CHAPTER 11

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CHAPTER 11

RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

Historical text, tables, values and parameters are identified as such and provide the historical basis for facility design.

Radionuclides generated in the reactor core, neutron activation of nuclides in the reactor coolant system (RCS), and the air surrounding the reactor vessel are the sources of all radioactivity that occurs in the process streams of the various radioactive systems.

Sections 11.2 and 11.3 describe the liquid and gaseous waste management systems and give annual liquid and gaseous radioactive releases for expected and design cases.

11.1.1 Radionuclide Inventory in the Core

The specific activity of fission products in the core is calculated using the computer program ACTIVITY 2. This program calculates the contribution from parent, daughter, and granddaughter isotopes by solving the following differential equations:

1. First order nuclides:

$$\frac{dN_{c_i}(t)}{dt} = f\alpha_i - \left(\lambda_i + h\gamma_i + \beta_i\right) N_{c_i}(t) \qquad (11.1\text{-}1)$$

2. Second order nuclides:

$$\frac{dN_{c_j}(t)}{dt} = F\alpha_j + \lambda_i f_{ij} N_{c_i}(t) - (\lambda_j + h\gamma_j + \beta_j) N_{c_j}(t) \qquad (11.1\text{-}2)$$

3. Third order nuclides:

$$\frac{dN_{c_k}(t)}{dt} = F\alpha_k + \lambda_i f_{ik} N_{c_i}(t) + \lambda_j f_{jk} N_{c_j}(t) - \left(\lambda_k + h\gamma_k + \beta_k\right) N_{c_k}(t) \qquad (11.1\text{-}3)$$

where:

i,j,k = First, second, and third order nuclide parameters

$N_{c_i}(t)$ = Concentration of nuclide i per fuel region at time t (atoms per reqion)

t = Time (seconds)

f = Fission rate (fissions per second in fuel region)

α_i = Fission yield for isotope i (atoms per fission)

λ_i = Decay constant for isotope i (seconds^{-1})

γ_i = Escape rate coefficient (seconds^{-1})

β_i = $\sigma_{a_i} \phi_{th}$ = Burnup rate (seconds^{-1})

f_{ij} = Branching fraction from i to j

h = Fraction of failed fuel.

The program has a basic library of 167 nuclides with a capability of 200 nuclides. Library data include decay scheme information, production information, and decay gamma spectra in seven energy groups. Input data include time intervals, initial source inventory in the fuel, neutron flux, and power level. The program output describes the system analyzed, as well as the operating history, the activities, and associated gamma spectral information for the input time interval.

The calculation of the core iodine fission product inventory is consistent with the inventories given by DiNunno (et al 1962). The fission product inventories are calculated using the appropriate data from Meek and Rider (1974), Lederer (et al 1968), Nucleonics Handbook of Nuclear Research and Technology (1963), Goldberg (et al 1966), and Perkins (1963). The core iodine and noble gas fission product inventories are presented in Table 11.1-1 (historical) based on continuous operation of the unit at 2,766 MWt. These inventories represent the values used by the original license application to establish plant design and are retained for historical purposes.

In 1999, a new core inventory, including primary and secondary coolant design activity concentrations was developed using updated, conservative facility design and operating parameter values as analysis inputs.

The isotopic inventory in the core and coolant were again revised as part of implementation of containment conversion to atmospheric operation, and extended power uprate. The analysis was performed assuming 518 days of continuous operation at a core thermal power of 2918 MWt, and using the isotope generation and depletion computer code ORIGEN-S, utilizing the Control Module SAS2H of the ORNL SCALE 4.3 computer code. The uprated core inventory and the associated design and technical specification coolant inventory are provided in Tables 15.0-7a, 15.0-8b and 15.0-8c, respectively.

Fuel assembly source terms for shielding design are calculated using the ACTIVITY 2 computer code and are presented in Chapter 12.

Fuel element heat loadings and stresses, as well as fuel operating experience, are presented in Chapter 4.

11.1.2 Radionuclide Inventory in Fuel Element Gap

The gap activity is that fraction of the gaseous activity in the core that diffuses to the fuel gaps.

For the fuel handling accident, gap activity fractions are from Regulatory Guide 1.183 (0.1 for Kr-85, 0.08 for I-131 and 0.05 for others). Table 15.7-6a presents the associated core gap activities. The gap activity used for the Loss of Coolant Accident (LOCA), the Locked Rotor Accident (LRA) and the Control Rod Ejection Accident (CREA) are also based on Regulatory Guide 1.183, and were performed assuming an uprated core thermal power of 2918 MWt. The core inventory used to establish the gap activity for these accidents is provided in Table 15.0-7a.

11.1.3 Primary Coolant Equilibrium Activities

11.1.3.1 Fission Product Activities

The design fission product activities with 1 percent failed fuel in the reactor coolant are also calculated with the ACTIVITY 2 program. The following differential equations are used:

1. First order nuclides:

$$\frac{dN_{w_i}}{dt} = \frac{hn\gamma_i}{V_w} N_{c_i}(t) - \left(\lambda_i + \frac{PF_{EQ_i} Q_1}{V_w} + \beta_i \frac{T_1}{T_2}\right) N_{w_i}(t) \tag{11.1-4}$$

2. Second order nuclides:

$$\frac{dN_{w_j}}{dt} = \frac{hn\gamma_j}{V_w} N_{c_j}(t) + \lambda_i f_{ij} N_{w_i}(t)$$

$$-\left(\lambda_j + \frac{PF_{EQ_j} Q_1}{V_w} + \beta_j \frac{T_1}{T_2}\right) N_{w_j}(t) \tag{11.1-5}$$

3. Third order nuclides:

$$\frac{dN_{w_k}}{dt} = \frac{hn\gamma_k}{V_w} N_{c_k}(t) + \lambda_j f_{jk} N_{w_j}(t)$$

$$-\left(\lambda_k + \frac{PF_{EQ_k} Q_1}{V_w} + \beta_k \frac{T_1}{T_2}\right) N_{w_k}(t) \qquad (11.1\text{-}6)$$

where:

$N_{w_i}(t)$ = Concentration of nuclide i in the main coolant at time t (atoms/cm^3)

n = Total number of fuel regions

β_i = $\sigma_{a_i} \phi_{th}$ = Burnup rate (seconds^{-1})

V_w = Volume of main coolant (cm^3)

PF_{EQ_i} = Equivalent purification factor (fraction) for i

T_1 = Coolant residence time in core (seconds)

T_2 = Coolant circulation time (seconds)

Q_1 = Equivalent flow into purification stream (cm^3/sec)

= $Q_p \dfrac{\rho_p}{\rho_w}$

Q_p = Actual flow entering purification stream at coolant loop density (cm^3/sec)

ρ_w = Density of the main coolant (g/cm^3)

ρ_p = Density of the purification flow (g/cm^3)

In addition to the library and input data described in Section 11.1.1, the following information is required for the calculation of primary coolant activities: 1) library data which include purification factors for typical demineralizers and fuel escape rate coefficients, and 2) input data which includes the fraction of fuel defects and the density of reactor coolant. The chemical and volume control system (Section 9.3.4) may also be described in terms of flow rates, densities, and operating intervals.

The RCS design basis equilibrium radioactivities presented in Table 11.1-2 (historical) are based on continuous operation of the core at 2,766 MWt with 1.0 percent fuel cladding defects. These reactor coolant activities which are based on parameters given in Table 11.1-3 (historical) represent the values utilized by the original license application to establish the plant shielding design, and to develop the original licensing basis source terms required for the evaluation of the liquid, gaseous, and solid waste systems.

In these original calculations, the fuel rods with cladding defects were assumed to be present in the initial core and uniformly distributed throughout the core. Thus, the fission product escape rate coefficients were based upon average fuel temperature. The calculations were performed using the average temperature of the reactor coolant. The reactor coolant density correction of 1.4 was made in order to obtain the correct radionuclide concentrations downstream of the letdown heat exchanger.

The RCS design basis equilibrium concentrations for isotopes not calculated by the ACTIVITY 2 program were assumed to be the NUREG-0017 (U.S. Nuclear Regulatory Commission (USNRC) 1976) expected values multiplied by three, or the values suggested by the nuclear steam system supplier, whichever is higher. Also included in Table 11.1-2 are the original licensing basis expected equilibrium concentrations for the RCS. These results are based on measured and calculated concentrations reported in NUREG-0017 (USNRC 1976) and the parameters listed in Table 11.1-3.

The expected reactor coolant activities were used to develop the source terms for gaseous and liquid effluents in Sections 11.2 and 11.3.

Both expected and design reactor coolant activities were used to evaluate the ventilation design in Chapter 12.

Power uprate to a core power level of 2900 Mwt represents a change from the original design basis. The design and technical specification primary coolant source terms estimated for the power uprate are presented in Table 15.0-8b and 15.0-8c, respectively. The assessment of impact of power uprate, on adequacy of existing plant shielding, and on radwaste effluents, is discussed in Section 12.3 and Sections 11.2A and 11.3.3, respectively.

11.1.3.2 Tritium Activity

There are two principal contributors to tritium production within the pressurized water reactor (PWR) system: the ternary fission source and the dissolved boron in the reactor coolant. Additional contributions are made by Li-6, Li-7, and deuterium in the reactor water. Tritium is also produced by nuclear reactions with boron contained in burnable poison rods. These rods were used only during the first operating cycle. Table 11.1-4 (Historical) presents tritium production from different sources.

11.1.3.2.1 Fission Source

This tritium is formed within the fuel material and may:

1. Remain in the fuel rod uranium matrix.
2. Diffuse into the cladding and become fixed there, as zirconium tritide.
3. Diffuse through the cladding and be released into the primary coolant.

4. Be released to the coolant through microscopic cracks or failures in the fuel cladding.

Previous Westinghouse fuel design has conservatively assumed that the ratio of fission tritium released into the coolant to the total fission tritium formed was approximately 0.30 for Zircaloy clad fuel. The operating experience at the R.E. Ginna Plant of the Rochester Gas and Electric Company, and at other operating PWRs using Zircaloy clad fuel, has shown that the tritium release through the Zircaloy fuel cladding is less than the earlier estimates. Consequently, the release fraction has been revised downward from 30 percent to 10 percent based on these data (Westinghouse 1974).

11.1.3.2.2 Control Rod Source

There are no reactions in this absorber material which produce tritium, thus the control rods are not a source of tritium.

11.1.3.2.3 Boric Acid Source

A direct contribution to the reactor coolant tritium concentration is made by neutron reaction with the boron in solution. The concentration of boric acid varies with core life and load follow so that this is a steadily decreasing source during core life. The principal boron reactions are the B-10 (n, 2α) H-3 and B-10 (n,α) Li-7 (n, nα) H-3 reactions.

11.1.3.2.4 Burnable Shim Rod Source

These rods were in the core only during the first operating cycle and their potential tritium contribution was only during this period.

11.1.3.2.5 Lithium and Deuterium

Lithium and deuterium reactions contribute only minor quantities to the tritium inventory, as shown in Table 11.1-4. These sources are due to the activation of the lithium and deuterium in the RCS as they pass through the reactor. Li-6 is essentially excluded from the system by utilizing 99.9 percent Li-7.

11.1.3.2.6 Design Bases

The design intent is to reduce the tritium sources in the RCS to a practical minimum in order to permit longer retention of the reactor coolant within Beaver Valley Power Station Unit-2 (BVPS-2) without compromising operator exposures. Reduction of source terms is provided by utilizing control rods composed of materials that do not contribute to tritium production and the determination that the quantity of tritium released from the fuel rods with Zircaloy cladding is less than originally expected.

11.1.3.2.7 Design Evaluation

Table 11.1-4 compares a typical design basis tritium production that has been used in the past to establish system and operational requirements of BVPS-2, and present expected values. There are two principal contributors to the tritium production: ternary fission source and the dissolved boron in the reactor coolant.

Review of plants operating with Zircaloy-clad fuel indicates that less than 10 percent of the fission produced tritium will be released to the reactor coolant. The operating levels of boron concentration during start-up will be approximately 1,100 to 1,200 ppm boron. Burnable poison rods in the core also contained boron which contributed some tritium to the coolant, but only during the first cycle. Operating plants with Zircaloy-clad cores have reported low tritium concentrations in the reactor coolant system after considerable periods of operation. This clearly indicates that the present design tritium source term is conservative.

This quantity of tritium becomes uniformly distributed in the RCS, the primary grade water system, and the boron recovery system. During refueling operations, the tritium is further diluted when the refueling canal is filled from the borated refueling water storage tank.

The expected tritium concentration in the primary coolant is based on the value provided in NUREG-0017 (USNRC 1976). For the purpose of radioactive liquid waste analysis, it is assumed that 50 percent of the activity produced and entering the coolant in 1 year is released in the radioactive liquid waste system. The remaining 50 percent is released in the radioactive gaseous waste system and ventilation effluents.

The design tritium concentration in the primary coolant is selected to allow limited access to the containment during normal operation. Chemistry and radiation protection procedures control tritium concentrations to design levels and also allow for continuous containment access during refueling with operation of the containment ventilation system.

The following conclusions have been reached:

1. The tritium levels in plants operating with Zircaloy clad cores is lower then previous design predictions.

2. The tritium source is reduced by using control rods composed of materials that do not contribute to tritium production.

11.1.3.3 Corrosion Products

Corrosion products in the reactor coolant become activated when they pass through the core. The most important corrosion products are Cr-51, Mn-54, Mn-56, Fe-55, Fe-59, Co-58, and Co-60. The corrosion product activity is dependent on many factors, including the type of plant and the materials of construction. The mass transport process is complex and stochastic, and calculative methods to predict corrosion product activity accurately have not been successfully correlated with operational data (Barlett 1969). Analytical predictions of the corrosion product activity levels are approximations. Therefore, design corrosion product levels utilized in the original plant design are assumed to be the NUREG-0017 (USNRC 1976) expected values multiplied by three, or the values measured at operating reactors (Westinghouse 1974), whichever is greater.

Table 11.1-2 gives the corrosion product activities in the reactor coolant utilized in the original plant design.

Note that the design corrosion product levels in the reactor coolant estimated for the power uprate and presented in Table 15.0-8b are assumed to be NUREG 0017 Revision 1 expected values multiplied by three.

11.1.3.4 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 in the primary coolant by neutron irradiation in the core. Reactions with all three oxygen isotopes, O-16 (99.76 percent), O-17 (0.037 percent), and O-18 (0.204 percent) result in the production of N-16.

Nitrogen-16 emits high energy gammas in 75 percent of the disintegrations (70 percent at 6.13 MeV and 5 percent at 7.11 MeV).

Table 11.1-5 (Historical) gives the N-16 activity at various points in the RCS.

11.1.4 Radioactivity in the Secondary Side

The concentrations of principal radioisotopes in the secondary side of the steam generators are listed for both the design and expected cases in Table 11.1-6 (Historical) for liquid and Table 11.1-7 (Historical) for steam. These secondary coolant activities which are based on parameters given in Table 11.1-3 represent the values utilized by the original license application to establish the plant shielding design, and to develop the original licensing basis source terms required for the evaluation of the liquid, gaseous, and solid waste systems. Based on parameters in Table 11.1-3, the design results for fission and activation products were calculated with the computer program IONEXCHANGER, which solves the following differential equations for secondary liquid activities:

1. First order nuclides:

$$\frac{dN_i}{dt} = R_i - \left(\lambda_i + \frac{Q_B}{V}\right) N_i(t) \tag{11.1-7}$$

2. Second order nuclides:

$$\frac{dN_j}{dt} = R_j + \lambda_i f_{ij} N_i(t) - \left(\lambda_j + \frac{Q_B}{V}\right) N_j(t) \tag{11.1-8}$$

3. Third order nuclides:

$$\frac{dN_k}{dt} = R_k + \lambda_i f_{ik} N_i(t) + \lambda_j f_{jk} N_j(t) - \left(\lambda_k + \frac{Q_B}{V}\right) N_k(t) \tag{11.1-9}$$

where:

N_i = Number of atoms of nuclide i (atoms)

R_i = Feed rate of nuclide i (atoms/sec)

λ_i = Radioactive decay constant for nuclide i (seconds^{-1})

f_{ij} = Branching fraction from i to j

V = Volume of steam generator liquid (cm^3)

Q_B = Steam generator radioactivity removal rate (cm^3/sec)

t = Time (seconds)

Secondary side steam activities are obtained by using the following relationship:

$$A_i = P_i A_{O_i} \tag{11.1-10}$$

where:

A_i = Steam equilibrium activity for isotope i (μCi/g)

A_{O_i} = Liquid equilibrium activity for isotope i (μCi/g)

P_i = Partition factor (liquid to steam) for isotope i

The expected secondary liquid and steam activities are based on the concentrations reported in NUREG-0017 (USNRC 1976) and the parameters in Table 11.1-3.

Power uprate to a core power level of 2900 Mwt represents a change from the original design basis. The design and technical specification secondary coolant source terms estimated for the power uprate are presented in Table 15.0-8b and 15.0-8c, respectively. The assessment of impact of power uprate, on adequacy of existing plant shielding, and on radwaste effluents, is addressed in Section 12.3 and Sections 11.2 through 11.4, respectively.

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Tables for Section 11.1

TABLE 11.1-1 [HISTORICAL]

IODINE AND NOBLE GAS INVENTORY IN REACTOR CORE AND FUEL ROD GAPS*

Isotope	Core (Ci)	Fraction of Core Activity in Gap	Fuel Rod Gap Activity (Ci)
I-131	6.9×10^{7}	0.1	6.9×10^{6}
I-132	9.9×10^{7}	0.1	9.9×10^{6}
I-133	1.6×10^{8}	0.1	1.6×10^{7}
I-134	1.8×10^{8}	0.1	1.8×10^{7}
I-135	1.4×10^{8}	0.1	1.4×10^{7}
Kr-83m	1.2×10^{7}	0.1	1.2×10^{6}
Kr-85m	3.0×10^{7}	0.1	3.0×10^{6}
Kr-85	6.8×10^{5}	0.1	6.8×10^{4}
Kr-87	5.9×10^{7}	0.1	5.9×10^{6}
Kr-88	8.3×10^{7}	0.1	8.3×10^{6}
Kr-89	1.1×10^{8}	0.1	1.1×10^{7}
Xe-131m	4.2×10^{5}	0.1	4.2×10^{4}
Xe-133m	3.7×10^{6}	0.1	3.7×10^{5}
Xe-133	1.6×10^{8}	0.1	1.6×10^{7}
Xe-135m	4.2×10^{7}	0.1	4.2×10^{6}
Xe-135	4.1×10^{7}	0.1	4.1×10^{6}
Xe-137	1.4×10^{8}	0.1	1.4×10^{7}
Xe-138	1.4×10^{8}	0.1	1.4×10^{7}

NOTES:

* The information presented in the above table is based on 650 days of operation at 2,766 MWt. It was developed in support of the original license and is considered historical.

TABLE 11.1-2 [HISTORICAL]

REACTOR COOLANT EQUILIBRIUM CONCENTRATIONS

Nuclide	Design (μCi/g)	Expected (μCi/g)
Noble Gases		
Kr-83m	4.3×10^{-1}	2.0×10^{-2}
Kr-85m	2.1	8.9×10^{-2}
Kr-85	1.1×10^{1}	1.2×10^{-1}
Kr-87	1.2	6.2×10^{-2}
Kr-88	3.2	1.8×10^{-1}
Kr-89	1.0×10^{-1}	5.8×10^{-3}
Xe-131m	1.1×10^{-1}	6.0×10^{-3}
Xe-133m	3.1	4.2×10^{-2}
Xe-133	2.7×10^{1}	1.7
Xe-135m	1.1	1.5×10^{-2}
Xe-135	3.3	2.1×10^{-1}
Xe-137	1.6×10^{-1}	9.9×10^{-3}
Xe-138	6.8×10^{-1}	5.0×10^{-2}
Subtotal	5.3×10^{1}	2.5
Halogens		
Br-83	8.5×10^{-2}	5.5×10^{-3}
Br-84	3.9×10^{-2}	3.0×10^{-3}
Br-85	5.4×10^{-3}	3.5×10^{-4}
I-130	6.9×10^{-3}	2.3×10^{-3}
I-131	2.5	2.9×10^{-1}
I-132	8.8×10^{-1}	1.1×10^{-1}
I-133	4.0	4.2×10^{-1}
I-134	5.5×10^{-1}	5.4×10^{-2}
I-135	2.1	2.1×10^{-1}
Subtotal	1.0×10^{1}	1.1
Corrosion Products		
Cr-51	6.0×10^{-3}	2.0×10^{-3}
Mn-54	9.9×10^{-4}	3.3×10^{-4}
Mn-56	2.9×10^{-2}	-
Fe-55	5.1×10^{-3}	1.7×10^{-3}
Fe-59	3.3×10^{-3}	1.1×10^{-3}
Co-58	5.1×10^{-2}	1.7×10^{-2}
Co-60	6.3×10^{-3}	2.1×10^{-3}
Subtotal	1.0×10^{-1}	2.4×10^{-2}

TABLE 11.1-2 (Cont)

Nuclide	Design (μCi/g)	Expected (μCi/g)
Other		
Rb-86	$2.7x10^{-4}$	$9.1x10^{-5}$
Rb-88	3.2	$2.3x10^{-1}$
Rb-89	$9.9x10^{-2}$	-
Sr-89	$4.1x10^{-3}$	$3.7x10^{-4}$
Sr-90	$1.6x10^{-4}$	$1.1x10^{-5}$
Sr-91	$1.9x10^{-3}$	$7.2x10^{-4}$
Sr-92	$7.3x10^{-4}$	-
Y-90	$2.0x10^{-4}$	$1.3x10^{-6}$
Y-91m	$1.1x10^{-3}$	$4.2x10^{-4}$
Y-91	$6.5x10^{-4}$	$6.8x10^{-5}$
Y-92	$7.1x10^{-4}$	-
Y-93	$3.3x10^{-4}$	$3.8x10^{-5}$
Zr-95	$6.7x10^{-4}$	$6.3x10^{-5}$
Nb-95	$6.8x10^{-4}$	$5.3x10^{-5}$
Mo-99	3.3	$9.0x10^{-2}$
Tc-99m	1.8	$5.4x10^{-2}$
Ru-103	$3.3x10^{-4}$	$4.7x10^{-5}$
Ru-106	$3.1x10^{-5}$	$1.1x10^{-5}$
Rh-103m	$3.3x10^{-4}$	$5.2x10^{-5}$
Rh-106	$3.1x10^{-5}$	$1.2x10^{-5}$
Te-125m	$9.3x10^{-5}$	$3.1x10^{-5}$
Te-127m	$1.9x10^{-3}$	$3.0x10^{-4}$
Te-127	$9.9x10^{-4}$	$9.4x10^{-4}$
Te-129m	$3.7x10^{-2}$	$1.5x10^{-3}$
Te-129	$2.1x10^{-2}$	$1.8x10^{-3}$
Te-131m	$2.2x10^{-2}$	$2.7x10^{-3}$
Te-131	$1.1x10^{-2}$	$1.3x10^{-3}$
Te-132	$2.6x10^{-1}$	$2.9x10^{-2}$
Te-134	$3.0-10^{-2}$	-
Cs-134	$2.6x10^{-1}$	$2.7x10^{-2}$
Cs-136	$1.5x10^{-1}$	$1.4x10^{-2}$
Cs-137	1.6	$1.9x10^{-2}$
Cs-138	$9.3x10^{-1}$	-
Ba-137m	1.5	$1.9x10^{-2}$
Ba-140	$4.2x10^{-3}$	$2.3x10^{-4}$
La-140	$1.4x10^{-3}$	$1.6x10^{-4}$
Ce-141	$6.6x10^{-4}$	$7.4x10^{-5}$
Ce-143	$5.0x10^{-4}$	$4.3x10^{-5}$
Ce-144	$4.7x10^{-4}$	$3.5x10^{-5}$
Pr-143	$6.4x10^{-4}$	$5.3x10^{-5}$
Pr-144	$4.7x10^{-4}$	$3.8x10^{-5}$
Np-239	$3.9x10^{-3}$	$1.3x10^{-3}$
Subtotal	$1.2x10^{1}$	$4.9x10^{-1}$

TABLE 11.1-2 (Cont)

Nuclide	Design (μCi/g)	Expected (μCi/g)
H-3	3.5	1.0
Subtotal	3.5	1.0
Total (excluding H-3)	7.5×10^{1}	4.1
Total (including H-3)	7.9×10^{1}	5.1

The information presented in the above table is based on 650 days of operation at 2,766 MWt. It was developed in support of the original license and is considered historical.

TABLE 11.1-3 [HISTORICAL]

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT, SECONDARY SIDE LIQUID, AND SECONDARY SIDE STEAM FISSION AND ACTIVATION PRODUCT ACTIVITY

Parameter	Value
Core thermal power, MWt (licensed power level, plus 4%)	2,766
Fuel cladding defects, percent of core thermal power (design)	1.0
Fission product escape rate coefficients:	
Noble gas nuclides (sE-1)	6.5×10^{-8}
Br, Rb, I, and Cs nuclides (sE-1)	1.3×10^{-8}
Te nuclides (sE-1)	1.0×10^{-9}
Mo nuclides (sE-1)	2.0×10^{-9}
Sr and Ba nuclides (sE-1)	1.0×10^{-11}
Y, La, Ce, Pr, Nb, Zr, Rh, Ru nuclides (sE-1)	1.6×10^{-12}
Reactor coolant liquid mass (lb) (without pressurizer)	4.2×10^{5}
Reactor coolant liquid volume (ft^3)	9,400
Reactor coolant full power average temperature (°F)	580
Purification flow rate (normal) (gpm)	60
Mixed bed demineralizer decontamination factors:	
Noble gases, N-16, H-3	1.0
Cations	
Design: Cs, Mo, Y	1.0
Expected: Cs, Rb	2.0
All other nuclides including activation products	10.0
Cation bed demineralizer decontamination factors:	
Noble gases, N-16, H-3	1.0
Cations	
Design: Cs, Y, Mo	10.0
Expected: Cs, Rb	10.0

TABLE 11.1-3 (Cont)

Parameter	Value
Cation bed demineralizer decontamination factors:	
All other nuclides, including activation products	
Design	1.0
Expected	10.0
Expected (Halogens)	1.0
Ratio of cation bed demineralizer flow to mixed bed (purification) demineralizer flow	0.1
Reactor coolant letdown discharged via boron recovery system (lb/hr)	
Design	296
Expected	500
Steam flow rate at max rated power (lb/hr)	1.16×10^{7}
Primary to secondary leak rate (lb/day)	
Design	1,188
Expected	100
Steam generator partition factor (Recirculating U-tube)	
Noble gases, N-16, H-3	1.0
Halogens	0.01
Cs, Rb (expected)	0.001
Cs, Rb (design)	0.0025
Others (expected)	0.001
Others (design)	0.0025
Condensate polishing demineralizer decontamination factors	
Cations	
Design: Cs, Y, Mo	2.0
Expected: Cs, Rb	2.0
All other Nuclides, including activations products	
Design	10.0
Expected	10.0
Condensate polishing flow rate (lb/hr)	8.5×10^{6}

TABLE 11.1-3 (Cont)

Parameter	Value
Thermal neutron flux (n/cm^2-sec)	$4.16x10^{13}$
Operating time (650 EFPD) (hr)	15,600
Coolant cycle time (sec)	11.3
Coolant in core time (sec)	0.9
Degasification factor	1.0
Secondary side equilibrium time (hr)	1.0×10^4
Volume control tank volumes	
Vapor (ft^3)	175
Liquid (ft^3)	125
Total secondary liquid per steam generator (lb)	$9.93x10^4$
Total steam generator blowdown flow (lb/hr) (15 gpm/steam generator is the minimum flow corresponding to secondary side design activities).	$2.24x10^4$
Fraction removed from steam generator blowdown (purification factors for design and expected cases)	
Noble gases	0.0
Halogens	0.855
Cs, Rb	0.500
Others	0.899
Tritium	0.0
Ratio of condensate Demineralizer flow rate to total steam flow rate	0.733

The information presented in the above table is based on 650 days of operation at 2,766 MWt. It was developed in support of the original license and is considered historical.

TABLE 11.1-4 (HISTORICAL)

TRITIUM PRODUCTION

Tritium Source	Total Produced (Ci/yr)	Release Expected to Reactor Coolant (Ci/yr)
Ternary fissions		
(initial cycle)	10,800	1,080
(equilibrium cycle)	8,100	810
Burnable poison rods		
(initial cycle)	760	76
Coolant (soluble boron)		
(initial cycle)	324	324
(equilibrium cycle)	229	229
Lithium, deuterium		
(initial cycle)	107	107
(equilibrium cycle)	79	79
Total initial cycle	11,991	1,587
Total equilibrium cycle	8,408	1,118

NOTES:

Core power level = 2,766 MWt.
Capacity power factor = 0.8.
Release fraction from fuel = 10 percent.
Release fraction from burnable poison rods = 10 percent.
Initial cycle weight B-10 = 2,374 grams.
Initial cycle boron = 910 ppm.
Equilibrium cycle boron = 1,100 ppm.

The production rate for tritium entering the coolant for use in evaluating effluent releases is 0.4 Ci/yr per MWt (USNRC 1976). At 2,766 MWt, this results in 1,106 curies of tritium per year available for release to the environment, which is consistent with the equilibrium cycle value given previously.

TABLE 11.1-5 (HISTORICAL)

REACTOR COOLANT N-16 ACTIVITY

Position in Loop	Loop Transit Time (sec)	N-16 Activity, (μCi/g)
Leaving core	0.0	132
Leaving reactor vessel	1.0	119
Entering steam generator	1.4	110
Leaving steam generator	6.6	74
Entering reactor coolant pump	7.2	70
Entering reactor vessel	7.9	65
Entering core	10.4	52
Leaving core	11.3	132

TABLE 11.1-6 [HISTORICAL]

SECONDARY SIDE LIQUID EQUILIBRIUM CONCENTRATIONS

Nuclide	Design* (μCi/g)	Expected** (μCi/g)
Br-83	$2.3x10^{-5}$	$1.3x10^{-7}$
Br-84	$4.0x10^{-6}$	$2.7x10^{-8}$
Br-85	$6.4x10^{-8}$	$3.5x10^{-10}$
I-130	$3.0-10^{-6}$	$8.5x10^{-8}$
I-131	$1.3x10^{-3}$	$1.2x10^{-5}$
I-132	$3.8x10^{-4}$	$3.5x10^{-6}$
I-133	$1.9x10^{-3}$	$1.6x10^{-5}$
I-134	$8.6x10^{-6}$	$6.9x10^{-7}$
I-135	$8.3x10^{-4}$	$7.1x10^{-6}$
Cr-51	$7.5x10^{-6}$	$2.9x10^{-7}$
Mn-54	$1.3x10^{-6}$	$6.5x10^{-8}$
MN-56	$1.2x10^{-5}$	
Fe-55	$6.4x10^{-6}$	$2.6x10^{-7}$
Fe-59	$4.2x10^{-6}$	$1.9x10^{-7}$
Co-58	$6.4x10^{-5}$	$2.6x10^{-6}$
Co-60	$8.0x10^{-6}$	$2.9x10^{-7}$
Rb-86	$3.4x10^{-7}$	$2.4x10^{-8}$
Rb-88	$2.2x10^{-4}$	$1.4x10^{-6}$
Rb-89	$5.7x10^{-6}$	
Sr-89	$5.2x10^{-6}$	$6.5x10^{-8}$
Sr-90	$1.9x10^{-7}$	$1.3x10^{-9}$
Sr-91	$1.6x10^{-6}$	$5.3x10^{-8}$
Sr-92	$3.1x10^{-7}$	
Y-90	$4.1x10^{-7}$	$2.5x10^{-10}$
Y-91m	$7.7x10^{-7}$	$1.9x10^{-8}$
Y-91	$1.5x10^{-6}$	$9.8x10^{-9}$
Y-92	$6.7x10^{-7}$	
Y-93	$3.9x10^{-7}$	$2.7x10^{-9}$
Zr-95	$8.4x10^{-7}$	$1.3x10^{-8}$
Nb-95	$8.6x10^{-7}$	$1.3x10^{-8}$
Mo-99	$6.5x10^{-3}$	$1.3x10^{-5}$
Tc-99m	$3.9x10^{-3}$	$7.5x10^{-6}$
Ru-103	$4.1x10^{-7}$	$6.5x10^{-9}$
Ru-106	$3.9x10^{-8}$	$1.3x10^{-9}$
Rh-103m	$4.1x10^{-7}$	$3.9x10^{-9}$
Rh-106	$3.9x10^{-8}$	$7.0x10^{-10}$
Te-125m	$1.2x10^{-7}$	$3.3x10^{-9}$
Te-127m	$2.3x10^{-6}$	$3.3x10^{-8}$
Te-127	$1.7x10^{-6}$	$8.0x10^{-8}$
Te-129m	$4.7x10^{-5}$	$1.9x10^{-7}$
Te-129	$4.3x10^{-5}$	$1.2x10^{-7}$
Te-131m	$2.4x10^{-5}$	$3.0x10^{-7}$
Te-131	$5.4x10^{-6}$	$3.7x10^{-8}$
Te-132	$3.1x10^{-4}$	$3.2x10^{-6}$

TABLE 11.1-6 (Cont)

Nuclide	Design* (μCi/g)	Expected** (μCi/g)
Te-134	$4.5x10^{-6}$	
Cs-134	$5.9x10^{-4}$	$7.3x10^{-6}$
Cs-136	$3.3x10^{-4}$	$3.7x10^{-6}$
Cs-137	$3.4x10^{-3}$	$5.3x10^{-6}$
Cs-138	$1.1x10^{-4}$	
Ba-137m	$3.1x10^{-3}$	$1.6x10^{-6}$
Ba-140	$5.2x10^{-6}$	$3.2x10^{-8}$
La-140	$2.2x10^{-6}$	$2.1x10^{-8}$
Ce-141	$8.4x10^{-7}$	$1.3x10^{-8}$
Ce-143	$5.4x10^{-7}$	$3.0x10^{-9}$
Ce-144	$5.7x10^{-7}$	$6.5x10^{-9}$
Pr-143	$8.1x10^{-7}$	$6.5x10^{-9}$
Pr-144	$5.7x10^{-7}$	$3.6x10^{-9}$
Np-239	$4.5x10^{-6}$	$1.9x10^{-7}$
H-3	$4.4x10^{-3}$	$1.0x10^{-3}$
Total (excluding H-3)	$2.2x10^{-2}$	$8.7x10^{-5}$
Total (including H-3)	$2.6x10^{-2}$	$1.1x10^{-3}$

NOTES:

*Based on 1,188 lb/day primary to secondary leak rate.
**Based on 100 lb/day primary to secondary leak rate.

The information presented in the above table is based on 650 days of operation at 2,766 MWt. It was developed in support of the original license and is considered historical.

TABLE 11.1-7 [HISTORICAL]

SECONDARY SIDE STEAM EQUILIBRIUM CONCENTRATIONS

Nuclide	Design* (μCi/g)	Expected** (μCi/g)
Kr-83m	$1.9x10^{-6}$	$7.3x10^{-9}$
Kr-85m	$9.1x10^{-6}$	$3.2x10^{-8}$
Kr-85	$4.8x10^{-5}$	$4.3x10^{-8}$
Kr-87	$5.2x10^{-6}$	$2.1x10^{-8}$
Kr-88	$1.4x10^{-5}$	$6.5x10^{-8}$
Kr-89	$4.4x10^{-7}$	$2.1x10^{-9}$
Xe-131m	$4.6x10^{-7}$	$2.3x10^{-9}$
Xe-133m	$1.3x10^{-5}$	$1.5x10^{-8}$
Xe-133	$1.1x10^{-4}$	$6.2x10^{-7}$
Xe-135m	$4.7x10^{-6}$	$5.3x10^{-9}$
Xe-135	$1.4x10^{-5}$	$7.8x10^{-8}$
Xe-137	$7.0x10^{-7}$	$3.8x10^{-9}$
Xe-138	$2.9x10^{-6}$	$1.8x10^{-8}$
Br-83	$2.3x10^{-7}$	$1.3x10^{-9}$
Br-84	$4.0x10^{-8}$	$2.7x10^{-10}$
Br-85	$6.4x10^{-10}$	$3.5x10^{-12}$
I-130	$3.0x10^{-8}$	$8.5x10^{-10}$
I-131	$1.3x10^{-5}$	$1.2x10^{-7}$
I-132	$3.8x10^{-6}$	$3.5x10^{-8}$
I-133	$1.9x10^{-5}$	$1.6x10^{-7}$
I-134	$8.6x10^{-8}$	$6.9x10^{-9}$
I-135	$8.3x10^{-6}$	$7.1x10^{-8}$
Cr-51	$1.9x10^{-8}$	$2.9x10^{-10}$
Mn-54	$3.1x10^{-9}$	$6.5x10^{-11}$
Mn-56	$3.0x10^{-8}$	-
Fe-55	$1.6x10^{-8}$	$2.6x10^{-10}$
Fe-59	$1.0x10^{-8}$	$1.9x10^{-10}$
Co-58	$1.6x10^{-7}$	$2.6x10^{-9}$
Co-60	$2.0x10^{-8}$	$2.9x10^{-10}$
Rb-86	$8.4x10^{-10}$	$2.4x10^{-11}$
Rb-88	$5.4x10^{-7}$	$1.4x10^{-9}$
Rb-89	$1.4x10^{-8}$	-
Sr-89	$1.3x10^{-8}$	$6.5x10^{-12}$
Sr-90	$4.6x10^{-10}$	$1.3x10^{-12}$
Sr-91	$3.9x10^{-9}$	$5.3x10^{-11}$
Sr-92	$7.8x10^{-10}$	-
Y-90	$1.0x10^{-9}$	$2.5x10^{-13}$
Y-91m	$1.9x10^{-9}$	$1.9x10^{-11}$
Y-91	$3.7x10^{-9}$	$9.8x10^{-12}$
Y-92	$1.7x10^{-9}$	-
Y-93	$9.7x10^{-10}$	$2.7x10^{-12}$
Z-95	$2.1x10^{-9}$	$1.3x10^{-11}$
Nb-95	$2.1x10^{-9}$	$1.3x10^{-11}$
Mo-99	$1.6x10^{-5}$	$1.3x10^{-8}$

TABLE 11.1-7 (Cont)

Nuclide	Design* (μCi/g)	Expected** (μCi/g)
Tc-99m	9.7×10^{-6}	7.5×10^{-9}
Ru-103	1.0×10^{-9}	6.5×10^{-12}
Ru-106	9.7×10^{-11}	1.3×10^{-12}
Rh-103m	1.0×10^{-9}	3.9×10^{-12}
Rh-106	9.7×10^{-11}	7.0×10^{-13}
Te-125m	2.9×10^{-10}	3.3×10^{-12}
Te-127m	5.9×10^{-9}	3.3×10^{-11}
Te-127	4.1×10^{-9}	8.0×10^{-11}
Te-129m	1.2×10^{-7}	1.9×10^{-10}
Te-129	1.1×10^{-7}	1.2×10^{-10}
Te-131m	6.0×10^{-8}	3.0×10^{-10}
Te-131	1.4×10^{-8}	3.7×10^{-11}
Te-132	7.7×10^{-7}	3.2×10^{-9}
Te-134	1.1×10^{-8}	-
Cs-134	1.5×10^{-6}	7.3×10^{-9}
Cs-136	8.3×10^{-7}	3.7×10^{-9}
Cs-137	8.4×10^{-6}	5.3×10^{-9}
Ba-137m	7.8×10^{-6}	1.6×10^{-9}
Ba-140	1.3×10^{-8}	3.2×10^{-11}
La-140	5.5×10^{-9}	2.1×10^{-11}
Ce-141	2.1×10^{-9}	1.3×10^{-11}
Ce-143	1.4×10^{-9}	3.0×10^{-12}
Ce-144	1.4×10^{-9}	6.5×10^{-12}
Pr-143	2.0×10^{-9}	6.5×10^{-12}
Pr-144	1.4×10^{-9}	3.6×10^{-12}
Np-239	1.1×10^{-8}	1.9×10^{-10}
H-3	4.4×10^{-3}	1.0×10^{-3}
Total (excluding H-3)	3.1×10^{-4}	1.4×10^{-6}
Total (including H-3)	4.7×10^{-3}	1.0×10^{-3}

NOTES:

*Based on 1,188 lb/day primary to secondary leak rate.
**Based on 100 lb/day primary to secondary leak rate.

The information presented in the above table is based on 650 days of operation at 2,766 MWt. It was developed in support of the original license and is considered historical.

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

Historical text, tables, values and parameters are identified as such and provide the historical basis for facility design.

This section describes the capabilities of Beaver Valley Power Station - Unit 2 (BVPS-2) to control, collect, process, handle, store, recycle, and dispose of liquid radioactive waste generated as a result of normal BVPS-2 operations, including anticipated operational occurrences. The liquid waste management system (LWMS) includes the radioactive liquid waste system and utilizes some of the components of the steam generator blowdown system (SGBS) for processing.

11.2.1 Design Bases

The equipment, instrumentation, and operating procedures utilized in the BVPS-2 LWMS ensure that radwastes are safely processed, and that discharges from the site are within the limits set forth in 10 CFR 20 and meet the requirements of the Annex to Appendix I of 10 CFR 50. Actual concentrations of radioactive material in plant effluents are limited in accordance with the Offsite Dose Calculation Manual.

The information presented below outlines the original licensing basis of the liquid waste system design:

1. The various sources of radioactive liquids and their daily flows for the expected and design cases for Beaver Valley Power Station - Unit 1 (BVPS-1) and BVPS-2 are given in Tables 11.2-1 and 11.2-2.

   The expected and design flows for the condensate demineralizer rinse water are obtained from vendor operation instructions. The remaining expected flows are based on the values given in NUREG-0017 (USNRC 1976). The remaining design flows are based upon the simultaneous occurrence of the expected flows and the maximum volume per event given in American National Standards Institute (ANSI) N199-1976.

2. The NUREG-0017 flows are the basis for both the expected and design radioactive liquid releases presented herein. The radioactivity concentration and release information discussed below have been retained for historical purposes and represent values associated with the original plant license. As noted earlier, actual concentrations of radioactive material in plant effluents are limited in accordance with the Offsite Dose Calculation Manual.

The expected liquid releases presented herein are based upon the activity (fraction of primary coolant) given in NUREG-0017. One exception, condensate demineralizer rinse water activity, is assumed to be 10^{-8} μCi/g in accordance with ANSI N199-1976. Tables 11.2-3, 11.2-4, 11.2-5, and 11.2-6 give detailed isotopic listings of these sources leaving the LWMS and yard drain system. The design releases are based upon the activity (fraction of primary coolant) during operation at design basis fuel leakage. Tables 11.2-7 and 11.2-8 give detailed isotopic listings of the turbine building drains design concentrations which are routed to the LWMS. Provisions described in Section 11.2.2 ensure that the turbine building drains effluent will not be released to the environment without processing if activity concentrations exceed the limits set forth in the ODCM. Tables 11.2-9 and 11.2-10 give detailed isotopic listings of effluent leaving the LWMS.

Detailed isotopic listings of sources (expected and design) entering the liquid waste system are given in Section 11.1. Fractions of the concentrations given in Section 11.1 are used as sources to the liquid waste system. All of the tables discussed above are considered historical.

3. Radiation protection criteria for the LWMS are given in Section 12.1.

4. The LWMS is designated non-nuclear safety (NNS) class and QA Category II, as defined in Section 3.2.

5. The seismic design classifications of structures housing the LWMS conform to the intent of guidelines of Regulatory Guide 1.143 except as noted in Section 1.8, BVPS-2 position on Regulatory Guide 1.143. Portions of the LWMS are located in the auxiliary building and the waste handling building, which are Seismic Category I and Category II structures, respectively.

6. Pressure-retaining components in the systems utilize welded construction to the maximum practicable extent. Flanged joints, or suitable quick-disconnect fittings, are used only where maintenance or operational requirements clearly indicate that such construction is preferable. Screwed connections in which threads provide the only seals are not used except for instrumentation connections where welded connections are not suitable. Process lines will not be less than 3/4 inch. Lines 3/4 inch and greater, but less than 2 1/2 inches nominal size, are socket-welded, with the exception of the evaporator bottoms piping, which is butt-welded to minimize crud trapping. For lines of 2 1/2 inches and above, piping is butt-welded. Backing rings are not used in lines carrying resins or other

particulate material. All welding constituting the pressure boundary of pressure-retaining components will be performed in accordance with ANSI B31.1, 1967, including all addenda through June 1972, and ASME Code, Section VIII, July 1971, including addenda of Winter, 1972.

7. The process piping systems are hydrostatically tested. Testing of piping systems is performed in accordance with ANSI B31.1, 1967, and all addenda through June 1972.

The LWMS is designed to meet the anticipated processing requirements of BVPS-2. Adequate storage capacity is provided to process liquid wastes during periods when major processing equipment may be down for maintenance and during periods of excessive waste generation. There are two waste drain tanks of 7,500 gallons each, and two backup steam generator blowdown (SGB) hold tanks of 50,000 gallons each, for a total capacity of 115,000 gallons, all of which can be processed via the evaporators.

In addition, the BVPS-1 and BVPS-2 liquid waste systems are cross-connected for added availability.

There is no expected effluent release from the BVPS-1 boron recovery system (BRS) to the BVPS-2 SGB evaporator during normal base-loaded operation. A cross-connect is provided that can be used to accommodate additional liquid waste generated as a result of the original plant design for load-following from the LWMS.

Releases to the environment from the LWMS are monitored prior to discharge. Process and effluent radiological monitoring systems are described in Section 11.5.

Additional information about implementation of General Design Criteria 60 and 64 of Appendix A to 10 CFR 50 is included in Sections 3.1.2.60, 3.1.2.64, 11.3, 11.4, and 11.5.

The following design features are incorporated to reduce equipment maintenance downtime, liquid leakage, and gaseous releases of radioactive materials to the environment, and to facilitate cleaning and improve radwaste operations:

1. Materials of construction are chosen to resist corrosion and erosion.

2. Conical-bottom tanks and vessels are used where a potential exists for high activity and suspended solids to minimize buildup of radioactive sludge and to facilitate cleaning.

   Although the SGB hold tanks are flat-bottom tanks, provisions are taken to prevent crud buildup in the bottom of the tanks. These provisions include a second suction for the waste drain tank pumps that is located higher up on the waste drain tanks to minimize crud being pumped to SGB hold tanks.

3. Components handling potentially high activity liquids (that is, evaporator bottoms hold tank, evaporator circulation pumps, and waste drain tanks) are located in individually shielded cubicles. Other components handling lower activity liquids are shielded, as necessary, to minimize exposure to operators and maintenance personnel.

4. Pumps handling concentrated radioactive liquids (evaporator bottoms and hold tank pumps) are fitted with double mechanical seals with the exception of the evaporator circulation pumps, which are equipped with primary water injected inleakage single mechanical seals. In the event of seal failure, the leakage is directed to a radioactive sump through a drain connection.

5. Piping is designed to minimize crud pockets where activity could accumulate.

6. Valves are chosen to minimize crud pockets where activity could accumulate.

7. Tanks that are expected to handle highly radioactive liquids are vented to the aerated vent system (Section 11.3) to minimize the potential for gaseous releases into working areas.

8. Alternate processing capability is provided, allowing alternate processing paths to be used in the event of equipment failure. Equipment for additional processing is also provided to ensure liquid activities are reduced to acceptable levels.

9. Controls are provided to allow system operation and monitoring from the main control room except for evaporator sampling and bottom transfer to solid waste, which must be performed locally.

10. The possibility and consequences of tank overflow are minimized by various monitor, alarm, and process functions, as described in Table 11.2-11. Monitoring for radioactivity is described in Section 11.5.

11.2.2 System Description

The LWMS is illustrated on Figures 11.2-1 through 11.2-6. A tabulation of the system components and their design parameters is presented in Table 11.2-12.

The LWMS includes two evaporators which have adequate capacity to process all of the anticipated liquid waste. (Although these evaporators are designated steam generator blowdown (SGB) evaporators, they can be used as liquid waste evaporators.) Steam generator blowdown is not normally processed by the liquid waste system (Section 10.4.8).

Two 7,500-gallon waste drain tanks accept and store all liquid waste to be processed and allow sufficient time for sampling the liquid waste prior to processing the tank contents through the BVPS-2 clean-up ion exchangers or SGB evaporators, or transferring it to the high level waste drain tanks in BVPS-1. Liquid waste can be transferred to the two 50,000 gallon SGB hold tanks. Liquid waste can also be transferred from the waste drain tank pumps directly to the evaporators or ion exchangers.

The evaporator systems, which include the evaporators and the clean-up ion exchangers, produce a distillate suitable for reuse or discharge.

Distillate from the evaporators is collected in test tanks, sampled, and, if within allowable chemistry and activity limits, recycled to the BVPS-1 boron recovery test tanks for reuse or discharge to BVPS-1 or BVPS-2 cooling tower blowdowns in accordance with the limits specified in the Offsite Dose Calculation Manual.

Liquid from the test tanks can be recycled. However, for the purpose of evaluating the radiological impact on the environment, 100 percent of test tank contents is assumed to be discharged, as well as 140,000 gallons per year (gal/yr) of BRS distillate released to control the buildup of tritium in the reactor coolant. Assurance that waste above predetermined activity limits is not inadvertently discharged to the environment is provided through sampling of the test tanks and by the liquid waste effluent monitor. This monitor alarms when activity levels in the effluent exceed Offsite Dose Calculation Manual limits and automatically terminates the release.

Each batch is analyzed prior to release using gamma spectroscopy and the activity of each radionuclide to be discharged is recorded. Isotopic analyses and composites of retained samples are made in accordance with the Offsite Dose Calculation Manual. Detailed administrative records of all radioactive liquid releases are maintained. Tables 11.2-3 and 11.2-5 present annual expected discharge activities for significant isotopes. The tables are retained for Historical information.

Process flexibility is increased by allowing liquid waste of low activity, as verified by sampling, from the liquid waste drain tanks or the steam generator hold tanks to be filtered prior to discharge without evaporation and processing.

The turbine building floor drains and sumps discharge to collection manholes in the yard and gravity-drain through oil separators and normally go to the environment via the BVPS-2 yard drainage system.

Grab samples are obtained from the turbine building drains system as required by the Offsite Dose Calculation Manual (ODCM). If the turbine building drains activity concentrations at the discharge point are determined to exceed the limits set forth in the ODCM, then the turbine building drains will be isolated from the storm drainage system via manual isolation valves and transferred to the liquid waste system. Therefore, the design case turbine building drains concentrations given in Table 11.2-8 will not be released to the environment without appropriate dilution.

11.2.3 System Evaluation

The LWMS has been designed with sufficient storage capacity to accommodate liquid input equivalent to design system flows based upon the frequency of events specified by Table 6 of ANSI N199-1976 (information summarized in Table 11.2-2) and total 45,670 gallons. These design flows do not include turbine building drains or steam generator blowdown since they are not normally processed by the liquid waste system. Processing of steam generator blowdown is administratively controlled by a valve between the steam generator blowdown tank and the blowdown tank drain cooler (Figure 10.4-23). Normal liquid waste system storage inventory is 115,000 gallons and therefore has the capability to store this design input.

Assuming that two liquid waste storage tanks (one waste drain tank, 7,500 gallons and one SGB hold tank, 50,000 gallons) are full or isolated and the BVPS-1 high level waste tanks cannot accept any BVPS-2 waste, system storage capacity decreases to 57,500 gallons. Assuming also that evaporator operation is secured during this period of liquid input, it can be seen that system storage capacity is sufficient to store this design input.

System component sizing is based upon the expected inputs shown in Table 11.2-2.

The design liquid flows of Tables 11.2-1 and 11.2-2 assume a maximum volume event occurring in each input category during the same day (tank overflow, pump seal failure, etc).

An estimate of liquid waste inputs is 4,060 gallons per day (gpd) total expected flow rate into the liquid waste system. This volume can be easily accommodated in the remaining storage volume of 57,500 gallons, and can be processed by one evaporator operating approximately 15 percent of the time. This allows for satisfactory system operation even if one evaporator and associated equipment were isolated for maintenance at all times. System redundancy is provided by including two evaporators, two waste drain tanks and pumps, two steam generator hold tanks, two evaporator feed pumps, two test tanks and pumps, and two cleanup ion exchangers, each of which can handle system requirements while the other is isolated for maintenance.

The BVPS-2 LWMS can operate as intended during periods of surge waste flows, anticipated operational occurrences, and equipment downtime.

Expected liquid radioactive releases should not be affected during periods of equipment downtime. System operation would be as described previously, using redundant equipment/alternate evaporator operation to produce effluent and equivalent releases, as given in Tables 11.2-3, 11.2-4, 11.2-5, and 11.2-6. The tables are retained for Historical information.

11.2.4 Radioactive Releases

In support of the original license application, models and assumptions from NUREG-0017 were used to calculate the expected radioactivity in the liquid discharge. Tables 11.2-3 and 11.2-5 give the expected yearly releases of radionuclides in Ci/yr/reactor, and Tables 11.2-4 and 11.2-6 give the expected yearly releases of radionuclides in μCi/g. The above tables support the original license and are retained for historical purposes.

There are two potential liquid waste release points. These are the cooling tower discharge point and a separate release point for the turbine building drains.

The liquid waste discharge is diluted by blowdown from BVPS-1 and BVPS-2 cooling towers of approximately 15,000 gpm and 7,800 gpm, respectively. These blowdown rates are the yearly minimum values considering the largest drift and evaporation rates.

Stream routes, including process equipment used, are shown on Figures 11.2-7 and 11.2-8.

The calculated expected effluents did not exceed the concentration limits of the Offsite Dose Calculation Manual. The doses due to these effluents did not exceed the numerical design objectives of Appendix I to 10 CFR 50 and the dose limits of 10 CFR 20. The calculated annual doses for the previously mentioned liquid effluent releases are summarized in Appendix 11A. Appendix 11A supports the original license and is retained for historical purposes.

Scaling techniques, based on NUREG-0017, Revision 1 methodology, were utilized to assess the impact of core power uprate on radioactive liquid effluents at BVPS.

The conservatively performed power uprate analysis utilized the plant core power operating history during the years 1997 to 2001, the reported liquid effluent and dose data during that period, NUREG-0017 equations and assumptions, and conservative methodology, to estimate the impact of operation at the analyzed uprate core power level of 2918 MWt, over that of operation at the previously licensed power level, on radioactive liquid effluents and consequent normal operation off-site doses.

The licensed reactor core power level during the 1997 to 2001 time frame was 2652 MWt (note that for a portion of the year 2001, BVPS operated at a core power level of 2689 MWt, but the assessment conservatively assumed a core power level of 2652 MWt for the entire period). For the uprate condition, the system parameters utilized in the power uprate analysis reflected the flow rates and coolant masses at an analyzed NSSS power level of 2910 MWt and a core power level of 2918 MWt. For the pre-uprate condition, the evaluation utilized offsite doses based on an average 5 year set of organ and whole body doses calculated using data presented in the BVPS Annual Radioactive Effluent Release Reports for the years 1997 through 2001, taking into consideration the associated average annual core power level, extrapolated to 100 percent availability at the licensed power level.

Using the methodology and equations found in NUREG-0017, Revision 1, and based on a comparison of the change in power level and in plant coolant system parameters (e.g., reactor coolant mass, steam generator liquid mass, steam flow rate, reactor coolant letdown flow rate, flow rate to the cation demineralizer, letdown flow rate for boron control, steam generator blowdown flow rate, steam generator moisture carryover, etc.) for both pre-uprate and uprate conditions, the maximum potential percentage increase in coolant activity levels due to the uprate, for each chemical group identified in NUREG-0017, was estimated.

To estimate an upper bound impact on off-site doses, the highest factor found for any chemical group pertinent to the release pathway was applied to the average doses previously determined as representative of operation at pre-uprate conditions. This approach was utilized to estimate the maximum potential increase in effluent doses due to the uprate, and demonstrate that the estimated off-side doses following the uprate, although increased will continue to remain below the regulatory limits set by 10CFR50, Appendix I.

It is noted that, for an operating plant such as BVPS, the actual liquid effluent isotopic release curie and dose information are provided in the Annual Radioactive Effluent Release Reports.

11.2.5 References for Section 11.2

American National Standards Institute 1976. American National Standard Liquid Radioactive Waste Processing System for Pressurized Water Reactor Plants, ANSI N199-1976.

U.S. Nuclear Regulatory Commission 1976. Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors, NUREG-0017.

U.S. Nuclear Regulatory Commission 1985. Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors, NUREG-0017, Rev. 1.

Tables for Section 11.2

TABLE 11.2-1 [HISTORICAL]

LIQUID WASTE FLOWS
BVPS-1

Waste Input	Expected (gpd)*	Design (gpd)**
Containment sump	40	15,000
Auxiliary building sump	200	13,500
Lab drains	400	5,000
Reactor plant samples	35	1,200
Miscellaneous sources	700	6,500
CVCS letdown (BVPS-1)***	1,440	2,880
CVCS letdown (BVPS-2)***	1,440	2,880
BVPS-1 turbine building drains	7,200	151,000
Total input****	4,255	46,960

NOTES:

* Expected flows are obtained from NUREG-0017 and ANSI N199-1976.

** Design values are based upon simultaneous occurrence of expected flows and maximum daily volume per event from ANSI N199-1976 and NUREG-0017. These values assume a worst case of an operational occurrence in each flow category in the same day. These flows are only used to evaluate system storage capacity.

*** Contribution to the liquid waste system from an 86,500 gpd expected or 173,000 gpd design letdown rate.

**** Turbine building drains are not processed by the liquid waste system and, therefore, are not included as a waste input.

TABLE 11.2-2 [HISTORICAL]

LIQUID WASTE FLOWS
BVPS-2

Waste Input	Expected (gpd)*	Design (gpd)**
Containment sump	40	15,000
Auxiliary building sump	200	13,500
Miscellaneous sources	700	6,500
Reactor plant samples	35	1,200
Lab drains	400	5,000
Condensate demineralized rinse water***	2,685	4,470
BVPS-2 turbine building drains	7,200	151,000
Total input****	4,060	45,670

NOTES:

* Expected flows are obtained from NUREG-0017, with exception of condensate demineralizer waste.

** Design values, with the exception of condensate demineralizer rinse water, are based upon simultaneous occurrence of expected flows and maximum daily volume per event from ANSI 199-1976. These values assume a worst case of an operational occurrence in each flow category in the same day. These flows are only used to evaluate system storage capacity.

*** NUREG-0017 value is not used in this case due to use of Powdex type demineralizer in lieu of deep bed regenerant type. Expected daily input is based upon backflushing at a rate of 200 flushes/year (4,900 gal/flush). Design flow is based upon backflushing at a rate of 333 flushes/year.

**** Turbine building drains are not normally processed by the liquid waste system and therefore are not included as a waste input.

TABLE 11.2-3 [HISTORICAL]

TURBINE BUILDING DRAIN RELEASES
EXPECTED CASE (Ci/yr)

Nuclide	BVPS-1 Releases	BVPS-2 Releases*
Cr-51	2.3×10^{-6}	5.2×10^{-6}
Mn-54	5.1×10^{-7}	1.2×10^{-6}
Fe-55	2.0×10^{-6}	4.7×10^{-6}
Fe-59	1.5×10^{-6}	3.4×10^{-6}
Co-58	2.0×10^{-5}	4.7×10^{-5}
Co-60	2.3×10^{-6}	5.2×10^{-6}
Sr-89	5.1×10^{-7}	1.2×10^{-7}
Sr-90	9.9×10^{-9}	2.3×10^{-7}
Sr-91	4.9×10^{-7}	9.5×10^{-7}
Y-90	2.0×10^{-9}	4.5×10^{-9}
Y-91m	2.3×10^{-7}	3.4×10^{-7}
Y-91	7.5×10^{-7}	1.8×10^{-7}
Y-93	2.4×10^{-7}	4.7×10^{-7}
Zr-95	9.9×10^{-7}	2.3×10^{-7}
Nb-95	9.9×10^{-7}	2.3×10^{-7}
Mo-99	9.9×10^{-5}	2.3×10^{-4}
Tc-99m	7.3×10^{-5}	1.4×10^{-4}
Ru-103	5.1×10^{-7}	1.2×10^{-7}
Ru-106	9.9×10^{-9}	2.3×10^{-7}
Rh-103m	4.7×10^{-7}	7.0×10^{-7}
Rh-106	9.1×10^{-9}	1.3×10^{-7}
Te-125m	2.4×10^{-7}	5.9×10^{-7}
Te-127m	2.4×10^{-7}	5.9×10^{-7}
Te-127	7.5×10^{-7}	1.4×10^{-6}
Te-129m	1.5×10^{-6}	3.4×10^{-6}
Te-129	1.4×10^{-6}	2.1×10^{-6}
Te-131m	2.4×10^{-6}	5.4×10^{-6}
Te-131	4.7×10^{-7}	6.6×10^{-7}

TABLE 11.2-3 (Cont)

Nuclide	BVPS-1 Releases	BVPS-2 Releases*
Te-132	$2.4x10^{-5}$	$5.7x10^{-5}$
Ba-137m	$2.1x10^{-5}$	$2.9x10^{-5}$
Ba-140	$2.4x10^{-7}$	$5.7x10^{-5}$
La-140	$1.8x10^{-7}$	$3.8x10^{-7}$
Ce-141	$9.9x10^{-7}$	$2.3x10^{-7}$
Ce-143	$2.4x10^{-7}$	$5.4x10^{-7}$
Ce-144	$5.1x10^{-7}$	$1.2x10^{-7}$
Pr-143	$5.1x10^{-7}$	$1.2x10^{-7}$
Pr-144	$4.6x10^{-7}$	$6.6x10^{-7}$
Np-239	$1.5x10^{-6}$	$3.4x10^{-6}$
Br-83	$2.3x10^{-5}$	$2.3x10^{-5}$
Br-84	$3.9x10^{-6}$	$4.7x10^{-6}$
Br-85	$4.7x10^{-7}$	$6.3x10^{-7}$
I-130	$2.1x10^{-5}$	$1.5x10^{-5}$
I-131	$3.4x10^{-3}$	$2.1x10^{-3}$
I-132	$6.2x10^{-4}$	$6.3x10^{-4}$
I-133	$4.1x10^{-3}$	$2.9x10^{-3}$
I-134	$1.0x10^{-4}$	$1.2x10^{-4}$
I-135	$1.5x10^{-3}$	$1.3x10^{-3}$
Rb-86	$2.0x10^{-7}$	$4.3x10^{-7}$
Rb-88	$1.6x10^{-5}$	$2.5x10^{-5}$
Cs-134	$5.5x10^{-5}$	$1.3x10^{-4}$
Cs-136	$2.9x10^{-5}$	$6.6x10^{-5}$
Cs-137	$4.1x10^{-5}$	$9.5x10^{-5}$
Total	$1.0x10^{-2}$	$8.0x10^{-3}$

NOTE:

* Some of the liquid releases generated as a result of BVPS-2 opera-tion are included in BVPS-2 results even though they will be pro-cessed and discharged from BVPS-1. These include CVCS letdown releases.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.2-4 [HISTORICAL]

TURBINE BUILDING DRAIN CONCENTRATION
EXPECTED CASE (μCi/g)

Nuclide	BVPS-1 Concentrations	BVPS-2 Concentrations*
Cr-51	$1.1x10^{-11}$	$2.5x10^{-11}$
Mn-54	$2.5x10^{-12}$	$5.3x10^{-12}$
Fe-55	$9.7x10^{-12}$	$1.2x10^{-11}$
Fe-59	$7.3x10^{-12}$	$1.6x10^{-11}$
Co-58	$9.7x10^{-11}$	$2.3x10^{-10}$
Co-60	$1.1x10^{-11}$	$2.5x10^{-11}$
Sr-89	$2.5x10^{-13}$	$5.8x10^{-13}$
Sr-90	$4.8x10^{-14}$	$1.1x10^{-13}$
Sr-91	$2.4x10^{-12}$	$4.6x10^{-12}$
Y-90	$9.7x10^{-15}$	$2.2x10^{-14}$
Y-91m	$1.1x10^{-12}$	$1.6x10^{-12}$
Y-91	$3.6x10^{-13}$	$8.7x10^{-13}$
Y-93	$1.2x10^{-13}$	$2.3x10^{-13}$
Zr-95	$4.8x10^{-13}$	$1.1x10^{-12}$
Nb-95	$4.8x10^{-13}$	$1.1x10^{-12}$
Mo-99	$4.8x10^{-10}$	$1.1x10^{-9}$
Tc-99m	$3.5x10^{-10}$	$6.8x10^{-10}$
Ru-103	$2.5x10^{-13}$	$5.8x10^{-13}$
Ru-106	$4.8x10^{-14}$	$1.1x10^{-13}$
Rh-103m	$2.3x10^{-13}$	$3.4x10^{-13}$
Rh-106	$4.4x10^{-14}$	$6.3x10^{-14}$
Te-125m	$1.2x10^{-13}$	$2.9x10^{-13}$
Te-127m	$1.2x10^{-12}$	$2.9x10^{-12}$
Te-127	$3.6x10^{-12}$	$6.8x10^{-12}$
Te-129m	$7.3x10^{-12}$	$1.6x10^{-11}$
Te-129	$6.8x10^{-12}$	$1.0x10^{-11}$

TABLE 11.2-4 (Cont)

Nuclide	BVPS-1 Concentrations	BVPS-2 Concentrations*
Te-131m	$1.2x10^{-11}$	$2.6x10^{-11}$
Te-131	$2.3x10^{-12}$	$3.2x10^{-12}$
Te-132	$1.2x10^{-10}$	$2.8x10^{-10}$
Ba-137m	$1.0x10^{-10}$	$1.4x10^{-10}$
Ba-140	$1.2x10^{-12}$	$2.8x10^{-10}$
La-140	$8.7x10^{-13}$	$1.8x10^{-12}$
Ce-141	$4.8x10^{-13}$	$1.1x10^{-12}$
Ce-143	$1.2x10^{-13}$	$2.6x10^{-13}$
Ce-144	$2.5x10^{-13}$	$5.8x10^{-13}$
Pr-143	$2.5x10^{-13}$	$5.8x10^{-13}$
Pr-144	$2.2x10^{-13}$	$3.2x10^{-13}$
Np-239	$7.3x10^{-12}$	$1.6x10^{-11}$
Br-83	$1.1x10^{-10}$	$1.1x10^{-10}$
Br-84	$1.9x10^{-11}$	$2.3x10^{-11}$
Br-85	$2.3x10^{-13}$	$3.0x10^{-13}$
I-130	$1.0x10^{-10}$	$7.2x10^{-11}$
I-131	$1.6x10^{-8}$	$1.0x10^{-8}$
I-132	$3.0x10^{-9}$	$3.0x10^{-9}$
I-133	$2.0x10^{-8}$	$1.4x10^{-8}$
I-134	$4.8x10^{-10}$	$5.8x10^{-10}$
I-135	$7.3x10^{-9}$	$6.3x10^{-9}$
Rb-86	$9.7x10^{-13}$	$2.1x10^{-12}$
Rb-88	$7.7x10^{-11}$	$1.2x10^{-10}$
Cs-134	$2.7x10^{-10}$	$6.3x10^{-10}$
Cs-136	$1.4x10^{-10}$	$3.2x10^{-10}$
Cs-137	$2.0x10^{-10}$	$4.6x10^{-10}$
Total	$4.8x10^{-8}$	$3.9x10^{-8}$

NOTE:

* Some of the liquid releases generated as a result of BVPS-2 opera-tion are included in BVPS-2 results even though they will be processed and discharged from BVPS-1. These include CVCS letdown releases.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.2-5 [HISTORICAL]

LIQUID RADIOACTIVE SYSTEM
EXPECTED ANNUAL RELEASES (Ci/yr)
FROM COOLING TOWER DISCHARGE POINT

Nuclide	BVPS-1 Releases	BVPS-2 Releases*	Total
Cr-51	$1.3x10^{-3}$	$1.0x10^{-4}$	$1.4x10^{-3}$
Mn-54	$3.1x10^{-4}$	$2.5x10^{-5}$	$3.4x10^{-4}$
Fe-55	$1.6x10^{-3}$	$1.3x10^{-4}$	$1.7x10^{-3}$
Fe-59	$8.3x10^{-4}$	$6.5x10^{-5}$	$9.0x10^{-4}$
Co-58	$1.4x10^{-2}$	$1.1x10^{-3}$	$1.5x10^{-2}$
Co-60	$2.0x10^{-3}$	$1.6x10^{-4}$	$2.2x10^{-3}$
Sr-89	$2.9x10^{-4}$	$2.2x10^{-5}$	$3.1x10^{-4}$
Sr-90	$1.1x10^{-5}$	$8.5x10^{-7}$	$1.2x10^{-5}$
Sr-91	$1.3x10^{-5}$	$5.3x10^{-6}$	$1.8x10^{-5}$
Y-90	$9.4x10^{-6}$	$6.0x10^{-7}$	$1.0x10^{-5}$
Y-91m	$8.7x10^{-6}$	$3.6x10^{-6}$	$1.2x10^{-5}$
Y-91	$5.7x10^{-5}$	$4.4x10^{-6}$	$6.1x10^{-5}$
Y-93	$7.4x10^{-7}$	$3.0x10^{-7}$	$1.0x10^{-6}$
Zr-95	$5.1x10^{-5}$	$4.0x10^{-6}$	$5.5x10^{-5}$
Nb-95	$5.2x10^{-3}$	$4.0x10^{-6}$	$5.6x10^{-5}$
Mo-99	$1.1x10^{-2}$	$2.3x10^{-3}$	$1.3x10^{-2}$
Tc-99m	$1.1x10^{-2}$	$2.1x10^{-3}$	$1.3x10^{-2}$
Ru-103	$3.4x10^{-5}$	$2.7x10^{-6}$	$3.7x10^{-5}$
Ru-106	$1.0x10^{-5}$	$8.2x10^{-7}$	$1.1x10^{-6}$
Rh-103m	$3.4x10^{-5}$	$2.7x10^{-6}$	$3.7x10^{-5}$
Rh-106	$1.0x10^{-5}$	$8.2x10^{-7}$	$1.1x10^{-5}$
Te-125m	$2.5x10^{-5}$	$1.9x10^{-6}$	$2.7x10^{-5}$
Te-127m	$2.6x10^{-4}$	$2.1x10^{-5}$	$2.8x10^{-4}$
Te-127	$2.7x10^{-4}$	$2.5x10^{-5}$	$3.0x10^{-4}$
Te-129m	$1.1x10^{-3}$	$8.2x10^{-5}$	$1.2x10^{-3}$
Te-129	$6.7x10^{-4}$	$5.3x10^{-5}$	$7.2x10^{-4}$
Te-131m	$1.6x10^{-4}$	$5.2x10^{-5}$	$2.1x10^{-4}$
Te-131	$3.0x10^{-5}$	$9.4x10^{-6}$	$3.9x10^{-5}$
Te-132	$4.3x10^{-3}$	$7.8x10^{-4}$	$5.1x10^{-3}$

TABLE 11.2-5 (Cont)

Nuclide	BVPS-1 Releases	BVPS-2 Releases*	Total
Ba-137m	3.1×10^{-2}	2.1×10^{-2}	5.2×10^{-2}
Ba-140	1.1×10^{-4}	9.3×10^{-6}	1.2×10^{-4}
La-140	1.1×10^{-4}	8.4×10^{-6}	1.2×10^{-4}
Ce-141	5.1×10^{-5}	4.0×10^{-6}	5.5×10^{-5}
Ce-143	2.8×10^{-6}	8.6×10^{-7}	3.7×10^{-6}
Ce-144	3.2×10^{-5}	2.6×10^{-6}	3.5×10^{-5}
Pr-143	2.7×10^{-5}	2.3×10^{-6}	2.9×10^{-5}
Pr-144	3.2×10^{-5}	2.6×10^{-6}	3.5×10^{-5}
Np-239	1.4×10^{-4}	3.2×10^{-5}	1.7×10^{-4}
Br-83	2.5×10^{-5}	2.9×10^{-5}	5.4×10^{-4}
Br-84	2.7×10^{-6}	5.9×10^{-9}	2.7×10^{-6}
Br-85	2.8×10^{-8}	0.0	2.8×10^{-8}
I-130	1.2×10^{-4}	2.3×10^{-4}	3.5×10^{-4}
I-131	1.6×10^{-1}	1.0×10^{-1}	2.6×10^{-1}
I-132	4.9×10^{-3}	2.3×10^{-3}	7.2×10^{-3}
I-133	4.0×10^{-2}	6.5×10^{-2}	1.1×10^{-1}
I-134	8.0×10^{-5}	4.6×10^{-6}	8.5×10^{-5}
I-135	4.3×10^{-3}	9.2×10^{-3}	1.4×10^{-2}
Rb-86	7.5×10^{-5}	3.7×10^{-5}	1.1×10^{-4}
Rb-88	1.2×10^{-4}	0.0	1.2×10^{-4}
Cs-134	4.6×10^{-2}	3.0×10^{-2}	7.6×10^{-2}
Cs-136	8.9×10^{-3}	3.9×10^{-3}	1.3×10^{-2}
Cs-137	3.3×10^{-2}	2.2×10^{-2}	5.5×10^{-2}
H-3	5.5×10^{2}	5.5×10^{2}	1.1×10^{3}
Total (except H-3)	3.8×10^{-1}	2.6×10^{-1}	6.4×10^{-1}

NOTE:

* Some of the liquid releases generated as a result of BVPS-2 operation are included in BVPS-2 results even though they will be processed and discharged from BVPS-1. These include CVCS letdown releases.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.2-6 [HISTORICAL]

LIQUID RADIOACTIVE SYSTEM
EXPECTED ANNUAL RELEASE CONCENTRATIONS (μCi/g)
FROM COOLING TOWER DISCHARGE POINT

Nuclide	BVPS-1 Release Conc	BVPS-2 Release Conc*	Combined Release Conc**
Cr-51	4.4×10^{-11}	6.7×10^{-12}	3.1×10^{-11}
Mn-54	1.0×10^{-11}	1.6×10^{-12}	7.5×10^{-12}
Fe-55	5.4×10^{-11}	8.4×10^{-12}	3.8×10^{-11}
Fe-59	2.8×10^{-11}	4.2×10^{-12}	2.0×10^{-11}
Co-58	4.7×10^{-10}	7.1×10^{-11}	3.3×10^{-10}
Co-60	6.7×10^{-11}	7.0×10^{-11}	4.9×10^{-11}
Sr-89	9.6×10^{-12}	1.4×10^{-12}	6.8×10^{-12}
Sr-90	3.5×10^{-13}	5.5×10^{-14}	2.6×10^{-13}
Sr-91	4.4×10^{-13}	3.4×10^{-13}	4.0×10^{-13}
Y-90	3.1×10^{-13}	3.9×10^{-14}	2.2×10^{-13}
Y-91m	2.9×10^{-13}	2.3×10^{-13}	2.6×10^{-13}
Y-91	1.9×10^{-12}	2.8×10^{-13}	1.3×10^{-12}
Y-93	2.5×10^{-14}	2.0×10^{-14}	2.2×10^{-14}
Zr-95	1.7×10^{-12}	2.6×10^{-13}	1.2×10^{-12}
Nb-95	1.7×10^{-12}	2.6×10^{-13}	1.2×10^{-12}
Mo-99	3.8×10^{-10}	1.5×10^{-10}	2.9×10^{-10}
Tc-99m	3.6×10^{-10}	1.4×10^{-10}	2.9×10^{-10}
Ru-103	1.2×10^{-12}	1.7×10^{-13}	8.2×10^{-13}
Ru-106	3.4×10^{-13}	5.3×10^{-14}	2.4×10^{-14}
Rh-103m	1.1×10^{-12}	1.7×10^{-13}	8.1×10^{-13}
Rh-106	3.4×10^{-13}	5.3×10^{-14}	2.4×10^{-13}
Te-125m	8.3×10^{-13}	1.2×10^{-13}	5.9×10^{-13}
Te-127m	8.7×10^{-12}	1.3×10^{-12}	6.2×10^{-12}
Te-127	8.9×10^{-12}	1.6×10^{-12}	6.6×10^{-12}
Te-129m	3.5×10^{-11}	5.3×10^{-12}	2.6×10^{-11}
Te-129	2.3×10^{-11}	3.4×10^{-12}	1.6×10^{-11}
Te-131m	5.4×10^{-12}	3.3×10^{-12}	4.6×10^{-12}
Te-131	1.0×10^{-12}	6.1×10^{-13}	8.6×10^{-13}
Te-132	1.4×10^{-10}	5.1×10^{-11}	1.1×10^{-10}
Ba-137m	1.0×10^{-9}	1.3×10^{-9}	1.1×10^{-9}
Ba-140	3.5×10^{-12}	6.0×10^{-13}	2.6×10^{-12}

TABLE 11.2-6 (Cont)

Nuclide	BVPS-1 Release Conc	BVPS-2 Release Conc*	Combined Release Conc**
La-140	3.8×10^{-12}	5.4×10^{-13}	2.6×10^{-12}
Ce-141	1.7×10^{-12}	2.6×10^{-13}	1.2×10^{-12}
Ce-143	9.5×10^{-14}	5.6×10^{-14}	8.2×10^{-14}
Ce-144	1.1×10^{-12}	1.7×10^{-13}	7.7×10^{-13}
Pr-143	9.1×10^{-13}	1.5×10^{-13}	6.4×10^{-13}
Pr-144	1.1×10^{-12}	1.7×10^{-13}	7.7×10^{-13}
Np-239	4.8×10^{-12}	2.1×10^{-12}	3.7×10^{-12}
Br-83	8.4×10^{-13}	1.9×10^{-12}	1.2×10^{-12}
Br-84	9.0×10^{-14}	3.8×10^{-16}	5.9×10^{-14}
Br-85	9.5×10^{-16}	0.0	6.2×10^{-16}
I-130	4.0×10^{-12}	1.5×10^{-11}	7.7×10^{-12}
I-131	5.3×10^{-9}	6.5×10^{-9}	5.7×10^{-9}
I-132	1.7×10^{-10}	1.5×10^{-10}	1.6×10^{-10}
I-133	1.4×10^{-9}	4.2×10^{-9}	2.4×10^{-9}
I-134	2.7×10^{-12}	2.9×10^{-13}	1.9×10^{-12}
I-135	1.5×10^{-10}	6.0×10^{-10}	3.1×10^{-10}
Rb-86	2.5×10^{-12}	2.4×10^{-12}	2.4×10^{-12}
Rb-88	3.9×10^{-12}	0.0	2.6×10^{-12}
Cs-134	1.5×10^{-9}	2.0×10^{-9}	1.7×10^{-9}
Cs-136	3.0×10^{-10}	2.5×10^{-10}	2.9×10^{-10}
Cs-137	1.1×10^{-9}	1.4×10^{-9}	1.2×10^{-9}
H-3	1.9×10^{-5}	3.6×10^{-5}	2.4×10^{-5}
Total (except H-3)	1.3×10^{-8}	1.7×10^{-8}	1.4×10^{-8}

NOTE:

*Some of the liquid releases generated as a result of BVPS-2 operation are included in BVPS-2 results even though they will be processed and discharged from BVPS-1. These include CVCS letdown releases.

**The BVPS-1 and BVPS-2 flow rates are different, therefore, the combined concentration does not equal the sum of BVPS-1 concentrations and BVPS-2 concentrations.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.2-7 [HISTORICAL]

TURBINE BUILDING DRAIN RELEASES
TO LIQUID WASTE MANAGEMENT SYSTEM
DESIGN CASE (Ci/yr)**

Nuclide	BVPS-1 Releases	BVPS-2 Releases*
Cr-51	$1.5x10^{-4}$	$1.6x10^{-4}$
Mn-54	$2.6x10^{-5}$	$2.6x10^{-5}$
Fe-55	$1.3x10^{-4}$	$1.4x10^{-4}$
Fe-59	$8.2x10^{-5}$	$8.4x10^{-5}$
Co-58	$1.3x10^{-3}$	$1.4x10^{-3}$
Co-60	$1.6x10^{-4}$	$1.7x10^{-4}$
Sr-89	$1.0x10^{-4}$	$1.1x10^{-4}$
Sr-90	$2.4x10^{-6}$	$2.4x10^{-6}$
Sr-91	$3.2x10^{-5}$	$3.3x10^{-5}$
Y-90	$2.8x10^{-6}$	$4.9x10^{-6}$
Y-91m	$2.1x10^{-5}$	$1.6x10^{-5}$
Y-91	$1.6x10^{-5}$	$3.0x10^{-5}$
Y-93	$5.7x10^{-6}$	$8.1x10^{-6}$
Zr-95	$1.8x10^{-5}$	$1.8x10^{-5}$
Nb-95	$1.8x10^{-5}$	$1.8x10^{-5}$
Mo-99	$7.8x10^{-2}$	$1.4x10^{-1}$
Tc-99m	$5.7x10^{-2}$	$8.1x10^{-2}$
Ru-103	$8.2x10^{-6}$	$8.4x10^{-6}$
Ru-106	$5.7x10^{-7}$	$5.8x10^{-7}$
Rh-103m	$8.2x10^{-6}$	$8.4x10^{-6}$
Rh-106	$5.7x10^{-7}$	$5.8x10^{-7}$
Te-125m	$2.4x10^{-6}$	$2.4x10^{-6}$
Te-127m	$4.8x10^{-5}$	$4.9x10^{-5}$
Te-127	$3.4x10^{-5}$	$3.5x10^{-5}$
Te-129m	$9.8x10^{-4}$	$1.0x10^{-3}$
Te-129	$9.1x10^{-4}$	$9.2x10^{-4}$
Te-131m	$4.8x10^{-4}$	$5.0x10^{-4}$
Te-131	$1.1x10^{-4}$	$1.2x10^{-4}$
Te-132	$6.2x10^{-3}$	$6.4x10^{-3}$
Ba-137m	$3.0x10^{-2}$	$5.5x10^{-2}$
Ba-140	$1.0x10^{-4}$	$1.1x10^{-4}$
La-140	$4.4x10^{-5}$	$4.6x10^{-5}$
Ce-141	$1.8x10^{-5}$	$1.8x10^{-5}$
Ce-143	$1.1x10^{-5}$	$1.2x10^{-5}$
Ce-144	$9.1x10^{-6}$	$9.2x10^{-6}$
Pr-143	$1.6x10^{-5}$	$1.7x10^{-5}$
Pr-144	$9.1x10^{-6}$	$9.2x10^{-6}$
Np-239	$9.1x10^{-5}$	$9.2x10^{-5}$
Br-83	$2.8x10^{-3}$	$1.9x10^{-3}$
Br-84	$3.8x10^{-4}$	$3.4x10^{-4}$
Br-85	$5.4x10^{-6}$	$5.4x10^{-6}$
I-130	$5.2x10^{-4}$	$2.5x10^{-4}$

TABLE 11.2-7 (Cont)

Nuclide	BVPS-1 Releases	BVPS-2 Releases*
I-131	2.6×10^{-1}	1.1×10^{-1}
I-132	4.6×10^{-2}	3.2×10^{-2}
I-133	3.4×10^{-1}	1.6×10^{-1}
I-134	8.2×10^{-3}	7.1×10^{-4}
I-135	1.2×10^{-1}	6.9×10^{-2}
Rb-86	7.5×10^{-6}	7.0×10^{-6}
Rb-88	4.5×10^{-3}	4.5×10^{-3}
Cs-134	7.4×10^{-3}	1.3×10^{-2}
Cs-136	4.2×10^{-3}	6.9×10^{-3}
Cs-137	3.4×10^{-2}	5.9×10^{-2}
Total	9.7×10^{-1}	7.5×10^{-1}

NOTE:

*Some of the liquid releases generated as a result of BVPS-2 operation are included in BVPS-2 results even though they will be processed and discharged from BVPS-1. These include CVCS letdown releases.

**See Section 11.2.2

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.2-8 [HISTORICAL]

TURBINE BUILDING DRAINS CONCENTRATIONS
ROUTED TO LIQUID WASTE MANAGEMENT SYSTEM
DESIGN CASE (μCi/g)*****

Nuclide	BVPS-1 Concentrations	BVPS-2 Concentrations****	MPC*	Fraction**
Cr-51	7.2×10^{-10}	7.7×10^{-10}	2.0×10^{-3}	3.9×10^{-7}
Mn-54	1.3×10^{-10}	1.3×10^{-10}	1.0×10^{-4}	1.3×10^{-6}
Fe-55	6.3×10^{-10}	6.8×10^{-10}	8.0×10^{-4}	8.5×10^{-7}
Fe-59	4.0×10^{-10}	4.1×10^{-10}	5.0×10^{-5}	8.2×10^{-6}
Co-58	6.3×10^{-9}	6.8×10^{-9}	9.0×10^{-5}	7.6×10^{-5}
Co-60	7.7×10^{-10}	8.2×10^{-10}	3.0×10^{-5}	2.7×10^{-5}
Sr-89	4.8×10^{-10}	5.3×10^{-10}	3.0×10^{-6}	1.8×10^{-4}
Sr-90	1.2×10^{-11}	1.2×10^{-11}	3.0×10^{-7}	4.0×10^{-5}
Sr-91	1.5×10^{-10}	1.6×10^{-10}	5.0×10^{-5}	3.2×10^{-6}
Y-90	1.4×10^{-11}	2.4×10^{-11}	2.0×10^{-5}	1.2×10^{-6}
Y-91m	1.0×10^{-10}	7.7×10^{-11}	3.0×10^{-3}	2.6×10^{-8}
Y-91	7.7×10^{-11}	1.4×10^{-10}	3.0×10^{-5}	4.7×10^{-6}
Y-93	2.8×10^{-11}	3.9×10^{-11}	3.0×10^{-5}	1.3×10^{-6}
Zr-95	8.7×10^{-10}	8.7×10^{-10}	6.0×10^{-5}	1.5×10^{-5}
Nb-95	8.7×10^{-10}	8.7×10^{-10}	1.0×10^{-4}	8.7×10^{-6}
Mo-99	3.8×10^{-7}	6.8×10^{-7}	4.0×10^{-5}	1.7×10^{-2}
Tc-99m	2.8×10^{-7}	3.9×10^{-7}	3.0×10^{-3}	1.3×10^{-4}
Ru-103	4.0×10^{-11}	4.1×10^{-11}	8.0×10^{-5}	5.1×10^{-7}
Ru-106	2.8×10^{-12}	2.8×10^{-12}	1.0×10^{-5}	2.8×10^{-7}
Rh-103m	4.0×10^{-11}	4.1×10^{-11}	1.0×10^{-2}	4.1×10^{-9}
Rh-106	2.8×10^{-12}	2.8×10^{-12}	***	-
Te-125m	1.2×10^{-11}	1.2×10^{-11}	1.0×10^{-4}	1.2×10^{-7}
Te-127m	2.3×10^{-10}	2.4×10^{-10}	5.0×10^{-5}	4.8×10^{-6}
Te-127	1.6×10^{-10}	1.7×10^{-10}	2.0×10^{-4}	8.5×10^{-7}
Te-129m	4.7×10^{-9}	4.8×10^{-9}	2.0×10^{-5}	2.4×10^{-4}
Te-129	4.4×10^{-9}	4.4×10^{-9}	8.0×10^{-4}	5.5×10^{-6}
Te-131m	2.3×10^{-9}	2.4×10^{-9}	4.0×10^{-5}	6.0×10^{-5}
Te-131	5.3×10^{-10}	5.8×10^{-10}	3.0×10^{-6}	1.9×10^{-4}
Te-132	3.0×10^{-8}	3.1×10^{-8}	2.0×10^{-5}	1.6×10^{-3}
Ba-137m	1.4×10^{-7}	2.7×10^{-7}	***	-
Ba-140	4.8×10^{-10}	5.3×10^{-10}	2.0×10^{-5}	2.7×10^{-5}
La-140	2.1×10^{-10}	2.2×10^{-10}	2.0×10^{-5}	1.1×10^{-5}
Ce-141	8.7×10^{-11}	8.7×10^{-11}	9.0×10^{-5}	9.7×10^{-7}
Ce-143	5.3×10^{-11}	5.8×10^{-11}	4.0×10^{-5}	1.5×10^{-6}
Ce-144	4.4×10^{-11}	4.4×10^{-11}	1.0×10^{-5}	4.4×10^{-6}
Pr-143	7.7×10^{-11}	8.2×10^{-11}	5.0×10^{-5}	1.6×10^{-6}
Pr-144	4.4×10^{-11}	4.4×10^{-11}	***	-
Np-239	4.4×10^{-10}	4.4×10^{-10}	1.0×10^{-4}	4.4×10^{-6}
Br-83	1.4×10^{-8}	9.2×10^{-9}	3.0×10^{-6}	3.1×10^{-3}
Br-84	1.8×10^{-9}	1.6×10^{-9}	***	-
Br-85	2.6×10^{-11}	2.6×10^{-11}	***	-

TABLE 11.2-8 (Cont)

Nuclide	BVPS-1 Concentrations	BVPS-2 Concentrations****	MPC*	Fraction**
I-130	$2.5x10^{-9}$	$1.2x10^{-9}$	$3.0x10^{-6}$	$4.0x10^{-4}$
I-131	$1.3x10^{-6}$	$5.3x10^{-7}$	$3.0x10^{-7}$	1.8
I-132	$2.2x10^{-7}$	$1.5x10^{-7}$	$8.0x10^{-6}$	$1.9x10^{-2}$
I-133	$1.6x10^{-6}$	$7.7x10^{-7}$	$1.0x10^{-6}$	$7.7x10^{-1}$
I-134	$4.0x10^{-8}$	$3.4x10^{-9}$	$2.0x10^{-5}$	$1.7x10^{-4}$
I-135	$5.8x10^{-7}$	$3.3x10^{-7}$	$4.0x10^{-6}$	$8.3x10^{-2}$
Rb-86	$3.6x10^{-11}$	$3.4x10^{-11}$	$2.0x10^{-5}$	$1.7x10^{-6}$
Rb-88	$2.2x10^{-8}$	$2.2x10^{-8}$	***	-
Cs-134	$3.6x10^{-8}$	$6.3x10^{-8}$	$9.0x10^{-6}$	$7.0x10^{-3}$
Cs-136	$2.0x10^{-8}$	$3.3x10^{-8}$	$6.0x10^{-5}$	$5.5x10^{-4}$
Cs-137	$1.6x10^{-7}$	$2.9x10^{-7}$	$2.0x10^{-5}$	$1.5x10^{-2}$
Total	$4.7x10^{-6}$	$3.6x10^{-6}$		2.71

NOTES:

* Maximum permissible concentration (MPC) from 10 CFR 20 Appendix B
** Ratio of BVPS-2 concentration/MPC
*** No value given in 10 CFR 20
**** Some of the liquid releases generated as a result of BVPS-2 operation are included in BVPS-2 results even though they will be processed and discharged from BVPS-1. These include CVCS letdown releases.
***** See Section 11.2.2

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.2-9 [HISTORICAL]

LIQUID RADIOACTIVE SYSTEM
DESIGN ANNUAL RELEASES (Ci/yr)
FROM COOLING TOWER DISCHARGE POINT

Nuclide	BVPS-1 Releases	BVPS-2 Releases*	Total
Cr-51	$2.6x10^{-3}$	$1.5x10^{-4}$	$2.8x10^{-3}$
Mn-54	$5.9x10^{-4}$	$3.4x10^{-5}$	$6.2x10^{-4}$
Fe-55	$3.1x10^{-3}$	$1.8x10^{-4}$	$3.3x10^{-3}$
Fe-59	$1.6x10^{-3}$	$9.1x10^{-5}$	$1.7x10^{-3}$
Co-58	$2.7x10^{-2}$	$1.5x10^{-3}$	$2.9x10^{-2}$
Co-60	$3.9x10^{-3}$	$2.2x10^{-4}$	$4.1x10^{-3}$
Sr-89	$2.1x10^{-3}$	$1.2x10^{-4}$	$2.2x10^{-3}$
Sr-90	$5.7x10^{-5}$	$3.3x10^{-6}$	$6.0x10^{-5}$
Sr-91	$2.2x10^{-5}$	$8.4x10^{-6}$	$3.0x10^{-5}$
Y-90	$5.8x10^{-5}$	$3.5x10^{-6}$	$6.2x10^{-5}$
Y-91m	$1.5x10^{-5}$	$5.7x10^{-6}$	$2.1x10^{-5}$
Y-91	$3.4x10^{-4}$	$1.9x10^{-5}$	$3.6x10^{-4}$
Y-93	$4.1x10^{-6}$	$1.6x10^{-6}$	$5.7x10^{-6}$
Zr-95	$3.5x10^{-4}$	$2.0x10^{-5}$	$3.7x10^{-4}$
Nb-95	$4.1x10^{-4}$	$2.3x10^{-5}$	$4.3x10^{-4}$
Mo-99	$2.7x10^{-1}$	$4.2x10^{-2}$	$3.1x10^{-1}$
Tc-99m	$2.5x10^{-1}$	$3.7x10^{-2}$	$2.9x10^{-1}$
Ru-103	$1.6x10^{-4}$	$8.8x10^{-6}$	$1.7x10^{-4}$
Ru-106	$1.3x10^{-5}$	$7.6x10^{-7}$	$1.4x10^{-5}$
Rh-103m	$1.5x10^{-4}$	$8.7x10^{-6}$	$1.6x10^{-4}$
Rh-106	$1.3x10^{-5}$	$7.6x10^{-7}$	$1.4x10^{-5}$
Te-125m	$4.8x10^{-5}$	$2.7x10^{-6}$	$5.1x10^{-5}$
Te-127m	$1.1x10^{-3}$	$6.0x10^{-5}$	$1.2x10^{-3}$
Te-127	$1.0x10^{-3}$	$5.5x10^{-5}$	$1.1x10^{-3}$
Te-129m	$1.7x10^{-2}$	$9.5x10^{-4}$	$1.8x10^{-2}$
Te-129	$1.1x10^{-2}$	$6.0x10^{-4}$	$1.2x10^{-2}$
Te-131m	$8.4x10^{-4}$	$2.2x10^{-4}$	$1.1x10^{-3}$
Te-131	$1.6x10^{-4}$	$4.0x10^{-5}$	$2.0x10^{-4}$
Te-132	$2.5x10^{-2}$	$3.4x10^{-3}$	$2.8x10^{-2}$

TABLE 11.2-9 (Cont)

Nuclide	BVPS-1 Releases	BVPS-2 Releases*	Total
Ba-137m	1.3	$6.1x10^{-1}$	1.9
Ba-140	$1.2x10^{-3}$	$8.0x10^{-5}$	$1.3x10^{-3}$
La-140	$1.3x10^{-3}$	$5.4x10^{-5}$	$1.4x10^{-3}$
Ce-141	$3.0x10^{-4}$	$1.7x10^{-5}$	$3.2x10^{-4}$
Ce-143	$2.1x10^{-5}$	$5.1x10^{-6}$	$2.7x10^{-5}$
Ce-144	$2.1x10^{-4}$	$1.2x10^{-5}$	$2.2x10^{-4}$
Pr-143	$2.1x10^{-4}$	$1.3x10^{-5}$	$2.2x10^{-4}$
Pr-144	$2.1x10^{-4}$	$1.2x10^{-5}$	$2.2x10^{-4}$
Np-239	$2.7x10^{-4}$	$4.7x10^{-5}$	$3.2x10^{-4}$
Br-83	$2.4x10^{-4}$	$3.0x10^{-4}$	$5.4x10^{-4}$
Br-84	$2.2x10^{-5}$	$5.1x10^{-8}$	$2.2x10^{-5}$
Br-85	$2.7x10^{-7}$	0.0	$2.7x10^{-7}$
I-130	$2.3x10^{-4}$	$4.0x10^{-4}$	$6.3x10^{-4}$
I-131	$8.6x10^{-1}$	$4.1x10^{-1}$	1.3
I-132	$2.8x10^{-2}$	$1.0x10^{-2}$	$3.8x10^{-2}$
I-133	$2.4x10^{-1}$	$3.3x10^{-1}$	$5.7x10^{-1}$
I-134	$5.2x10^{-4}$	$3.1x10^{-5}$	$5.5x10^{-4}$
I-135	$2.8x10^{-2}$	$5.9x10^{-2}$	$8.7x10^{-2}$
Rb-86	$1.4x10^{-4}$	$5.0x10^{-5}$	$1.9x10^{-4}$
Rb-88	$1.0x10^{-3}$	$3.9x10^{-10}$	$1.0x10^{-3}$
Cs-134	$2.8x10^{-1}$	$1.4x10^{-1}$	$4.2x10^{-1}$
Cs-136	$6.1x10^{-2}$	$2.0x10^{-2}$	$8.1x10^{-2}$
Cs-137	1.3	$6.5x10^{-1}$	2.0
H-3	$5.5x10^{2}$	$5.5x10^{2}$	$1.1x10^{3}$
Total (except H-3)	4.7	2.3	7.0

NOTE:

* Some of the liquid releases generated as a result of BVPS-2 operation are included in BVPS-2 results even though they will be processed and discharged from BVPS-1. These include CVCS letdown releases.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.2-10 [HISTORICAL]

LIQUID RADIOACTIVE SYSTEM
DESIGN ANNUAL RELEASE CONCENTRATIONS (μCi/g)
FROM COOLING TOWER DISCHARGE POINT

Nuclide	BVPS-1 Release Conc	BVPS-2 Release Conc****	Combined Released Conc*****	MPC*	Fraction MPC for Combined Release Conc**
Cr-51	$8.6x10^{-11}$	$9.4x10^{-12}$	$6.2x10^{-11}$	$2.0x10^{-3}$	$3.1x10^{-8}$
Mn-54	$2.0x10^{-11}$	$2.2x10^{-12}$	$1.4x10^{-11}$	$1.0x10^{-4}$	$1.4x10^{-7}$
Fe-55	$1.0x10^{-10}$	$1.2x10^{-11}$	$7.3x10^{-11}$	$8.0x10^{-4}$	$9.1x10^{-8}$
Fe-59	$5.4x10^{-11}$	$5.9x10^{-12}$	$3.8x10^{-11}$	$5.0x10^{-5}$	$7.6x10^{-7}$
Co-58	$9.1x10^{-10}$	$9.9x10^{-11}$	$6.4x10^{-10}$	$9.0x10^{-5}$	$7.1x10^{-6}$
Co-60	$1.3x10^{-10}$	$1.4x10^{-11}$	$9.3x10^{-11}$	$3.0x10^{-5}$	$3.1x10^{-6}$
Sr-89	$6.9x10^{-11}$	$7.5x10^{-12}$	$4.9x10^{-11}$	$3.0x10^{-6}$	$1.6x10^{-5}$
Sr-90	$1.9x10^{-12}$	$2.1x10^{-13}$	$1.3x10^{-12}$	$3.0x10^{-7}$	$4.3x10^{-6}$
Sr-91	$7.4x10^{-13}$	$5.4x10^{-13}$	$6.6x10^{-13}$	$5.0x10^{-5}$	$1.3x10^{-8}$
Y-90	$2.0x10^{-12}$	$2.3x10^{-13}$	$1.4x10^{-12}$	$2.0x10^{-5}$	$7.0x10^{-8}$
Y-91m	$4.9x10^{-13}$	$3.7x10^{-13}$	$4.6x10^{-13}$	$3.0x10^{-3}$	$1.5x10^{-10}$
Y-91	$1.1x10^{-11}$	$1.2x10^{-12}$	$7.9x10^{-12}$	$3.0x10^{-5}$	$2.6x10^{-7}$
Y-93	$1.4x10^{-13}$	$1.0x10^{-13}$	$1.3x10^{-13}$	$3.0x10^{-5}$	$4.3x10^{-9}$
Zr-95	$1.2x10^{-11}$	$1.3x10^{-12}$	$8.2x10^{-12}$	$6.0x10^{-5}$	$1.4x10^{-7}$
Nb-95	$1.4x10^{-11}$	$1.5x10^{-12}$	$9.5x10^{-12}$	$1.0x10^{-4}$	$9.5x10^{-8}$
Mo-99	$9.0x10^{-9}$	$2.7x10^{-9}$	$6.8x10^{-9}$	$4.0x10^{-5}$	$1.7x10^{-4}$
Tc-99m	$8.4x10^{-9}$	$2.4x10^{-9}$	$6.4x10^{-9}$	$3.0x10^{-3}$	$2.1x10^{-6}$
Ru-103	$5.2x10^{-12}$	$5.7x10^{-13}$	$3.8x10^{-12}$	$8.0x10^{-5}$	$4.8x10^{-8}$
Ru-106	$4.4x10^{-13}$	$4.9x10^{-14}$	$3.1x10^{-13}$	$1.0x10^{-5}$	$3.1x10^{-8}$
Rh-103m	$5.2x10^{-12}$	$5.6x10^{-13}$	$3.5x10^{-12}$	$1.0x10^{-2}$	$3.5x10^{-10}$
Rh-106	$4.4x10^{-13}$	$4.9x10^{-14}$	$3.1x10^{-13}$	***	-
Te-125m	$1.6x10^{-12}$	$1.7x10^{-13}$	$1.1x10^{-12}$	$1.0x10^{-4}$	$1.1x10^{-8}$
Te-127m	$3.6x10^{-11}$	$3.9x10^{-12}$	$2.6x10^{-11}$	$5.0x10^{-5}$	$5.2x10^{-7}$
Te-127	$3.5x10^{-11}$	$3.6x10^{-12}$	$2.4x10^{-11}$	$2.0x10^{-4}$	$1.2x10^{-7}$
Te-129m	$5.6x10^{-10}$	$6.1x10^{-11}$	$4.0x10^{-10}$	$2.0x10^{-5}$	$2.0x10^{-5}$
Te-129	$3.6x10^{-10}$	$3.9x10^{-11}$	$2.6x10^{-10}$	$8.0x10^{-4}$	$3.3x10^{-7}$
Te-131m	$2.8x10^{-11}$	$1.4x10^{-11}$	$2.4x10^{-11}$	$4.0x10^{-5}$	$6.0x10^{-7}$
Te-131	$5.3x10^{-12}$	$2.6x10^{-12}$	$4.4x10^{-12}$	$3.0x10^{-6}$	$1.5x10^{-6}$
Te-132	$8.3x10^{-10}$	$2.2x10^{-10}$	$6.2x10^{-10}$	$2.0x10^{-5}$	$3.1x10^{-5}$
Ba-137m	$4.2x10^{-8}$	$3.9x10^{-8}$	$4.2x10^{-8}$	***	-
Ba-140	$4.2x10^{-11}$	$5.1x10^{-12}$	$2.9x10^{-11}$	$2.0x10^{-5}$	$1.5x10^{-6}$
La-140	$4.2x10^{-11}$	$3.5x10^{-12}$	$3.1x10^{-11}$	$2.0x10^{-5}$	$1.6x10^{-6}$
Ce-141	$9.9x10^{-12}$	$1.1x10^{-12}$	$7.1x10^{-12}$	$9.0x10^{-5}$	$7.9x10^{-8}$
Ce-143	$7.1x10^{-13}$	$3.3x10^{-13}$	$6.0x10^{-13}$	$4.0x10^{-5}$	$1.5x10^{-8}$
Ce-144	$7.0x10^{-12}$	$7.7x10^{-13}$	$4.9x10^{-12}$	$1.0x10^{-5}$	$4.9x10^{-7}$
Pr-143	$7.0x10^{-12}$	$8.3x10^{-13}$	$4.9x10^{-12}$	$5.0x10^{-5}$	$9.8x10^{-8}$
Pr-144	$7.0x10^{-12}$	$7.7x10^{-13}$	$4.9x10^{-12}$	***	-
Np-239	$9.2x10^{-12}$	$3.1x10^{-12}$	$7.1x10^{-12}$	$1.0x10^{-4}$	$7.1x10^{-8}$
Br-83	$8.2x10^{-12}$	$1.9x10^{-11}$	$1.2x10^{-11}$	$3.0x10^{-6}$	$4.0x10^{-6}$

TABLE 11.2-10 (Cont)

Nuclide	BVPS-1 Release Conc	BVPS-2 Release Conc****	Combined Released Conc*****	MPC*	Fraction MPC for Combined Release Conc**
Br-84	7.4×10^{-13}	3.3×10^{-15}	4.9×10^{-13}	***	-
Br-85	9.2×10^{-15}	0.0	6.0×10^{-15}	***	-
I-130	7.6×10^{-12}	2.6×10^{-11}	1.4×10^{-11}	3.0×10^{-6}	4.7×10^{-6}
I-131	2.9×10^{-8}	2.6×10^{-8}	2.9×10^{-8}	3.0×10^{-7}	9.7×10^{-2}
I-132	9.4×10^{-10}	6.7×10^{-10}	8.4×10^{-10}	8.0×10^{-6}	1.1×10^{-4}
I-133	8.2×10^{-9}	2.1×10^{-8}	1.3×10^{-8}	1.0×10^{-6}	1.3×10^{-2}
I-134	1.7×10^{-11}	2.0×10^{-12}	1.2×10^{-11}	2.0×10^{-5}	6.0×10^{-7}
I-135	9.2×10^{-10}	3.8×10^{-9}	1.9×10^{-9}	4.0×10^{-6}	4.8×10^{-4}
Rb-86	4.8×10^{-12}	3.3×10^{-12}	4.2×10^{-12}	2.0×10^{-5}	2.1×10^{-7}
Rb-88	3.4×10^{-11}	2.5×10^{-17}	2.2×10^{-11}	***	-
Cs-134	9.5×10^{-9}	8.8×10^{-9}	9.1×10^{-9}	9.0×10^{-6}	1.0×10^{-3}
Cs-136	2.1×10^{-9}	1.3×10^{-9}	1.8×10^{-9}	6.0×10^{-5}	3.0×10^{-5}
Cs-137	4.5×10^{-8}	4.2×10^{-8}	4.4×10^{-8}	2.0×10^{-5}	2.2×10^{-3}
H-3	1.9×10^{-5}	3.6×10^{-5}	2.4×10^{-5}	3.0×10^{-3}	8.0×10^{-3}
Total (except H-3)	1.6×10^{-7}	1.5×10^{-7}	1.6×10^{-7}		1.2×10^{-1}

NOTES:

* From 10 CFR 20 Appendix B.

** Ratio of combined release concentration/MPC.

*** No value given in 10 CFR 20.

**** Some of the liquid releases generated as a result of BVPS-2 operation are included in BVPS-2 results even though they will be processed and discharged from BVPS-1. These include CVCS letdown releases.

***** The BVPS-1 and BVPS-2 flow rates are different, therefore, the combined concentration does not equal the sum of BVPS-1 concentrations and BVPS-2 concentrations.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.2-11

TANK OVERFLOW PROTECTION

Tank	UFSAR Section No.	Level Monitoring and Alarms	Monitoring or Alarm Location	Overflow Provisions	Processing of Overflow
Refueling water storage tank	6.2.2	Recorder/ Indicator: high below normal low low-low	Main control room	Overflows to liquid waste system	Liquid waste system (LWS)
Primary component cooling water surge tanks	9.2.2	Indicator: high low	Main control room	Overflows to floor drains then to aux. bldg. sump	Aux. bldg. sump to LWS
Primary drains transfer tank (inside contain-ment)	9.3.3	Recorder/ indicator: high low	Main control room	Pressurized vessel-NA	NA
Primary drains transfer tank outside contain-ment)	9.3.3	Indicator: high-high low-low	Main control room	Pressurized vessel-NA	NA
Sludge tank (condensate polishing)	10.4.6	Indicator: (ultra-sonic) high	Control room (Condensate polishing bldg.)	Overflows to backwash hold tank	Condensate polishing bldg. sump to LWS
Backwash hold tank (condensate polishing)	10.4.6	Indicator: high	Control room (Condensate polishing bldg.)	Overflows to floor drains then to condensate polishing bldg. sump	Condensate Polishing bldg. sump to LWS
Backwash feed tank (condensate polishing)	10.4.6	Low	Control room (Condensate polishing bldg.)	Overflows to floor drains then to condensate polishing bldg. sump	Condensate polishing bldg. sump to LWS

TABLE 11.2-11 (Cont)

Tank	UFSAR Section No.	Level Monitoring and Alarms	Monitoring or Alarm Location	Overflow Provisions	Processing of Overflow
Liquid waste drain tanks	11.2	Indicator: high low	Main control room	Overflow to aux. bldg. sump via floor drain	Aux. bldg. sump to LWS
Steam generator blowdown evaporator test tanks	11.2	Indicator: high low	Main control room	Overflows to floor drains then to aux. bldg. sump	Aux. bldg. sump to LWS
Steam generator blowdown hold tanks	11.2	Indicator: high low	Main control room	Overflows to second tank then to floor drains then to waste handling bldg. sump	Waste handling bldg. Sump to LWS
Steam generator blowdown evaporator bottoms hold tank	11.2	Indicator: high	Main control room Control room (Condensate polishing bldg.)	Overflows to cubicle (cubicle has 6" curb at entrance)	Cubicle floor drain then to waste handling bldg. sump to LWS
Spent resin hold tank	11.4	Indicator: high low	Control room (Condensate polishing bldg.)	Overflows to decant tank	Refer to decant tank
Decant tank	11.4	Indicator: high-high high low	Control room (Condensate polishing bldg.)	Condensate polishing bldg. sump	Condensate polishing bldg. sump to LWS
Turbine plant demineralized water storage tank	10.4.7	Indicator: high low	Main control room	Overflows to turbine bldg. floor drains	Oil separator

TABLE 11.2-12

LIQUID WASTE SYSTEM COMPONENT DESIGN DATA

Component		Design Parameters
Steam Generator Blowdown Evaporator		
Quantity		2
Capacity, each (gpm)		20
Design pressure (psig/vacuum)		100/full
Design temperature (°F)		350
Operating pressure (psig)		15
Operating temperature (°F)		250
Material	Bottom:	Incoloy 825
	Top:	Type 316L stainless steel
Liquid Waste Drain Tank		
Quantity		2
Capacity, each (gal)		7,500
Design pressure (psig)		25
Design temperature (°F)		150
Operating pressure		Atmospheric
Operating temperature (°F)		100
Material		SA-240 Type 304L stainless steel
Steam Generator Blowdown Hold Tank		
Quantity		2
Capacity, each (gal)		50,000
Design pressure (psig)		0.5
Design temperature (°F)		200
Operating pressure		Atmospheric
Operating temperature (°F)		130
Material		A-240 Type 316 stainless steel
Steam Generator Blowdown Test Tank		
Quantity		2
Capacity, each (gal)		18,000
Design pressure (psig)		0.5
Design temperature (°F)		200
Operating pressure		Atmospheric
Operating temperature (°F)		130
Material		A-240 Type 304 stainless steel

TABLE 11.2-12 (Cont)

Component	Design Parameters
Steam Generator Blowdown Evaporator Distillate Accumulator	
Quantity	2
Capacity, each (gal)	200
Design pressure (psig/vacuum)	100/full
Design temperature (°F)	300
Operating pressure (psig)	15
Operating temperature (°F)	250
Material	SA-240 Type 304 stainless steel
Steam Generator Blowdown Evaporator Bottoms Hold Tank	
Quantity	1
Capacity (gal)	2,200
Design pressure (psig/vacuum)	50/full
Design temperature (°F)	260
Operating pressure	Atmospheric
Operating temperature (°F)	170
Material	SA-240 Type 316 stainless steel
Liquid Waste Drain Tank Pump	
Quantity	2
Type	Horizontal centrifugal
Motor (hp)	5
Seal type	Single mechanical
Capacity, each (gpm)	30
Head at rated capacity (ft)	114
Design pressure (psig)	200 @ 100°F
Material	
Pump casing	A-296 Gr. CF8M
Shaft	Type 316 stainless steel
Impeller	A-296 Gr. CF8M

TABLE 11.2-12 (Cont)

Component	Design Parameters
Steam Generator Blowdown Test Tank Pump	
Quantity	2
Type	Horizontal centrifugal
Motor (hp)	7.5
Seal type	Single mechanical
Capacity, each (gpm)	100
Head at rated capacity (ft)	118
Design pressure (psig)	200 @ 100°F
Materials	
Pump casing	A-296 Gr. CF8M
Shaft	Type 316 stainless steel
Impeller	A-296 Gr. CF8M
Steam Generator Blowdown Evaporator Bottoms Hold Tank Pump	
Quantity	1
Type	Horizontal centrifugal
Motor (hp)	2
Seal type	Double mechanical
Capacity (gpm)	50
Head at rated capacity (ft)	60
Design pressure (psig)	200 @ 100°F
Material	
Pump casing	A-296 Gr. CF8M
Shaft	Type 316 stainless steel
Impeller	A-296 Gr. CF8M
Steam Generator Blowdown Evaporator Bottoms Pump	
Quantity	2
Type	Horizontal centrifugal
Motor (hp)	2
Seal type	Double mechanical
Capacity, each (gpm)	20
Head at rated capacity (ft)	65
Design pressure (psig)	200 @ 100°F
Materials	
Pump casing	A-296 Gr. CF8M
Shaft	Type 316 stainless steel
Impeller	A-296 Gr. CF8M

TABLE 11.2-12 (Cont)

Component	Design Parameters
Steam Generator Blowdown Evaporator Bottoms Coolant Recirculation Pump	
Quantity	1
Type	Horizontal centrifugal
Motor (hp)	7.5
Seal type	Single mechanical
Capacity (gpm)	212
Head at rated capacity (ft)	72
Design pressure (psig)	200 @ 100°F
Materials	
Pump casing	A-296 Gr. CF8M
Shaft	Type 316 stainless steel
Impeller	A-296 Gr. CF8M
Steam Generator Blowdown Evaporator Distillate Pump	
Quantity	2
Type	Horizontal centrifugal
Motor (hp)	7.5
Seal type	Single mechanical
Capacity, each (gpm)	30
Head at rated capacity (ft)	139
Design pressure (psig)	200 @ 100°F
Materials	
Pump casing	A-296 Gr. CF8M
Shaft	Type 316 stainless steel
Impeller	A-296 Gr. CF8M
Steam Generator Blowdown Evaporator Feed Pump	
Quantity	2
Type	Horizontal centrifugal
Motor (hp)	5
Seal type	Single mechanical
Capacity, each (gpm)	50
Head at rated capacity (ft)	100
Design pressure (psig)	200 @ 100°F
Materials	
Pump casing	A-296 Gr. CF8M
Shaft	Type 316 stainless steel
Impeller	A-296 Gr. CF8M

TABLE 11.2-12 (Cont)

Component	Design Parameters
Steam Generator Blowdown Evaporator Circulation Pump	
Quantity	2
Type	Horizontal centrifugal
Motor (hp)	50
Seal type	Injected single mechanical
Capacity, each (gpm)	2,200
Head at rated capacity (ft)	64
Design pressure (psig)	225 @ 253°F
Materials	
Pump casing	A-296 Gr. CN7M
Shaft	Incoloy 825
Impeller	A-296 Gr. CN7M
Steam Generator Blowdown Evaporator Overhead Condenser	
Quantity	2
Duty, each (Btu/hr)	10,400,000

	Shell	Tube
Flow (lb/hr)	11,000	520,000
Design pressure (psig)	100	150
Design temperature (°F)	350	350
Operating pressure (psig)	15	60
Operating temperature, in/out (°F)	250/250	100/120
Material	Type 304 stainless steel	Type 304 stainless steel
Fluid	Saturated steam	Inhibited demin-eralized water

TABLE 11.2-12 (Cont)

Component	Design Parameters	
Steam Generator Blowdown Evaporator Reboiler		
Quantity	2	
Duty, each (Btu/hr)	12,330,000	
	Shell	Tube
Flow (lb/hr)	14,000	1,040,000
Design pressure (psig)	165	150
Design temperature (°F)	375	360
Operating pressure (psig)	100	15
Operating temperature, in/out (°F)	338/338	250/263
Material	Carbon steel	Incoloy 825
Fluid	Saturated steam	0-12% boric acid and other contamination
Steam Generator Blowdown Evaporator Distillate Cooler		
Quantity		2
Duty, each (Btu/hr)		1,200,000
	Shell	Tube
Flow (lb/hr)	60,000	9,950
Design pressure (psig)	150	150
Design temperature (°F)	340	340
Operating pressure (psig)	60	50
Operating temperature, in/out (°F)	100/120	250/129
Material	Type 304 stainless steel	Type 304 stainless steel
Fluid	Inhibited demineralized water	Water up to 60 ppm boric solution

TABLE 11.2-12 (Cont)

Component	Design Parameters	
Steam Generator Blowdown Evaporator Bottoms Cooler		
Quantity	1	
Total duty (Btu/hr)	1,143,000	
	Shell	Tube
Flow (lb/hr)	76,200	12,700
Design pressure (psig)	150	150
Design temperature (°F)	300	300
Operating pressure (psig)	60	45
Operating temperature, in/out (°F)	140/155	250/160
Material	Carbon steel	Type 316 stainless steel
Fluid	Inhib. demin. water pH 9-11	12% boric acid and other contamination
Cleanup Filter		
Quantity	1	
Retention size, (microns)	5 nominal	
Filter element type	Wound fiber	
Capacity, maximum (gpm)	150	
Design material, housing	Type 304 stainless steel	
Design pressure (psig)	200	
Design temperature (°F)	250	
Cleanup Ion Exchanger		
Quantity	2	
Design flow (gpm/ft^2)	21.2	
Resin type	H-OH	
Resin active volume (ft^3)	35	
Design pressure (psig)	175	
Design temperature (°F)	200	
Material	Type 304 stainless steel	

TABLE 11.2-13

EQUIPMENT LEAKAGE ESTIMATES*

Item	Leakage Rate Expected/Design***	Conc. (Fraction PCA)
Inside containment**	10 gpd/6 gpm	1.00
Pumps****	50 gpd/10 gpm	0.100
Valves*****	30 gpd/10 gpm	0.100
Fuel pit liner	700 gpd/4 gpm	0.001

NOTES:

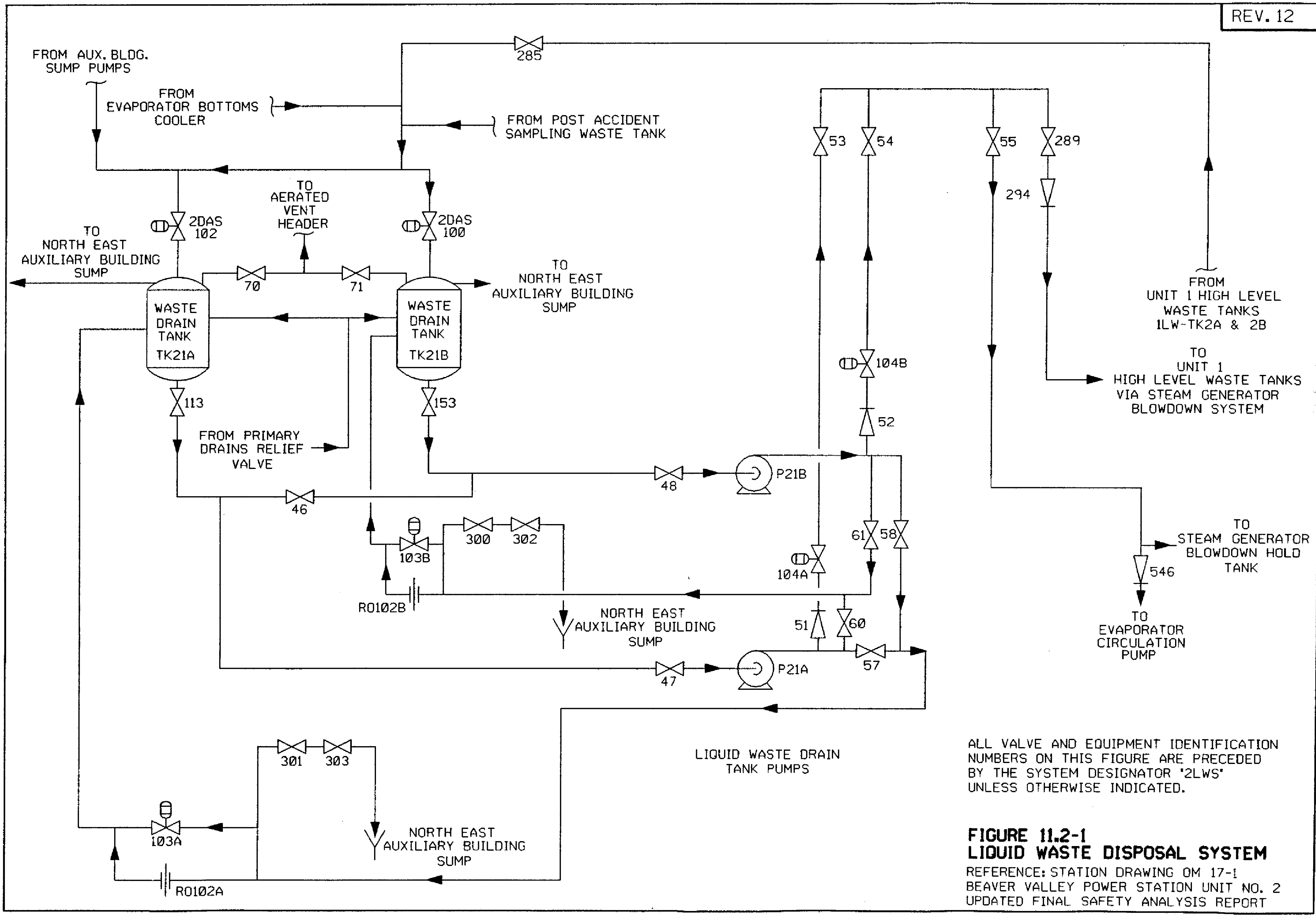
* Based upon values given in Table 6 of ANSI N199-1976.

** Design rate based on coolant leakage that does not flash. Expected rate based on valve packing leakage, flanges, and instrument connection leakage.

*** Design leakage rate duration is approximately 1 day, with exception of 30 days for fuel pit liner leakage.

**** Design rate based on seal failure.

***** Expected rate based on all valves outside containment leaking at a rate of 10 cc/hr/in stem diameter. Maximum rate based on packing failure.



ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR '2LWS' UNLESS OTHERWISE INDICATED.

FIGURE 11.2-1
LIQUID WASTE DISPOSAL SYSTEM

REFERENCE: STATION DRAWING OM 17-1
BEAVER VALLEY POWER STATION UNIT NO. 2
UPDATED FINAL SAFETY ANALYSIS REPORT

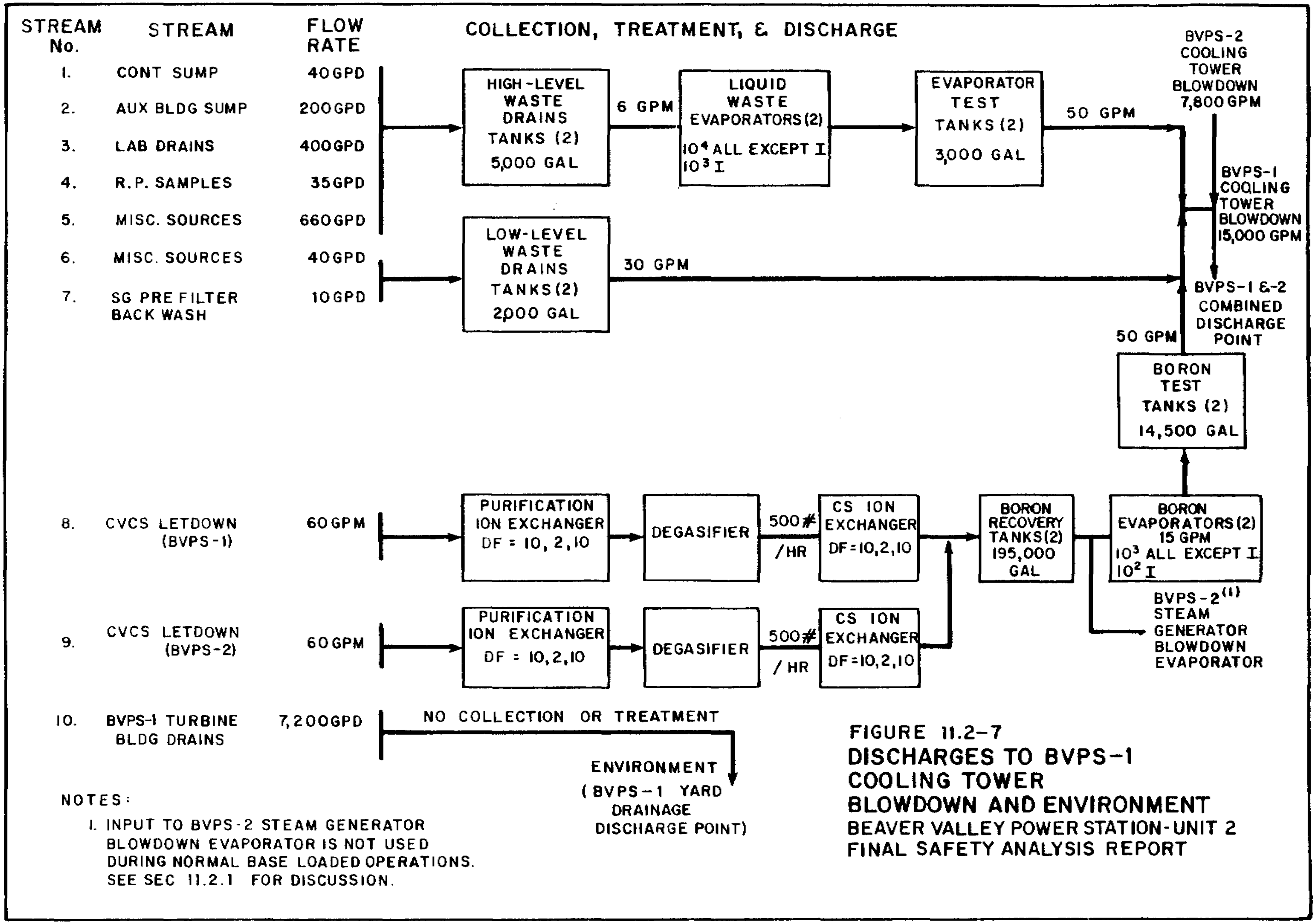
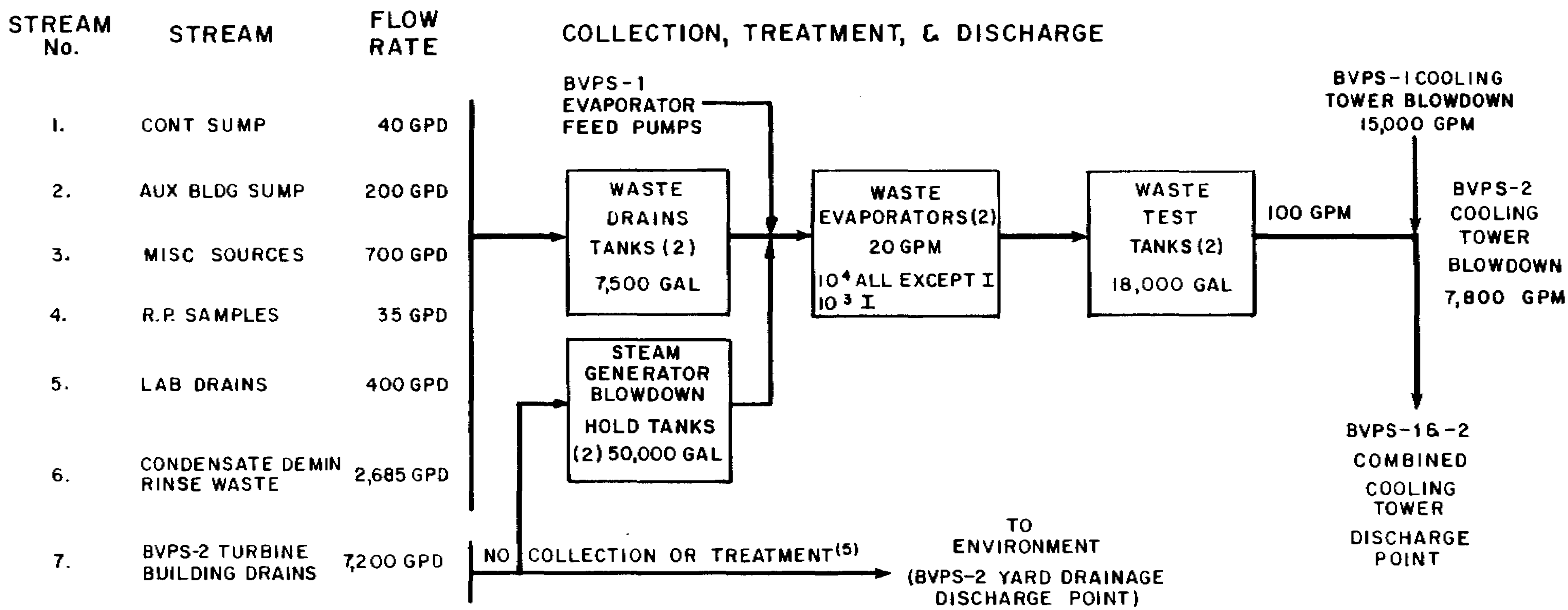


FIGURE 11.2-7
DISCHARGES TO BVPS-1 COOLING TOWER BLOWDOWN AND ENVIRONMENT
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



NOTES:

1. NO STEAM GENERATOR BLOWDOWN IS DISCHARGED.
2. THE WASTE EVAPORATORS ARE ACTUALLY TITLED STEAM GENERATOR BLOWDOWN EVAPORATORS.
3. DECONTAMINATION FACTORS STATED ARE FOR EVAPORATOR SYSTEM INCLUDING THE CLEAN-UP ION EXCHANGERS.
4. THERE IS NO EXPECTED INPUT FROM BVPS-1 EVAPORATOR FEED PUMPS DURING NORMAL BASE LOADED OPERATION. OPERATIONS IS DISCUSSED IN SECTION 11.2.1.
5. TURBINE BUILDING DRAINS ARE PROCESSED BY THE LIQUID WASTE SYSTEM UPON HIGH ACTIVITY.

FIGURE 11.2-8
DISCHARGES TO BVPS-2 COOLING TOWER BLOWDOWN AND ENVIRONMENT
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

Historical text, tables, values and parameters are identified as such and provide the historical basis for facility design.

This section describes the capabilities of Beaver Valley Power Station - Unit 2 (BVPS-2) to control, collect, process, store, recycle, and dispose of gaseous radioactive waste generated from normal operation and anticipated operational occurrences. Process and radiological monitoring systems are described in Section 11.5. The gaseous waste management systems include the radioactive gaseous waste disposal (GWD) system and ventilation systems.

The radioactive GWD system continuously processes and monitors all waste gas streams at Beaver Valley Power Station - Unit 1 (BVPS-1) and BVPS-2 prior to discharge to the atmosphere. The system provides decay time for the BVPS-2 boron recovery system (BRS) degasifier gaseous effluent (Section 9.3.4.6) and the offgas stream from the air ejector system (Section 10.4.2) for BVPS-1 and BVPS-2, as necessary. The provision for storing gases generated from either unit when going to cold shutdown is also provided. The system further provides for recycling the major portion of the hydrogen present in the degasifier overheads back to the volume control tank (VCT).

All gaseous waste effluent not recycled is directed to the BVPS-1 radioactive GWD system or the gaseous waste storage tanks (GWSTs) for discharge.

The aerated vent system for BVPS-2 is filtered before being discharged to the BVPS-1 radioactive GWD system.

The BVPS-2 radioactive GWD system is shown on Figures 11.3-1, 11.3-2, and 11.3-3.

The ventilation systems which service the safety-related and/or potentially contaminated areas described in Section 9.4 are designed to provide a suitable environment for personnel and equipment, and to prevent the spread of radioactive contamination. The ventilation systems which exhaust air from areas subject to radioactive contamination are designed with the provision for diverting the flow through the main filter banks in the supplementary leak collection and release system (SLCRS).

11.3.1 Design Bases

The GWD system and its associated ventilation systems are designed to meet the following criteria:

1. The systems are designed to meet the requirements of 10 CFR 20, and the dose design objectives specified in the Annex to Appendix I to 10 CFR 50, including provisions to treat gaseous radioactive wastes such that:

    a. The calculated annual total quantity of all radioactive material released from the site to the atmosphere does not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 mRems to the total body, or 15 mRems to the skin.

b. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form, released from the site to the atmosphere, does not result in an estimated annual dose or dose commitment for any individual in an unrestricted area from all pathways of exposure in excess of 15 mRems to any organ.

c. The instantaneous dose rates of radioactive materials in gaseous effluents released to an unrestricted area do not exceed the limits in the Offsite Dose Calculation Manual.

2. The radioactive GWD system is designed to meet the anticipated processing requirements of BVPS-2. Adequate capacity is provided to process gaseous wastes during periods when major processing equipment may be down for maintenance (single failures), and during periods of excessive waste generation.

3. The GWD system is classified non-nuclear (NNS) class and is designated QA Category II. The GWD system and structures housing the GWD system meet the seismic requirements of Regulatory Guide 1.143 with the exceptions noted in Section 1.8.

Tables 11.3-1 and 11.3-2 give expected annual quantities of radioactive material (by radionuclide) released, averaged over the life of BVPS-2, for BVPS-1 and BVPS-2, respectively. The above tables are historical and were developed in support of the original license. Similarly, Tables 11.3-3 and 11.3-4 are historical and give the design annual quantities of radioactive material for BVPS-1 and BVPS-2. The associated expected doses to individuals at or beyond the site boundary are given in Section 5.2 of the Environmental Report-Operating License Stage (ER-OLS).

Section 3.5.3 of the ER-OLS showed that the proposed systems met the numerical design objectives of Appendix I to 10 CFR 50.

The GWD system components and their design parameters and materials are given in Table 11.3-5.

A portion of the BRS (Section 9.3.4.6) is designed to process letdown from the chemical and volume control system (CVCS) (Section 9.3.4), or the primary drains transfer tanks (PDTT) in the nuclear equipment vent and drain system (Section 9.3.3), through either of the two degasifiers at a maximum rate of 75 gallons per minute (gpm) each. Maximum letdown from the CVCS is 120 gpm, and the maximum flow rate from the nuclear equipment drains system is no more than 50 gpm. Gaseous effluents from the degasifiers are directed to the GWD system.

The GWD system meets General Design Criteria 60 and 64 of Appendix A to 10 CFR 50, as discussed in Sections 3.1.2.60, 3.1.2.64, and 11.5.

The following design features are incorporated to minimize maintenance, equipment downtime, leakage, and radioactive gaseous releases in order to facilitate radwaste operation, and to assist in maintaining occupational exposures as low as is reasonably achievable.

1. Redundant degasifiers provide minimum outage (Section 9.3.4.6).

2. Components requiring servicing are placed in individual shielded cubicles to minimize personnel exposure during maintenance.

3. Leakage from pumps is piped to floor drains.

4. Corrosion-resistant materials are used throughout.

5. The radioactive GWD system can be operated remotely from the GWD control panel located in the auxiliary building.

6. A radiation monitor provides a signal that will automatically terminate all flow from the BVPS-1 decay tanks and BVPS-2 storage tanks should the radioactivity of the discharging stream exceed a predetermined level.

7. Automatic shutdown of the overhead gas compressors on abnormal oxygen concentration is provided to preclude the formation or buildup of explosive hydrogen/oxygen mixtures.

11.3.2 System Description

System design provides that all the gaseous effluent from the degasifiers is directed to the gaseous waste charcoal delay beds for decay of short-lived radioactive isotopes prior to compressing, with provisions to recycle the stripped gases back to the VCT. The compressed waste gas is directed to the gas decay tanks of the BVPS-1 radioactive GWD system for control of the equilibrium level of the coolant fission product gas inventory of krypton-85. Short-lived isotopes are controlled by decaying in the charcoal delay beds. The system also includes air ejector vent charcoal delay beds for treatment of the gaseous effluents, as necessary, from both BVPS-1 and BVPS-2 main condenser air ejector vents. During normal system operation the effluent from the BVPS-2 air ejectors are discharged through a flow path that is located downstream of the radiation monitor that is used to indicate whether the discharge is acceptable (Section 10.4.2.2). The treated gaseous effluent from the air ejector vent charcoal delay beds is filtered by the air ejector particulate filters. After filtration, it is sent to the BVPS-1 GWD system upstream of the GWD filters for discharge. The air from the aerated vent header is filtered by a vent filter and sent to BVPS-1, where it is filtered by the BVPS-1 GWD filters, prior to discharge via the process vent on top of the BVPS-1 cooling tower.

Figures 11.3-1, 11.3-2, and 11.3-3 show the system. Table 11.3-5 lists the capacities of the radioactive GWD system components.

Table 11.3-6 is historical and gives parameters used in calculating effluent release activities for both BVPS-1 and BVPS-2 in support of the original license. As noted earlier, Tables 11.3-1 and 11.3-2 provide the associated effluent releases from each source for BVPS-1 and BVPS-2, respectively and are also considered historical. Expected effluent calculations are based on NUREG-0017 (U.S. Nuclear Regulatory Commission 1976).

11.3.2.1 Degasifier Gaseous Effluent Portion of Gaseous Waste Disposal System

Reactor coolant letdown, containing dissolved hydrogen and fission gases, is normally directed to a degasifier from the letdown line upstream of the VCT in the CVCS. Liquid collected by the nuclear equipment vent and drain system (Section 9.3.3) is directed to the other degasifier. Dissolved gases are separated from the liquid in the degasifier.

The degasifiers of the BRS (Section 9.3.4.6) are designed to process reactor coolant letdown continuously. However, reactor coolant letdown may bypass the degasifiers to the BVPS-1 coolant recovery tanks, if desired. The degasifiers design flow of 150 gpm total (75 gpm each), exceeds the maximum expected throughout for the liquid portion of the process gas subsystem. Separation of dissolved gases at all reactor coolant letdown flow rates is thus ensured. The degasifier operates at a pressure of approximately 2 psig.

Effluent gases from the degasifier contain primarily hydrogen and water vapor. A small amount of nitrogen, and traces of xenon, krypton, and iodine, are also present in the effluent gases. These gases are dehumidified (dew point approximately 52°F) in one of the two sets of process exchangers (degasifier vent chiller in the BRS and gaseous waste chiller). Condensation effluent from the water trap is returned to the PDTT, located outside containment, via a liquid seal. The liquid seal is protected by a check valve and proper piping arrangement. The gas stream is passed through and filtered by the ambient temperature process gas charcoal bed adsorbers (cubicle temperature maintained at approximately 85°F) and one of two redundant prefilters. The heat due to radioactive decay is small and does not affect the adsorption of noble gases on the charcoal. The charcoal bed adsorbers are designed to delay xenon isotopes for a minimum of 30 days, and provide a 2 day delay for krypton isotopes for the normal letdown flow rates (for example, 60 gpm) when operated in series. In addition, decontamination of iodine to negligible levels is obtained during passage through the charcoal beds. The charcoal is divided evenly between two pairs of vertical tanks in series. The tanks are piped so that either pair may be bypassed, if necessary, with a corresponding decrease in decay time; however, normal flow is through the four beds in series. The only radioisotope present in any quantity in the predominantly hydrogen stream after the decay period is krypton-85.

One of the two overhead gas compressors directs the radioactive gas stream to a gas surge tank at a system pressure of about 65 psig. The gas may be reduced in pressure and returned to the VCT in the CVCS (Section 9.3.4). Periodically, the surge tank gas is bled to either the BVPS-1 gaseous waste decay tanks or the BVPS-2 GWSTs.

The compressors operate automatically in response to the suction pressure, thus maintaining the degasifier’s overhead components at a pressure between established limits.

The degasifier effluent portion of the GWD system is designed to include hermetically-sealed valves and welded pipe. In addition, the gas flow is monitored for oxygen content. At an indication of a high oxygen content, the compressors are shut off. These precautions are used to preclude potentially explosive mixtures of oxygen/hydrogen.

In the event of modes of fuel failure which might result in abnormal concentrations of fission products in the reactor coolant, storage space is provided in BVPS-1 decay tanks or BVPS-2 GWSTs. The tanks will be allowed to go to a higher holding pressure and will thus be able to accommodate a larger volume of gas. The higher pressure will not exceed the design pressure of the system.

11.3.2.2 Air Ejector Effluent Portion of Gaseous Waste Disposal System

The gaseous effluent stream from the main condenser air ejectors of BVPS-1 and BVPS-2 is directed, as necessary, to the air ejector vent charcoal delay beds which provide sufficient holdup for decay of short-lived radioactive components. Radiation monitoring and flow metering in the air ejector after condenser effluent discharge piping provide indication of the air ejector effluent release rate. The plant will be operated such that the dose limits of 10 CFR 50 Appendix I will not be exceeded.

Prior to entering the charcoal beds, the gas stream has a dew point of 55°F and the humidity is decreased by allowing the gas stream to heat up before entering the charcoal beds (cubicle maintained at approximately 77°F). Normally, the effluent from the air ejectors is not contaminated.

Gaseous effluent from both the BVPS-2 containment vacuum system and BVPS-2 aerated vents (from the vent and drain system) is directed to BVPS-1. It is passed through charcoal and high efficiency particulate air (HEPA) filters and repressurized prior to atmospheric discharge through the process vent on top of the BVPS-1 cooling tower. The effluent stream is monitored for flow rate, temperature, particulate, and gaseous radioactive effluent activity prior to this discharge.

11.3.2.3 Gaseous Waste Storage Tanks

The gaseous waste storage tankage is designed to handle all the gas generated by either BVPS-1 or BVPS-2 when going to a cold shutdown condition, from the following sources:

1. Noncondensible gases in the pressurizer steam space,

2. Hydrogen in the reactor coolant, and

3. Nitrogen used as an inert cover gas in the VCT.

Additional provisions are included to allow the unit which is operating to discharge to the BVPS-1 gaseous waste decay tanks while the tanks receive input from the preceding sources.

The system utilizes seven tanks with the ability for individual isolation. Pressure-relieving devices are provided on each tank. The total rated relieving capacity of each pressure-relief device is sufficient to prevent pressurization greater than 100 psi. These pressure-relieving devices discharge to the process vent release point on BVPS-1.

An off-line radiation monitor is provided to detect the contained activity in the tanks.

The discharge path from the tanks is routed via the BVPS-1 gaseous waste decay tanks discharge path. This path is maintained by a flow control valve and is provided with automatic isolation upon receiving a high radiation signal from the process vent final release radiation monitor.

11.3.2.4 Ventilation Systems

The following safety-related and/or potentially contaminated areas are provided with ventilation systems: control building (Section 9.4.1), fuel building (Section 9.4.2), auxiliary building (Section 9.4.3), waste handling building (Section 9.4.3), containment structure (Section 9.4.7), main steam valve area (Section 9.4.9), safeguards area (Section 9.4.11), cable vault and rod control area (Section 9.4.12), decontamination building (Section 9.4.13), and condensate polishing building (Section 9.4.16).

All radiation-controlled area ventilation is processed, if necessary, by HEPA and charcoal filters in BVPS-2 and released in the ventilation vent on top of the primary auxiliary building (Section 6.5.1) or the release point atop the BVPS-2 containment structure. An alternate containment purge line is provided which can route containment atmosphere to the BVPS-1 process vent subsystem, upstream of the GWD filters. Containment purge releases are normally exhausted from the ventilation vent. In the event of high containment concentrations, the containment purge would be routed out the BVPS-1 process vent. The lower release rate and better dispersion coefficient from the BVPS-1 process vent release point would keep site boundary concentrations below maximum permissible levels, as shown in Table 11.3-7 (Historical). Building ventilation systems are discussed in Section 9.4. Special containment ventilation systems to be used during refueling are discussed in Section 9.4.7.

11.3.2.5 Steam and Power Conversion Systems

The steam generator blowdown system (Section 10.4.8) is normally not a source of gaseous effluent. Blowdown is reduced in pressure, with flashed steam routed to a feedwater heater or the main condenser and cooled liquid routed to the condenser for treatment by the steam generator blowdown demineralizer(s).

The turbine gland seal system (Section 10.4.3) is a potential source of radioactivity. The effluent is monitored as described in Section 9.4.15.

11.3.3 Radioactive Releases

In support of the original license application, the expected and design releases for each potentially radioactive feed stream were presented in Tables 11.3-1, 11.3-2, 11.3-3, and 11.3-4 in curies per year per nuclide, for BVPS-1 and BVPS-2. Values used in these calculations are listed in Table 11.3-6 for BVPS-1 and BVPS-2. The above tables support the original license and are retained for historical purposes. Releases from the containment vacuum pump operation, the condensate polishing building vent, and the decontamination building vent, including the waste gas storage vault vent, are monitored and recorded, but are not significant, and are therefore not included in the tables.

Design gaseous releases were compared with the concentration limits of 10 CFR 20 in Tables 11.3-7, 11.3-9, and 11.3-10. Diffusion factors used are those given in Section 2.3.5. These tables support the original license and are retained for historical purposes.

Any gaseous waste effluent not recycled is directed to the BVPS-1 GWD system process vent for disposal. The process vent discharge is located at the top of the BVPS-1 circulating water cooling tower, approximately 500 feet above grade elevation. The release point inside diameter is 10 inches. The nominal exit velocity and maximum temperature are 2,000 ft/min and approximately 106°F, respectively.

Ventilation from the auxiliary building, waste handling building, fuel building, containment structure, and contiguous areas including the cable vault, rod control building, pipe tunnel, and the north and south safeguards area, is discharged through the SLCRS. The supplementary leak collection filter exhaust fans discharge through a duct to a release point 150 feet above grade. The release point is located on top of the containment structure. The leak collection release point inside diameter is 42 inches. A venturi effect is produced by a reduction in ductwork from a rectangular cross section of 48 x 54 inches to a circular exit point 42 inches in diameter. The nominal exit velocity is 6,100 ft/min and maximum temperature is approximately 139°F.

The doses due to these effluents did not exceed the numerical design objectives of Appendix I to 10 CFR 50 and the dose limits of 10 CFR 20. The calculated annual doses for the gaseous effluent releases are summarized in Appendix 11A. Appendix A supports the original license and is retained for historical purposes.

Scaling techniques, based on NUREG-0017, Revision 1 methodology, were utilized to assess the impact of core power uprate on radioactive gaseous effluents at BVPS. The conservatively performed power uprate analysis utilized the plant core power operating history during the years 1997 to 2001, the reported gaseous effluent and dose data during that period, NUREG-0017 equations and assumptions, and conservative methodology, to estimate the impact of operation at the analyzed uprate core power level of 2918 MWt, over that of operation at the previously licensed power level, on radioactive gaseous effluents and consequent normal operation off-site doses.

The licensed reactor core power level during the 1997 to 2001 time frame was 2652 MWt (note that for a portion of the year 2001, BVPS operated at a core power level of 2689 MWt, but the assessment conservatively assumed a core power level of 2652 MWt for the entire period). For the uprate condition, the system parameters utilized in the power uprate analysis reflected the flow rates and coolant masses at an analyzed NSSS power level of 2910 MWt and a core power level of 2918 MWt. For the pre-uprate condition, the evaluation utilized offsite doses based on an average 5 year set of organ and whole body doses calculated using data presented in the BVPS Annual Radioactive Effluent Release Reports for the years 1997 through 2001, taking into consideration the associated average annual core power level, extrapolated to 100 percent availability at the licensed power level.

Using the methodology and equations found in NUREG-0017, Revision 1, and based on a comparison of the change in power level and in plant coolant system parameters (e.g., reactor coolant mass, steam generator liquid mass, steam flow rate, reactor coolant letdown flow rate, flow rate to the cation demineralizer, letdown flow rate for boron control, steam generator blowdown flow rate, steam generator moisture carryover, etc.) for both pre-uprate and uprate conditions, the maximum potential percentage increase in coolant activity levels due to the uprate, for each chemical group identified in NUREG-0017, was estimated.

To estimate an upper bound impact on off-site doses, the highest factor found for any chemical group pertinent to the release pathway was applied to the average doses previously determined as representative of operation at pre-uprate conditions. This approach was utilized to estimate the maximum potential increase in effluent doses due to the uprate, and demonstrate that the estimated off-site doses following the uprate, although increased, will continue to remain below the regulatory limits set by 10CFR50, Appendix I.

It is noted that, for an operating plant such as BVPS, the actual gaseous effluent isotopic release curie and dose information are provided in the Annual Radioactive Effluent Release Reports.

11.3.4 Waste Gas System Failure

11.3.4.1 Identification of Causes and Descriptions

The waste gas system failure is classified as an American Nuclear Society (ANS) Condition III event, an incident which may occur during the lifetime of a plant.

The failure and subsequent release of radioactive gases from various points in the gaseous waste system (GWS) has been evaluated. The design basis for maintaining the integrity of the GWS is described in Section 11.3. The following locations were considered for evaluation: the failure of an inlet line to the waste gas charcoal delay beds and a rupture of one gaseous waste storage tank (Section 11.3).

Letdown flow is processed by the mixed bed demineralizers. One degasifier will process letdown flow at 60 gpm. The radionuclide inventories are based on reactor coolant equilibrium concentrations following operation with 1 percent failed fuel.

The design provisions provided to control the release of radioactive materials in the GWS meet General Design Criterion (GDC) 60 and are described in Section 11.3

11.3.4.2 Analysis of the Effects and Consequences

11.3.4.2.1 Method of Analysis

Evaluation of the waste gas system failure is based on Standard Review Plan (Section 11.3) Gaseous Waste Management System.

The fraction of the radioactivity released from the charcoal delay beds is based on the theoretical and experimental studies described by Underhill (1972).

11.3.4.3 Radiological Consequences

The radiological consequences for a postulated accidental release from the GWS are evaluated using the design assumptions listed in Table 11.3.4-1. For comparison purposes, the expected parameters are also listed in Table 11.3.4-1.

The inlet line rupture analysis assumes that the noble gas inventory released to the environment from the ruptured line is based on the sum of the following: a 1-hour release to the environment of degasifier effluent and the release of a fraction of the radioactivity absorbed on the charcoal delay beds. The fraction of noble gases released from the charcoal delay beds are given in Table 11.3.4-2.

For the GWST rupture, it is assumed that considering holdup and decay, the noble gases produced from the complete degasification of the primary coolant is contained in the seven GWSTs. Since the tanks are not isolated from each other, the rupture of one tank is assumed to cause the release of the contents of all seven tanks. The accident atmospheric dispersion values are given in Table 15.0-14.

The methodology used in computing the doses is discussed in Appendix 15A. The calculated whole body doses at the exclusion area boundary are presented in Table 15.0-12, and do not exceed 0.5 Rem.

11.3.5 Reference for Section 11.3

U.S. Nuclear Regulatory Commission 1976. Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Pressurized Water Reactors (PWR-GALE Code), NUREG-0017.

Tables for Section 11.3

TABLE 11.3-1 [HISTORICAL]

BVPS-1 GASEOUS WASTE MANAGEMENT SYSTEM AND VENTILATION SYSTEM, EXPECTED GASEOUS RELEASES (Ci/yr)

Nuclide	Containment Building	Auxiliary Building	Turbine Building	Main Condenser/ Air Ejector	Blowdown Flash Tank	Radioactive Gaseous Waste System	Total
Kr-83m	$2.2x10^{-2}$	$4.2x10^{-1}$	$3.9x10^{-5}$	$2.7x10^{-1}$	0.0	0.0	$7.1x10^{-1}$
Kr-85m	$1.5x10^{-1}$	1.9	$1.7x10^{-4}$	1.2	0.0	$7.3x10^{-2}$	3.3
Kr-85	$6.1x10^{1}$	2.5	$2.3x10^{-4}$	1.6	0.0	$2.3x10^{2}$	$3.0x10^{2}$
Kr-87	$5.4x10^{-2}$	1.3	$1.1x10^{-4}$	$8.2x10^{-1}$	0.0	0.0	2.2
Kr-88	$2.4x10^{-1}$	3.8	$3.5x10^{-4}$	2.4	0.0	0.0	6.4
Kr-89	$4.7x10^{-4}$	$1.2x10^{-1}$	$1.1x10^{-5}$	$7.7x10^{-2}$	0.0	0.0	$2.0x10^{-1}$
Xe-131m	$7.4x10^{-1}$	$1.3x10^{-1}$	$1.2x10^{-5}$	$8.0x10^{-2}$	0.0	1.3	2.2
Xe-133m	$8.9x10^{-1}$	$8.9x10^{-1}$	$8.1x10^{-5}$	$5.6x10^{-1}$	0.0	0.0	2.3
Xe-133	$8.9x10^{1}$	$3.6x10^{1}$	$3.4x10^{-3}$	$2.3x10^{1}$	0.0	$2.3x10^{1}$	$1.7x10^{2}$
Xe-135m	$4.5x10^{-3}$	$3.2x10^{-1}$	$2.9x10^{-5}$	$2.0x10^{-1}$	0.0	0.0	$5.2x10^{-1}$
Xe-135	$7.0x10^{-1}$	4.5	$4.2x10^{-4}$	2.8	0.0	0.0	7.9
Xe-137	$1.0x10^{-3}$	$2.1x10^{-1}$	$2.1x10^{-5}$	$1.3x10^{-1}$	0.0	0.0	$3.5x10^{-1}$
Xe-138	$1.5x10^{-2}$	1.1	$9.7x10^{-5}$	$6.6x10^{-1}$	0.0	0.0	1.7
I-131	$1.2x10^{-3}$	$4.6x10^{-2}$	$1.4x10^{-3}$	$1.6x10^{-2}$	0.0	0.0	$6.5x10^{-2}$
I-133	$2.0x10^{-4}$	$6.7x10^{-2}$	$1.7x10^{-3}$	$2.3x10^{-2}$	0.0	0.0	$9.2x10^{-2}$
Co-58	$7.5x10^{-4}$	$6.0x10^{-2}$	0.0	0.0	0.0	0.0	$6.1x10^{-2}$
Co-60	$3.4x10^{-4}$	$2.7x10^{-2}$	0.0	0.0	0.0	0.0	$2.7x10^{-2}$
Mn-54	$2.2x10^{-4}$	$1.8x10^{-2}$	0.0	0.0	0.0	0.0	$1.8x10^{-2}$
Fe-59	$7.5x10^{-5}$	$6.0x10^{-3}$	0.0	0.0	0.0	0.0	$6.1x10^{-3}$
Sr-89	$1.7x10^{-5}$	$1.3x10^{-3}$	0.0	0.0	0.0	0.0	$1.3x10^{-3}$
Sr-90	$3.0x10^{-6}$	$2.0x10^{-4}$	0.0	0.0	0.0	0.0	$2.0x10^{-4}$
Cs-134	$2.2x10^{-4}$	$1.8x10^{-2}$	0.0	0.0	0.0	0.0	$1.8x10^{-2}$
Cs-137	$3.8x10^{-4}$	$3.0x10^{-2}$	0.0	0.0	0.0	0.0	$3.0x10^{-2}$
C-14	1.0	0.0	0.0	0.0	0.0	7.0	8.0
Ar-41	$2.5x10^{1}$	0.0	0.0	0.0	0.0	0.0	$2.5x10^{1}$
H-3	$1.0x10^{2}$	$4.5x10^{2}$	0.0	0.0	0.0	0.0	$5.5x10^{2}$

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.3-2 [HISTORICAL]

BVPS-2 GASEOUS WASTE MANAGEMENT SYSTEM AND VENTILATION SYSTEM,
EXPECTED GASEOUS RELEASES (Ci/yr)

Nuclide	Containment Building	Auxiliary Building	Turbine Building	Main Condenser/ Air Ejector	Blowdown Flash Tank	Radioactive Gaseous Waste System	Total
Kr-83m	$4.0x10^{-5}$	$4.2x10^{-1}$	$3.9x10^{-5}$	$2.7x10^{-1}$	0.0	0.0	$6.9x10^{-1}$
Kr-85m	$1.4x10^{-2}$	1.9	$1.7x10^{-4}$	1.2	0.0	$1.2x10^{-2}$	3.1
Kr-85	$6.1x10^{1}$	2.5	$2.3x10^{-4}$	1.6	0.0	$2.3x10^{2}$	$3.0x10^{2}$
Kr-87	$5.3x10^{-6}$	1.3	$1.1x10^{-4}$	$8.2x10^{-1}$	0.0	0.0	2.1
Kr-88	$4.1x10^{-3}$	3.8	$3.5x10^{-4}$	2.4	0.0	0.0	6.2
Kr-89	0.0	$1.2x10^{-1}$	$1.1x10^{-5}$	$7.7x10^{-2}$	0.0	0.0	$2.0x10^{-1}$
Xe-131m	$7.2x10^{-1}$	$1.3x10^{-1}$	$1.2x10^{-5}$	$8.0x10^{-2}$	0.0	$8.3x10^{-1}$	1.8
Xe-133m	$7.6x10^{-1}$	$8.9x10^{-1}$	$8.1x10^{-5}$	$5.6x10^{-1}$	0.0	0.0	2.2
Xe-133	$8.4x10^{1}$	$3.6x10^{1}$	$3.4x10^{-3}$	$2.3x10^{1}$	0.0	8.2	$1.5x10^{2}$
Xe-135m	0.0	$3.2x10^{-1}$	$2.9x10^{-5}$	$2.0x10^{-1}$	0.0	0.0	$5.2x10^{-1}$
Xe-135	$2.4x10^{-1}$	4.5	$4.2x10^{-4}$	2.8	0.0	0.0	7.5
Xe-137	0.0	$2.1x10^{-1}$	$2.1x10^{-5}$	$1.3x10^{-1}$	0.0	0.0	$3.4x10^{-1}$
Xe-138	0.0	1.1	$9.7x10^{-5}$	$6.6x10^{-1}$	0.0	0.0	1.7
I-131	$7.7x10^{-4}$	$4.6x10^{-3}$	$6.5x10^{-4}$	$2.1x10^{-2}$	0.0	0.0	$2.7x10^{-2}$
I-133	$7.4x10^{-5}$	$6.7x10^{-3}$	$8.7x10^{-4}$	$3.0x10^{-2}$	0.0	0.0	$3.8x10^{-2}$
Co-58	$1.9x10^{-3}$	$6.0x10^{-4}$	0.0	0.0	0.0	0.0	$2.5x10^{-3}$
Co-60	$8.4x10^{-4}$	$2.7x10^{-4}$	0.0	0.0	0.0	0.0	$1.1x10^{-3}$
Mn-54	$5.5x10^{-4}$	$1.8x10^{-4}$	0.0	0.0	0.0	0.0	$7.3x10^{-4}$
Fe-59	$1.9x10^{-4}$	$6.0x10^{-5}$	0.0	0.0	0.0	0.0	$2.5x10^{-4}$
Sr-89	$4.2x10^{-5}$	$1.3x10^{-5}$	0.0	0.0	0.0	0.0	$5.5x10^{-5}$
Sr-90	$7.4x10^{-6}$	$2.0x10^{-6}$	0.0	0.0	0.0	0.0	$9.4x10^{-6}$
Cs-134	$5.5x10^{-4}$	$1.8x10^{-4}$	0.0	0.0	0.0	0.0	$7.3x10^{-4}$
Cs-137	$9.4x10^{-4}$	$3.0x10^{-4}$	0.0	0.0	0.0	0.0	$1.2x10^{-3}$
C-14	1.0	0.0	0.0	0.0	0.0	7.0	8.0
Ar-41	$2.5x10^{1}$	0.0	0.0	0.0	0.0	0.0	$2.5x10^{1}$
H-3	$1.0x10^{2}$	$4.5x10^{2}$	0.0	0.0	0.0	0.0	$5.5x10^{2}$

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.3-3 [HISTORICAL]

BVPS-1 GASEOUS WASTE MANAGEMENT SYSTEM AND VENTILATION SYSTEM, DESIGN CASE GASEOUS RELEASES (Ci/yr)

Nuclide	Containment Building	Auxiliary Building	Turbine Building	Main Condenser/ Air Ejector	Blowdown Flash Tank	Radioactive Gaseous Waste System	Total
Kr-83m	$4.9x10^{-1}$	9.1	$1.0x10^{-2}$	$6.8x10^{1}$	0.0	0.0	$7.7x10^{1}$
Kr-85m	4.1	$4.5x10^{1}$	$5.0x10^{-2}$	$3.3x10^{2}$	0.0	1.8	$3.8x10^{2}$
Kr-85	$6.1x10^{3}$	$2.3x10^{2}$	$2.6x10^{-1}$	$1.8x10^{3}$	0.0	$2.2x10^{4}$	$3.0x10^{4}$
Kr-87	1.1	$2.5x10^{1}$	$2.8x10^{-2}$	$1.9x10^{2}$	0.0	0.0	$2.2x10^{2}$
Kr-88	4.6	$6.8x10^{1}$	$7.6x10^{-2}$	$5.0x10^{2}$	0.0	$2.6x10^{-2}$	$5.7x10^{2}$
Kr-89	$8.9x10^{-3}$	2.1	$2.4x10^{-3}$	$1.5x10^{1}$	0.0	0.0	$1.7x10^{1}$
Xe-131m	$1.4x10^{1}$	2.3	$2.7x10^{-3}$	$1.8x10^{1}$	0.0	$2.4x10^{1}$	$5.8x10^{1}$
Xe-133m	$7.2x10^{1}$	$6.6x10^{1}$	$7.0x10^{-2}$	$4.9x10^{2}$	0.0	$4.3x10^{-2}$	$6.3x10^{2}$
Xe-133	$1.5x10^{3}$	$5.7x10^{2}$	$5.9x10^{-1}$	$4.3x10^{3}$	0.0	$3.9x10^{2}$	$6.3x10^{3}$
Xe-135m	$3.8x10^{-1}$	$2.3x10^{1}$	$2.5x10^{-2}$	$1.8x10^{2}$	0.0	0.0	$2.0x10^{2}$
Xe-135	$1.2x10^{1}$	$7.0x10^{1}$	$7.6x10^{-2}$	$5.2x10^{2}$	0.0	0.0	$6.0x10^{2}$
Xe-137	$1.7x10^{-2}$	3.4	$3.8x10^{-3}$	$2.5x10^{1}$	0.0	0.0	$2.8x10^{1}$
Xe-138	$2.2x10^{-1}$	$1.4x10^{1}$	$1.6x10^{-2}$	$1.1x10^{2}$	0.0	0.0	$1.2x10^{2}$
I-131	$1.1x10^{-2}$	$4.0x10^{-1}$	$1.7x10^{-1}$	1.7	0.0	0.0	2.3
I-133	$2.1x10^{-3}$	$6.4x10^{-1}$	$2.3x10^{-1}$	2.6	0.0	0.0	2.5
Co-58	$7.5x10^{-4}$	$6.0x10^{-2}$	0.0	0.0	0.0	0.0	$6.1x10^{-2}$
Co-60	$3.4x10^{-4}$	$2.7x10^{-2}$	0.0	0.0	0.0	0.0	$2.7x10^{-2}$
Mn-54	$2.2x10^{-4}$	$1.8x10^{-2}$	0.0	0.0	0.0	0.0	$1.8x10^{-2}$
Fe-59	$7.5x10^{-5}$	$6.0x10^{-3}$	0.0	0.0	0.0	0.0	$6.1x10^{-3}$
Sr-89	$1.4x10^{-4}$	$1.1x10^{-2}$	0.0	0.0	0.0	0.0	$1.1x10^{-2}$
Sr-90	$2.5x10^{-5}$	$1.7x10^{-3}$	0.0	0.0	0.0	0.0	$1.7x10^{-3}$
Cs-134	$1.8x10^{-3}$	$1.5x10^{-1}$	0.0	0.0	0.0	0.0	$1.5x10^{-1}$
CS-137	$3.2x10^{-3}$	$2.5x10^{-1}$	0.0	0.0	0.0	0.0	$2.5x10^{-1}$
C-14	1.0	0.0	0.0	0.0	0.0	7.0	8.0
Ar-41	$2.5x10^{1}$	0.0	0.0	0.0	0.0	0.0	$2.5x10^{1}$
H-3	$1.0x10^{2}$	$4.5x10^{2}$	0.0	0.0	0.0	0.0	$5.5x10^{2}$

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.3-4 [HISTORICAL]

BVPS-2 GASEOUS WASTE MANAGEMENT SYSTEM AND VENTILATION SYSTEM, DESIGN CASE GASEOUS RELEASES (Ci/yr)

Nuclide	Containment Building	Auxiliary Building	Turbine Building	Main Condenser/ Air Ejector	Blowdown Flash Tank	Radioactive Gaseous Waste System	Total
Kr-83m	9.1×10^{-4}	9.1	1.0×10^{-2}	6.8×10^{1}	0.0	0.0	7.7×10^{1}
Kr-85m	3.6×10^{-1}	4.5×10^{1}	4.9×10^{-2}	3.3×10^{2}	0.0	3.0×10^{-1}	3.8×10^{2}
Kr-85	6.1×10^{3}	2.3×10^{2}	2.6×10^{-1}	1.8×10^{3}	0.0	2.2×10^{4}	3.0×10^{4}
Kr-87	1.1×10^{-4}	2.5×10^{1}	2.8×10^{-2}	1.9×10^{2}	0.0	0.0	2.2×10^{2}
Kr-88	7.8×10^{-2}	6.8×10^{1}	7.6×10^{-2}	5.0×10^{2}	0.0	0.0	5.7×10^{2}
Kr-89	0.0	2.1	2.4×10^{-3}	1.5×10^{1}	0.0	0.0	1.7×10^{1}
Xe-131m	1.4×10^{1}	2.3	2.7×10^{-3}	1.8×10^{1}	0.0	1.5×10^{1}	4.9×10^{1}
Xe-133m	6.2×10^{1}	6.6×10^{1}	7.0×10^{-2}	4.9×10^{2}	0.0	0.0	6.2×10^{2}
Xe-133	1.5×10^{3}	5.7×10^{2}	5.9×10^{-1}	4.3×10^{3}	0.0	1.4×10^{2}	6.5×10^{3}
Xe-135m	0.0	2.3×10^{1}	2.5×10^{-2}	1.8×10^{2}	0.0	0.0	2.0×10^{2}
Xe-135	4.1	7.0×10^{1}	7.6×10^{-2}	5.2×10^{2}	0.0	0.0	5.9×10^{2}
Xe-137	0.0	3.4	3.8×10^{-3}	2.5×10^{1}	0.0	0.0	2.8×10^{1}
Xe-138	0.0	1.4×10^{1}	1.6×10^{-2}	1.1×10^{2}	0.0	0.0	1.2×10^{2}
I-131	7.3×10^{-3}	4.0×10^{-2}	7.0×10^{-2}	2.1	0.0	0.0	2.2
I-133	7.7×10^{-4}	6.4×10^{-2}	1.0×10^{-1}	3.4	0.0	0.0	3.6
Co-58	1.9×10^{-3}	6.0×10^{-4}	0.0	0.0	0.0	0.0	2.5×10^{-3}
Co-60	8.4×10^{-4}	2.7×10^{-4}	0.0	0.0	0.0	0.0	1.1×10^{-3}
Mn-54	5.5×10^{-4}	1.8×10^{-4}	0.0	0.0	0.0	0.0	7.3×10^{-4}
Fe-59	1.9×10^{-4}	6.0×10^{-5}	0.0	0.0	0.0	0.0	2.5×10^{-4}
Sr-89	3.5×10^{-4}	1.1×10^{-4}	0.0	0.0	0.0	0.0	4.6×10^{-4}
Sr-90	6.2×10^{-5}	1.7×10^{-5}	0.0	0.0	0.0	0.0	7.9×10^{-5}
Cs-134	4.5×10^{-3}	1.5×10^{-3}	0.0	0.0	0.0	0.0	6.0×10^{-3}
Cs-137	7.9×10^{-3}	2.5×10^{-3}	0.0	0.0	0.0	0.0	1.0×10^{-2}
C-14	1.0	0.0	0.0	0.0	0.0	7.0	8.0
Ar-41	2.5×10^{1}	0.0	0.0	0.0	0.0	0.0	2.5×10^{1}
H-3	1.0×10^{2}	4.5×10^{2}	0.0	0.0	0.0	0.0	5.5×10^{2}

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.3-5

GASEOUS WASTE DISPOSAL SYSTEM
COMPONENT DESIGN PARAMETERS

Component Characteristic	Design Parameters
Waste Gas Charcoal Delay Beds (2GWS-TK22A-D)	
Quantity	4
Capacity, height (max.) x diam. (ft x ft)	12 x 2.5
Weight of charcoal (lb) each	1,000
Operating pressure (psig)	2
Design pressure (psig)	100/full vacuum
Operating temperature (°F)	85
Design temperature (°F)	240
Material of construction	Carbon Steel
Overhead Gas Compressor Prefilters (2GWS-FLT24A&B)	
Quantity	2
Capacity (scfm) each	4.2
Operating pressure (psig)	2
Design pressure (psig)	150
Design temperature (°F)	200
Material of construction	Type 304 stainless steel
Waste Gas Chillers (2GWS-E21A&B)	
Quantity	2
Duty (Btu/hr) each	100
Tube side, degasified gas system (lb/hr) each	$<$ 11.4
Operating pressure (psig)	2
Design pressure (psig)	100
Operating temperature, in/out (°F)	100/52
Design temperature (°F)	240
Material of construction	Type 304 stainless steel
Shell side, chilled water (gpm) each	$<$ 1
Operating pressure (psig)	93
Design pressure (psig)	150
Operating temperature, in/out (°F)	45/55
Design temperature (°F)	150
Material of construction	Type 304 stainless steel

TABLE 11.3-5 (Cont)

Component Characteristic	Design Parameters
Overhead Gas Compressors (2GWS-C21A&B)	
Quantity	2
Capacity (scfm) each	2
Motor (hp) each	3
Operating pressure (psig) (discharge @ rated capacity)	65
Design pressure (psig)	220
Design temperature (°F) (inlet)	110
Material of construction	Type 304 stainless steel
Surge Tank (2GWS-TK21)	
Quantity	1
Capacity (ft^3)	52
Design pressure (psig)	110/full vacuum
Operating pressure (psig)	65
Design temperature (°F)	300
Operating temperature (°F)	100
Material of construction	Type 304 stainless steel
Surge Tank Discharge Filter (2GWS-FLT25)	
Quantity	1
Capacity (scfm)	4.2
Operating pressure (psig)	65
Design pressure (psig)	150
Design temperature (°F)	200
Material of construction	Type 304 stainless steel
Air Ejector Vent Chiller (2GWS-E22)	
Quantity	1
Duty (Btu/hr)	17,900
Tube side, chilled water (lb/hr)	3,580
Operating pressure (psig)	50
Design pressure (psig)	150pjs
Operating temperature, in/out (°F)	45/50
Design temperature (°F)	150
Material of construction	Type 304 stainless steel

TABLE 11.3-5 (Cont)

Component Characteristic	Design Parameters
Air Ejector Vent Chiller (2GWS-E22) (Cont)	
Shell Side, Air Ejector Discharge Gases (lb/hr)	196.2
Operating pressure (psig)	1.6
Design pressure (psig)	150
Operating temperature, in/out (°F)	125/55
Design temperature (°F)	150
Material of construction	Type 304 stainless steel
Air Ejector Vent Charcoal Delay Beds (2GWS-TK23A-M)	
Quantity	12
Capacity, height (max.) x diam. (ft x ft) each	14.5 x 3.5
Weight of charcoal (lb) each	3,000
Operating pressure (psig)	1.6
Design pressure (psig)	53
Operating temperature (°F)	77
Design temperature (°F)	150
Material of construction	Type 304 stainless steel
Air Ejector Vent Filters (2GWS-FLT23A&B)	
Quantity	2
Capacity (scfm) each	43
Operating pressure (psig)	1.6
Design pressure (psig)	50
Operating temperature (°F)	55-77
Design temperature (°F)	250
Material of construction	Type 304 stainless steel
Sweep Gas Chiller (2GWS-E23)	
Quantity	1
Duty (Btu/hr)	66,750
Tube side, saturated air (lb/hr)	618.5
Operating pressure (psig)	-0.9
Design pressure (psig)	100
Operating temperature, in/out (°F)	130/65
Design temperature (°F)	250
Material of construction	Type 304 stainless steel

TABLE 11.3-5 (Cont)

Component Characteristic	Design Parameters
Sweep Gas Chiller (2GWS-E23) (Cont)	
Shell side, chilled water (lb/hr)	6,675
Operating pressure (psig)	60
Design pressure (psig)	150
Operating temperature, in/out (°F)	45/55
Design temperature (°F)	250
Material of construction	Carbon steel
Vent Filter (2GWS-FLT22)	
Quantity	1
Capacity (scfm)	1,000
Operating pressure (in WG)	-25
Design pressure (in WG)	-35
Operating temperature (°F)	65
Design temperature (°F)	250
Material of construction	Type 304 stainless steel
Sweep Gas Blowers (2GWS-FN22A&B)	
Quantity	2
Capacity (scfm) each	125
Motor (hp) each	5
Static pressure (in WG)	34
Operating temperature (°F)	68-104
Material of construction	Type 316 stainless steel
Alternate Containment Purge Blower (2GWS-FN23)	
Quantity	1
Capacity (scfm)	1,000
Motor (hp)	7.5
Static pressure (in WG)	22.2
Operating temperature (°F)	90-105
Material of construction	Type 316 stainless steel
Gaseous Waste Storage Tanks (2GWS-TK25A-G)	
Quantity	7
Capacity (ft^3) each	132
Operating pressure (psig)	65
Design pressure (psig)	100
Operating temperature (°F)	180

TABLE 11.3-5 (Cont)

Component Characteristic	Design Parameters
Gaseous Waste Storage Tanks (2GWS-TK23A-G) (Cont)	
Design temperature (°F)	200
Material of construction	Carbon steel

TABLE 11.3-6 [HISTORICAL]

GASEOUS WASTE MANAGEMENT SYSTEMS,
EFFLUENT RELEASE PARAMETERS

Characteristic	Parameter BVPS-1	BVPS-2
Plant capacity factor	0.8	0.8
Containment building		
Noble gas release to containment building (fraction/day of primary coolant activity)	0.01	0.01
Iodine release to containment building (fraction/day of primary coolant activity)	10^{-5}	10^{-5}
Purge exhaust ventilation rate (cfm)	$3x10^4$	$3x10^4$
Purge exhaust ventilation time (hrs)	8	8
Recirculation rate during purge (cfm)	0	0
Iodine exhaust filter efficiency (%)	90	0*
Particulate exhaust filter efficiency (%)	99	0*
Number of hot purges/year	2	0
Number of cold purges/year	2	4
Continuous ventilation exhaust rate (cfm)	0	0
Free containment volume (ft^3)	$1.8x10^6$	$1.8x10^6$
Containment internal cleanup system		
Containment internal cleanup system operates prior to purging (hrs)	16	16
Containment internal cleanup system mixing efficiency (%)	70	70
Recirculation rate prior to purge (cfm)	$2x10^3$	$1x10^4$
Iodine filter efficiency (%)	90	90
Particulate filter efficiency (%)	0*	99

TABLE 11.3-6 (Cont)

Characteristic	Parameter BVPS-1	BVPS-2
Auxiliary building		
Iodine exhaust filter efficiency (%)	0*	90
Particulate exhaust filter efficiency (%)	0*	95
Primary coolant leakage rate into building (lb/day)	160	160
Iodine partition factor	7.5×10^{-3}	7.5×10^{-3}
Turbine building		
Iodine exhaust filter efficiency (%)	0*	0*
Particulate exhaust filter efficiency (%)	0*	0*
Steam leakage (lb/hr)	1,700	1,700
Main condenser/air ejector (MC/AE)		
Volatile iodine/total iodine in primary system	0.05	0.05
Volatile iodine is treated as noble gas in steam generator		
Primary to secondary leak rate (expected) (lb/day)	100**	100**
MC/AE volatile iodine partition factor	0.15	0.15
Volatile iodine condenser bypass fraction	0.44	0.27
Steam generator blowdown flash tank overheads are vented to the second point feedwater heaters which are vented and drained to the main condenser (included in above listed leak rate)		

TABLE 11.3-6 (Cont)

Characteristic	Parameter BVPS-1	Parameter BVPS-2
Radioactive gaseous waste system (process gas system)		
Letdown flow to degasifier (lb/hr)	$3x10^4$	$3x10^4$
Process gas released to the ATM	500	500
Holdup time prior to charcoal beds (minutes)	0	0
Krypton dynamic adsorption coefficient (cm^3/g)	16.0	18.5
Xenon dynamic adsorption coefficient (cm^3/g)	292	330
System flow rate to charcoal delay beds (cfm)	0.33	0.31
Total mass of charcoal in beds (lbs)	4,000	4,000
Charcoal depth (ft, each of 2 in series)	14	14
No iodine or particulates are released from system		
Krypton holdup time in delay bed (days)	2	2.6
Xenon holdup time in delay bed (days)	39	46
Number of complete primary system degasifications/year (complete degasification is handled by the same equipment as normal operation)	2	2

NOTES:

*Efficiency of zero indicates the actual parameter is not used in the release analysis.

**Design leak rate is 1,188 lb/day.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.3-7 [HISTORICAL]

DESIGN GASEOUS RELEASES
FROM PROCESS VENT
BVPS-1 AND BVPS-2

Nuclide	Concentration (Ci/cm^3)***	MPC (Ci/cm^3)*	Fraction MPC**
Kr-83m	$2.2x10^{-11}$	$3.0x10^{-8}$	$7.3x10^{-4}$
Kr-85m	$1.1x10^{-10}$	$1.0x10^{-7}$	$1.1x10^{-3}$
Kr-85	$7.7x10^{-8}$	$3.0x10^{-7}$	$2.6x10^{-1}$
Kr-87	$5.9x10^{-11}$	$2.0x10^{-8}$	$3.0x10^{-3}$
Kr-88	$1.6x10^{-10}$	$2.0x10^{-8}$	$7.8x10^{-3}$
Kr-89	$4.7x10^{-12}$	$3.0x10^{-8}$	$1.6x10^{-4}$
Xe-131m	$1.7x10^{-10}$	$4.0x10^{-7}$	$4.3x10^{-4}$
Xe-133m	$8.6x10^{-10}$	$3.0x10^{-7}$	$2.9x10^{-3}$
Xe-133	$1.8x10^{-8}$	$3.0x10^{-7}$	$6.1x10^{-2}$
Xe-135m	$5.6x10^{-11}$	$3.0x10^{-8}$	$1.9x10^{-3}$
Xe-135	$2.0x10^{-10}$	$1.0x10^{-7}$	$2.0x10^{-3}$
Xe-137	$7.8x10^{-12}$	$3.0x10^{-8}$	$2.6x10^{-4}$
Xe-138	$3.4x10^{-11}$	$3.0x10^{-8}$	$1.1x10^{-3}$
I-131	$6.7x10^{-13}$	$1.0x10^{-10}$	$6.7x10^{-3}$
I-132	$6.2x10^{-18}$	$3.0x10^{-9}$	$2.1x10^{-9}$
I-133	$9.4x10^{-13}$	$4.0x10^{-10}$	$2.4x10^{-3}$
I-134	$1.5x10^{-18}$	$6.0x10^{-9}$	$2.5x10^{-10}$
I-135	$4.4x10^{-17}$	$1.0x10^{-9}$	$4.4x10^{-8}$
Co-58	$2.2x10^{-14}$	$2.0x10^{-9}$	$1.1x10^{-5}$
Co-60	$9.5x10^{-15}$	$3.0x10^{-10}$	$3.2x10^{-5}$
Mn-54	$6.2x10^{-15}$	$1.0x10^{-9}$	$6.2x10^{-6}$
Fe-55	$1.1x10^{-15}$	$2.0x10^{-9}$	$5.7x10^{-7}$
Sr-89	$4.0x10^{-15}$	$3.0x10^{-10}$	$1.3x10^{-5}$
Sr-90	$7.0x10^{-16}$	$3.0x10^{-11}$	$2.3x10^{-5}$
Cs-134	$5.1x10^{-14}$	$4.0x10^{-10}$	$1.3x10^{-4}$
Cs-137	$9.0x10^{-14}$	$5.0x10^{-10}$	$1.8x10^{-4}$
C-14	$1.4x10^{-11}$	$1.0x10^{-7}$	$1.4x10^{-4}$
Ar-41	$2.8x10^{-10}$	$4.0x10^{-8}$	$7.1x10^{-3}$
H-3	$1.1x10^{-9}$	$2.0x10^{-7}$	$5.7x10^{-3}$

NOTES:

* Maximum permissible concentration from 10 CFR 20, Appendix B.
** Ratio of release concentration to MPC.
*** Includes containment purge releases for BVPS-2.
The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.3-9 [HISTORICAL]

BVPS-2
DESIGN GASEOUS RELEASES FROM
ELEVATED RELEASE

Nuclide	Concentration ($\mu Ci/cm^3$)	MPC ($\mu Ci/cm^3$)*	Fraction MPC**
Kr-83m	$2.7x10^{-11}$	$3.0x10^{-8}$	$8.9x10^{-4}$
Kr-85m	$1.3x10^{-10}$	$1.0x10^{-7}$	$1.3x10^{-3}$
Kr-85	$6.7x10^{-10}$	$3.0x10^{-7}$	$2.3x10^{-3}$
Kr-87	$7.3x10^{-11}$	$2.0x10^{-8}$	$3.7x10^{-3}$
Kr-88	$2.0x10^{-10}$	$2.0x10^{-8}$	$1.0x10^{-2}$
Kr-89	$6.2x10^{-12}$	$3.0x10^{-8}$	$2.1x10^{-4}$
Xe-131m	$6.7x10^{-12}$	$4.0x10^{-7}$	$1.7x10^{-5}$
Xe-133m	$1.9x10^{-10}$	$3.0x10^{-7}$	$6.4x10^{-4}$
Xe-133	$1.7x10^{-9}$	$3.0x10^{-7}$	$5.6x10^{-3}$
Xe-135m	$6.7x10^{-11}$	$3.0x10^{-8}$	$2.3x10^{-3}$
Xe-135	$2.1x10^{-10}$	$1.0x10^{-7}$	$2.1x10^{-3}$
Xe-137	$1.0x10^{-11}$	$3.0x10^{-8}$	$3.3x10^{-4}$
Xe-138	$4.1x10^{-11}$	$3.0x10^{-8}$	$1.4x10^{-3}$
I-131	$1.2x10^{-13}$	$1.0x10^{-10}$	$1.2x10^{-3}$
I-133	$1.9x10^{-13}$	$4.0x10^{-10}$	$4.7x10^{-4}$
Co-58	$1.8x10^{-15}$	$2.0x10^{-9}$	$8.8x10^{-7}$
Co-60	$7.9x10^{-16}$	$3.0x10^{-10}$	$2.6x10^{-6}$
Mn-54	$5.3x10^{-16}$	$1.0x10^{-9}$	$5.3x10^{-7}$
Fe-59	$1.8x10^{-16}$	$2.0x10^{-9}$	$8.8x10^{-8}$
Sr-89	$3.2x10^{-16}$	$3.0x10^{-10}$	$1.1x10^{-6}$
Sr-90	$5.0x10^{-17}$	$3.0x10^{-11}$	$1.7x10^{-6}$
Cs-134	$4.4x10^{-15}$	$4.0x10^{-10}$	$1.1x10^{-5}$
Cs-137	$7.3x10^{-15}$	$5.0x10^{-10}$	$1.5x10^{-5}$
C-14	0.0	$1.0x10^{-7}$	0.0
Ar-41	0.0	$4.0x10^{-8}$	0.0
H-3	$1.3x10^{-9}$	$2.0x10^{-7}$	$6.6x10^{-3}$

NOTES:

* Maximum permissible concentration from 10 CFR 20, Appendix B.
** Ratio of BVPS-2 concentration to MPC.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.3-10 [HISTORICAL]

BVPS-2
DESIGN GASEOUS RELEASES FROM
TURBINE BUILDING VENT

Nuclide	Concentration ($\mu Ci/cm^3$)	MPC ($\mu Ci/cm^3$)*	Fraction MPC**
Kr-83m	$3.0x10^{-14}$	$3.0x10^{-8}$	$1.0x10^{-6}$
Kr-85m	$1.5x10^{-13}$	$1.0x10^{-7}$	$1.5x10^{-6}$
Kr-85	$7.7x10^{-13}$	$3.0x10^{-7}$	$2.6x10^{-6}$
Kr-87	$8.3x10^{-14}$	$2.0x10^{-8}$	$4.2x10^{-6}$
Kr-88	$2.3x10^{-13}$	$2.0x10^{-8}$	$1.2x10^{-5}$
Kr-89	$7.1x10^{-15}$	$3.0x10^{-8}$	$2.4x10^{-7}$
Xe-131m	$8.0x10^{-15}$	$4.0x10^{-7}$	$2.0x10^{-8}$
Xe-133m	$2.1x10^{-13}$	$3.0x10^{-7}$	$6.9x10^{-7}$
Xe-133	$1.8x10^{-12}$	$3.0x10^{-7}$	$5.8x10^{-6}$
Xe-135m	$7.4x10^{-14}$	$3.0x10^{-8}$	$2.5x10^{-6}$
Xe-135	$2.3x10^{-13}$	$1.0x10^{-7}$	$2.3x10^{-6}$
Xe-137	$1.1x10^{-14}$	$3.0x10^{-8}$	$3.7x10^{-7}$
Xe-138	$4.7x10^{-14}$	$3.0x10^{-8}$	$1.6x10^{-6}$
I-131	$2.1x10^{-13}$	$1.0x10^{-10}$	$2.1x10^{-3}$
I-133	$3.0x10^{-13}$	$4.0x10^{-10}$	$7.4x10^{-4}$

NOTES:

* Maximum permissible concentration from 10 CFR 20, Appendix B.
** Ratio of BVPS-2 concentration to MPC.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.3.4-1

PARAMETERS USED FOR THE
WASTE GAS SYSTEM FAILURE

Power (MWt)	2,918
Fraction of fuel with defects	0.01
Letdown flow (gpm)	135
Charcoal delay beds holdup times (hrs)	
Kr	9.6
Xe	174
Duration of release for the inlet line rupture (hrs)	1

TABLE 11.3.4-2

FRACTIONS OF NOBLE GASES
RELEASED FROM CHARCOAL DELAY BEDS

Nuclide	Expected - Design
Kr-83m	0.650
Kr-85m	0.353
Kr-85	0.0444
Kr-87	0.786
Kr-88	0.502
Kr-89	1.000
Xe-131m	0.00739
Xe-133m	0.0361
Xe-133	0.0156
Xe-135m	1.000
Xe-135	0.195
Xe-137	1.000
Xe-138	1.000

TABLE 11.3.4-3

POTENTIAL DOSES FROM WASTE GAS SYSTEM RUPTURE (WGSR)

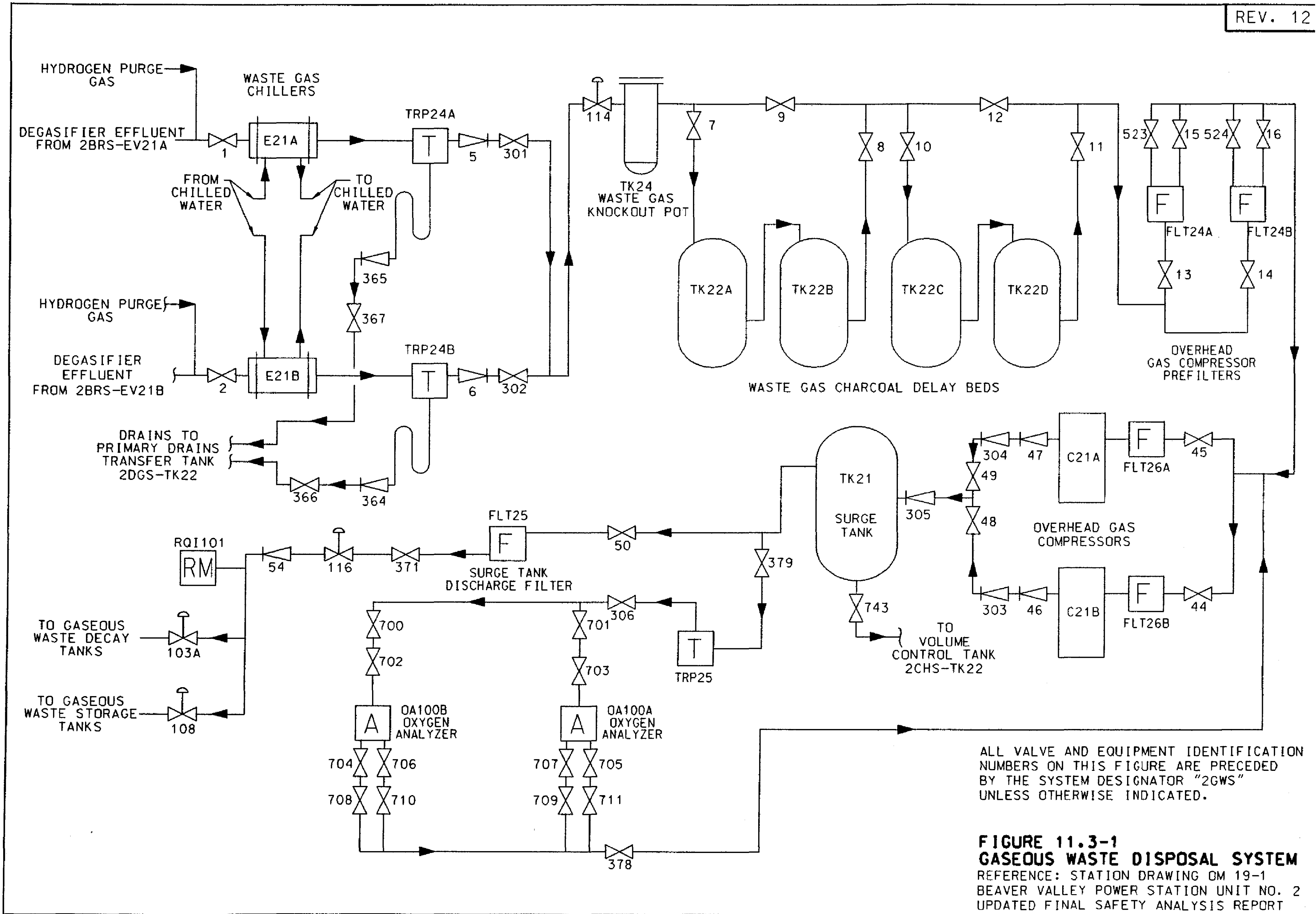
	EAB Whole Body Dose (rem)
Tank Rupture	3.99E-2
Line Rupture	2.99E-1

	LPZ Whole Body Dose (rem)
Tank Rupture	1.93E-3
Line Rupture	1.44E-2

	CR Whole Body Dose (rem)
Tank Rupture	1.91E-3
Line Rupture	7.37E-3

	CR Skin Dose (rem)
Tank Rupture	4.16
Line Rupture	4.74E-1

REV. 12



ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR "2GWS" UNLESS OTHERWISE INDICATED.

FIGURE 11.3-1
GASEOUS WASTE DISPOSAL SYSTEM
REFERENCE: STATION DRAWING OM 19-1
BEAVER VALLEY POWER STATION UNIT NO. 2
UPDATED FINAL SAFETY ANALYSIS REPORT

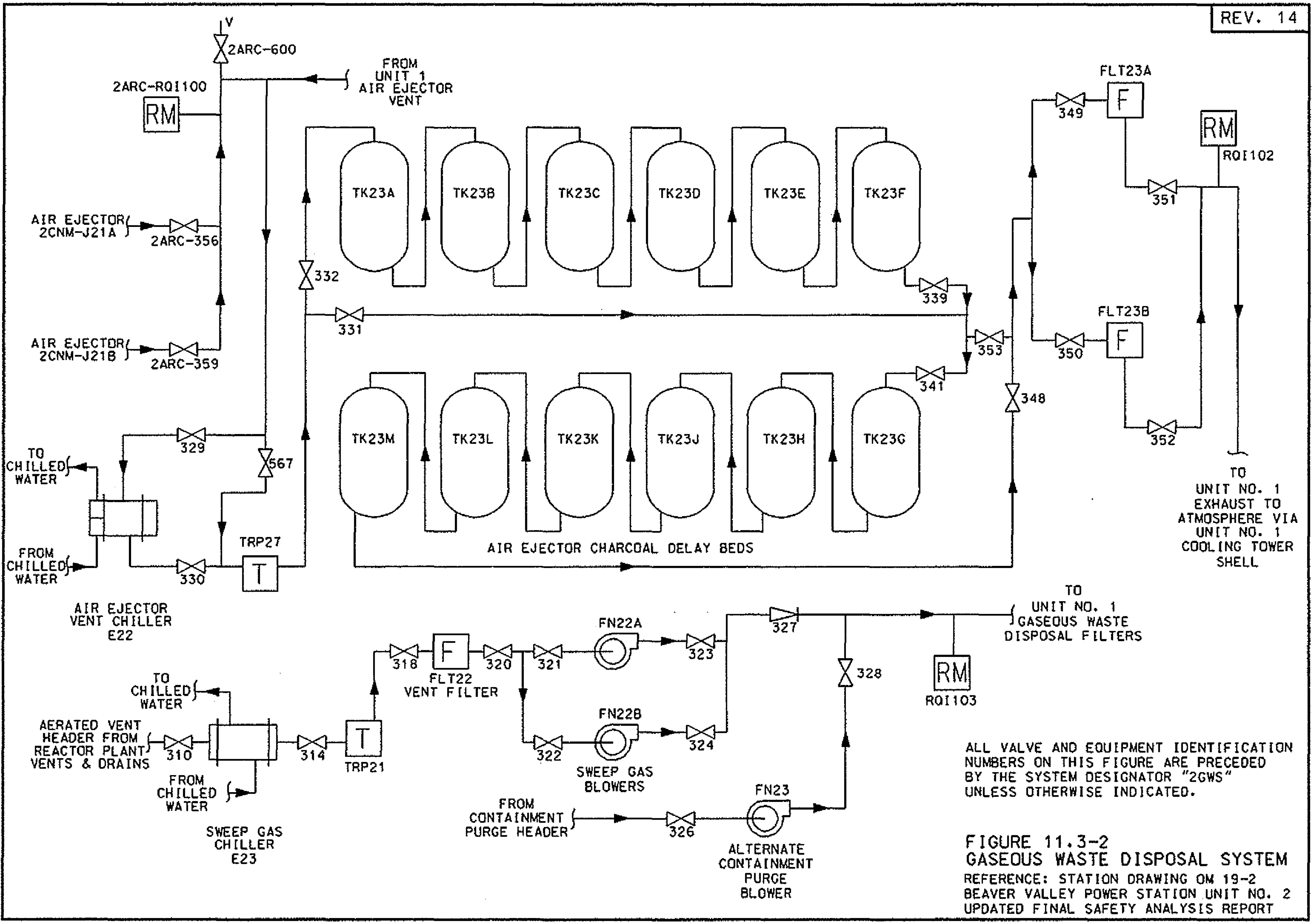
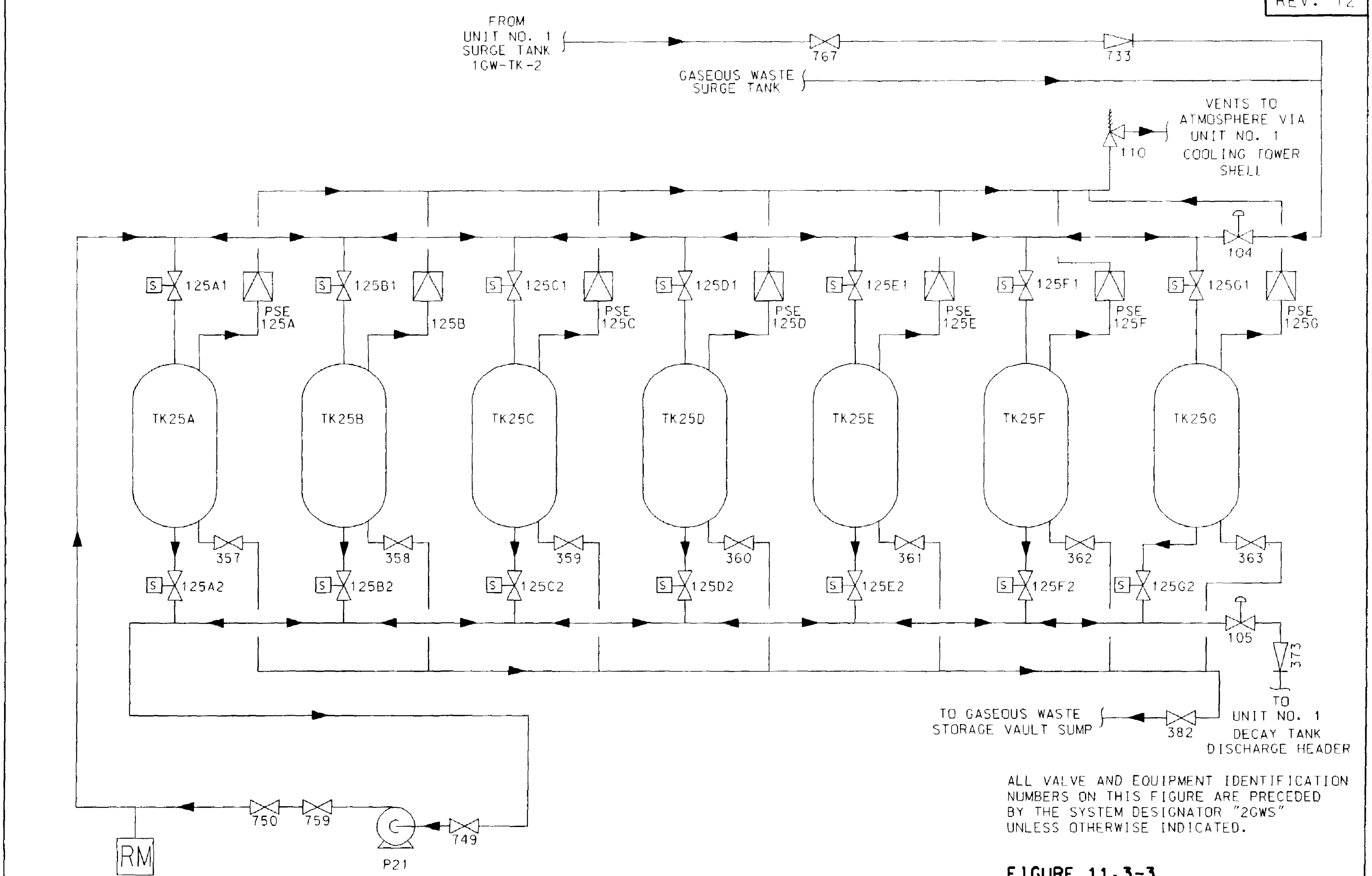


FIGURE 11.3-2
GASEOUS WASTE DISPOSAL SYSTEM
REFERENCE: STATION DRAWING OM 19-2
BEAVER VALLEY POWER STATION UNIT NO. 2
UPDATED FINAL SAFETY ANALYSIS REPORT



ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR "2GWS" UNLESS OTHERWISE INDICATED.

FIGURE 11.3-3
GASEOUS WASTE STORAGE TANKS

REFERENCE: STATION DRAWING OM 19-3
BEAVER VALLEY POWER STATION UNIT NO. 2
UPDATED FINAL SAFETY ANALYSIS REPORT

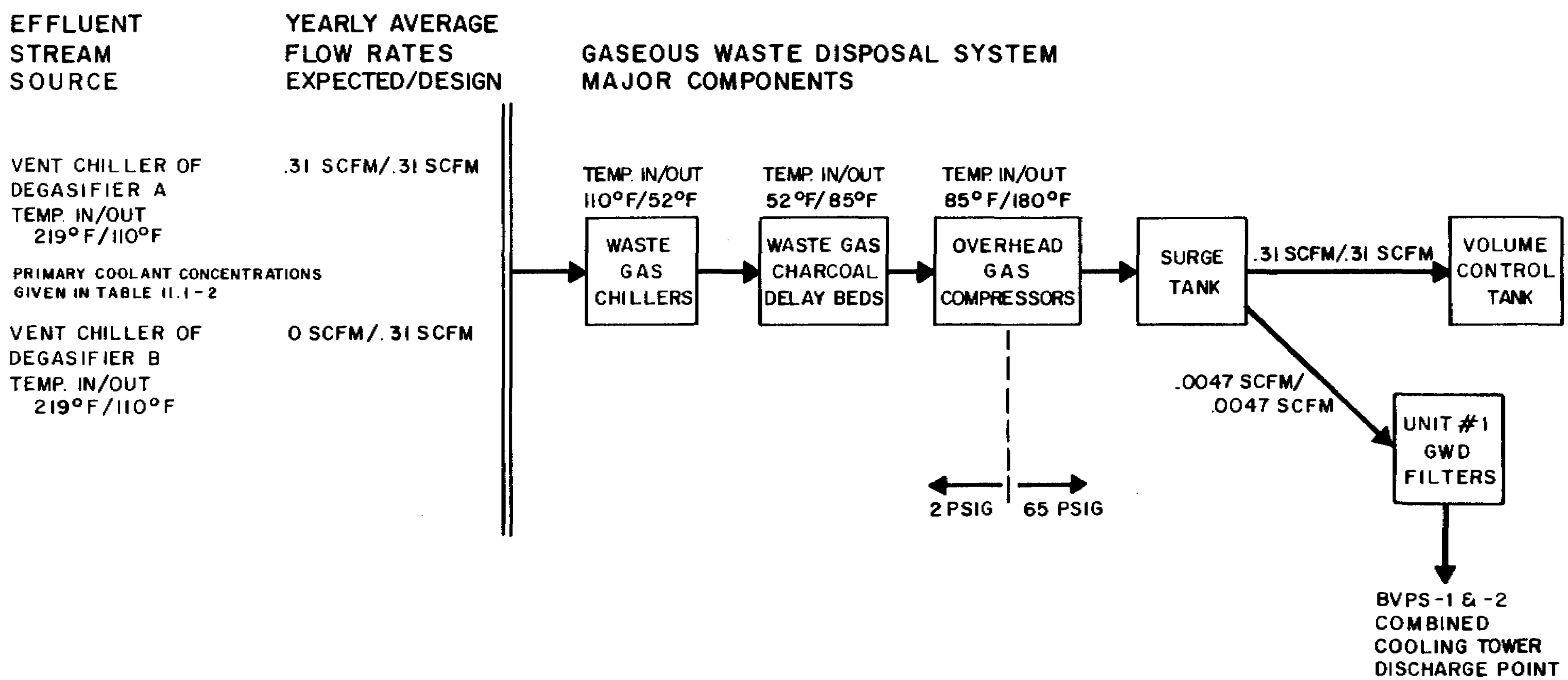
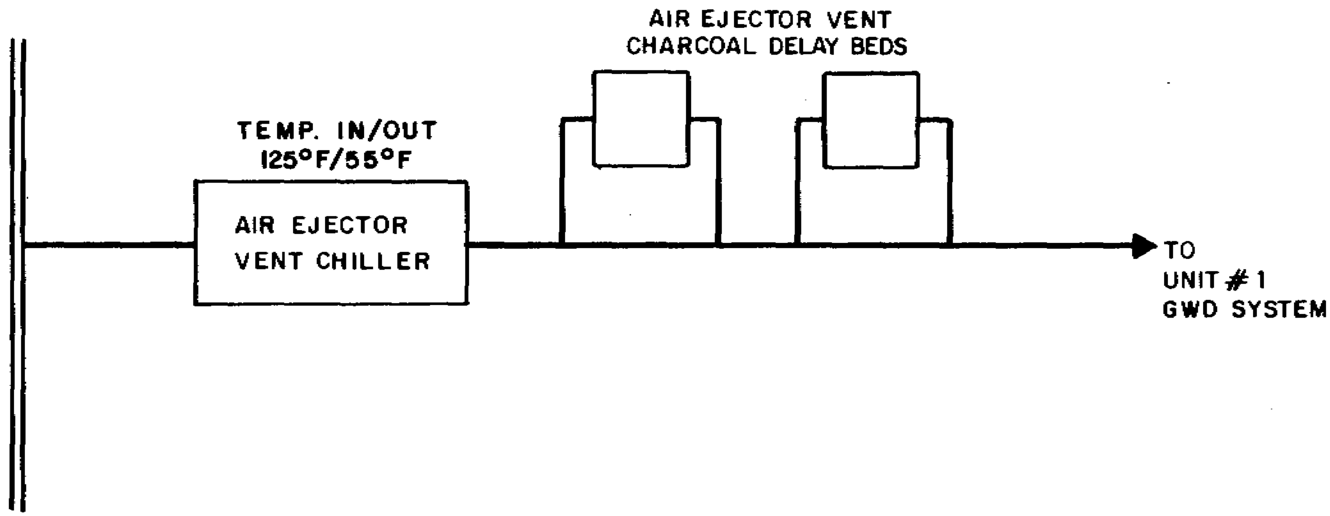


FIGURE 11.3-4
DEGASIFIER GASEOUS EFFLUENT PORTION OF THE GASEOUS WASTE SYSTEM
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

EFFLUENT STREAM SOURCE	EXPECTED FLOW RATES (SEE NOTE 1)
#1 UNIT #1 AIR EJECTOR	5 SCFM
SECONDARY COOLANT CONCENTRATIONS GIVEN IN TABLE 11.1-6 AND TABLE 11.1-7	
#2 UNIT #2 AIR EJECTOR	5 SCFM



NOTES:

1. EXPECTED AIR INLEAKAGE FOR EACH UNIT IS 5 SCFM
2. EXPECTED SYSTEM OPERATING PRESSURE IS 1.6 PSIG

FIGURE 11.3-5
AIR EJECTOR EFFLUENT PORTION
OF THE GASEOUS WASTE SYSTEM
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

11.4 SOLID RADWASTE MANAGEMENT SYSTEM

The solid waste management system provides facilities for the collection and preparation of radioactive waste materials for shipment to processing and disposal facilities. The various waste streams are prepared for shipment by filtration, dewatering, solidification, segregation, compaction, packaging, and/or storage.

The information provided in this section relative to the design and expected volume and radioactivity levels of solid radwaste generated is based on a core power level of 2766 MWt. It was developed in support of the original license and is retained herein for historical purposes.

Core power uprate to 2900 MWt will not appreciably impact equipment performance, nor will it require drastic changes in system operation or maintenance. Consequently, only minor changes, if any, are expected in the volume of waste generated.

However, it is expected that the activity levels for most of the solid waste would increase proportionately to the increase in inventory of long half-life isotopes in the coolant, which is expected to be bounded by the percentage of the uprate.

11.4.1 Design Bases

The solid radwaste system is designed in accordance with the following criteria:

1. The system design parameters are based on radionuclide concentrations and volumes consistent with reactor operating experience for similar designs, and with the source terms of Section 11.1.

2. All radwastes that are wet and being shipped directly to a burial facility will be solidified or dewatered prior to their shipment offsite. A process control program and operating procedures are used to verify the absence of free liquid in the containers. Provisions exist to reprocess containers having free liquid in accordance with Branch Technical Position (BTP) ETSB 11-3 (USNRC 1981). Processing equipment is sized to handle the design condition inputs without the need to ship bulk liquids. Onsite waste storage facilities provide sufficient storage capacity to allow time for shipping delays. Storage capacity will exceed 30 days' accumulation at expected generation rates in accordance with BTP ETSB 11-3.

3. Solid waste containers, shipping casks, and methods of packaging meet applicable federal regulations (10 CFR 71 and 49 CFR 171 through 178). Wastes to be shipped to a licensed burial site are in accordance with applicable U.S. Nuclear Regulatory Commission, U.S. Department of Transportation (USDOT), and state regulations.

4. The solid radwaste system components and piping are designed in accordance with the criteria of BTP ETSB 11-3 and Regulatory Guide 1.143, except as noted in Section 1.8, BVPS-2 position on Regulatory Guide 1.143.

5. The system contains provisions to control leakage and to facilitate operations and maintenance in accordance with BTP ETSB 11-3 and Regulatory Guides 1.143 and 8.8.

6. The filling of containers, the solidification, and the storage of radioactive solid wastes conform to 10 CFR 20 and 10 CFR 50, Appendix A, General Design Criteria 60, 63 and 64. In addition, 10 CFR 50.34a requirements and Regulatory Guide 8.8 guidelines are met to ensure occupational radiation exposures are as low as is reasonably achievable (ALARA).

7. The waste storage facilities in the condensate polishing building are shielded to provide protection for operating personnel in accordance with the radiation protection design considerations in Section 12.1.2. Solidification agents are stored in accordance with BTP ETSB 11-3.

8. The system is not safety-related and is classified as non-nuclear safety class, and QA Category II.

9. That part of the foundation and adjacent walls of the condensate polishing building up to a height sufficient to contain the liquid inventory in the building is designed to the seismic criteria of Paragraph 5, and to the quality assurance criteria of Paragraph 6, of Regulatory Guide 1.143. The specific building design criteria are given in Section 3.8.

10. The solid waste system and the structures housing the solid waste system meet the seismic requirements of Regulatory Guide 1.143 with the exceptions noted in Section 1.8, BVPS-2 position on Regulatory Guide 1.143.

11. Those portions of the system that handle liquid radwaste meet the applicable design bases of Section 11.2.1.

12. The system is designed to process, handle, and store the waste types and quantities generated as a result of normal operation, including anticipated operational occurrences, as described in Section 11.4.2.

11.4.2 System Description

Solid radwaste system components are listed in Table 11.4-1, which also summarizes design and operating conditions.

Beaver Valley Power Station - Unit 1 (BVPS-1) and Beaver Valley Power Station - Unit 2 (BVPS-2) radwaste systems have interconnecting piping for additional capability. This allows BVPS-2 radwaste to be dewatered or solidified, as an option, by BVPS-1.

11.4.2.1 System Inputs

The materials which are handled as radioactive solid waste include depleted resins from process ion exchangers, concentrated waste solutions from the BVPS-2 evaporators, radioactive powdered resin sludges, spent filter cartridges, and miscellaneous contaminated or irradiated solid materials (other than fuel).

Figures 11.4-3 and 11.4-4 are flow charts of the expected and design quantities and the activity levels of the solid radwaste generated by the unit. Gross activities and weights or volumes for solid radwaste sources are also shown on Figures 11.4-3 and 11.4-4.

The principal nuclides and isotopic inventories in the solid radwaste system are presented in Table 11.4-2 and 11.4-3 for the expected and design operating conditions, respectively.

The prediction of the nuclide concentrations for the solid radwaste isotopic inventories is based on the following variables:

1. The rate of transport, within the plant systems, of the insoluble fractions of radioactive corrosion products is one to two orders of magnitude slower than for soluble fission products, and is more sensitive to changes than to steady operation.

2. The normal expected isotopic mix of solid waste from predictive mathematical and empirical models are compared to data giving isotopic mixtures in solid waste.

11.4.2.1.1 Spent Resins

Figures 11.4-3 and 11.4-4 estimate the volume of spent demineralizer and ion exchanger resins generated per year in the expected and design cases. These resins come from a variety of different services, and the total volume estimated is based on the individual resin bed volumes and the frequency of replacement (Tables 11.4-4 and 11.4-5).

As indicated earlier, the figures and tables referenced above were developed in support of the original license and are considered historical.

11.4.2.1.2 Evaporator Bottoms

The evaporators in the liquid radwaste system can be operated to discharge bottoms, at an expected concentration of 12 percent by weight solids, to the solid radwaste system for solidification and ultimate shipment offsite. The volume of bottoms solution to be shipped offsite is shown on Figures 11.4-3 and 11.4-4. The calculated activity of these bottoms, also shown on Figures 11.4-3 and 11.4-4, is based on the inputs of the liquid radwaste system (Section 11.2).

As indicated earlier, the figures referenced above were developed in support of the original license and are considered historical.

11.4.2.1.4 Miscellaneous Radioactive Solid Wastes

Based on expected and design operating conditions, it is estimated that, on an annual basis, 6,700 ft^3 and 11,900 ft^3 of additional waste, respectively, would be generated. These volumes, consisting of spent filters, contaminated clothing, and other radioactive material and their activities are shown on Figures 11.4-3 and 11.4-4, and were estimated from operating experience at other similar nuclear facilities.

As indicated earlier, the figures referenced above were developed in support of the original license and are considered historical.

11.4.2.2 Equipment Description

The solid radwaste system equipment is operated on a batch basis, and the equipment capabilities are able to meet design rates throughout. The solidification equipment is estimated to be able to process two containers (e.g.: 55 gallon drums) per hour.

The solidification system consists of a drumming station mechanism (drum processor), a power panel, control console, decant tank, sampling equipment, cement metering and weighing equipment, caustic buffering tank, and the piping, pumps, and process equipment modules required for transfer and solidification of the wastes. No equipment uses compressed gases. The solidification equipment requires a minimum of manual action and, in conjunction with the building layout, is designed to minimize occupational radiation exposures. Solidified containers are handled entirely by an overhead bridge crane. The crane is operated from the control console by the operator in combination with closed circuit television. A grapple device is available for use with the overhead bridge crane when handling high integrity containers.

The spent resin handling system consists of a spent resin holding tank and a spent resin transfer booster pump. The system provides a reusable source of water to sluice resins from various radioactive demineralizers to the hold tank where they are stored until transfer to the solidification portion of the system or to a high integrity container (HIC).

Primary grade water under pressure is available, if required, for flushing the piping in the solid radwaste system.

Remote handling equipment and a filter transfer shield are available for removing used filters before transferring for solidification or dewatering in preparation for shipment.

Contaminated tools and other compressible and incompressible solid wastes are packaged in various containers, such as 55 gallon drums or other acceptable packaging.

Piping that connects the solid radwaste systems of BVPS-1 and BVPS-2 is provided for greater flexibility. This allows BVPS-2 radwaste to be dewatered or solidified at the BVPS-1 solid radwaste facility, if required.

11.4.2.2.1 Evaporator Bottoms Packaging

Waste concentrates, at an expected concentration of 12 percent solids from the evaporators in the steam generator blowdown system (SGBS), are stored in the evaporator bottoms tank in the SGBS. These concentrates are adjusted for pH as required, using caustic from the caustic buffering system that is located in the waste handling building. The concentrates are then circulated to the solidification equipment in the condensate polishing building and back to the tank. The waste is drawn from this line by a positive displacement metering pump and injected into containers (55 gallon drums) in the drumming station. The containers will have been previously loaded with a measured weight of dry cement. The waste and dry cement are then mixed by tumbling the container within the drumming station processor. Solidification occurs in the container. The drum is removed from the processor and placed in storage onsite for future labeling, offsite shipment, and disposal. The drumming operation and container handling are both accomplished remotely.

11.4.2.2.2 Spent Resin Handling and Packaging

Resin in a demineralizer or ion exchanger is considered spent when the decontamination factor falls below a predetermined value, when the pressure drop becomes excessive, or when the surface dose rate approaches a predetermined limit.

The demineralizer or ion exchanger is then isolated and sluice water from the spent resin holding tank is used to flush the spent resin from the demineralizer or ion exchanger into the spent resin holding tank or HIC, using the spent resin transfer booster pump. The flush water passes out of the spent resin holding tank through retention screens, to prevent resin carry over, and is recirculated.

Resins sluiced from demineralizers and ion exchangers are stored in the spent resin holding tank. The resins are then slurried to the decant tank where they are allowed to settle. Excess water is removed by the decant pump. The pH of the spent resin can be adjusted using the caustic buffering system. The spent resin holding tank can hold more than 10 weeks accumulation of resins at the expected processing rates.

The resin or sludge is then injected into a container (55 gallon drum or equivalent), which had been previously loaded with a measured weight of dry cement. The waste and cement are mixed by tumbling the container in the drumming station. Solidification occurs in the container. The container is removed from the processor and is placed in storage onsite, for future labeling, offsite shipment, and disposal. If desired, the resin or sludge can be injected into an HIC and dewatered for offsite disposal in lieu of solidification.

The decant water is reused in the sluicing system. Sluice water may be pumped, if necessary, to the waste evaporator feed tanks in the liquid radwaste system via the waste handling building sump. Decant water from powdered resin sludges is returned to the sludge tank in the condensate polishing system. Water pumped from the HIC will be returned to the resin hold tank or the decant tank.

11.4.2.2.3 Solidification and Packaging Process Controls

The BVPS-2 radioactive waste solidification system provides a comprehensive process to solidify and package liquid radwastes. Solidification or dewatering of resins, sludges, or evaporator bottoms is ensured by the implementation of a process control program, as described in Sections 9.1.3 and 10.4 of ANSI/ANS 55.1 (1979). This program provides reasonable assurance of a consistent quality waste product which is acceptable for shipment and burial.

The Process Control Program (PCP) contains the methodology and boundary conditions to assure that all activities related to waste form are controlled and directed toward an end product meeting all regulatory requirements.

The potential for spills by dropping and damaging a high integrity container (HIC) being handled by the multi-size line grapple is considered very small. The grapple can be controlled from either a local pendant or the solid waste system control panel. The operator has positive indication at the control board or pendant that the grapple is fully engaged to the HIC. Once engaged, two separate buttons must be pressed to release the grapple from the HIC reducing the possibility of an accidental release.

The filling of containers, storage and shipment of radioactive solid wastes conform with 10 CFR 20, 10 CFR 61, 10 CFR 71, and the Department of Transportation Regulations, Title 49 CFR Parts 171-179.

11.4.2.2.4 Filter Handling

Filter elements are removed from service when the surface dose on the filter housing reaches a predetermined level, or when the pressure drop becomes excessive. The operation is carried out using remote handling equipment and a filter transfer shield when required. High activity filter cartridges are raised into the shield, and transported to the solid waste building using a filter transfer cart. The filters are placed in approved packages and prepared for shipment.

11.4.2.2.5 Waste Compaction Operation

Contaminated dry compressible materials are stored at specified locations in the plant, and subsequently may be transported to the waste compaction area for packaging.

11.4.2.2.6 Incompressible Waste Handling

Components with low activity, such as contaminated tools, are handled in the waste compaction area or in an appropriate designated area.

Spent core components, whose activity levels are very high, are handled under water within the reactor refueling cavity and fuel transfer canal. They are stored in the fuel pool until adequate packaging is provided for offsite shipment.

11.4.2.3 Expected Volumes

Tables 11.4-4 and 11.4-5 present a listing of the expected and design volumes of spent resins from various sources entering the solid radwaste system. Figures 11.4-3 and 11.4-4 present gross activities and weights or volumes for solid radwaste sources. Containers of evaporator bottoms, resins, and sludges are considered to be wet and are solidified with dry cement in containers or dewatered in HICs. Compacted or compressible waste is considered to be dry and is packaged in appropriately sized containers.

As indicated earlier, the figures and tables referenced above were developed in support of the original license and are considered historical.

11.4.2.4 Packaging

Based on the gross activities shown on Figures 11.4-3 and 11.4-4 and the USDOT Limits, container activity will vary between negligible, for most compressible or compacted wastes, to an expected activity of less than 300 $\mu Ci/cm^3$ or a design activity of less than 1,500 $\mu Ci/cm^3$, for the reactor water purification demineralizer resins. Estimates of the specific radionuclide content of the solid wastes are given in Tables 11.4-2 and 11.4-3.

As indicated earlier, the figures and tables referenced above were developed in support of the original license and are considered historical.

The filling of containers and the storage of solid radwaste conform with 10 CFR 20 and 10 CFR 50 requirements. Packages meet shipping and burial regulations of 49 CFR 171 through 178, 10 CFR 71, and 10 CFR 61, as applicable.

11.4.2.5 Storage Facilities

The solidification area, empty container storage, and waste storage areas are located on the ground floor of the condensate polishing building.

Wastes are normally packaged to allow for shipment after processing. Wastes in shipping containers with high radiation levels are stored in a shielded storage area until ready for shipment.

An average of 29 (expected quantity) or 53 (design quantity) containers (55 gallon drums) of solidified waste will be processed each week, based on the waste quantities shown on Figures 11.4-3 and 11.4-4, respectively. Using HICs (High Integrity Containers) to dispose of dewatered spent resins, an average of 6 (expected quantity) or 11 (design quantity) containers will be processed annually, with a corresponding decrease in the total number of containers of solidified waste. The storage facilities allow approximately 28 weeks storage capacity based on the expected average rate of solid waste accumulation or 15 weeks of storage capacity based on the design average rate of solid waste accumulation.

As indicated earlier, the figures referenced above were developed in support of the original license and are considered historical.

All processed waste containers can be stored in the condensate polishing building until shipped offsite.

11.4.2.6 Shipment and Disposal

All packages containing radioactive material and the procedures used to prepare these for offsite shipment conform with U.S. Department of Transportation and U.S. Nuclear Regulatory Commission Regulations. All waste material is either transferred to a licensed disposal contractor, to a licensed waste processor, or to a common carrier for delivery to a licensed disposal contractor, as appropriate.

Table 11.4-6 summarizes the annual number of shipments and shipped containers for the expected and design cases. As indicated earlier, the table referenced above was developed in support of the original license and is considered historical.

11.4.2.7 Protection Against Uncontrolled Releases

Protection against uncontrolled releases of radioactive material from the radioactive solid waste system is achieved through the use of indicators, monitors, alarms, interlocks, and retaining structures. The solid waste system is almost completely automated; therefore, the potential for operator error is minimal. In case of equipment malfunction, provisions exist for operator override.

The spent resin decant tank, the evaporator bottom tank, and the drumming station are located in curbed cubicles where any leakage will be retained. The spent resin hold tank is located in a totally enclosed cubicle, with the only access being the use of a ladder. In the event of a spillage of radioactive liquid from the pumps or tanks of the solid waste system, the liquid is collected by a network of floor drains. The walls and floors of areas with potential for contamination are suitably finished to facilitate decontamination. The floor is pitched towards the floor drains. The drains are piped to the building sumps and the liquid is pumped to the radioactive liquid waste system for processing.

The spent resin hold tank, the spent resin decant tank, and the evaporator bottoms tank are provided with low and high level indicators at the solid waste panel. Radiation detectors are provided at the decant station and the drumming station. A pump meters the specified quantity of resin-water slurry (at the required concentration) that is fed into the 55 gallon drums. The filling operation is automatic with remote control provided for operator override. If the shipping container is filled above the maximum safe levels, a level sensor automatically stops the process. A spray system is provided in the drum station to wash down the outer surface of the containers if necessary.

In the event of a loss of air supply from the compressors, all the valves and the metering pumps fail safe to the "closed" position.

The spent resin transfer booster pump will automatically shut off in the event of a low pressure signal on the suction side. The resin decanting pump will automatically shut off on a resin decant tank low level signal.

The most limiting single operator error or equipment failure would result in the spillage of the spent resin or evaporator bottoms tank contents. The drainage system in the condensate polishing building and the waste handling building are capable of handling such an event.

11.4.3 Process Control Program (PCP)

11.4.3.1 Procedures

Written procedures shall be established, implemented, and maintained covering the implementation of the PCP.

11.4.3.2 Changes to the PCP

Changes to the PCP shall be documented and records of reviews performed shall be retained in accordance with the applicable record retention provision of the quality assurance program description referenced in the Updated Final Safety Analysis Report. This documentation shall contain:

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

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

Changes to the PCP shall become effective after review and acceptance by the PORC and the approval of the plant manager, predesignated alternate or a predesignated manager to whom the plant manager has assigned in writing the responsibility for review and approval of specific subjects.

11.4.4 Reference for Section 11.4

U.S. Nuclear Regulatory Commission 1981. Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Reactor Plants, Branch Technical Position ETSB 11-3.

Tables for Section 11.4

TABLE 11.4-1

SOLID RADWASTE SYSTEM COMPONENT DATA

Component	Parameters
Spent Resin Holding Tank[(2)]	
Quantity	1
Capacity (gal)	1,500
Operating pressure	Atmospheric
Design temperature (°F)	250
Material[(1)] of construction	Stainless steel, SA-240, Type 304
Cement Day Tank	
Quantity	1
Capacity (ft^3)	55
Operating pressure	Atmospheric
Design temperature (°F)	120
Material[(1)] of construction	Carbon steel
Decant Tank[(2)]	
Quantity	1
Capacity (gal)	500
Operating pressure	Atmospheric
Design temperature (°F)	150
Material[(1)] of construction	Stainless steel, SA-240, Type 304
Caustic Buffering Tank	
Quantity	1
Capacity (gal)	1,500
Operating pressure	Atmospheric
Design temperature (°F)	104
Material[(1)] of construction	Stainless steel, SA-240, Type 304
Shipping Container (Drum)	
Quantity	As required
Capacity (gal)	55
Operating pressure	Atmospheric
Design temperature (°F)	212
Material[(1)] of construction	Low carbon commercial steel

TABLE 11.4-1 (Cont)

Component	Parameters
Spent Resin Transfer Booster Pump	
Quantity	1
Capacity (gpm)	65
Operating pressure (psig)	50
Design temperature (°F)	150
Material[1] of construction	Stainless steel, A-296 Gr. CF8M
Metering Pump	
Quantity	2
Capacity (gpm) each	15
Operating pressure (psig)	105
Design temperature (°F)	212
Material[1] of construction	Stainless steel, Type 316L
Decanting Pump	
Quantity	1
Capacity (gpm)	16
Operating pressure (psig)	40
Design temperature (°F)	212
Material[1] of construction	Stainless steel, Type 304L
Caustic Buffering Pump	
Quantity	1
Capacity (gph)	230
Operating pressure (psig)	94
Design temperature (°F)	80
Material[1] of construction	Stainless steel, Type 316
Mixer - Spent Resin Hold	
Quantity	1
Motor (hp)	2
Design temperature (°F)	212
Material[1] of construction	Stainless steel, Type 304L
Mixer - Decant Tank	
Quantity	1
Motor (hp)	1
Design temperature (°F)	212
Material[1] of construction	Stainless steel, Type 304/316

TABLE 11.4-1 (Cont)

Component	Parameters
Drumming Station Tumbler	
Quantity	1
Capacity	Two 55 gal drums/hr
Design temperature (°F)	212
Material[1] of construction	Stainless steel, Type 304L
Multi-Size Liner Grapple	
Quantity	1
Capacity	24,000 lbs.
Material[1] of construction	Carbon steel ASTM-A36

NOTES:

(1) Materials listed in this table may have been replaced with materials of equivalent design characteristics. The term equivalent is described in UFSAR Section 1.12, "Equivalent Materials".

(2) Designed in accordance with ASME Code, Section VIII, Division 1.

TABLE 11.4-2 [HISTORICAL]

SOLID WASTE ISOTOPIC CONTENT - EXPECTED CASE

Isotope	Spent Resins (Ci/yr)*	Waste Evaporator Bottoms (Ci/yr)	Condensate Polishing Demineralizer (Ci/yr)	Spent Filter Cartridges (Ci/yr)*	Totals (Ci/yr)
Cr-51		$2.8x10^{-1}$	$2.9x10^{-2}$	$7.1x10^{1}$	$7.1x10^{1}$
Mn-54		$4.6x10^{-2}$	$7.1x10^{-3}$	$3.3x10^{1}$	$3.3x10^{1}$
Fe-55		$2.4x10^{-1}$	$2.9x10^{-2}$	$1.8x10^{2}$	$1.8x10^{2}$
Fe-59		$1.5x10^{-1}$	$2.0x10^{-2}$	$5.6x10^{1}$	$.6x10^{1}$
Co-58		2.4	$2.8x10^{-1}$	$1.2x10^{3}$	$1.2x10^{3}$
Co-60		$2.9x10^{-1}$	$3.2x10^{-2}$	$2.3x10^{2}$	$2.3x10^{2}$
Sr-89	$2.0x10^{1}$	$5.2x10^{-2}$	$5.1x10^{-3}$	2.0	$2.2x10^{1}$
Sr-90	1.3	$1.5x10^{-3}$	$1.4x10^{-4}$	$1.3\ 10^{-1}$	1.4
Sr-91	$3.8x10^{-1}$	$6.8x10^{-2}$	$4.6x10^{-4}$	$3.8x10^{-2}$	$4.9x10^{-1}$
Y-90	1.2	$2.5x10^{-4}$	$9.1x10^{-5}$	$1.2x10^{-1}$	1.3
Y-91m	$2.5x10^{-1}$	$4.5x10^{-2}$	$3.0x10^{-4}$	$2.5x10^{-2}$	$3.2x10^{-1}$
Y-91	4.2	$9.6x10^{-3}$	$1.0x10^{-3}$	$4.2x10^{-1}$	4.6
Y-93	$2.1x10^{-2}$	$3.7x10^{-3}$	$2.5x10^{-5}$	$2.1x10^{-3}$	$2.7x10^{-2}$
Zr-95	4.0	$8.8x10^{-3}$	$1.4x10^{-3}$	$4.0x10^{-1}$	4.4
Nb-95	5.1	$7.4x10^{-3}$	$1.4x10^{-3}$	$5.1x10^{-1}$	5.6
Mo-99	$3.3x10^{2}$	$1.2x10^{1}$	$6.5x10^{-1}$	$3.3x10^{1}$	$3.8x10^{2}$
Tc-99m	$3.1x10^{2}$	9.0	$6.1x10^{-1}$	$3.1x10^{1}$	$3.5x10^{2}$
Ru-103	2.2	$6.6x10^{-3}$	$6.8x10^{-4}$	$2.2x10^{-1}$	2.4
Ru-106	1.1	$1.5x10^{-3}$	$1.4x10^{-4}$	$1.1x10^{-1}$	1.2
Rh-103m	2.1	$6.6x10^{-3}$	$6.6x10^{-4}$	$2.1x10^{-1}$	2.3
Rh-106	1.1	$1.5x10^{-3}$	$1.4x10^{-4}$	$1.1x10^{-1}$	1.2
Te-125m	1.8	$4.3x10^{-3}$	$3.5x10^{-4}$	$1.8x10^{-1}$	2.0
Te-127m	$2.3x10^{1}$	$4.2x10^{-2}$	$3.6x10^{-3}$	2.3	$2.5x10^{1}$
Te-127	$2.3x10^{1}$	$1.0x10^{-1}$	$3.9x10^{-3}$	2.3	$2.5x10^{1}$
Te-129m	$6.1x10^{1}$	$2.1x10^{-1}$	$1.9x10^{-2}$	6.1	$6.7x10^{1}$
Te-129	$3.9x10^{1}$	$1.5x10^{-1}$	$1.2x10^{-2}$	3.9	$4.3x10^{1}$
Te-131m	4.5	$3.3x10^{-1}$	$8.0x10^{-3}$	$4.5x10^{-1}$	5.0
Te-131	$8.5x10^{-1}$	$6.7x10^{-2}$	$1.5x10^{-3}$	$8.5x10^{-2}$	1.0
Te-132	$1.3x10^{2}$	3.9	$1.8x10^{-1}$	$1.3x10^{1}$	$1.5x10^{2}$
Ba-137m	$1.2x10^{3}$	2.5	$3.1x10^{-1}$	$1.2x10^{2}$	$1.3x10^{3}$
Ba-140	3.9	$3.2x10^{-2}$	$2.9x10^{-3}$	$3.9x10^{-1}$	4.3
La-140	4.2	$2.3x10^{-2}$	$2.8x10^{-3}$	$4.2x10^{-1}$	4.6
Ce-141	2.9	$1.0x10^{-2}$	$1.3x10^{-3}$	$2.9x10^{-1}$	3.2
Ce-143	$8.0x10^{-2}$	$5.4x10^{-3}$	$8.7x10^{-5}$	$8.0x10^{-3}$	$9.3x10^{-2}$
Ce-144	3.4	$4.9x10^{-3}$	$7.1x10^{-4}$	$3.4x10^{-1}$	3.7
Pr-143	1.0	$7.4x10^{-3}$	$6.2x10^{-4}$	$1.0x10^{-1}$	1.1
Pr-144	3.4	$4.9x10^{-3}$	$7.0x10^{-4}$	$3.4x10^{-1}$	3.7
Np-239	4.1	$1.7x10^{-1}$	$8.6x10^{-3}$	$4.1x10^{-1}$	4.7
Br-83	$7.3x10^{-1}$	$2.2x10^{-1}$	$9.8x10^{-4}$	$7.3x10^{-2}$	1.0
Br-84	$8.7x10^{-2}$	$2.8x10^{-2}$	$4.5x10^{-5}$	$8.7x10^{-3}$	$1.2x10^{-1}$
Br-85	$9.3x10^{-4}$	$2.9x10^{-4}$	$5.2x10^{-8}$	$9.3x10^{-5}$	$1.3x10^{-3}$

TABLE 11.4-2 (Cont)

Isotope	Spent Resins (Ci/yr)*	Waste Evaporator Bottoms (Ci/yr)	Condensate Polishing Demineralizer (Ci/yr)	Spent Filter Cartridges (Ci/yr)*	Totals (Ci/yr)
I-130	1.6	$2.4x10^{-1}$	$3.3x10^{-3}$	$1.6x10^{-1}$	$2.0x10^{o}$
I-131	$3.1x10^{3}$	$4.0x10^{1}$	3.4	$3.1x10^{2}$	$3.5x10^{3}$
I-132	$1.4x10^{2}$	7.1	$2.0x10^{-1}$	$1.4x10^{1}$	$1.6x10^{2}$
I-133	$4.8x10^{2}$	$4.9x10^{1}$	1.0	$4.8x10^{1}$	$5.8x10^{2}$
I-134	2.6	$8.2x10^{-1}$	$1.9x10^{-3}$	$2.6x10^{-1}$	3.7
I-135	$7.7x10^{1}$	$1.7x10^{1}$	$1.5x10^{-1}$	7.7	$1.0x10^{2}$
Rb-86	1.3	$1.3x10^{-2}$	$1.3x10^{-3}$	$1.3x10^{-1}$	1.4
Rb-88	2.1	1.2	$2.1x10^{-4}$	$2.1x10^{-1}$	3.5
Cs-134	$1.8x10^{3}$	3.8	$4.4x10^{-1}$	$1.8x10^{2}$	$2.0x10^{3}$
Cs-136	$1.5x10^{2}$	1.9	$1.9x10^{-1}$	$1.5x10^{1}$	$1.7x10^{2}$
Cs-137	$1.3x10^{3}$	2.7	$3.2x10^{-1}$	$1.3x10^{2}$	$1.4x10^{3}$
Totals	$9.2x10^{3}$	$1.6x10^{2}$	8.0	$2.6x10^{3}$	$1.2x10^{4}$

NOTE:

*Corrosion products conservatively assumed to be deposited on filter cartridges.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.4-3 [HISTORICAL]

SOLID WASTE ISOTOPIC CONTENT - DESIGN CASE

Isotope	Spent Resins (Ci/yr)*	Waste Evaporator Bottoms (Ci/yr)	Condensate Polishing Demineralizer (Ci/yr)	Spent Filter Cartridges (Ci/yr)*	Totals (Ci/yr)
Cr-51	-	$8.4x10^{-1}$	1.1	$3.5x10^{2}$	$3.5x10^{2}$
Mn-54	-	$1.4x10^{-1}$	$1.9x10^{-1}$	$1.0x10^{2}$	$1.0X10^{2}$
Fe-55	-	$7.2x10^{-1}$	1.0	$5.6x10^{2}$	$5.6x10^{3}$
Fe-59	-	$4.6x10^{-1}$	$6.1x10^{-1}$	$2.4x10^{2}$	$2.4x10^{2}$
Co-58	-	7.1	9.8	$4.3x10^{3}$	$4.3x10^{3}$
Co-60	-	$8.9x10^{-1}$	1.3	$6.9x10^{2}$	$6.9x10^{3}$
Sr-89	$3.2x10^{2}$	$5.7x10^{-1}$	$7.8x10^{-1}$	$3.2x10^{1}$	$3.5x10^{2}$
Sr-90	$1.0x10^{1}$	$1.3x10^{-2}$	$1.8x10^{-2}$	1.0	$1.1x10^{1}$
Sr-91	2.0	$1.8x10^{-1}$	$3.2x10^{-2}$	$2.0x10^{-1}$	2.4
Y-90	9.7	$1.5x10^{-2}$	$2.0x10^{-2}$	$9.7x10^{-1}$	$1.1x10^{1}$
Y-91m	1.3	$1.2x10^{-1}$	$2.1x10^{-2}$	$1.3x10^{-1}$	1.6
Y-91	5.5	$9.1x10^{-2}$	$1.3x10^{-1}$	$5.5x10^{-1}$	6.3
Y-93	$2.7x10^{-2}$	$3.2x10^{-2}$	$4.7x10^{-3}$	$2.7x10^{-3}$	$6.6x10^{-2}$
Zr-95	$5.6x10^{1}$	$9.4x10^{-2}$	$1.3x10^{-1}$	5.6	$6.2x10^{1}$
Nb-95	$6.9x10^{1}$	$9.6x10^{-2}$	$1.3x10^{-1}$	6.9	$7.6x10^{1}$
Mo-99	$1.7x10^{3}$	$4.4x10^{2}$	$3.3x10^{2}$	$1.7x10^{2}$	$2.7x10^{3}$
Tc-99m	$2.7x10^{3}$	$3.1x10^{2}$	$3.3x10^{2}$	$2.7x10^{2}$	$3.7x10^{3}$
Ru-103	$2.3x10^{1}$	$4.6x10^{-2}$	$6.0x10^{-2}$	2.3	$2.5x10^{1}$
Ru-106	2.3	$3.1x10^{-3}$	$4.3x10^{-3}$	$2.3x10^{-1}$	2.5
Rh-103m	$2.2x10^{1}$	$4.6x10^{-2}$	$6.0x10^{-2}$	2.2	2.4
Rh-106	2.3	$3.1x10^{-3}$	$4.3x10^{-3}$	$2.3x10^{-1}$	2.5
Te-125m	7.4	$1.3x10^{-2}$	$1.8x10^{-2}$	$7.4x10^{-1}$	8.1
Te-127m	$1.8x10^{2}$	$2.7x10^{-1}$	$3.6x10^{-1}$	$1.8x10^{1}$	$2.0x10^{2}$
Te-127	$1.7x10^{2}$	$1.8x10^{-1}$	$3.4x10^{-1}$	$1.7x10^{1}$	$1.9x10^{2}$
Te-129m	$2.3x10^{3}$	5.2	7.1	$2.3x10^{2}$	$2.5x10^{3}$
Te-129	$1.5x10^{3}$	3.2	4.5	$1.5x10^{2}$	$1.7x10^{3}$
Te-131m	$7.3x10^{1}$	2.7	1.4	7.3	$8.5x10^{1}$
Te-131	$1.4x10^{1}$	$5.5x10^{-1}$	$2.6x10^{-1}$	1.4	$1.5x10^{1}$
Te-132	$2.2x10^{3}$	$3.5x10^{1}$	$3.1x10^{1}$	$2.2x10^{2}$	$2.5x10^{3}$
Ba-137m	$1.4x10^{4}$	$1.6x10^{2}$	$2.3x10^{2}$	$1.4x10^{3}$	$1.6x10^{4}$
Ba-140	$1.4x10^{2}$	$5.8x10^{-1}$	$7.2x10^{-1}$	$1.4x10^{1}$	$1.5x10^{2}$
La-140	$1.5x10^{2}$	$2.3x10^{-1}$	$5.6x10^{-1}$	$1.5x10^{1}$	$1.7x10^{2}$
Ce-141	$4.1x10^{1}$	$9.2x10^{-2}$	$1.3x10^{-1}$	4.1	$4.5x10^{1}$
Ce-143	1.8	$6.2x10^{-2}$	$3.4x10^{-2}$	$1.8x10^{-1}$	2.0
Ce-144	$3.7x10^{1}$	$4.9x10^{-2}$	$6.8x10^{-2}$	3.7	$4.1x10^{1}$
Pr-143	$2.4x10^{1}$	$9.0x10^{-2}$	$1.2x10^{-1}$	2.4	$2.6x10^{1}$
Pr-144	$3.6x10^{1}$	$4.9x10^{-2}$	$6.8x10^{-2}$	3.6	$4.0x10^{1}$
Np-239	$2.4x10^{1}$	$5.1x10^{-1}$	$3.9x10^{-1}$	2.4	$2.7x10^{1}$
Br-83	$2.2x10^{1}$	3.4	$2.9x10^{-1}$	2.2	$2.8x10^{1}$
Br-84	2.3	$3.6x10^{-1}$	$1.1x10^{-2}$	$2.3x10^{-1}$	2.9
Br-85	$2.8x10^{-2}$	$4.5x10^{-3}$	$1.6x10^{-5}$	$2.8x10^{-3}$	$3.6x10^{-2}$

TABLE 11.4-3 (Cont)

Isotope	Spent Resins (Ci/yr)*	Waste Evaporator Bottoms (Ci/yr)	Condensate Polishing Demineralizer (Ci/yr)	Spent Filter Cartridges (Ci/yr)*	Totals (Ci/yr)
I-130	9.5	7.1×10^{-1}	2.0×10^{-1}	9.5×10^{-1}	1.1×10^{1}
I-131	5.4×10^{4}	3.4×10^{2}	4.1×10^{2}	5.4×10^{3}	6.0×10^{4}
I-132	2.4×10^{3}	5.9×10^{1}	3.6×10^{1}	2.4×10^{2}	2.7×10^{3}
I-133	9.2×10^{3}	4.7×10^{2}	2.0×10^{2}	9.2×10^{2}	1.1×10^{4}
I-134	5.4×10^{1}	8.4	3.9×10^{-2}	5.4	6.8×10^{1}
I-135	1.6×10^{3}	1.7×10^{2}	2.9×10^{1}	1.6×10^{2}	2.0×10^{3}
Rb-86	1.2×10^{1}	3.8×10^{-2}	4.9×10^{-2}	1.2	1.3×10^{1}
Rb-88	1.0×10^{2}	1.6×10^{1}	1.4×10^{-1}	1.0×10^{1}	1.3×10^{2}
Cs-134	3.1×10^{3}	3.6×10^{1}	5.1×10^{1}	3.1×10^{2}	3.5×10^{3}
Cs-136	3.7×10^{2}	2.1×10^{1}	2.5×10^{1}	4.0×10^{1}	4.6×10^{2}
Cs-137	1.5×10^{4}	1.7×10^{2}	2.4×10^{2}	1.5×10^{3}	1.7×10^{4}
Totals	1.0×10^{5}	2.3×10^{3}	2.0×10^{3}	1.7×10^{4}	1.2×10^{5}

NOTE:

*Corrosion products conservatively assumed to be deposited on filter cartridges.

The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.4-4 [HISTORICAL]

SPENT RESIN GENERATED ANNUALLY
AS EXPECTED

Demineralizers	Number of Beds Total	Number of Beds Operating*	Volume (ft^3/bed	Frequency of Replacement Total beds/yr	Spent Resin Generated (ft^3/yr)
Mixed bed demineralizer**	2	2	30	6	180
Cation bed demineralizer**	1	1	20	2	40
Cesium removal ion exchanger (mixed bed)**	2	2	35	4	140
Deborating demineralizer**,***	2	2	35	2	70
Steam generator blowdown cleanup demineralizer****	2	2	35	4	140
Fuel pool demineralizer	1	1	17	1	17
Total spent resin	-	-	-	-	587
Condensate treatment powdered resin sludge	5	4	12	200	2,400

NOTES:

*On-line during normal operation
**Reactor plant service bed resin demineralizers
***Used only at end of core burnup to reduce boron concentration in the reactor coolant system
****Part of liquid waste treatment system
The information presented in the above table was developed in support of the original license and is considered historical.

TABLE 11.4-5 [HISTORICAL]

SPENT RESIN GENERATED ANNUALLY
AS DESIGNED

Demineralizers	Number of Beds Total	Number of Beds Operating*	Volume (ft^3/bed	Frequency of Replacement Total beds/yr	Spent Resin Generated (ft^3/yr)
Mixed bed demineralizer**	2	2	30	12	360
Cation bed demineralizer**	1	1	20	4	80
Cesium removal ion exchanger (mixed bed)**	2	2	35	8	280
Deborating demineralizer**,***	2	2	35	4	140
Steam generator blowdown cleanup demineralizer****	2	2	35	8	280
Fuel pool demineralizer	1	1	17	2	34
Total spent resin	-	-	-	-	1,174
Condensate treatment powdered resin sludge	5	4	12	333	4,000

NOTES:

*On-line during normal operation
**Reactor plant service bed resin demineralizers
***Used only at end of core burnup to reduce boron concentration in the reactor coolant system
****Part of liquid waste treatment system
The information presented in the above table was developed in support of the original license and is considered historical.

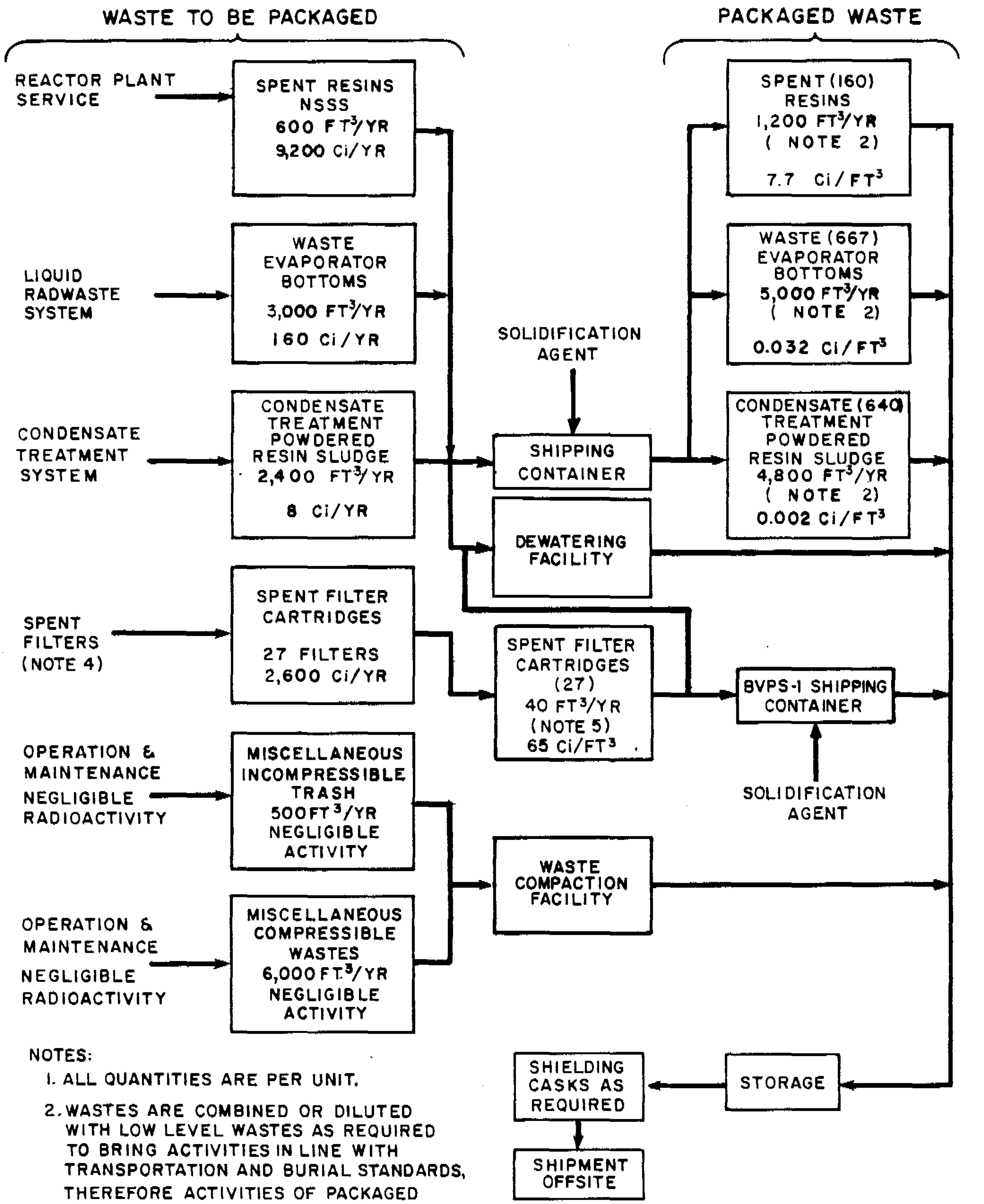
TABLE 11.4-6 [HISTORICAL]

SOLID RADWASTE ANNUAL SHIPMENTS

Type of Waste and Packaging	Expected	Design
Solidified wet wastes and filter cartridges		
Total containers	1,494 (1,334)*	2,773 (2,453)*
Total shipments	71	132
Dewatered spent resins		
Total containers (HICs)	6	11

*Number in ( ) is total if spent resins are shipped in HICs.

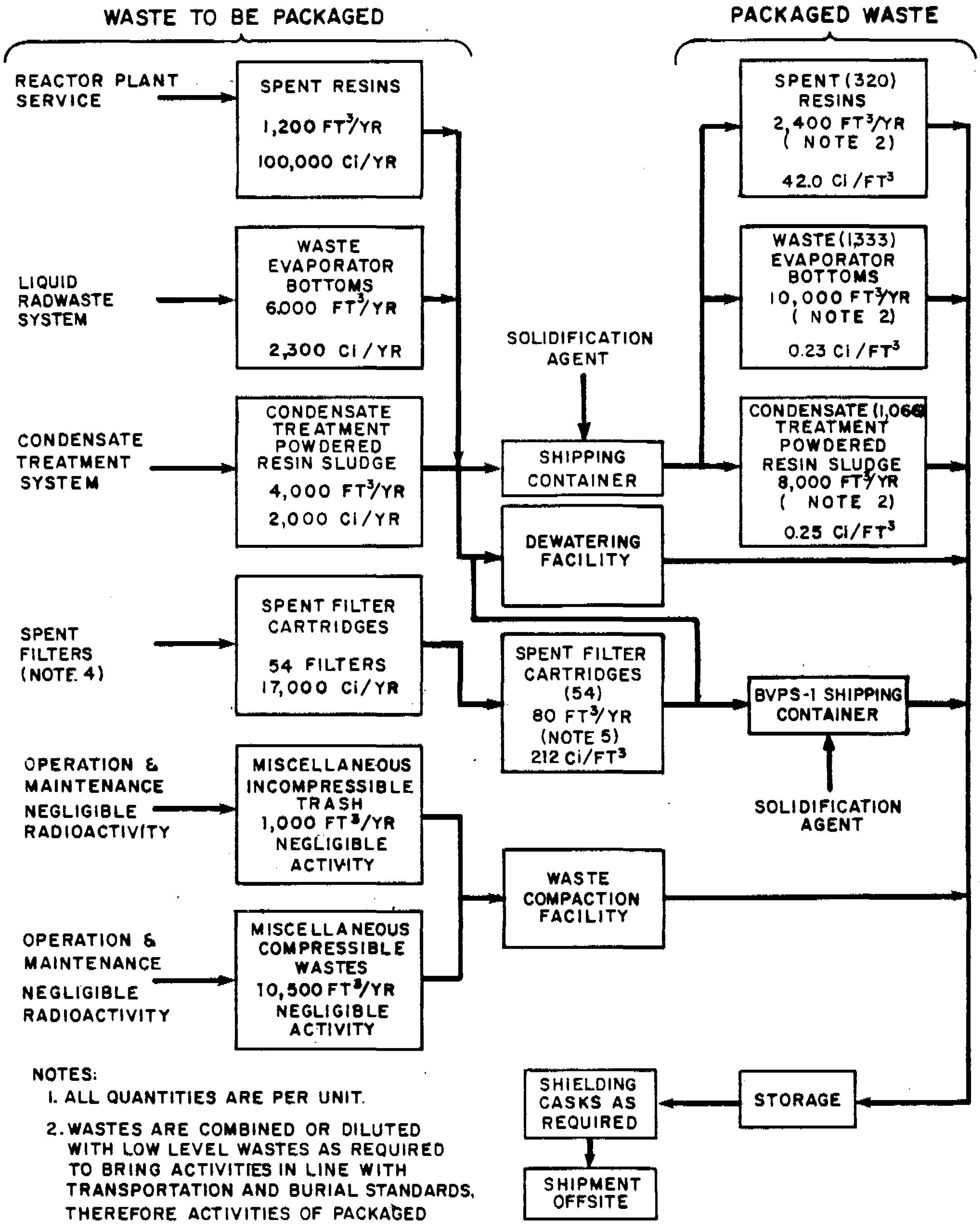
The information presented in the above table was developed in support of the original license and is considered historical.



NOTES:

1. ALL QUANTITIES ARE PER UNIT.
2. WASTES ARE COMBINED OR DILUTED WITH LOW LEVEL WASTES AS REQUIRED TO BRING ACTIVITIES IN LINE WITH TRANSPORTATION AND BURIAL STANDARDS, THEREFORE ACTIVITIES OF PACKAGED WASTES DO NOT DIRECTLY CORRELATE WITH ACTIVITIES OF THEIR UNPACKAGED SOURCES.
3. NUMBER OF CONTAINERS/YR INDICATED IN ( ) PARENTHESES.
4. SPENT FILTERS TRANSFERRED TO BVPS-1 FOR DISPOSAL.
5. VOLUME SHOWN IS FOR UNPACKAGED WASTE.
6. USING HICS: SPENT RESIN(6), 600 FT^3/ YR, 15 CI / FT^3

FIGURE 11.4-3
SOLID RADWASTE SYSTEM
EXPECTED QUANTITIES PER UNIT
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT



NOTES:

1. ALL QUANTITIES ARE PER UNIT.
2. WASTES ARE COMBINED OR DILUTED WITH LOW LEVEL WASTES AS REQUIRED TO BRING ACTIVITIES IN LINE WITH TRANSPORTATION AND BURIAL STANDARDS. THEREFORE ACTIVITIES OF PACKAGED WASTES DO NOT DIRECTLY CORRELATE WITH ACTIVITIES OF THEIR UNPACKAGED SOURCES.
3. NUMBER OF CONTAINERS/YR INDICATED IN ( ) PARENTHESES.
4. SPENT FILTERS TRANSFERRED TO BVPS-1 FOR DISPOSAL.
5. VOLUME SHOWN IS FOR UNPACKAGED WASTE
6. USING HICS: SPENT RESIN(11), 1200 FT^3/YR, 88 CI/FT^3

FIGURE 11.4-4
SOLID RADWASTE SYSTEM
DESIGN QUANTITIES PER UNIT
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING INSTRUMENTATION AND SAMPLING SYSTEMS

11.5.1 Design Bases

The process and effluent radiation monitoring system is a digital computer-based data processing system designed to detect, indicate, annunciate, store, record, and control selected radioactive process streams and effluents throughout Beaver Valley Power Station - Unit 2 (BVPS-2). Potential paths for release of radioactive materials are monitored (Figures 11.5-1 and 11.5-2) during normal operation, anticipated operational occurrences, and during post-accident conditions. These releases are continuously monitored to ensure compliance with U.S. Nuclear Regulatory Commission (USNRC) General Design Criteria (GDC) 60, 63, and 64, the requirements of 10 CFR 20, 10 CFR 50, and Regulatory Guide 1.21, and meets the intent of Regulatory Guide 1.97. Sections 11.5.2, 11.5.3, and 11.5.4 describe the design features provided to ensure agreement with Regulatory Guide 1.21. Quality Assurance Category I/Seismic Category I equipment will be Class 1E, environmentally qualified to Regulatory Guide 1.89 in accordance with the Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974, and seismically qualified in accordance with Regulatory Guide 1.100 to IEEE Standard 344-1975. Quality Assurance Category II/Seismic Category II equipment will be non-Class 1E. Regulatory Guide 1.97 instrument types and variables are listed in Section 7.5. Section 3.11 discusses environmental equipment qualification. Potential pathways for release of radioactive materials during accident conditions are continuously monitored.

The design objectives for the process and effluent radiation monitors for normal operation are as follows:

1. To respond to the anticipated form of radiation (that is, electro-magnetic or corpuscular),

2. To provide continuous indication and recording of dose rates and radioactive concentrations in radioactive and potentially radioactive systems,

3. To provide local and remote alarms to give audible and visual warning to operating plant personnel when radioactivity levels have reached predetermined set points,

4. For certain fluid and ventilation streams, to provide contacts for automatic actuation of components external to the process and effluent radiation monitors, and

5. To record, process, and provide data to be used in reporting releases of radioactive materials in liquid and gaseous effluents in accordance with the guidelines of Regulatory Guide 1.21.

The design objectives for the post-accident monitors (PAMs) which assess plant conditions during and following an accident are based on the guidelines of Regulatory Guide 1.97 and meet the intent of NUREG-0737 (USNRC 1980), Action Item II.F.1, Attachments 1 and 2.

11.5.2 System Description

11.5.2.1 Digital Radiation Monitoring System

The digital radiation monitoring system (DRMS) consists of three modules, including:

1. The radiation monitors,
2. The local processors, and
3. The central processors.

The radiation monitors are divided into the following functional classifications:

1. Process monitors which determine concentrations of radioactive material in plant fluid systems,
2. Effluent monitors which measure radioactivity discharged directly to the environs at final release points,
3. Airborne monitors which provide operator information relative to airborne concentrations of radioactive gases and particulate radioactivity at various locations in the plant (a detailed discussion of these monitors is provided in Section 12.3.4),
4. Area monitors which provide operator information relative to external gamma radiation levels at fixed points throughout the plant (a detailed discussion of these monitors is described in Section 12.3.4), and
5. Post-accident (or high-range) monitors that are designed to assess radiological conditions which require an extended range to envelop anticipated abnormal radiation levels.

The process and effluent portions of the DRMS include the following radiation monitors, grouped according to their classification (quantity in parentheses):

1. Effluent Radiation Monitors

    a. QA Category I/Seismic Category I

1) Main steam discharge high range (1) (post-accident)

2) Elevated release (2)

b. QA Category II/Seismic Category II

1) Ventilation vent (1)

2) Liquid waste process effluent (1)

3) Condensate polishing building vent stack (1)

4) Waste gas storage vault (1)

5) Decontamination building (1)

2. Process Radiation Monitors

a. QA Category I/Seismic Category I

1) Recirculation spray heat exchanger service water (4)

2) Containment purge (2)

b. QA Category II/Seismic Category II

1) Air ejector delay bed exhaust (1)

2) Aerated vent transfer line (1)

3) Component cooling service water (1)

4) Component cooling heat exchanger service water (1)

5) Component cooling water (1)

6) Steam generator blowdown sample (1)

7) Reactor coolant letdown high range (1)

8) Reactor coolant letdown low range (1)

9) Gaseous waste surge tank transfer line (1)

10) Auxiliary steam condensate (1)

11) Evaporator reboiler condensate (1)

12) Waste gas storage tanks (1)

13) Air ejector discharge (1)

11.5.2.2 Process and Effluent Radiation Monitors

This section describes the normal operation and post-accident process and effluent monitors. Gaseous effluent releases are discussed in Section 11.3.2.2 and gaseous effluent monitoring in the Environmental Report, Section 3.5.5. The airborne and area monitors are described in Section 12.3.4. The local processors and central processors are described in Sections 11.5.2.7 and 11.5.2.8, respectively.

In addition to classifying radiation monitors according to their function (Section 11.5.1), monitors are also classified according to monitor type. All process and effluent monitors fall into one of the following four monitor types:

1. Off-line gas and particulate,

2. Off-line gas,

3. Off-line liquid,

4. In-line gas or liquid (which includes pipe and duct-mounted detectors).

Table 11.5-1 lists the process and effluent monitors in the DRMS and their respective monitor type.

11.5.2.2.1 Off-Line Gas and Particulate Monitors

The off-line gas and particulate monitors provide a continuous off-line sampling of gaseous and airborne particulate effluents. A continuous sample of effluent or airborne activity is withdrawn from a ventilation duct via an isokinetic nozzle and then drawn through moving filter paper for airborne particulate collection. The activity of the material deposited on the filter paper is continuously scanned by a detector. For effluent monitors, a separate isokinetic path through a fixed filter system provides a composite sample for measuring integrated flow volumes.

After passing through the moving filter paper the sample passes through an easily removable charcoal filter cartridge, and into a shielded sample chamber, where the gases are monitored by a detector, and then discharged back to the ventilation line.

The sample line and all surfaces of the sampler exposed to the sample are stainless steel. The monitor electronics are enclosed in a splash-resistant enclosure.

The sampling system includes a pump with motor, a purge unit for purging the gas sample chamber, a pressure gage, flowmeter, manual or automatic flow control adjustment unit, and isolation valve(s).

Detector types, ranges, set points, and sampling location for specific off-line gas and particulate monitors are further defined in Tables 11.5-1, 11.5-2 and 11.5-3.

11.5.2.2.2 Off-Line Gas Monitors

The off-line gas monitors are similar to the off-line gas and particulate monitors except that the moving filter particulate detector and isokinetic collection are not required. All other requirements for the gaseous detector associated with purge filters, and the sampling system are described in Section 11.5.2.2.1.

The elevated release off-line gas monitor consists of a wide range off-line gas monitor and collection assemblies for normal and high range particulate and iodine. Moving paper filter capability is provided on an off-line particulate assembly.

The elevated release off-line gas monitor is designed to the guidelines of Regulatory Guide 1.97 and enables the plant operators to diagnose and follow the course of abnormal occurrences.

11.5.2.2.3 Off-Line Steam Monitor

The main steam discharge off-line steam monitor consists of three adjacent-to-line detectors viewing the sampling lines for each of the main steam loops.

The main steam discharge off-line steam monitor is designed to the guidelines of Regulatory Guide 1.97 and enables the plant operators to diagnose and follow the course of abnormal occurrences.

11.5.2.2.4 Off-Line Liquid Monitors

Off-line liquid monitors provide a continuous off-line sampling and monitoring of liquid streams. A continuous liquid sample is withdrawn from the liquid process or effluent stream, passed through an off-line shielded, liquid sampler assembly where the sample is monitored by a radiation detector, and then returned to the liquid process or effluent stream or discharged to local floor drains.

Most off-line liquid monitors are supplied with pumps, a pressure gage, temperature indicator, flow meter, flow switch, manual flow control adjustment, isolation valves, and flanges at the inlet and outlet of the sampler assembly to allow for its removal.

Sample chambers have sufficient volume to achieve the required sensitivity and include readily replaceable liners capable of being decontaminated for reuse.

11.5.2.2.5 In-Line Gas or Liquid Monitors

In-line gas or liquid monitors are designed to monitor process or effluent streams continuously with either a radiation detector located in the well or a sampler, which is designed to fit directly in a pipe through the use of standard pipe flanges, or with a detector mounted directly to a duct.

Duct or pipe-mounted monitors are designed to monitor process and effluent streams continuously via a radiation detector mounted on or in the ventilation duct or pipe. A moisture-proof housing, designed to protect the detector from the environment in which it is located, and adequate radiation shielding is also provided to achieve required detector sensitivity.

11.5.2.3 Description of Monitor Components

Each monitor consists of a detector (or detectors) and associated shielding, filters, sampling pumps, check sources, isokinetic nozzles, etc, as described in Section 11.5.2.2. The following sections describe these major components of the process and effluent monitors. The number and type of components required for each monitor are presented in Table 11.5-1.

All radiation monitors are designed to withstand the sample and environmental temperatures and pressures listed in Table 11.5-3. Environment qualification of the equipment is in accordance with IEEE Standard 323-1974. Monitors provided for post-accident measurements are able to withstand the expected radiation dose environment.

11.5.2.3.1 Detectors

Detector type, range, sensitivity, and radiation background are listed in Table 11.5-2.

Detector ranges are selected to ensure maximum range and sensitivity, as stated in Table 11.5-2. The range of the high-range monitors is sufficient to cover all postulated releases.

Beta or gamma detectors are employed by all process and effluent monitors. The elevated release monitor includes three gas detectors - low range, mid range, and high range.

11.5.2.3.2 Detector Shielding Requirements

An adequate amount of shielding is provided around each detector to reduce the background radiation to a level which will not interfere with the detector sensitivity stated in Table 11.5-2. All off-line detectors have four pi geometry shielding.

11.5.2.3.3 Detector Check Source(s)

A remotely-operated radiation check source is furnished as an integral part of each detector unit. This is a long half-life source (at least 2 years) having emissions similar to those being detected, and an activity sufficient to indicate in the approximate mid range of the next decade above the specified background reading (Table 11.5-2). This source is operable from the local processor, or remotely from the main control room and radiation protection laboratory via the DRMS central processor's displays and keyboards. Class 1E monitor check sources are operable from the control and display modules, as indicated in Section 11.5.2.6.5.

Adequate shielding is provided between the detector and check source so that, in the de-energized position, the radiation from the check source will not interfere with the detector sensitivity. Each channel is automatically reset after check source withdrawal.

All check sources are itemized and shipped separately from the monitors. Those which require a USNRC license are identified.

11.5.2.3.4 Detector Signal Accuracy and Accuracy Guarantees

The accuracy of all radiation monitors is within ±33 percent of the true value, assuming normal operating conditions. During accident conditions, overall system accuracy is within a factor of 2 in accordance with the guidelines of Regulatory Guide 1.97. Overall electronic system response time from 10 to 90 percent of maximum reading is less than, or equal to, 3 seconds. Specific accuracies are determined for each detector.

11.5.2.3.5 Isokinetic Nozzles

Isokinetic nozzles are provided for all off-line gas and particulate monitors to assure uniform quality sample control. The nozzle design is determined from duct sizes and flow rates listed in Table 11.5-1. The American National Standards Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities (ANSI N13.1-1969) is used for guidance.

Two sample flow control options are available for the ventilation vent effluent monitor, the elevated release effluent monitor, the condensate polishing building effluent monitor, the decontamination building effluent monitor and the waste gas storage vault ventilation effluent monitor. Setpoint sample flow control may be used to control sample flow to a fixed flowrate based on the sample pump design and sample delivery line diameter, or Isokinetic sample flow control may be used to control sample flow to a dynamic, calculated flowrate based on the real-time duct or stack velocity. Both methods utilize flow rate control valves, flow meters, duct or stack velocity probes, etc. and display sample flow and duct or stack flow on the system.

11.5.2.3.6 Moving/Fixed Particulate Filters

Moving filters are included in all off-line gas and particulate monitors. Fixed filter papers are used in each effluent gas and particulate monitor in conjunction with a moving filter paper system. Shielded fixed filters are included on high-range off-line gas monitors for personnel protection. Filters are mounted in positions which assure sufficient retention of particulates.

For moving filters, an electromechanical assembly, which controls the movement of the filter paper or tape, is provided as an integral part of the detector unit. This assembly has a variable speed filter drive capable of continuous and step advance movement, and operation in a fixed position. The filter paper or tape can be easily removed with little likelihood of instrument contamination during removal and has a 30-day minimum capacity at normal speeds. The entire assembly is enclosed in a splash-resistant housing. In the event of a malfunction in the electromechanical assembly (for example, torn paper, loss of power), the channel failure alarm is initiated locally and remotely.

A filter alarm is initiated when the filter paper supply reaches 3 days. The channel failure alarm is initiated when all filter paper is expended.

Moving and fixed filter papers have relatively smooth finishes on the collection face to enhance alpha counting. In addition, the material composition of all filters is selected to withstand high humidity conditions. All fixed filter paper is readily burnable and is capable of being easily ashed.

Flow is monitored for the fixed filter flow paths of effluent monitors. This flow signal is input to the local processor and integrated over time, as described in Section 11.5.2.7.

11.5.2.3.7 Charcoal Filters

Charcoal filters/cartridges are provided for iodine collection for all off-line gas and particulate and off-line gas monitors, as indicated in Table 11.5-1.

11.5.2.3.8 Tritium Sampling

Tritium sampling capability is provided for each monitor which contains a gas sample.

11.5.2.3.9 Sample Pumps

All off-line gas and off-line gas and particulate monitors include vacuum pumps designed to provide long life and reliability with leakproof and contamination-free performance. Off-line liquid radiation monitors which require a pump are identified in Table 11.5-1. All liquid pumps are self-priming. The liquid pump capacity is sufficient to provide a flow rate of 2 sample volumes per minute. For those pumps which operate intermittently (Section 11.5.2.5), provisions are made for permissive start contacts.

11.5.2.4 Effluent Monitors - Functional Description

Effluent monitors (those monitors which measure radioactivity discharged directly to the environment at final release points) are off-line gas and particulate monitors, off-line steam, or off-line liquid monitors. The exceptions are effluent PAMs, also referred to as high-range monitors, which are off-line gas monitors or in-line gas/liquid monitors. Monitor technical requirements are listed in Table 11.5-1. Detector technical requirements are listed in Table 11.5-2. Conditions of service for these monitors are listed in Table 11.5-3.

Alarm set points for the monitors are dependent on detector response for the expected nuclide mix. Alarm set point determinations are based on process and effluent release limits and personnel protection.

11.5.2.4.1 Ventilation Vent Monitor

The ventilation vent off-line gas and particulate monitor continuously samples effluent releases via the ventilation vent. Normally, the air flowing to the ventilation vent is composed of effluent from the area contiguous to the containment and purge exhaust which are each individually monitored. A high activity signal from the containment purge source will cause automatic isolation of the purge system and indicate to the operator the need to divert the effluent through one of the main filter banks in the supplementary leak collection and release system (SLCRS) (Section 6.5.2) before discharge to the environment via the elevated release point.

11.5.2.4.2 Elevated Release Monitor

The elevated release off-line gas and particulate monitor withdraws a continuous sample from the elevated release exhaust line upstream of the point of discharge of the effluent to the environment.

This monitor is provided with sampling capability and necessary range to meet the intent of NUREG-0737, Action Item II.F.1, Attachments 1 and 2.

Normally, the sample is composed of effluent from the leak collection areas, auxiliary building, and fuel building, which are each individually monitored. The effluent passes through one of the main filter banks in the SLCRS before discharge via the elevated release point.

The high range portion of this monitor assists in monitoring the radiation concentration in various systems during and following an accident, in which the elevated release may be used as a potential discharge path to the environment.

The isokinetic nozzle is designed and fabricated to Seismic Category I requirements.

11.5.2.4.3 Liquid Waste Process Effluent Monitor

The liquid waste process effluent off-line liquid detector monitors the discharge from the liquid waste process system. The pump for this monitor starts on a flow signal from a flow transmitter. This discharge may be released via the Beaver Valley Power Station - Unit 1 (BVPS-1) or BVPS-2 cooling tower blowdown or sent to the BVPS-1 boron recovery test tank. Upon detection of high radiation, this monitor actuates isolation to prevent the release of this discharge.

11.5.2.4.4 Condensate Polishing Building Vent Stack Monitor

The condensate polishing building vent stack off-line gas and particulate monitor withdraws a continuous sample from a point upstream of the condensate polishing building vent stack effluent release.

11.5.2.4.5 Main Steam Discharge High-Range Monitor

The main steam discharge high-range off-line steam detectors assist in monitoring plant effluents during and following an accident in which the atmospheric dump valves and the main steam safety valves may be used as potential discharge paths to the environment of radioactive material derived from primary to secondary leakage.

Additionally, these detectors will help detect steam generator tube rupture depending upon activity level and assist the operator to follow emergency response guidelines.

11.5.2.4.6 Waste Gas Storage Vault Effluent Monitor

The waste gas storage vault effluent off-line gas monitor samples effluent releases via the ventilation duct of the gaseous waste storage tank area. Upon high alarm, flow is diverted through filters before being released to the atmosphere.

11.5.2.4.7 Decontamination Building Effluent Monitor

The decontamination building off-line gas and particulate monitor withdraws a continuous sample from a point upstream of the decontamination building effluent release.

11.5.2.5 Process Monitors - Functional Description

Process monitors (those monitors which determine concentrations of radioactive material in BVPS-2 fluid systems) (gas or liquid) may be off-line gas, off-line liquid, or in-line gas monitors depending upon the monitor's function and the sample fluid monitored. Monitor technical requirements are listed in Table 11.5-1. Detector technical requirements are listed in Table 11.5-2. Conditions of service for these monitors are listed in Table 11.5-3.

Alarm set points for the monitors are dependent on detector responses for the expected nuclide mix. Alarm set point determinations are based on process and effluent release limits and personnel protection.

11.5.2.5.1 Recirculation Spray Heat Exchanger Service Water Monitor

Four recirculation spray heat exchanger service water off-line monitors individually monitor the service water discharge of the four recirculation spray heat exchangers. Each heat exchanger has 50 percent capacity and they are operated in pairs. These monitors provide redundancy for detecting heat exchanger leakage. They are designed for intermittent operation (Section 11.5.2.3.9).

Form C contacts are supplied for each of these four monitors and are used to start the pump and actuation signals are taken from the service water motor-operated valves to place the monitor into operation.

11.5.2.5.2 Containment Purge Monitor

The containment purge exhaust is monitored by two redundant in-line, duct-mounted detectors. During the initial containment purge (prior to allowing personnel entry into containment) containment air is monitored. If the containment air is within allowable limits, it is directed to the ventilation vent exhaust. If airborne activity is present, flow is directed through a filter bank to the elevated release. When the containment airborne activity has been reduced to levels at which personnel respiratory protection is not required, the containment is opened to atmosphere permitting personnel entry. During refueling, the containment purge exhaust normally maintains the containment at a slightly negative pressure. A high activity alarm from the containment purge exhaust monitor during refueling operations may be configured to automatically close the purge supply and exhaust isolation dampers in the containment building. The airborne activity in the containment atmosphere may then be subsequently discharged at a controlled rate through one of the main filter banks in the SLCRS to the elevated release point above the containment structure.

11.5.2.5.3 Air Ejector Discharge Monitor

The air ejector discharge in-line gas monitor continuously monitors the gaseous effluent from the condenser air ejector discharge upstream of the air ejector discharge charcoal delay beds. The detector is located in a well in an in-line sampler. An alarm from this detector indicates a primary-to-secondary system leak. A high activity alarm signals the operator to divert manually to the charcoal delay beds.

11.5.2.5.4 Air Ejector Delay Beds Exhaust Monitor

The air ejector charcoal delay beds detector continuously monitor the gaseous effluent from the discharge of the air ejector charcoal delay beds. The detector is located in a well in an off-line sampler. An alarm from this detector indicates a primary-to-secondary system leak and also a possible malfunction of the charcoal delay beds.

11.5.2.5.5 Aerated Vent Transfer Line Monitor

The aerated vent transfer line off-line gas monitor detects potential sources of radioactivity in environmental releases. The various flow paths monitored are the containment vacuum pump discharge and the sweep gas and purge exhaust from the waste gas charcoal delay beds.

11.5.2.5.6 Component Cooling Water Monitor

The component cooling water (CCW) monitor continuously analyzes a composite sample drawn from the two upstream legs of the CCW heat exchangers. The sample is drawn through an off-line sampler system containing a detector inserted in a well located in the flow stream of the off-line sampler. Detection of radioactivity indicates leakage of radioactive effluent from the reactor coolant system, or an auxiliary system, into the CCW system.

11.5.2.5.7 Component Cooling Heat Exchanger Service Water Monitor

The component cooling heat exchanger service water off-line liquid monitor analyzes a continuous composite sample of the service water discharged from the component cooling heat exchangers in service. The sample drawn is a mixture of the service water from those heat exchangers in service. The detector is located in a well in an off-line sampler. A high alarm indicates a heat exchanger leak. A valving arrangement permits individual sampling of each heat exchanger to determine the location of the leak.

11.5.2.5.8 Component Cooling Service Water Monitor

The component cooling service water off-line liquid monitor analyzes a continuous sample of the service water discharged from the CCW heat exchangers during normal operation. The detector is located in a well in an off-line sampler. An activity measurement is indicative of a heat exchanger leak. This detector acts as a redundant channel. Activity in the service water is initially detected by the component cooling heat exchanger service water monitor (Section 11.5.2.5.7), which is upstream of the component cooling service water monitor.

11.5.2.5.9 Steam Generator Blowdown Sample Monitor

The steam generator blowdown (SGB) sample monitor analyzes a composite sample of blowdown effluent. The sample comes from a common header which acts as a manifold for effluent taken from the three steam generators. The detector is located in a well in an off-line liquid sampler. An alert activity alarm alerts the operator to sample each of the SGB effluents. In addition, a high alarm isolates the SGB sample lines to the primary sample system. The individual sampling is accomplished by a valving arrangement which permits the operator to determine the source of the high activity. The SGB sample monitoring system can be flushed with clean water from a flush line inlet upstream of the sample monitoring area. A high activity alarm from this monitor will also isolate flow to the steam generator blowdown demineralizers.

11.5.2.5.10 Reactor Coolant Letdown Monitor (High and Low Range)

The gross activity of the reactor coolant is continuously monitored by a reactor coolant letdown off-line liquid monitor consisting of a low-range detector and a high-range detector. Samples are drawn from the reactor coolant letdown line and delayed to permit sufficient decay of the N-16 isotope before they pass by the detectors. In this system, large variations in the activity levels are possible depending upon the amount of fission products leaked to the coolant. The detectors are located adjacent to the off-line sample tubing. The ranges and sensitivities of the detectors provide one decade of overlap at the highest decade of the low-range detector. Collimating plugs are used with the low-range detector, after the reactor coolant activity builds up, to provide redundancy for the high-range channel.

The high-range detector continuously monitors the reactor letdown during normal operation. The monitor is provided with three adjustable single channel analyzers with variable energy settings. Each of the two detectors has its own separate indication. This system can be flushed with clean water from a flush line upstream of the sample monitoring area.

11.5.2.5.11 Gaseous Waste Surge Tank Transfer Line Monitor

The gaseous waste surge tank transfer line on-line gas monitor monitors radioactive gas composed mostly of hydrogen before flow is transferred to BVPS-1 or BVPS-2 waste gas storage tanks.

11.5.2.5.12 Auxiliary Steam Condensate Monitor

The auxiliary steam condensate monitor analyzes a continuous sample drawn from the condensate header. The detector is located in a well in an off-line liquid sampler. An activity indication alerts the operator to a leak in any of the radioactive systems serviced by the auxiliary steam system.

11.5.2.5.13 Evaporator Reboiler Steam Condensate Monitor

The evaporator reboiler is serviced by the evaporator reboiler steam condensate monitor during shutdown from the closed steam and condensate system of the auxiliary boiler. The purpose of the monitor is to detect any radioactive contamination in the clean service side resulting from a tube rupture. This evaporator is used to process liquid waste; therefore, the reboiler process fluid side is radioactive.

11.5.2.5.14 Gaseous Waste Storage Tank Monitor

The gaseous waste storage tank off-line gas monitor is used to monitor gases contained within the gaseous waste storage tanks prior to discharge to the BVPS-1 gaseous waste disposal system.

11.5.2.5.15 N-16 Main Steam Line Monitors

The N-16 main steam line monitors, one for each line, are adjacent-to-line monitors located in the main steam valve area. The purpose of these monitors is to provide an early indication of a steam generator primary to secondary tube leak that may be a precursor to a tube rupture and to identify the affected steam generator.

The detectors are located on the 32" Main Steam Lines, in the Main Steam Valve Area. The processing equipment is located one elevation below, in the Rod Control Cable Vault Area.

The N-16 concentration in the reactor coolant system and the steam flow is a function of reactor power. An input to the processor, supplied from the Plant Computer System, adjusts the monitor response for changes in reactor power. The monitors are capable of detecting steam generator primary to secondary leakage of from 1 to 500 gallons per day on each steam generator. At plant power levels of 1160 MWt to 2900 MWt (~40% to 100% of maximum power) the monitor shall be capable of detecting N-16 at a minimum concentration of 1.7×10^{-7} μCi/cc, which corresponds to a nominal leak rate of 5 gallons per day.

Leak rate data for each steam generator and alarms generated by the processors actuate on the Main Control Room DRMS equipment.

11.5.2.6 Digital System Description

11.5.2.6.1 General

The overall DRMS is designed to provide the operator with display and recording capability and alarm information.

The DRMS includes the following major equipment:

1. Monitors and local processors for each process, effluent, airborne, and area monitor. The term local processor means that the processor is located in the BVPS-2 environs in the vicinity of the detector.

2. Main control room cabinets, as necessary, to house required DRMS equipment.

3. Control and display modules for Class 1E monitors and effluent monitors.

4. Safety-related recorders and related interface equipment.

5. Digital central processors.

6. Display and printers for location in the main control room.

7. Display and printer for location in the radiation protection office.

Each radiation monitor transfers information to the main control room via individual data links. The architecture of the overall system is redundant (but not safety grade) such that the failure of any one cable or component will not preclude communication between the central processors and monitors. The central processors incorporate redundant features so that they remain functional upon experiencing certain types of failures (such as central processing unit or disc memory failure). Loss of one monitor or display capability is tolerable due to a single failure.

All radiation monitors normally interface with the main control room operator and the health physicist via printers and CRTs connected to the central processor.

All Category I seismically-qualified Class 1E radiation monitors normally provide information via the central processor printers and displays but also read out, alarm, and record in the main control room via safety-related control and display modules and printers. All effluent monitors also read out and alarm in the main control room cabinets via individual control and display modules.

Local digital readout and audible and visual alarms are provided for each process and effluent monitor.

The DRMS initiates alarm messages when parameters being measured have exceeded a reference value. Equipment malfunctions, such as conditions of low flow and power failure, also generate system alarms. Administrative controls are used to ensure the low process flow alarm on the Liquid Waste Process Effluent monitor performs automatically. All data pertaining to these alarm conditions are automatically displayed and recorded in the main control room and in the radiation protection laboratory for Class 1E and effluent monitors, the alarms are also displayed via their respective control and display modules. Alarm data for the Class 1E monitors are automatically recorded on the Class 1E printers.

The DRMS is capable of electronically checking all monitors, both automatically and upon demand. These checks are logged, and in the event of a mechanical failure or the inability to obtain a proper reading, an alarm is generated.

11.5.2.6.2 Power Supplies to Electronic Equipment

Category I radiation monitoring instrumentation has available power from Class 1E buses in the event of a power failure or postulated accidents Category II radiation monitoring instrumentation is powered from non-Class 1E power, which is a highly reliable power source, and is backed-up by the non-safety onsite diesel generator.

Each monitor's DAU is provided with a 48 hour battery backup.

11.5.2.6.3 Safety-Related Control/Display and Recording Modules

Each Class 1E monitor has a control/display module and a recording module located in the main control room cabinet. Each control/display module has separate alarm lights and horns, and power indication lights in duplication of those required locally.

11.5.2.6.4 Local Processors

Each local processor has a local control and indication panel with control switches, a light emitting diode (LED) digital display meter, and alarm lights.

The radiation reading exhibited on this LED display is in units of μCi/cc for process and airborne monitors, and mRem/hr or Rem/hr for area monitors.

Separate alarm lights are provided for high and alert radiation levels and channel failure. A power indication light is illuminated during normal operation for UPS.

Each area monitor has its own LED digital display indicator, alarm lights, and power indication light.

All local processor programming is performed on read only memory. All set points and dynamic values are stored on random access memory.

Optical isolation, or its equivalent, is provided for all signals between monitors or between monitors and the main control room. The isolators prevent damage to electronic circuits if spurious 1,500 V or 480 V dc is applied directly to or electrostatically, electromagnetically, or inductively introduced into, the signal cable conductors. Additionally, these isolators prevent ground loop current flows.

11.5.2.7 Central Processors (DPS)

11.5.2.7.1 General

This section describes the functional requirements of the central processors and peripheral equipment.

11.5.2.7.2 Digital Radiation Monitoring System Central Processor Software Development and Documentation

Computer software has been developed for data handling activities such as display formats, typewriter print formats, calculations, trending, report generation, logging, and historical data storage.

Adequate safeguards are provided to protect against inadvertent or unauthorized program changes.

11.5.2.7.3 Summary of Digital Central Processor Functions

The two DRMS central processors include programming to perform the following functions:

1. Alarm calculation, reporting, and sequence of events logging,

2. Communications,

3. Command initiation (message transmission and acknowledgment for locally performed control functions),

4. Individual, group, and graphic displays,

5. Utility typewriter,

6. Periodic and demand logs,

7. Radioisotope release report generation,

8. Analyses and calculations,

9. Historical data recording and retrieval,

10. Standard computer trending, averaging, etc., and

11. Automatic error checking, correction, and alarming.

11.5.2.7.4 Data Files

A data file maintained on redundant bulk storage at the central processors for each radiation monitor contains the following information for all monitors:

1. Monitor identification number,
2. Monitor name,
3. Monitor address,
4. Latest radiation value,
5. Engineering unit conversion factor,
6. Radiation value in engineering units,
7. Radiation data statistical classification (A, B, C, or D),
8. Background radiation value,
9. High alarm set point,
10. High alarm status,
11. Alert alarm set point,
12. Alert alarm status,
13. Power failure alarm status,
14. Channel failure alarm status,
15. Bypass alarm status (monitor in calibration),
16. Check source status (in or out),
17. Latest high voltage value, and
18. High voltage scan interval.

The data file for off-line particulate monitors includes the following additional information:

1. Filter paper drive speed (off, normal, or fast),
2. Filter paper alarm status,
3. No flow alarm status,
4. Purge status (purge or no purge), and

5. Pump status (on or off).

The data file for off-line liquid monitors includes the following additional information:

1. Purge status (purge or no purge),

2. Pump status (on or off),

3. No flow alarm status, and

4. High temperature alarm status.

The information described previously is stored on file so that graphic displays and print formats can be developed using any of the information stored in the files.

11.5.2.8 In-service Inspection, Calibration, and Maintenance

The operability of each channel of the process and effluent DRMS is checked frequently with a source which is positioned by controls in the main control room or manually at the monitor. Both systems are functionally tested periodically and calibrated during each refueling.

The periodic test is used to verify the operability of alarms, automatic valves, and pumps. This is performed by lowering the set point below background, starting pumps where necessary, and verifying the operation of alarms, valves, and pumps.

Calibration of all monitors is conducted in accordance with the Offsite Dose Calculation Manual. Generally, this procedure is conducted during refueling to minimize the effect on BVPS-2 operation and to limit occupational exposures. Calibration is conducted using a source which will allow indication in a low-, mid-, and high-range, except for the extended range monitors where high-range calibration could result in undesirable exposures to plant personnel. For these monitors, calibration is conducted in the low range of the monitor, and linearity is verified by the manufacturer at the time of purchase and by a simulated electronic signal introduced as close to the radiation detector as practical.

Provisions for minimizing sample deposition and decontaminating sample chambers and monitor components are detailed in Sections 11.5.2.2 and 11.5.2.3.

11.5.2.9 Sampling

Section 9.3.2 discusses the various process and effluent samples taken periodically for chemical and radiochemical analysis.

Section 9.3.2.1 lists liquid process and effluent samples to be taken periodically and monitored for radioactivity at the primary sample sink. Sampling not discussed in Section 9.3.2 is discussed in the sections describing the individual system designs. The fluid systems are sampled via local sampling connections. The Offsite Dose Calculation Manual describes the various liquid samples analyzed for gross beta-gamma activities, including the sampling frequencies. Section 11.2 gives the expected and design concentrations of radionuclides in the various process and effluent liquid samples.

Prior to collecting a sample, liquid sample lines are purged of stagnant water and undissolved solids to ensure that a representative sample is obtained.

Sample taps suitable for connection to a sampling chamber are provided for all off-line effluent, gas, and particulate monitors to obtain a sample for laboratory gamma spectrum analysis. This analysis provides the data needed to evaluate the gross radioactivity measurements.

11.5.3 Effluent Monitoring and Sampling

In accordance with the requirements of GDC 64 (Section 3.1.2.64), each effluent discharge path is continuously monitored for radioactive effluents resulting from normal operations, anticipated operational occurrences, and postulated accidents (Figures 11.5-1 and 11.5-2). Each effluent monitor alarms if the radionuclide concentration exceeds a predetermined level.

The elevated release high-range and main steam discharge high-range monitors are capable of monitoring all postulated primary system accident releases through normal gaseous effluent paths.

All steam releases are monitored by the main steam discharge high-range detectors.

The main steam discharge high-range monitors and the elevated release high-range monitors have the required range to meet the intent of NUREG-0737, Action Item II.F.1, Attachment 1. The elevated release high-range monitor has the sampling capabilities required to meet the intent of NUREG-0737, Action Item II.F.1, Attachment 2.

11.5.4 Process Monitoring and Sampling

The sampling system (Section 11.5.2.9) allows periodic sampling of radioactive waste process streams to determine the radiation levels in the radioactive waste process systems. Section 11.2 gives recirculation pumps and tank capacities for liquid waste tanks.

The isokinetic nozzles used for obtaining uniform samples (Section 11.5.2.2) are designed according to the guidelines of ANSI N13.1-1969.

A sampling room (Section 9.3.2) is provided for remote sampling. Recirculation loops are separated from the sample taps by shielding. Sample sink drains are collected and sent to various systems, depending on the nature of the sample being taken, for reclaiming or processing, as necessary. Sample sinks are provided with ventilation exhaust hoods.

Monitors which initiate automatic functions (valve closure, pump start, flow diversion, etc.) are listed in Table 11.5-1, and the function initiated is described in Sections 11.5.2.4 and 11.5.2.5.

Monitoring of fuel storage and handling areas for radioactive waste is described in Section 12.3.4.

11.5.5 References for Section 11.5

Institute of Electrical and Electronics Engineers (IEEE) 1971. Criteria for Protection Systems for Nuclear Power Generating Stations, IEEE Standard 279-1971.

Institute of Electrical and Electronics Engineers 1974. Qualifying Class 1E Equipment for Nuclear Power Generating Stations, IEEE Standard 323-1974.

Institute of Electrical and Electronics Engineers 1975. Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations, IEEE Standard 344-1975.

U.S. Nuclear Regulatory Commission (USNRC) 1979. Draft Radiological Technical Specification for PWRS. NUREG-0472.

U.S. Nuclear Regulatory Commission 1980. Clarification of TMI Action Plan Requirements. NUREG-0737.

Tables for Section 11.5

TABLE 11.5-1

TECHNICAL REQUIREMENTS FOR PROCESS AND EFFLUENT RADIATION MONITORS

Monitor Designation	Monitor Type	Seismic Category	Number of Monitors Required	Fixed & Moving Filter & Charcoal Cartridge	Pump Required	Effluent/ Process Flow (cfm)	Automatic Control Function	Duct Size/[1] Pipe Dia. (In)/Mtl. /Schd	Electrical Class 1E
Effluent									
Main steam discharge high range	Off-line steam	I	1	(3)	(3)	600 (lb/hr)	No	3/8/SS/ Tubing 10/CS/60	Yes
Ventilation vent	Off-line gas and particulate	II	1	X	X	26,000	No	48	No
Elevated release	Off-line gas	I	1	X	X	49,500 (normal) 43,000 (accident)	No	54 x 58	Yes
Liquid waste process effluent [8]	Off-line liquid	II	1	(3)	X	(3, 4)	Yes	3/4/SS/10S (Sample line)	No
Waste gas storage vault	Off-line gas and particulate	II	1	X	X	1,000	Yes	14	No
Condensate polishing building vent stack	Off-line gas and particulate	II	1	X	X	25,500	Yes	36	No
Decontamination building	Off-line gas and particulate	II	1	X	X	12,400	Yes	48 x 18	No

TABLE 11.5-1 (Cont)

Monitor Designation	Monitor Type	Seismic Category	Number of Monitors Required	Fixed & Moving Filter & Charcoal Cartridge	Pump Required	Effluent/ Process Flow (cfm)	Automatic Control Function	Duct Size/(1) Pipe Dia. (In)/Mtl. /Schd	Electrical Class 1E
Process									
Recirculation spray heat exchanger service water	Off-line liquid	I	4	(3)	X	(4) 0-6,000 gpm	Yes	1/SS/tubing 1/SS/40S (Sample line)	Yes
Containment purge	In-line gas	I	2	(3)	(3)	7,500(9)	Yes(9)	30	Yes
Air ejector discharge	In-line gas	II	1	(3)	(3)	3-20	No	6/CS/40	No
Air ejector delay bed exhaust	Off-line gas	II	1	(3)	X	3-40(4)	No	3/4/SS/tubing	No
Aerated vent transfer line	Off-line gas	II	1	(2)	X	125 (4) (normal) 1,125 (maximum)	No	3/4/SS/tubing	No
Component cooling service water	Off-line liquid	II	1	(3)	X	(3, 4)	No	1/SS/tubing (Sample line)	No
Component cooling heat exchanger service water	Off-line liquid	II	1	(3)	X	(3, 4)	No	1/SS/tubing (Sample line)	No
Steam generator blowdown sample (5)	Off-line liquid	II	1	(3)	X	(3, 4)	Yes	1/2/SS/tubing (Sample line)	No
Component cooling	Off-line liquid	II	1	(3)	X	(3, 4)	No	1/SS/tubing (Sample line)	No

TABLE 11.5-1 (Cont)

Monitor Designation	Monitor Type	Seismic Category	Number of Monitors Required	Fixed & Moving Filter & Charcoal Cartridge	Pump Required	Effluent/ Process Flow (cfm)	Automatic Control Function	Duct Size/(1) Pipe Dia. (In)/Mtl. /Schd	Electrical Class 1E
Reactor coolant letdown high range and low range(6)	ATL liquid	II	1	(3)	(3)	(3)	No	(3)	No
Gaseous waste surge tank transfer line	ATL gas	II	1	(3)	(3)	(3)	No	(3)	No
Auxiliary steam condensate (7, 8)	Off-line liquid	II	1	(3)	X	(3, 4)	No	1/SS/tubing (Sample line)	No
Evaporator reboiler condensate (7, 8)	Off-line liquid	II	1	(3)	X	(3, 4)	No	1/SS/tubing (Sample line)	No
Waste gas storage tanks	ATL gas	II	1	(3)	(3)	(3)	No	(3)	No
Main Steam Line N-16 Monitors	Adjacent to line	II	3	(3)	(3)	3.780×10^6 lb/hr @ 100% Rx Pwr	No	32"/CS/Main Steam Lines	No

NOTES:

1. CS - Carbon Steel.
   SS - Stainless Steel.
   ATL - Adjacent to Line.
2. Only fixed filter and charcoal cartridge holders to be provided.
3. Not applicable.
4. Sample line flow is nominally 2 cfm for gases and 3 gpm for liquids.

NOTES (continued):

5. This monitor automatically actuates external equipment - Form C output contacts required.
6. The two reactor coolant letdown monitors are combined into a two-detector/one-sampler configuration with separate channel readout.
7. This monitor is provided with a temperature sensor and automatic shutoff of the liquid effluent into the monitor sample volume in the event of high temperature.
8. This monitor is provided with a sample cooling heat exchanger.
9. Flow and/or automatic control are not requirements and may differ from that listed.

TABLE 11.5-2

DETECTOR TECHNICAL REQUIREMENTS FOR PROCESS AND EFFLUENT RADIATION MONITORS

Monitor Designation	Detector Description	Maximum Background (mRem/hr)	Reference Isotope	Minimum Sensitivity (μCi/cc)	Detectable Range (μCi/cc)
Effluent					
Main steam discharge high range	Gas (Steam)	2.5	Kr-88	$1x10^{-2}$	10^{-2} to 10^{+3}
Ventilation vent	Particulate	1.0	I-131	$1x10^{-10}$	10^{-10} to 10^{-5}
	Gas		Xe-133, Kr-85	$1x10^{-7}$	10^{-7} to 10^{-2}
Elevated release particulate skid	Particulate	1.0	I-131	$1x10^{-10}$	10^{-10} to 10^{-5}
Elevated release gas detector skid	Gas 1	1.0	Xe-133	$1x10^{-7}$	10^{-7} to 10^{-1}
	Gas 2		Xe-133	$1x10^{-4}$	10^{-4} to 10^{+2}
	Gas 3		Xe-133	$1x10^{-1}$	10^{-1} to 10^{+5}
Liquid waste process effluent	Liquid	2.5	Cs-137	$1x10^{-7}$	10^{-7} to $^{-2}$
Component cooling heat exchanger service water	Liquid	2.5	Cs-137	$1x10^{-6}$	10^{-6} to 10^{-1}
Waste gas storage vault	Particulate	2.5	I-131	$1x10^{-10}$	10^{-10} to 10^{-5}
	Gas		Xe-133, Kr-85	$1x10^{-7}$	10^{-7} to 10^{-2}
Condensate polishing vent stack	Particulate	2.5	I-131	$1x10^{-10}$	10^{-10} to 10^{-5}
	Gas		Xe-133, Kr-85	$1x10^{-7}$	10^{-7} to 10^{-2}
Decontamination Building	Particulate	2.5	I-131	$1x10^{-10}$	10^{-10} to 10^{-5}
	Gas		Xe-133, Kr-85	$1x10^{-7}$	10^{-7} to 10^{-2}
Process					
Recirculation spray heat exchanger service water	Liquid	5.0	Cs-137	$1x10^{-4}$	$10x^{-4}$ to 10^{+1}
Containment purge	Gas	2.5	Xe-133	$1x10^{-6}$	10^{-6} to 10^{-1}
Air ejector discharge	Gas	0.75	Xe-133	$1x10^{-6}$	10^{-6} to 10^{-1}

TABLE 11.5-2 (Cont)

Monitor Designation	Detector Description	Maximum Background (mRem/hr)	Reference Isotope	Minimum Sensitivity (μCi/cc)	Detectable Range (μCi/cc)
Air ejector delay bed exhaust	Gas	2.5	Xe-133	$1x10^{-6}$	10^{-6} to 10^{-1}
Aerated vent transfer line	Gas	2.5	Xe-133	$1x10^{-6}$	10^{-6} to 10^{-1}
Component cooling service water	Liquid	2.5	Cs-137	$1x10^{-6}$	10^{-6} to 10^{-1}
Steam generator blowdown sample	Liquid	2.5	Cs-137	$1x10^{-6}$	10^{-6} to 10^{-1}
Reactor coolant letdown high range and low range	Liquid	2.5	Cs-137 Cs-137	$1x10^{-1}$ $1x10^{-3}$	10^{-1} to 10^{4} 10^{-3} to 10^{1}
Gaseous waste surge tank transfer line	Gas	2.5	Kr-85	$1x10^{-2}$	10^{-2} to 10^{3}
Component cooling water	Liquid	2.5	Cs-137	$1x10^{-6}$	10^{-6} to 10^{-1}
Auxiliary steam condensate	Liquid	2.5	Cs-137	$1x10^{-6}$	10^{-6} to 10^{-1}
Evaporator reboiler condensate	Liquid	2.5	Cs-137	$1x10^{-6}$	10^{-6} to 10^{-1}
Waste gas storage tanks	Gas	2.5	Xe-133	$5x10^{-2}$	10^{-2} to 10^{4}
Main Steam Line N-16 Monitors	Adjacent to Line	*	N-16	$2x10^{-8}$	10^{-8} to 10^{-2}

* Background radiation level is not limiting due to the detector energy range.

TABLE 11.5-3

CONDITIONS OF SERVICE FOR PROCESS AND EFFLUENT RADIATION MONITORS

Monitor Designation	Sample Fluid	Sample Fluid: Operating/ Design Temperature (°F)	Sample Fluid: Operating/ Design Pressure	Ambient Conditions: Dry Bulb Temp Range (°F)	Ambient Conditions: Max Pressure (psig)	Relative Humidity Range (%)	Building/Location Elevation (ft-in)
Effluent							
Main steam discharge high range	Gas (steam)	517/560	790/1085 psig	55 to 120	0	20 to 90	MSVH/796-10
Ventilation vent	Air	50/140	8 inches WG	65 to 120	0	20 to 90	Auxiliary bldg/ 773 - 6
Waste gas storage vault	Air	60/120	-6 inches WG	65 to 150	-1/4 inch WG	50 to 100	Decontamination bldg
Elevated release	Air	50/140	8 inches WG	65 to 120	0	20 to 90	Auxiliary bldg/ 773 - 6
Liquid waste process effluent*,****	Borated water	130/200	70/100 psig	65 to 120	0	20 to 90	Auxiliary bldg/ 718 - 6
Condensate polishing bldg vent stack	Air	50/140	8 inches WG	50 to 120	0	30 to 95	Condensate polishing bldg/794 - 6
Decontamination bldg	Air	70/120	-21 inches WG	65 to 150	-1/4 inch WG	50 to 100	Fuel bldg/766-4
Process							
Recirculation spray heat exchanger service water	River water	40/160	25/150 psig	60 to 122	0	20 to 90	Diesel bldg/759
Containment purge**	Air	35/140	-6.3/45.7 psig	85 to 135	11.6	30 to 60	Containment/782

TABLE 11.5-3 (Cont)

Monitor Designation	Sample Fluid	Sample Fluid: Operating/ Design Temperature (°F)	Sample Fluid: Operating/ Design Pressure	Ambient Conditions: Dry Bulb Temp Range (°F)	Ambient Conditions: Max Pressure (psig)	Relative Humidity Range (%)	Building/Location Elevation (ft-in)
Air ejector discharge	Process gas	125/140	1.5/3 psig	50 to 120	0	30 to 100	Turbine bldg/761 - 4
Air ejector delay bed exhaust	Process gas	77/140	1.5/3 psig	65 to 130	0	20 to 100	Auxiliary bldg/ 755 - 6
Aerated vent transfer line	Process gas	100/140	9 to 14 inches WG	65 to 120	0	20 to 90	Auxiliary bldg/ 735 - 6
Component cooling service water	River water	40/120	55/150 psig	65 to 120	0	20 to 90	Auxiliary bldg/ 710 - 6
Component cooling heat exchanger service water	River water	40/120	55/150 psig	65 to 120	0	20 to 90	Auxiliary bldg/ 710 - 6
Steam generator blowdown sample***	Water	110/120	35/50 psig	65 to 120	0	20 to 90	Auxiliary bldg/ 718 - 6
Component cooling	Water	85/125	110/150 psig	65 to 120	0	20 to 90	Auxiliary bldg/ 718 - 6
Reactor coolant letdown high range and low range***	Borated water	115/130	75/200 psig	65 to 120	0	20 to 90	Auxiliary bldg/ 735 - 6
Gaseous waste surge tank transfer line*	Process gas (hydrogen)	180/180	65 to 100/100 psig	65 to 120	0	20 to 90	Auxiliary bldg/ 735 - 6
Auxiliary steam condensate*,****	Water	212/212	10/25 psig	65 to 120	0	20 to 90	Auxiliary bldg/ 718 - 6
Evaporator reboiler steam condensate*,****	Water	212/212	10/25 psig	50 to 120	0	30 to 75	Waste handling bldg/ 722 - 6
Waste gas storage tanks	Process gas (hydrogen)	65/180	65/100	65 to 150	-1/4 inch WG	50 to 100	Decontamination bldg/735

TABLE 11.5-3 (Cont)

		Sample Fluid		Ambient Conditions			
Monitor Designation	Sample Fluid	Operating/ Design Temperature (°F)	Operating/ Design Pressure	Dry Bulb Temp Range (°F)	Max Pressure (psig)	Relative Humidity Range (%)	Building/Location Elevation (ft-in)
Main Steam Line N-16 Monitors	Does not sample (Adjacent to Line)	517 deg F steam temperature	N/A	40 - 130	0	20-90	Main Steam Valve Area

NOTES:

* This monitor is provided with a temperature sensor and automatic shutoff of the liquid effluent into the monitor sample volume in the event of high temperature.

** This monitor is located within the reactor containment but does not require post-accident environmental qualification.

*** This monitor does not require automatic temperature protection. Purchaser supplies temperature sensing equipment and automatic shutoff valves external to radiation monitor

**** This monitor is provided with a sample cooling heat exchanger.

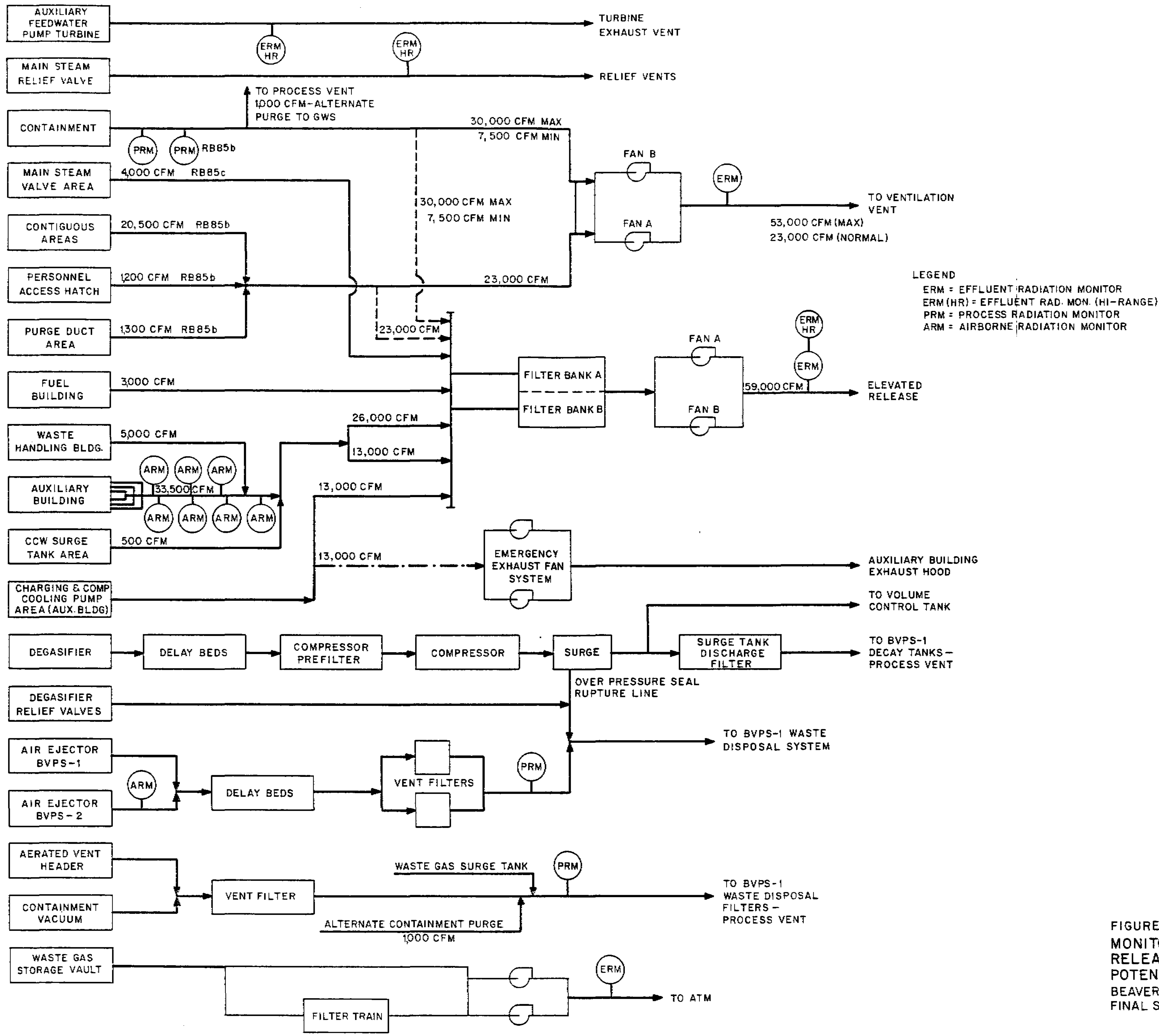


FIGURE 11.5-1
MONITOR LOCATIONS FOR RELEASE PATHS FOR POTENTIALLY RADIOACTIVE GASES
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

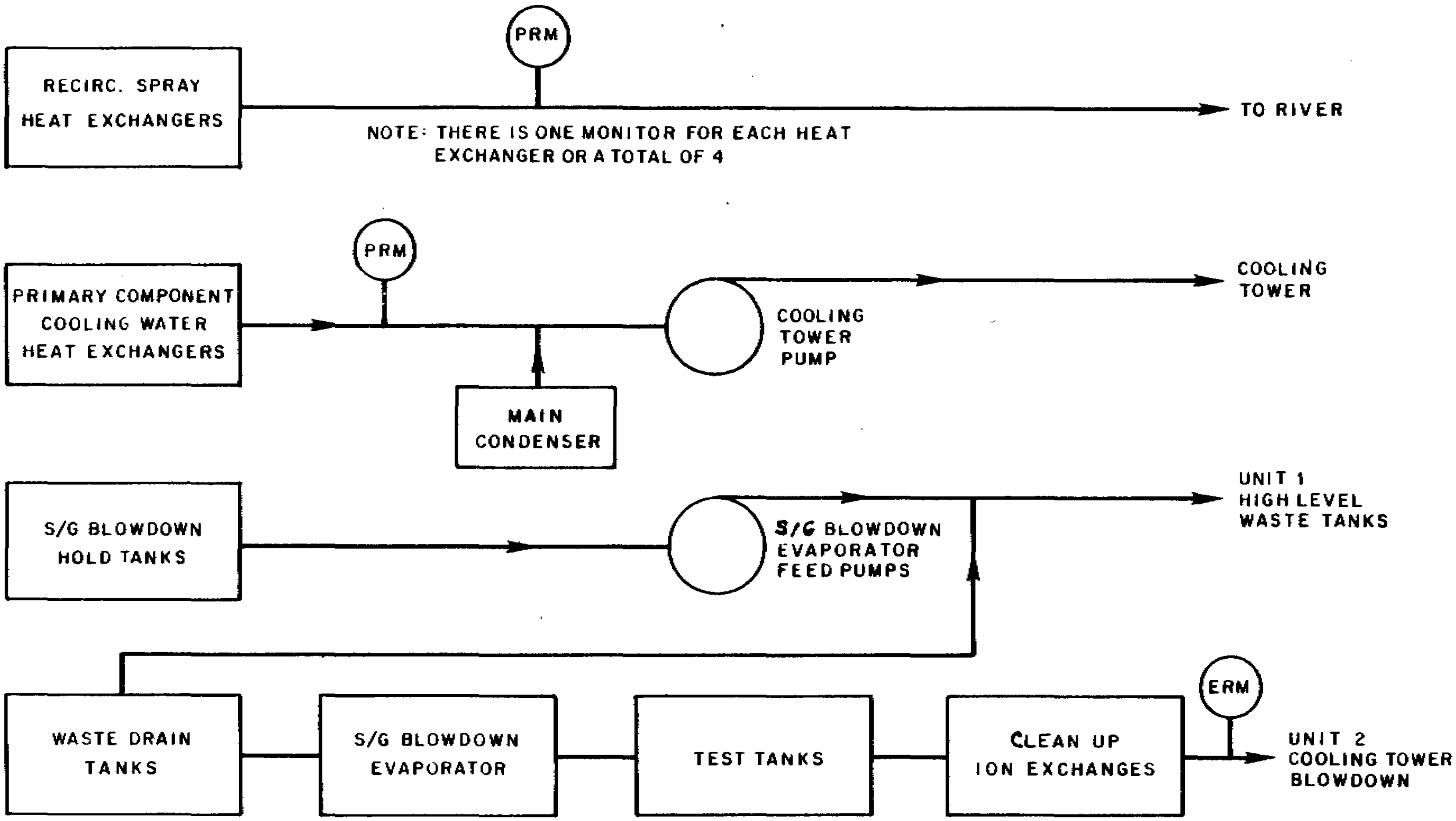


FIGURE 11.5-2
MONITORING LOCATIONS FOR RELEASE PATHS FOR POTENTIAL RADIOACTIVE LIQUIDS
BEAVER VALLEY POWER STATION-UNIT 2
FINAL SAFETY ANALYSIS REPORT

NOTE
PRM - PROCESS RADIATION MONITOR
ERM - EFFLUENT RADIATION MONITOR

APPENDIX 11A

SUMMARY OF ANNUAL RADIATION DOSES

APPENDIX 11A

SUMMARY OF ANNUAL RADIATION DOSES

Appendix 11A evaluates the radiation exposure to the general public or to individuals thereof due to radwaste effluents discharged to the environment based on the original design of the liquid and gaseous radwaste system. Appendix 11A has been retained to provide historical information. Actual concentrations of radioactive material in effluents and compliance with design objectives of 10 CFR 50 Appendix I is controlled in accordance with the Offsite Dose Calculation Manual. Actual effluent isotopic release curie and dose information is provided in the Annual Radioactive Effluent Release Reports.

The calculated annual radiation doses to maximum individuals from normal operation of Beaver Valley Power Station - Unit 1 (BVPS-1) and Beaver Valley Power Station - Unit 2 (BVPS-2) are presented in Tables 11A-1, 11A-2, 11A-3, 11A-4, 11A-5, 11A-6, 11A-7, 11A-8, 11A-9, 11A-10, 11A-11, 11A-12, 11A-13, 11A-14, 11A-15, 11A-16, 11A-17, 11A-18, 11A-19 and 11A-20. Table 11A-21 demonstrates that the calculated annual radiation doses are below the design objectives of 10 CFR 50, Appendix I Annex (USNRC 1975). In providing guidance for the implementation of Appendix I, the U.S. Nuclear Regulatory Commission has made use of the maximum exposed individual approach. "Maximum" individuals are characterized as maximum with regard to food consumption, occupancy, and other usage of the region in the vicinity of the plant site and, as such, represent individuals with reasonable deviations from the "average" individuals considered representative of the population in general.

For gaseous radioactive releases, the pathways that were analyzed include: standing on contaminated ground, ingestion of vegetation, and inhalation of and submersion in gaseous effluents. These pathways were considered for all of the resident locations. Resident locations having cows, goats, and meat animals were also analyzed for ingestion of cow milk, goat milk, and beef meat, respectively. Additionally, the doses associated with ingestion of deer, rabbit, pheasant, and grouse were conservatively added to all resident locations analyzed.

The highest calculated organ dose due to radioiodines and particulates is 7.7 mRem/yr. This represents the dose to the thyroid of a child who lives at the residence location 1,432 meters northwest of the site. The majority of this dose is due to consumption of stored produce. The highest calculated external exposure rates to the whole body and skin from immersion in noble gases at an occupied location are 1.1 and 2.4 mRem/yr, respectively. These also occur at the residence location 1,432 meters northwest of the site.

The highest calculated beta and gamma air doses at an unoccupied location from noble gas releases are 6.8 and 6.3 mrad/yr, respectively. These occur at the northern shore of the Ohio River, 567 meters northwest of the site.

For liquid releases, the maximum individual consumed fish whose principal habitat was assumed to be the edge of the initial mixing zone. This location was also conservatively used in calculating doses from swimming. Boating was assumed to take place only in the discharge outfall area, while shoreline recreation was analyzed at the junction of Little Beaver Creek with the Ohio River, 4 miles down river. The Chester, West Virginia, public water supply, 7.1 miles down river, was analyzed for ingestion of potable water. The maximum individual was also assumed to consume fresh and stored vegetation that was irrigated with river water withdrawn from this water supply.

The highest calculated whole body dose to an individual from liquid pathways is 3.5 mRem/yr. This occurs in the adult age group. The highest calculated organ dose to an individual from liquid pathways was 4.7 mRem/yr to a teen liver. These doses are primarily due to ingestion of fish.

The calculated annual doses to the population residing within a 50 mile radius of the site are presented in Table 11A-22. Population doses were calculated for a projected population of 3,949,000 people residing within 50 miles of the site in the year 2010.

Gaseous pathways considered in the population dose analysis include: inhalation, exposure to ground deposits, ingestion of vegetation, cow milk, and beef meat, and submersion in noble gases. The calculated annual dose to the 50-mile population is 14.9 man-Rem/yr whole body and 22.3 man-Rem/yr thyroid.

Liquid pathways considered in the population dose analysis include: ingestion of potable water and fish, shoreline recreation, and swimming and boating. The calculated annual dose to the 50-mile population is 0.34 man-Rem/yr whole body and 1.5 man-Rem/yr thyroid.

The calculated annual gaseous and liquid doses to the contiguous United States population are also presented in Table 11A-22. For liquid effluents, the calculated doses to the contiguous United States population are 0.35 man-Rem/yr whole body and 1.5 man-Rem/yr thyroid. For gaseous effluents, the calculated doses to the contiguous United States population are 55 man-Rem/yr whole body and 63 man-Rem/yr thyroid.

Details of the models and assumptions used in calculating the dose information provided in this section are discussed in Appendix 5C of the BVPS-2 Environmental Report - Operating License Stage.

References for 11A

U.S. Nuclear Regulatory Commission (USNRC) 1975. Concluding Statement of Position of the Regulatory Staff. Docket RM-50-2, Washington, D.C. 40 FR 40816.

U.S. Environmental Protection Agency (USEPA) 1972. Estimates of Ionizing Radiation Doses in the United States. ORP-CSD-72-1.

Tables for Appendix 11A

TABLE 11A-1

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP
FROM GASEOUS EFFLUENTS

Residence Location at 1,432 m NW
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		6.2×10^{-1}	7.3×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}
Inhalation		1.6	0.0	1.4×10^{-1}	1.6	2.3	1.6	1.8	1.6
Fresh vegetation		3.6×10^{-1}	0.0	4.7×10^{-2}	3.7×10^{-1}	9.6×10^{-1}	3.4×10^{-1}	3.3×10^{-1}	3.4×10^{-1}
Stored vegetation		1.9	0.0	2.4×10^{-1}	2.0	1.8	1.8	1.8	1.8
Deer	1,577 m ESE	2.4×10^{-3}	0.0	8.1×10^{-3}	2.5×10^{-3}	2.4×10^{-3}	2.3×10^{-3}	2.2×10^{-3}	2.4×10^{-3}
Rabbit	567 m NW	2.1×10^{-4}	0.0	2.6×10^{-5}	2.2×10^{-4}	3.3×10^{-4}	2.0×10^{-4}	1.9×10^{-4}	2.2×10^{-4}
Grouse	567 m NW	1.7×10^{-5}	0.0	1.7×10^{-6}	1.7×10^{-5}	2.4×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	1.7×10^{-5}
Pheasant	567 m NW	1.4×10^{-4}	0.0	1.4×10^{-5}	1.4×10^{-4}	1.9×10^{-4}	1.3×10^{-4}	1.3×10^{-4}	1.4×10^{-4}
Total dose		4.5	7.3×10^{-1}	1.1	4.6	5.7	4.4	4.6	4.4

TABLE 11A-2

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE TEEN GROUP
FROM GASEOUS EFFLUENTS

Residence Location at 1,432 m NW
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		6.2×10^{-1}	7.3×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}
Inhalation		1.6	0.0	1.9×10^{-1}	1.6	2.5	1.6	1.9	1.6
Fresh vegetation		2.3×10^{-1}	0.0	4.1×10^{-2}	2.5×10^{-1}	7.2×10^{-1}	2.3×10^{-1}	2.2×10^{-1}	2.2×10^{-1}
Stored vegetation		2.5	0.0	4.0×10^{-1}	2.7	2.3	2.4	2.3	2.4
Deer	1,577 m ESE	3.1×10^{-3}	0.0	1.2×10^{-2}	3.3×10^{-3}	3.1×10^{-3}	3.0×10^{-3}	2.9×10^{-3}	3.1×10^{-3}
Rabbit	567 m NW	1.2×10^{-4}	0.0	2.1×10^{-5}	1.4×10^{-4}	2.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}	1.3×10^{-4}
Grouse	567 m NW	9.9×10^{-6}	0.0	1.4×10^{-6}	1.1×10^{-5}	1.5×10^{-5}	9.7×10^{-6}	9.5×10^{-6}	1.0×10^{-5}
Pheasant	567 m NW	8.2×10^{-5}	0.0	1.2×10^{-5}	8.8×10^{-5}	1.2×10^{-4}	8.0×10^{-5}	7.8×10^{-5}	8.4×10^{-5}
Total dose		5.0	7.3×10^{-1}	1.3	5.2	6.1	4.9	5.0	4.8

TABLE 11A-3

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP
FROM GASEOUS EFFLUENTS

Residence Location at 1,432 m NW
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		6.2×10^{-1}	7.3×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}
Inhalation		1.4	0.0	2.6×10^{-1}	1.5	2.4	1.4	1.7	1.4
Fresh vegetation		2.7×10^{-1}	0.0	6.9×10^{-2}	3.0×10^{-1}	1.0	2.7×10^{-1}	2.6×10^{-1}	2.6×10^{-1}
Stored vegetation		3.8	0.0	9.2×10^{-1}	4.3	3.7	3.9	3.7	3.7
Deer	1,577 m ESE	4.1×10^{-3}	0.0	1.7×10^{-2}	4.4×10^{-3}	4.2×10^{-3}	4.1×10^{-3}	4.0×10^{-3}	4.0×10^{-3}
Rabbit	567 m NW	1.4×10^{-4}	0.0	3.9×10^{-5}	1.6×10^{-4}	2.9×10^{-4}	1.5×10^{-4}	1.4×10^{-4}	1.4×10^{-4}
Grouse	567 m NW	1.2×10^{-5}	0.0	2.6×10^{-6}	1.3×10^{-5}	2.0×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}
Pheasant	567 m NW	9.7×10^{-5}	0.0	2.1×10^{-5}	1.1×10^{-4}	1.6×10^{-4}	9.7×10^{-5}	9.4×10^{-5}	9.6×10^{-5}
Total dose		6.1	7.3×10^{-1}	1.9	6.7	7.7	6.2	6.3	6.0

TABLE 11A-4

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP
FROM GASEOUS EFFLUENTS

Residence Location at 1,432 m NW
Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	6.2×10^{-1}	7.3×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}	6.2×10^{-1}
Inhalation	8.3×10^{-1}	0.0	1.8×10^{-1}	8.6×10^{-1}	1.8	8.4×10^{-1}	1.0	8.3×10^{-1}
Total dose	1.4	7.3×10^{-1}	8.0×10^{-1}	1.5	2.4	1.5	1.6	1.4

TABLE 11A-5

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP
FROM GASEOUS EFFLUENTS

Residence With Cow - Location at 4,538 m WNW
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		6.0×10^{-2}	7.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}
Inhalation		2.4×10^{-1}	0.0	2.4×10^{-2}	2.4×10^{-1}	3.4×10^{-1}	2.4×10^{-1}	2.7×10^{-1}	2.4×10^{-1}
Fresh vegetation		5.8×10^{-2}	0.0	1.6×10^{-2}	5.9×10^{-2}	1.3×10^{-1}	5.6×10^{-2}	5.5×10^{-2}	5.6×10^{-2}
Stored vegetation		3.1×10^{-1}	0.0	8.4×10^{-2}	3.2×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}
Cow milk		1.5×10^{-1}	0.0	4.7×10^{-2}	1.5×10^{-1}	2.7×10^{-1}	1.4×10^{-1}	1.3×10^{-1}	1.3×10^{-1}
Deer	1,577 m ESE	2.4×10^{-3}	0.0	8.1×10^{-3}	2.5×10^{-3}	2.4×10^{-3}	2.3×10^{-3}	2.2×10^{-3}	2.4×10^{-3}
Rabbit	567 m NW	2.1×10^{-4}	0.0	2.6×10^{-5}	2.2×10^{-4}	3.3×10^{-4}	2.0×10^{-4}	1.9×10^{-4}	2.2×10^{-4}
Grouse	567 m NW	1.7×10^{-5}	0.0	1.7×10^{-6}	1.7×10^{-5}	2.4×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	1.7×10^{-5}
Pheasant	567 m NW	1.4×10^{-4}	0.0	1.4×10^{-5}	1.4×10^{-4}	1.9×10^{-4}	1.3×10^{-4}	1.3×10^{-4}	1.4×10^{-4}
Total dose		8.2×10^{-1}	7.0×10^{-2}	2.4×10^{-1}	8.3×10^{-1}	1.1	8.0×10^{-1}	8.2×10^{-1}	7.9×10^{-1}

TABLE 11A-6

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE TEEN GROUP
FROM GASEOUS EFFLUENTS

Residence With Cow - Location at 4,538 m WNW
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		6.0×10^{-2}	7.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}
Inhalation		2.5×10^{-1}	0.0	3.3×10^{-2}	2.5×10^{-1}	3.6×10^{-1}	2.5×10^{-1}	2.9×10^{-1}	2.5×10^{-1}
Fresh vegetation		3.9×10^{-2}	0.0	1.5×10^{-2}	4.1×10^{-2}	9.6×10^{-2}	3.8×10^{-2}	3.7×10^{-2}	3.8×10^{-2}
Stored vegetation		4.1×10^{-1}	0.0	1.5×10^{-1}	4.3×10^{-1}	3.9×10^{-1}	4.0×10^{-1}	3.9×10^{-1}	4.0×10^{-1}
Cow milk		1.9×10^{-1}	0.0	8.5×10^{-2}	2.1×10^{-1}	3.9×10^{-1}	1.9×10^{-1}	1.8×10^{-1}	1.8×10^{-1}
Deer	1,577 m ESE	3.1×10^{-3}	0.0	1.2×10^{-2}	3.3×10^{-3}	3.1×10^{-3}	3.0×10^{-3}	2.9×10^{-3}	3.1×10^{-3}
Rabbit	567 m NW	1.2×10^{-4}	0.0	2.1×10^{-5}	1.4×10^{-4}	2.2×10^{-4}	1.2×10^{-4}	1.2×10^{-4}	1.3×10^{-4}
Grouse	567 m NW	9.9×10^{-6}	0.0	1.4×10^{-6}	1.1×10^{-5}	1.5×10^{-5}	9.7×10^{-6}	9.5×10^{-6}	1.0×10^{-5}
Pheasant	567 m NW	8.2×10^{-5}	0.0	1.2×10^{-5}	8.8×10^{-5}	1.2×10^{-4}	8.0×10^{-5}	7.8×10^{-5}	8.4×10^{-5}
Total dose		9.5×10^{-1}	7.0×10^{-2}	3.6×10^{-1}	9.9×10^{-1}	1.3	9.4×10^{-1}	9.6×10^{-1}	9.3×10^{-1}

TABLE 11A-7

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP
FROM GASEOUS EFFLUENTS

Residence with Cow - Location at 4,538 m WNW
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		6.0×10^{-2}	7.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}
Inhalation		2.2×10^{-1}	0.0	4.4×10^{-2}	2.2×10^{-1}	3.6×10^{-1}	2.2×10^{-1}	2.5×10^{-1}	2.2×10^{-1}
Fresh vegetation		4.6×10^{-2}	0.0	2.6×10^{-2}	5.0×10^{-2}	1.3×10^{-1}	4.7×10^{-2}	4.5×10^{-2}	4.5×10^{-2}
Stored vegetation		6.7×10^{-1}	0.0	3.7×10^{-1}	7.2×10^{-1}	6.5×10^{-1}	6.7×10^{-1}	6.5×10^{-1}	6.5×10^{-1}
Cow milk		3.0×10^{-1}	0.0	2.1×10^{-1}	3.5×10^{-1}	7.2×10^{-1}	3.1×10^{-1}	3.0×10^{-1}	2.9×10^{-1}
Deer	1,577 m ESE	4.1×10^{-3}	0.0	1.7×10^{-2}	4.4×10^{-3}	4.2×10^{-3}	4.1×10^{-3}	4.0×10^{-3}	4.0×10^{-3}
Rabbit	567 m NW	1.4×10^{-4}	0.0	3.9×10^{-5}	1.6×10^{-4}	2.9×10^{-4}	1.5×10^{-4}	1.4×10^{-4}	1.4×10^{-4}
Grouse	567 m NW	1.2×10^{-5}	0.0	2.6×10^{-6}	1.3×10^{-5}	2.0×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}
Pheasant	567 m NW	9.7×10^{-5}	0.0	2.1×10^{-5}	1.1×10^{-4}	1.6×10^{-4}	9.7×10^{-5}	9.4×10^{-5}	9.6×10^{-5}
Total dose		1.3	7.0×10^{-2}	7.3×10^{-1}	1.4	1.9	1.3	1.3	1.3

TABLE 11A-8

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP
FROM GASEOUS EFFLUENTS

Residence with Cow - Location at 4,538 m WNW
Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	6.0×10^{-2}	7.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}	6.0×10^{-2}
Inhalation	1.3×10^{-1}	0.0	3.1×10^{-2}	1.3×10^{-1}	2.5×10^{-1}	1.3×10^{-1}	1.5×10^{-1}	1.3×10^{-1}
Cow milk	4.7×10^{-1}	0.0	3.9×10^{-1}	5.7×10^{-1}	1.5	4.9×10^{-1}	4.7×10^{-1}	4.6×10^{-1}
Total dose	6.6×10^{-1}	7.0×10^{-2}	4.8×10^{-1}	7.6×10^{-1}	1.8	6.8×10^{-1}	6.8×10^{-1}	6.5×10^{-1}

TABLE 11A-9

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP
FROM GASEOUS EFFLUENTS

Residence with Goat - Location at 2,865 m ESE
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		$5.8x10^{-2}$	$6.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$
Inhalation		$3.2x10^{-2}$	0.0	$7.0x10^{-3}$	$3.2x10^{-2}$	$4.9x10^{-2}$	$3.2x10^{-2}$	$3.5x10^{-2}$	$3.2x10^{-2}$
Fresh vegetation		$1.3x10^{-2}$	0.0	$3.3x10^{-2}$	$1.4x10^{-2}$	$9.2x10^{-2}$	$1.2x10^{-2}$	$1.1x10^{-2}$	$1.2x10^{-2}$
Stored vegetation		$6.9x10^{-2}$	0.0	$1.8x10^{-1}$	$7.1x10^{-2}$	$6.1x10^{-2}$	$6.3x10^{-2}$	$6.0x10^{-2}$	$6.3x10^{-2}$
Goat milk		$6.6x10^{-2}$	0.0	$1.1x10^{-1}$	$7.5x10^{-2}$	$2.2x10^{-1}$	$5.3x10^{-2}$	$4.5x10^{-2}$	$4.2x10^{-2}$
Deer	1,577 m ESE	$2.4x10^{-3}$	0.0	$8.1x10^{-3}$	$2.5x10^{-3}$	$2.4x10^{-3}$	$2.3x10^{-3}$	$2.2x10^{-3}$	$2.4x10^{-3}$
Rabbit	567 m NW	$2.1x10^{-4}$	0.0	$2.6x10^{-5}$	$2.2x10^{-4}$	$3.3x10^{-4}$	$2.0x10^{-4}$	$1.9x10^{-4}$	$2.2x10^{-4}$
Grouse	567 m NW	$1.7x10^{-5}$	0.0	$1.7x10^{-6}$	$1.7x10^{-5}$	$2.4x10^{-5}$	$1.6x10^{-5}$	$1.6x10^{-5}$	$1.7x10^{-5}$
Pheasant	567 m NW	$1.4x10^{-4}$	0.0	$1.4x10^{-5}$	$1.4x10^{-5}$	$1.9x10^{-4}$	$1.3x10^{-4}$	$1.3x10^{-4}$	$1.4x10^{-4}$
Total dose		$2.4x10^{-1}$	$6.8x10^{-2}$	$4.0x10^{-1}$	$2.5x10^{-1}$	$4.8x10^{-1}$	$2.2x10^{-1}$	$2.1x10^{-1}$	$2.1x10^{-1}$

TABLE 11A-10

BVPS NUCLEAR POWER PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE TEEN GROUP
FROM GASEOUS EFFLUENTS

Residence with Goat - Location 2,865 m ESE
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		$5.8x10^{-2}$	$6.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$
Inhalation		$3.3x10^{-2}$	0.0	$9.9x10^{-3}$	$3.3x10^{-2}$	$5.4x10^{-2}$	$3.3x10^{-2}$	$3.7x10^{-2}$	$3.3x10^{-2}$
Fresh vegetation		$1.0x10^{-2}$	0.0	$3.1x10^{-2}$	$1.1x10^{-2}$	$7.4x10^{-2}$	$9.9x10^{-3}$	$9.1x10^{-3}$	$9.4x10^{-3}$
Stored vegetation		$1.0x10^{-1}$	0.0	$3.2x10^{-1}$	$1.2x10^{-1}$	$9.8x10^{-2}$	$1.0x10^{-1}$	$9.7x10^{-2}$	$9.9x10^{-2}$
Goat milk		$8.7x10^{-2}$	0.0	$2.0x10^{-1}$	$1.2x10^{-1}$	$3.5x10^{-1}$	$8.4x10^{-2}$	$7.0x10^{-2}$	$6.4x10^{-2}$
Deer	1,577 m ESE	$3.1x10^{-3}$	0.0	$1.2x10^{-2}$	$3.3x10^{-3}$	$3.1x10^{-3}$	$3.0x10^{-3}$	$2.9x10^{-3}$	$3.1x10^{-3}$
Rabbit	567 m NW	$1.2x10^{-4}$	0.0	$2.1x10^{-5}$	$1.4x10^{-4}$	$2.2x10^{-4}$	$1.2x10^{-4}$	$1.2x10^{-4}$	$1.3x10^{-4}$
Grouse	567 m NW	$9.9x10^{-6}$	0.0	$1.4x10^{-6}$	$1.1x10^{-5}$	$1.5x10^{-5}$	$9.7x10^{-6}$	$9.5x10^{-6}$	$1.0x10^{-5}$
Pheasant	567 m NW	$8.2x10^{-5}$	0.0	$1.2x10^{-5}$	$8.8x10^{-5}$	$1.2x10^{-4}$	$8.0x10^{-5}$	$7.8x10^{-5}$	$8.4x10^{-5}$
Total dose		$2.9x10^{-1}$	$6.8x10^{-2}$	$6.3x10^{-1}$	$3.5x10^{-1}$	$6.4x10^{-1}$	$2.9x10^{-1}$	$2.7x10^{-1}$	$2.7x10^{-1}$

TABLE 11A-11

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP
FROM GASEOUS EFFLUENTS

Residence with Goat - Location at 2,865 m ESE
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		$5.8x10^{-2}$	$6.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$	$5.8x10^{-2}$
Inhalation		$3.0x10^{-2}$	0.0	$1.4x10^{-2}$	$3.0x10^{-2}$	$5.5x10^{-2}$	$3.0x10^{-2}$	$3.3x10^{-2}$	$3.0x10^{-2}$
Fresh vegetation		$1.5x10^{-2}$	0.0	$5.6x10^{-2}$	$1.7x10^{-2}$	$1.1x10^{-1}$	$1.6x10^{-2}$	$1.5x10^{-2}$	$1.5x10^{-2}$
Stored vegetation		$2.2x10^{-1}$	0.0	$8.0x10^{-1}$	$2.4x10^{-1}$	$2.1x10^{-1}$	$2.2x10^{-1}$	$2.1x10^{-1}$	$2.1x10^{-1}$
Goat milk		$1.5x10^{-1}$	0.0	$4.9x10^{-1}$	$2.3x10^{-1}$	$6.9x10^{-1}$	$1.6x10^{-1}$	$1.4x10^{-1}$	$1.3x10^{-1}$
Deer	1,577 m ESE	$4.1x10^{-3}$	0.0	$1.7x10^{-2}$	$4.4x10^{-3}$	$4.2x10^{-3}$	$4.1x10^{-3}$	$4.0x10^{-3}$	$4.0x10^{-3}$
Rabbit	567 m NW	$1.4x10^{-4}$	0.0	$3.9x10^{-5}$	$1.6x10^{-4}$	$2.9x10^{-4}$	$1.5x10^{-4}$	$1.4x10^{-4}$	$1.4x10^{-4}$
Grouse	567 m NW	$1.2x10^{-5}$	0.0	$2.6x10^{-6}$	$1.3x10^{-5}$	$2.0x10^{-5}$	$1.2x10^{-5}$	$1.2x10^{-5}$	$1.2x10^{-5}$
Pheasant	567 m NW	$9.7x10^{-5}$	0.0	$2.1x10^{-5}$	$1.1x10^{-4}$	$1.6x10^{-4}$	$9.7x10^{-5}$	$9.4x10^{-5}$	$9.6x10^{-5}$
Total dose		$4.8x10^{-1}$	$6.8x10^{-2}$	1.4	$5.8x10^{-1}$	1.1	$4.9x10^{-1}$	$4.6x10^{-1}$	$4.5x10^{-1}$

TABLE 11A-12

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP
FROM GASEOUS EFFLUENTS

Residence With Goat - Location at 2,865 m ESE
Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	5.8×10^{-2}	6.8×10^{-2}	5.8×10^{-2}	5.8×10^{-2}	5.8×10^{-2}	5.8×10^{-2}	5.8×10^{-2}	5.8×10^{-2}
Inhalation	1.8×10^{-2}	0.0	9.9×10^{-3}	1.8×10^{-2}	4.0×10^{-2}	1.8×10^{-2}	2.0×10^{-2}	1.8×10^{-2}
Goat milk	2.6×10^{-1}	0.0	9.2×10^{-1}	4.3×10^{-1}	1.6	2.9×10^{-1}	2.6×10^{-1}	2.4×10^{-1}
Total Dose	3.4×10^{-1}	6.8×10^{-2}	9.9×10^{-1}	5.1×10^{-1}	1.7	3.7×10^{-1}	3.4×10^{-1}	3.2×10^{-1}

TABLE 11A-13

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP
FROM GASEOUS EFFLUENTS

Residence with Beef Animal - Location at 1,577 m ESE
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		1.5×10^{-1}	1.8×10^{-1}	1.5×10^{-1}	1.5×10^{-1}	1.5×10^{-1}	1.5×10^{-1}	1.5×10^{-1}	1.5×10^{-2}
Inhalation		7.9×10^{-2}	0.0	1.8×10^{-2}	8.0×10^{-2}	1.2×10^{-1}	7.9×10^{-2}	8.5×10^{-2}	7.9×10^{-2}
Fresh vegetation		3.2×10^{-2}	0.0	8.3×10^{-2}	3.4×10^{-2}	2.4×10^{-1}	3.0×10^{-2}	2.8×10^{-2}	3.0×10^{-2}
Stored vegetation		1.7×10^{-1}	0.0	4.4×10^{-1}	1.8×10^{-1}	1.5×10^{-1}	1.6×10^{-1}	1.5×10^{-1}	1.6×10^{-1}
Beef		5.6×10^{-2}	0.0	2.1×10^{-1}	5.6×10^{-2}	6.7×10^{-2}	5.4×10^{-2}	5.3×10^{-2}	5.6×10^{-2}
Deer	1,577 m ESE	2.4×10^{-3}	0.0	8.1×10^{-3}	2.5×10^{-3}	2.4×10^{-3}	2.3×10^{-3}	2.2×10^{-3}	2.4×10^{-3}
Rabbit	567 m NW	2.1×10^{-4}	0.0	2.6×10^{-5}	2.2×10^{-4}	3.3×10^{-4}	2.0×10^{-4}	1.9×10^{-4}	2.2×10^{-4}
Grouse	567 m NW	1.7×10^{-5}	0.0	1.7×10^{-6}	1.7×10^{-5}	2.4×10^{-5}	1.6×10^{-5}	1.6×10^{-5}	1.7×10^{-5}
Pheasant	567 m NW	1.4×10^{-4}	0.0	1.4×10^{-5}	1.4×10^{-4}	1.9×10^{-4}	1.3×10^{-4}	1.3×10^{-4}	1.4×10^{-4}
Total dose		4.9×10^{-1}	1.8×10^{-1}	9.1×10^{-1}	5.0×10^{-1}	7.3×10^{-1}	4.8×10^{-1}	4.7×10^{-1}	4.8×10^{-1}

TABLE 11A-14

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE TEEN GROUP
FROM GASEOUS EFFLUENTS

Residence with Beef Animal - Location at 1,577 m ESE
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		$1.5x10^{-1}$	$1.8x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$
Inhalation		$8.1x10^{-2}$	0.0	$2.5x10^{-2}$	$8.2x10^{-2}$	$1.3x10^{-2}$	$8.1x10^{-2}$	$9.0x10^{-2}$	$8.1x10^{-2}$
Fresh vegetation		$2.5x10^{-2}$	0.0	$7.7x10^{-2}$	$2.8x10^{-2}$	$2.0x10^{-1}$	$2.5x10^{-2}$	$2.3x10^{-2}$	$2.4x10^{-2}$
Stored vegetation		$2.6x10^{-1}$	0.0	$8.1x10^{-1}$	$2.9x10^{-1}$	$2.5x10^{-1}$	$2.5x10^{-1}$	$2.4x10^{-1}$	$2.5x10^{-1}$
Beef		$4.3x10^{-2}$	0.0	$1.7x10^{-1}$	$4.5x10^{-2}$	$5.2x10^{-2}$	$4.3x10^{-2}$	$4.2x10^{-2}$	$4.4x10^{-2}$
Deer	1,577 m ESE	$3.1x10^{-3}$	0.0	$1.2x10^{-2}$	$3.3x10^{-3}$	$3.1x10^{-3}$	$3.0x10^{-3}$	$2.9x10^{-3}$	$3.1x10^{-3}$
Rabbit	567 m NW	$1.2x10^{-4}$	0.0	$2.1x10^{-5}$	$1.4x10^{-4}$	$2.2x10^{-4}$	$1.2x10^{-4}$	$1.2x10^{-4}$	$1.3x10^{-4}$
Grouse	567 m NW	$9.9x10^{-6}$	0.0	$1.4x10^{-6}$	$1.1x10^{-5}$	$1.5x10^{-5}$	$9.7x10^{-6}$	$9.5x10^{-6}$	$1.0x10^{-5}$
Pheasant	567 m NW	$8.1x10^{-5}$	0.0	$1.2x10^{-5}$	$8.8x10^{-5}$	$1.2x10^{-4}$	$8.0x10^{-5}$	$7.8x10^{-5}$	$8.4x10^{-5}$
Total dose		$5.6x10^{-1}$	$1.8x10^{-1}$	1.2	$6.0x10^{-1}$	$7.9x10^{-1}$	$5.5x10^{-1}$	$5.5x10^{-1}$	$5.5x10^{-1}$

TABLE 11A-15

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP
FROM GASEOUS EFFLUENTS

Residence with Beef Animal - Location at 1,577 m ESE
Annual Dose (mRem/yr)

Pathway		Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground		$1.5x10^{-1}$	$1.8x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$
Inhalation		$7.4x10^{-2}$	0.0	$3.4x10^{-2}$	$7.5x10^{-2}$	$1.3x10^{-1}$	$7.4x10^{-2}$	$8.1x10^{-2}$	$7.3x10^{-2}$
Fresh vegetation		$3.8x10^{-2}$	0.0	$1.4x10^{-1}$	$4.4x10^{-2}$	$2.9x10^{-1}$	$3.9x10^{-2}$	$3.6x10^{-2}$	$3.6x10^{-2}$
Stored vegetation		$5.4x10^{-1}$	0.0	2.0	$6.1x10^{-1}$	$5.4x10^{-1}$	$5.5x10^{-1}$	$5.3x10^{-1}$	$5.2x10^{-1}$
Beef		$7.4x10^{-2}$	0.0	$3.3x10^{-1}$	$7.7x10^{-3}$	$8.9x10^{-2}$	$7.5x10^{-2}$	$7.4x10^{-2}$	$7.4x10^{-2}$
Deer	1,577 m ESE	$4.1x10^{-3}$	0.0	$1.7x10^{-2}$	$4.4x10^{-3}$	$4.2x10^{-3}$	$4.1x10^{-3}$	$4.0x10^{-3}$	$4.0x10^{-3}$
Rabbit	567 m NW	$1.4x10^{-4}$	0.0	$3.9x10^{-5}$	$1.6x10^{-4}$	$2.9x10^{-4}$	$1.5x10^{-4}$	$1.4x10^{-4}$	$1.4x10^{-4}$
Grouse	567 m NW	$1.2x10^{-5}$	0.0	$2.6x10^{-6}$	$1.3x10^{-5}$	$2.0x10^{-5}$	$1.2x10^{-5}$	$1.2x10^{-5}$	$1.2x10^{-5}$
Pheasant	567 m NW	$9.7x10^{-5}$	0.0	$2.1x10^{-5}$	$1.1x10^{-4}$	$1.6x10^{-4}$	$9.7x10^{-5}$	$9.4x10^{-5}$	$9.6x10^{-5}$
Total dose		$8.8x10^{-1}$	$1.8x10^{-1}$	2.7	$9.6x10^{-1}$	1.2	$8.9x10^{-1}$	$8.8x10^{-1}$	$8.6x10^{-1}$

TABLE 11A-16

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP
FROM GASEOUS EFFLUENTS

Residence With Beef Animal - Location 1,577 m ESE
Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	$1.5x10^{-1}$	$1.8x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$	$1.5x10^{-1}$
Inhalation	$4.4x10^{-2}$	0.0	$2.5x10^{-2}$	$4.4x10^{-2}$	$9.9x10^{-2}$	$4.4x10^{-2}$	$4.9x10^{-2}$	$4.3x10^{-2}$
Total Dose	$1.9x10^{-1}$	$1.8x10^{-1}$	$1.7x10^{-1}$	$1.9x10^{-1}$	$2.5x10^{-1}$	$1.9x10^{-1}$	$2.0x10^{-1}$	$1.9x10^{-1}$

TABLE 11A-17

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP
FROM LIQUID EFFLUENTS

Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Potable water	4.3×10^{-3}	0.0	3.5×10^{-4}	4.5×10^{-3}	2.0×10^{-2}	4.2×10^{-3}	3.9×10^{-3}	4.0×10^{-3}
Fish consumption	3.5	0.0	2.4	4.6	9.6×10^{-1}	1.5	5.1×10^{-1}	1.4×10^{-1}
Shoreline recreation	1.2×10^{-5}	1.4×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}
Fresh vegetation	6.8×10^{-4}	0.0	2.4×10^{-4}	7.9×10^{-4}	4.6×10^{-3}	5.1×10^{-4}	3.9×10^{-4}	4.2×10^{-4}
Stored vegetation	4.7×10^{-3}	0.0	1.6×10^{-3}	5.4×10^{-3}	2.6×10^{-3}	3.4×10^{-3}	2.7×10^{-3}	2.7×10^{-3}
Swimming exposure	4.7×10^{-4}	6.2×10^{-4}	4.7×10^{-4}	4.7×10^{-4}	4.7×10^{-4}	4.7×10^{-4}	4.7×10^{-4}	4.7×10^{-4}
Boating exposure	3.7×10^{-4}	4.9×10^{-4}	3.7×10^{-4}	3.7×10^{-4}	3.7×10^{-4}	3.7×10^{-4}	3.7×10^{-4}	3.7×10^{-4}
Total dose	3.5	1.1×10^{-3}	2.4	4.6	9.9×10^{-1}	1.5	5.2×10^{-1}	1.5×10^{-1}

TABLE 11A-18

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE TEEN GROUP
FROM LIQUID EFFLUENTS

Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Potable water	$3.0x10^{-3}$	0.0	$3.4x10^{-4}$	$3.3x10^{-3}$	$1.7x10^{-2}$	$3.1x10^{-3}$	$2.8x10^{-3}$	$2.8x10^{-3}$
Fish consumption	2.0	0.0	2.5	4.7	$8.9x10^{-1}$	1.6	$5.9x10^{-1}$	$9.5x10^{-2}$
Shoreline recreation	$6.8x10^{-5}$	$8.0x10^{-5}$	$6.8x10^{-5}$	$6.8x10^{-5}$	$6.8x10^{-5}$	$6.8x10^{-5}$	$6.8x10^{-5}$	$6.8x10^{-5}$
Fresh vegetation	$3.9x10^{-4}$	0.0	$2.2x10^{-4}$	$6.2x10^{-4}$	$3.6x10^{-3}$	$4.5x10^{-4}$	$2.7x10^{-4}$	$2.4x10^{-4}$
Stored vegetation	$5.3x10^{-3}$	0.0	$2.9x10^{-3}$	$8.3x10^{-3}$	$3.4x10^{-3}$	$5.4x10^{-3}$	$3.8x10^{-3}$	$3.3x10^{-3}$
Swimming exposure	$4.7x10^{-4}$	$6.2x10^{-4}$	$4.7x10^{-4}$	$4.7x10^{-4}$	$4.7x10^{-4}$	$4.7x10^{-4}$	$4.7x10^{-4}$	$4.7x10^{-4}$
Boating exposure	$3.7x10^{-4}$	$4.9x10^{-4}$	$3.7x10^{-4}$	$3.7x10^{-4}$	$3.7x10^{-4}$	$3.7x10^{-4}$	$3.7x10^{-4}$	$3.7x10^{-4}$
Total dose	2.0	$1.2x10^{-3}$	2.5	4.7	$9.1x10^{-1}$	1.6	$6.0x10^{-1}$	$1.0x10^{-1}$

TABLE 11A-19

BVPS NUCLEAR PLANT
ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP
FROM LIQUID EFFLUENTS

Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Potable water	5.5×10^{-3}	0.0	9.6×10^{-4}	6.4×10^{-3}	3.9×10^{-2}	5.8×10^{-3}	5.3×10^{-3}	5.2×10^{-3}
Fish consumption	8.0×10^{-1}	0.0	3.1	4.0	9.3×10^{-1}	1.3	4.6×10^{-1}	4.1×10^{-2}
Shoreline recreation	1.4×10^{-5}	1.7×10^{-5}	1.4×10^{-5}	1.4×10^{-5}	1.4×10^{-5}	1.4×10^{-5}	1.4×10^{-5}	1.4×10^{-5}
Fresh vegetation	3.7×10^{-4}	0.0	3.9×10^{-4}	7.6×10^{-4}	5.3×10^{-3}	4.4×10^{-4}	3.2×10^{-4}	2.7×10^{-4}
Stored vegetation	6.7×10^{-3}	0.0	6.8×10^{-3}	1.4×10^{-2}	5.6×10^{-3}	7.8×10^{-3}	6.0×10^{-3}	5.1×10^{-3}
Swimming exposure	2.6×10^{-4}	3.5×10^{-4}	2.6×10^{-4}	2.6×10^{-4}	2.6×10^{-4}	2.6×10^{-4}	2.6×10^{-4}	2.6×10^{-4}
Boating exposure	2.1×10^{-4}	2.7×10^{-4}	2.1×10^{-4}	2.1×10^{-4}	2.1×10^{-4}	2.1×10^{-4}	2.1×10^{-4}	2.1×10^{-4}
Total dose	8.1×10^{-1}	6.4×10^{-4}	3.1	4.0	9.8×10^{-1}	1.3	4.7×10^{-1}	5.2×10^{-2}

TABLE 11A-20

BVPS NUCLEAR PLANT
ANNUAL DOES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP
FROM LIQUID EFFLUENTS

Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Potable water	$5.3x10^{-3}$	0.0	$1.0x10^{-3}$	$6.6x10^{-3}$	$5.9x10^{-2}$	$5.7x10^{-3}$	$5.3x10^{-3}$	$5.1x10^{-3}$
Total dose	$5.3x10^{-3}$	0.0	$1.0x10^{-3}$	$6.6x10^{-3}$	$5.9x10^{-2}$	$5.7x10^{-3}$	$5.3x10^{-3}$	$5.1x10^{-3}$

TABLE 11A-21

COMPARISON OF MAXIMUM CALCULATED DOSES FROM BEAVER VALLEY POWER STATION WITH RM-50-2 DESIGN OBJECTIVES

Criterion	RM-50-2 Design Objective[1,2]	Calculated Dose
Gaseous Effluents		
Gamma air dose	10 mrad/year	6.3 mrad/yr[3]
Beta air dose	20 mrad/year	6.8 mrad/yr[3]
Noble gas, total body	5 mRem/year	1.1 mRem/yr[4]
Noble gas, skin	15 mRem/year	2.4 mRem/yr[4]
Iodines and particulates, any organ	15 mRem/year	7.7 mRem/yr[5]
Liquid Effluents		
Total body	5 mRem/year	3.5 mRem/yr[6]
Any organ	5 mRem/year	4.7 mRem/yr[7]

NOTES:

1. U.S. Atomic Energy Commission 1975.
2. Per site.
3. Northern shore of the Ohio River, 567 m northwest.
4. Residence location with the highest calculated χ/Q, 1,432 m northwest.
5. Child thyroid dose, residence location, 1,432 m northwest.
6. Dose to adult from consumption of fish.
7. Dose to teen liver from consumption of fish.

TABLE 11A-22

CALCULATED POPULATION DOSE COMMITMENT

50-Mile Population Dose

	Total Body (man-Rem)	Thyroid (man-Rem)
Natural radiation background*	4.9×10^{5}	4.9×10^{5}
Annual dose per site		
Liquid effluents	3.4×10^{-1}	1.5
Noble gas effluents	9.4×10^{-1}	9.4×10^{-1}
Radioiodines and particulates**	1.4×10^{1}	2.1×10^{1}

Contiguous U.S. Population Dose

	Total Body (man-Rem)	Thyroid (man-Rem)
Annual dose per site		
Liquid effluents	3.5×10^{-1}	1.5
Noble gas effluents	9.9×10^{-1}	1.7
Radioiodines and particulates**	5.4×10^{1}	6.1×10^{1}

NOTES:

*Estimates of Ionizing Radiation Doses in the United States, U.S. Environmental Protection Agency 1972, ORP-CSD-72-1, Using the average state background dose rate (125 mRem/yr), and year 2010 projected population of 3,949,000.

**Carbon-14 and tritium have been added to this category.