.48 x 10 ⁻²	3.18 x 10 ⁻⁸
Sr-90	1.00 x 10 ⁻⁸	1.00 x 10 ⁻⁸	Xe-133m	5.50 x 10 ⁻⁶	1.58 x 10 ⁻¹¹
Sr-91	1.00 x 10 ⁻⁸	1.00 x 10 ⁻⁸	Xe-133	7.84 x 10 ⁻¹	2.43 x 10 ⁻⁷
Sr-92	1.00 x 10 ⁻⁸	1.00 x 10 ⁻⁸	<u>Halogens</u>		
Y-90	1.00 x 10 ⁻⁸	1.00 x 10 ⁻⁸	I-131	4.38 x 10 ⁻⁷	5.70 x 10 ⁻¹³
Y-91	1.00 x 10 ⁻⁸	1.00 x 10 ⁻⁸			
Mo-99	1.03 x 10 ⁻⁷	6.10 x 10 ⁻⁷			
I-131	5.39 x 10 ⁻⁷	2.40 x 10 ⁻⁶			
I-132	3.75 x 10 ⁻⁷	1.67 x 10 ⁻⁶			
I-133	6.30 x 10 ⁻⁷	2.80 x 10 ⁻⁶	I-132	0	2.83 x 10 ⁻¹³
I-134	7.64 x 10 ⁻⁸	3.40 x 10 ⁻⁷	I-133	0	4.73 x 10 ⁻¹³
I-135	3.17 x 10 ⁻⁷	1.41 x 10 ⁻⁶	I-134	0	5.78 x 10 ⁻¹⁴
Cs-134	1.03 x 10 ⁻⁷	6.10 x 10 ⁻⁷	I-135	0	2.38 x 10 ⁻¹³
Cs-136	1.85 x 10 ⁻⁷	1.10 x 10 ⁻⁷	<u>Particulates</u>		
			Rb-87	1.57 x 10 ⁻¹²	
			Cs-135	1.45 x 10 ⁻⁷	

TABLE 11.4-2 (Continued)

Specific Activities at Sample Points in the Liquid & Gaseous Systems
(Average Activities with 0.1% Failed Fuel)

Radionuclide	Activities in Liquid Systems		
	Purification Demineralizer Inlet ($\mu\text{Ci/ml}$)	Clean Waste Monitor Tank ($\mu\text{Ci/ml}$)	Miscellaneous Waste Monitor Tank ($\mu\text{Ci/ml}$)
<u>Fission Products (gaseous)</u>			
Kr-83m	2.88×10^{-2}	2.88×10^{-7}	
Kr-85m	0.153	1.53×10^{-6}	
Kr-85	0.315	3.15×10^{-6}	
Kr-87	8.37×10^{-2}	8.37×10^{-7}	
Kr-88	0.268	2.68×10^{-6}	
Xe-131m	0.225	2.25×10^{-6}	
Xe-133m	0.284	2.84×10^{-6}	
Xe-133	24.9	2.49×10^{-4}	
Xe-135m	9.26×10^{-7}	9.26×10^{-7}	
Xe-135	0.508	5.08×10^{-6}	
Xe-138	5.11×10^{-2}	5.11×10^{-7}	
Cs-137	1.27	3.18×10^{-7}	1.88×10^{-6}
Cs-138	7.34×10^{-2}	1.84×10^{-8}	1.09×10^{-7}
Ba-137m	1.117	2.93×10^{-7}	1.74×10^{-6}
Ba-139	7.60×10^{-5}	1.00×10^{-8}	1.13×10^{-8}
Ba-140	5.63×10^{-6}	1.00×10^{-8}	1.00×10^{-8}
La-140	2.25×10^{-6}	1.00×10^{-8}	1.00×10^{-8}
Ce-144	5.18×10^{-7}	1.00×10^{-8}	1.00×10^{-8}
<u>Insoluble Corrosion Products</u>			
Cr-51	3.69×10^{-3}	9.23×10^{-3}	1.38×10^{-5}
Mn-54	4.23×10^{-4}	1.06×10^{-8}	1.58×10^{-6}
Fe-55	1.46×10^{-2}	3.65×10^{-7}	5.45×10^{-5}
Fe-59	4.23×10^{-4}	1.06×10^{-8}	1.58×10^{-6}
Co-58	2.22×10^{-2}	5.55×10^{-7}	8.25×10^{-5}
Co-60	1.19×10^{-4}	1.00×10^{-8}	4.45×10^{-7}
Zr-95	2.91×10^{-2}	7.28×10^{-7}	1.08×10^{-4}

TABLE 11.4-3

Parameters to be Monitored and Sample Frequency

<u>Sample Point</u>	<u>Parameters to be Monitored</u>	<u>Frequency</u>
1. Miscellaneous Waste Monitor Tank, Detergent Waste Drain Tank, and Clean Waste Monitor Tanks	a) Gamma Isotopic Analysis b) Dissolved and Entrained Noble Gases c) Tritium Concentration, Gross Alpha d) Strontium-89/90, Fe-55	Prior to release of each batch 1 batch/Month Monthly (2) Quarterly (2)
2. Waste Gas Decay Tanks	a) Gamma Isotopic Analysis, Tritium	Prior to release of each batch
3. Station Vent	a) Iodine – 131 and Particulate Radioactivity (1) b) Gamma Isotopic Analysis, Tritium c) Gross Gamma d) Strontium-89/90	Weekly (3) Monthly grab Noble Gas Monitor Quarterly (4)
4. Circulating Water	Gross Beta – Gamma Activity	Monthly

- (1) When iodine or particulate radioactivity levels exceed 10 percent of the limit in the Offsite Dose Calculation Manual (ODCM) the sampling frequency is increased to a minimum of once each day.
- (2) This analysis is performed on a monthly or quarterly composite sample made up of representative portions from each batch.
- (3) The station vent is sampled continuously; the sample media is changed and analyzed weekly.
- (4) The analysis is performed on a monthly or quarterly composite sample made up of representative portions from each weekly analysis.

11.5 SOLID WASTE SYSTEM

11.5.1 Design Objectives

Davis-Besse commits to the regulatory position of Regulatory Guide 4.15 (Revision 1, February 1979).

The solid waste system was designed to receive, process, package, and store all of the solid wastes generated at the Davis-Besse Nuclear Power Station. The system was designed to accomplish this task in a manner so simple as to minimize the possibility of radioactive material release and personnel exposure.

The materials handled by the solid waste system include bead-type resins, spent filter cartridges, powdered resins, and miscellaneous solid waste such as paper, rags, contaminated clothing, gloves and shoe coverings. The solid waste system area was designed to provide the necessary shielding to prevent the overexposure of operating personnel to radioactive sources. This is accomplished through the use of lead shielding, concrete shielding, and safe operating procedures.

11.5.2 System Inputs

This section is the original design basis for the solid radwaste system, and as such, has been left as originally written.

As previously noted, the primary sources of solid radwaste are:

- Spent bead resins
- Spent powdered resins
- Spent filter cartridges
- Misc. paper, clothing, etc.

Table 11.5-1 gives estimates of the total maximum and average quantities of the various radionuclides, excluding tritium contributed annually by each of these sources during a fuel cycle. In compiling this table, it was generally assumed that the estimates of maximum and average radionuclide inventories were based, respectively, on 1% and 0.1% failed fuel conditions occurring annually during an equilibrium fuel cycle. Whenever reactor coolant specific activities were needed for calculation, the values used for dissolved radionuclides were the maximum levels, corrected to correspond to a coolant density of 1 g/ml, attained annually in the primary system during an equilibrium fuel cycle. For corrosion products, the values given in Table 11.1-7, also modified to correspond to a liquid density of 1 g/ml, were used. The estimated quantities (volumes) of the above wastes are given in Table 11.5-2.

Other assumptions used in Tables 11.5-1 and 11.5-2 are as follows:

a. Spent Bead Resins:

Quality: The quality is based on each demineralizer handling potentially radioactive liquids being flushed once a year. This should be conservative since duplicate units, each sized to last for an entire year are provided in several places.

Activity: Although some spent resins will have processed spent fuel pool water, it is assumed that primary system coolant is, directly or indirectly, the only source of activity on resins. Further, to be conservative, it is assumed that the inventories of ionic and insoluble radionuclides found on the resins are essentially the maximum amounts that could accumulate from incore fuel leakage during a year of reactor operations. Estimates of the quantities involved were derived as follows:

Case 1: Dissolved radionuclides with Short Half-lives which are Normally Removed by the Purification Demineralizer and Which may have Relatively Long Lived Parent Nuclides Present in Significant Amounts (Rb-88, Sr-89, Sr-91, Sr-92, I-131, I-132, I-133, I-134, I-135, Ba-139, Ba-140, La-140)

The total activity due to each of the radionuclides listed above that can accumulate on the bead resins used for processing, directly or indirectly, reactor coolant is estimated using the following expression:

$$\text{Total Act.} = \frac{(V \times M_i)}{1} + \frac{[(L \times M_i / \lambda_i) + (L \times M_p / \lambda_p)]}{2}$$

where the subscript (i) denotes the nuclide under consideration and (p) the parent nuclide (if the parent is present in a significant amount). Other symbols are:

V = Primary System Volume (cold), ml

M = Max. Sp. Act. Attained in Reactor Coolant (cold), μ Ci/ml

L = Normal Letdown Rate (cold), ml/sec

B = Max. Leak Rate of Radionuclide from Core, Ci/sec

A breakdown of the terms is as follows:

Term 1: This represents the maximum inventory of the radionuclide under consideration found in the primary system. It takes into account continuous removal of radionuclides by the purification demineralizer.

Term 2: This represents the maximum amount of activity that can exist outside of the primary system and, therefore, that can accumulate on resins as a result of normal primary system letdown (45 gpm). Mathematically this term is the equilibrium, or steady state, portion of the expression. It describes the buildup of a radionuclide in an isolated system into which the radionuclide and where applicable, its parent nuclide, are being introduced at a constant rate.

The only activity estimates in which the presence of a parent nuclide is taken into account are those for La-140 and I-132. In the case of the latter, the situation is complicated by the fact that the concentration of the precursor (Te-132) in the coolant is not listed in Tables 11.1-4 and 11.1-5. This problem is resolved by assuming that the coolant concentration of Te-132 is the same as that given for I-132. This approach should be slightly conservative since it presupposes that there is no direct production of I-132 during the fission process.

Case 2: Ionic Radionuclides with Short Half-lives Which are Not Normally Removed by the Mixed Bed Purification Demineralizer and which may have Relatively Long Lived Ionic Parent Nuclides Present in Significant Amounts (Y-91, Mo-99, Cs-136, Cs-138)

The maximum total accumulation of activity on the resins due to each of these radionuclides is estimated using the following expression:

$$\text{Total Act.} = (B_i/\lambda_i) + (B_p/\lambda_p)$$

where all symbols are as previously defined.

This is the maximum buildup of the radionuclide external to the fuel due to leakage from the fuel.

Mathematically the above expression is similar to Term 2 of the equation in Case 1. The only difference is that the constant addition of daughter and parent nuclides into an isolated system is directly from the fuel rather than indirectly through primary system letdown.

Case 3: Ionic Radionuclides with Long Half-Lives or Those with Long Lived Ionic Parent Nuclides Present in Significant Amounts (Sr-90, Y-90, Cs-137, Ba-137m, Ce-144)

Estimates of the maximum accumulation of each of these radionuclides on the resins are determined using the following equation:

$$\begin{aligned} \text{Total Act.} = & \left[B_p / (\lambda_i - \lambda_p) - B_i / \lambda_i \right] e^{-\lambda_i T^*} + B_p / \lambda_p \\ & + B_i / \lambda_i - \left[B_p \lambda_i / \lambda_p (\lambda_i - \lambda_p) \right] e^{-\lambda_p T^*} \end{aligned}$$

where (T*) is the length of a nominal fuel cycle (292 EFPD) in seconds and all other notations have the same meanings as in the previous cases.

The right side of this equation is actually an estimate of the maximum amount of activity that can exist external to the fuel due to 292 days of leakage from the fuel. It is conservative in that the constant leak rate assumed is the maximum level attained in an entire fuel cycle. From Table 11.1-2 it is seen that the long lived radionuclides, due to their constantly increasing inventory in the fuel, would tend to leak at a constantly increasing rate. For one radionuclide, Cs-137, a more accurate estimate is used which takes into account this time dependent nature of the leakage (note: Ba-137m activity is assumed equal to that of Cs-137).

Case 4: Insoluble Radionuclides (Cr-51, Mn-54, Fe-55, Fe-59, Co-58, Co-60, Zr-95)

The activity present is equal to all the corrosion products contained in 346 days of normal, 45 gpm, letdown flow from the primary system.

b. Spent Resins:

Quantity: The quantity is based on an expected usage rate of one (1) condensate polishing demineralizer resin change per week for 52 weeks. Each change produces about 12 ft³ of spent resin.

Activity: Total radionuclide activities are based on the following assumptions:

1. All crud and dissolved activity in primary-to-secondary leakage is picked up by the resins.
2. The primary-to-secondary leakage is 100 gpd.
3. No credit is taken for decay after the leakage leaves the primary system.

c. Resin From Liquid Radwaste Treatment:

Quantity: The quantity is based on all the non-detergent liquid waste listed in Table 11.2-5 being treated by demineralization. All of the radioactivity is assumed to be absorbed by the resins, with a final resin concentration of 1 μ Ci/cc.

Activity: Total activity in this class of waste consists of all that is contained in the non-detergent miscellaneous liquid waste. Average total activities are found by multiplying the quantities estimated for each source, or group of related sources, times the average specific activities listed for them in Table 11.2-7 and then adding the results. Maximum ionic activity levels are simply 10 times the average values. Total crud inventories are the same in both cases.

d. Spent Filter Cartridges:

Quantity: The number of filter cartridges that must be disposed of each year is based on the following projected usage schedule:

Davis-Besse Unit 1 Updated Final Safety Analysis Report

Filter	Estimated Change Freq	Total Use Year
Purification demineralizer filter	1/30 days	10
Makeup filter	1/100 days	3
Primary Demineralizer filter	1/100 days	3
Clean waste monitor tank filter	1/100 days	3
Misc waste monitor tank filter	1/100 days	3
Spent fuel pool filter	1/100 days	3
Spent fuel pool skimmer filter	1/100 days	3
Refueling canal skimmer filter	1/30 days	1
	Total	29

Activity: Equal to all the corrosion products contained in 292 days of normal 45 gpm (@120 °F) letdown flow from the reactor primary system.

e. Miscellaneous Paper, Clothing, etc:

Quantity: The quantity is based on the production of 55 ft³ of loose waste per week plus an additional 2100 ft³ per year.

Activity: It does not appear possible to establish realistic, justifiable assumptions that can be used to estimate the activity levels in this type of waste. The reason for this is that the incidents which result in the contamination of this material are essentially random in their occurrence as well as in their severity. Therefore, for the purpose of these estimates, it is arbitrarily assumed that 200 ml of degassed reactor coolant are contained in each 10 ft³ of loose waste material. Based on a waste compaction ratio of 5:1, an average total coolant specific activity of about 4.867 µCi/ml (excluding tritium) and taking into account no decay, this amounts to about 0.017 µCi/ml of compacted waste. While extensive comparisons with similar waste from operating stations were not made due to a lack of information, it is noted that this value compares favorably with the 0.018 µCi/ml contained in at least one shipment of compacted waste from the Connecticut Yankee Station (ref. 40).

11.5.3 System Description

The waste solidification system originally installed by Protective Packaging, Inc. (PP) is no longer used to process and solidify radwaste. Likewise, the Chem-Nuclear Systems, Inc. (CNSI) solidification system subsequently used is also not being employed for solidification. The present practice is to eliminate the use of any solidification system in the interests of easier handling and lower occupational exposures. For similar reasons, evaporation of liquid radwaste is also no longer carried out. The following practices are being followed in handling the wastes described above:

- a. Spent Bead Resins are transferred to a high-integrity container. This container consists of cross-linked polyethylene. The polyethylene container is contained within a shipping cask that is provided with dewatering capability. The resins are dewatered to the point where residual water is less than 1% of the container volume. The 1% criteria applies only to resins shipped for burial. For resins shipped for processing the vendor criteria apply. When the container is full and dewatered it is closed. The shipping cask is also closed and shipped to a burial

site. Current regulations allow shipment of dewatered resins of high specific activity (greater than 1 $\mu\text{Ci/cc}$) provided a high-integrity container is used. This process eliminates solidification costs.

As required by definition, the polyethylene high-integrity containers:

- are capable of maintaining their contents until the radionuclides have decayed;
- withstand all pressures and stresses encountered during handling, lifting, loading, off-loading, backfilling and burial;
- are not susceptible to chemical, galvanic or other reactions from contents or the burial environment;
- will not deteriorate when subject to extreme variances in temperature, from the waste material itself, from processing waste inside the containers, or during storage, transportation and burial;
- will not degrade or have their characteristics diminished by radiation emitted from the sun during storage;
- have lids, caps fitting and closures of equivalent materials to meet all of the above requirements and can be completely sealed to prevent any loss of contents.

b. Powdered Resins are transferred to a metal container as long as they have specific activities less than 1 $\mu\text{Ci/cc}$. Otherwise these resins are shipped in a high integrity container, as described above. Resins shipped in metal containers are dewatered until residual water is less than 0.5% of the container volume.

c. Resin from Liquid Waste Treatment have specific activities less than 1 $\mu\text{Ci/cc}$. These resins are produced by the onsite Demineralization Unit. This unit processes the liquid radwaste at a rate of 25 gal/min. The liquid effluent from the system is recirculated for four hours to produce a homogeneous fluid, from which a reliable representative sample is taken. When it is verified that the liquid is within permissible limits for release to the environment, it is released to Lake Erie.

The spent resin from the demineralization unit are transferred to a High-Integrity Container (HIC) and handled as described in Section 11.5.3a.

d. Filter Cartridges are transferred to shielded containers for storage and placed in high integrity containers for shipments, burial or processing.

11.5.4 Packaging

11.5.4.1 Shielded Cask with Disposable Liner

Dewatered resins are packaged in a High Integrity Container (HIC) and are then placed into a shielded cask for disposal.

11.5.4.2 Shielded Container

Spent filter cartridges are stored in shielded containers. Examples of these are standard drums which have lead liners. The lining thickness is varied as required to keep radiation levels low in the storage area.

Portable shielding is used as needed to prevent the surface radiation levels from exceeding the limits prescribed by the Department of Transportation. In general, because filters are normally changed before large amounts of radioactivity have built up, extensive external shielding is not needed.

11.5.4.3 Unshielded Storage Containers

Solid wastes such as paper, rags, clothing, etc. may be collected. Since most waste of this type is usually only slightly contaminated, the storage containers do not contain excessive amounts of radioactivity when completely full. Once filled, the containers are stored in the Low Level Radwaste Storage Facility (LLRWSF) until ready for offsite shipment and disposal.

11.5.5 Storage Facilities

Solid waste storage handling occurs in the Low Level Radwaste Storage facility as shown in Figure 1.2-12.

The Low Level Radioactive Waste Storage Facility (LLRWSF) provides interim on-site storage for dry active waste (DAW) boxes and liners/High-Integrity Containers (HIC) and also provides DAW compaction and segregation areas. The following activities are also permitted, with administrative controls in place, in the LLRWSF: opening of DAW containers in the cell area for inventory, sorting and/or re-packaging, loading a Sea Land container with DAW and preparing the container for off-site shipment in the truck bay, opening of Radioactive Material (RAM) containers in the cell area for retrieval of tools and equipment, and refurbishing and/or minor repair of tools and equipment in the cell area. Approximately five years of storage area is available in the LLRWSF. The facility has separate radiation monitoring and floor drain collection systems.

A barricade enclosing the spent resin transfer pump room has been installed to prevent access to the room and to provide shielding from radiation. During resin transfer from the spent resin storage tank, the levels of radiation in this pump room may exceed 1000 mr/hr. Access is controlled.

11.5.6 Deleted

11.5.7 Shipment*

All radioactive solid radwastes are shipped from the site in Department of Transportation approved containers by Department of Transportation approved carriers. The containers are transported from the fill area or the storage area to the loading area. Personnel shields are provided for movement of the casks within the solid waste area. In the loading area, the casks are lifted by a crane onto a truck. Additional shielding, if required, is placed around the cask once it is secured on the truck, and it is then taken to an approved disposal site.

Once the casks, drums, or boxes, are loaded and secured on the trucks, radiation readings are taken to ensure that Department of Transportation limits are not exceeded. Surface

contamination readings are also taken to ensure no leakage of radioactive contents from the containers has occurred during handling.

Figure 1.2-4 shows the loading area for trucks. Trucks are not normally stored onsite for the shipment of solid radwastes. The trucks are supplied by the licensed carrier.

Empty liners are stored either in storage stalls or in an empty liner storage area. Drums are stored in an empty drum storage area.

To assure the safe transfer, packaging, and transport of low-level radioactive waste, the licensee conforms with the requirements of IE Bulletin No. 79-19 which details specifications for operating training and retraining programs, audit functions and current regulations available at the plant.

* Shipments of radioactive waste, for burial or volume reduction, are dependent upon operation and availability of licensed facilities.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11.5-1
Estimated Total Activity of Solid Waste (1)

Activity from Indicated Sources of Solid Waste, Ci

Radionuclide	Spent Bead-Type Resins		Spent Powdered Resins		Resins from Liquid Radwaste Treatment		Spent Filter Cartridges		Paper, Clothing, etc.	
	Maximum	Average	Maximum	Average	Maximum	Average	Maximum	Average	Maximum	Average
<u>Corrosion Products*</u>										
Cr-51	313.07	313.07	0.44	0.44	313.07	313.07	313.07	313.07	3.69×10^{-4}	3.69×10^{-4}
Mn-54	35.83	35.83	0.06	0.06	35.83	35.83	35.83	35.83	4.23×10^{-5}	4.23×10^{-5}
Fe-55	1237.97	1237.97	1.74	1.74	1237.97	1237.97	1237.97	1237.97	1.46×10^{-3}	1.46×10^{-3}
Fe-59	35.83	35.83	0.06	0.06	35.83	35.83	35.83	35.83	4.23×10^{-5}	4.23×10^{-5}
Co-58	1880.76	1880.76	2.65	2.65	1880.76	1880.76	1880.76	1880.76	2.22×10^{-3}	2.22×10^{-3}
Co-60	10.08	10.08	0.02	0.02	10.08	10.08	10.08	10.08	1.19×10^{-5}	1.19×10^{-5}
Zr-95	2464.03	2464.03	3.47	3.47	2464.03	2464.03	2464.03	2464.03	2.91×10^{-3}	2.91×10^{-3}
<u>Ionic Radionuclides</u>										
Rb-88	635.45	63.55	321.95	32.20	690.46	69.05	---	---	0.27	2.70×10^{-2}
Sr-89	87.57	8.76	0.54	0.06	1.16	0.12	---	---	4.51×10^{-4}	4.51×10^{-5}
Sr-90	19.66	1.97	0.02	1.74×10^{-3}	0.04	3.74×10^{-3}	---	---	1.46×10^{-5}	1.46×10^{-6}
Sr-91	10.66	1.07	3.40	0.34	7.29	0.73	---	---	2.85×10^{-3}	2.85×10^{-4}
Sr-92	2.37	0.24	1.04	0.11	2.24	0.23	---	---	8.72×10^{-4}	8.72×10^{-5}
Y-90	19.40	1.94	1.21	0.13	2.59	0.26	---	---	1.01×10^{-3}	1.01×10^{-4}
Y-91	12.96	1.30	6.75	0.68	14.50	1.45	---	---	5.66×10^{-3}	5.66×10^{-4}
Mo-99	979.17	97.92	491.28	49.13	1052.96	105.30	---	---	0.42	4.12×10^{-2}
I-131	9954.04	995.41	385.15	38.52	825.97	82.60	---	---	0.33	3.23×10^{-2}
I-132	3185.53	318.56	268.30	26.83	575.68	57.57	---	---	0.23	2.25×10^{-2}
I-133	2033.27	203.33	450.73	45.08	966.65	96.67	---	---	0.38	3.78×10^{-2}
I-134	111.71	11.18	54.62	5.47	117.38	11.74	---	---	4.58×10^{-2}	4.58×10^{-3}
I-135	625.51	62.56	226.56	22.66	485.92	48.60	---	---	0.19	1.90×10^{-2}
Cs-134	3925.06	392.51	487.70	48.77	1052.96	105.30	---	---	0.41	4.09×10^{-2}
Cs-136	200.80	20.08	88.24	8.83	189.88	18.99	---	---	7.40×10^{-2}	7.40×10^{-3}
Cs-137	8900.00	890.00	1514.36	151.44	3245.16	324.52	---	---	1.27	0.13
Cs-138	279.76	27.98	87.53	8.76	188.16	18.82	---	---	7.34×10^{-2}	7.34×10^{-3}
Ba-137m	8900.00	890.00	1395.12	139.52	2994.49	299.49	---	---	1.17	0.12
Ba-139	19.15	1.92	9.07	0.91	19.51	1.96	---	---	7.60×10^{-5}	7.60×10^{-6}
Ba-140	26.81	2.69	0.68	0.07	1.45	0.15	---	---	5.63×10^{-6}	5.63×10^{-7}
La-140	27.36	2.74	0.27	0.03	0.58	0.06	---	---	2.25×10^{-6}	2.25×10^{-7}
Ce-144	25.92	2.60	0.07	6.18×10^{-3}	0.14	0.02	---	---	5.18×10^{-7}	5.18×10^{-8}

* Due to the difficulty of determining a realistic distribution of crud in solid waste, the same activity is assigned to more than one source.

(1) The following information pertaining to quantities was provided as part of the initial safety analysis. Operating experience has demonstrated that the reactor coolant activity levels are much lower than the activity levels assumed in the radwaste system design and the following quantity information is considered historical.

TABLE 11.5-2

Quantities of Solid Wastes Per Year

<u>Source</u>	<u>Waste Input to Solid Radwaste System</u>	<u>Solid Waste Vol. Shipped from Station</u>
Spent Bead Resins	600 ft ³ gross displacement vol.*	600 ft ³
Powdered Resins	600 ft ³	600 ft ³
Resins from Radwaste Demineralizer	500 ft ³	500 ft ³
Filter Cartridges	29 cartridges	29 drums (215 ft ³)
Misc. Paper, Cloth, etc. (compactable)	4960 ft ³	992 ft ³

* The spent resin bead waste input volume of 600 ft³ is based on the assumption of 1% failed fuel. Actual resin waste volume approximates 200 ft³/yr.

11.6 OFFSITE RADIOLOGICAL MONITORING PROGRAM

Davis-Besse commits to the regulatory position of Regulatory Position 4.15 (Revision 1, February 1979).

11.6.1 Background Radiation

The preoperational radiological monitoring program commenced on August, 1972 and provided approximately five full years of background data prior to station operation.

Concentrations of background radioactivity for the Davis-Besse site are listed in Table 11.6-1. These concentrations are based on data from the 1981 annual radiological environmental monitoring program.

The major objective of the preoperational monitoring program was to accumulate data for two years on the background radiation at the Davis-Besse site. In an effort to make the program as comprehensive and up-to-date as possible, minor changes were made as necessary to incorporate proposed future monitoring requirements. The program complied with Regulatory Guide 4.8, "Environmental Technical Specification For Nuclear Power Plants."

The operational radiological monitoring program is similar to the pre-operational program.

11.6.2 Critical Pathways

The design bases for the critical pathways for exposure to humans due to the small quantity of radioactivity released from the station during operation are described in Appendix 11A together with the calculational models used in arriving at exposure estimates to the populace.

11.6.3 Sampling Media, Locations and Frequency

Ambient radiation measurements are made using thermoluminescent dosimeters (TLD). Dosimeter packets are located at each of twenty-seven locations. Each dosimeter consists of up to four individual exposure areas to provide a statistical analysis of the data. The dosimeter packets at each location are changed quarterly. There are five locations for determining airborne particulates and iodine which are changed weekly. Iodine is one of the most restrictive radionuclides which could be released to the environment. Weekly surface water samples are collected from two locations. These samples should provide the first indication of radioactive effluent leakage into the environment from the station discharge. Drinking water samples are taken from two public water supplies to detect any increases over background radioactivity concentrations. Ground water samples are collected quarterly from one location. Two species of fish are sampled annually at two locations because they are a food source and certain radionuclides concentrate in their flesh. Broad leaf vegetation samples are collected monthly, when available, since this is a direct pathway to man, should any radionuclides be present in the crops. Milk samples are collected semi-monthly during the grazing season (May-Oct.) and monthly at other times (Nov.-Apr.) in accordance with Table 11.6-2. Samples are collected from the milking animals closest to the station and from milking animals at least 15-30 km away. Milk is analyzed since radioactive iodine in air can be concentrated by the milk exposure pathway.

Lake shoreline sediments from one location with existing or potential recreational value are collected semi-annually to determine the concentration of radioactivity that might result from solids precipitating in the lake.

Table 11.6-2 lists the types of samples collected, the number of samples and sampling locations, a description of the sampling locations, the collection frequency, and the type of analyses to be performed on these samples.

The selection of sampling locations was based upon meteorological data, indicating the directions of the prevailing wind in the area, hydrological data, and the locations of the main intakes of water from Lake Erie which are used by the close-in and nearby populations. The selection of sampling locations was also based on the analysis of critical pathways in the environment. This site region is an agricultural area; thus, milk and food crops growing in the area must be sampled.

The sampling frequency for the various environmental media was chosen based upon the critical radionuclides, their half-lives, and their behavior in the biological and physical system. In the specific case of iodine, it is possible because of its short half-life in pasture (5 days) not to detect its presence in milk when sampled on a monthly schedule. However, I-131 is also sampled in the air on a weekly schedule at ten locations around the site. If there is a significant release of I-131 from this station, or any other station or from a weapons test in the atmosphere, I-131 will be detected.

The location of the nearest milking animals to the site is verified annually via the Annual Land Use Census.

The farms closest to the station site and those in the prevailing wind directions were included in the monitoring program. Since fishing activities take place in this area, two species of fish are sampled.

Consideration of the surrounding population was also incorporated in the selection of air sampling locations and the placement of thermoluminescent dosimeters. To provide a control during the operational program, one air sampler was placed in the least prevailing wind direction at a sufficient distance from the station to remove the influence of the station effluent.

The number of sample locations, and the type of samples and analyses, described above are the minimum required to be taken. Additional types of samples, sample locations, and analysis may be added to the radiological environmental monitoring program at the discretion of Environmental and Chemistry Management. The current enhancements to the radiological environmental monitoring program are outlined in Station procedures.

11.6.4 Analytical Sensitivity

Based upon the quantity of sample and the sensitivity of the counting procedure being used for the analysis, any significant increases over background will be detected. Table 11.6-3 gives the size of sample analyzed, type of analysis, detection method and minimum sensitivity and the various types of samples.

11.6.5 Data Analysis and Presentation

Results of the Radiological Environmental Monitoring Program are reported monthly and each monthly report includes all analytical data from samples taken during the current year. A radioactive release report covering the operation of the unit shall be submitted in accordance with 10CFR50.36. This radioactive release report includes a summary of the quantities of liquid and gaseous effluents and solid waste released as outlined in Regulatory Guide 1.21, June 1974 with data summarized on a quarterly basis following the format of Appendix B of the

Regulatory Guide. Additional information as required in the Davis-Besse Technical Specifications is also included. An annual report on the radiological environmental surveillance program for the Previous calendar year of operation is submitted to the Director of the NRC Regional Office prior to May 1 of each year. The period of the first report begins with the date of initial criticality. The report includes summaries, interpretations, and analysis of trends 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 unit operation on the Environment. In addition, the report provides the summarized and tabulated results of analysis of radiological environmental samples and of environmental radiation measurements taken during the period. The report also includes the results of the land use census required by the specification. If harmful effects or evidence of irreversible damage are detected by the monitoring, the licensee must provide an analysis of the problem and a proposed course of action to alleviate the problem.

Results of all radiological environmental samples taken are summarized and tabulated on an annual basis following a format similar to Table 1 of Regulatory Guide 4.8, December 1975. In the event that some results are not available the report is submitted by May 1 noting and explaining the reasons for the missing results. The missing data is submitted as soon as possible in a supplementary report.

11.6.6 Program Statistical Sensitivity

Sample size, procedure and instrumentation, counting geometry and counting time are such that dose can be estimated including any significant dose resulting from radioactive effluents released to the environment from station operation.

TABLE 11.6-1

Background Radioactivity and Radiation

<u>Type of Sample</u>	<u>Concentration of Background Radioactivity</u>
Airborne Particulate	0.1 pCi beta emitting radioactivity/m ³
Surface Water	4 x 10 ⁻⁹ µCi beta emitting radioactivity /ml
Drinking Water	3 x 10 ⁻⁹ µCi beta emitting radioactivity/ml
Ambient Radiation	8-22 mR/qtr. ^(a)
Fish	2 x 10 ⁻⁶ µCi beta emitting radioactivity/gram, wet

^(a) The ambient radiation measurement represents the ambient gamma dose rate, not a concentration of radioactivity.

TABLE 11.6-2

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations ^(a)</u>	<u>Collection Frequency</u>	<u>Type and Frequency of Analyses</u>
1. DIRECT RADIATION ^(c)	27 routine monitoring stations either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows: an inner ring of stations, generally one in each meteorological sector in the general area of the SITE BOUNDARY; an outer ring of stations, one in each meteorological sector in the 6- to 8-km range from the site, except the sectors over Lake Erie; the balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in 1 or 2 areas to serve as control stations	Quarterly	Gamma dose quarterly
2. AIRBORNE Radioiodine and Particulates	Samples from 5 locations 3 samples from close to the 3 SITE BOUNDARY locations, in different sectors, generally from areas of higher calculated annual average groundlevel D/Q. 1 sample from the vicinity of a nearby community, generally in the area of higher calculated annual average groundlevel D/Q. 1 sample from a control location, 15-30 km distant.	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading	<u>Radioiodine Canister:</u> I-131 analysis weekly. <u>Particulate Sampler:</u> Gross beta radioactivity analysis following filter change ^(b) ; Gamma isotopic analysis of composite (by location) quarterly.

TABLE 11.6-2 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

	<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations ^(a)</u>	<u>Collection Frequency</u>	<u>Type and Frequency of Analyses</u>
3.	WATERBORNE			
a.	Surface	2 samples	Weekly composite sample (Indicator location is a continuous, composite trickle-feed)	Composite for tritium and gamma isotopic analysis monthly.
b.	Ground	Sample from one source only if likely to be affected	Quarterly	Gamma isotopic and tritium analysis quarterly.
c.	Drinking	1 sample from the nearest source. 1 sample from a control location	Weekly composite sample.	Gross beta on monthly composite. Tritium and gamma isotopic analysis on quarterly composite. I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.
d.	Sediment from Shoreline	1 sample from area with existing or potential recreational value	Semiannually	Gamma isotopic analysis semiannually.
4.	INGESTION			
a.	Milk	If available, samples from milking animals up to 2 locations within 8 km distance having the highest dose potential. 1 sample from milking animals at a control location 15-30 km distant and generally in a less prevalent wind direction.	Semimonthly when animals are on pasture, monthly at other times	Gamma isotopic and I-131 analysis semi-monthly when animals are on pasture; monthly at other times.

TABLE 11.6-2 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

	<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations ^(a)</u>	<u>Collection Frequency</u>	<u>Type and Frequency of Analyses</u>
b.	Fish	1 sample each of 2 commercially and/or recreationally important species in vicinity of site. 1 sample of same species in areas not influenced by plant discharge.	1 sample in season.	Gamma isotopic analysis on edible portions.
c.	Food Products	Samples of up to 3 different kinds of broad leaf vegetation growth in two different offsite locations of higher predicted annual average ground-level D/Q if milk sampling is not performed.	Monthly when available.	Gamma isotopic and I-131 analysis.
		1 sample of each of the similar broad leaf vegetation grown 15-30 km distant in a less prevalent wind direction if milk sampling is not performed.	Monthly when available.	Gamma isotopic and I-131 analysis.

^(a) Specific parameters of distance and direction sector from the centerline of the reactor. Descriptions of specific sampling locations may be found in the Offsite Dose Calculation Manual.

^(b) Airborne particulate filters are analyzed for beta emitting radioactivity (gross beta analysis) 24 hours or more after sampling, to allow for radon and thoron daughter decay. If the concentration of beta-emitting radioactivity in air particulate samples is greater than ten times the yearly mean of control samples, gamma isotopic analysis is performed on the individual samples.

^(c) A thermoluminescent dosimeter (for purposes of this table) is considered to be one phosphor. Two or more phosphors in a packet are considered as two or more dosimeters.

TABLE 11.6-3

Environmental Monitoring Program Analytical Program

<u>Type of Sample</u>	<u>Amount of Sample</u>	<u>Type of Analysis</u>	<u>Measuring Equipment</u>	<u>LLD</u>
Airborne Particulate	Approximately 300 m ³ of air	Gross beta Gamma isotopic (Cs-134, Cs-137)	LBPC ^a Ge ^b	0.06pCi/m ³ 0.06pCi/m ³
Airborne Iodine	Approximately 300 m ³ of air	I-131	Ge	0.07pCi/m ³
Ambient Radiation	-----	Gamma dose	TLD Reader	1.0 mrem
Surface water, Untreated ^g	1 gallon	Gross beta (DS) Gross beta (SS) Tritium Gamma isotopic	LBPC LBPC LS Ge	4.0pCi/l 4.0pCi/l 2000pCi/l 15-30pCi/l
Surface water Treated ^g	1 gallon	Gross beta(TR) ^f Tritium Gamma isotopic	LBPC LS Ge	4.0pCi/l 2000pCi/l 15-30pCi/l
Shoreline Sediments	4 pounds	Gamma isotopic (CS-134, CS-137)	Ge	150pCi/kg dry
Groundwater	1 gallon	Tritium Gamma isotopic	LS ^e Ge	2000 pCi/l 10-30 pCi/l
Fish	1-2 pounds of flesh of each of two species	Gamma isotopic	Ge	130-260 pCi/Kg wet (I-131)
Broad Leaf Vegetation	2-5 pounds of each type of vegetation	Gamma isotopic I-131	Ge Ge	60pCi/Kg wet (I-131) 60pCi/Kg wet
Milk	1 gallon Goat 1 gallon Cow milk	I-131 Gamma Isotopic	LBPC Ge	1.0pCi/-1 15pCi/1

For any significant peaks observed from the Ge spectra, a quantitative determination will be made for the specific radionuclide.

^a LBPC = low background proportional counter

^b Ge = germanium detector

^c DS = dissolved solids

^d SS = suspended solids

^e Ls = liquid scintillation counter

^f Tr = total residue

^g Analysis for I-131 is performed on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.

TABLE 11.6-3 (Continued)

Environmental Monitoring Program Analytical Program

The Lower Limits of Detection (LLD) is defined by the following equation:

$$LLD = \frac{4.66 S_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \text{EXP}(-\lambda \Delta t)}$$

where

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

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

E is the counting efficiency (as counts per transformation).

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

2.22 is the number of transformations per minute per picocurie.

Y is the fractional radiochemical yield (when applicable).

λ is the radioactive decay constant for the particular radionuclide.

ΔT is the elapsed time between end of the sample collection period and time of counting.

Typical values of E, V, Y, and Δt should be used in the calculations.

11.7 REFERENCES

1. Frank, P.W., et al., Radiochemistry of Third PWR Fuel Material Test - X-1 Loop NRX Reactor, WAPD-TM-29, February 1957.
2. Eichenberg, J.D., et al., Effects of Irradiation of Bulk UO₂, WAPD-183, October 1957.
3. Allison, G.M. and Robertson, R.F.S., The Behavior of Fission Products in Pressurized-Water Systems. A Review of Defect Test on UO₂ Fuel Elements at Chalk River, AECL-1338, 1961.
4. Allison, G.M. and Roe, H.K., The Release of Fission Gases & Iodines from Defected UO₂ Fuel Elements of Different Lengths, AECL-2206, June 1965.
5. Fletcher, W.D. and Picone, L.F., Fission Products from Fuel Defect Test at Saxton, WCAP-3269-63, April 1966.
6. Ray, J.W., et al., Investigation of Tritium Generation and Release in PM Nuclear Power Plants, BMI-1787, October 31, 1966.
7. Goode, J.H., and Cox, C.M., The Distribution of Fission Product Tritium in a Zircaloy-Clad UO₂ Blanket Rad from PWR-1, ORNL-TM-2994, June 1970.
8. Barlett, John W., Stochastics of Coolant Crud, ANS Transactions, Volume 12, 1969.
9. Graves, R.H., "Chemistry and Waste Management at the Connecticut Yankee Atomic Power Plant", NACE Proceedings of the 25th Annual Conference, 1970.
10. Perry, R.H., et al., Chemical Engineer's Handbook, 4th Edition, McGraw-Hill, New York, 1963, Table 14-24.
11. Hartley, F.W., "Connecticut Yankee, Chemistry and Radiochemistry Effects on Plant Operations and Maintenance- Cores I, II and III, paper present at ANS Meeting on Reactor Operating Experience, Denver, Colorado, 1971.
12. Docket No. 50-298, Draft Environmental Statement to Nebraska Public Power District for Cooper Nuclear Station, Table III-4 (November 1972).
13. Fletcher, W.D., Ion Exchange in Boric Acid Solutions with Radioactive Decay, WCAP-3716, November 1962.
14. Goldman, M.I., "Radioactive Waste Management and Radiation Exposure", Nuclear Technology, 14, 157 (May 1972).
15. Decontamination Test Data for AMF Beaird Radioactive Waste Evaporator Installed at Rochester Gas & Electric Robert Ginna Nuclear Generating Station, AMF Bulletin RW-5, AMF Beaird, Inc. (October 1971).
16. Dickinson, B.N. and Higgins, I.R., "Selective Ion-Exchange Sorption of Cesium-137 and Iodine-131 from Borated Solutions", Nuclear Science and Engineering, 27, 131 (1967).

Davis-Besse Unit 1 Updated Final Safety Analysis Report

17. Hicks, J.H., Mravich, N.J., and Pocock, F.J., "Nuclear Steam Supply Water Chemistry Research", Proceedings of the American Power Conference, 33, 794 (1971).
18. Weisman, J. and Bartnoff, S., "The Saxon Chemical Shim Experiment", WCAP-3269-24 (July 1965).
19. Lindsay, W.T. Jr. and Abrams, C.S., "Ion-Exchange Removal of Fission Products from High Purity Water", Bettis Technical Review, WAPD-BT-16, pp 73 (December 1959).
20. Yokotsuka, S., Akatsu, E. and Ueno, K., "Flow Rate and Decontamination Factor of Cs-137 and I-131 through Mixed Resin Bed", J. Nuc. Sci. Tech., 8, 477 (September 1971).
21. Sharpelos, J.M., "Progress in Treatment of a Radioactive Condensate Waste", HW-79174 (October 1963).
22. DeLora y Scria, F. and Yee, W.C., "Decontamination of Radioactive Waste by Means of Ion Exchange", ORNL-tr-1771
23. Rhodes, D.W. and Lohse, G.E., "Treatment of Radioactive Waste Streams at the Idaho Chemical Processing Plant by Evaporation and Ion Exchange", CONF-710509-1 (1969).
24. Lindsay, W.T., Simon, G.P. and Abrams, C.S., "The Performance of Base-Form Ion Exchangers for pH Control and Removal of Radioisotopes from a Pressurized Water Reactor System", WAPD-TM-215 (July 1960).
25. Mercer, B.W., "Treatment of Radioactive Wastes by Ion Exchange", BNWL-SA-2369, paper presented at the 66th National AIChE meeting, Portland, Oregon, August 1969.
26. Ion Exchange Resins in Nuclear Power Technology, Duolite Data Bulletin No. 30, Diamond Shamrock Chemical Company, (April 1970).
27. Ryan, L.F. and Brown, R.M., "Powdered Ion Exchange in Nuclear Cycles", Proceedings of the American Power Conference, 30, 907 (1968).
28. Schultz, B.G., Comp., Chap. 12, "Corrosion Products in Recirculating Systems", Corrosion and Wear Handbook for Water-Cooled Reactors, DePaul, D.J., Ed., McGraw-Hill Book Company, Inc., New York (1957).
29. Parsont, M.A., and Goldman, M.I., "Effects of Estimated Radioactive Effluents from the Davis-Besse Nuclear Power Station for the Toledo Edison Company", NUS-729A, November, 1970.
30. Okubo, A. and Farlow, J.S., "Analysis of Great Lakes Drogue Studies", Proceedings of the 10th Conference on Great Lakes Research, 1967.
31. Docket No. 50-346, PSAR for Davis-Besse Nuclear Power Station, Appendix 2D, p. 2D-82.
32. Docket No. 50-346, Supplement to the Environmental Report for Davis-Besse Nuclear Power Station, Volume I, Appendix 2G.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

33. Radioactive Gas Evolved from Demineralizer, Nucl. Safety, 13 (6): 491 (November - December 1972).
34. Wilhelm, J.G., Dillmann, H., and Gerlach, K., "Testing of Iodine Filter Systems under Normal and Post-Accident Conditions", Proceedings of the Twelfth AEC Air Cleaning Conference, 1972.
35. Pence, D.T., Duce, F.A. and Maeck, W.J., "Developments in the Removal of Airborne Iodine Species with Metal-Substituted Zeolites", Proceedings of the Twelfth AEC Air Cleaning Conference, 1972.
36. Ibid
37. Parsly, L.F., "Design Considerations of Reactor Containment Spray Systems - Part IV, Calculation of Iodine - Water Partition Coefficients".
38. Eggleton, A.E.J., "A Theoretical Examination of Iodine - Water Partition Coefficients", AERE-R 4887, February, 1967.
39. Ibid.
40. Docket No. 50213-119, Connecticut Yankee Atomic Power Company Haddam Neck Plant Operation Report No. 71-10 for the Month of October, p. 7 (October 1971).
41. Chow, V.T., et al., Handbook of Applied Hydrology, McGraw-Hill, New York, 1964, p.11-4.

APPENDIX 11A

OFF-SITE RADIATION EXPOSURE DUE TO OPERATION
OF THE
DAVIS-BESSE NUCLEAR POWER STATION
(NUS-976)

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APPENDIX 11A

OFF-SITE RADIATION EXPOSURE DUE TO OPERATION OF THE
DAVIS-BESSE NUCLEAR POWER STATION

TABLE OF CONTENTS

<u>Section</u>	<u>Title</u>	<u>Page</u>
11A.1.0	<u>Introduction</u>	11A-1
11A.2.0	<u>Summary and Conclusions</u>	11A-2
11A.3.0	<u>Radiation Exposure from Gaseous Effluents</u>	11A-4
11A.3.1	<u>Estimated Gaseous Effluents</u>	11A-4
11A.3.2	<u>External Exposure from Gaseous Effluents</u>	11A-4
11A.3.3	<u>Internal Exposure from Gaseous Effluents</u>	11A-7
11A.3.4	<u>Computational Methods for Doses Resulting from Gaseous Effluents</u>	11A-12
11A.3.4.1	Whole Body and Body Surface Cloud Immersion Dose	11A-12
11A.3.4.2	External Exposure from Particulate Deposition	11A-13
11A.3.4.3	Internal Exposure Due to Inhalation	11A-14
11A.3.4.4	Exposure Due to Food Chain Transport	11A-16
11A.3.5	<u>Summary of Exposure from Gaseous Releases</u>	11A-16
11A.4.0	<u>Radiation Exposure from Liquid Effluents</u>	11A-27
11A.4.1	<u>Estimated Liquid Effluents and Concentrations</u>	11A-27
11A.4.2	<u>Maximum Individual Radiation Exposure from Liquid Effluents</u>	11A-28
11A.4.3	<u>Population Radiation Exposure from Liquid Effluents</u>	11A-34
11A.4.4	<u>Radiation Exposure of Aquatic Bio-Systems</u>	11A-35
11A.4.5	<u>Computational Methods for Doses Resulting from Liquid Effluents</u>	11A-37
11A.4.5.1	Whole Body and Body Organ Exposure from Water Ingestion	11A-37
11A.4.5.2	Whole Body and Body Organ Exposure from Fish Ingestion	11A-38
11A.4.5.3	Models for Calculation of Dose to Aquatic Biota	11A-39
11A.5.0	<u>Dose Totals and Comparison with Federal Regulations and Natural Background</u>	11A-48
11A.6.0	References	11A-51

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
11A.3-1	Estimated Airborne Effluents	11A-17
11A.3-2	External Exposure Due to Ground Deposition of Particulates and Halogens	11A-19
11A.3-3	Estimated Population Distributions Within 50 Miles of the Station	11A-20
11A.3-4	External Population Exposure Due to Gaseous Releases	11A-23
11A.3-5	Applicable Annual Average X/Q Values for the Davis-Besse Site	11A-24
11A.3-6	Annual Average Wind Direction Frequencies at the Davis-Besse Site	11A-25
11A.3-7	Summary of Maximum Individual Exposure Rates Due to Gaseous Releases	11A-26
11A.4-1	Estimated Liquid Effluents and Concentrations	11A-40
11A.4-2	Internal Body Organ Exposure From Ingestion of Water and Fish	11A-41
11A.4-3	Concentration Factors for Effluent Radionuclides in Fish	11A-42
11A.4-4	Summary of Maximum Individual Exposure Rates Due to Liquid Releases	11A-43
11A.4-5	Data Used to Estimate Population Water Ingestion Exposure	11A-44
11A.4-6	Radiation Exposure of Aquatic Biota	11A-45
11A.4-7	ICRP Values for Organ Mass and Effective Radius	11A-46
11A.4-8	Estimate of Body Water Distribution by Organ in Standard Man	11A-47
11A.5-1	Maximum Individual Dose Totals and Comparison with Federal Regulations and Natural Background Exposure	11A-49
11A.5-2	Whole Body Population Dose Totals and Comparison with Federal Regulations and Natural Background Exposure	11A-50

LIST OF FIGURES

<u>Figure</u>	<u>Title</u>
11A.1-1	Exposure Pathways

APPENDIX 11A

OFF-SITE RADIATION EXPOSURE DUE TO OPERATION OF THE
DAVIS-BESSE NUCLEAR POWER STATION

11A.1.0 Introduction

The Davis-Besse Nuclear Power Station occupies a site area of approximately 954 acres situated on the southwest shore of Lake Erie (ref. 1). The site itself is mostly marshland and is partially used as a natural wildlife refuge. The nearest large population center is Toledo, Ohio, located about 20 miles west of the plant. Smaller population centers near the site are Fremont, to the south, and Sandusky, to the southeast. Large areas of land around the site are only sparsely populated and support a mostly agricultural economy. Recreational activities such as sport fishing and duck hunting are popular in the region of the site.

This appendix evaluates the radiological impact that the Davis-Besse Station will have upon the population and wildlife living in the region of the site. Maximum potential individual radiation exposure rates are evaluated as well as the total integrated population exposure within a fifty mile radius of the reactor. Exposure pathways considered include all significant environmental and biological mechanisms by which activity released from the Station could conceivably reach the public. Figure 11A.1-1, Exposure Pathways, presents a visual illustration of these exposure pathways of greatest significance for central station nuclear generating plants such as Davis-Besse.

Annual activity releases by isotope are presented in Chapter 11 of the Davis-Besse USAR. Included in Chapter 11 is a full explanation of the derivation of these source terms. Of considerable importance is the fact that two sets of source terms have been established based on different values of the percentage of defective fuel cladding assumed to occur. The lower value, 0.1%, is considered the likely average value over the life of the Station. This value was used to calculate 'expected' activity releases. The higher value, 1.0%, is considered the limiting value of defective fuel cladding beyond which continued Station operation would be unlikely. The 1.0% failed fuel criterion has been used to calculate "design basis" activity releases. The following convention is utilized in the following sections:

Numerical data (concentrations, doses, etc.) for which the defective fuel cladding fraction makes a significant difference are presented in the form of 0.1% (1.0%), where the number appropriate to design basis releases are presented in parenthesis immediately following the number appropriate to expected releases.

Since the applicable regulations are based on a consideration of average annual dose, this study is directed at determining the doses delivered in a typical year of operation. The estimated doses to individuals and the population are compared to existing and proposed dose guidelines, to normal background radiation doses, and to other ordinarily acceptable radiation exposure levels.

Calculational methods employed are, in general, those suggested or recommended by such bodies as the IAEA, the USAEC or the ICRP where such methods are available. In situations where the lack of knowledge has precluded accuracy, conservative assumptions have been made in an attempt to eliminate the possibility of underestimating the particular exposure level.

11A.2.0 Summary and Conclusions

The maximum expected radiation exposure to the general public or to individuals thereof resulting from the operation of the Davis-Besse Nuclear Power Station was evaluated. The quantities of the various radionuclides released from the station in gaseous particulate, or liquid form were obtained from data prepared for the Davis-Besse FSAR. Expected release rates were based on 0.1% defective fuel cladding (ref. 2). In addition, conservative environmental dilution and concentration factors have been used so that the annual doses calculated in this study are the maximum values which could reasonably be expected to occur due to the normal operation of the Davis-Besse Nuclear Power Station.

All calculated human exposure levels resulting from the operation of Davis-Besse have been found to amount to only small fractions of normally accepted levels of radiation exposure. The applicable limits set by law in 10CFR20 (ref. 3) for maximum permissible exposure have not been exceeded or even closely approached.

a. Maximum Individual Exposure:

For gaseous releases, exposure rates have been calculated for the following exposure pathways: external exposure from cloud immersion and particulate deposition; internal exposure from inhalation of tritium, iodine and particulates; and internal exposure from food chain transport of iodines and particulates. For liquid releases the exposure pathways examined include: water ingestion, fish ingestion, swimming and sunbathing. The maximum potential whole body exposure to an adult individual from all these sources has been calculated to be 5.38 mrem per year, based on the maximum expected release rates and milk ingestion at the nearest known grazing location. For design basis release rates the exposure total from all these sources is 35.5 mrem per year. The critical organ (liver) exposure total from all sources has been estimated to be 7.49 mrem per year for expected releases and 56.2 mrem per year for design basis releases.

In assessing the significance of the above exposure totals it should be noted that they represent the sum of the maximum exposure rates for several different exposure pathways, each of which has been evaluated with a high degree of conservatism. It is highly improbable that any individual will ever receive the maximum potential exposure from any of the pathways examined. The probability that any individual will receive them all simultaneously is entirely negligible. Thus, the totals presented represent only upper bounds to the potential exposure, and should not be construed to represent the likely exposure rates.

These maximum potential exposure rates are small in comparison with ordinary and acceptable individual radiation exposure levels. For instance, the annual whole body exposure from naturally occurring background radiation in the United States averages about 130 mrem per year (ref. 4). The exposure to an individual from a single chest X-ray may be as much as 170 mrem (ref. 4). Wearing a particular kind of watch may yield an incremental annual whole body dose of 4 mrem (ref. 4). The inhabitation of a stone or masonry dwelling rather than a frame house has been calculated to yield an average additional annual whole body dose of 40 mrem and a maximum annual incremental exposure of over 500 mrem (ref. 5). Thus the maximum potential incremental exposure to man from plant contributed radioactivity represents only a small fraction of exposure increments common to ordinary experience.

b. Population Exposure:

The largest source of population exposure from the expected plant effluents is due to the external radiation received from the released radioactive gases. The annual exposure to the surrounding population from this pathway, combined with that due to the particulate releases is calculated to total 2.52 (25.2) man-rem in the year 2010. This total is about twenty (fifty-five) times the population exposure calculated to result in that year from liquid releases, including exposure from water and fish ingestion.

Total exposure from both gaseous and liquid release is calculated to range from 1.52 (14.8) man-rem per year in 1970 to 2.65 (25.7) man-rem per year in 2010.

c. Exposure of Aquatic and Other Organisms:

The maximum potential radiation exposure of aquatic biota has been evaluated on an extremely conservative basis. The calculated exposure levels, presented in Table 11A.4-1, are based on the continuous presence in Lake Erie of the annual average discharge concentrations of all radionuclides. Also, it has been assumed that radioactivity concentrations in the sediment may be as much as a thousand times those in the ambient water. These assumptions maximize the potential exposure levels to aquatic biota and minimize the probability of occurrence. The maximum resulting exposure rate is about 261 (712) mrad per year to bottom dwelling invertebrates.

The dose to bottom feeding fish is calculated to be about 135 (451) mrad per year, using the same conservative basis. Most of this exposure is due to the high levels of radioactivity assumed to exist in the bottom sediment. For a fish which does not normally dwell at the lake bottom, the annual exposure from all internal and external sources is calculated to be 34 (162) mrad.

If credit is taken for dilution of the discharge in the lake, the dose estimates are substantially reduced. These reduced exposure levels are much more realistic in terms of expectancy than those based on widely pervasive undiluted discharge concentrations.

Regarding terrestrial animals, they will be subject to radiation exposure via the same pathways that have been extensively evaluated for man. There will be essentially no difference in the calculated external exposure rates, and it is not expected that internal exposure will be significantly different.

This analysis has resulted in no estimated exposure level which exceeds the guideline in 10CFR20, (ref. 3) for the maximum permissible radiation exposure of man (500 mrem/yr.). In view of this and the extreme conservatism of the models and assumptions employed in the derivation of the doses, it is concluded that no significant radiological hazard to the aquatic population of Lake Erie or to other wildlife, will occur due to operation of the Davis-Besse Nuclear Power Station.

d. Conclusions:

In summary it can be stated that all known environmental and biological phenomena which could possibly lead to significant radiation exposure of man have been investigated in detail. The calculation models, assumed biological parameters, and environmental concentration and dilution factors have all been conservative in nature. Therefore, the calculated exposure rates to which man might potentially be subject are the highest which could reasonably be expected

to occur due to the Davis-Besse Nuclear Power Station. No current Federal regulation regarding radiation exposure of members of the public will be exceeded or even approached.

11A.3.0 Radiation Exposure from Gaseous Effluents

There are a number of pathways through which persons off-site may be exposed to the gaseous radioactivity released from nuclear power plants. From Figure 11A.3-1, three general pathways may be identified; direct radiation exposure, inhalation exposure, and exposure through food chains. The relative importance of these exposure pathways is determined by the radionuclide composition and quantities of the released radioactive gases as well as the site environment.

11A.3.1 Estimated Gaseous Effluents

The expected and design basis annual gaseous releases estimated for the Davis-Besse Station are presented in Table 11A.3-1, 'Estimated Gaseous Effluents'. The values presented have been provided by Bechtel Corporation (ref. 2). These values have been produced for presentation and use in Chapter 11 of the Davis-Besse FSAR. As explained in Chapter 11, the greatest single source of gaseous activity is due to continuous gas stripping of the reactor coolant let-down flow. This single source accounts for essentially all of the released noble gases, although significant amounts of radioiodines and particulates have been estimated to occur through the release of aerated gases from various equipment vents, steam leakage and via escape of holding tank cover gases.

All gaseous releases are assumed to occur at ground level and to be continuous rather than intermittent, although intermittent releases under more favorable meteorological conditions could further reduce the calculated exposures.

11A.3.2 External Exposure from Gaseous Effluents

From Table 11A.3-1 it can be seen that the gaseous activity releases are expected to be comprised of noble gases, iodines and other particulate matter, and tritium. Two modes of external exposure have been considered for these releases. For tritium and the noble gases the critical mode of external exposure is from direct radiation due to the ambient air concentration. For particulates and iodines the critical external exposure is from the buildup of these materials on or in the ground due to the phenomenon of ground deposition.

Neither the noble gases nor tritium will deposit onto the ground under normal conditions. This exposure pathway is therefore treated only for iodines and particulates. The calculation of external exposure due to the radioactive cloud has, however, included all released radioactive materials.

1. Maximum Individual External Exposure:

The site boundary external dose rate depends in general on the source terms, the applicable atmospheric diffusion, and the receptor characteristics. The distance to the nearest site boundary from the release point near the reactor building is a minimum of 2400 feet at Davis-Besse (ref. 9). For the purpose of estimating the maximum potential annual exposure a hypothetical "maximum individual" will be assumed to reside at the site boundary continuously over the full period of one year. The individual will be assumed to be unshielded by housing which would normally provide a dose reduction factor of 2 or more.

All forms of external exposure will depend directly upon the annual average site boundary air concentration, which in turn depend on the atmospheric dispersion parameter, X/Q (sec/m^3). The applicable value of X/Q at the Davis-Besse site boundary, for annual average meteorology is taken to be $4.17 \times 10^{-6} \text{ sec}/\text{m}^3$ (ref. 7).

This value is based on the assumption of winds into the direction of interest with an annual average frequency of 15%. According to the annual average wind rose in Figure 2.3-4 of the USAR, the highest frequency of on-shore winds occurs for winds out of the east (towards the west) and amounts to only about 6.3%. Thus, the site boundary X/Q given in the USAR appears to be high by a factor of two or more.

a. External exposure from the radioactive cloud

When immersed in a cloud of radioactive material, a receptor individual will be subject to penetrating gamma radiation as well as short range beta radiation. Irradiation by gamma rays will generally yield a relatively homogeneous exposure of the whole body. The beta rays of concern here will yield exposure only of the outermost tissues of the human body (ref. 8) this is due to their short range in human flesh, which is on the order of one centimeter or less. Human tissues at risk from beta radiation included the skin, the lens of the eye, and, for the more energetic betas, the gonads.

In order to estimate the potential exposure rate the “infinite sphere” model has been used, as recommended by the International Commission of Radiological Protection (ICRP) and other authoritative bodies (ref. 9, 10). The basic assumption of the model is that the absorbed dose rate at any point inside an infinite homogeneous sphere of uniform radioactivity concentration is equal to the energy release per unit mass. The concentrations are taken to be equal to the annual average at the location of interest. The dose rate to a ground level receptor from a surrounding cloud of radioactivity is taken to be one half that given by the infinite sphere approach. This is because, for gamma radiation, the receptor is only irradiated from one half the total available solid angle; for beta radiation the factor of 0.5 accounts for the self shielding of the human body. The dose rate is also adjusted upward by a factor of 1.13 to account for the increased stopping power of human tissue relative to air (ref. 9). Use of this model has been demonstrated to lead to conservative results (ref. 11).

Assuming that only gamma radiation will contribute to whole body exposure, the resulting site boundary whole body exposure rate due to cloud immersion has been calculated to be 0.593 (5.93) mrem per year. This exposure rate is based upon the site boundary X/Q value of $4.17 \times 10^{-6} \text{ sec}/\text{m}^3$, the release rates presented in Table 11A.3-1, and the use of the infinite sphere model. The sources of disintegration energies and formulas used in the calculation are presented in Subsection 11A.3.4. The exposure to the skin, lens of the eye, and gonads, based on both beta and gamma exposure, is calculated to be 1.30 (12.7) mrem per year. Even weak tritium betas have been assumed to contribute to this body surface exposure. It may be noted that of the total exposure, only a very small fraction is due to materials other than the noble gases.

b. External exposure from ground deposition of particulates and halogens

The natural deposition of particulates and iodines onto grasses and soils is a familiar and important phenomenon. The rate of deposition can be obtained from the annual average air concentration by application of a quantity known as the deposition velocity. The continuous deposition onto ground of a radioactive material will cause a monotonically increasing ground concentration which will eventually approach a limiting, or equilibrium value. This equilibrium

ground concentration will be reached when the loss due to radioactive decay just balances the rate of deposition. For the very long lived radionuclides this condition will not be reached within the expected plant lifetime of 40 years. Thus ground concentrations at the end of 40 years have been calculated at the site boundary for each of the particulate and iodine isotopes in Table 11A.3-1. Proper dose conversion parameters have then been applied to estimate the beta and gamma exposure rates at the critical height of one meter above ground level (ref. 12, 13). The maximum ground concentrations and resulting exposure rates resulting from these calculations are presented in Table 11A.3-2. Beta exposure has been reduced by a factor of two to account for the limited range of beta radiation in the human body.

As before, only the gamma component of the total exposure rate is assumed to yield whole body radiation. It can be seen from Table 11A.3-2 that only three radionuclides contribute significantly to the whole body exposure rate. These radionuclides and their individual whole body exposure yields are: Cs-134, 0.118 (1.18) mrem per year; Cs-137, 0.455 (4.55) mrem per year; and I-131, 0.011 (0.11) mrem per year. The total whole body exposure rate is calculated to be 0.588 (5.88) mrem per year.

The exposure rate to the body surface will be increased by the beta component and is therefore higher than the whole body exposure rate. As before, the radionuclides contributing significant beta exposure are: Cs-134, 0.067 (0.67) mrem per year; Cs-137, 0.520 (5.20) mrem per year; I-131, 0.0094 (0.094) mrem per year. The total exposure rate at one meter due to all beta radiation is 0.604 (6.04) mrem per year. The total body surface exposure from beta and gamma sources is calculated to be 1.19 (11.9) mrem per year.

In analyzing the exposure resulting from this pathway, it has been assumed that once deposited on the ground, all isotopes except Cs-137 would remain at the surface. Due to its very long half-life and the natural phenomena of percolation cesium fallout on soil eventually establishes a concentration ($\mu\text{Ci}/\text{cm}^3$) in soil which is depth dependent. Analysis of Cs-137 concentration in soil due to fallout has established that the typical distribution is such that the ground concentration decreases exponentially as the depth increases (ref. 14, 15). An investigation by Beck (ref. 15) has revealed that this distribution is attained in a relatively short interval (1 to 2 years) and can be described mathematically by a relaxation length which will vary from place to place depending upon local conditions of soil and precipitation. A normal range is from 3 to 8 centimeters (ref. 15). As shorter relaxation lengths imply that the average depth of penetration is less, thus reducing the available shielding, a value of 3 centimeters has been used in this analysis. The details of the calculation and the equations used are presented in Subsection 11A.3.4.

A most important parameter for this exposure pathway is the average deposition velocity. The rate of deposition, and thus the exposure, depends directly on this quantity. Particulate matter has been assumed to deposit at a rate of 2×10^3 meters per second (ref. 16, 17) in this analysis, while halogens have been assumed to deposit at 1×10^{-2} meters per second (ref. 17, 18, 19).

1. External Population Exposure:

The total exposure to the population within 50 miles of the Davis-Besse site has been analyzed for both of the important external exposure pathways discussed above. This exposure necessarily depends upon the population distribution estimated to exist around the station location.

The total population of permanent residents living within 50 miles of the station in 1970 was estimated to have been 2,066,703 persons, distributed as shown in Figures 2.1-5 and 2.1-6 of

the Davis-Besse USAR. The distribution gives the number of people residing in each of 160 different population segments, where a segment is defined as that area within a 22.5° sector (centered on one of the 16 compass points) and within one of the 10 annuli into which the area within a 50 mile radius is subdivided. Within 10 miles of the site the “summer only” population constituted a significant perturbation to the calculation. These part time residents were accounted for by assuming that 100 people present during only the summer would receive the same dose as 25 permanent residents. For each segment with a non-negligible summer-only population, the permanent resident population was increased by one for every four summer-only residents. The summer-only population distribution about the site, estimated for 1970, 1990 and 2010, was obtained from material for Subsection 2.1.3 of the Davis-Besse USAR. The resulting population distributions upon which the population doses are based are presented in Table 11A.3-3.

The total exposure occurring within any one segment is the product of the average dose rate for the segment, and the segment population. The total population exposure within 50 miles of the site is then the sum of the population exposures within each of the 160 population segments. Following this general procedure, the NUS computer code GASDOS performed the necessary calculations. Results were obtained not only for estimated 1970 population distribution, but for the projected population distributions for the years 1990 and 2010 as well.

The results of the calculations are presented in Table 11A.3-4. As for maximum individual exposure, the doses resulting from cloud immersion are slightly higher than those due to particulate deposition. No allowance has been made for plume depletion as the particulate matter is transported away from the site. At the large distances involved, plume depletion during transport will be effective in reducing the dose rates although no credit for this phenomena is taken here. It is interesting that although the total population exposure is shown to increase in time, the per capita exposure decreases. This is due to an apparent shift outward of the average distance of the population from the plant.

The segment average dose rate was taken to be the dose rate of the geometric midpoint of the segment. Annual average X/Q values presented in Subsection 2.3.5.2 of the Davis-Besse USAR were used as the basis. These values were logarithmically interpolated to arrive at X/Q values at the needed distances, which are presented in Table 11A.3-5. Since the X/Q values from Subsection 2.3.5.2 of the USAR were appropriate only for a sector into which winds blew with an annual average frequency of 15%, further corrections were made to account for the actual annual average wind direction frequency distribution as depicted in USAR Figure 2.3-4. The abstracted wind direction frequencies, used in the calculations, are given in Table 11A.3-6.

11A.3.3 Internal Exposure from Gaseous Effluents

The radioactive content of the gases released at Davis-Besse must be either inhaled, ingested, or absorbed through the skin in order to yield internal radiation exposure. Skin absorption is significant only for the released tritium which in the form of HTO can pass through the skin with relative ease (ref. 20). Exposure through ingestion requires the physical transport of the radioactive materials through some form of food chain. In this regard only the released iodines and particulates are of concern.

Since the noble gases do not react chemically with other substances under normal conditions, there is no physical basis for their transport through food chains or reconcentration within the human body for these gases.

In terms of continued inhalation and absorption in the body, both krypton and xenon may develop in physical solution, chiefly in the body water and fat (ref. 21). Several human exposure experiments revealed that inhalation of relatively large amounts of radioactive noble gases resulted in very low tissue exposures (ref. 22, 23). In general, it may be estimated that the internal dose from radioactive noble gases dissolved in body tissue following inhalation from a cloud is negligible, i.e., less than 1% of the associated external whole body dose (ref. 24). The resultant doses from exposure to noble gases, therefore, are considered to be external whole body doses only.

Potential exposure due to inhalation is of relative importance only for the released tritium and iodine. However, it has been evaluated for other particulate discharges as well.

1. Exposure from Inhalation:

Exposure due to inhalation of tritium, iodine, and other particulates will depend directly on the estimated daily inhalation intake. This quantity can be obtained by multiplying the annual average air concentration times the volumetric breathing rate. The necessary equation is as below:

$$A(\mu\text{Ci/day}) = Q(\text{Ci/yr}) \times X/Q(\text{sec/m}^3) \times \text{BR}(\text{m}^3/\text{sec}) \times \frac{1\text{yr}}{365.25\text{days}} \times 10^6 (\mu\text{Ci/Ci})$$

The quantity BR is the breathing rate which for the standard man is equal to $2.32 \times 10^{-4} \text{ m}^3/\text{sec}$ (ref. 9).

a. Iodine inhalation

The formula above has been used to estimate the rate of iodine activity intake at the site boundary by a maximum individual. Due to the preferential concentration of iodine in the thyroid gland, the resulting whole body dose is negligible in comparison to the resulting thyroid exposure, which is calculated to be 0.220 (2.20) mrem per year for an adult. Due to the small size of the thyroid gland in a child (the thyroid exposure is inversely proportional to the mass of the thyroid), the thyroid exposure via this pathway is increased for a small child, amounting to 0.382 (3.82) mrem per year. The calculational model and equations used to arrive at these exposure estimates, are described in Subsection 11A.3.4. The dose conversion parameters used to estimate adult thyroid exposure are those contained in the AEC publication TID 14844, "Calculation of Distance Factors for Power and Test Reactor Sites" (ref. 25). These parameters are presented below and illustrate the relative importance of I-131 for this exposure pathway:

<u>Iodine Isotopes</u>	<u>Rem per Curie Inhaled (ref. 25)</u>
I-131	1.48×10^6
I-132	5.35×10^4
I-133	4.00×10^5
I-134	2.50×10^4
I-135	1.24×10^5

The greater dose-effectiveness of I-131 is due primarily to its relatively long 8.05 day half-life in comparison to the other isotopes.

b. Tritium inhalation

The amount of tritium inhaled daily by a standard man at the site boundary has been determined to be 1.03×10^{-3} microcuries per day. However, due to the phenomenon of skin absorption this intake does not represent the total source. As estimated by the ICRP (ref. 9) the total tritium intake from inhalation and skin absorption can be conservatively determined to be equal to twice that due to inhalation alone. As there is essentially no difference in the exposure from this mode of tritium uptake by the body and that due to tritium intake via water ingestion, the resulting exposure has been computed using methods and equations outlined in Subsection 11A.4.5. The total daily tritium intake has been taken to be 2.07×10^{-3} microcuries per day.

The resulting exposure has been determined to be 0.096 mrem per year to the whole body.¹ The maximum organ dose has been determined to be 0.131 mrem per year to the testes. Doses to all other organs except the bone are between the exposure rates for the testes and whole body. The bone dose has been computed to be 0.058 mrem per year.

c. Particulate inhalation

Later in this section a discussion of the potential particulate intake by man due to food chain transport is presented. It is determined there that the potential adult intake rate through the milk route for Cs-137 is about 1.1×10^{-6} microcuries per day. As Cs-137 is by far the most critical radionuclide for particulate inhalation and particulate food chain transport (with the exception of the iodine isotopes), the relative importance of the inhalation pathway can be determined by examining the potential Cs-137 intake via inhalation. This amounts to 1.47×10^{-7} microcuries per day. Thus, the Cs-137 potential inhalation intake amounts to only about 13% of the potential intake through food chain transport. Thus, inhalation intake is relatively insignificant.

The resulting whole body and critical organ exposure due to particulate inhalation has been computed by determining the daily intake of each isotope and applying the methods and equations used to compute liquid ingestion exposure as given in Subsection 11A.4.5. The calculated dose estimate is 0.0028 (0.028) mrem per year to the whole body. The critical organ has been determined to be the liver, for which the calculated exposure is 0.0064 (0.064) mrem per year.

2. Exposure from Food Chain Transport:

The radioactive particulate and iodine emissions from Davis-Besse are expected to be involved in food chain transport to man after ground deposition. The potential uptake in food crops of several of the released radionuclides is a well documented phenomena (ref. 26). However, the critical exposure pathway to man is via the milk route, (ref. 16, 18) especially for the radionuclides of greatest concern here, Cs-134, Cs-137 and I-131. As can be seen in Table 11A.3-2 these three isotopes constitute very nearly all of the ground activity resulting from particulate deposition. Cesium 137 accounts for 95% of this activity by itself. Material deposited on the ground may be present in the grasses ingested by grazing cows. This will be caused by direct foliar retention and uptake from the soil by roots. After ingestion by the cow these radioactive materials will be transferred to the cow's milk which may subsequently be ingested by man. The potential intake by man will depend on the assumed milk ingestion rate and the activity concentration of the milk produced. In estimating the potential activity levels in

¹The exposure due to tritium inhalation is identical for expected and design basis release rates, for the whole body and all body organs.

milk, the method of Bryant (ref. 16) has been used for Sr-90, Cs-137. The potential I-131 concentration has been obtained using a slightly different model.

Due to its great dose effectiveness Sr-90 has been included in the calculation. The potential Sr-90 activity level has been estimated using the following equation².

$$C(pCi/l) = P_r F_r + P_d F_d$$

where:

C = the resulting milk concentration, pCi/l

F_r = the rate of deposition, mCi / km² – yr

F_d = the cumulative deposit, mCi / km²

P_r = the deposition rate factor (pCi / l / (mCi / km² – yr)

P_d = the long term soil factor (pCi/l) / (mCi / km²)

The above formulation has arisen from an analysis of data from surveys of milk contamination due to Sr-90 fallout on pasture. The parameters P_r and P_d are proportionality factors and are designed to account for grass contamination by direct foliar retention and plant base absorption (P_d). Values of these proportionality factors have been estimated by the method of least squares to be 0.8 for P_r and 0.2 for P_d (ref. 16). Application of these values and the above model leads to an estimated potential Sr-90 concentration in milk of 2.76 x 10⁻¹¹ (2.76 x 10⁻¹⁰) microcuries per liter, at the site boundary.

A similar formula, also due to Bryant (ref. 16), has been used to estimate the potential Cs-134 and Cs-137 concentrations. The equation is:

$$C(pCi/l) = P_r F_r + P_d F_d$$

Where:

C = the resulting milk concentration, pCi/l

F_r = the rate of deposition, mCi / km² – yr

F_d = the deposit in the previous two years, mCi / km²

P_r = the deposition rate factor, (pCi / l / (mCi / km² – yr)

P_d = the soil factor, (pCi/l) / (mCi / km²)

The variation in the meaning of the parameter F_d is due to the ability of cesium to percolate downward in soil, as previously discussed in relation to external exposure from particulate deposition. The similarly obtained values of P_r and P_d are 3.6 and 0.65, respectively. It should be noted that both of Bryant's formulas used here are strictly empirical in nature; i.e., they are designed for the exclusive purpose of describing the best available scientific data. The estimated maximum potential concentrations in milk derived using the above formulation are 7.35 x 10⁻⁷ (7.35 x 10⁻⁶) microcuries per liter for Cs-134 and 2.27 x 10⁻⁶ (2.27 x 10⁻⁵) microcuries per liter for Cs-137, both at the site boundary. Concentrations in milk have been estimated for cesium and strontium using an assumed deposition velocity of 2 x 10⁻³ meters per second (ref. 16, 17).

The potential I-131 milk concentration is based on a slightly different model, using an assumed deposition velocity of 1 x 10⁻² meters per second (ref. 17, 18). The formula necessary to compute this concentration is presented below:

$$C(\mu Ci/l) = Q(Ci/yr) \times Q \frac{\text{sec}}{\text{m}^3} V_g \frac{\text{m}}{\text{sec}} \frac{T_g(\text{days})}{0.693} K \frac{[(\mu Ci/l)(\mu Ci/m^2)]}{365.25 \text{ days/yr.}} \times 10^6 (\mu Ci/Ci)$$

²It has been assumed that there is approximately 1 gram of calcium per liter of milk.

where :

- C = the resulting milk concentration, $\mu\text{Ci/l}$
- Q = the I-131 source term, Ci/yr
- X/Q = the atmospheric dispersion factor, sec/m^3
- V_g = the deposition velocity, m/sec
- T_g = the effective halflife on the ground, days, and
- K = the milk to grass activity ratio, $(\mu\text{Ci/l}) / (\mu\text{Ci/m}^2)$

The effective halflife on grass for I-131 is obtained by combining the halflife due to weathering, about 13 to 14 days, (ref. 18, 27) with the radiological halflife of about 8 days. The resulting value of T_g is 5 days. The value of K has been estimated from field data obtained by Booker (ref. 19) following the Windscale accident of 1959 and is taken to be $0.091(\mu\text{Ci/l}) / (\mu\text{Ci/m}^2)$. Use of these values and the above formula leads to an estimated maximum potential I-131 milk concentration of 8.92×10^{-6} (8.92×10^{-5}) microcuries per liter, again at the site boundary. I-133 may also be taken into account using proper values of the necessary parameters. For I-133 the value of T_g is approximately equal to the radiological halflife of 0.867 days. The value of K for I-133 is taken to be half that for I-131, (ref. 28) or $0.045 (\mu\text{Ci/l}) / (\mu\text{Ci/m}^2)$. The resulting site boundary potential I-133 concentration is then 6.75×10^{-7} (6.75×10^{-6}) microcuries per liter.

If it can be assumed that an adult will drink no more than 0.5 liters of milk daily, (ref. 18) and that a child will drink no more than 1.0 liters of milk daily (ref. 29) the maximum rates of activity intake via this pathway are as presented below:

Maximum Activity Intake Via Milk Ingestion³

<u>Radionuclide</u>	<u>Adult ($\mu\text{Ci/day}$)</u>	<u>Child ($\mu\text{Ci/day}$)</u>
Sr-90	1.38×10^{-11}	2.76×10^{-11}
Cs-134	3.68×10^{-7}	7.35×10^{-7}
Cs-137	1.14×10^{-6}	2.27×10^{-6}
I-131	2.23×10^{-6}	4.46×10^{-6}
I-133	1.69×10^{-7}	3.38×10^{-7}

In the above intake tabulation it has been assumed that cows would graze outdoors only about half the calendar year at the most (ref. 30). Thus, the maximum potential iodine intake has been reduced by a factor of two. The cesium and strontium intakes have been assumed to be unaffected as they may also be contained in stored feed due to their long halflives.

The annual exposure to adults and children due to the above rates of intake has been computed using methods and equations outlined in Subsections 11A.3.4 and 11A.4.5. The resulting whole body exposure to an adult due to milk ingestion from cows located at the site boundary is computed to be 3.12×10^{-2} (3.12×10^{-1}) mrem per year. The critical organ is the thyroid for which the computed adult exposure amounts to 1.64 (16.4) mrem per year, almost entirely due to I-131 and I-133. The resulting exposure to a child's thyroid due to milk ingestion from cows stationed at the site boundary is computed to be 16.2 (162) mrem per year, the associated whole body exposure is negligible in comparison, amounting to only 0.102 (1.02) mrem per year.

³Intake rates presented here are for maximum expected release rates. Intake rates for design basis releases are a factor of ten higher in each case.

The potential exposure limits presented above are perhaps artificially high for the following reason. No cows are presently known to be located either at any portion of the site boundary or in close proximity to the site boundary. In fact the nearest location at which grazing cattle have been reported (in the Davis-Besse PSAR, Chapter 2) is about 1.5 miles to the south-southwest of the station. The AEC in its Draft Environmental Statement has stated for the Davis-Besse Nuclear Power Station, (ref. 6) "The nearest dairy herd is pastured about two miles to the south, and this also represents the nearest probable pasturage." Table 2.7 of the AEC's Draft Environmental Statement lists all locations of dairy cattle within five miles of the site, the nearest location given being 2.5 miles to the west-southwest. The distance to the nearest dairy herd to the south is given in the AEC's Table 2.7 as four miles.

As there are considerable discrepancies in the above data the distance and direction of the nearest cow have been taken to be as given in Chapter 2 of the Davis-Besse PSAR (1.5 miles to the south-southwest).

The applicable X/Q value has been obtained via logarithmic interpolation between the values provided in Section 2.3 of the Davis-Besse USAR for the site boundary (distance = 0.73 kms, X/Q = 4.17×10^{-6}) and for the Low Population Zone (distance = 3.2 kms, X/Q = 3.60×10^{-7}). The atmospheric dispersion value thus determined is 7.96×10^{-7} sec/m³.

Child and adult doses due to milk ingestion have been determined based on this value of X/Q. They can be considered much more realistic in terms of the probability of occurrence than those presented earlier.

At the assumed location milk ingestion results in the following thyroid exposure rates: 0.307 (3.07) mrem per year for an adult; 3.07 (30.7) mrem per year for a small child. The resulting whole body exposure is negligible in comparison. The effect of delay between the production and consumption of the milk was not included nor was the effect of possible dilution by other milk supplies. Throughout this analysis it has been apparent that the resulting exposure from milk ingestion has been dominated by the released radioiodines. Exposure due to strontium and cesium has been negligible in comparison to that due to iodine.

11A.3.4 Computational Methods for Doses Resulting from Gaseous Effluents

11A.3.4.1 Whole Body and Body Surface Cloud Immersion Dose

The gaseous radioactive effluents consist primarily of the noble gases krypton and xenon. Exposure of a man to an atmosphere contaminated with radioactive isotopes of these elements is assumed to result only in an external, whole body dose, from submersion in the radioactive cloud. Since these elements are not incorporated into the human body to a significant degree, there are only negligible resultant internal doses (ref. 24). The noble gases and all other gaseous releases have been included in the calculation of external exposure due to immersion in the radioactive cloud. The resulting dose is proportional to the ground level concentration of radioactivity and can be computed using the ICRP recommended semi-infinite sphere model (ref. 9). The following relationship was used to determine the dose rate from this exposure pathway:

$$D(\text{rem/year}) = 0.259 \times (X/Q) \times \sum_i E_i Q_i$$

where:

X/Q = the applicable annual average atmospheric dispersion parameter, sec/m³;

Davis-Besse Unit 1 Updated Final Safety Analysis Report

E_i = the average disintegration energy⁴ of the i th radionuclide, either beta or gamma;

Q_i = the annual average activity release for the i th radionuclide, Ci/year;

0.259 = the constant necessary to yield the dose rate in rem/year.

The normalization constant, 0.259, is given in the following equation:

$$0.259 = 1/2(1.6 \times 10^{-4} \text{ ergs/MeV}) (10^{-2} \text{ gram rads/erg})(1 \text{ rem/rad})$$

$$(1.13) (3.7 \times 10^{10} \text{ dis/sec Ci}) \dots\dots\dots \left(\frac{1}{1.293 \times 10^3 \text{ grams air / m}^3} \right)$$

where:

1/2 = the geometry factor accounting for the fact that the receptor is irradiated from half the total available solid angle; and

1.13 = a factor to account for the increased stopping power of tissue relative to air for β 's and secondary electrons produced by x- and -y radiation (ref. 9).

The above formula was used for maximum individual exposure and for population exposure. Beta radiation was assumed to irradiate only the body surface. The estimated population distributions within 50 miles of the Station for the years 1970, 1990 and 2010 were used for this purpose. For each of the 160 population segments (16 sectors x 10 annuli) into which the 50 mile population is distributed, an annual average segment dose rate was established. This was taken to be the dose rate at the geometric midpoint of the segment. The annual segment population exposure is thus the product of the segment population and the segment average dose rate. Summing the individual segment population exposure over all 160 segments yields the total population exposure within 50 miles of the Station.

The average value of X/Q for a specific segment was taken to be that for the distance of the midpoint of that segment from the Station. For example, the average X/Q for a sector 10-20 miles in a given direction was taken to be that for a distance of 15 miles in the given direction. The numerical values for X/Q used in this evaluation were obtained from the meteorological data presented in Section 2.3 of the Davis-Besse USAR. Corrections were made to account for the actual annual average wind direction frequencies as depicted in USAR Figure 2.3-4.

11A.3.4.2 External Exposure From Particulate Deposition

The following equation was used to compute the ground activity concentration at the end of the assumed 40-year plant lifetime for each particulate and iodine isotope released:

$$C_g = (\mu\text{Ci/cm}^2) = Q \text{ X/Q } V_g \left(\frac{T}{0.693} \right) \left(\frac{1 \text{ yr}}{365.25 \text{ days}} \right) \left(\frac{10^6 \mu\text{Ci}}{\text{Ci}} \right) \left(\frac{10^{-4} \text{ m}^2}{\text{cm}^2} \right)$$

where :

Q = the release rate, Ci/yr;

X/Q = the applicable atmospheric dispersion parameter, sec/m³;

V_g = the deposition velocity, m/sec; and

⁴Data on disintegration energies was obtained primarily from NEDO-12037, "Summary of Gamma and Beta Energy and Intensity Data", by Meek and Gilbert (January 1970); where necessary data was obtained from the "Table of Isotopes" (ref. 45).

T = the radiological half-life, days.

After calculating the end of 40 years ground concentrations, proper dose conversion factors were applied to obtain the resulting beta and gamma exposure rates at the critical height of one meter. Dose conversion factors for gamma radiation for all radionuclides but Cs-137 were obtained directly out of "Factors for the Calculation of Infinite Plane Exposure Rates From Gamma Radiation", by Crocker, et al. (ref. 12). Only gamma radiation was assumed to irradiate the whole body.

Beta dose conversion factors were calculated using the equation given in EXREM II (ref. 13). The calculated beta exposure rates were then reduced by a factor of two to account for the effect of self-shielding of the human body. Maximum and average beta energies were obtained from the " (ref. 46).

The critical isotope for this exposure pathway was found to be Cs-137. Various analyses of ground concentrations due to fallout Cs-137 have made apparent the ability of cesium to penetrate into soil after initial deposition (ref. 14). The concentration distribution with depth can usually be described in terms of the relaxation length, the volume concentration decreasing exponentially with soil depth. This distribution is obtained relatively rapidly after deposition, usually within one to two years (ref. 15). The range of the relaxation length is usually from three to eight centimeters, the value in a particular instance depending on local precipitation, soil type, and other variables (ref. 15). As the external dose rates vary inversely to the value of the relaxation length, the value used in this analysis was conservatively taken to be three centimeters.

The dose conversion factor for gamma radiation from Cs-137 is necessarily dependent upon the relaxation length. The dose conversion factor appropriate for a three centimeter relaxation length was obtained from data published by Beck, (ref. 47) and is equal to 0.0429 (r/hr) / (μCi/cm²). The beta component of the Cs-137 radiation was conservatively estimated by assuming that only the activity in the top centimeter of soil could contribute to the dose. A distance of one centimeter in soil represents about 80 half-thicknesses for 0.2 MeV betas; the average beta disintegration energy for Cs-137 is 0.195 MeV (ref. 46).

11A.3.4.3 Internal Exposure Due to Inhalation

Exposure rates have been computed for inhalation of iodine, tritium and particulates. In each case the first step in the calculation has been to estimate the daily activity intake via inhalation. The amount of activity reaching the body following inhalation has been estimated using the following formula:

$$A (\mu\text{Ci/day}) = Q \times X/Q \times B \left(\frac{1 \text{ yr}}{365.25 \text{ days}} \right) \left(\frac{10^4 \mu\text{Ci}}{\text{Ci}} \right) f_s$$

where:

Q = the release rate, Ci/yr;

X/Q = the applicable atmospheric dispersion parameter, sec/m³;

B = the breathing rate, m³/sec; and

f_s = the fraction reaching the thyroid via inhalation

The value of B for an adult was taken to be 2.32 x 10⁻⁴ m³/sec which is equivalent to the value of 20 m³/day recommended by the ICRP (ref. 9). Since the more critical receptor for this pathway

is a young child, especially for I-131, an estimate of child exposure was also made. The breathing rate for a young child was taken to be 6 m³/day (refs. 18, 48).

a. Exposure Due to Tritium Inhalation

As tritium can rapidly be absorbed through the human skin, inhalation is not the only source of tritium intake (ref. 20). As recommended by the ICRP (ref. 9) the total tritium intake via inhalation and skin absorption is taken to be twice that due to inhalation alone.

Having established the rate of tritium intake, the resulting adult exposure was obtained using methods outlined in Subsection 11A.4.5. Exposure to a child was assumed to be identical to that for an adult.

b. Exposure Due to Radioiodine Inhalation

For adults, the daily intake via inhalation of the various radioiodines was estimated using the formula given above. The dose conversion factors already presented in previous sections to arrive at the thyroid exposure rates. Other organ doses and the whole body dose were obtained using methods given in Subsection 11A.4.5. For child thyroid exposure it was necessary to calculate proper dose conversion factors. The equation used is given below:

$$DCF_i \text{ (rem/Ci)} = f_a \left(\frac{T_{\text{eff } i}}{0.693} \right) \left(\frac{3.7 \times 10^{10} \frac{\text{dis}}{\text{sec}} - \mu\text{Ci}}{m} \right) \left(8.64 \times 10^4 \frac{\text{sec}}{\text{day}} \right) E_{\text{eff}} \frac{(\text{MeV})}{\text{dis}} \left(1.6 \times 10^{-8} \frac{\text{q-rem}}{\text{MeV}} \right)$$

where:

DCF_i (rem/Ci) = the dose conversion factor for the i th iodine isotope, rem/Ci inhaled;
 f_a = the fraction reaching the thyroid via inhalation;
 $T_{\text{eff } i}$ = the effective biological half-life in the thyroid, days;
 $E_{\text{eff } i}$ = the effective disintegration energy, MeV/dis; and
 m = the mass of the thyroid gland, grams.

The value of f_a was taken to be 0.23, the value given in ICRP II for adult iodine inhalation (ref. 9). The fraction of iodine reaching the thyroid does not vary significantly with age (ref. 49). The values of T_{eff} for each iodine isotope were obtained by combining the radiological half-life with the assumed 8-day biological half-life for iodine in the thyroid (ref. 48). The values of the effective energy were taken to be those given by Bryant (ref. 18) for all iodine isotopes except I-131. The effective energy of I-131 was taken to be 0.21 MeV as estimated by Rohwer and Kaye (ref. 48). The dose conversion factors so calculated were applied to estimate the maximum potential child thyroid exposure via inhalation. It was found that the child thyroid dose from this route was about 74% greater than the equivalent adult thyroid exposure.

c. Exposure Due to Particulate Inhalation

Using formulas already given, the daily activity intake of each particulate radionuclide for an adult via inhalation was determined. These rates of daily intake were then converted to whole body and body organ exposure rates by the DOSCAL computer code, which is described in Subsection 11A.4.5. From the DOSCAL output it was determined that the critical radionuclides for this exposure pathway were Cs-134 and Cs-137; these two isotopes contributed about 98% of the adult whole body exposure. Using data given by Bryant (ref. 18) the whole body exposure for a child was estimated. The body surface and thyroid exposure for a child were assumed to be equal to the calculated whole body exposure, as was the case for an adult.

11A.3.4.4 Exposure Due to Food Chain Transport

The manner in which the daily activity intakes for this pathway were established is described fully in this appendix. For adult exposure these intake rates were converted to exposure rates by the DOSCAL code. For child thyroid exposure from I-131 and I-133 the dose conversion parameters already developed to compute inhalation thyroid exposure were used. It was necessary only to replace f_a with f_w ; the fraction of ingested iodine reaching the thyroid. The value of f_w was taken to be 0.3, as given by the ICRP (ref. 9) and recommended by the FRC (ref. 49). The whole body child exposure due to Cs-134 and Cs-137 was calculated in the same manner as for particulate inhalation, using the dosimetric data provided by Bryant (ref. 18).

11A.3.5 Summary of Exposure from Gaseous Releases

In an effort to unify the results of this section, Table 11A.3-7 has been drawn up, illustrating the calculated maximum individual exposure rates from the maximum expected gaseous releases for the whole body, body surface, and thyroid. Results are presented for adult and child exposure and are primarily based on exposure at the site boundary.

Based on milk ingestion from a cow at the site boundary, the total whole body exposure from all gaseous releases to a child is 1.41 mrem per year, about equal to the equivalent adult exposure of 1.32 mrem per year. The whole body exposure rates are about a factor of two less than the calculated body surface exposure totals of 2.74 mrem per year for child and 2.64 mrem per year for an adult. Higher still are the thyroid exposure totals of 17.0 mrem per year (child) and 3.17 mrem per year (adult).

Regarding the thyroid exposure, it should be remembered that the exposure rates are greatly reduced if it is not assumed that cows graze at the site boundary. Based on milk ingestion at the nearest reported grazing location the total thyroid exposure from all sources is 4.77 mrem per year for a child and 1.84 mrem per year for an adult.

Population exposure totals have been calculated for external exposure and have been presented earlier in Table 11A.3-4. As internal exposure is generally less significant than external exposure, the exception being the milk ingestion pathway, population exposure has not been calculated for internal exposure. Exposure to iodine, via inhalation or ingestion does not cause a significant whole body exposure which is the primary concern for exposure of large population groups (ref. 9).

TABLE 11A.3-1

Estimated Airborne EffluentsTotal Annual Activity Release, Curies (ref. 2)

<u>Radionuclide</u>	<u>Expected Release*</u>	<u>Design Basis Release*</u>
Noble Gases:		
Ar-41	1.47 + 1	1.47 + 1
Kr-83m	3.94	3.94 + 1
Kr-85m	2.06 + 1	2.06 + 2
Kr-85	2.95 + 2	2.95 + 3
Kr-87	1.12 + 1	1.12 + 2
Kr-88	3.60 + 1	3.60 + 1
Xe-131m	5.56 + 1	5.56 + 2
Xe-133m	4.09 + 1	4.09 + 2
Xe-133	3.77 + 3	3.77 + 4
Xe-135m	3.05 + 2	3.05 + 3
Xe-135	1.07 + 2	1.07 + 3
Xe-138	6.85	6.85 + 1
Iodines:		
I-131	1.19 - 1	1.19
I-132	5.20 - 2	5.20 - 1
I-133	1.05 - 1	1.05
I-134	8.61 - 3	8.61 - 2
I-135	4.96 - 2	4.96 - 1
Tritium:		
H-3	3.90 + 2	3.90 + 2
Particulates:		
Cr-51	1.63 - 4	1.63 - 4
Mn-54	1.86 - 5	1.86 - 5
Fe-55	6.40 - 4	6.40 - 4
Fe-59	1.86 - 5	1.86 - 5
Co-58	9.80 - 4	9.80 - 4
Co-60	5.20 - 6	5.20 - 6
Rb-87	2.22 - 10	2.22 - 9
Rb-88	2.72 - 1	2.72
Sr-89	1.98 - 5	1.98 - 4
Sr-90	6.40 - 7	6.40 - 6
Sr-91	1.20 - 4	1.20 - 3
Sr-92	3.30 - 5	3.30 - 4
Y-90	4.42 - 5	4.42 - 4
Y-91	2.49 - 4	2.49 - 3
Zr-95	1.28 - 3	1.28 - 3
Mo-99	1.80 - 2	1.80 - 1

* Numerical data are presented in the form $1.47 + 1 = 1.47 \times 10^1$.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.3-1 (Continued)

Estimated Airborne Effluents

Total Annual Activity Release, Curies (ref. 2)

<u>Radionuclide</u>	<u>Expected Release*</u>	<u>Design Basis Release*</u>
Particulates, (Cont'd):		
Cs-134	1.80 - 2	1.80 - 1
Cs-135	2.06 - 5	2.06 - 4
Cs-136	3.25 - 3	3.25 - 2
Cs-137	5.55 - 2	5.55 - 1
Cs-138	1.77 - 2	1.77 - 1
Ba-137m	4.50 - 3	4.50 - 2
Ba-139	2.53 - 4	2.53 - 3
Ba-140	2.47 - 5	2.47 - 4
La-140	9.80 - 6	9.80 - 5
Ce-144	2.28 - 6	2.28 - 5

* Numerical data are presented in the form $1.47 + 1 = 1.47 \times 10^1$.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.3-2

External Exposure Due to Ground Deposition of Particulates and Halogens

Radionuclide	Estimated Release Rate,* Ci/yr	Half-life, days	Site Boundary Ground Concentration After 40 Years, $\mu\text{Ci}/\text{cm}^2$	Exposure Rate at 1 Meter, mrem/yr	
				Beta	Gamma
Cr-51	1.63 - 4	2.78 + 1	1.49 - 11	0	7.73 - 7
Mn-54	1.86 - 5	3.13 + 2	1.92 - 11	0.	2.68 - 5
Fe-55	6.40 - 4	9.86 + 2	2.08 - 9	0.	0.
Fe-59	1.86 - 5	4.50 + 1	2.76 - 12	3.65 - 6	5.19 - 6
Co-58	9.80 - 4	7.14 + 1	2.31 - 10	0.	3.19 - 4
Co-60	5.20 - 6	1.92 + 3	3.27 - 11	3.06 - 5	1.26 - 4
Rb-87	2.22 - 10	1.83 + 13	7.41 - 15	8.90 - 15	0.
Rb-88	2.72 - 1	1.23 - 2	1.10 - 11	1.67 - 4	1.11 - 5
Sr-89	1.98 - 5	5.08 + 1	3.31 - 12	1.76 - 5	0.
Sr-90	6.40 - 7	1.06 + 4	1.38 - 11	2.62 - 6	0.
Sr-91	1.20 - 4	4.03 - 1	1.59 - 13	1.11 - 6	2.16 - 7
Sr-92	3.30 - 5	1.12 - 1	1.22 - 14	2.93 - 8	2.33 - 8
Y-90	4.42 - 5	2.67 + 0	3.89 - 13	4.29 - 6	0.
Y-91	2.49 - 4	5.88 + 1	4.82 - 11	3.40 - 4	2.70 - 7
Zr-95	1.28 - 3	6.55 + 1	2.76 - 10	2.36 - 4	3.38 - 4
Mo-99	1.80 - 2	2.78 + 0	1.65 - 10	5.40 - 4	3.52 - 5
Cs-134	1.80 - 2	7.52 + 2	4.46 - 8	6.65 - 2	1.18 - 1
Cs-135	2.06 - 5	8.40 + 8	6.87 - 10	9.05 - 14	0.
Cs-136	3.25 - 3	1.30 + 1	1.39 - 10	2.00 - 5	5.27 - 4
Cs-137	5.55 - 2	1.10 + 4	1.21 - 6	5.20 - 1	4.55 - 1
Cs-138	1.77 - 2	2.24 - 2	1.31 - 12	1.41 - 5	4.16 - 6
Ba-137m	4.50 - 3	1.77 - 3	2.62 - 14	0.	2.70 - 8
Ba-139	2.53 - 4	5.78 - 2	4.82 - 14	5.15 - 7	3.99 - 9
Ba-140	2.47 - 5	1.28 + 1	1.04 - 12	1.77 - 6	3.91 - 7
La-140	9.80 - 6	1.68 + 0	5.42 - 14	3.16 - 7	2.00 - 7
Ce-144	2.28 - 6	2.84 + 2	2.14 - 12	8.35 - 11	8.54 - 8
I-131	1.91 - 1	8.07 + 0	1.58 - 8	9.35 - 3	1.13 - 2
I-132	5.20 - 2	9.52 - 2	8.16 - 11	4.48 - 4	2.96 - 4
I-133	1.05 - 1	8.67 - 1	1.50 - 9	7.15 - 3	1.53 - 3
I-134	8.61 - 3	3.63 - 2	5.15 - 12	3.97 - 5	2.04 - 5
I-135	4.96 - 2	2.79 - 1	2.28 - 10	7.10 - 4	1.15 - 3
Totals:			1.276-6	6.054-1	5.884-1

* All numbers presented in this table are based on the expected release rates.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.3-3

Estimated Population Distributions Within 50 Miles of the Station for the Year 1970

<u>DIR.</u>	Number of Persons at Indicated Distance										<u>DIRECTION TOTAL</u>
	<u>0-1 MILES</u>	<u>1-2 MILES</u>	<u>2-3 MILES</u>	<u>3-4 MILES</u>	<u>4-5 MILES</u>	<u>5-10 MILES</u>	<u>10-20 MILES</u>	<u>20-30 MILES</u>	<u>30-40 MILES</u>	<u>40-50 MILES</u>	
NNE	40	0	0	0	0	0	0	908	17327	18920	37195
NE	7	0	0	0	0	0	0	54	5337	13125	18523
ENE	0	0	0	0	0	0	507	479	0	0	986
E	0	0	0	0	0	0	1596	97	0	27099	28792
ESE	0	0	0	0	0	0	10799	41039	15472	100699	168009
SE	0	25	0	0	98	2356	5257	10864	22566	9543	50709
SSE	0	47	52	70	75	1324	4000	15256	5022	16358	42204
S	49	47	71	65	83	2198	26958	8591	27933	7707	75502
SSw	82	43	43	64	109	2316	3943	5326	24728	10186	46840
SW	30	0	35	59	65	1302	5860	10134	8928	45329	71742
WSW	0	10	10	43	60	786	8179	17889	32124	9131	68232
W	19	62	36	57	115	817	13322	231673	38667	16079	300847
WNW	0	58	296	0	0	803	11664	203406	36725	10656	263608
NW	4	244	63	0	0	0	0	37635	12968	15287	66201
NNW	113	35	0	0	0	0	0	12854	32769	153120	198891
N	19	0	0	0	0	0	0	248	50431	578697	629395
ANNULUS TOTAL	363	571	606	358	605	11902	92085	596453	332797	1031936	2067676

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.3-3 (Continued)

Estimated Population Distributions Within 50 Miles of the Station for the Year 1990

<u>DIR.</u>	Number of Persons at Indicated Distance										<u>DIRECTION TOTAL</u>
	<u>0-1 MILES</u>	<u>1-2 MILES</u>	<u>2-3 MILES</u>	<u>3-4 MILES</u>	<u>4-5 MILES</u>	<u>5-10 MILES</u>	<u>10-20 MILES</u>	<u>20-30 MILES</u>	<u>30-40 MILES</u>	<u>40-50 MILES</u>	
NNE	40	0	0	0	0	0	0	1760	33580	36667	72047
NE	7	0	0	0	0	0	0	54	10343	25436	35840
ENE	0	0	0	0	0	0	557	479	0	0	1036
E	0	0	0	0	0	0	1764	120	0	37509	39393
ESE	0	0	0	0	0	0	13330	50655	19098	138879	221962
SE	0	25	0	0	107	2764	6233	13236	24992	10451	57808
SSE	0	47	52	85	81	1595	4640	17210	5378	18470	47558
S	49	47	71	65	89	2660	31270	9602	31056	8384	83293
SSw	82	43	43	64	113	2771	4574	5835	30277	11226	55028
SW	30	0	35	59	65	1547	6694	14155	13270	59038	94893
WSW	0	10	10	43	60	917	9815	26911	47476	11977	97219
W	19	62	36	57	115	959	17124	263278	45436	20326	347412
WNW	0	38	296	0	0	962	13706	280802	95775	16500	408079
NW	4	244	63	0	0	0	0	168647	58111	50019	277088
NNW	113	35	0	0	0	0	0	57781	87518	196850	342297
N	19	0	0	0	0	0	0	764	73268	746552	820603
ANNULUS TOTAL	363	551	606	373	630	14175	109707	911289	575578	1388284	3001556

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.3-3 (Continued)

Estimated Population Distributions Within 50 Miles of the Station for the Year 2010

<u>DIR.</u>	Number of Persons at Indicated Distance										<u>DIRECTION TOTAL</u>
	<u>0-1 MILES</u>	<u>1-2 MILES</u>	<u>2-3 MILES</u>	<u>3-4 MILES</u>	<u>4-5 MILES</u>	<u>5-10 MILES</u>	<u>10-20 MILES</u>	<u>20-30 MILES</u>	<u>30-40 MILES</u>	<u>40-50 MILES</u>	
NNE	50	0	0	0	0	0	0	2865	54667	59693	117275
NE	9	0	0	0	0	0	0	54	16838	41410	58311
ENE	0	0	0	0	0	0	602	479	0	0	1081
E	0	0	0	0	0	0	1919	143	0	49261	51323
ESE	0	0	0	0	0	0	15895	60405	22774	181923	280997
SE	0	31	0	0	119	3177	7198	15626	27214	11261	64626
SSE	0	58	65	94	91	1833	5253	19045	5691	20448	52578
S	60	58	88	93	107	3128	35401	10556	32184	9014	90689
SSW	101	53	53	79	136	3229	5179	6308	36052	12280	63470
SW	38	0	44	74	82	1807	7486	18953	18560	73727	120771
WSW	0	12	12	50	74	1074	11614	37981	66214	15279	132310
W	24	76	45	71	141	1121	21501	294476	52264	25036	394755
WNW	0	72	366	0	0	1125	15766	372660	170682	30301	590972
NW	6	302	78	0	0	0	0	337670	116351	95918	550325
WNW	140	43	0	0	0	0	0	115780	158242	254428	528633
N	24	0	0	0	0	0	0	1426	103042	967473	1071965
ANNULUS TOTAL	452	705	751	461	750	16494	127814	1294427	880775	1847452	4170081

TABLE 11A.3-4

External Population Exposure Due to Gaseous Releases*

Year	Population Exposure Due to Cloud Immersion, man-rem/year				Average Per Capita Exposure Due to Cloud Immersion, mrem/year			
	Whole Body Exposure		Body Surface Exposure		Whole Body Exposure		Body Surface Exposure	
	E.R.	D.B.R.	E.R.	D.B.R.	E.R.	D.B.R.	E.R.	D.B.R.
1970	0.731	7.31	1.605	15.71	3.54-4	3.54-3	7.76-4	7.60-3
1990	0.958	9.58	2.103	20.58	3.19-4	3.19-3	7.00-4	6.85-3
2010	1.266	12.66	2.779	27.20	3.04-4	3.04-3	6.66-4	6.52-3
	Population Exposure Due to Particulate Deposition, man-rem/year				Average Per Capita Exposure Due to Particulate Deposition, mrem/year			
	Whole Body Exposure		Body Surface Exposure		Whole Body Exposure		Body Surface Exposure	
	E.R.	D.B.R.	E.R.	D.B.R.	E.R.	D.B.R.	E.R.	D.B.R.
1970	0.725	7.25	1.474	14.74	3.51-4	3.51-3	7.15-4	7.15-3
1990	0.951	9.51	1.932	19.32	3.17-4	3.17-3	6.44-4	6.44-3
2010	1.256	12.56	2.552	25.52	3.02-4	3.02-3	6.13-4	6.13-3
	Total External Population Exposure, man-rem/year				Total Average Per Capita External Exposure, m-rem/year			
	Whole Body Exposure		Body Surface Exposure		Whole Body Exposure		Body Surface Exposure	
	E.R.	D.B.R.	E.R.	D.B.R.	E.R.	D.B.R.	E.R.	D.B.R.
1970	1.456	14.56	3.079	30.45	7.05-4	7.05-3	1.49-3	1.48-2
1990	1.909	19.09	4.035	39.90	6.36-4	6.36-3	1.34-3	1.33-2
2010	2.522	25.22	5.331	52.72	6.06-4	6.06-3	1.28-3	1.27-2

* The abbreviations E.R. and D.B.R. are used to denote maximum expected releases and design basis releases.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.3-5

Applicable Annual Average X/Q Values for the Davis-Besse Site (ref. 7)

Distance		Annual Average X/Q Value, sec/m ³ *
<u>Miles</u>	<u>Meters</u>	
0.5	800	3.89×10^{-6}
1.5	2,400	7.96×10^{-7}
2.5	4,000	3.06×10^{-7}
3.5	5,600	2.21×10^{-7}
4.5	7,200	1.59×10^{-7}
7.5	12,000	5.97×10^{-8}
15	24,000	1.56×10^{-8}
25	40,000	6.85×10^{-9}
35	56,000	4.12×10^{-9}
45	72,000	2.80×10^{-9}

* These values are appropriate for a direction into which the wind blows with an annual average frequency of 15%.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.3-6

Annual Average Wind Direction Frequencies at the Davis-Besse Site

<u>Direction Winds Are From</u>	<u>Frequency, % of Year</u>
NNE	3.9
NE	4.0
ENE	5.3
E	6.3
ESE	4.8
SE	2.6
SSE	2.4
S	6.6
SSW	10.7
SW	12.7
WSW	12.5
W	6.8
WNW	4.8
NW	4.3
NNW	3.9
N	<u>3.0</u>
Total	94.8
Calms	<u>5.2</u>
TOTAL	100.0

TABLE 11A.3-7

Summary of Maximum Individual Exposure Rates Due to Gaseous Releases

Exposure Rates Presented are Based on Expected Releases

	<u>Child mr/yr*</u>			<u>Adult mr/yr*</u>		
	<u>W.B.</u>	<u>Body Surface</u>	<u>Thyroid</u>	<u>W.B.</u>	<u>Body Surface</u>	<u>Thyroid</u>
Cloud immersion	0.593	1.30	0.593	0.593	1.30	0.593
External from particulate deposition	0.588	1.194	0.588	0.588	1.194	0.588
Iodine inhalation	0.000	0.000		0.000		
	4	4	0.382	4	0.0004	0.220
Tritium inhalation	0.096	0.113	0.125	0.096	0.113	0.125
Particulate inhalation	0.001	0.001		0.002		0.002
	2	2	0.0012	8	0.0028	8
Food chain transport:						
At nearest cow	0.0195	0.0195	3.07	0.0060	0.0060	0.307
At site boundary	0.102	0.102	16.2	0.0312	0.0312	1.64
Exposure totals:						
Milk ingestion from Nearest cow:	1.33	2.66	4.77	1.29	2.62	1.84
Milk ingestion from a cow located at the site Boundary:	1.41	2.74	17.9	1.32	2.64	3.17

* Exposure rated due to design basis releases are roughly a factor of ten high in each case, except for exposure due to tritium which is unchanged.

11A.4.0 Radiation Exposure from Liquid Effluents

When it becomes necessary to discharge liquid radwaste from the Davis-Besse Nuclear Power Station, it is planned that the liquid waste will be injected into a dilution flow of approximately 20,000 gpm (ref. 31). This dilution flow consists of the discharge from one of the cooling tower makeup pumps and the cooling tower blowdown flow. A mixing factor of 0.8 is assumed (ref. 31) which yields an effective dilution flow of 16,000 gpm. Thus diluted, the liquid radwaste will be discharged to Lake Erie. This section evaluates the potential radiation exposure occurring as a result of the anticipated discharge of small amounts of radioactive materials in this manner.

11A.4.1 Estimated Liquid Effluents and Concentrations

Table 11A.4-1 presents the maximum expected and design basis annual average activity release and discharge concentrations for liquid radwaste from Davis-Besse. Estimates of annual activity releases have been provided by Bechtel Corporation (ref. 2). Annual average discharge concentrations have been obtained applying the effective dilution flow of 16,000 gpm. These discharge concentrations form the basic data upon which all maximum individual exposure rates due to liquid releases are based.

Also of concern is the total integrated population exposure due to liquid releases of activity. Since the population groups affected are at relatively large distances from the site, it is necessary to obtain some estimate of the maximum radioactivity concentrations due to the plant as a function of lake transport distance and direction. The following locations are of particular interest due to their use of Lake Erie as a source of potable water:

<u>Location Served</u>	<u>Distance from Station Discharge to Intake (ref. 32)</u>
Erie Industrial Park	3.6 mi. (5.79 km.)
Camp Perry	4.6 mi. (7.4 km.)
Port Clinton	8.6 mi. (14 km.)
Oregon	12.0 mi. (19.3 km.)
Toledo	12.0 mi. (19.3 km.)

Subsection 11.2.7.2 of the Davis-Besse USAR provides an adequate mathematical model for predicting minimum dispersion effects in Lake Erie at the distances needed. The modified Fickian dispersion model presented there yields the following equation for predicting the plume centerline concentration:

$$C(x) = \frac{1.59 \times 10^{-3}}{x} Qf$$

where:

$C(x)$ = average centerline concentration, $\mu\text{Ci/cc}$
 Q = point source continuous release source term, $\mu\text{Ci/sec}$
 x = the downstream distance, centimeters, and
 f = frequency of flow in the direction of interest.

Due to the mathematical nature of the model, concentration predictions obtained with it are inaccurate near the point of release. Its use here is restricted to distances large enough to avoid the region of initial mixing.

As explained in USAR Subsection 11.2.7.2 the concentration predictions must be weighted by an assumed frequency of flow in the direction of interest. Wind directional frequencies given in Table 11A.3-6 of this report have been used to establish the necessary data. Winds out of the following directions are assumed to produce a surface current lake flow towards Toledo and Oregon: SSE, SE, ESE, and E. Winds out of the ENE, NE, NNE, N, and NNW are assumed to produce currents towards Camp Perry and Port Clinton. Thus, the frequency of flow to the northwest, after accounting for calms, is about 17%, and about 21% to the southeast towards Camp Perry.

Effective dilution factors have been obtained for the locations of concern by taking the ratio of the annual average discharge concentration for tritium and the concentrations predicted by the model. At the maximum expected release rate of 346 curies per year the annual average discharge concentration of tritium is $1.087 \times 10^{-5} \mu\text{Ci/cc}$. The predicted tritium concentrations at the given locations and the derived dilution factors are presented below:

<u>Location</u>	<u>H-3 Concentration</u>	<u>Effective Dilution Factor</u>
Erie Industrial Park	$6.38 \times 10^{-9} \mu\text{Ci/cc}$	1700
Camp Perry	$5.00 \times 10^{-9} \mu\text{Ci/cc}$	2170
Port Clinton	$2.67 \times 10^{-9} \mu\text{Ci/cc}$	4070
Oregon	$1.54 \times 10^{-9} \mu\text{Ci/cc}$	7080
Toledo	$1.54 \times 10^{-9} \mu\text{Ci/cc}$	7080

These derived effective dilution factors are utilized to predict the annual average radioactivity concentrations at the potable water intakes for the given locations.

11A.4.2 Maximum Individual Radiation Exposure from Liquid Effluents

In the region of the Davis-Besse site, Lake Erie contains pollutants from industrial and municipal wastes. In the immediate vicinity of the site (at the point of discharge) it is not expected that any individual will consume as potable water the untreated cooling tower blowdown either before or after discharge to Lake Erie. Nor is it expected that any individual will swim extensively in these same waters, although this is more likely than the water ingestion mentioned above. However, for the purpose of establishing the maximum potential individual radiation exposure due to the normal operation of the Davis-Besse Nuclear Power Station, the following "maximum individual" is hypothesized:

1. the maximum individual is assumed to consume 1.2 liters daily of the cooling tower blowdown flow. This water ingestion represents the tap water intake that will adequately sustain an average adult male, as determined by the ICRP (ref. 9);
2. the maximum individual is assumed to consume 50 grams daily of fish grown since birth in waters containing annual average discharge concentrations of all radionuclides; and
3. the maximum individual is assumed to swim 200 hours per year in waters containing the annual average discharge concentrations of all radionuclides.

Other less significant exposure pathways are also considered. The above assumptions are discussed in more detail in the following paragraphs:

1. Internal Exposure from Water Ingestion:

Although it is extremely unlikely that any individual will ever drink water from the cooling tower blowdown, maximum individual exposure is calculated on this basis and presented as an upper limit to the potential drinking water ingestion dose. The assumed intake rate is 1.2 liters per day. This represents the entire daily tap water requirement of a "standard man", as recommended by the ICRP (ref. 9).

The ingestion of radionuclides will generally cause an uneven distribution of dose within the human body. Some elements, hydrogen in particular, become rather evenly distributed. Others, such as iodine or cesium, are preferentially taken up by certain body organs. These phenomena produce particular organ doses which can be either higher or lower than the associated whole body dose.

The array of particular organ doses due to the hypothesized water ingestion has been calculated using the NUS computer code DOSCAL. The results of this calculation are summarized in Table 11A.4-2 "Internal Body Organ Exposure from Ingestion of Water and Fish". Calculational methods used in DOSCAL are generally those advocated by such bodies as the ICRP, the IAEA (International Atomic Energy Agency) and the USAEC.

The highest dose to the maximum individual from this route of exposure is calculated to be 1.27 (5.78) mrem per year to the thyroid. Of this total about 62% (17%) is due to tritium. The remainder is almost entirely due to the five isotopes of iodine. The total annual whole body exposure from this route is calculated to be 0.64 (1.15) mrem of which over 93% (76%) is due to tritium alone.

It should be noted that the nearest known potable water intake, from the point of discharge, is located about 5.8 kilometers distant to the southeast. Concentrations of radionuclides at the water intakes at this location have been conservatively estimated to be a factor of 1700 lower than at discharge. Thus, the actual expected doses from water ingestion are at least a factor of 1700 lower than the maximum hypothetical doses. Also, standard water treatment facilities often have the capacity to reduce radioactive content in water prior to distribution to the public (ref. 33). No credit is assumed either for this or for radioactive decay in transit.

2. Internal Exposure from Fish Ingestion:

A number of factors are required to compute the internal doses from ingestion of fish which might conceivably contain radioactivity from the plant liquid effluents. In addition to the activity concentrations in the waters of interest, the concentration of the activity in the fish and the amount consumed must be established.

Aquatic organisms, through biological processes, have the ability to concentrate radionuclides released from the plant. This concentration of activity in aquatic organisms, which are in turn ingested by man, must be considered in determining the possible dose to man. The ratio of the concentration of a radionuclide in an aquatic organism to that in the ambient water is known as the concentration factor (CF). The concentration factor varies among the different species of aquatic life and, for a given specie, varies with the different radionuclides. Also, the concentration may vary considerably between different organs of an organism. For the dose calculations, appropriate concentration factors were used for the edible portions of fish and are shown in Table 11A.4-3.

In order to determine the dose to humans, the quantity of fish eaten must be estimated. The dose model used postulates that the maximum individual consumes 50 grams of fish flesh every day. This is about equal to the seafood consumption reported for commercial fishermen (ref. 35) and about four times the annual per capita consumption of seafood in the United States (ref. 36).

The various maximum doses to an individual have been calculated by the NUS computer code DOSCAL and are presented in Table 11A.4-2. The ingested fish was assumed to have concentrations of all radionuclides in the edible flesh equal to the annual average discharge concentrations times the appropriate concentration factors, as presented in Table 11A.4-3. In the case of expected release rates the highest dose so derived amounts to 3.49 mrem per year to the liver, over 97% of which is due to Cs-134 and Cs-137. In the case of design basis release rates based on 1.0% defective fuel cladding the organ receiving the maximum dose remains the liver, receiving 34.2 mrem per year. Whole body exposure from this exposure pathway amounts to 1.56 (15.0) mrem per year.

There is a low probability of any individual receiving the exposures summarized above, for several reasons. First, rapid dispersion of the heated effluents containing these radionuclides will occur, thus eliminating any sizeable area of Lake Erie as having annual average concentrations of radioactivity approaching the discharge concentrations. Second, the probability of an individual eating fish obtained only from a single location is very low. Finally, fish are highly mobile and are not likely to remain near the discharge point throughout the year, particularly in the warmer seasons; in winter, access for fishing is likely to be limited. In view of these facts, fish ingestion doses presented above are considered to be quite conservative.

3. External Exposure from Swimming:

The external exposure of an individual by submersion in waters containing the radioactive effluents of the Davis-Besse Power Station is presented only to further insure the completeness of this study. Not only is such exposure extremely unlikely, but the direct irradiation of humans while swimming presents only a minor exposure pathway. The radionuclide that will be present in the cooling tower blowdown in the highest concentration is tritium which yields only weak beta radiation upon decay. The beta particles emitted from decaying tritium nuclei do not have sufficient energy to penetrate the human skin and therefore, cannot contribute externally to a whole body dose. As with external exposure from gaseous releases, the beta component of the total exposure is assumed to irradiate only the body surface. Only gamma radiation is assumed to contribute to whole body exposure.

The external exposure to swimmers may be conservatively estimated by assuming that the swimmer is completely immersed in an infinite medium of uniform concentration and receives the same dose as the water itself. The expression for the dose rate is given for each radionuclide by:

$$\begin{aligned} R_{\text{rem/hour}} &= \frac{C_w}{P_w} \bar{E}_{\beta,\gamma} \left(3600 \frac{\text{sec}}{\text{hour}} \right) \left(3.7 \times 10^4 \frac{\text{dis}}{\text{sec} - \mu\text{Ci}} \right) \dots \\ &\dots \left(1.6 \times 10^{-6} \frac{\text{ergs}}{\text{MeV}} \right) \left(10^{-2} \frac{\text{gm} - \text{rad}}{\text{erg}} \right) (\text{RBE})(\text{gf}) \end{aligned}$$

where:

C_w = the concentration in water of the radionuclide, $\mu\text{Ci/cc}$;

P_w = the density of the water, 1.0 grams/cc;

Davis-Besse Unit 1 Updated Final Safety Analysis Report

$E_{\beta,\gamma}$ = the average disintegration energy (beta or gamma) of the radionuclide, MeV;
RBE = the "relative biological effectiveness" for the radiation involved, rem/rad;
gf = the "geometry factor".

Utilization of the model to compute human exposure from swimming involves the following assumptions, all of which tend to over-estimate the dose:

- a. The receptor is surrounded by a body of water of infinite dimension;
- b. The concentrations of radioactivity are everywhere uniform.

The value of the RBE is unity for all radiations of interest except for the weak tritium betas for which the RBE is equal to 1.7 (ref. 9). The geometry factor is designed to account for the fraction of the total solid angle from which radiation may be received. For gamma radiation the value of "gf" is taken to be unity for all radionuclides. This is conservative because the receptor is usually near the surface where gamma radiation from the water comes only from half the total solid angle. For beta particles the gf is taken to be equal to 0.5. This is done to account for the significant self-shielding of the human body against fission product beta radiation; the fission product betas of concern here have a range in human flesh on the order of one centimeter or less.

Tritium, which emits no gammas and only very soft betas, contributes only an external skin dose. The infinite sphere model remains applicable and the dose may be calculated with the equation given above. However, due to the fact that the normally held water in the skin and the water⁵ external to the skin rapidly equilibrate through exchange processes (ref. 20) the value of the "gf" is taken to be unity; i.e., beta irradiation of the skin from tritium comes from all directions.

Application of the above method yields a whole body submersion dose rate from gamma radiation of 3.76×10^{-3} (1.47×10^{-2}) mrem per hour. The associated body surface exposure rate is much higher, largely due to the tritium component, amounting to 4.77×10^{-2} (6.28×10^{-2}) mrem per hour. Assuming that the maximum individual will swim 200 hours per year in the discharge region, the potential annual exposure from this pathway is 0.0038 (0.0147) mrem per year to the whole body and 0.0515 (0.0775) mrem per year to the body surface.

4. Radiation Exposure Other Sources:

Two remaining exposure pathways for liquid radwaste are discussed in this section. These pathways are of lesser significance than those previously discussed. Other exposure pathways of negligible significance and therefore not discussed further are:

- a. The drinking of lake water by animals, flesh from which is subsequently eaten by man;
- b. The uptake of lake water in crops, via irrigation, which are subsequently eaten by man; and
- c. the seepage of lake water into the ground water table from which water may be taken via wells.

⁵ Tritiated water has been found to pass through the skin with a half-time of about three minutes (ref. 20).

These exposure pathways are ignored due to their relative insignificance in comparison to the pathways considered.

A potentially significant exposure pathway does exist for persons sunbathing or walking along the shore of the lake. The radionuclides in the liquid wastes discharged via the cooling tower blowdown flow at the Davis-Besse Nuclear Station can be expected to accumulate to some degree on the bottom sediments and the shoreline sand. The affected area would be limited to approximately the area of the beach between the low and high water marks.

The concentrations of the radionuclides on the shore are influenced by the dilution that occurs in the lake, the chemical composition of the effluent as well as the aquatic environment, and the sorptive capacity of the shoreline soil, which is usually expressed as a concentration factor between water and soil. The ability of soils to concentrate radioactive materials differs widely from one element to another. Radionuclides are removed from solution primarily by absorption and ion exchange. Generally, the fine grained bottom sediments are more effective sorbers of radionuclides than are the coarser grained sands.

The largest discharges of radioactive liquids to the environment have been made from the Windscale site in England and at Hanford in the United States. Smaller discharges have been made from the Oak Ridge National Laboratory into the Clinch River and from the Savannah River Plant. At Windscale, where the principal radioactive effluents are fission products derived from the reprocessing of spent fuel elements, the discharges are made directly to the marine environment; whereas, the Hanford wastes are discharged to the Columbia River some several hundred miles from the Pacific Ocean.

Extensive environmental monitoring programs of these sites have resulted in the development of appropriate concentration factors for only a very few radionuclides (ref. 37, 38). The highest concentration factor reported by Preston and Jeffries (ref. 37) is 530 and applies only to zirconium and niobium. The same source lists the measured concentration factor for cesium in shoreline sand as 65, and ten times that figure for cesium in bottom silt.

However, Saddington suggests a CF of 100 for cesium in bottom sediment, (ref. 38) and although this is not an experimental value, for conservatism it is used here not only for cesium, but for all other radionuclides, except tritium. Tritium, released in the form of water, cannot concentrate in shoreline sand.

For gamma radiation, it was assumed that an individual lying on the shore would receive half the dose that he would receive if he were completely immersed in the sand.⁶ A calculation similar to the swimming dose calculation with sand as the infinite medium and with the concentration factor of 10^3 applied results in a maximum potential whole body dose rate of 9.41×10^{-3} (3.67×10^{-2}) mrem per hour for sunbathing along the lakeshore. Assuming an exposure period of 200 hours per year, the annual whole body dose to a sunbather would be 1.88 (7.35) mrem. This dose is calculated on the basis of annual average discharge concentrations, neglecting significant dilution occurring during transport to the nearest beach. The associated annual body surface exposure is 2.32 (9.57) mrem per year.

The remaining pathway to consider involves the transfer of activity ingested from lake water by a cow and secreted to the milk, which is then ingested by a human. In order to evaluate this route of exposure, it is assumed that the milking cattle obtain all of their drinking water from the

⁶The "geometry factor is taken to be 0.5 for both beta and gamma radiation from this source.

cooling tower blowdown flow. It can be estimated that the water requirement (exclusive of milk) for a cow is approximately 50 milliliters per kilogram of body weight (ref. 39). For a 100 pound (454 kg) animal this yields a water intake of about 22.7 liters/day.

Considering the lactating cow to produce about 18.2 liters of milk per day and taking the milk to be about 87% water, the total water intake for a lactating animal is estimated to be approximately 38.5 liters/day.

Of the radionuclides present in the water only the alkaline earths, alkali metals, and iodines need to be considered. The other elements are effectively discriminated against by the cow (ref. 40). The concentration in the milk is estimated as the product of the concentration in water, the cow's intake of water, and the coefficient of transfer. The coefficient of transfer is defined as the fraction of the daily intake secreted per liter of milk. The values used are listed below:

<u>Element</u>	<u>Coefficient of Transfer (ref. 40)</u>
Sr	.001
I	.01
Cs	.025
Ba	.002

The concentration of tritium in the secreted milk is assumed to be 0.87 times the concentration in the water ingested by the cow. The highest transfer coefficient is that of cesium (0.025). Since a person may be assumed to ingest a maximum of one liter of milk daily, on the average, the daily activity intake of Cs in microcuries will be given by:

$$I(\mu\text{Ci/day}) = C_w \left(\text{Cs} \frac{\mu\text{Ci}}{\text{cc}} \right) \left(38.5 \times 10^3 \frac{\text{cc}}{\text{day}} \right) \left(.025 \frac{\mu\text{Ci/liter}}{\mu\text{Ci/day}} \right) (1.0 \text{ liter/day}) = \left(962.5 \frac{\text{cc}}{\text{day}} \right) \times C_w \left(\frac{\mu\text{Ci}}{\text{cc}} \right)$$

where: $C_w \left(\text{Cs} \frac{\mu\text{Ci}}{\text{cc}} \right)$ is the concentration of cesium activity in the water ingested by the cow.

This can be compared to the daily cesium intake due to the man drinking 1.2 liter/day of the same water the cow drinks, which is given by:

$$I (\mu\text{Ci/day}) = (1200 \text{ cc/day}) \times C_w (\mu\text{Ci/cc})$$

This brief analysis demonstrates that the dose from the ingestion of milk from a cow drinking the lake water is somewhat less than the human dose from drinking the water directly. Since drinking one liter of milk per day might replace an equivalent amount of water intake in the individual liquid ingestion rate of 1.2 liters per day discussed earlier, the doses from this pathway are not analyzed further.

5. Summary of Maximum Individual Exposure From Liquid Releases:

Table 11A.4-4 presents the various exposure rates calculated to result from Davis-Besse's liquid activity releases. All exposure rates presented are based on an adult receptor. They are also based on the annual average discharge concentrations which is extremely conservative. Substantial dilution immediately upon discharge will effectively reduce the activity concentrations in the immediate vicinity of the site, thus reducing the potential exposure rates.

The maximum calculated individual exposure from all sources considered amounts to 6.20 (43.5) mrem per year to the liver. The total whole body exposure amounts to 4.09 (23.5) mrem per year.

11A.4.3 Population Radiation Exposure From Liquid Effluents

As a measure of the radiological well-being of the general public, the total integrated population exposure, in units of man-rem per year, is the most pertinent index of the impact of the liquid effluents. Population exposure may accrue from all the individual exposure pathways previously considered. However, only two of these pathways are considered significant in terms of population exposure from liquid effluents.

Detailed consideration is given to the population exposure derived from potable water intakes and fish ingestion.

Sources of exposure not considered significant in terms of population exposure are:

1. swimming in, sunbathing near, or walking along the lake;
2. ingestion of milk from cows drinking from the lake;
3. other exposure pathways not considered in terms of maximum individual exposure.

The exclusion of these three categories of radiation exposure is primarily based on their relative insignificance in comparison to the pathways considered.

1. Population Exposure from Water Ingestion:

Table 11A.4-5 presents a tabulation of the pertinent data necessary to estimate the total plant induced population exposure from water ingestion. The data includes the names of locations using Lake Erie as a source of potable water, the estimated number of people served, the distance of the intake from the Station, the applicable dilution factor, and the estimated per capita rate via water ingestion. Also given is the population exposure in man-rem, obtained by multiplying the per capita dose rate by the number of exposed individuals.

Since the population dose rate obtained is applicable only for 1970, an indication of the potential growth of this dose rate is desired. It can be assumed that the increase of population exposure from water ingestion with time will be approximately equal to that expected for population exposure due to noble gas releases. The justification for this assumption is that both routes of exposure depend directly on the number of people affected, and since the growth rates of the affected populations are expected to be roughly equal, the doses would increase in proportion with each other in time. Using this method to extrapolate the 1970 estimated dose rate, the following results are obtained:

Population Exposure – Water Ingestion

<u>Year</u>	<u>Man-rem per year</u>	
	<u>E.R.</u>	<u>D.B.R.</u>
1970	4.95×10^{-2}	8.85×10^{-2}
1990	7.19×10^{-2}	1.28×10^{-1}
2010	9.98×10^{-2}	1.78×10^{-1}

The above data also illustrate the relative stability of water ingestion exposure with changing defective fuel fractions. This is due to the dominance of tritium as the major dose producing radionuclide for this pathway. The Davis-Besse tritium release rate is not highly dependent on failed fuel fraction.

In general, all the data presented in Table 11A.4-2, with the exception of the dose rates, are taken from material presented in Chapter 2 of the USAR. However, at the Erie Industrial Park, which has a large daytime employee population (850) but only a small permanent population (less than ten), it has been necessary to assume that each non-resident individual consumes half his or her daily water requirement during the working hours (ref. 9). In the case of Camp Perry, used mainly for National Guard training activities, (ref. 32) an annual average population of about 1000 has been conservatively estimated from data presented in the FSAR.

Per capita dose rates have been obtained by assuming each individual to drink 1.2 liters daily of water obtained from Lake Erie. The radioactivity concentrations of the water ingested are assumed to be given by the discharge concentration divided by the effective dilution factor.

2. Population Exposure From Fish Ingestion:

An accurate estimate of the population exposure to be expected from fish ingestion cannot be derived from the data available. However, an upper limit to this exposure has been estimated based on the following assumptions:

- a. only the commercial fish catch in Ohio Fishing District 1 (from Toledo to Huron) contributes to the dose;
- b. one fourth of the total annual sport fish catch from the Ohio waters of Lake Erie contributes to the dose;
- c. of all the fish caught, only one third of the live weight is edible; and
- d. the applicable dilution factor is taken to be 5000 in all cases. Of the above four assumptions, only the last requires further explanation. The dilution factor chosen is applicable for a direction into which the surface discharge flows 10% of the time, at a distance of five miles. Thus, on the average, all fish contributing to the dose are conservatively assumed to be caught at a distance of only five miles from the Station discharge.

The 1970 commercial fish catch from Ohio Fishing District 1 totaled 2,380,543 pounds (ref. 41), over 29% of the total catch in all five Ohio districts for that year. One fourth of the estimated 1970 sport fish catch for all Ohio waters of Lake Erie amounts to 3,243,750 pounds (ref. 32). Thus the total annual fish catch assumed to be affected by the Station discharge is 5,624,293 pounds. Application of the assumed average effective dilution factor of 5000 yields a whole body population exposure of 1.47×10^{-2} (1.40×10^{-1}) man-rem, based on 1970 fishing statistics.

11A.4.4 Radiation Exposure Of Aquatic Bio-Systems

This section is directed toward an evaluation of the radiological impact of liquid effluents upon the various life forms inhabiting Lake Erie in the region of the site. The long range effect of continuous small releases of radioactive materials will be an incremental increase in the

normally occurring environmental levels. These increases will be most pronounced in situations where the radionuclides may tend to concentrate.

An organism inhabiting the lake near the site will be subject to plant contributed radiation from three different sources. Internally, radiations from radionuclides which may have concentrated within the organism will contribute to the total annual exposure. Externally, this organism will be affected by the radiation from the surrounding water body, and possibly also from the lake bottom where large concentration factors (CF's) may be effective in increasing radiation fields.

The receptor organisms at risk from this exposure vary in size, feeding habits, habitual location and other characteristics important in the determination of dose. Calculations have shown that the maximum radiation exposure received by any aquatic organism will be that delivered to life forms inhabiting the lake bottom where sediment concentrations of the released radionuclides may be orders of magnitude above those in the ambient water.

In the immediate vicinity of the discharge, stationary biota may be subject to annual average concentrations approaching in magnitude those projected for the discharge itself. This situation can only occur in a very small region owing to the rapid dilution. Using the CF for shoreline sands of 1000 (discussed earlier) for all discharged radionuclides except tritium, the maximum external dose rate from gamma radiation, to a receptor point at the lake bottom, is 82.5 (322.3) mrad per year. Almost all of this dose arises from the assumption of concentrations in bottom sediments a factor of 100 higher than annual average discharge concentrations. The gamma component is discussed separately because the beta component is much less significant, but more complex to deal with.

For very small organisms such as algae and microorganisms the beta dose can be considered a whole body dose, with even weak tritium betas contributing. In such a case the external beta dose at the lake bottom amounts to 21.1 (100.3) mrad per year. If the organism is large enough so that external H-3 causes no whole body exposure, the dose rate is reduced slightly to 19.0 (97.7) mrad per year.

The above external dose rates apply only to organisms normally dwelling at the lake bottom. For all organisms normally dwelling at sufficient height above the lake bottom (about 5 meters) so that they are shielded from the gamma radiation from material sorbed on the lake bottom, the external dose rates are reduced by about a factor of 500.

Using computer code DOSCAL, the annual internal exposure received by a fish living in waters containing discharge concentrations of radioactivity has been calculated to be 33.9 (160.9) mrad per year. Invertebrates are calculated to receive internal exposure amounting to 159.1 (292.5) mrad per year. Microorganisms are assumed to receive essentially no internal radiation.

The doses from all the sources considered are compiled and presented in Table 11A.4-6, "Radiation Exposure of Aquatic Biota." These doses were calculated using the annual average concentrations at discharge for all radionuclides and take no credit for any dilution of the discharge in the lake. These dose estimates are thus maximum potential exposure rates at the point of discharge.

In the evaluation of the magnitude of the dose rates presented, the conservatism of the calculational procedures and assumptions should be kept in mind. Also to be considered is the relative radiosensitivity of the aquatic population. Within the aquatic population, the more primitive forms, such as unicellular organisms and macrophytes, are the most radioresistant; fish and amphibians are more radiosensitive. Eggs and larvae in their early developmental

states are the most radiosensitive, and therefore, constitute the most critical group in the aquatic population (ref. 41).

A multitude of studies have been made to determine the effect on fish eggs and larvae chronically irradiated by immersion in low-level radionuclide solutions. An investigation by Polikarpov (ref. 42) has yielded illuminating results. It was found that the death rate of fresh water fish eggs was not increased even when immersed in a 10^{-6} $\mu\text{Ci/cc}$ solution of Sr-90, about 10^6 times the maximum expected discharge concentration estimated for Davis-Besse.

11A.4.5 Computational Methods For Doses Resulting From Liquid Effluents

11A.4.5.1 Whole Body and Body Organ Exposure from Water Ingestion

The ICRP has established the maximum average liquid intake of the standard man at 1.2 liters per day (ref. 9). The standard man is assumed to have a total body mass of 70 kilograms (ref. 9). Hence, the 1.2 liter daily liquid intake represents a daily renewal of 1.7% of the body mass.

The daily intake of the i th radionuclide, A_i ($\mu\text{Ci/day}$) from the hypothesized 1.2 liter daily ingestion of cooling tower blowdown water, can be found from the maximum expected discharge concentration, C_i ($\mu\text{Ci/cc}$), by the following formula:

$$A_i (\mu\text{Ci/day}) = 1200 (\text{cc/day}) \times C_i (\mu\text{Ci/cc})$$

For the whole body, and for all body organs of interest for the particular element, the equilibrium body (organ) burden, Q (μCi), can be computed from the daily intake and other information.

$$Q (\mu\text{Ci}) = \frac{A_{f_w}}{\lambda_{\text{eff}}} = \frac{A_{f_w} T_{\text{eff}}}{.693}$$

where:

A = intake rate, $\mu\text{Ci/day}$;

F_w = fraction of intake reaching organ of interest (ref. 9);

λ_{eff} = effective decay constant (radioactive decay plus biological decay constant, days^{-1})

T_{eff} = effective half-life, (ref. 9) days.

This can be combined with the general equation for determining dose rate, R mrem/year, to an organ from an organ burden Q :

$$R = \left(\sum E_f (\text{RBE}) n \frac{\text{MeV}}{\text{dis}} \right) \left(\frac{3.7 \times 10^4 \text{ dis}}{\text{sec} - \mu\text{Ci}} \right) \left(\frac{1}{M - \text{grams}} \right) Q \mu\text{Ci}$$

$$\left(\frac{1.6 \times 10^{-6} \text{ ergs}}{\text{MeV}} \right) \left(\frac{\text{gm} - \text{rem}}{100 \text{ ergs}} \right) \left(\frac{3.16 \times 10^7 \text{ sec}}{\text{year}} \right) \left(\frac{10^3 \text{ mrem}}{\text{rem}} \right)$$

$$R = \frac{1.87 \times 10^7 \sum E_f (\text{RBE}) n Q}{M} \text{mrem / year}$$

where:

$\sum E_f (\text{RBE}) n$ = effective energy, MeV, for the body organ of interest;

M = mass of body organ of interest.

Therefore, for the total body with a mass of 70,000 grams (ref. 9), the result is:

$$R = 3.845 \times 10^2 \Sigma E_f (RBE) n f_w T_{eff} A$$

For a given radionuclide, the values of the effective energy, effective half-life, fraction reaching the organ, and the organ mass vary with the particular organ under consideration. Table 11A.4-7 lists the ICRP values of mass and effective radius of the body considered in this report.

Using data given in the ICRP Report of Committee II (ref. 9) for values of $\Sigma E_f (RBE)$, n , T_{eff} and f_w the annual dose rate due to each ingested radionuclide was determined for the whole body and all the body organs of interest for the particular element, with the exception of the gastrointestinal tract.

Doses to the GI-tract were calculated by the method of "MPC ratioing" wherein the ratio of the actual concentration and the MPC (ref. 9) multiplied by the dose derived by the GI-tract from ingestion of water at MPC levels (15 rem/year).

Once the above outlined calculations are accomplished, the doses may be totaled. The dose to a particular organ is made up of the following three components:

1. the dose to the particular organ from all those radionuclides for which the particular organ dose is calculable;
2. The whole body dose from all other radionuclides except tritium; and
3. the dose from tritium.

The third component of the organ dose, the dose from tritium, depends on the organ. For tritium, released in the chemical form of water, it is possible from ICRP data to compute a whole body dose and a "body water" dose. The "body water" is a fictitious organ hypothesized for the purpose of establishing MPC values for tritium. In actuality water is distributed throughout the body, most body organs being comprised of roughly 80% water (ref. 39). The calculation of a dose to the body water from tritium assumes that 100% of the tritium disintegration energy is absorbed in the body water, not accounting for energy absorbed in non-aqueous tissues. The computational method used assumes that the body organ dose from tritium is equal to the body water dose, calculated using ICRP II criteria, times the fractional water content of the organ (grams water/gram organ). The mass water fractions used for the various organs are presented in Table 11A.4-8 of this document. The body water distribution as presented in Table 11A.4-8 and used in the NUS computer code DOSCAL appears to be quite adequate; the total body mass and total body water mass are very close to the values estimated in ICRP II (ref. 9).

11A.4.5.2 Whole Body and Body Organ Exposure from Fish Ingestion

Aquatic life forms will concentrate various elements within their bodies in proportion to the concentration of the element in the water in which they live. The ratio of the concentration of the element in the organism to the concentration in water is defined as the concentration factor, or CF. The CF generally depends on the element being concentrated, the species of organism, and the environment in which it lives. Maximum CF's for fish, shellfish and plants in seawater have been tabulated (ref. 34) and are used here to estimate the maximum activity of individual radionuclides in fish.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

For the purpose of estimating the maximum dose an individual might obtain from eating fish it was assumed that an individual might consume a maximum average of 50 grams of fish daily. Thus, the daily activity intake of the i th radionuclide would be given by:

$$A_i (\mu\text{Ci/day}) = C_i (\mu\text{Ci/cc}) (CF)_i 1.0 (\text{cc/gram}) (50 \text{ grams/day})$$

Once the daily activity intake is established, the equilibrium body (organ) burden and the concurrent whole body and organ doses are calculated in the same manner as presented in Subsection 11A.4.5.1.

11A.4.5.3 Models for Calculation of Dose to Aquatic Biota

External doses to biota were computed using the infinite sphere model as presented earlier. For biota not habitually at the lake bottom, such as free swimming fish, the exposure was assumed to arise from all directions with activity concentrations taken to be those estimated for the station discharge. For bottom dwelling organisms the dose from half the total solid angle was assumed to arise from the surrounding hemisphere of water. The dose from the remaining half of the solid angle was assumed to arise from a surrounding hemisphere of sediment. Concentrations of all radionuclides except tritium in sediment were conservatively assumed to be equal to 10^3 times the discharge concentrations.

Internal exposure was conservatively evaluated by assuming that all of the energy released in each gram of fish flesh was absorbed in the fish. The following formula was utilized.

$$D = C_w CF (3.7 \times 10^4) \bar{E} (1.6 \times 10^{-8}) (3.154 \times 10^7)$$

where:

D = the absorbed dose rate, rad/yr.

C_w = the concentration at discharge, $\mu\text{Ci/cc}$

CF = the applicable concentration factor, $(\mu\text{Ci/cc})/(\mu\text{Ci/cc})$

3.7×10^4 = disintegrations per second per microcurie

\bar{E} = the average beta and total gamma disintegration energy, MeV/dis

1.6×10^{-8} = gram-rads/MeV

3.154×10^7 = seconds per year.

Concentration factors used in the analysis were those given in "Concentration Factors of Stable Elements in Edible Aquatic Organisms", UCRL-50564, by Chapman, Fisher, and Pratt (ref. 34).

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.4-1

Estimated Liquid Effluents and Concentrations

	Expected Releases 0.1% Failed Fuel		Design Basis Releases 1.0% Failed Fuel	
	<u>Curies/Year</u>	<u>μCi/cc</u>	<u>Curies/Year</u>	<u>μCi/cc</u>
Corrosion Products:				
Cr 51	1.60-2	5.02-10	1.60-2	5.02-10
Mn 54	1.84-3	5.78-11	1.84-3	5.78-11
Fe 55	6.31-2	1.98-9	6.31-2	1.98-9
Fe 59	1.84-3	5.78-11	1.84-3	5.78-11
Co 58	9.55-2	3.00-9	9.55-2	3.00-9
Co 60	5.22-4	1.64-11	5.22-4	1.64-11
Zr 95	1.26-1	3.96-9	1.26-1	3.96-9
Fission Products, Gaseous:				
Ar 41	1.09-3	3.42-11	1.09-3	3.42-11
Kr 83m	2.82-4	8.86-12	2.82-3	8.86-11
Kr 85m	1.50-3	4.71-11	1.50-2	4.71-10
Kr 85	3.08-3	9.68-11	3.08-2	9.68-10
Kr 87	8.18-4	2.57-11	8.18-3	2.57-10
Kr 88	2.62-3	8.23-11	2.62-2	8.23-10
Xe 131m	2.20-3	6.91-11	2.02-2	6.91-10
Xe 133m	2.78-3	8.73-11	2.78-2	8.73-10
Xe 133	2.44-1	7.67-9	2.44	7.67-8
Xe 135m	9.05-4	2.84-11	9.05-3	2.84-10
Xe 135	4.97-3	1.56-10	4.97-2	1.56-9
Xe 138	5.00-4	1.57-11	5.00-3	1.57-10
Fission Products, Solid-Ionic:				
Rb 88	3.44-3	1.08-10	3.44-2	1.08-9
Sr 89	4.60-5	1.44-12	4.60-4	1.44-11
Sr 90	3.36-5	1.05-12	3.36-4	1.05-11
Sr 91	1.11-4	3.48-12	1.11-3	3.48-11
Sr 92	5.44-5	1.71-12	5.44-4	1.71-11
Y 90	6.17-5	1.94-12	6.17-4	1.94-11
Y 91	1.95-4	6.12-12	1.95-3	6.12-11
Mo 99	1.23-2	3.86-10	1.23-1	3.86-9
I 131	1.20-2	3.77-10	1.20-1	3.77-9
I 132	7.29-3	2.29-10	7.29-2	2.29-9
I 133	1.37-2	4.30-10	1.37-1	4.30-9
I 134	1.26-3	3.96-11	1.26-2	3.96-10
I 135	6.69-3	2.10-10	6.69-2	2.10-9
Cs 134	1.23-2	3.86-10	1.23-1	3.86-9
Cs 136	2.23-3	7.00-11	2.23-2	7.00-10
Cs 137	3.79-2	1.19-9	3.79-1	1.91-8
Cs 138	1.27-3	3.99-11	1.27-2	3.99-10
Ba 137m	3.55-2	1.11-9	3.55-1	1.11-8
Ba 139	1.97-4	3.18-12	1.97-3	6.18-11
Ba 140	4.91-5	1.54-12	4.91-4	1.54-11
La 140	3.95-5	1.24-12	3.95-4	1.24-11
Ce 144	3.47-5	1.09-12	3.47-4	1.09-11
Tritium:				
H 3	3.46+2	1.09-5	4.43+2	1.39-5

TABLE 11A.4-2

Internal Body Organ Exposure from Ingestion of Water and Fish

Body Organ	Exposure From Water Ingestion, rem/year		Exposure From Fish Ingestion, rem/year	
	E.R.	D.B.R.	E.R.	D.B.R.
Whole Body	6.44-4	1.15-3	1.56-3	1.50-2
Lungs	7.90-4	1.11-3	4.67-4	3.92-3
Liver	8.25-4	1.80-3	3.49-3	3.43-2
Spleen	8.39-4	1.64-3	2.71-3	2.64-2
Kidney	8.07-4	1.32-3	1.25-3	1.18-2
Bone	4.34-4	1.15-3	2.22-3	2.16-2
Muscle	8.45-4	1.66-3	2.75-3	2.68-2
Pancreas	8.26-4	1.39-3	1.57-3	1.50-2
Thyroid	1.27-3	5.78-3	1.59-3	1.52-2
Small Intestine	7.94-4	1.06-3	1.62-3	9.11-4
Upper Large Intestine	8.26-4	1.39-3	1.57-3	1.50-2
Lower Large Intestine	9.05-4	1.47-3	2.29-3	1.58-2

Notes about this table: Doses presented here are based on the undiluted annual average radioactivity concentration of the station discharge. Liquid exposure is based on an intake of 1.2 liters per day. Fish ingestion is postulated to be 50 grams per day. Other body organs for which doses were calculated but not found to substantially differ from those presented are the testes, ovaries, prostate, heart, brain, skin, and stomach .

The abbreviations E.R. and D.B.R. stand for maximum expected releases and design basis releases.

TABLE 11A.4-3

Concentration Factors for Effluent Radionuclides in Fish

<u>Element</u>	<u>Concentration Factor in Freshwater Fish *</u>
H	0.926
Ar	1
Cr	200
Mn	25
Co	500
Fe	300
Kr	1
Rb	2000
Sr	40
Y	100
Zr	100
Mo	100
I	1
Xe	1
Cs	1000
Ba	10
La	100
Ce	100

* All concentration factors used in this report for freshwater organisms are those compiled by Chapman, Fisher, and Pratt and presented in UCRL-50564, "Concentration Factors of Chemical Elements in Edible Aquatic Organisms" (ref. 34).

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.4-4

Summary of Maximum Individual Exposure Rates Due to Liquid Releases

Exposure Rates Presented Are Those Calculated For Adults, mrem/year								
<u>Exposure Pathway</u>	<u>Exposure from Expected Releases</u>				<u>Exposure from Design Basis Releases</u>			
	<u>Whole Body</u>	<u>Body Surface</u>	<u>Liver</u>	<u>Thyroid</u>	<u>Whole Body</u>	<u>Body Surface</u>	<u>Liver</u>	<u>Thyroid</u>
Water Ingestion	0.644	0.753	0.825	1.27	1.15	1.29	1.80	5.78
Fish Ingestion	1.56	1.57	3.49	1.59	15.0	15.0	34.3	15.2
Swimming	0.0038	0.0515	0.0038	0.0038	0.0147	0.0075	0.0147	0.0147
Sunbathing	1.88	2.32	1.88	1.88	7.35	9.57	7.35	7.35
Exposure Totals	4.09	4.69	6.20	4.74	23.5	25.9	43.5	28.3

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.4-5

Data Used to Estimate Population Water Ingestion Exposure (ref. 32)

Location	Number of People Affected	Distance of Intake from Station, mi.	Applicable Dilution Factor	Per Capita Dose Rate, mrem/yr.		Population Dose Rate, man-rem/yr.	
				E.R.	D.B.R.	E.R.	D.B.R.
1. Erie Industrial Park	430	3.6	1700	3.79-4	6.77-4	1.63-4	2.91-4
2. Camp Perry	1,000	4.6	2170	2.97-4	5.30-4	2.97-4	5.30-4
3. Port Clinton	7,600	8.6	4070	1.58-4	2.83-4	1.20-3	2.15-3
4. Oregon	16,000	12.0	7080	9.09-5	1.63-4	1.45-3	2.60-3
5. Toledo	510,000	12.0	7080	9.09-5	1.63-4	4.64-2	8.29-2
TOTALS	535,030					4.95-2	8.85-2

TABLE 11A.4-6

Radiation Exposure of Aquatic Biota

Maximum Potential Annual Radiation Exposure, mrad/year

Receptor	External Exposure		Internal Exposure		From All Sources	
	<u>E.R.</u>	<u>D.B.R.</u>	<u>E.R.</u>	<u>D.B.R.</u>	<u>E.R.</u>	<u>D.B.R.</u>
Sediment Level Micro-organisms	103.5	422.2	-0	-0	103.5	422.2
Bottom-feeding Fish	101.5	419.9	33.9	160.9	135.4	580.8
Bottom Dwelling Invertebrates	101.5	419.9	159.1	292.5	260.6	712.4
Free Floating Micro-organisms	2.27	3.49	0	0	2.27	3.49
Free Swimming Fish	0.203	0.839	33.9	160.9	34.1	161.7
Free Swimming Invertebrates	0.203	0.839	159.1	292.5	159.3	293.3

Note: Maximum potential doses are calculated using the annual average discharge concentrations.

TABLE 11A.4-7

ICRP Values for Organ Mass and Effective Radius (ref. 9)

<u>Body Organ</u>	<u>Organ Mass</u> <u>grams</u>	<u>Effective Radius</u> <u>cm</u>
Whole Body	70,000	30
Body Water	43,000	30
Lungs (ref. 39)	1,000	10
Liver	1,700	10
Spleen	150	7
Kidneys (ref. 39)	300	7
Bone	7,000	5
Muscle	30,000	30
Pancreas	70	5
Thyroid	20	3
Testes (ref. 39)	40	3
Ovaries (ref. 39)	8	3
Prostate	20	3
Stomach	250	10
Small Intestine	1,100	30
Upper Large Intestine	135	5
Lower Large Intestine	150	5

TABLE 11A.4-8

Estimate of Body Water Distribution by Organ in Standard Man

<u>Body Organ</u>	<u>Organ Mass</u>	<u>Organ Mass Water Fraction,</u>		<u>Organ Water</u>
	<u>grams⁽¹⁾</u>	<u>%</u>		<u>Content, grams</u>
		range ⁽²⁾	value used ^(3,5)	
DOSCAL organs:				
brain	1,500	70-85	77.5	1,163
lungs ⁽²⁾	1,000	78-80	79.0	790
heart	300	71-80	75.5	227
muscle	30,000	74-84	79.0	23,700
bone	7,000	30-44	37.0	2,590
skin	6,100	70-75	72.5	4,423
liver	1,700	73-77	75.0	1,275
spleen	150	76-81	78.5	118
kidneys ⁽²⁾	300	78-79	78.5	236
pancreas	70	none given	80.0	56
thyroid	20	none given	80.0	16
GI-tract	2,000	none given	80.0	1,600
testes ⁽²⁾	40	84.0	84.0	34
ovaries ⁽²⁾	8	80.5	80.5	6.4
prostate	20	82.5	82.5	17
Other organs:				
blood	5,400	77-81	70.6	4,244
fat	10,000	none given	-20 ⁽⁴⁾	-2,000
lymphatic tissue	700	none given	80.0	560
bone marrow	3,000	14.0	14.0	420
Totals:	-70,000			-43,000

Notes:

- 1) Values for organ mass are from ICRP
- 2) Range values for organ mass water fraction are from the Handbook of Biological Data (ref. 39), except as noted.
- 3) The arithmetic mean value of the range is used in DOSCAL, unless only a single value is given.
- 4) This value is from "The Physiology and Toxicology and Tritium in Man" by Pinson and Langham (ref. 20).
- 5) Where better data is not available, the organ is estimated to be 80% water.

11A.5.0 Dose Totals And Comparison With Federal Regulations And Natural Background

The doses calculated to occur due to the operation of the Davis-Besse Nuclear Power Station may best be brought into perspective by a comparison with exposure levels already present from naturally occurring background radiation.

The Environmental Protection Agency (ref. 4) has published estimates of statewide average whole body radiation doses from three sources; 1) cosmic radiation, 2) naturally occurring terrestrial radiation, and 3) internal radiation from naturally occurring radionuclides incorporated into the human body. For the state of Ohio, the respective annual doses from these three sources are 50 mrem, 65 mrem, and 26 mrem, respectively. Thus in Ohio, the average whole body dose from natural background radiation amounts to 140 mrem annually, slightly higher than the national average of 130 mrem per year (ref. 4).

In other states the annual whole body background dose is reported to range upwards from 100 mrem/year in Louisiana to 250 mrem/year in Colorado. The Special Studies Group has also found that medical exposure in the USA averaged 90 mrem/year in 1970 (ref. 4). The annual average exposure from television sets is reported to be 0.1 mrem. The total whole body population exposure occurring as a result of increased cosmic radiation during commercial air travel is estimated at 90,000 man-rem in the USA during 1969 (ref. 43).

Title 10 Part 20 of the Code of Federal Regulations limits the allowable exposure to individuals of the general population to 500 millirems, annually.

The per capita dose to any substantial section of the population is not allowed to exceed one third of this value, about 170 millirems annually. These legal guidelines together with the above exposure levels due to natural and man-made radiation provide a basis on which comparison of the incremental doses due to the operation of the Davis-Besse Nuclear Power Station may be adequately drawn. Such a comparison is presented in Tables 11A.5-1 and 11A.5-2.

In Table 11A.5-1 the calculated total adult exposures from gaseous and liquid releases are compiled and compared to applicable standards. For maximum expected release rates, the total whole body exposure from all considered pathways amounts to 5.38 mrem per year, only about 1% of the current legal limit imposed under 10CFR20. This exposure represents less than 4% of the estimated average whole body exposure from natural background over the entire State of Ohio. Critical organ exposure totals have not been found to be significantly larger than whole body exposure totals. Exposure totals due to design basis releases are roughly a factor of ten higher than those due to maximum expected releases, following the proportion of the assumed fraction of failed fuel.

The total population exposure due to both gaseous and liquid releases is presented in Table 11A.5-2. It can be seen there that the resulting exposures represent only very small fractions of the exposure permissible under 10CFR20, or the exposure due to naturally occurring background radiation.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.5-1

Maximum Individual Dose Totals and Comparison with Federal Regulations and Natural Background Exposure

exposures presented are for adults and are in mrem per year.

	Expected Releases				Design Basis Releases			
	Whole Body	Body Surface	Liver	Thyroid	Whole Body	Body Surface	Liver	Thyroid
I. Exposure Due to Davis-Besse								
Gaseous Exposure*	1.29	2.62	1.29	1.84	12.0	24.8	12.7	14.2
Liquid Exposure	4.09	4.69	6.20	4.74	23.5	25.9	43.5	28.3
Davis-Besse Totals	5.38	7.31	7.49	6.58	35.5	50.7	56.2	42.5
II. Allowable Exposure Under 10CFR20	500	500	500	500	500	500	500	500
III. Natural Background Exposure in Ohio	140	140	140	140	140	140	140	140

* It has been assumed that milk will be ingested from the nearest known cow, not at the site boundary.

Davis-Besse Unit 1 Updated Final Safety Analysis Report

TABLE 11A.5-2

Whole Body Population Dose Totals and Comparison with Federal Regulations and Natural Background Exposure

All exposures presented are for adults and are in man-rem/year.

	<u>Expected Releases</u>			<u>Design Basis Releases</u>		
	<u>Year 1970</u>	<u>Year 1990</u>	<u>Year 2010</u>	<u>Year 1970</u>	<u>Year 1990</u>	<u>Year 2010</u>
I. Exposure Due to Davis-Besse						
Gaseous exposure	1.456	1.909	2.522	14.56	19.09	25.22
Liquid exposure *	0.0642	0.0932	0.129	0.229	0.332	0.461
Davis-Besse Totals	1.52	2.00	2.65	14.8	19.4	25.7
II. Allowable Exposure Under 10CFR20	351,500	510,500	709,000	351,500	510,500	709,000
III. Exposure Due to Natural Background **	289,500	420,000	584,000	289,500	420,000	584,000

* Population exposure due to fish ingestion has been projected into the future using the same method as for water ingestion.

** Natural background exposure is based on 140 mrem per year per person.

11A.6.0 References

1. "Supplement to the Environmental Report", Volume I, Davis-Besse Nuclear Power Station.
2. Personal communications from Mr. Alex Hawes, Bechtel Corporation, Gaithersburg, Md., to Mr. Dan E. Martin, NUS Corporation, February 13, 1972; February 26, 1972; and March 18, 1972.
3. Chapter 10, Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation".
4. Special Studies Group, Office of Radiation Programs, EPA, "Estimates of Ionizing Radiation Doses in the United States, 1960-2000", 2nd printing, (1971).
5. Blatz, H., Radiation-Hygiene Handbook, McGraw-Hill (1959).
6. "Draft Environmental Statement for the Davis-Besse Nuclear Power Station", U.S. AEC, Directorate of Licensing, Docket No. 50-346 (November, 1972).
7. Personal communication from Mr. Ted Quay, Bechtel Corporation, Gaithersburg, Md., to Mr. Dan E. Martin, NUS Corporation, Rockville, Md. (March 23, 1973)
8. Hine, G.J., and G.L. Brownell (editors), Radiation Dosimetry, Academic Press, New York (1956).
9. ICRP Publication 2, "Report of Committee II on Permissible Dose for Internal Radiation", International Commission on Radiological Protection, Pergamon Press, New York (1959).
10. "Radioactive Waste Disposal Into the Sea", International Atomic Energy Agency, Safety Series No. 5, Vienna (1961).
11. Slade, D. H. (editor), "Meteorology and Atomic Energy." 1968." U.S. AEC, Division of Technical Information, TID-24190 (July 1968).
12. Crocker, G.R., et al., "Factors for the Calculation of Infinite Plane Exposure Rates from Gamma Radiation", Health Physics, Vol. 12, No. 9 (September 1966).
13. Turner, W.D., "The EXREM II Computer Code for Estimating External Doses to Populations from Construction of a Sea-Level Canal with Nuclear Explosives", Union Carbide Corporation, Computing Technology Center, Oak Ridge, Tennessee, CTC-8 (July 1969).
14. Beck, H.L. "Environmental Gamma Radiation from Deposited Fission Products", Health Physics, Vol. 12, pp.313-323 (March 1966).
15. Personal communication from Mr. Harold L. Beck, Health and Safety Laboratories, New York to Mr. Dan E. Martin, NUS Corporation, Rockville, MD. (March 19, 1973).

Davis-Besse Unit 1 Updated Final Safety Analysis Report

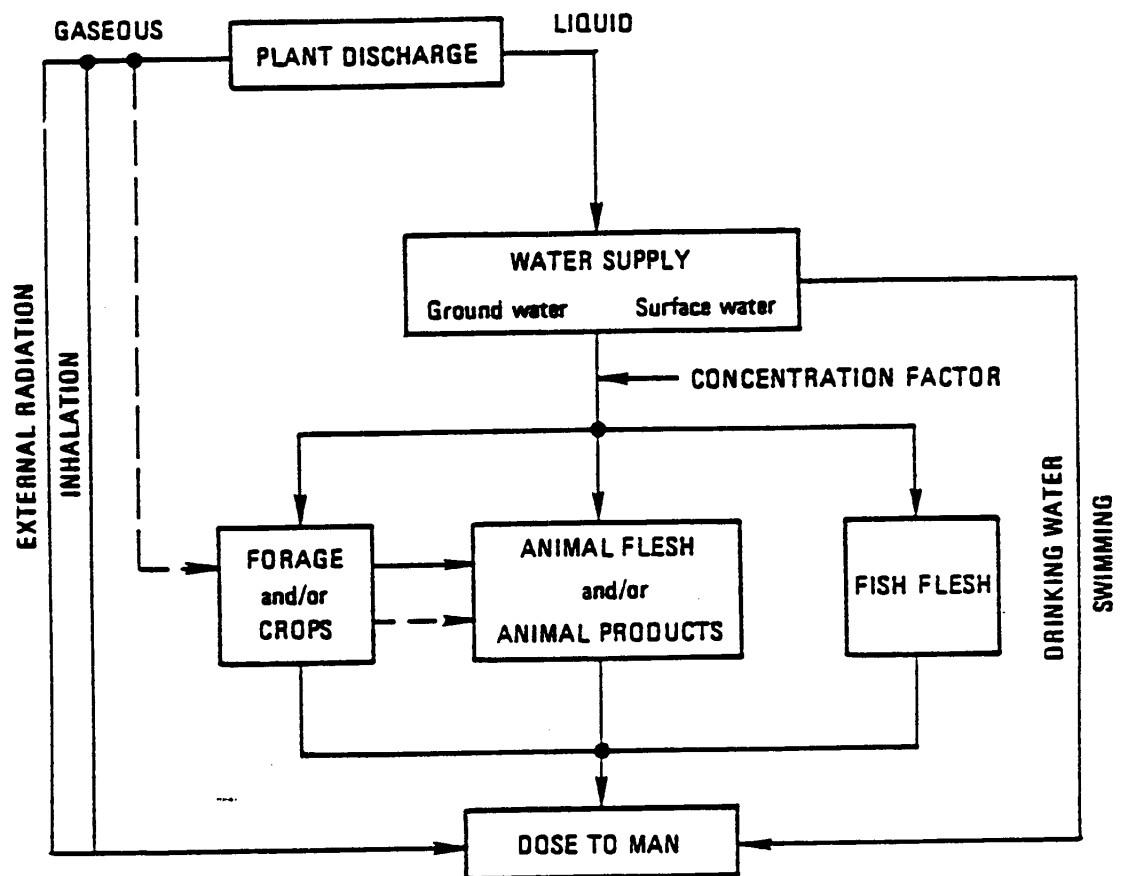
16. Bryant, P.M., "Derivation of Working Limits for Continuous Release Rates of Sr-90 and Cs-137 to Atmosphere in a Milk Producing Area", Health Physics, Vol. 12, pp. 1393-1405 (1966).
17. Gamertsfelder, Carl, "Additional Testimony of Dr. Carl Gamertsfelder Concerning Requests for Information and Data", U.S. AEC Rulemaking Hearings on the Proposed Appendix I to 10CFR50, Staff Exhibit 8 (April 1972).
18. Bryant, P.M., "Data for Assessments Concerning Controlled and Accidental Releases of I-131 and Cs-137 to Atmosphere", Health Physics, Vol. 17, No. 1 (July 1969).
19. Burnett, J.J., "A Derivation of the 'Factor of 700' for I-131", Health Physics, Vol. 18, No. 1 (January 1970).
20. Pinson, E.A., and W.H. Langham, "Physiology and Toxicology of Tritium in Man", Journal-of-Applied Physiology, Vol. 10, (January 1957 - May 1957).
21. Hytten, F.E., K. Taylor and N. Taggart, "Measurement of Total Body Fat in Man by Absorption of Kr-85", Clinical Science, Vol. 31 (1966).
22. Tobias, C.A., H.B. Jones, H.H. Lawrence and J.G. Hamilton, "The Uptake and Elimination of Krypton and other Inert Gases by the Human Body", Journal of Clinical Investigation 28:1375-1385.
23. Lassen, N.A., "Assessment of Tissue Radiation Dose in Clinical Use of Radioactive Inert Gases, With Examples of Absorbed Doses from $3(\text{sub})\text{H}$, $85(\text{sub})\text{Kr}$, and $133(\text{sub})\text{Xe}$ ", Minerva Nucleare Vol. 8 (1964).
24. Hendrickson, M.M., "The Dose From $85(\text{sub})\text{Kr}$ Released to the Earth's Atmosphere", in "Environmental Aspects of Nuclear Power Stations", IAEA Symposium, New York (August 1970).
25. DiNunno, J.J., et al., "Calculation of Distance Factors for Power and Test Reactor Sites", TID-14844, U.S. AEC (1962).
26. Garner, R. John, "Transfer of Radioactive Materials from the Terrestrial Environment of Animals and Man", Environmental Protection Agency, CRC Press (1972).
27. "Background Material for the Development of Radiation Protection Standards"; Federal Radiation Council, Report No. 5 (July 1964).
28. Garner, R.J., "A Mathematical Analysis of the Transfer of Fission Products to Cow's Milk", Health Physics, Vol. 13, pp. 295-212 (1967).
29. "Background Material for the Development of Radiation Protection Standards", Federal Radiation Council, Report No. 2 (September 1961).
30. Personal communication from Mr. Glenn Maddy, Sandusky County Extension Agent, Fremont, Ohio, and Mr. Lawrence Kolbicka, NUS Corporation, Rockville, Md. (March 21, 1973).

Davis-Besse Unit 1 Updated Final Safety Analysis Report

31. Personal communication from R.C.L. Olson, Bechtel Corporation, to A.W. De Agazio, NUS Corporation, (October 4, 1972).
32. "Final Safety Analysis Report", Chapter 2, Davis Besse Nuclear Power Station.
33. Fletcher, J.F., and W. L. Dotson, "HERMES, A Digital Computer Code for Estimating Regional Radiological Effects from the Nuclear Power Industry", U.S.,AEC, Hanford Engineering Development Laboratory (December 1971).
34. Chapman, W.H., et al., "Concentration Factors of Chemical Elements in Edible Aquatic Organisms", UCRL-50564 (December 1968).
35. Cowser, K.W., and W.S. Snyder, "Safety Analysis of Radionuclide Release to the Clinch River", ORNL-3721, Supplement 3 (1966).
36. Riley, F., "Fisheries of the United States, 1970", U. S. Department of Commerce, NOAA, NMFS-5600 (March 1971).
37. Preston, A. and D.F. Jeffereis, "Aquatic Aspects in Chronic and Acute Contamination Situations", in Environmental Contaminations by Radioactive Materials, IAEA Symposium, Vienna (1969).
38. Glueckauf, E. (editor), "Atomic Energy Waste", published by Interscience Publishers Inc., New York (1961).
39. Spector, W.S., "Handbook Biological Data", W.B. Saunders Company (1961).
40. Perez, L.J. and P.B. Robinson, "Estimates of Iodine-131 Thyroid Doses from Gross Beta Radioactivity in Milk Samples Collected at St. George, Utah, 1953", USDHEW, PHS. December 1967), SIB- 6705.
41. Donaldson, L.R. and R.F. Foster, "Effects of Radiation on Aquatic Organisms" in Chapter 10 of The Effects of Atomic Radiation on Oceanography and Fisheries, NAS NRC Publication 551 (1957).
42. Polikarpov, G.G., Radioecology of Aquatic Organisms, Reinhold, New York (1966).
43. Goldman, M.I., "Statement by Dr. Morton I. Goldman on Behalf of the Consolidated Utility Group in the U. S. Atomic Energy Commission Rulemaking Hearing on Proposed Appendix I", Germantown, Maryland (March 17, 1972).
44. Langham, W.H., and Pinson, E.A., "Physiology and Toxicology of Tritium in Man", Journal of Applied Physiology, Volume 10 (January 1957-May 1957).
45. Lederer, C.M., et al., "Table of Isotopes", Sixth Edition, Wiley (1967).
46. "Radiation Health Handbook", Public Health Service, U.S. DHEW, (Revised January, 1970).
47. Beck, H.L., J. DeCampo, C. Gogalak, "In Situ Ge (Li) and Na I (T1) Gamma Ray Spectrometry", Health and Safety Laboratory, HASL-258 (September 1972).

Davis-Besse Unit 1 Updated Final Safety Analysis Report

48. Rohwer, P.S., S.V. Kaye, "Age-Dependent Models for Estimating Internal Dose in Feasibility Evaluations of Plowshare Events", Oak Ridge National Laboratory, ORNL-TM-2229 (1968).
49. FRC Report No. 2, "Background Material for the Development of Radiation Protection Standards", (September 1961).



DAVIS-BESSE NUCLEAR POWER STATIO
EXPOSURE PATHWAYS
FIGURE 11A.1-1

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