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CHAPTER 11
RADIOACTIVE WASTE MANAGEMENT

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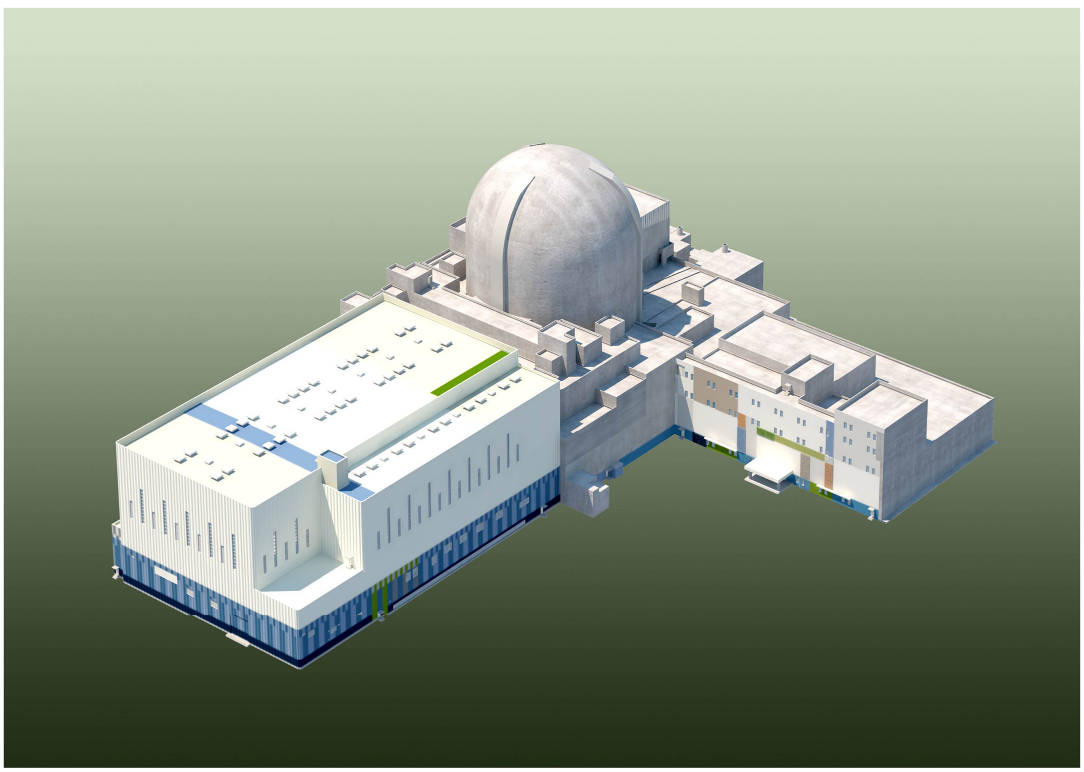

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CHAPTER 11 – RADIOACTIVE WASTE MANAGEMENT

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ACRONYM AND ABBREVIATION LIST

AB	auxiliary building
ACU	air cleaning unit
ADV	atmospheric dump valve
AHU	air handling unit
ALARA	as low as is reasonably achievable
ALI	annual limit on intake
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
API	american petroleum institute
APR	Advanced Power Reactor
ARMS	area radiation monitoring system
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing And Materials
BAC	boric acid concentrator
BTP	branch technical position
CCW	component cooling water
CCWS	component cooling water system
CEA	control element assembly
CFR	Code of Federal Regulations
COL	combined license
COLA	combined license application
CREVAS	control room emergency ventilation actuation signal
CVCS	chemical and volume control system
CWT	chemical waste tank
DAC	derived air concentration
DAW	dry active waste
DC	design certification
DCD	Design Control Document

DF	decontamination factor
DOT	U.S. Department of Transportation
EAB	exclusion area boundary
EDT	equipment drain tank
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
ESW	essential service water
EWT	equipment waste tank
FDS	floor drain system
FDT	floor drain tank
GDC	general design criteria (of 10 CFR 50, Appendix A)
GI	gastrointestinal
GRS	gaseous radwaste system
GWMS	gaseous waste management system
FSAR	Final Safety Analysis Report
HEPA	high-efficiency particulate air
HIC	high-integrity container
HPS	Health Physics Society
HVAC	heating, ventilation, and air conditioning
HX	heat exchanger
IE	Inspection and Enforcement
IPS	information processing system
IRSF	interim radwaste storage facility
IRWST	in-containment refueling water storage tank
ISG	Interim Staff Guidance
IX	ion exchange
LASRT	low-activity spent resin tank
LOCA	loss-of-coolant accident
LPZ	low-population zone
LRS	liquid radwaste system

LWMS	liquid waste management system
LWR	light water reactor
MCR	main control room
MF	membrane filter
NEI	Nuclear Energy Institute
NNS	non-nuclear safety
NRC	United States Nuclear Regulatory Commission
NSSS	nuclear steam supply system
NUREG	NRC technical report designation
OBE	operating basis earthquake
ODCM	offsite dose calculation manual
P&ID	piping and instrumentation diagram
PCA	primary coolant activity
PERMSS	process and effluent radiation monitoring and sampling systems
PTS	primary-to-secondary
PWR	pressurized water reactor
QA	quality assurance
QAP	quality assurance procedure
QIAS	qualified indication and alarm system
RCA	radiologically controlled area
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RDT	reactor drain tank
RG	Regulatory Guide
RMS	radiation monitoring system
R/O	reverse osmosis
SC	shutdown cooling
SFP	spent fuel pool
SFPCCS	spent fuel pool cooling and cleanup system
SG	steam generator

SGBDS	steam generator blowdown system
SGTR	steam generator tube rupture
SRLST	spent resin long-term storage tank
SRP	Standard Review Plan
SRS	solid radwaste system
SRST	spent resin storage tank
SSC	structure, system, or component
SWMS	solid waste management system
TEDE	total effective dose equivalent
TEMA	Tubular Exchanger Manufacturers Association
TID	total integrated dose
TMI	Three Mile Island
VCT	volume control tank
WCT	waste collection tank

CHAPTER 11 – RADIOACTIVE WASTE MANAGEMENT

11.1 Source Terms

This section presents information on the sources of radioactivity that serve as design bases for the various radioactive waste treatment systems for normal operation, including anticipated operational occurrences (AOOs) (expected source term), as well as for design basis conditions (design basis source term). The application of the source terms and the mathematical models and parameters used to calculate source terms for normal operation and for design basis conditions are different. A clear distinction is made between the design basis source term and the expected source term. The design basis source term used for the radiation shielding design is addressed in Section 12.2, and the accident source term used for the radiological consequence analysis is addressed in Chapter 15.

Definitions

a. Design basis source term

The design basis source term is used for the design of the radioactive waste management system and for determining design lifetime integrated doses for the design specifications of plant equipment. The design basis source term is based on design basis data used for calculating the maximum reactor coolant activity as shown in Table 11.1-1.

b. Expected source term

The expected or operating basis source term is used for describing annual releases from the plant to the environment on an average basis. Site boundary doses due to releases from the plant ventilation exhausts, liquid discharges, and offsite shipment of solid radioactive material are examples of calculations that use this source term. The expected source term is based on a realistic model as described in ANSI/ANS 18.1 (Reference 1) for reactor coolant activity during normal operation as represented in Table 11.1-1. Calculations pertaining to releases described in 10 CFR 50, Appendix I (Reference 2), conform with the methods and parameters described in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 1.112 (Reference 3).

11.1.1 Design Basis Source Term

11.1.1.1 Fission Product Activities in the Reactor Coolant

The DAMSAM Code (Reference 4) is used to calculate the design basis source term of fission products in the reactor coolant for the design of the radioactive waste management system and to determine design lifetime integrated doses for plant equipment. The isotopes considered in the maximum case are those that are significant for design purposes by reason of a combination of energy, half-life, and/or abundance.

The mathematical model used to determine the concentration of nuclides in the reactor coolant system (RCS) involves a group of linear, first-order differential equations. These equations are obtained by applying a mass balance for production and removal in both the fuel pellet region and the reactor coolant region.

In the fuel pellet region, the mass balance includes fission product production by direct fission yield, by parent fission product decay, and by neutron activation, while removal includes decay, neutron activation, and escape to the reactor coolant.

In the reactor coolant region, the fission product is introduced when it escapes from the fuel pellet through defective fuel rod cladding, parent decay in the reactor coolant, and neutron activation of the fission products in the reactor coolant. The fission products are removed by decay; coolant purification; boron feed-and-bleed operations (to accommodate fuel burnup); leakage and other feed-and-bleed operations during startups, shutdowns, and load-following operation; and neutron activation.

The expression to determine the fission product inventory in the fuel pellet region is:

$$\frac{dN_{c,i}}{dt} = (F)(Y_i)(P) + (f_{i-1}\lambda_{i-1})N_{p,i-1} + \sigma_j\phi N_{p,j} - (\lambda_i + Dv_i + \sigma_i\phi)N_{p,i} \qquad \text{(Eq. 11.1-1)}$$

The expression to determine the fission product inventory in the reactor coolant region is:

$$\frac{dN_{c,i}}{dt} = (D)(v_i)(N_{p,i}) + (f_{i-1}\lambda_{i-1})N_{c,i-1} + (\sigma_j\phi CVR)\,N_{c,j}$$
$$-(\dot{Q}\lambda_i + \frac{\dot{Q}}{w}\eta_i + \frac{(1-\eta_i)\dot{C}}{C_0 - t\dot{C}} + \frac{L}{W} + \sigma_i\phi CVR)\,N_{c,i} \qquad \text{(Eq. 11.1-2)}$$

Where the variables are defined as:

N	=	nuclide population, atoms
F	=	average fission rate, fissions/MWt-sec
Y	=	core averaged fission yield of nuclide, fraction
P	=	core power, MWt
λ	=	decay constant, sec^{-1}
σ	=	microscopic capture cross section, cm^2
ϕ	=	thermal neutron flux, neutrons/cm^2-sec
v	=	escape rate coefficient, sec^{-1}
f	=	branching fraction
t	=	time, seconds
D	=	defective fuel cladding, fraction
CVR	=	ratio of core coolant volume to reactor coolant volume
$\dot{Q}$	=	chemical and volume control system (CVCS) purification mass flow rate during power operation, kg/sec
W	=	RCS mass during power operation, kg
η	=	resin efficiency of CVCS ion exchanger and gas stripper efficiency
C_0	=	boron concentration at the beginning of core life, ppm
$\dot{C}$	=	boron concentration reduction rate due to feed and bleed, ppm/sec
L	=	mass flow rate of reactor coolant leakage or the other feed and bleed, kg/sec

Where the subscripts are identified as:

i	=	i^{th} nuclide
i-1	=	precursor to i^{th} nuclide for decay
j	=	j^{th} nuclide to i^{th} nuclide for neutron activation
p	=	pellet region
c	=	reactor coolant region

This model does not involve the fuel plenum and gap region. Instead, escape rate coefficients are used to represent the overall release from the fuel pellets to the reactor coolant. The escape rate coefficient is an empirical value derived from experiments involving the Nuclear Research Reactor and Material Testing Reactor that were initiated at the Bettis Plant (Reference 5). The escape rate coefficients were obtained from test rods that were operated at high linear heat rates.

The linear heat rates were uniform over test sections of 26.035 cm (10.25 in) in length. The exact linear heat rates were not known, but post-irradiation inspection showed that some test specimens had exhibited centerline melting. Supplemental tests were conducted in Canada to determine the effect of rod length on the release of fission gases and iodines from the defective fuel rods (Reference 6). The experiments also determined the relationship between linear heat rate and the escape rate coefficient. Because the average heat rate for a fuel rod is below the linear heat rate of 591 W/cm (18 kW/ft), which corresponds to the selected escape rate coefficients for halogens and noble gases shown in Table 11.1-1, the current escape rate coefficients are conservative.

Table 11.1-1 shows the values of the parameters that are used to calculate the reactor coolant fission product source term. The maximum reactor coolant fission product source term used for the design is presented in Table 11.1-2.

The design basis reactor coolant activities are based on 1 percent fuel cladding defects. The total activities of iodine and noble gases are 3.6 μCi/g (I-131 dose equivalent) and 580 μCi/g (Xe-133 dose equivalent), respectively. The activities of iodine and noble gases are limited to 1.0 μCi/g (I-131 dose equivalent) and 300 μCi/g (Xe-133 dose equivalent) by the plant Technical Specifications. The reactor coolant activities that are limited by the Technical Specifications during normal power operation are lower than the design values.

11.1.1.2 <u>Spent Fuel Pool and Refueling Pool Activities</u>

Table 11.1-3 contains the assumptions used in calculating the design basis specific activities of the fission and corrosion products in the spent fuel pool for the start of the refueling period, and Table 11.1-4 contains the results of the calculations. The RCS is assumed to cool down for 2 days upon shutdown for refueling. During this period, the primary coolant is let down through the purification filter, purification ion exchanger, gas stripper, and volume control tank. The letdown serves two purposes: (1) removing the noble gases in the gas stripper prevents large activity releases from the refueling pool to the reactor containment building following reactor vessel head removal, and (2) reducing, through ion exchange and filtration, the dissolved fission and corrosion products in the reactor coolant that would otherwise enter the spent fuel pool and refueling pool.

At the end of this period, the reactor coolant above the reactor vessel flange is partially drained. The reactor vessel head is unbolted, and the refueling pool is filled with water from the in-containment refueling water storage tank (IRWST). The remaining reactor coolant containing radioactivity is then mixed with water in the refueling pool and spent fuel pool. Refueling pool water is cooled by the shutdown cooling system and cleaned of radioactivity by the spent fuel pool cooling and cleanup systems. The spent fuel pool water is cooled and cleaned of radioactivity by the spent fuel pool cooling and cleanup systems.

After refueling, the spent fuel pool is isolated, and the water in the refueling pool is returned to the IRWST. The total activity in the spent fuel pool is determined through these serial processes.

Leakage of radionuclides into the spent fuel pool from damaged fuel stored in the pool is not considered a significant contributor to the radionuclide concentration in the spent fuel pool water because of the extremely low escape rate coefficients of the spent fuel in the spent fuel pool. The low escape rate coefficients are due in part to the low spent fuel pool temperature. Most of the activity are released from the defective fuel elements during shutdown and cooldown of the reactor prior to removal of the reactor vessel head. If significant releases from the defective fuel are detected, the defective fuel elements are isolated in a separate container so the released activity does not contribute to the specific activity in the spent fuel pool water. The primary source of radioactivity in the spent fuel pool water, after refueling operations have been completed, is due to displaced activation products, or crud, from the surfaces of the spent fuel assemblies.

11.1.1.3 Secondary System Activity

For the purpose of the design, the steam generator (SG) is assumed to have tube leaks, and radionuclides are moved to the secondary system from the primary system. In determining design basis sources in the secondary system, a total SG tube leakage is assumed to be 3,270 L/day (0.6 gal/min), which is the accident-induced SG leakage criterion (Chapter 16, Subsection 5.5.9.b.2). This value is conservatively used even though the limiting condition of operation for SG operational leakage in the Technical Specifications is 562 L/day (0.1 gal/min) (Chapter 16, Subsection 3.4.12).

Radionuclides are removed from the secondary system by the following mechanisms:

a. Steam generator blowdown demineralizer treatment

b. Condensate polishing demineralizer treatment

c. Radioactive decay

d. Exhaust through the main condenser vacuum pumps

e. Main steam leakage

Primary coolant activities used to determine the design basis sources in the secondary system are addressed in Subsection 11.1.1.1. Assumptions used in determining the secondary system activities are listed in Table 11.1-5. Design basis equilibrium radionuclide concentrations in the secondary system are determined as described below:

a. Steam generator liquid activity

The following expression determines the concentration of nuclides in the SG liquid (i.e., blowdown source):

$$M_{sl}\frac{dN_{sl}}{dt} = RN_w - TN_{sg} - BN_{sl} - \lambda N_{sl}M_{sl} + TN_{sg}(1-F)$$

$$+TFN_{sg}\left[\frac{0.8333}{DF_d} + \frac{0.1667}{DF_dDF_c}\right] + BN_{sl}\frac{1}{DF_b}\left[0.8333 + \frac{0.1667}{DF_c}\right] \qquad \text{(Eq. 11.1-3)}$$

Where the variables are defined as:

N = nuclide concentration, Bq/g

R = primary-to-secondary leakage rate, g/sec

T = main steam flow rate, g/sec

F = fraction of radionuclide in the main steam reaching to the main condenser

M = secondary liquid mass in SG, g

B = SG blowdown rate, g/sec

DF = decontamination factor

λ = decay constant, sec^{-1}

0.1667 = fraction of condensate water processed in condensate polishing system (0.8333 means fraction of bypassing the condensate polishing system)

Where the subscripts are defined as:

s = SG

w = RCS

c = condensate polishing system

d = condenser vacuum system

b = SG blowdown system

l = liquid in the secondary system

g = steam in the secondary system

Also, $N_{sg} = \alpha N_{sl}$

Where:

α = SG partition coefficient (the ratio of concentration-in-steam to concentration-in-liquid)

Therefore, the equilibrium concentration of nuclides in the SG liquid is given by:

$$N_{sl} = \frac{RN_W}{\lambda M_{sl} + \alpha TF\left\{1-\left(\frac{0.8333}{DF_d}+\frac{0.1667}{DF_d DF_c}\right)\right\} + B\left\{1-\frac{1}{DF_b}\left(0.8333+\frac{0.1667}{DF_c}\right)\right\}} \qquad \text{(Eq. 11.1-4)}$$

Parameter definition and units are shown in Eq. 11.1-3.

The design basis radionuclide concentrations in the SG liquid are listed in Table 11.1-6.

a. Main steam activity

To obtain the peak noble gas concentration in the main steam leaving the SGs, all of the noble gases that enter the SG via primary coolant leakage are assumed to exit with the steam (i.e., no noble gases are in the SG secondary liquid). The noble gas activities in the steam exiting the SG are the ratio of the primary-to-secondary (PTS) leakage rate multiplied by the radionuclide concentration in the primary coolant to the steam flow rate out of the SG.

This is expressed as:

$$N_{sg} = \frac{R \cdot N_W}{T} \qquad \text{(Eq. 11.1-5)}$$

The parameter definition and units are shown in Eq. 11.1-3.

The equilibrium concentration of non-noble gas radionuclides in the steam exiting the SG is the equilibrium concentration of the radionuclides in the SG liquid multiplied by the SG partition coefficient, which is the ratio of concentration in the SG steam to the concentration in the SG water.

This is expressed as:

$$N_{sg} = \alpha \cdot N_{sl} \qquad \text{(Eq. 11.1-6)}$$

Parameter definition and units are shown in Eq. 11.1-3.

The design basis radionuclide concentrations in the main steam are determined using Equations 11.1-5 and 11.1-6 and are listed in Table 11.1-6.

b. High-capacity blowdown liquid activity

Approximately once a week, a high-capacity blowdown is performed to remove accumulated crud in the SG. The high-capacity blowdown is performed for 2 minutes per week. For the two SGs, the high-capacity blowdown rate is 5 percent of the nuclear steam supply system (NSSS) maximum steaming rate (113 kg/sec [249.0 lb/sec]) based on cold leg temperature, and the minimum value is 3.6 percent (81.2 kg/sec [179.2 lb/sec]) based on hot leg temperature. For conservatism, the flow rate of 81.2 kg/sec (179.2 lb/sec) is used to calculate radionuclide crud concentration in the high-capacity blowdown liquid. To obtain the radionuclide crud activity in the high-capacity blowdown liquid, it is assumed that the crud radionuclides (i.e., Mn, Co, Fe, Cr, Zr) due to PTS coolant leakage remain in the SGs between high-capacity blowdown operations. The accumulated radionuclide crud is diluted with 9.78×10^3 kg (2.15×10^4 lb) of the high-capacity blowdown water and discharged to the high-capacity blowdown flash tank. The radionuclide crud concentrations in the high-capacity blowdown liquid are calculated as follows:

$$N_h = \frac{R \cdot N_W}{\lambda \cdot M_h}\left\{1 - e^{-\lambda t}\right\} \quad \text{(Eq. 11.1-7)}$$

Where:

N_h = radionuclide concentration of crud within the high-capacity blowdown water, Bq/g

t = period of high-capacity blowdown, sec

λ = decay constant, sec^{-1}

M_h = mass of high-capacity blowdown water, g

R and N_w are described in the variable identifications for Eq. 11.1-3.

The remaining radionuclide concentrations in the high-capacity blowdown liquid (i.e., non-noble gas and non-crud radionuclides) are determined using Eq. 11.1-4, which determines the equilibrium radionuclide concentration in the SG liquid.

Radionuclide crud concentrations in the high-capacity blowdown liquid are listed in Table 11.1-7.

11.1.1.4 Radwaste System Activities

Source terms for the liquid waste management system (LWMS) are described in Section 11.2. Source terms for the gaseous radwaste system (GRS) are described in Section 11.3.

11.1.1.5 Volume Control Tank Activity

The total activity inventory in the volume control tank (VCT) is based on an expected maximum water volume, 15,725 L (4,154 gal), of reactor coolant letdown and an expected maximum vapor volume of 17,500 L (618 ft^3). The design basis specific activities of gaseous sources vented from the VCT to the gaseous radwaste system (GRS) are provided in Table 11.1-8.

11.1.1.6 Reactor Drain Tank Activity

The total activity inventory in the reactor drain tank (RDT) is based on an expected maximum water volume of 11,962 L (3,160 gal) and an expected maximum vapor volume of 9,085 L (321 ft^3). The design basis specific activities of gaseous sources vented from the RDT to the GRS are provided in Table 11.1-8.

11.1.1.7 Gas Stripper Activity

The total activity inventory in the gas stripper is based on the summation of activity in the aftercooler, the heat recovery exchanger, the overhead condenser, the reboiler, the stripper column, and the feed preheater within the gas stripper package. The design basis specific activities of gaseous sources vented from the gas stripper to the GRS are provided in Table 11.1-8.

11.1.1.8 Equipment Drain Tank Activity

The total activity inventory in the equipment drain tank (EDT) is based on an expected maximum water volume of 13,306 L (3,515 gal) and an expected maximum vapor volume of 33,085 L (1,168 ft^3). The design basis specific activities of gaseous sources vented from the EDT to the GRS are provided in Table 11.1-8.

11.1.2 Expected Source Term

11.1.2.1 Reactor Coolant Activities

The data in Table 11.1-9 represent the expected normal fission and corrosion product specific activities in the reactor coolant with no gas stripping. The data are used in evaluating only normal operations including AOOs. The expected specific activities in the reactor coolant are based on ANSI/ANS 18.1 using the normal operating parameters provided in Table 11.1-1.

11.1.2.2 Spent Fuel Pool and Refueling Pool Activities

The model used to determine the spent fuel pool and refueling pool radionuclide activities is described in Subsection 11.1.1.2. The model used to predict expected activities is the same as the analysis model of the design basis source term except that the expected source terms in the primary coolant are used. The expected specific activities for the spent fuel pool and refueling pool are shown in Table 11.1-4.

11.1.2.3 Secondary System Activities

The equilibrium radionuclide concentrations in the SG liquid and in the main steam during the normal operation are determined using the method described in Subsection 11.1.1.3. The SG tube leak rate from the primary to the secondary system is assumed to be 34 kg/day (75 lb/day), based on ANSI/ANS 18.1. Additional assumptions used to determine the secondary activity are provided in Table 11.1-5.

The expected specific activities for the secondary system are provided in Table 11.1-10.

11.1.3 Neutron Activation Products

11.1.3.1 Deposited Crud Activities

Deposited crud activities on primary system surfaces have been evaluated using measured data from various operating pressurized water reactors (PWRs). Even though these reactors have different water chemistries and different materials in contact with the primary coolant, their crud activities (Bq/g-crud), crud film thicknesses, and dose rates are remarkably similar. The half-lives, reactions, and gamma decay energies for each of the long-lived isotopes in the radioactive crud are as provided in Table 11.1-11.

The radioactive crud originates from in-core and out-of-core surfaces. The radioactive crud deposits on the in-core surfaces and erodes after a short irradiation period. This irradiation period or core residence time (t_{res}) for each isotope is determined by the following equations. See Appendix 11A for the derivation of these equations.

Circulating crud:

$$t_{res} = \frac{1}{\lambda_i} \ln\left(1 - \frac{A_i A_T}{\Sigma_i \phi A_c}\right) \qquad \text{(Eq. 11.1-8)}$$

Deposited crud:

$$t_{res} = \frac{1}{\lambda_j} \ln\left(1 - \frac{A_j}{\Sigma_j \phi}\right) \qquad \text{(Eq. 11.1-9)}$$

Where:

A_i, A_j = crud activities for each isotope, Bq/g-crud

$\Sigma_i\phi$, $\Sigma_j\phi$ = activation rate for each isotope, reaction/g-sec

A_T = total primary system area, cm^2

A_C = core surface area, cm^2

λ_i, λ_j = decay constant for each isotope, sec^{-1}

t_{res} = core residence time, sec

The activation cross section (Σ_i) is:

$$\Sigma_i = \frac{(a/o)_i (w/o)_i N_o \sigma_i}{\{A\}_i} \qquad \text{(Eq. 11.1-10)}$$

Where:

$(a/o)_i$ = isotopic abundance, fraction

$(w/o)_i$ = elemental abundance in the crud or the elemental abundance in the base metal, fraction

N_o = Avogadro's number, 6.023×10^{23} atoms/g-mole

$\{A\}_i$ = atomic weight of isotope (i)

σ_i = microscopic cross section, cm^2

Σ_i = activation cross section, cm^2/g

The core residence times are determined by applying the measured average and maximum crud activities (Bq/g-crud) from various operating reactors, system parameters, and activation rates to the above expressions. The core residence times are shown in Table 11.1-12. The crud activities (A_i) are determined by applying the averages (t_{res}) of the maximum core residence times in Table 11.1-12, the system parameters, and the activation rates to the following equation. Because all of the Fe-59 residence times are long, the activity (A_j) is assumed to be saturated.

$$A_i = \Sigma_i \phi \left(1 - e^{-\lambda_i t_{res}}\right) \frac{A_C}{A_T} \qquad \text{(Eq. 11.1-11)}$$

Where:

A_i = crud activities, Bq/g-crud

As the averages (t_{res}) of the maximum core residence times are, in general, a factor of 2 to 4 greater than a straight average residence time, the resulting calculated crud activities are conservative. These calculated crud activities of the long-lived isotopes are as shown in Table 11.1-13. These calculated crud activities are applied to both the circulating crud and out-of-core deposited crud.

Applying the average crud level (0.075 ppm) in the reactor coolant of various operating reactors to the calculated crud activities (Bq/g-crud) in Table 11.1-13, the crud specific activities in the reactor coolant, as shown in Table 11.1-14, are determined. The partial crud activities in the reactor coolant from the above conservative evaluation can be less than the expected crud activities (Table 11.1-9) in the reactor coolant during normal operations. In this case, the expected crud activities during normal operations are used as design basis maximum activities. These circulating crud activities in the reactor coolant are listed in Table 11.1-2.

The maximum coolant activities can be greater due to “crud bursts” during shutdown or changes in reactor power level. However, these “bursts” occur over short periods, and the average values are therefore more reasonable for use during long-term operation.

11.1.3.2 Carbon-14 Production

Carbon-14 is produced by neutron activation of O^{17} and N^{14} isotopes in the RCS. The greatest amount of C-14 is produced by the O^{17} (n, α) C^{14}reaction, and a lower amount of C-14 is produced by the N^{14}(n, p)C^{14} reaction. The production rate of C-14 (Q, Bq/cycle) from both reactions can be calculated by using the following equation:

$$Q = \lambda\ t\ m\ N\ (\sigma_{th}\phi_{th} + \sigma_f\ \phi_f) \qquad \text{(Eq. 11.1-12)}$$

Where:

λ	=	decay constant, 3.84×10^{-12} sec^{-1}
t	=	reactor operating time, 4.15×10^{7} sec
m	=	mass of active core water, 1.64×10^{7} g
N	=	atom concentration in the RCS water $\{N(O^{17}) = 1.27 \times 10^{19}$ atoms/g H_2O, $N(N^{14})=1.31 \times 10^{17}$ atoms/g $H_2O\}$
σ_{th}	=	microscopic effective thermal cross section, cm^2 $\{\sigma_{th}\ (O^{17}) = 1.21 \times 10^{-25}, \sigma_{th}\ (N^{14})=9.51 \times 10^{-25}\}$
σ_f	=	microscopic effective fast cross section, cm^2 $\{\sigma_f\ (O^{17}) = 4.79 \times 10^{-26}, \sigma_f\ (N^{14})=3.92\times 10^{-26}\}$
ϕ_{th}	=	thermal neutron flux, 6.32×10^{13} n/cm^2-sec
ϕ_f	=	fast neutron flux, 3.06×10^{14} n/cm^2-sec

The production rate of the C-14 by $O^{17}(n, \alpha)C^{14}$ reaction is 7.4×10^{11} Bq/cycle. The production rate of the C-14 by $N^{14}(n, p)C^{14}$ reaction is 3.0×10^{10} Bq/cycle. The production rate of C-14 from these sources during reactor operation is 7.7×10^{11} Bq/cycle.

11.1.3.3 Argon-41 Production and Releases

Argon-41 (Ar-41) is formed in the reactor containment building air by neutron activation of naturally occurring Ar-40 in the air surrounding the reactor vessel and could be produced within the reactor coolant by the Ar-40 dissolved in the primary coolant. Ar-41 is released to the environment via the reactor containment building vent when the reactor containment building is vented or purged. The annual release amount of Ar-41 from a PWR is assumed to be 34 Ci/yr (Reference 7).

11.1.3.4 Nitrogen-16 Production

Nitrogen-16 (N-16) is produced by the neutron reaction with oxygen-16. N-16 is not a significant radiation source outside the reactor containment building due to its short half-life (7.13 seconds). N-16 activities for the shielding design inside the reactor containment building are provided in Subsection 12.2.1.1.2.

11.1.4 Tritium Production in Reactor Coolant

The principal sources of tritium production in a PWR are from ternary fission and neutron-induced reactions in boron, lithium, and deuterium that are present in the reactor coolant and control element assemblies (CEAs). The tritium produced in the reactor coolant contributes immediately to the overall tritium concentration, while the tritium produced by fission and neutron capture in the CEAs contributes to the overall tritium concentration via release through the fuel cladding.

11.1.4.1 Activation Sources of Tritium

The activation reactions producing tritium are shown in Table 11.1-15. The tritium production from B-11 and N-14 sources is insignificant due to low reaction cross section and abundance and can be neglected. The activation reactions from B-10, lithium, and deuterium are the major sources of tritium in the reactor coolant and CEAs.

The tritium production from the above sources is determined by the following expressions:

$$\frac{dN}{dt} = \Sigma_a \phi - \lambda N \qquad \text{(Eq. 11.1-13)}$$

$$N = \frac{\Sigma_a \phi}{\lambda}\left(1 - e^{-\lambda t}\right)$$

Activity (Bq) = $V\lambda N = \Sigma_a\phi\,(1 - e^{-\lambda t})\,V$

Where:

N	=	tritium concentration, atoms/cm^3
$\Sigma_a\phi$	=	production rate, atoms/cm^3-sec
λ	=	decay constant, sec^{-1}
t	=	reactor operating period of interest, sec
V	=	effective core volume or CEA volume, cm^3

The parameters used in the calculation are shown in Table 11.1-16. Based on these parameters, the tritium produced from activation sources in the reactor coolant is provided in Table 11.1-17.

11.1.4.2 Tritium from Fission

The ternary fission production of tritium in the core is calculated using the ORIGEN-S Computer Code (Reference 8). Tritium as a product of fission is released to the reactor coolant through the fuel cladding. One percent of an average expected tritium release from the fuel and 2 percent of a maximum design value are used to estimate the tritium production in the reactor coolant. Tritium production is shown in Table 11.1-17.

11.1.4.3 Tritium Concentrations in the Secondary System

In determining the tritium activity concentrations in the secondary system, it is assumed that tritium that enters the secondary system from the primary system via SG tube leakage is uniformly mixed in the secondary system steam and liquid masses. In the equilibrium condition, the decay and leakage losses of tritium from the secondary system are equal to the primary-to-secondary system tritium leakage.

The tritium activity concentrations in the secondary system are calculated using the following equation:

$$N_s = \frac{R}{L_s} N_w \qquad \text{(Eq. 11.1-14)}$$

Where:

N_w = tritium activity concentrations in the primary system, Bq/g

R = primary-to-secondary leak rate, g/sec

L_s = steam leak rate, g/sec

N_s = tritium activity concentrations in the secondary system, Bq/g

11.1.5 Leakage Sources

Systems containing radioactive liquids and gases are potential sources of leakage and discharge to the environment. Liquid leakage is from potential sources such as pump seals and valve packings. Expected leakage of primary coolant into the reactor containment building is at a rate that would result in the release of 3 percent per day of the primary coolant noble gas inventory and 8.0×10^{-4} percent per day of the primary coolant iodine inventory. The expected primary coolant leak rate into the auxiliary building is 72.6 kg/day (160 lb/day), and the expected leak rate of steam into the turbine generator building is 771 kg/hr (1,700 lb/hr). The expected primary-to-secondary leakage rate across the SG tubes is 34 kg/day (75 lb/day). Table 11.1-18 provides maximum anticipated leak rates from NSSS-related valves and pumps.

Liquid radioactive releases are further addressed in Section 11.2. Gaseous radioactive effluents to the environment are further addressed in Section 11.3. Concentrations of airborne radioactive nuclides in cubicles are addressed in Subsection 12.2.2.3.

11.1.6 Combined License Information

No combined license (COL) information is required with regard to Section 11.1.

11.1.7 References

1. ANSI/ANS 18.1, "Radioactive Source Term for Normal Operation of Light Water Reactors," American Nuclear Society, 1999.

2. 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," U.S. Nuclear Regulatory Commission.

3. Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," Rev. 1, U.S. Nuclear Regulatory Commission, March 2007.

4. P. D. Maloney, "DAMSAM: A Digital Computer Program to Calculate Primary and Secondary Activity Transients," Combustion Engineering, Inc., 1972.

5. J. D. Eichenberg et al., WAPD-183, "Effects of Irradiation on Bulk UO_2," Bettis Plant, October 1957.

6. G. M. Allison and H. K. Rae, "The Release of Fission Gases and Iodines from Defected UO_2 Fuel Elements of Different Lengths," AECL-2206, June 1965.

7. NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," Rev.1, U.S. Nuclear Regulatory Commission, 1985.

8. NUREG/CR-0200, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," Rev. 6, RSICC, Oak Ridge National Laboratory, 1998.

Table 11.1-1

Parameter Values Used to Calculate
the Reactor Coolant Fission Product Source Term

Parameter	Maximum[(1)]	Normal[(2)]
Core power level (MWt)	4,063	3,983
Duration of reactor operation (core cycles)	5	-
Equilibrium fuel cycle (effective full-power days)	480	-
Thermal neutron flux, n/cm^2-sec	6.32E+13	-
Average thermal fission rate (fission/MW-sec)	3.12E+16	-
Fraction of fuel defect (fraction)	0.01	-
Reactor coolant mass, kg (lb)	2.92E+05 (6.43E+05)	2.92E+05 (6.43E+05)
Core-to-reactor coolant volume ratio (fraction)	0.073	-
Purification flow, kg/sec (lb/sec)	5.02 (11.07)	5.02 (11.07)
Purification flow for boron control, kg/sec (lb/sec), cycle average	-	2.65E-02 (5.85E-02)
Boron concentration at BOC (ppm), minimum	1,110	-
Ion exchanger and gas stripper removal efficiency		
CVCS purification ion exchanger		
Xe, Kr, tritium	0.0	0.0
Cs, Rb	0.5	0.5
Anion	0.99	0.99
Others	0.98	0.98
CVCS gas stripper		
Xe, Kr	0.999	-
Others	0.0	-
CVCS gas stripper operation	Continuous	None
Fission product escape rate coefficients (sec^{-1})		
Xe, Kr	6.5E-08	-
I, Br, Rb, Cs	1.3E-08	-
Mo	2.0E-09	-
Te	1.0E-09	-
Sr, Ba	1.0E-11	-
Y, Zr, Nb, Tc, Ru, La, Ce	1.6E-12	-

(1) Design basis source term (1 % fuel defect, DAMSAM code input)

(2) Expected source term (ANSI/ANS 18.1)

Table 11.1-2

Maximum Reactor Coolant Fission Product Source Term
(Core Power: 4,063 MWt, 1.0 % Fuel Defect, Continuous Gas Stripping)

Nuclide	Specific Activity (Bq/g)	Nuclide	Specific Activity (Bq/g)	Nuclide	Specific Activity (Bq/g)
Kr-85m	3.00E+04	Cs-136	1.89E+03	Nb-95	2.04E+01
Kr-85	7.40E+02	Cs-137	1.63E+04	Mo-99	1.11E+04
Kr-87	2.92E+04	N-16	8.22E+06(3)	Tc-99m	6.66E+03
Kr-88	7.40E+04	H-3	1.30E+05(4)	Ru-103	7.03E+00
Xe-131m	7.40E+03	Na-24	1.81E+03(1)	Ru-106	3.00E+00
Xe-133m	1.92E+03	Cr-51	5.48E+02	Ag-110m	5.15E+01(1)
Xe-133	9.62E+05	Mn-54	6.34E+01(1)	Te-129m	2.37E+02
Xe-135m	2.29E+04	Fe-55	4.75E+01(1)	Te-129	2.52E+02
Xe-135	1.30E+05	Fe-59	1.19E+01(1)	Te-131m	1.11E+03
Xe-137	5.55E+03	Co-58	1.82E+02(1)	Te-131	4.44E+02
Xe-138	1.96E+04	Co-60	2.10E+01(1)	Te-132	7.77E+03
Br-84	7.77E+02	Zn-65	2.02E+01(1)	Ba-137m	1.55E+04
I-131	9.99E+04	Sr-89	1.30E+02	Ba-140	1.59E+02
I-132	2.66E+04	Sr-90	8.88E+00	La-140	5.55E+01
I-133	1.41E+05	Sr-91	1.92E+02	Ce-141	5.92E+00
I-134	1.67E+04	Y-91m	1.11E+02	Ce-143	1.67E+01
I-135	7.77E+04	Y-91	1.89E+01	Ce-144	1.70E+01
Rb-88	7.40E+04	Y-93	4.44E+00	W-187	9.70E+01(1)
Cs-134	1.41E+04	Zr-95	2.40E+01(2)	Np-239	8.62E+01(1)

(1) Expected source terms based on ANSI/ANS 18.1 (Reference 1) are used when these values are higher than the design basis source terms, for added conservatism.

(2) Summation of fission and corrosion product specific activity

(3) Specific activity at the reactor vessel outlet nozzle

(4) Based on the tritium measurement in domestic operating reactors in Korea

Table 11.1-3

Assumptions Used in Determining Activities
in the Spent Fuel Pool Cooling and Cleanup System

1. During the 2 days after shutdown, the primary coolant is purified by the purification filter, purification ion exchanger, and gas stripper of the CVCS. The purified primary coolant is then diluted by the spent fuel and refueling pool cooling water.

2. During the first 30 days after shutdown, the primary coolant, spent fuel pool cooling water, and refueling pool cooling water are simultaneously purified by the spent fuel pool cooling and cleanup system; thereafter, the spent fuel pool cooling water is only purified.

3. The capacity of the spent fuel pool cooling and cleanup system is 1,324.9 L/min (350 gpm).

4. The decontamination factors of the spent fuel pool cooling and cleanup system demineralizers are as follows:

Noble gases	1
I, Br	100
Cs, Rb	2
Others	100

5. No credit is taken for removal of activity by filters in the spent fuel pool cooling and cleanup system when calculating the spent fuel pool and refueling pool cooling water activities. In calculating the cumulative activities of filters, the filter element is assumed to have a decontamination factor of 10.

Table 11.1-4

Maximum and Expected Specific Activities
in the Spent Fuel Pool and Refueling Pool (Bq/g)

Nuclide	Maximum[(1)]	Expected[(2)]`
H-3	4.0E+04	1.1E+04
N-16	0.0E+00	0.0E+00
Kr-85m	0.0E+00	0.0E+00
Kr-85	0.0E+00	0.0E+00
Kr-87	0.0E+00	0.0E+00
Kr-88	0.0E+00	0.0E+00
Xe-131m	0.0E+00	0.0E+00
Xe-133m	0.0E+00	0.0E+00
Xe-133	0.0E+00	0.0E+00
Xe-135m	0.0E+00	0.0E+00
Xe-135	0.0E+00	0.0E+00
Xe-137	0.0E+00	0.0E+00
Xe-138	0.0E+00	0.0E+00
Br-84	0.0E+00	0.0E+00
Rb-88	0.0E+00	0.0E+00
Sr-89	1.1E+00	4.4E–02
Sr-90	3.5E–01	1.9E–02
Sr-91	6.4E–02	1.2E–02
Y-91m	0.0E+00	0.0E+00
Y-91	2.5E+00	2.6E–02
Y-93	6.6E–03	2.4E–01
Zr-95	2.1E–01	1.3E–01
Nb-95	1.5E–01	8.3E–02
Tc-99m	3.3E–01	8.9E–03
Mo-99	5.3E+01	1.2E+00
Ru-103	5.4E–02	2.3E+00
Ru-106	6.5E–02	7.6E+01
Ag-110m	9.1E–01	8.9E–01
Te-129m	1.8E+00	5.5E–02

Nuclide	Maximum[(1)]	Expected[(2)]
Te-129	0.0E+00	0.0E+00
I-131	6.3E+02	5.2E–01
Te-131m	2.9E+00	1.5E–01
Te-131	0.0E+00	0.0E+00
Te-132	3.9E+01	3.4E–01
I-132	3.8E–04	3.2E–05
I-133	2.5E+02	1.8E+00
I-134	0.0E+00	0.0E+00
Cs-134	3.2E+02	3.6E–02
I-135	6.5E+00	1.8E–01
Cs-136	2.4E+01	4.7E–01
Cs-137	4.4E+02	6.2E–02
Ba-140	1.1E+00	3.5E+00
La-140	2.0E–01	3.5E+00
Ce-141	4.4E–02	4.4E–02
Ce-143	5.0E–02	3.3E–01
Ce-144	3.2E–01	2.9E+00
Na-24	1.8E+00	1.8E+00
Cr-51	4.0E+00	9.0E–01
Mn-54	1.3E+00	1.2E+00
Fe-55	1.5E+00	1.5E+00
Fe-59	9.4E–02	9.3E–02
Co-58	1.6E+00	1.6E+00
Co-60	7.4E–01	7.3E–01
Zn-65	3.5E–01	3.4E–01
Ba-137m	4.4E+02	6.2E–02
W-187	2.1E–01	2.1E–01
Np-239	3.8E–01	3.8E–01

(1) Design basis source term (1 % fuel defect)

(2) Expected source term (ANSI/ANS 18.1)

Table 11.1-5

Assumptions Used in Determining Secondary System Activities

1. Primary coolant activities are described in Subsection 11.1.1.1 for the design basis case and in Subsection 11.1.2.1 for the expected case.

2. Primary-to-secondary leak rates:

Design basis	3,270 L/day (0.6 gal/min)
Expected	34 kg/day (75 lb/day)

3. Flow rates in the secondary system (based on the two SGs):

Steam flow rate, kg/hr (lb/hr)	8.14×10^6 (1.80×10^7)
Continuous blowdown rate, kg/hr (lb/hr)	1.63×10^5 (3.59×10^5)
High-capacity blowdown rate (hot leg), kg/sec (lb/sec)	8.18×10^1 (1.80×10^2)

4. Liquid masses in the secondary system of two SGs, kg (lb): 2.41×10^5 (5.32×10^5)

5. Steam generator internal partition coefficients (Reference 7):

H-3	1.0
I, Br	0.01
Others	0.005

All noble gases are assumed to be in the steam.

6. Fractions of radionuclide in the main steam reaching the main condenser (Reference 7):

I, Br	0.2
Noble gases	1.0
Others	0.1

7. Decontamination factors of the blowdown demineralizer and condensate polishing demineralizer (Reference 7):

Demineralizer	Decontamination Factor				
	Noble Gases	I, Br	Cs, Rb	H-3	Others
Blowdown	1	100	100	1	100
Condensate polishing	1	10	2	1	10

Table 11.1-6

Design Basis Radionuclide Concentrations
in the Secondary System (Bq/g) (1 % Fuel Defect)

Nuclide	Steam Generator	
	Liquid	Steam
Kr-85m	-	3.71E-01
Kr-85	-	9.16E-03
Kr-87	-	3.61E-01
Kr-88	-	9.16E-01
Xe-131m	-	9.16E-02
Xe-133m	-	2.38E-02
Xe-133	-	1.19E+01
Xe-135m	-	2.83E-01
Xe-135	-	1.61E+00
Xe-137	-	6.87E-02
Xe-138	-	2.43E-01
Br-84	3.03E-01	3.03E-03
I-131	1.20E+02	1.20E+00
I-132	2.17E+01	2.17E-01
I-133	1.62E+02	1.62E+00
I-134	8.86E+00	8.86E-02
I-135	8.08E+01	8.08E-01
Rb-88	1.93E+01	9.64E-02
Cs-134	1.91E+01	9.56E-02
Cs-136	2.55E+00	1.28E-02
Cs-137	2.21E+01	1.11E-01
Cr-51	6.78E-01	3.39E-03
Mn-54	7.85E-02	3.93E-04
Fe-55	5.88E-02	2.94E-04
Fe-59	1.47E-02	7.36E-05
Co-58	2.25E-01	1.13E-03
Co-60	2.60E-02	1.30E-04
Zr-95	2.97E-02	1.49E-04
Zn-65	2.50E-02	1.25E-04

Nuclide	Steam Generator	
	Liquid	Steam
N-16	7.38E-01	3.69E-03
Na-24	2.08E+00	1.04E-02
Sr-89	1.61E-01	8.04E-04
Sr-90	1.10E-02	5.50E-05
SR-91	2.13E-01	1.06E-03
Y-91m	5.81E-02	2.91E-04
Y-91	2.34E-02	1.17E-04
Y-93	4.95E-03	2.47E-05
Nb-95	2.52E-02	1.26E-04
Mo-99	1.35E+01	6.76E-02
Tc-99M	6.93E+00	3.46E-02
Ru-103	8.70E-03	4.35E-05
Ru-106	3.72E-03	1.86E-05
Ag-110m	6.38E-02	3.19E-04
Te-129m	2.93E-01	1.47E-03
Te-129	1.57E-01	7.84E-04
Te-131m	1.32E+00	6.62E-03
Te-131	1.48E-01	7.38E-04
Te-132	9.49E+00	4.74E-02
Ba-137m	6.92E-01	3.46E-03
Ba-140	1.96E-01	9.81E-04
La-140	6.69E-02	3.34E-04
Ce-141	7.32E-03	3.66E-05
Ce-143	2.00E-02	1.00E-04
Ce-144	2.11E-02	1.05E-04
W-187	1.15E-01	5.73E-04
Np-239	1.05E-01	5.23E-04
H-3	1.69E+04	1.69E+04

Table 11.1-7

Radionuclide Crud Concentrations
in the High-Capacity Blowdown Liquid (Bq/g)

Nuclide	Design Basis Source	Expected Source
Cr-51	3.26E+02	3.12E+01
Mn-54	2.43E+04	2.75E+00
Fe-55	2.73E+03	5.35E+01
Fe-59	2.82E+04	3.93E+00
Co-58	8.71E+02	1.95E+02
Co-60	1.09E+02	1.09E+02
Zr-95	8.21E+01	8.21E+01
Zn-65	1.95E+01	1.95E+01

Table 11.1-8

Design Basis Radionuclide Concentrations of Sources to GRS (Bq/cm^3)(1)

Nuclide	Reactor Drain Tank(2)	Volume Control Tank	Gas Stripper(2)	Equipment Drain Tank(3)
H-3	1.0E+01	1.0E+01	1.7E+01	1.0E+00
Br-84	7.6E–01	1.0E–02	2.6E–01	7.2E–03
Kr-85m	2.7E+04	1.0E+03	1.0E+06	2.5E+03
Kr-85	6.6E+02	2.7E+01	2.5E+04	6.2E+01
Kr-87	2.6E+04	8.9E+02	9.8E+05	2.5E+03
Kr-88	6.6E+04	2.5E+03	2.5E+06	6.2E+03
Xe-131m	6.6E+03	1.5E+02	2.5E+05	6.2E+02
Xe-133m	1.7E+03	3.9E+01	6.4E+04	1.6E+02
Xe-133	8.6E+05	2.0E+04	3.2E+07	8.1E+04
Xe-135m	2.0E+04	2.5E+02	7.7E+05	1.9E+03
Xe-135	1.2E+05	2.5E+03	4.4E+06	1.1E+04
Xe-137	4.9E+03	2.7E+01	1.9E+05	4.7E+02
Xe-138	1.7E+04	2.0E+02	6.6E+05	1.7E+03
I-131	9.8E+01	1.7E+00	3.3E+01	9.3E–01
I-132	2.6E+01	4.2E–01	8.9E+00	2.5E–01
I-133	1.4E+02	2.3E+00	4.7E+01	1.3E+00
I-134	1.6E+01	2.4E–01	5.6E+00	1.5E–01
I-135	7.6E+01	1.2E+00	2.6E+01	7.2E–01

(1) 1.0 % fuel defect and continuous gas stripping are applied.

(2) Reactor drain tank and gas stripper specific activities are based on continuous venting at 0.680 L/m (0.024 scfm) and 9.061 L/m (0.32 scfm) to the GRS.

(3) Equipment drain tank specific activities are based on continuous venting at 0.14 L/m (0.005 scfm) to the GRS.

Table 11.1-9

Expected Specific Activities of Reactor Coolant During Normal Operation[(1)]
(Core Power: 3,983 MWt, No Gas Stripping)

Nuclide	Specific Activity (Bq/g)	Nuclide	Specific Activity (Bq/g)	Nuclide	Specific Activity (Bq/g)
Kr-85m	5.96E+02	Cs-136	3.70E+01	Nb-95	1.11E+01
Kr-85	4.33E+04	Cs-137	2.27E+00	Mo-99	2.51E+02
Kr-87	6.32E+02	N-16	1.48E+06[(2)]	Tc-99m	1.79E+02
Kr-88	6.70E+02	H-3	3.70E+04[(3)]	Ru-103	2.97E+02
Xe-131m	3.27E+04	Na-24	1.81E+03	Ru-106	3.57E+03
Xe-133m	2.71E+03	Cr-51	1.23E+02	Ag-110m	5.15E+01
Xe-133	1.18E+03	Mn-54	6.34E+01	Te-129m	7.52E+00
Xe-135m	4.83E+03	Fe-55	4.75E+01	Te-129	8.97E+02
Xe-135	2.51E+03	Fe-59	1.19E+01	Te-131m	5.84E+01
Xe-137	1.26E+03	Co-58	1.82E+02	Te-131	2.87E+02
Xe-138	2.27E+03	Co-60	2.10E+01	Te-132	6.68E+01
Br-84	5.97E+02	Zn-65	2.02E+01	Ba-137m	2.27E+00
I-131	8.23E+01	Sr-89	5.54E+00	Ba-140	5.14E+02
I-132	2.27E+03	Sr-90	4.75E-01	La-140	9.77E+02
I-133	1.04E+03	Sr-91	3.67E+01	Ce-141	5.94E+00
I-134	3.74E+03	Y-91m	1.72E+01	Ce-143	1.09E+02
I-135	2.13E+03	Y-91	2.06E-01	Ce-144	1.58E+02
Rb-88	7.07E+03	Y-93	1.61E+02	W-187	9.70E+01
Cs-134	1.59E+00	Zr-95	1.54E+01	Np-239	8.62E+01

(1) Expected source term (ANSI/ANS 18.1)

(2) Specific activity at the reactor coolant entering the letdown line

(3) The concentration of tritium is a function of the inventory of tritiated liquids in the plant, rate of production of tritium due to activation in the reactor coolant, rate of release from the fuel, and extent to which tritiated water is recycled or discharged from the plant. The value of tritium concentration listed in this table is typical in PWRs with the assumption that a moderate amount of tritium is recycled (Reference 1).

Table 11.1-10

Expected Radionuclide Concentrations in the Secondary System (Bq/g)

Nuclide	Steam Generator	
	Liquid	Steam
Kr-85M	-	1.04E-04
Kr-85	-	7.54E-03
Kr-87	-	1.10E-04
Kr-88	-	1.17E-04
Xe-131m	-	5.69E-03
Xe-133 m	-	4.72E-04
Xe-133	-	2.05E-04
Xe-135 m	-	8.41E-04
Xe-135	-	4.37E-04
Xe-137	-	2.19E-04
Xe-138	-	3.95E-04
Br-84	3.28E-03	3.28E-05
I-131	1.39E-03	1.39E-05
I-132	2.61E-02	2.61E-04
I-133	1.68E-02	1.68E-04
I-134	2.79E-02	2.79E-04
I-135	3.12E-02	3.12E-04
Rb-88	2.59E-02	1.30E-04
Cs-134	3.03E-05	1.52E-07
Cs-136	7.03E-04	3.52E-06
Cs-137	4.33E-05	2.17E-07
Cr-51	2.14E-03	1.07E-05
Mn-54	1.10E-03	5.52E-06
Fe-55	8.28E-04	4.14E-06
Fe-59	2.07E-04	1.04E-06
Co-58	3.17E-03	1.58E-05
Co-60	3.66E-04	1.83E-06
Zr-95	2.68E-04	1.34E-06
Zn-65	3.52E-04	1.76E-06

Nuclide	Steam Generator	
	Liquid	Steam
N-16	1.87E-03	9.35E-06
Na-24	2.93E-02	1.47E-04
Sr-89	9.65E-05	4.82E-07
Sr-90	8.28E-06	4.14E-08
Sr-91	5.72E-04	2.86E-06
Y-91m	1.27E-04	6.33E-07
Y-91	3.59E-06	1.79E-08
Y-93	2.52E-03	1.26E-05
Nb-95	1.93E-04	9.66E-07
Mo-99	4.30E-03	2.15E-05
Tc-99m	2.62E-03	1.31E-05
Ru-103	5.17E-03	2.58E-05
Ru-106	6.22E-02	3.11E-04
Ag-110m	8.97E-04	4.49E-06
Te-129m	1.31E-04	6.54E-07
Te-129	7.86E-03	3.93E-05
Te-131m	9.81E-04	4.90E-06
Te-131	1.34E-03	6.71E-06
Te-132	1.15E-03	5.74E-06
Ba-137m	1.43E-06	7.13E-09
Ba-140	8.92E-03	4.46E-05
La-140	1.66E-02	8.28E-05
Ce-141	1.03E-04	5.17E-07
Ce-143	1.84E-03	9.18E-06
Ce-144	2.75E-03	1.38E-05
W-187	1.61E-03	8.07E-06
Np-239	1.47E-03	7.36E-06
H-3	6.81E+01	6.81E+01

Table 11.1-11

Long-Lived Isotopes in Crud

Isotope	Half-Life	λ (d^{-1})	Parent	Reaction	γ/dis(1)	E (MeV)
Cr-51	27.70 days	2.50E-02	Cr-50	n, γ	0.1	0.32
Mn-54	312.3 days	2.22E-03	Fe-54	n, p	1	0.84
Fe-59	44.50 days	1.56E-02	Fe-58	n, γ	1	1.18
Co-60	5.272 years	3.60E-04	Co-59	n, γ	2	1.25
Co-58	70.82 days	9.77E-03	Ni-58	n, p	1	0.81
Zr-95	64.02 days	1.08E-02	Zr-94	n, γ	2	0.75

(1) gamma/disintegration

Table 11.1-12

Parameters for Crud Activity

Parameter	Value
Thermal neutron flux, n/cm^2-sec	6.32E+13
Fast neutron flux, n/cm^2-sec	3.06E+14
RCS surface area / core surface area, A_T/A_C	4.8

Core Residence Times and Activation Rates

Isotope	Core Residence Time (t_{res}, day)	Activation Rate (reactions/g-sec)
Cr-51	12	1.34E+11
Mn-54	110	4.37E+08
Fe-59	Saturated	1.99E+08
Co-58	23	4.18E+10
Co-60	197	4.32E+09
Zr-95	29	8.65E+08

Table 11.1-13

Long-Lived Crud Activity

Isotope	Half-Life	Activity (Bq/g-crud)
Cr-51	27.70 days	7.31E+09
Mn-54	312.3 days	1.99E+07
Fe-59	44.50 days	4.18E+07
Co-58	70.82 days	1.77E+09
Co-60	5.272 years	6.22E+07
Zr-95	64.02 days	4.90E+07

Table 11.1-14

Calculated Average Crud Activity
in the Reactor Coolant

Isotope	Activity (Bq/g-coolant)
Cr-51	5.48E+02
Mn-54	1.49E+00
Fe-59	3.14E+00
Co-58	1.33E+02
Co-60	4.66E+00
Zr-95	3.67E+00

Table 11.1-15

Tritium Activation Reactions

Reaction	Threshold Energy, MeV	Cross Section, $cm^{2(1)}$
$B^{10}(n, 2\alpha)T$	1.4	1.26E-26
$Li^{7}(n, n\alpha)T$	3.9	7.79E-27
$Li^{6}(n, \alpha)T$	Thermal	9.44E-22
$H^{2}(n, \gamma)T$	Thermal	5.50E-28
$B^{11}(n, T)Be^{9}$	10.4	7.30E-30
$N^{14}(n, T)C^{12}$	4.3	3.00E-28

(1) Spectrum averaged value for neutrons of energy greater than 0.625 eV

Table 11.1-16

Parameters Used for Calculating Tritium Production

Parameter	Value
Active core water volume, cm^3	3.00E+07
Thermal neutron flux, n/cm^2-sec	6.32E+13
Fast neutron flux, n/cm^2-sec	3.06E+14
Average lithium concentration in the reactor coolant, ppm	
Expected	2.2
Maximum	3.5
Lithium-6 abundance, a/o	0.1
Average boron concentration in the reactor coolant, ppm	
Expected	652
Maximum	755
Power level, MWt	4,063
Tritium release from fuel, %	
Expected	1.0
Maximum	2.0
Tritium release from CEA, %	50.0

Table 11.1-17

Tritium Production in the Reactor Coolant

Source	Average (Bq/cycle)	Maximum (Bq/cycle)
$H^2(n, \gamma)T$	2.57E+11	2.57E+11
$Li^6(n, \alpha)T$	1.09E+13	1.73E+13
$Li^7(n, n\alpha)T$	6.90E+11	1.10E+12
$B^{10}(n, 2\alpha)T$	4.28E+13	4.95E+13
Fission products(1)	1.04E+13	2.07E+13
CEAs	2.30E+12	1.29E+13
Total	6.73E+13	1.02E+14

(1) Tritium production from ternary fission (ORIGEN-S code)

Table 11.1-18

Maximum Anticipated Leakage Rates from NSSS-Related Components to the Building Environment

Component	Assumed Leakage Rates
Valves	
Disk leakage	4 cm^3/hr/cm of seat diameter
Stem leakage	4 cm^3/hr/cm of stem diameter
Pumps	
Centrifugal (mechanical seals) (except SI and SC pumps)	50 cm^3/hr per seal during normal operating conditions with availability of seal cooling water
	100 cm^3/hr per seal during loss of externally supplied cooling water
Positive displacement	3,785 cm^3/hr (1 gal/hr)
SI and SC pumps	1,000 cm^3/hr per seal (each pump)
Flanges	30 cm^3/hr

11.2 Liquid Waste Management System

The liquid waste management system (LWMS) is designed to monitor, control, collect, process, handle, store, and dispose of liquid radioactive waste generated during normal plant conditions, including anticipated operational occurrences (AOOs) based on NRC RG 1.143 (Reference 1), NUREG-0017 (Reference 2), and other applicable codes and regulations delineated in this section. The radionuclide concentration of liquid effluent releases at the site discharge point during normal operation including AOOs is below the radionuclide concentration limit in 10 CFR 20, Appendix B (Reference 3), and conforms with as low as is reasonably achievable (ALARA) criteria of 10 CFR 50, Appendix I (Reference 4) based on the use of industry-proven technologies within the structures, systems, and components (SSCs) incorporated into the design. The design also incorporates segregation of liquid waste collection and processing to provide reasonable assurance that the treatment and release objectives are met.

The LWMS design is supplemented with operating procedures, programs, and operator actions to provide assurance that the SSC integrity and functions are maintained, and the releases are within the limits specified in 10 CFR 20, Appendix B (Reference 3).

The lessons learned program provides guidance on integrating industry, operating, and construction experience into the APR1400 design. Under this program, NRC generic communications and industry operating and construction experience are maintained in a database that is reviewed, assessed, and integrated into the design as appropriate. The construction and operating experience of nuclear power plants has been incorporated into the database for design improvement.

The LWMS is divided into the following major subsystems:

a. Floor drain subsystem

b. Equipment waste subsystem

c. Chemical waste subsystem

d. Detergent waste subsystem

The radiation level in the processed liquid is verified by radiation monitors and sampling and analysis, in accordance with NRC RG 1.21 (Reference 5) and RG 4.15 (Reference 6), prior to release to the environment. Process sampling and effluent radiation monitoring systems are described in Section 11.5.

11.2.1 Design Bases

11.2.1.1 Design Objectives

The LWMS meets the following design objectives:

a. Capability to process floor drain wastes, equipment wastes, chemical wastes, and detergent wastes to meet release radionuclide concentration limits in accordance with 10 CFR 20, Appendix B (Reference 3), prior to discharge to the environment using industry proven technologies including filtration, reverse osmosis (R/O), and ion exchange

b. Capability to recycle treated water to minimize the liquid radwaste effluent releases to the environment in accordance with GDC 60 (Reference 7) through the use of redundant and independent tanks and pumps

c. Capability to segregate the liquid waste streams by the use of separate waste drain headers and waste collection sumps or tanks for each waste stream category

d. Capability to store, sample, and analyze processed liquid to confirm conformance with the release limits prior to the discharge being released to the environment or returned to the waste collection tank for further treatment, with the provision of redundant tanks and pumps and the sampling capability

e. Capability to prevent unplanned discharges of processed liquid by the use of an administratively controlled discharge valve and dual radiation monitors to provide assurance that the radioactivity concentrations of effluents and the resultant doses are within the limits of 10 CFR 20, Appendix B (Reference 3), and 10 CFR 50, Appendix I (Reference 4), respectively, and that there are no uncontrolled and/or unmonitored releases

f. Capability to provide early detection of leaks and overflows through the use of the cubicle drainage design and instruments that initiate alarms in the radwaste control room for operator actions in accordance with NRC RG 4.21 (Reference 8)

11.2.1.2 Design Criteria

The LWMS design criteria are as follows:

a. The LWMS provides sufficient capacity, redundancy, and flexibility to treat liquid radwaste to reduce radionuclide concentration to the concentration limit of effluents in 10 CFR 20, Appendix B (Reference 3), during equipment downtime and during operation at design basis fission product leakage levels (leakage from fuel producing 1 percent of the reactor power).

b. The LWMS is designed so that releases of radioactive materials to the environment are controlled and monitored in accordance with General Design Criteria (GDC) 60 (Reference 7), GDC 61 (Reference 9), and GDC 64 (Reference 10) with the incorporation of an administratively controlled valve and dual radiation monitors on the sole discharge line for liquid effluent. The release is to be controlled by the offsite dose calculation manual (ODCM) following the Nuclear Energy Institute (NEI) 07-09A template, which is to be developed by COL applicant (COL 11.2(1)). Offsite radiation doses measured on an annual basis resulting from the effluents during normal operation and AOOs are maintained within the limits of 10 CFR 50, Appendix I (Reference 4). The release is controlled in accordance with 10 CFR 50.34a (Reference 11).

c. The LWMS is designed, constructed, and tested to the codes and standards listed in Table 11.2-7 in accordance with Regulatory Positions C.1.1.1 and C.4 of NRC RG 1.143 (Reference 1).

d. The LWMS is designed with adequate storage capabilities for normal operation, including AOOs, in accordance with RG 1.143 (Reference 1) and ANSI-55.6 (Reference 12), and the capability to connect to and return liquid waste from the installed liquid waste processing system to accommodate and treat anticipated waste surge volumes. The interconnections between plant systems and the liquid waste processing equipment are designed to avoid the contamination of nonradioactive systems and uncontrolled releases of radioactivity to the environment. This design feature meets the requirements of Inspection and Enforcement (IE) Bulletin No. 80-10 (Reference 13).

The LWMS is designed with hard piping between radioactive and nonradioactive systems in accordance with IE Bulletin 80-10 (Reference 13). Chemical addition piping for pH adjustment is provided with hard piping, and demineralized water is provided for flushing the pipes after each transfer of contaminated fluid. These connections are hard pipes and are equipped with double barriers to prevent unintended contamination in accordance with NRC RG 4.21 (Reference 8).

e. The LWMS is not designed for abnormal or accident plant conditions but is designed to hold up and process the liquid wastes generated during normal conditions and AOOs.

f. The LWMS is designed to operate in batch mode during normal operating conditions and AOOs, but can be operated continuously when there is a need. For equipment sizing and process capability determination, the LWMS is designed to store and process the maximum design basis input in 1 day. The LWMS includes eight collection tanks (386,112 L [102,000 gal] gross volume) and two reverse osmosis (R/O) packages, each with 227 L/min (60 gpm) of processing capacity. Additionally, the LWMS contains two monitor tanks (102,206 L [27,000 gal] gross volume each) that hold the treated water prior to discharge. The equipment capacities of the LWMS are listed in Table 11.2-6.

g. Design features are included to reduce equipment maintenance, equipment downtime, and leakage of radioactive liquid into the building atmosphere using industry-proven components with proper material selections that are compatible with fluid conditions and the radioactive environment and lessons learned from nuclear industry as specified in NRC RG 1.143 (Reference 1), Figure 2. The structures, systems, and components (SSCs) of the LWMS are classified in accordance with the safety classification process described in NRC RG 1.143 (Reference 1), Figure 2, which includes RW-IIa, RW-IIb, and RW-IIc. The SSCs are designed in conformance with the applicable codes and standards and the guidelines in NRC RG 1.143 (Reference 1). The seismic design criteria and quality group classification applicable to the design of the LWMS are described in

Section 3.2. The LWMS design is in conformance with ANSI/ANS-55.6 (Reference 12).

h. In accordance with ANSI/ANS-55.6 (Reference 12) and RG 1.143 (Reference 1), the portions of the compound building that house the LWMS equipment are designed to maintain SSC integrity in the event of an operating basis earthquake (OBE).

i. The LWMS is designed with proven technologies that include filtration, R/O, ion exchange, and the associated piping and controls for automated operation upon manual initiation. The number and the capacities of the components are designed to provide flexibility for single and/or parallel train operation. This design approach provides simple but efficient processing for effective removal of radionuclides from the contaminated liquid in compliance with 10 CFR 20, Appendix B (Reference 3) and 10 CFR 50, Appendix I (Reference 4).

j. The components in the LWMS, except for the microfiltration unit, are designed in accordance with the applicable industry codes with pressure relief valves and vents opening into the cubicles and operating at ambient temperatures without heat generation or a vacuum drawing device. The microfiltration unit is designed to use a vacuum-drawing device equipped with a rupture disc and relief valve.

k. Each LWMS tank is provided with vent piping that is terminated at the vicinity of the inlet duct of the heating, ventilation, and air conditioning (HVAC) system in the compound building. The HVAC system in the compound building is described in Subsection 9.4.7.

l. The LWMS is designed with two radiation monitors in the sole discharge line inside the compound building. When a predetermined setpoint is exceeded, the monitors generate alarm signals in the radwaste control room that close the discharge line isolation valves to prevent release if either monitor detects activity in excess of the predetermined setpoint or if there are equipment malfunctions during the release. The LWMS is designed with enough redundancy and capacity that, if the isolation valves are closed, it can operate without discharging until the alarm condition is resolved. Administrative control of the lock-closed release path adds confidence that operator error will not cause inadvertent discharges of liquid waste that contains activity in excess of the limits for release. Offsite effluent concentrations and radiation doses to the public, which are controlled by the ODCM and measured annually, are within the limits of 10 CFR 20, Appendix B (Reference 3), and 10 CFR 50, Appendix I (Reference 4), respectively.

m. The LWMS is designed in accordance with NRC RG 4.21 (Reference 8) as it contains radioactive liquid from the plant. The required design and operational objectives of NRC RG 4.21 (Reference 8) are addressed in Section 12.4.2. The LWMS features and programs that meet these objectives are described in Subsection 11.2.2.4.

n. The LWMS is designed in accordance with 10 CFR 20.1406 (Reference 14) and NRC RG 4.21 (Reference 8). The design includes early leak detection features for overflows, unintended leakage, and releases. The early leak detection instruments are provided for LWMS components that contain significant amounts of contaminated fluid and are designed to provide alarm signals in the radwaste control room for operator actions. Procedures are to be developed by the COL applicant to provide periodic inspection and calibration for the maintenance of the instruments. Procedures will be further developed that govern the timely assessment and appropriate responses for the leakage mitigation, including investigation, rerouting of inputs, transferring of tank contents, decontamination, and the required maintenance on the SSCs, as required. The procedures are to be integrated into a plant-wide RG 4.21 Program in accordance with NEI 08-08A, Generic FSAR Template Guidance for Life Cycle Minimization of Contamination (Reference 15) (COL 11.2(2)).

o. Each waste collection tank and monitor tank is provided with an overflow connection at least as large as the inlet. The location of the overflow is above the high-level alarm setpoint. Each cubicle housing these tanks is coated with an impermeable epoxy liner (coating), up to the top of the cubicle wall, to facilitate decontamination of the facility in the event of tank leakage. Epoxy coatings in the cubicles are Service Level II coatings as defined in NRC RG 1.54 (Reference 16). This design feature, in conjunction with early leak detection and operator actions to drain and transfer collected leakage serves to minimize the release of the radioactive liquid to the groundwater and environment in accordance with Branch

Technical Position (BTP) 11-6 (Reference 17), 10 CFR 20.1406 (Reference 14), and NRC RG 4.21 (Reference 8).

p. The LWMS is designed in conformance with ALARA principles for minimizing occupational doses as described in NRC RG 8.8 (Reference 18) with the provision of individual cubicles for components that contain significant amounts of contaminated fluid, the leak detection and drainage design, the provision of area and component decontamination capability to minimize the buildup of contamination, and the adequate layout and efficient ingress and egress pathways. Sufficient shielding is provided for all equipment in the radiologically controlled area (RCA) to minimize radiation exposure.

q. The quality assurance (QA) program for the design, installation, procurement, and fabrication of LWMS components conforms with Regulatory Position C.7 of NRC RG 1.143 (Reference 1) and NRC RG 1.33 (Reference 19). Table 3.2-1 in this Design Control Document (DCD) identifies the seismic category, quality group, and safety class for components of the LWMS. The QA program is designed in accordance with ANSI/ANS-55.6 (Reference 12).

11.2.1.3 Method of Treatment

The LWMS provides for the segregated collection of floor drainage, equipment drainage, chemical drainage, and detergent drainage and has permanently installed equipment to store and treat the influent and allow sampling of the contents in the waste collection tanks and monitor tanks. The results of the sample analysis are used to confirm whether the treatment requirements and product specifications are met. Two 100 percent capacity R/O package systems are included in the LWMS to provide redundancy and operating flexibility. Each R/O package contains a pretreatment module that removes suspended solids, oil, and organic contaminants; an R/O module that removes soluble salts and radioactive ions from the waste passing through the pretreatment module; and a demineralizer module that polishes the R/O permeate for further removal of residual ionic radionuclides. This combination of processing equipment provides the necessary treatment and purification of the liquid effluent to meet the discharge requirements.

The LWMS components are equipped with piping, pumps, valves, and the necessary instrumentation to operate automatically. The components and piping are housed in the compound building, which provides shielding and temporary containment of leakage and drainage through the use of cubicles with epoxy coating, leak detection instruments, and operator actions. The design provides options for operator initiation and termination of the process operations and the selection of treatment components to achieve treatment objectives and effluent specifications to meet the set of regulations, including 10 CFR 20, Appendix B (Reference 3) and 10 CFR 50, Appendix I (Reference 4) efficiently. Due to equipment redundancy, the components can be divided into separate trains that can operate in series, or parallel, based on the contamination levels and treatment needs. The design also provides recycle capability in the event that additional processing is required to meet the release specifications. Details of the system and component descriptions and operations are provided in Subsection 11.2.2.

Detergent wastes from personnel decontamination showers and detergent-type decontamination solutions drainage are unlikely to have high radioactivity. The detergent waste is collected, filtered, and released through a monitored pathway. In the unlikely event that the radionuclide concentration is above a setpoint, the detergent waste is diverted to the chemical waste tank (CWT) for additional processing.

Depending on site-specific requirements, the COL applicant is to determine whether contaminated laundry is sent to an offsite facility for cleaning or for disposal (COL 11.2(3)).

Tanks are equipped with high-level and high-high level alarms that alert operators of high liquid levels in order to minimize the potential for overflow. If an operator action is not taken, the overflow over the high-high level can be directed to the other storage tank through the cross-connections.

11.2.1.4 Radioactive Source Terms in LWMS

Radioactive sources in the radwaste systems include fission and activation radionuclides produced in the core and the reactor coolant. The radioactive source terms in each LWMS component are determined using the DIJESTER Computer Code (Reference 20).

The DIJESTER Code (Reference 20) determines radionuclide inventories and concentrations by solving the differential equations of flow through the component, taking into account the flow rate, liquid source concentrations, decontamination factors, process time, equipment volume, and decay constant.

Each of the three liquid radwaste streams (chemical waste, floor drain waste, and equipment waste) has several input flows, and each input has a different source concentration. The main flow paths of the liquid radwaste system are shown in Figure 11.2-2. The COL applicant is to provide the piping and instrumentation diagrams (P&IDs) (COL 11.2(4)). The specific activities for each radwaste stream are determined with a fraction of primary coolant activity.

For the purposes of the radioactivity calculation, resin beds in the LWMS demineralizers are assumed to have a service life of 1 year. Although the service life of filters and resins in the LWMS may vary according to the operational conditions, filters and resins are likely to be replaced based on the media performance, such as pressure drop and media integrity. Source term strength is conservatively assumed for 1 year of fluid processing to provide reasonable assurance that occupational exposures associated with radwaste system operations remain ALARA. The decontamination factors used in the source term calculations are presented in Table 11.2-3. The expected and design basis (1 percent fuel defect) radioactive inventories in LWMS components are provided in Tables 11.2-11 through 11.2-14.

The structure and components of the LWMS are classified according to the safety classification guidance in NRC RG 1.143 (Reference 1). Analyses are performed based on the method stipulated in Figure 2 of NRC RG 1.143 (Reference 1). For conservative analysis, the radiological impact is assumed to be greater than the criteria of either 5 mSv at the unprotected area boundary or 0.05 Sv to facility personnel within the protected boundary. The facility is thus considered RW-IIa.

The radioactive inventories in the LWMS components are determined based on 1 percent fuel defect and compared with the A1 and A2 values in Appendix A of 10 CFR 71 (Reference 21). If the radioactivity inventories of a component exceed the A1 quantities, the component is classified as RW-IIa. If the radioactivity inventories are less than A1 quantities and greater than A2 quantities, the component is classified as RW-IIb. All other components are classified as RW-IIc. The results are included in Tables 11.2-13 and 11.2-14.

11.2.1.5 Site-Specific Cost-Benefit Analysis

The cost-benefit analysis approach stipulated by Paragraph II.D of 10 CFR 50, Appendix I (Reference 4) requires that a population dose analysis be performed to demonstrate that the LWMS is designed in accordance with the ALARA criterion.

Due to the site-specific nature of the population dose analyses, the cost-benefit analysis is deferred to the site-specific environmental reports.

The COL applicant is to perform a site-specific cost-benefit analysis to demonstrate conformance with the regulatory requirements of NRC RG 1.110 (COL 11.2(5)) (Reference 22).

11.2.1.6 Mobile or Temporary Equipment

The LWMS is designed with permanently installed equipment. The LWMS does not include the use of mobile or temporary equipment. To provide the flexibility for future use of mobile or temporary equipment in accordance with site-specific requirements, space and connections are provided for the installation of mobile equipment.

The COL applicant is to provide assurance that the use of mobile or temporary equipment and interconnections to plant systems conform with regulatory requirements and guidance such as 10 CFR 50.34a, 10 CFR 20.1406, and NRC RG 1.143 (COL 11.2(6)) (References 11, 14, and 1).

The COL applicant is responsible for the identification of mobile/portable LWMS connections that are considered nonradioactive but may later become radioactive through contact with contaminated radioactive systems, and for the preparation of operating procedures for mobile/portable LWMS connections in conformance with the guidance and information in IE Bulletin 80-10 (Reference 13) and ANSI/ANS 40.37 (Reference 23) (COL 11.2(6)).

11.2.2 System Description

The LWMS manages liquid wastes generated by the plant during normal operation including AOOs. The boundary of the LWMS starts at the interface valves for each of the input streams potentially containing radioactive material from other plant systems as shown in Figure 11.2-1. For many of these streams, the boundary of the LWMS starts at the sump pump discharge isolation valves from the respective building sump. The boundary of the LWMS ends at the isolation valve of the sole discharge line to the discharge header.

The LWMS consists of the following four main subsystems, which are based on characteristics of input streams: equipment waste, floor drain waste, chemical waste, and detergent waste. The subsystems have eight waste collection tanks, eight waste collection tank pumps, two R/O packages, two monitor tanks, and two monitor tank pumps to collect treated fluid for analysis. The waste collection tanks (WCTs) and monitor tanks and their associated pumps are located in the compound building. The R/O packages in the compound building are at elevations of 63 ft 0 in and 85 ft 0 in.

The detergent waste subsystem included in the LWMS has two detergent waste tanks, two detergent waste tank pumps, and one filter. The detergent waste tanks and their associated pumps are in the compound building at El. 63 ft 0 in.

Figure 11.2-1 shows a process flow diagram of the LWMS. For the purpose of this DCD, process flow diagrams are used to indicate key process equipment and primary control instrumentation to illustrate the process design, method of operation, and release monitoring.

Inputs to the LWMS include the following:

a. Equipment drainage

b. Floor drainage

c. Chemical drainage

d. Detergent drainage

For a more detailed discussion of the drainage systems, refer to Subsection 9.3.3.

11.2.2.1 Liquid Waste Processing System Operation

11.2.2.1.1 Waste Input Streams

Sources of radioactive liquid wastes include the following:

a. Floor drain wastes, including but not limited to the reactor containment and auxiliary building floor drains and compound building drains

b. Equipment wastes, including but not limited to the auxiliary building equipment drains

c. Chemical wastes, including but not limited to the radiochemistry laboratory, fuel handling area, and equipment decontamination drains

d. Detergent wastes, including but not limited to the wastes from the personnel decontamination station and detergent-type decontamination solutions, which occur in the unlikely event of high radioactivity surpassing the effectiveness of the detergent waste filter

e. All of the potentially radioactive waste streams, such as the auxiliary steam condensate receiver tank drains and condensate polishing area sump drains are monitored for radiological contamination. Contaminated drains from these sources are routed to the LWMS for processing prior to discharge to the environment

11.2.2.1.2 Waste Collection and Storage

The LWMS is not designed for abnormal or accident plant conditions, but the LWMS has the capacity to hold up the liquid wastes generated during normal conditions and AOOs.

Input to the LWMS is divided into four process trains: floor drain, equipment waste chemical waste, and detergent waste. Wastes are segregated according to the processing requirements and treatment objectives.

Floor drains, including the reactor containment and auxiliary building floor drains and compound building drains, are routed to the floor drain train and processed by the R/O package.

Equipment wastes, including the auxiliary building equipment drains, are routed to the equipment waste train where wastes are processed by the R/O package.

Chemical wastes, including the high-level laboratory, low-level laboratory, fuel handling area, and equipment decontamination drains, are directed to the chemical waste train where wastes are normally processed by the R/O package.

Two floor drain tanks and two equipment drain tanks are cross-connected by the common header and overflow piping on each tank. During normal operation, the various inputs from equipment drain and floor drain headers are separated and collected in the designated radwaste tanks.

Two chemical waste tanks receive the influent from a common inlet header. Normally, one tank is filled while the other is on standby. Tanks are also equipped with cross-connected and overflow piping. The chemical waste tank also collects borated waste from the boric acid concentrator in the chemical and volume control system.

The chemical drain sump in the auxiliary building collects chemical wastes from the power block and transfers it to the chemical waste tanks in the compound building.

Two detergent waste tanks receive detergent wastes, including but not limited to the wastes from personal decontamination stations and detergent-type decontamination solutions. These wastes are normally filtered and discharged directly through the LWMS discharge line and are radiologically monitored. When the drains are detected to be above a predetermined setpoint, they are routed to the chemical waste tanks where wastes are processed by the R/O package. Refer to DCD Section 11.5 for more detailed description of the radiation monitoring system.

The floor drain tanks, equipment drain tanks, and chemical waste tanks are housed in separate cubicles with individual leak detection instruments and sloped and epoxy coated floors to facilitate drainage and cleaning. In the event that a small amount of fluid is accumulated in the drain pipe and triggers the level detection instrument, an alarm is initiated in the radwaste control room for timely operator actions. This design approach meets the requirements of RG 4.21 (Reference 8).

11.2.2.1.3 Waste Processing

The LWMS is designed to operate with a tank-to-tank manual batch operation according to the plant condition. Therefore, the LWMS is designed with the high degree of flexibility illustrated on the process flow diagrams.

Floor drain and equipment waste are routed to the R/O package for processing. The chemical waste is also processed by the R/O package and discharged to the monitor tanks. Downstream of the chemical waste pumps, there is a provision to connect to a mobile chemical waste treatment system that processes boric acid concentrates, if required.

The liquid waste processing system has two R/O packages. Each R/O package has four modules: pre-treatment, R/O, demineralizer, and concentrate feed.

The liquid wastes collected in the floor drain tanks, equipment waste tanks, and chemical waste tanks are first passed through a pre-treatment module in which oily and suspended solids are removed to maintain optimal performance of the R/O module. The suspended solids include the corrosion (e.g., sodium and iron) and activation products (e.g., cobalt and strontium), SG cruds, and other particulate from the floor drains. The passed water is routed to the R/O module for the removal of soluble species, which includes the ionic activation products (e.g., cesium and technetium). The activation products can be in particulate or ionic form, depending on pH and other fluid chemistry species (such as carbonates, hydroxides, and nitrates) in the liquid waste. Between the pretreatment and the R/O units, most of the radionuclides are removed from the R/O permeate. Current advances in treatment technologies and industry experience demonstrate the effectiveness of these treatment units and provide reasonable assurance that the 10 CFR 20, Appendix B (Reference 3) release limits are met. To further enhance the permeate quality, the R/O permeate from the R/O module is processed by the demineralizer module for final polishing and then transferred to the monitor tank for sampling, analysis, and release.

The generation of mixed liquid wastes is minimized by process control and the controlled use of hazardous chemicals. If and when mixed wastes are generated, the waste is to be separately collected and shipped for offsite treatment by the COL applicant (COL 11.2(7)).

11.2.2.1.4 Waste Sampling

The waste sampling process is similar for all LWMS tanks including the detergent waste tanks. Once a tank is filled, the operator initiates the recirculation/mixing mode by aligning the appropriate valves and starting the pump with the objective of obtaining tank homogeneity and representative sampling. See Section 11.5 for more detailed description of the process sampling system.

Obtaining a representative sample involves recirculating the tank contents and mixing the equivalent of three or more tank volumes using a mixing eductor as described in Subsection 11.2.2.3.1 depending on the type of influent waste and size of the tank. Recirculation for mixing and purging of the sampling line is to follow the guidance in ASTM D-3370-07, "Standard Practices for Sampling Water from Closed Conduits," as stipulated in NRC RG 1.21 (Reference 5).

Once adequate mixing has been achieved, a sample is taken while the pump is running using the sample connection provided on the recirculation piping. A sample of the liquid waste, except detergent waste, is taken at the sample panel of the process sampling system. The sample of detergent waste is taken at the grab sample sink. The samples are analyzed for chemical composition, gross gamma activity, and pH at the sample laboratory in the compound building. The design of the sampling system conforms to NRC RG 4.15 (Reference 6). Operating procedures for calibration and inspection following the guidance in NRC RG 4.15 (Reference 6) are to be developed by the COL applicant to provide reasonable assurance of representative sampling (COL 11.2(8)).

11.2.2.1.5 Chemical Addition

Following the sample analysis, the operator determines the amount and type of chemicals to be added to the tank to obtain a balanced pH for the tank fluid. Through permanently piped connections to the equipment waste tanks, floor drain tanks, and chemical waste tanks, acid, caustic, and anti-foam agents can be added to the tanks using metering pumps as necessary.

The chemicals are then mixed with the tank contents using the recirculation/mixing mode, followed by sampling and further chemical addition, if necessary. This process is repeated until the tank contents meet the required fluid pH for discharge.

11.2.2.2 Monitoring and Discharge

LWMS monitor tanks collect liquid processed through the R/O package. Following the sample analysis as described in Subsection 11.2.2.1.4, the operator determines where the contents of the monitor tank are to be transferred. If the water quality and radionuclide concentrations of the contents in the monitor tank meet the water specifications for the holdup tank, and the plant load following operation is not affected by the recycle operation, the contents of the monitor tank are transferred to the holdup tanks for plant reuse. If the water is not recycled and it is determined to be acceptable for offsite release, the contents of the monitor tank are discharged to the offsite release point.

The LWMS is designed to control the release of the treated effluent. The effluent is stored in the monitor tank and is sampled and analyzed to confirm that the release nuclide concentrations are within the limits of 10 CFR 20, Appendix B, Table 2 (Reference 3), prior to release. The LWMS is designed with recycle capability for further treatment for any batch that exceeds the release specifications (Table 11.2-1 for normal operation). The release is also continuously monitored by dual in-line radiation monitors (RE-183 and RE-184) during release. Any portion of the flow that exceeds the predetermined setpoint will trigger alarms in the MCR and the radwaste control room for operator actions, simultaneously turn off the monitor tank pump, and close the effluent discharge valve that is under supervisory control. The design and setpoints of the radiation monitors are described in Section 11.5. The LWMS is designed with no release bypass.

11.2.2.3 Component Description

The LWMS components are determined for the radioactive safety classification in accordance with the guidance provided in NRC RG 1.143 (Reference 1). The component safety classification is summarized in Table 11.2-6. Accordingly, the LWMS is classified as RW-IIa, based on the highest safety classification for the components within the system boundary. The LWMS components are housed within the compound building, which has been determined to be RW-IIa.

The safety classification for the LWMS components applies to components up to and including the nearest valves, fittings, and/or welded/flanged nozzle connections.

Component design data of the LWMS are listed in Table 11.2-6. The component design data include equipment flow rates and capacity, construction materials, and design temperatures and pressures. The codes and standards that are applicable to the LWMS components are listed in Table 11.2-7 and are consistent with codes and standards in NRC RG 1.143, Table 1 (Reference 1).

11.2.2.3.1 Tanks

The equipment waste tanks are the vertical, cylindrical type. Two tanks are provided in the LWMS to receive equipment drainage that is radioactively contaminated but contains a low level of suspended solids. The equipment waste tanks are used as backup tanks for the floor drain tanks whenever needed.

The floor drain tanks are the vertical, cylindrical type. Two tanks are provided in the LWMS to receive floor drainage from the reactor containment building, auxiliary building, compound building, and turbine building, which are expected to contain low levels of undissolved solids and oily contaminants.

The chemical waste tank is the vertical, cylindrical type. Two tanks are provided in the LWMS to receive influent from a common inlet header. Normally, one tank is filled while the other is on standby. Tanks are also equipped with cross-connect and overflow piping. The chemical waste tank also collects borated waste from the boric acid concentrator in the chemical and volume control system.

The detergent waste tank is the vertical, cylindrical type. Two detergent waste tanks are provided in the detergent waste subsystem. When one of the two detergent waste tanks is filled, the operator directs the influent to the other empty tank. The tank is sized to accommodate the expected daily peak volume of wastes.

The monitor tank is the vertical, cylindrical type. Two monitor tanks are provided in the LWMS. The monitor tanks are the only tanks in the LWMS from which the processed waste can be released to the environment or returned to the plant for reuse.

Eductors are used to facilitate mixing and resuspension of settled solids using the tank pump to provide the recirculation flow for a predetermined amount of mixing. Following the guidance in ASTM 3370 (Reference 25), three or more turnovers of the tank contents are required to provide reasonable assurance of adequate mixing in the tanks. During tank mixing, the sampling line is also recirculated back to the tank header in order to eliminate any stagnant fluid remaining inside the sampling line. The recirculation also facilitates the cleaning and purging of the sampling line for residues and adjusts background contamination to provide reasonable assurance of representative sampling. Following the completion of sampling, the sample line is to be purged by at least three pipe volumes of demineralized water, as stipulated in ASTM D3370-10 (Reference 25). The sampling point locations and methodology for the process and effluent sampling system are described in Section 11.5.

The sample points are located on the recirculation piping near each monitor tank. The sampling analysis is also to be used to compare with the radiation monitor readings during release. This design therefore uses two methods to provide reasonable assurance that no releases are allowed above the radioactivity limits.

The LWMS tanks are provided with level instruments to monitor liquid content levels. High level alarms are provided that, upon activation, terminate fluid input by closing the inlet valve to prevent overfilling and overflow. A low level alarm is also provided to trip off pumps to prevent cavitation and damage to the pumps. Both alarms are annunciated as a common alarm in the radwaste control room.

As discussed, the boundary of the LWMS ends at the monitor tank pump discharge isolation valve downstream of the radiation monitor points. The final discharge point location, the type, shape, and size of the discharge orifice, and the sampling of the discharge at the outfall are site-specific and are to be performed by the COL applicant (COL Item 11.2(8)).

The acid storage tank is a vertical, cylindrical type and stores sulfuric acid (H_2SO_4). The caustic storage tank is a vertical, cylindrical type and stores sodium hydroxide (NaOH). The chemical additive tank is a vertical, cylindrical type. In the chemical additive tank, the anti-form or flocculant is mixed with demineralized water. The acid batch tank is a horizontal, cylindrical type. In the acid batch tank, the H_2SO_4 from the acid storage tank is diluted before the chemicals are sent to the waste collection tank. The caustic batch tank is a horizontal, cylindrical type. In the caustic batch tank, the NaOH from the caustic storage tank is diluted before the chemicals are sent to the waste collection tank. The seal water storage tank is a vertical, cylindrical type, and stores radwaste water pump seal water.

The construction material for all tanks in the LWMS is stainless steel except for the acid batch tank, which is constructed of Incoloy 825 material. All tanks are to be designed, fabricated, welded, inspected, and tested in accordance with RG 1.143 (Reference 1).

The cells/cubicles housing tanks that contain significant quantities of radioactive material are coated with epoxy to facilitate drainage and decontamination. The coatings are Service Level II as defined in NRC RG 1.54 (Reference 16) and are subject to the limited QA provisions, selection, qualification, application, testing, maintenance, inspection provisions, and referenced standards of NRC RG 1.54 (Reference 16), as applicable to Service Level II coatings. Post-construction initial inspection is performed by personnel qualified in ASTM D 4537 (Reference 26) and according to the inspection plan guidance of ASTM D 5163 (Reference 27).

11.2.2.3.2 Pumps

Radwaste pumps are a horizontal, centrifugal type and are constructed of stainless steel. Two waste pumps are provided in each waste collection subsystem of the LWMS and transfer radwaste water from the waste collection tank to the R/O package for processing. The detergent waste tank pumps are the centrifugal, horizontal type. Two detergent waste tank pumps are provided in the detergent waste subsystem. The monitor tank pumps are the horizontal, centrifugal type. Two monitor pumps are provided in the LWMS and transfer the processed wastewater from the monitor tank to the point for offsite release or to the CVCS holdup tank for plant reuse.

Pumps are sized to process the contents of the tank in a single shift. The pump circulates the tank contents (through a mixing-eductor network located in the tank) to provide reasonable assurance of thorough mixing and representative sampling. The pump continually recirculates the tank contents during processing and discharging to prevent settling and potential radioactive crud buildup within the tank. The pump is automatically tripped on a low tank level signal.

The acid batch pump is a positive displacement type and transfers diluted H_2SO_4 solution from the acid batch tank to the equipment waste tanks, floor drain tanks, and chemical waste tanks to meet the required pH. The caustic batch pump is a positive displacement type and transfers diluted NaOH solution from the caustic batch tank to the equipment waste tanks, floor drain tanks, and chemical waste tanks to meet the required pH. The seal water pump is a horizontal, centrifugal type and transfers the seal water from the seal water storage tank to the seal water system of each radwaste water pump.

All pumps are to be designed, fabricated, welded, inspected, and tested in accordance with NRC RG 1.143 (Reference 1).

11.2.2.3.3 Detergent Waste Filter

The detergent waste filter is used for removing particles from detergent wastes. The detergent waste filter uses a cartridge-type filter. Cartridge filters are housed in shielded enclosures that assist with a simplified change-out with minimal occupational dose in conformance with the ALARA principle. The filter is contained in a top-loading, vertical stainless steel pressure vessel. Inlet flow is forced into the inside of the filter cartridge from the bottom and directed up into the outside of the filter cartridge. A spent filter cartridge is replaced manually.

The detergent waste filter is installed in an open area (i.e., there are no walls around the filter). The filter vessel is provided with lifting lugs and structural legs for anchoring to the foundation bolts.

All filter housings are to be designed, fabricated, welded, inspected, and tested in accordance with NRC RG 1.143 (Reference 1).

11.2.2.3.4 Seal Water Heat Exchanger

The seal water heat exchanger is a shell-and-tube type and removes heat from the seal water, which is returned from the seal water system of each radwaste water pump.

All heat exchangers are to be designed, fabricated, welded, inspected, and tested in accordance with NRC RG 1.143 (Reference 1).

11.2.2.3.5 R/O Package

Two R/O packages are provided to remove suspended solids, particulate/insoluble radionuclides, some metal and organic complexes, and soluble species from the different streams of liquid radwaste generated during plant operations. Each R/O package is capable of processing a minimum of 227 L/min (60 gpm) in normal operation.

The R/O package consists of the pre-treatment module, R/O module, demineralizer module, and concentrate feed module. The pre-treatment module, which consists of an oil removal filter and membrane filter (MF), removes the impurities in the radwaste water to maintain optimal performance of the R/O module. The oil removal filter is a column holding an oil removal filter cartridge designed to remove organic contaminants. This serves to protect the downstream ion-exchange media from fouling.

The R/O module, which uses spiral-wound type membranes, removes soluble species in the radwaste water. The demineralizer module uses three ion exchangers to polish R/O permeates. The concentrate feed module receives and stores the concentrates generated from the pre-treatment module and R/O module and transfers the concentrates from the concentrate holding tank to the concentrate treatment system of the SWMS.

The R/O package is capable of allowing maintenance to be performed on one process train while the other train continues to operate. Any component that requires frequent or routine maintenance, inspection, test, and adjustment of calibration is designed so that radiation exposures to operating and maintenance personnel are maintained ALARA.

The R/O package is designed using a modular (skid-mounted) approach to the greatest possible extent to provide reasonable assurance of easy installation and proper arrangement of components. Each R/O package is provided with an appropriate vent, drain, and flush line. The R/O package can be continuously operated from a control panel in the radwaste control room in the compound building. The control panel contains the controls and instrumentation to control the sequencing of events in the processes.

The demineralizer vessels are stainless steel pressure vessels with inlet distributors, connections equipped with screens to prevent infiltration of resins into the piping, and sluice outlets in the R/O package. Each demineralizer module has three vessels and is sized to process the contents passed through the R/O module.

All R/O housings and demineralizer vessels are to be designed, fabricated, welded, inspected, and tested in accordance with NRC RG 1.143 (Reference 1).

Access is provided to manually load fresh resin into the demineralizer vessels when required. The normal disposition of fully expended (high differential pressure, high radiation or loss of desired isotopic removal capability) demineralizer media is sluicing to the low-activity spent resin tank in the SWMS for processing and shipment offsite, as described in Section 11.4.

11.2.2.4 Design Features for Minimization of Contamination

The APR1400 is designed with features to meet the requirements of 10 CFR 20.1406 (Reference 14) and NRC RG 4.21 (Reference 8). The basic principles of NRC RG 4.21 (Reference 8) and the methods of control suggested in the regulations are delineated in four design objectives and two operational objectives, which are defined in Subsection 12.4.2. The primary features that address the design and operational objectives for the LWMS are described below.

The LWMS SSCs, including the facility that houses the components, are designed to limit leakage and/or control the spread of contamination. In accordance with NRC RG 4.21 (Reference 8), the LWMS has been evaluated for leakage identification from the SSCs that contain radioactive or potentially radioactive materials, the areas and pathways where probable leakage may occur, and the methods of leakage control incorporated in the design of the system. The leak identification evaluation indicated that the LWMS is designed to facilitate early leak detection and has the capability to assess collected fluids and respond to manage the collected fluids quickly. Thus, unintended contamination of the facility and the environment is minimized by the SSC design, and by operational procedures and programs for inspection and maintenance activities.

Prevention/Minimization of Unintended Contamination

a. The system components, including the collection tanks and the monitor tanks, are fabricated of stainless steel material and are of welded construction for life-cycle planning, thus minimizing leakage and unintended contamination of the facility and the environment.

b. The LWMS tanks are designed with sufficient capacity to provide temporary storage of the liquid waste generated from normal operation including anticipated operational occurrences. The tanks are equipped with cross-connected inlet headers for the control of overflow and to provide reasonable assurance of timely processing. The design minimizes the interruption of normal processing operation, the spread of contamination, and waste generation.

c. The LWMS tanks are designed with mixing eductors to minimize settling of suspended solids. The tanks have polished internal surfaces to minimize crud traps.

d. The LWMS is designed with minimum embedded or buried piping. Piping between buildings is designed to be equipped with piping sleeves with leakage directed back to the building for collection, thus preventing the spread of contamination.

<u>Adequate and Early Leak Detection</u>

a. All LWMS tanks are designed with level instruments to provide reasonable assurance of safe operation of the SSCs. The instruments provide alarms to the main control room and the radwaste control room for operator action in the event of high liquid levels.

b. The cubicles in which the LWMS tanks are located are designed to include leak detection instrumentation to initiate alarms for operator actions in the event of leakage. The leak detection design has the capability to detect a small quantity of leakage for early detection.

<u>Reduction of Cross-Contamination, Decontamination, and Waste Generation</u>

a. The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing waste generation.

b. The floor drains, equipment drains, chemical drains, and the detergent wastes are collected in segregated tanks in separate cubicles. Because the LWMS is designed to operate in batches, treatment for these collected wastes is determined through sampling and analyses. This design approach minimizes cross-contamination and waste generation.

c. The process piping containing contaminated solids is sized to facilitate flow and to provide for velocities that are sufficient to prevent the settling of solids. The piping is designed to reduce crud traps, thus reducing decontamination and waste generation. Decontamination fluid is collected and processed.

d. Utility connections are designed with a minimum of two barriers to prevent contamination of nonradioactive systems from potentially radioactive systems.

Decommissioning Planning

a. The SSCs are designed with decontamination capabilities. Design features such as spargers, welding techniques, and surface finishes are included to minimize the need for decontamination and minimize waste generation.

b. The SSCs are designed for the full service life and are fabricated, to the maximum extent practicable, as individual assemblies for easy removal.

Operations and Documentation

a. The LWMS is designed for automated operation with manual initiation when sufficient liquid volumes are accumulated in each category of collection tanks to warrant processing. Hence, the LWMS is designed to operate in batch modes. Adequate instrumentation, including level, flow rate, and pressure elements, and a process radiation monitor, is provided to monitor and control the operations to prevent undue interruption and the spread of radiologically contaminated material.

b. Piping is flushed clean after each processing to prevent solids from settling and accumulating. The flushing procedure also minimizes and/or prevents unintended leakage when the piping is not in use.

c. Leak detection instruments are provided to detect leakage of individual tanks. Adequate clearance around each tank is provided to enable prompt assessment and response when required.

d. The COL applicant is to develop the leak identification program (COL 11.2(9)) to identify site-specific components that contain radioactive materials, buried piping, embedded piping, leak detection methods and capabilities, and the methods that are used to prevent unnecessary contamination of clean components, facility areas, and the environment. The leak identification program, as part of the process control program, is designed to facilitate timely identification of leaks, prompt assessment, and appropriate responses to isolate and mitigate leakage.

e. The COL applicant is to prepare the operational procedures and maintenance program as related to leak detection and contamination control (COL 11.2(2)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning.

f. The COL applicant is to maintain complete documentation of the system design, construction, design modification, field changes, and operations (COL 11.2(10)).

Site Radiological Environmental Monitoring

The LWMS is included in the site process control program and the site radiological environmental monitoring program for monitoring facility and environmental contamination. The COL applicant is to prepare the site process control program and the site radiological environmental monitoring program (COL 11.2(11)). The site radiological environmental monitoring program includes sampling and analysis of effluent to be released, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants.

11.2.3 Radioactive Effluent Releases

11.2.3.1 Radioactive Effluent Releases and Dose Calculation in Normal Operation

Radioactive liquid effluents are treated by the LWMS and discharged through the plant discharge channel. The design of the LWMS components incorporates the decontamination factors, as a minimum, provided in NUREG-0017 (Reference 2), which are presented in Table 11.2-3. The PWR-GALE Code used in the liquid effluents release calculation is modified in accordance with NRC RG 1.112 (Reference 24), which requires applying ANSI/ANS 18.1-1999 (Reference 28).

The treatment process and release point, effluent temperature, effluent flow rate, and size and shape of flow orifices are site specific and are to be presented in the site-specific detail design. The COL applicant is to provide the site-specific information of the LWMS including radioactive release points, effluent temperature, and the shapes of flow orifices (COL 11.2(8)).

Annual liquid release source terms are calculated using the PWR-GALE Code (Reference 2), and input parameters are provided in Table 11.2-2. The concentration calculation uses an assumed dilution flow rate of 37,854 L/min (10,000 gpm) provided by cooling tower blowdown, dilution pump, or other plant discharges at the discharge point. The COL applicant is to confirm the assumed dilution flow rate based on site-specific parameters (COL 11.2(12)).

The expected radionuclide release rates of the various liquid effluent streams are presented in Table 11.2-1, with corresponding daily flow rates in Table 11.2-2. The total annual radionuclide release rates of the various liquid streams and their corresponding concentrations are adjusted by the annual dilution flow rate to calculate the liquid effluent concentrations. Table 11.2-10 provides the release concentrations for the design basis fuel leakage and the effluent concentration limits in 10 CFR 20, Appendix B (Reference 3).

The equation for calculating the design basis concentrations of the liquid effluent is as follows:

$$C(i) = \frac{R(i) \times MF_i}{F_{dil}}$$

Where:

C(i) = design basis liquid effluent concentration for the i[th] isotope, Bq/L

R(i) = total annual release rate of the i[th] isotope, Bq/yr (Table 11.2-1)

MFi = multiplication factor for the ith isotope (ratio of 1 % fuel defect design basis radionuclide concentration to ANSI/ANS-18.1-1999 expected concentration)

F_{dil} = dilution flow rate at discharge point, L/yr

The sum of concentration ratios for the design basis fuel leakage is 0.18, as presented in Table 11.2-10. This value is less than 1.0, which indicates that the releases meet the regulatory limit.

Offsite doses received by individuals as a result of radioactive liquid releases are calculated using the LADTAP II Code (Reference 29). The input parameters of the LADTAP II Code (Reference 29) are presented in Table 11.2-4. The dilution factor for aquatic food, boating, shoreline, swimming, and drinking water is assumed to be 5 for the normal operating conditions. The results of the dose calculation are presented in Table 11.2-5. The values are compared with the corresponding limits of 10 CFR 50, Appendix I (Reference 4). The maximum individual dose to total body is 0.018 mSv/yr for an adult. This value is less than the regulatory limit of 0.03 mSv/yr presented in 10 CFR 50, Appendix I (Reference 4). The maximum dose to any individual organ is 0.023 mSv/yr, which is the dose to a child's liver. This value is less than the limitation of 0.1 mSv/yr presented in 10 CFR 50, Appendix I (Reference 4).

The COL applicant is to calculate the dose to members of the public following the guidance of NRC RG 1.109 (Reference 30) and NRC RG 1.113 (Reference 31) using site-specific parameters and to compare the doses due to liquid effluents with the numerical design objectives of Appendix I to 10 CFR 50 (Reference 4), 10 CFR 20.1302 (Reference 32), and 40 CFR 190 (Reference 33) (COL 11.2(13)).

11.2.3.2 Radioactive Effluent Release due to Failure of Radioactive Liquid Tank

For the assessment of the impacts of contamination levels on the nearest portable water supply located in an unrestricted area, a tank containing radioactive liquid is postulated to fail. The acceptance criteria and methods used for the assessment follow the guidance in BTP 11-6 (Reference 17) and the radionuclide concentration limits in 10 CFR 20, Appendix B (Reference 3). In addition, the Interim Staff Guidance (ISG) DC/COL-ISG-013 (Reference 34) stipulates that the COL applicant is to identify the site-specific parameters for the evaluation (COL 11.2(14)). In the absence of site-specific requirements, the minimum dilution factors are calculated using 10 percent of 10 CFR 20, Appendix B, Table 2 (Reference 3) concentration limits and compared with the corresponding expected release radionuclide concentration.

In evaluating the postulated liquid-containing tank failure, the CVCS holdup tank is selected because it contains the highest amount of radioactive inventory among the liquid waste collection tanks installed in the yard area. In accordance with Section B.3 of BTP 11-6 (Reference 17), credit for liquid retention by the tank house surrounding the holdup tank is not taken.

The radionuclide inventory in the holdup tank is based on the expected fuel defect. The concentration of radioactive liquid after a liquid tank failure is assumed to be unmitigated and diluted by mixing in receiving water. The concentration is divided by 10 CFR 20 Appendix B (Reference 3) limits. Table 11.2-9 summarizes the results of this evaluation and identifies the minimum dilution factor as 1.71×10^3 to sufficiently dilute the failed tank nuclides to 10 percent of the 10 CFR 20 Appendix B (Reference 3) concentration limits. Site-specific hydrologic characteristics related to dilution of liquid tank failure source terms are described in Subsection 2.4.13.

The COL applicant is to provide the site-specific volume of the mixing water and hydrogeological data for analysis; the results of the analysis are to demonstrate that the potential groundwater or surface water contamination concentration resulting from radioactive release due to liquid-containing tank failure meets the requirements in 10 CFR 20, Appendix B, Table 2 (Reference 3) (COL 11.2(14)).

11.2.3.3 Offsite Dose Calculation Manual

The release of the treated liquid effluent is to follow the surveillance, control and operations requirements of the offsite dose calculation manual. The COL applicant is to prepare the site-specific ODCM in accordance with the Nuclear Energy Institute (NEI) 07-09A, Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description (Reference 35) (COL 11.2(1)).

11.2.4 Testing and Inspection Requirements

Preoperational testing is described in Section 14.2. Prior to installation, the R/O package is tested to verify that it is functioning properly. The system control panels are shop tested. The remainder of the system components are tested and inspected prior to shipment. After installation, but prior to initial plant startup, the LWMS is tested to verify pressure integrity, flow characteristics at design conditions, and the operability of valves, instrumentation, and controls. During commissioning and initial power operation, samples are taken on a batch basis to verify the load and decontamination efficiency of the R/O package. Instrumentation is recalibrated periodically. The inspection and testing are implemented to enable periodic evaluation of system operability and required performance in accordance with NRC RG 1.143 (Reference 1).

Epoxy coatings in cubicles that contain significant quantities of radioactive material are Service Level II coatings as defined in NRC RG 1.54 (Reference 16), and are subject to the limited QA provisions, selection, qualification, application, testing, maintenance and inspection provisions of NRC RG 1.54 (Reference 16) and standards referenced therein, as applicable to Service Level II coatings. Post-construction initial inspection is performed by personnel qualified using ASTM D 4537 (Reference 26) in accordance with inspection plan guidance of ASTM D 5163 (Reference 27).

11.2.5 Combined License Information

COL 11.2(1) The COL applicant is to prepare the site-specific ODCM in accordance with NEI 07-09A.

COL 11.2(2) The COL applicant is to prepare operational procedures and programs related to operations, inspection, calibration, and maintenance of the contamination control program.

COL 11.2(3) The COL applicant is to determine whether contaminated laundry is sent to an offsite facility for cleaning or for disposal.

COL 11.2(4) The COL applicant is to prepare and provide the P&IDs.

COL 11.2(5) The COL applicant is to perform a site-specific cost-benefit analysis following the guidance in the regulatory requirements of NRC RG 1.110.

COL 11.2(6) The COL applicant is to provide reasonable assurance that the mobile or temporary equipment and interconnections to plant systems conform with the regulatory requirements and guidance of 10 CFR 50.34a, 10 CFR 20.1406, NRC RG 1.143, and ANSI/ANS 40.37.

COL 11.2(7) The COL applicant is to develop the procedure for the collection and shipment of mixed wastes, if and when they are generated, for offsite treatment. The generation of mixed liquid wastes is minimized by process control and the controlled use of hazardous chemicals.

COL 11.2(8) The COL applicant is to develop the interface design and provide the site-specific information for the LWMS effluent discharge, including radioactive release points, effluent temperature, the design (type, shape, and size) of flow orifices, and the sampling requirements following the guidance of NRC RG 1.21 and RG 4.15 and the standards incorporated therein by reference.

COL 11.2(9) The COL applicant is to develop a plant-wide NRC RG 4.21 Program following the guidance in NEI 08-08A for contamination control.

COL 11.2(10) The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations and make them available for decommissioning planning and implementation.

COL 11.2(11) The COL applicant is to prepare the site process control program and the site radiological environmental monitoring program.

COL 11.2(12) The COL applicant is to confirm the assumed dilution flow rate provided by cooling tower blowdown, dilution pump, or other plant discharges at the discharge point based on site-specific parameters.

COL 11.2(13) The COL applicant is to calculate dose to members of the public following the guidance of NRC RG 1.109 and NRC RG 1.113 using site-specific parameters and to compare the doses due to the liquid effluents with the numerical design objectives of Appendix I to 10 CFR 50, 10 CFR 20.1302, and 40 CFR 190.

COL 11.2(14) The COL applicant is to perform an analysis to demonstrate that the potential groundwater or surface water contamination concentrations resulting from radioactive release from the liquid-containing tank failure, are in compliance with the limits in 10 CFR 20, Appendix B, Table 2.

11.2.6 References

1. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Rev. 2, U.S. Nuclear Regulatory Commission, November 2001.

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

3. 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," U.S. Nuclear Regulatory Commission.

4. 10 CFR Part 50, Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low as is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents,” U.S. Nuclear Regulatory Commission.

5. Regulatory Guide 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” Rev. 2, U.S. Nuclear Regulatory Commission, June 2009.

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Table 11.2-1 (1 of 2)

Expected Liquid Radioactive Effluents During Normal Operations, Including AOOs (Bq/yr)

Nuclide	Primary Coolant Shim Bleed	Liquid Radwaste System	SGBD System	Turbine Building Drains	Adjusted Total	Detergent Waste	Total[(1)]
Na-24	3.85E+05	5.55E+05	0.00E+00	3.42E+06	6.92E+07	0.00E+00	7.03E+07
P-32	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.66E+06	6.66E+06
Cr-51	2.25E+06	3.64E+05	0.00E+00	4.59E+05	4.88E+07	1.74E+08	2.22E+08
Mn-54	2.12E+06	2.06E+05	0.00E+00	2.32E+05	4.07E+07	1.41E+08	1.81E+08
Fe-55	1.67E+06	1.56E+05	0.00E+00	1.75E+05	3.16E+07	2.66E+08	3.00E+08
Fe-59	2.76E+05	3.66E+04	0.00E+00	4.26E+04	5.62E+06	8.14E+07	8.88E+07
Co-58	4.92E+06	5.74E+05	0.00E+00	6.73E+05	9.81E+07	2.92E+08	4.07E+08
Co-60	7.44E+05	6.88E+04	0.00E+00	7.84E+04	1.41E+07	5.18E+08	5.18E+08
Ni-63	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	6.29E+07	6.29E+07
Zn-65	6.66E+05	6.55E+04	0.00E+00	7.47E+04	1.28E+07	0.00E+00	1.30E+07
W-187	4.66E+04	5.07E+04	0.00E+00	2.31E+05	5.22E+06	0.00E+00	5.18E+06
Np-239	1.38E+05	1.04E+05	0.00E+00	2.63E+05	7.99E+06	0.00E+00	8.14E+06
Sr-89	1.36E+05	1.72E+04	0.00E+00	2.02E+04	2.75E+06	3.26E+06	5.92E+06
Sr-90	1.70E+04	1.56E+03	0.00E+00	1.75E+03	3.22E+05	4.81E+05	8.14E+05
Sr-91	2.99E+03	6.51E+03	0.00E+00	5.18E+04	9.69E+05	0.00E+00	9.62E+05
Y-91m	1.93E+03	4.22E+03	0.00E+00	3.30E+04	6.22E+05	0.00E+00	6.29E+05
Y-91	1.20E+04	1.44E+03	0.00E+00	9.21E+02	2.27E+05	3.11E+06	3.33E+06
Y-93	1.49E+04	3.06E+04	0.00E+00	2.28E+04	1.08E+06	0.00E+00	1.07E+06
Zr-95	4.07E+05	4.85E+04	0.00E+00	5.70E+04	8.10E+06	4.07E+07	4.81E+07
Nb-95	4.18E+05	3.74E+04	0.00E+00	3.92E+04	7.81E+06	7.03E+07	7.77E+07
Mo-99	4.96E+05	3.44E+05	0.00E+00	7.99E+05	2.60E+07	2.22E+06	2.81E+07
Tc-99m	4.74E+05	3.26E+05	0.00E+00	5.03E+05	2.06E+07	0.00E+00	2.07E+07
Ru-103	6.55E+06	9.07E+05	0.00E+00	1.10E+06	1.35E+08	1.07E+07	1.44E+08
Rh-103m	6.55E+06	9.07E+05	0.00E+00	1.08E+06	1.35E+08	0.00E+00	1.37E+08
Ru-106	1.21E+08	1.16E+07	0.00E+00	1.32E+07	2.31E+09	3.29E+08	2.63E+09
Rh-106	1.21E+08	1.16E+07	0.00E+00	1.32E+07	2.31E+09	0.00E+00	2.29E+09
Ag-110m	1.70E+06	1.67E+05	0.00E+00	1.89E+05	3.26E+07	4.44E+07	7.77E+07
Ag-110	2.21E+05	2.17E+04	0.00E+00	2.46E+04	4.26E+06	0.00E+00	4.07E+06
Sb-124	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.59E+07	1.59E+07

Table 11.2-1 (2 of 2)

Nuclide	Primary Coolant Shim Bleed	Liquid Radwaste System	SGBD System	Turbine Building Drains	Adjusted Total	Detergent Waste	Total(1)
Te-129m	1.54E+05	2.27E+04	0.00E+00	2.76E+04	3.24E+06	0.00E+00	3.26E+06
Te-129	9.88E+04	1.67E+04	0.00E+00	2.99E+04	2.31E+06	0.00E+00	2.29E+06
Te-131m	4.00E+04	3.89E+04	0.00E+00	1.52E+05	3.65E+06	0.00E+00	3.66E+06
Te-131	7.25E+03	7.10E+03	0.00E+00	2.77E+04	6.66E+05	0.00E+00	6.66E+05
I-131	6.29E+04	2.04E+06	0.00E+00	3.65E+05	3.92E+07	5.92E+07	9.99E+07
Te-132	1.58E+05	1.01E+05	0.00E+00	2.14E+05	7.51E+06	0.00E+00	7.40E+06
I-132	1.63E+05	7.51E+05	0.00E+00	7.73E+05	2.68E+07	0.00E+00	2.66E+07
I-133	4.00E+04	4.55E+06	0.00E+00	3.25E+06	1.25E+08	0.00E+00	1.26E+08
I-134	1.27E-03	3.23E+04	0.00E+00	2.53E+04	9.14E+05	0.00E+00	9.25E+05
Cs-134	4.48E+05	2.62E+05	0.00E+00	6.62E+03	1.13E+07	4.07E+08	4.07E+08
I-135	6.29E+03	2.26E+06	0.00E+00	3.18E+06	8.62E+07	0.00E+00	8.51E+07
Cs-136	3.10E+06	4.96E+06	0.00E+00	1.51E+05	1.30E+08	1.37E+07	1.44E+08
Cs-137	6.55E+05	3.77E+05	0.00E+00	9.73E+03	1.65E+07	5.92E+08	5.92E+08
Ba-137m	6.14E+05	3.52E+05	0.00E+00	9.10E+03	1.54E+07	0.00E+00	1.55E+07
Ba-140	5.29E+06	1.36E+06	0.00E+00	1.81E+06	1.34E+08	3.37E+07	1.67E+08
La-140	6.48E+06	1.90E+06	0.00E+00	2.95E+06	1.80E+08	0.00E+00	1.81E+08
Ce-141	1.19E+05	1.79E+04	0.00E+00	2.16E+04	2.50E+06	8.51E+06	1.11E+07
Ce-143	8.55E+04	8.03E+04	0.00E+00	2.87E+05	7.18E+06	0.00E+00	7.03E+06
Pr-143	1.24E+05	2.36E+04	0.00E+00	3.85E+03	2.39E+06	0.00E+00	2.41E+06
Ce-144	5.29E+06	5.14E+05	0.00E+00	5.70E+05	1.01E+08	1.44E+08	2.44E+08
Pr-144	5.29E+06	5.14E+05	0.00E+00	5.70E+05	1.01E+08	0.00E+00	9.99E+07
Others	2.52E+04	1.97E+03	0.00E+00	2.84E+02	4.37E+05	0.00E+00	4.44E+05
Total (Except Tritium)	3.00E+08	4.85E+07	0.00E+00	5.03E+07	6.33E+09	3.32E+09	9.62E+09
H-3							5.40E+13

(1) The total release effluents include 5.92×10^9 Bq/yr considering AOOs.

Table 11.2-2 (1 of 3)

PWR-GALE Code Input Parameters
Used to Calculate Annual Gaseous and Liquid Effluent Releases

Card No	Parameter	Value
1	Name of Reactor	APR1400 NRC (PWR)
2	Thermal Power Level (MWt)	4.06E+03
3	Mass of Primary Coolant (10^6 lb)	6.43E+02
4	Primary System Letdown Rate (gpm)	8.00E+01
5	Letdown Cation Demineralizer Flow (gpm)	0.00E+00
6	Number of Steam Generators	2.00E+00
7	Total Steam Flow (10^6 lb/hr)	1.80E+01
8	Mass of Liquid in Each Steam Generator (10^3 lb)	2.18E+02
9	Blowdown Flow (10^3 lb/hr) Blowdown Treatment Input Option	3.590E+01 0
10	Condensate Demineralizer Regeneration Time (Days)	0.00E+00
11	Condensate Demineralizer Flow Fraction	1.67E-01
12	Shim Bleed Flow rate (gpd) Fraction of primary coolant activity (PCA)	 6.05E+02 1.0
13	DF (Iodine) DF (Cs) DF (Others)	1.0E+05 4.0E+03 1.0E+05
14	Collection Time (Days) Process Time (Days) Fraction Discharged	5.9E+01 8.5E-01 1.0
15	Equipment Drains Flow rate (gpd) Fraction of primary coolant activity (PCA)	 2.50E+02 1.0
16	DF (Iodine) DF (Cs) DF (Others)	1.0E+05 2.0E+03 1.0E+04
17	Collection Time (Days) Process Time (Days) Fraction Discharged)	5.9E+01 8.5E-01 1.0

Table 11.2-2 (2 of 3)

Card No	Parameter	Value
18	Clean Waste Flow rate (gpd) Fraction of primary coolant activity (PCA)	 8.660E+2 0.144
19	DF (Iodine) DF (Cs) DF (Others)	1.0E+04 2.0E+03 1.0E+05
20	Collection Time (Days) Process Time (Days) Fraction Discharged	8.3E+00 1.7E-01 1.0
21	Dirty Waste Flow rate (gpd) Fraction of primary coolant activity (PCA)	 1.406E+03 0.075
22	DF (Iodine) DF (Cs) DF (Others)	1.0E+04 2.0E+03 1.0E+05
23	Collection Time (Days) Process Time (Days) Fraction Discharged	7.6E+00 1.7E-01 1.0
24	Fraction of Blowdown Flow Processed	1.0E+00
25	DF (Iodine) DF (Cs) DF (Others)	1.0E+02 1.0E+01 1.0E+01
26	Collection Time (Days) Process Time (Days) Fraction Discharged	0.0E+00 0.0E+00 0.0
27	Regenerant Flow Rate (gpd)	3.40E+03
28	DF (Iodine) DF (Cs) DF (Others)	1.0E+00 1.0E+00 1.0E+00

Table 11.2-2 (3 of 3)

Card No	Parameter	Value
29	Collection Time (Days) Process Time (Days) Fraction Discharged	0.0E+00 0.0E+00 0.0
30	Is There Continuous Stripping of Full Letdown Flow	0
31	Holdup Time for Xenon (Days)	4.50E+01
32	Holdup Time for Krypton (Days)	3.50E+00
33	Fill Time of Decay Tanks for the Gas Stripper (Days)	0.00E+00
34	Gas Waste System HEPA Efficiency (%)	99.0
35	Fuel Handling Area Filter Efficiency (%) Charcoal HEPA	 0.0 99.0
36	Auxiliary Building Filter Efficiency (%) Charcoal HEPA	 90.0 99.0
37	Containment Volume (10^6 ft^3)	3.13E+00
38	Containment Atmosphere Cleanup System Charcoal efficiency (%) HEPA efficiency (%) Flow rate (10^3 cfm)	 0.0 0.0 0.0
39	Containment High Volume Purge System Charcoal efficiency (%) HEPA efficiency (%) Number/yr	 0.0 99.0 2.0
40	Containment Low Volume Purge System Charcoal efficiency (%) HEPA efficiency (%) Flow rate (cfm)	 90.0 99.0 1,500
41	Fraction of Iodine Released from Blowdown Tank Vent	0.00E+00
42	Percent of Iodine Removed from Air Ejector Release	0.00E+00
43	Partition Factor of Detergent Waste	1.00E+00

Table 11.2-3 (1 of 2)

Decontamination Factors for CVCS and LWMS

Shim Bleed Decontamination Factors

Component	Nuclide		
	Iodine	Cs, Rb	Others
CVCS purification IX	1	1	1
CVCS pre-holdup IX	10	2	10
Boric acid concentrator	1	1	1
Boric acid condensate IX	1	1	1
LWMS R/O module	10	10	10
LWMS cation IX	1	10	10
LWMS mixed IX	100	2	100
LWMS mixed IX	10	10	10
Sum	1×10^5	4×10^3	1×10^5 (1)

(1) For conservatism, 10 % of total decontamination factor is applied.

Equipment Drain Decontamination Factors

Component	Nuclide		
	Iodine	Cs, Rb	Others
CVCS purification IX	N/A	N/A	N/A
CVCS pre-holdup IX	10	1	1
Boric acid concentrator	1	1	1
Boric acid condensate IX	1	1	1
LWMS R/O module	10	10	10
LWMS cation IX	1	10	10
LWMS mixed IX	100	2	100
LWMS mixed IX	10	10	10
Sum	1×10^5	2×10^3	1×10^4(1)

(1) For conservatism, 10 % of total decontamination factor is applied.

Table 11.2-3 (2 of 2)

Clean and Dirty Waste Decontamination Factors

Component	Nuclide		
	Iodine	Cs, Rb	Others
LWMS R/O module	10	10	10
LWMS cation IX	1	10	10
LWMS mixed IX	100	2	100
LWMS mixed IX	10	10	10
Sum	1×10^4	2×10^3	1×10^5

Table 11.2-4

Input Parameters Used for LADTAP II Code

Parameter	Value	Basis
Water type selection	Freshwater	Assumed that the liquid effluents are discharged to river or lake
Liquid effluent discharge rate (L/min)	3.79×10^4	See description in Subsection 11.2.3.1
Shore-width factor	0.2	Table A-2 in RG 1.109
Reconcentration model index	0	Assumed that there is no reconcentration
Source terms	See Table 11.2-1	Based on PWR-GALE code calculation
Dilution factor for aquatic food and boating	5	For DC application, a conservative dilution factor of 5.0 is assumed for evaluating individual doses from liquid effluents.
Dilution factor for shoreline and swimming	5	
Dilution factor for drinking water	5	
Dilution factor for irrigation water usage location for the current food product	5	
Irrigation rate ($L/m^2 \cdot month$)	41.68	Average value (0.5 m/yr) used in RESRAD code analysis (Reference 36)
Fraction of animal feed not produced with contaminated irrigation water	0	Assumed for conservative evaluation
Fraction of animal drinking water not obtained from contaminated irrigation water	0	
Transit time for any exposure pathway	0	
Midpoint of plant life (years)	30	Half of the APR1400 design life of 60 years
Other parameters	Default values in RG 1.109	Since there is no site-specific information for DC application, default values in RG 1.109 are used.

Table 11.2-5 (1 of 2)

Individual Doses from Liquid Effluents (mSv/yr)

Age Group	Skin	Bone	Liver	T. Body	Thyroid	Kidney	Lung	GI-LLI
Fish								
ADULT	-	1.54E-02	1.53E-02	1.12E-02	3.37E-04	5.14E-03	1.77E-03	4.47E-03
TEEN	-	1.65E-02	1.57E-02	6.62E-03	2.89E-04	5.20E-03	2.02E-03	3.29E-03
CHILD	-	2.08E-02	1.37E-02	2.99E-03	2.76E-04	4.38E-03	1.60E-03	1.30E-03
Drinking								
ADULT	-	2.07E-04	6.68E-03	6.61E-03	6.85E-03	6.53E-03	6.44E-03	7.55E-03
TEEN	-	1.98E-04	4.77E-03	4.63E-03	4.89E-03	4.63E-03	4.55E-03	5.35E-03
CHILD	-	5.71E-04	9.18E-03	8.80E-03	9.60E-03	8.90E-03	8.73E-03	9.47E-03
INFANT	-	5.93E-04	9.14E-03	8.60E-03	9.97E-03	8.75E-03	8.58E-03	9.03E-03
Shoreline Activity								
ADULT	2.13E-05	1.81E-05	1.81E-05	1.81E-05	1.81E-05	1.81E-05	1.81E-05	1.81E-05
TEEN	1.19E-04	1.01E-04	1.01E-04	1.01E-04	1.01E-04	1.01E-04	1.01E-04	1.01E-04
CHILD	2.48E-05	2.12E-05	2.12E-05	2.12E-05	2.12E-05	2.12E-05	2.12E-05	2.12E-05
Irrigated food: Vegetables								
ADULT	-	4.12E-06	5.01E-05	4.88E-05	4.84E-05	4.72E-05	4.55E-05	6.65E-05
TEEN	-	6.85E-06	6.34E-05	5.86E-05	6.00E-05	5.86E-05	5.59E-05	8.22E-05
CHILD	-	1.63E-05	1.01E-04	9.02E-05	9.73E-05	9.31E-05	8.86E-05	1.09E-04
Irrigated food: Leafy Vegetables								
ADULT	-	4.12E-06	5.01E-05	4.88E-05	4.84E-05	4.72E-05	4.55E-05	6.65E-05
TEEN	-	6.85E-06	6.34E-05	5.86E-05	6.00E-05	5.86E-05	5.59E-05	8.22E-05
CHILD	-	1.63E-05	1.01E-04	9.02E-05	9.73E-05	9.31E-05	8.86E-05	1.09E-04
Irrigated food: Milk								
ADULT	-	2.78E-06	4.93E-05	4.81E-05	4.71E-05	4.64E-05	4.54E-05	4.52E-05
TEEN	-	4.65E-06	6.20E-05	5.78E-05	5.82E-05	5.73E-05	5.58E-05	5.52E-05
CHILD	-	1.11E-05	9.89E-05	8.93E-05	9.37E-05	9.09E-05	8.83E-05	8.72E-05

Table 11.2-5 (2 of 2)

Age Group	Skin	Bone	Liver	T. Body	Thyroid	Kidney	Lung	GI-LLI
Irrigated Food: Meat								
ADULT	-	8.86E-06	4.65E-05	4.71E-05	4.59E-05	6.02E-05	4.51E-05	5.44E-04
TEEN	-	1.52E-05	5.75E-05	5.77E-05	5.63E-05	8.12E-05	5.53E-05	6.93E-04
CHILD	-	3.74E-05	9.13E-05	9.23E-05	8.99E-05	1.32E-04	8.76E-05	5.97E-04
Sum of All Pathways								
ADULT	2.13E-05	1.56E-02	2.22E-02	1.80E-02	7.39E-03	1.19E-02	8.41E-03	1.28E-02
TEEN	1.19E-04	1.68E-02	2.08E-02	1.16E-02	5.51E-03	1.02E-02	6.89E-03	9.65E-03
CHILD	2.48E-05	2.15E-02	2.33E-02	1.22E-02	1.03E-02	1.37E-02	1.07E-02	1.17E-02
INFANT	0.00E+00	5.93E-04	9.14E-03	8.60E-03	9.97E-03	8.75E-03	8.58E-03	9.03E-03

Table 11.2-6 (1 of 6)

Equipment List in the LWMS

Tanks

Equipment Characteristic	Description
Equipment name Quantity (each) Design capacity, L (gal) Design pressure, kg/cm^2G (psig) Design temperature, ° C (° F) Material Radwaste Safety Class	Floor drain tank 2 68,137 (18,000) ATM 93.3 (200) Stainless steel RW-IIa
Equipment name Quantity (each) Design capacity, L (gal) Design pressure, kg/cm^2G (psig) Design temperature, ° C (° F) Material Radwaste Safety Class	Equipment waste tank 2 68,137 (18,000) ATM 93.3 (200) Stainless steel RW-IIa
Equipment name Quantity (each) Design capacity, L (gal) Design pressure, kg/cm^2G (psig) Design temperature, ° C (° F) Material Radwaste Safety Class	Chemical waste tank 2 [34,069 (9,000)] ATM 93.3 (200) Stainless steel RW-IIc
Equipment name Quantity (each) Design capacity, L (gal) Design pressure, kg/cm^2G (psig) Design temperature, ° C (° F) Material Radwaste Safety Class	Monitor tank 2 [102,206 (27,000)] ATM 93.3 (200) Stainless steel RW-IIc

Table 11.2-6 (2 of 6)

Tanks (cont.)

Equipment Characteristic	Description
Equipment name Quantity (each) Design capacity, L (gal) Design pressure, kg/cm^{2}G (psig) Design temperature, ° C (° F) Material Radwaste Safety Class	Acid storage tank 1 1,703 (450) ATM 100 (212) Stainless steel N/A(1)
Equipment name Quantity (each) Design capacity, L (gal) Design pressure, kg/cm^{2}G (psig) Design temperature, ° C (° F) Material Radwaste Safety Class	Acid batch tank 1 189 (50) ATM 100 (212) Incoloy 825 N/A(1)
Equipment name Quantity (each) Design capacity, L (gal) Design pressure, kg/cm^{2}G (psig) Design temperature, ° C (° F) Material Radwaste Safety Class	Caustic storage tank 1 1,703 (450) ATM 100 (212) Stainless steel N/A(1)
Equipment name Quantity (each) Design capacity, L (gal) Design pressure, kg/cm^{2}G (psig) Design temperature, ° C (° F) Material Radwaste Safety Class	Seal water storage tank 1 1,741 (460) ATM 93.3 (200) Stainless steel RW-IIc
Equipment name Quantity (each) Design capacity, L (gal) Design pressure, kg/cm^{2}G (psig) Design temperature, ° C (° F) Material Radwaste Safety Class	Caustic batch tank 1 189 (50) ATM 100 (212) Stainless steel N/A(1)

(1) The equipment classified as N/A is non-radwaste.

Table 11.2-6 (3 of 6)

Tanks (cont.)

Equipment Characteristic	Description
Equipment name Quantity (each) Design capacity, L (gal) Design pressure, kg/cm^{2}G (psig) Design temperature, ° C (° F) Material Radwaste Safety Class	Chemical additive tank 1 416 (110) ATM 93.3 (200) Stainless steel N/A[(1)]
Equipment name Quantity (each) Design capacity, L (gal) Design pressure, kg/cm^{2}G (psig) Design temperature, ° C (° F) Material Radwaste Safety Class	Detergent waste tank 2 22,712 (6,000) ATM 93.3 (200) Stainless steel RW-IIc

Pumps

Equipment Characteristic	Description
Equipment name Quantity (each) Design capacity, L/min (gpm) Design process flow rate, L/min (gpm) Design temperature, ° C (° F) Material Radwaste Safety Class	Floor drain pump 2 (100 % capacity per each unit) 568 (150) per each unit 227 (60) per each unit 93.3 (200) Stainless steel RW-IIc
Equipment name Quantity (each) Design capacity, L/min (gpm) Design process flow rate, L/min (gpm) Design temperature, ° C (° F) Material Radwaste Safety Class	Equipment waste pump 2 (100 % capacity per each unit) 568 (150) per each unit 227 (60) per each unit 93.3 (200) Stainless steel RW-IIc

(1) The equipment classified as N/A is non-radioactive.

Table 11.2-6 (4 of 6)

Pumps (cont.)

Equipment Characteristic	Description
Equipment name Quantity (each) Design capacity, L/min (gpm) Design process flow rate, L/min (gpm) Design temperature, ° C (° F) Material Radwaste Safety Class	Chemical waste pump 2 (100 % capacity per each unit) 416 (110) per each unit 227 (60) per each unit 93.3 (200) Stainless steel RW-IIc
Equipment name Quantity (each) Design capacity, L/min (gpm) Design process flow rate, L/min (gpm) Design temperature, ° C (° F) Material Radwaste Safety Class	Monitor tank pump 2 (100 % capacity per each unit) 1,060 (280) per each unit 341 (90) per each unit 93.3 (200) Stainless steel RW-IIc
Equipment name Quantity (each) Design capacity, L/min (gpm) Design temperature, ° C (° F) Material Radwaste Safety Class	Seal water pump 2 (100 % capacity per each unit) 227 (60) per each unit 93.3 (200) Stainless steel RW-IIc
Equipment name Quantity (each) Design capacity, L/min (gpm) Design temperature, ° C (° F) Material Radwaste Safety Class	Acid batch pump 1 19 (5) 100 (212) Incoloy 825 N/A(1)
Equipment name Quantity (each) Design capacity, L/min (gpm) Design temperature, ° C (° F) Material Radwaste Safety Class	Caustic batch pump 1 19 (5) 100 (212) Stainless steel N/A(1)
Equipment name Quantity (each) Design capacity, L/min (gpm) Design temperature, ° C (° F) Material Radwaste Safety Class	Chemical additive pump 1 38 (10) 93.3 (200) Stainless steel N/A(1)

(1) The equipment classified as N/A is non-radioactive.

Table 11.2-6 (5 of 6)

Pumps (cont.)

Equipment Characteristic	Description
Equipment name	Detergent waste tank pump
Quantity (each)	2 (100 % capacity per each unit)
Design capacity, L/min (gpm)	568 (150)
Design process flow rate, L/min (gpm)	189 (50)
Design temperature, ° C (° F)	93.3 (200)
Material	Stainless steel
Radwaste Safety Class	RW-IIc

Miscellaneous

Equipment Characteristic	Description
Equipment name	Detergent waste filter
Quantity (each)	1
Design capacity, L/min (gpm)	189 (50)
Design pressure, kg/cm^2G (psig)	ATM
Design temperature, ° C (° F)	93.3 (200)
Material	Stainless steel
Radwaste Safety Class	RW-IIc
Equipment name	LRS seal water heat exchanger
Quantity (for both units)	1
Type	Shell and tube
Heat transfer, kcal/hr (Btu/hr)	2.5×10^4 (1.0×10^{-5})
Design pressure, kg/cm^2G (psig)	14.1 (200)
Design temperature, ° C (° F)	93.3 (200)
Material	Stainless steel
Radwaste Safety Class	RW-IIc

Table 11.2-6 (6 of 6)

Miscellaneous (cont.)

Equipment Characteristic	Description
Equipment name	R/O package
Quantity (each)	100 % capacity 2 train
Design capacity, L/min (gpm)	227 (60)
Design process flow rate, L/min (gpm)	227 (60)
Design pressure, kg/cm^2G (psig)	14.1 (200)
Design temperature, ° C (° F)	93.3 (200)
Material	Stainless steel
Radwaste Safety Class(1)	
MF membrane	RW-IIc
R/O feed tank	RW-IIb
R/O feed pump	RW-IIb
R/O module	RW-IIa
IX feed tank	RW-IIc
IX feed pump	RW-IIc
Cation bed	RW-IIa
Mixed bed 1	RW-IIa
Mixed bed 2	RW-IIc
Concentrate holding tank	RW-IIa

(1) The R/O package is subject to vendor design and the Radwaste Safety Class for components within the R/O package may require adjustment for vendor-specific design.

Table 11.2-7

Codes and Standards for Equipment in the LWMS

Equipment	Design and Fabrication	Material	Welder Qualifications and Procedures	Inspection and Testing
Tanks: Atmospheric or 0 – 1.05 kg/cm^2 (0 – 15 psig) (steel)	 API 650 (atmospheric) API 620 (0 – 1.05 kg/cm^2 [0 – 15 psig])	ASME Sec. II	ASME Sec. IX	 API 650 (atmospheric) API 620 (0 –1 .05 kg/cm^2 [0 – 15 psig])
Pressure Vessels	ASME Sec. VIII, Div. 1 or Div. 2	ASME Sec. II	ASME Sec. IX	ASME Sec. VIII, Div. 1 or Div. 2
Pumps	API-610; API-674; API-675; ASME Sec. VIII, Div. 1 or Div. 2	ASME Sec. II	ASME Sec. IX	ASME Sec. III, Class 3
Piping and Valves	ASME B31.3	ASME Sec. II	ASME Sec. IX	ASME B31.3
Ion Exchangers	ASME Sec. VIII, Div. 1	ASME Sec. II	ASME Sec. IX	ASME Sec. VIII, Div. 1
Filters	ASME Sec. VIII, Div. 1	ASME Sec. II	ASME Sec. IX	ASME Sec. VIII, Div. 1

Table 11.2-8

Radioactive Atmospheric Tank Overflow Protection

Tanks	Level Monitoring	Potential Overflow Alarm	Method for Containing Overflow
Equipment waste tank	Radwaste control room	Radwaste control room	Overflow from one radwaste tank flows to the other radwaste tank with the ultimate overflow of both tanks directed to the compound building sump.
Floor drain tanks	Radwaste control room	Radwaste control room	Same as above
Chemical waste tanks	Radwaste control room	Radwaste control room	Same as above
Monitor tanks	Radwaste control room	Radwaste control room	Same as above

Table 11.2-9 (1 of 2)

Radioactive Concentrations in Nearest Portable Water
Due to Liquid Waste Containing Tank Failure

Nuclide	Holdup Tank Inventory (Bq/cm^3)	Concentration at Nearest Potable Water (1) (Bq/cm^3)	10 CFR 20, Appendix B (Bq/cm^3)	Ratio
Br-84	8.81E-05	5.14E-08	1.48E+01	3.47E-09
I-129(2)	1.41E-09	8.23E-13	7.40E-03	1.11E-10
I-131	2.26E-02	1.32E-05	3.70E-02	3.57E-04
I-132	1.64E-03	9.54E-07	3.70E+00	2.58E-07
I-133	1.38E-02	8.07E-06	2.59E-01	3.12E-05
I-134	9.43E-04	5.51E-07	1.48E+01	3.72E-08
I-135	5.79E-03	3.38E-06	1.11E+00	3.04E-06
Rb-88	2.83E-03	1.65E-06	1.48E+01	1.12E-07
Cs-134	8.81E-04	5.14E-07	3.33E-02	1.54E-05
Cs-136	9.43E-03	5.51E-06	2.22E-01	2.48E-05
Cs-137	1.32E-03	7.71E-07	3.70E-02	2.08E-05
Na-24	2.20E-02	1.28E-05	1.85E+00	6.94E-06
Cr-51	9.43E-02	5.51E-05	1.85E+01	2.98E-06
Mn-54	8.18E-02	4.77E-05	1.11E+00	4.30E-05
Fe-55	6.29E-02	3.67E-05	3.70E+00	9.92E-06
Fe-59	1.13E-02	6.61E-06	3.70E-01	1.79E-05
Co-58	1.95E-01	1.14E-04	7.40E-01	1.54E-04
Co-60	2.96E-02	1.73E-05	1.11E-01	1.55E-04
Zn-65	2.64E-02	1.54E-05	1.85E-02	8.33E-05
Sr-89	5.47E-03	3.19E-06	2.96E-01	1.08E-05
Sr-90	6.92E-04	4.04E-07	1.85E-02	2.18E-05
Sr-91	2.58E-04	1.50E-07	7.40E-01	2.03E-07
Y-91m	4.72E-03	2.75E-06	7.40E+01	3.72E-08
Y-91	2.01E-02	1.17E-05	2.96E-01	3.97E-05
Y-93	5.16E-01	3.01E-04	7.40E-01	4.07E-04
Zr-95	1.64E-02	9.54E-06	7.40E-01	1.29E-05
Nb-95	9.43E-03	5.51E-06	1.11E+00	4.96E-06

Table 11.2-9 (2 of 2)

Nuclide	Holdup Tank Inventory (Bq/cm^3)	Concentration at Nearest Potable Water (1) (Bq/cm^3)	10 CFR 20, Appendix B (Bq/cm^3)	Ratio
Mo-99	2.26E-02	1.32E+05	7.40E-01	1.79E-05
TC-99m	6.92E-04	4.04E-07	3.70E+01	1.09E-08
Tc-99(2)	1.65E-11	9.68E-15	2.22E+00	4.36E-15
Ru-103	2.70E-01	1.58E-04	1.11E+00	1.42E-04
Ru-106	4.78E+00	2.79E-03	1.11E-01	2.51E-02
Ag-110m	6.92E-02	4.04E-05	2.22E-01	1.82E-04
Te-129m	5.91E-03	3.45E-06	2.59E-01	1.33E-05
Te-129	2.96E-04	1.73E-07	1.48E+01	1.17E-08
Te-131m	1.26E-03	7.34E-07	2.96E-01	2.48E-06
Te-131	3.33E-05	1.95E-08	2.96E+00	6.57E-09
Te-132	6.16E-03	3.60E-06	3.33E-01	1.08E-05
Ba-137m	1.32E-03	7.71E-07	-	-
Ba-140	2.39E-01	1.39E-04	2.96E-01	4.71E-04
La-140	4.47E-02	2.61E-05	3.33E-01	7.83E-05
Ce-141	4.97E-03	2.90E-06	1.11E+00	2.61E-06
Ce-143	3.77E-03	2.20E-06	7.40E-01	2.98E-06
Ce-144	2.08E-01	1.21E-04	1.11E-01	1.09E-03
W-187	2.20E-03	1.28E-06	1.11E+00	1.16E-06
Np-239	6.16E-03	3.60E-06	7.40E-01	4.86E-06
H-3	4.53E+03	2.79E-03	1.11E-01	2.51E-02
			SUM	1.00E-01

(1) Dilution factor of 1.71E+03 is assumed to meet the 10 % of 10 CFR 20, Appendix B limits.

(2) In accordance with NRC's position, two radionuclides of I-129 and Tc-99, which may cause significant potential exposure, are included in the liquid tank failure analysis. The inventories of I-129 and Tc-99 in the holdup tank are determined based on the assumption that they have the same RCS existence ratios to I-131 and Tc-99m with those for a similar PWR. As shown in the results, the impact of these nuclides on the dose contribution is negligible.

Table 11.2-10 (1 of 2)

Design Basis Liquid Effluent Concentration at the Site Boundary

Nuclide	Design Basis Release(1) (Bq/yr)	Effluent Concentration (Bq/m^3)	10 CFR 20, Appendix B Limits (Bq/m^3)	Ratio
Na-24	7.03E+07	3.53E+00	1.85E+06	1.91E-06
P-32	6.66E+06	3.35E-01	3.33E+05	1.01E-06
Cr-51	9.61E+08	4.83E+01	1.85E+07	2.61E-06
Mn-54	1.81E+08	9.11E+00	1.11E+06	8.21E-06
Fe-55	3.00E+08	1.51E+01	3.70E+06	4.07E-06
Fe-59	8.88E+07	4.46E+00	3.70E+05	1.21E-05
Co-58	4.07E+08	2.05E+01	7.40E+05	2.76E-05
Co-60	5.18E+08	2.60E+01	1.11E+05	2.35E-04
Ni-63	6.29E+07	3.16E+00	3.70E+06	8.54E-07
Zn-65	1.30E+07	6.51E-01	1.85E+05	3.52E-06
W-187	5.18E+06	2.60E-01	1.11E+06	2.35E-07
Np-239	8.14E+06	4.09E-01	7.40E+05	5.53E-07
Sr-89	1.34E+08	6.74E+00	2.96E+05	2.28E-05
Sr-90	1.47E+07	7.38E-01	1.85E+04	3.99E-05
Sr-91	4.89E+06	2.46E-01	7.40E+05	3.32E-07
Y-91m	3.99E+06	2.01E-01	7.40E+03	2.71E-05
Y-91	2.96E+08	1.49E+01	2.96E+05	5.03E-05
Y-93	1.07E+06	5.39E-02	7.40E+05	7.29E-08
Zr-95	7.24E+07	3.64E+00	7.40E+05	4.92E-06
Nb-95	1.39E+08	6.97E+00	1.11E+06	6.28E-06
Mo-99	1.21E+09	6.06E+01	7.40E+05	8.19E-05
Tc-99m	7.55E+08	3.79E+01	3.70E+07	1.03E-06
Ru-103	1.44E+08	7.25E+00	1.11E+06	6.53E-06
Rh-103m	1.37E+08	6.88E+00	2.22E+08	3.10E-08
Ru-106	2.63E+09	1.32E+02	1.11E+05	1.19E-03
Rh-106	2.29E+09	1.15E+02	-	-
Ag-110m	7.77E+07	3.91E+00	2.22E+05	1.76E-05

Table 11.2-10 (2 of 2)

Nuclide	Design Basis Release(1) (Bq/yr)	Effluent Concentration (Bq/m^3)	10 CFR 20, Appendix B Limits (Bq/m^3)	Ratio
Ag-110	4.07E+06	2.05E-01	-	-
Sb-124	1.59E+07	8.00E-01	2.59E+05	3.09E-06
Te-129m	5.74E+07	2.89E+00	2.59E+05	1.11E-05
Te-129	2.29E+06	1.15E-01	1.48E+07	7.79E-09
Te-131m	6.78E+07	3.41E+00	2.96E+05	1.15E-05
Te-131	1.01E+06	5.09E-02	7.40E+02	6.88E-05
I-131	9.99E+09	5.02E+02	3.70E+04	1.36E-02
Te-132	8.35E+08	4.20E+01	3.33E+05	1.26E-04
I-132	3.08E+08	1.55E+01	3.70E+06	4.18E-06
I-133	1.71E+10	8.57E+02	2.59E+05	3.31E-03
I-134	4.05E+06	2.04E-01	1.48E+07	1.38E-08
Cs-134	3.47E+09	1.74E+02	3.33E+04	5.24E-03
I-135	3.08E+09	1.55E+02	1.11E+06	1.40E-04
Cs-136	7.09E+09	3.56E+02	2.22E+05	1.60E-03
Cs-137	5.92E+10	2.98E+03	3.70E+04	8.04E-02
Ba-137m	1.55E+07	7.81E-01	-	-
Ba-140	1.67E+08	8.37E+00	2.96E+05	2.83E-05
La-140	1.81E+08	9.11E+00	3.33E+05	2.74E-05
Ce-141	1.11E+07	5.58E-01	1.11E+06	5.03E-07
Ce-143	7.03E+06	3.53E-01	7.40E+05	4.77E-07
Pr-143	2.41E+06	1.21E-01	7.40E+05	1.63E-07
Ce-144	2.44E+08	1.23E+01	1.11E+05	1.11E-04
Pr-144	9.99E+07	5.02E+00	2.22E+07	2.26E-07
H-3	5.40E+13	2.72E+06	3.70E+07	7.34E-02
Pr-144	9.99E+07	5.02E+00	2.22E+07	2.26E-07
H-3	5.40E+13	2.72E+06	3.70E+07	7.34E-02
			SUM	1.80E-01

(1) Design basis release rate is adjusted from expected liquid radioactive effluents (Table 11.2-1) using multiplication factors that are the ratios of design basis primary coolant activity to expected activity.

Table 11.2-11 (1 of 2)

Expected Radioactive Source Terms for LWMS Tanks (Bq/cm^3)

Nuclide	Equipment Waste Tank	Floor Drain Tank	Chemical Waste Tank	Monitor Tank
Br-84	1.91E+02	2.63E+02	5.97E+00	2.71E-03
I-131	2.63E+01	3.62E+01	8.23E-01	3.23E-02
I-132	7.26E+02	9.99E+02	2.27E+01	4.67E-02
I-133	3.33E+02	4.58E+02	1.04E+01	1.71E-01
I-134	1.20E+03	1.65E+03	3.74E+01	2.77E-02
I-135	6.82E+02	9.37E+02	2.13E+01	1.22E-01
Rb-88	2.26E+03	3.11E+03	7.07E+01	8.96E-02
Cs-134	5.09E-01	7.00E-01	1.59E-02	3.53E-03
Cs-136	1.18E+01	1.63E+01	3.70E-01	7.57E-02
Cs-137	7.26E-01	9.99E-01	2.27E-02	5.04E-03
Na-24	5.79E+02	7.96E+02	1.81E+01	2.25E-02
Cr-51	3.94E+01	5.41E+01	1.23E+00	5.26E-03
Mn-54	2.03E+01	2.79E+01	6.34E-01	2.81E-03
Fe-55	1.52E+01	2.09E+01	4.75E-01	2.11E-03
Fe-59	3.81E+00	5.24E+00	1.19E-01	5.16E-04
Co-58	5.82E+01	8.01E+01	1.82E+00	7.96E-03
Co-60	6.72E+00	9.24E+00	2.10E-01	9.32E-04
Zn-65	6.46E+00	8.89E+00	2.02E-01	8.93E-04
Sr-89	1.77E+00	2.44E+00	5.54E-02	2.41E-04
Sr-90	1.52E-01	2.09E-01	4.75E-03	2.11E-05
Sr-91	1.17E+01	1.61E+01	3.67E-01	3.02E-04
Y-91m	5.50E+00	7.57E+00	1.72E-01	1.89E-04
Y-91	6.59E-02	9.06E-02	2.06E-03	1.83E-05
Y-93	5.15E+01	7.08E+01	1.61E+00	1.40E-03
Zr-95	4.93E+00	6.78E+00	1.54E-01	6.73E-04

Table 11.2-11 (2 of 2)

Nuclide	Equipment Waste Tank	Floor Drain Tank	Chemical Waste Tank	Monitor Tank
Nb-95	3.55E+00	4.88E+00	1.11E-01	4.98E-04
Mo-99	8.03E+01	1.10E+02	2.51E+00	7.75E-03
Tc-99m	5.73E+01	7.88E+01	1.79E+00	7.12E-03
Ru-103	9.50E+01	1.31E+02	2.97E+00	1.28E-02
Ru-106	1.14E+03	1.57E+03	3.57E+01	1.58E-01
Ag-110m	1.65E+01	2.27E+01	5.15E-01	2.28E-03
Te-129m	2.41E+00	3.31E+00	7.52E-02	3.24E-04
Te-129	2.87E+02	3.95E+02	8.97E+00	1.08E-03
Te-131m	1.87E+01	2.57E+01	5.84E-01	1.23E-03
Te-131	9.18E+01	1.26E+02	2.87E+00	3.23E-04
Te-132	2.14E+01	2.94E+01	6.68E-01	2.17E-03
Ba-137m	7.26E-01	9.99E-01	2.27E-02	4.71E-03
Ba-140	1.64E+02	2.26E+02	5.14E+00	2.10E-02
La-140	3.13E+02	4.30E+02	9.77E+00	3.38E-02
Ce-141	1.90E+00	2.61E+00	5.94E-02	2.55E-04
Ce-143	3.49E+01	4.80E+01	1.09E+00	2.44E-03
Ce-144	5.06E+01	6.95E+01	1.58E+00	6.99E-03
W-187	3.10E+01	4.27E+01	9.70E-01	1.76E-03

Table 11.2-12 (1 of 3)

Expected Radioactive Source Terms for Other LWMS Components (Bq)

Nuclide	Reverse Osmosis	Cation Bed	Mixed Bed 1	Mixed Bed 2
Na-24	2.14E+11	2.15E+10	2.36E+09	2.15E+07
Cr-51	1.25E+11	1.25E+10	1.38E+09	1.25E+07
Mn-54	7.11E+10	7.11E+09	7.82E+08	7.11E+06
Fe-55	5.36E+10	5.36E+09	5.90E+08	5.36E+06
Co-58	1.98E+11	1.98E+10	2.17E+09	1.98E+07
Fe-59	1.26E+10	1.26E+09	1.39E+08	1.26E+06
Co-60	2.37E+10	2.37E+09	2.61E+08	2.37E+06
Zn-65	2.26E+10	2.26E+09	2.49E+08	2.26E+06
Br-84	2.68E+09	3.75E+07	3.11E+08	2.86E+06
Rb-88	1.87E+10	2.09E+09	1.36E+08	1.05E+08
Sr-89	5.92E+09	5.93E+08	6.52E+07	5.93E+05
Y-89m	5.92E+05	5.92E+04	6.52E+03	5.92E+01
Sr-90	5.38E+08	5.38E+07	5.92E+06	5.38E+04
Y-90m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90	3.24E+08	3.24E+07	3.56E+06	3.24E+04
Sr-91	2.80E+09	2.81E+08	3.09E+07	2.81E+05
Y-91m	1.76E+09	1.76E+08	1.94E+07	1.76E+05
Y-91	4.85E+08	4.85E+07	5.34E+06	4.85E+04
Y-93	1.30E+10	1.30E+09	1.43E+08	1.30E+06
Zr-93	1.32E+02	1.32E+01	1.45E+00	1.32E-02
Nb-93m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.67E+10	1.67E+09	1.83E+08	1.67E+06
Nb-95m	1.69E+08	1.69E+07	1.86E+06	1.69E+04
Nb-95	1.29E+10	1.29E+09	1.42E+08	1.29E+06
Mo-99	1.17E+11	1.17E+10	1.29E+09	1.17E+07
Tc-99m	1.09E+11	1.09E+10	1.20E+09	1.09E+07
Tc-99	6.36E+03	6.36E+02	6.99E+01	6.36E-01
Ru-103	3.12E+11	3.12E+10	3.43E+09	3.12E+07

Table 11.2-12 (2 of 3)

Nuclide	Reverse Osmosis	Cation Bed	Mixed Bed 1	Mixed Bed 2
Rh-103m	3.10E+11	3.10E+10	3.41E+09	3.10E+07
Ru-106	4.01E+12	4.01E+11	4.41E+10	4.01E+08
Rh-106m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rh-106	4.01E+12	4.01E+11	4.41E+10	4.01E+08
Ag-110m	5.76E+10	5.76E+09	6.34E+08	5.76E+06
Ag-110	7.49E+08	7.49E+07	8.24E+06	7.49E+04
Te-129m	7.80E+09	7.81E+08	8.59E+07	7.81E+05
Te-129	1.33E+10	1.36E+09	1.49E+08	1.36E+06
I-129	1.16E+01	1.16E+00	1.28E-01	1.16E-03
Te-131m	1.37E+10	1.37E+09	1.51E+08	1.37E+06
Te-131	3.49E+09	3.58E+08	3.92E+07	3.58E+05
I-131	7.23E+10	6.59E+08	7.30E+09	6.64E+07
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	3.45E+10	3.46E+09	3.80E+08	3.46E+06
I-132	7.61E+10	3.57E+09	5.04E+09	4.59E+07
I-133	1.72E+11	6.54E+07	1.89E+10	1.72E+08
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	2.67E+10	2.35E+08	3.04E+09	2.78E+07
Cs-134	1.79E+09	1.79E+08	9.97E+06	8.97E+06
I-135	1.13E+11	1.34E+08	1.25E+10	1.14E+08
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-136	3.35E+10	3.36E+09	1.87E+08	1.68E+08
Cs-137	2.57E+09	2.57E+08	1.43E+07	1.29E+07
Ba-137m	2.40E+09	2.40E+08	1.34E+07	1.20E+07
Ba-140	4.65E+11	4.65E+10	5.11E+09	4.65E+07

Table 11.2-12 (3 of 3)

Nuclide	Reverse Osmosis	Cation Bed	Mixed Bed 1	Mixed Bed 2
La-140	6.53E+11	6.53E+10	7.19E+09	6.53E+07
Ce-141	6.15E+09	6.15E+08	6.77E+07	6.15E+05
Ce-143	2.80E+10	2.80E+09	3.08E+08	2.80E+06
Pr-143	8.09E+09	8.09E+08	8.89E+07	8.09E+05
Ce-144	1.77E+11	1.77E+10	1.95E+09	1.77E+07
Pr-144	1.77E+11	1.77E+10	1.94E+09	1.77E+07
W-187	1.82E+10	1.82E+09	2.00E+08	1.82E+06
Np-239	3.53E+10	3.53E+09	3.89E+08	3.53E+06

Table 11.2-13 (1 of 3)

Design Basis Radioactive Source Terms for LWMS Tanks and Pumps (Bq/cm^3) (1 % Fuel Defect)

Nuclide	Equipment Waste Tank	Floor Drain Tank	Chemical Waste Tank	Monitor Tank	Detergent Waste Tank	Seal Water Storage Tank	Equipment Waste Pump(Bq)	Floor Drain Pump (Bq)
Br-84	2.49E+02	3.42E+02	7.77E+00	3.52E-03	2.88E-02	3.36E-02	6.92E+06	9.50E+06
I-131	3.20E+04	4.40E+04	9.99E+02	3.89E+01	3.70E+00	4.32E+00	8.89E+08	1.22E+09
I-132	8.51E+03	1.17E+04	2.66E+02	7.67E-01	9.85E-01	1.15E+00	2.36E+08	3.25E+08
I-133	4.51E+04	6.20E+04	1.41E+03	2.32E+01	5.22E+00	6.08E+00	1.25E+09	1.72E+09
I-134	5.34E+03	7.35E+03	1.67E+02	1.24E-01	6.19E-01	7.21E-01	1.48E+08	2.04E+08
I-135	2.49E+04	3.42E+04	7.77E+02	4.45E+00	2.88E+00	3.36E+00	6.92E+08	9.50E+08
Rb-88	2.37E+04	3.26E+04	7.40E+02	9.38E-01	2.74E+00	3.20E+00	6.59E+08	9.06E+08
Cs-134	4.51E+03	6.20E+03	1.41E+02	3.13E+01	5.22E-01	5.31E-01	1.25E+08	8.89E+07
Cs-136	6.05E+02	8.32E+02	1.89E+01	3.87E+00	7.00E-02	8.17E-02	1.68E+07	2.31E+07
Cs-137	5.22E+03	7.17E+03	1.63E+02	3.62E+01	6.04E-01	7.05E-01	1.45E+08	1.99E+08
Na-24	5.79E+02	7.96E+02	1.81E+01	2.25E-02	4.82E+00	7.81E-02	1.61E+07	2.21E+07
Cr-51	1.75E+02	2.41E+02	5.48E+00	2.34E-02	6.71E-02	2.36E-02	4.86E+06	6.70E+06
Mn-54	2.03E+01	2.79E+01	6.34E-01	2.81E-03	2.03E-02	3.44E-03	5.64E+05	7.75E+05
Fe-55	1.52E+01	2.09E+01	4.75E-01	2.11E-03	2.35E-03	2.58E-03	4.22E+05	5.81E+05
Fe-59	3.81E+00	5.24E+00	1.19E-01	5.16E-04	1.76E-03	5.14E-04	1.06E+05	1.46E+05
Co-58	5.82E+01	8.01E+01	1.82E+00	7.96E-03	4.41E-04	7.86E-03	1.62E+06	2.23E+06
Co-60	6.72E+00	9.24E+00	2.10E-01	9.32E-04	6.74E-03	9.06E-04	1.87E+05	2.57E+05

Table 11.2-13 (2 of 3)

Nuclide	Equipment Waste Tank	Floor Drain Tank	Chemical Waste Tank	Monitor Tank	Detergent Waste Tank	Seal Water Storage Tank	Equipment Waste Pump (Bq)	Floor Drain Pump (Bq)
Zn-65	6.46E+00	8.89E+00	2.02E-01	8.93E-04	7.78E-04	8.72E-04	1.79E+05	2.47E+05
Sr-89	4.16E+01	5.72E+01	1.30E+00	5.65E-03	7.48E-04	5.61E-03	1.16E+06	1.59E+06
Sr-90	2.84E+00	3.91E+00	8.88E-02	3.94E-04	4.82E-03	3.83E-04	7.89E+04	1.09E+05
Sr-91	6.14E+01	8.45E+01	1.92E+00	1.58E-03	3.29E-04	8.28E-03	1.71E+06	2.35E+06
Y-91m	3.55E+01	4.88E+01	1.11E+00	1.01E-03	7.11E-03	4.79E-03	9.86E+05	1.36E+06
Y-91	6.05E+00	8.32E+00	1.89E-01	8.73E-04	4.11E-03	8.16E-04	1.68E+05	2.31E+05
Y-93	1.42E+00	1.95E+00	4.44E-02	3.85E-05	7.00E-04	1.91E-04	3.95E+04	5.42E+04
Zr-95	7.68E+00	1.06E+01	2.40E-01	1.05E-03	1.64E-04	1.04E-03	2.13E+05	2.95E+05
Nb-95	6.53E+00	8.98E+00	2.04E-01	9.10E-04	7.56E-04	8.81E-04	1.81E+05	2.50E+05
Mo-99	3.55E+03	4.88E+03	1.11E+02	3.43E-01	4.11E-01	4.79E-01	9.86E+07	1.36E+08
Tc-99m	2.13E+03	2.93E+03	6.66E+01	3.09E-01	2.47E-01	2.88E-01	5.92E+07	8.14E+07
Ru-103	2.25E+00	3.09E+00	7.03E-02	3.04E-04	2.60E-04	3.03E-04	6.25E+04	8.59E+04
Ru-106	9.60E-01	1.32E+00	3.00E-02	1.33E-04	1.11E-04	1.29E-04	2.67E+04	3.67E+04
Ag-110m	1.65E+01	2.27E+01	5.15E-01	2.28E-03	1.91E-03	2.23E-03	4.58E+05	6.31E+05
Te-129m	7.58E+01	1.04E+02	2.37E+00	1.02E-02	8.78E-03	1.02E-02	2.11E+06	2.89E+06
Te-129	8.06E+01	1.11E+02	2.52E+00	6.63E-03	9.34E-03	1.09E-02	2.24E+06	3.08E+06
Te-131m	3.55E+02	4.88E+02	1.11E+01	2.34E-02	4.11E-02	4.79E-02	9.86E+06	1.36E+07
Te-131	1.42E+02	1.95E+02	4.44E+00	4.36E-03	1.64E-02	1.91E-02	3.95E+06	5.42E+06

Table 11.2-13 (3 of 3)

Nuclide	Equipment Waste Tank	Floor Drain Tank	Chemical Waste Tank	Monitor Tank	Detergent Waste Tank	Seal Water Storage Tank	Equipment Waste Pump (Bq)	Floor Drain Pump (Bq)
Te-132	2.49E+03	3.42E+03	7.77E+01	2.52E-01	2.88E-01	3.36E-01	6.92E+07	9.53E+07
Ba-137m	4.96E+03	6.82E+03	1.55E+02	3.38E+01	5.74E-01	6.70E-01	1.38E+08	1.89E+08
Ba-140	5.09E+01	7.00E+01	1.59E+00	6.49E-03	5.89E-03	6.87E-03	1.41E+06	1.94E+06
La-140	1.78E+01	2.44E+01	5.55E-01	4.30E-03	2.06E-03	2.39E-03	4.95E+05	6.78E+05
Ce-141	1.89E+00	2.60E+00	5.92E-02	2.54E-04	2.19E-04	2.55E-04	5.25E+04	7.22E+04
Ce-143	5.34E+00	7.35E+00	1.67E-01	3.74E-04	6.19E-04	7.21E-04	1.48E+05	2.04E+05
Ce-144	5.44E+00	7.48E+00	1.70E-01	7.52E-04	6.30E-04	7.34E-04	1.51E+05	2.08E+05
W-187	3.10E+01	4.27E+01	9.70E-01	1.76E-03	3.59E-03	4.19E-03	8.61E+05	1.19E+06
Np-239	2.76E+01	3.79E+01	8.62E-01	2.50E-03	3.19E-03	3.08E-03	7.67E+04	1.05E+06
Sum of Fractions								
$\sum A_i/A_{1i}$	1.20E+01	1.60E+01	1.90E-01	1.30E-02	8.27E-11	4.17E-05	5.00E-03	6.70E-03
$\sum A_i/A_{2i}$	1.60E+01	2.20E+01	2.50E-01	2.20E-02	1.09E-10	5.53E-05	6.60E-03	8.90E-03
Radwaste Classification								
	RW-IIa	RW-IIa	RW-IIc (1)	RW-IIc (1)	RW-IIc	RW-IIc	RW-IIc	RW-IIc

(1) The radwaste classifications for the chemical waste pump and the monitor pump are conservatively considered to be same as those for the corresponding tanks since the failure of the pumps would cause the same effect as that for the tanks.

Table 11.2-14 (1 of 4)

Design Basis Radioactive Source Terms for Other Miscellaneous LWMS Components (Bq) (1 % Fuel Defect)

Nuclide	MF Membrane	R/O Feed Tank	Reverse Osmosis	Cation Bed	IX Feed tank	Mixed Bed 1	Mixed Bed 2	Detergent Waste Filter	Concentrate Holding Tank
Na-24	1.47E+09	3.04E+09	2.14E+11	2.15E+10	1.67E+08	2.36E+09	2.15E+07	0.00E+00	4.35E+12
Cr-51	4.45E+08	9.21E+08	5.58E+11	5.58E+10	5.06E+07	6.14E+09	5.58E+07	2.28E+06	1.13E+13
Mn-54	5.15E+07	1.07E+08	7.11E+10	7.11E+09	5.86E+06	7.82E+08	7.11E+06	1.65E+06	1.44E+12
Fe-55	3.86E+07	7.98E+07	5.36E+10	5.36E+09	4.39E+06	5.90E+08	5.36E+06	1.59E+06	1.09E+12
Co-58	1.48E+08	3.06E+08	1.98E+11	1.98E+10	1.68E+07	2.17E+09	1.98E+07	1.88E+06	4.02E+12
Fe-59	9.66E+06	2.00E+07	1.26E+10	1.26E+09	1.10E+06	1.39E+08	1.26E+06	7.93E+04	2.56E+11
Co-60	1.71E+07	3.53E+07	2.37E+10	2.37E+09	1.94E+06	2.61E+08	2.37E+06	7.46E+05	4.82E+11
Zn-65	1.64E+07	3.39E+07	2.26E+10	2.26E+09	1.87E+06	2.49E+08	2.26E+06	4.77E+05	4.59E+11
Br-84	6.31E+08	1.31E+09	3.48E+09	4.88E+07	7.18E+07	4.05E+08	3.73E+06	0.00E+00	7.07E+10
Rb-88	6.01E+10	1.24E+11	1.96E+11	2.19E+10	6.83E+09	1.42E+09	1.10E+09	0.00E+00	3.98E+12
Sr-89	1.06E+08	2.18E+08	1.39E+11	1.39E+10	1.20E+07	1.53E+09	1.39E+07	0.00E+00	2.82E+12
Y-89m	0.00E+00	0.00E+00	1.39E+07	1.39E+06	0.00E+00	1.53E+05	1.39E+03	0.00E+00	2.82E+08
Sr-90	7.21E+06	1.49E+07	1.00E+10	1.01E+09	8.20E+05	1.11E+08	1.01E+06	0.00E+00	2.03E+11
Y-90m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Y-90	0.00E+00	0.00E+00	6.06E+09	6.06E+08	0.00E+00	6.66E+07	6.06E+05	0.00E+00	1.23E+11
Sr-91	1.56E+08	3.23E+08	1.47E+10	1.47E+09	1.77E+07	1.62E+08	1.47E+06	0.00E+00	2.99E+11
Y-91m	9.01E+07	1.87E+08	9.33E+09	9.36E+08	1.03E+07	1.03E+08	9.36E+05	0.00E+00	1.90E+10

Table 11.2-14 (2 of 4)

Nuclide	MF Membrane	R/O Feed Tank	Reverse Osmosis	Cation Bed	IX Feed Tank	Mixed Bed 1	Mixed Bed 2	Detergent Waste Filter	Concentrate Holding Tank
Y-91	1.53E+07	3.18E+07	2.17E+10	2.17E+09	1.75E+06	2.39E+08	2.17E+06	0.00E+00	4.41E+11
Y-93	3.61E+06	7.46E+06	3.58E+08	3.59E+07	4.10E+05	3.95E+06	3.59E+04	0.00E+00	7.27E+09
Zr-93	0.00E+00	0.00E+00	3.63E+00	3.63E-01	0.00E+00	3.99E-02	3.63E-04	0.00E+00	7.38E+01
Nb-93m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	1.95E+07	4.03E+07	2.60E+10	2.60E+09	2.22E+06	2.86E+08	2.60E+06	2.26E+05	5.28E+11
Nb-95m	0.00E+00	0.00E+00	2.63E+08	2.63E+07	0.00E+00	2.90E+06	2.63E+04	0.00E+00	5.34E+09
Nb-95	1.66E+07	3.43E+07	2.33E+10	2.33E+09	1.88E+06	2.57E+08	2.33E+06	0.00E+00	4.73E+11
Mo-99	9.01E+09	1.87E+10	5.17E+12	5.17E+11	1.03E+09	5.69E+10	5.17E+08	0.00E+00	1.05E+14
Tc-99m	5.41E+09	1.12E+10	4.77E+12	4.77E+11	6.15E+08	5.25E+10	4.77E+08	0.00E+00	9.69E+13
Tc-99	0.00E+00	0.00E+00	2.77E+05	2.77E+04	0.00E+00	3.04E+03	2.77E+01	0.00E+00	5.63E+06
Ru-103	5.71E+06	1.18E+07	7.39E+09	7.39E+08	6.49E+05	8.13E+07	7.39E+05	0.00E+00	1.50E+11
Rh-103m	0.00E+00	0.00E+00	7.34E+09	7.34E+08	0.00E+00	8.07E+07	7.34E+05	0.00E+00	1.49E+11
Ru-106	2.44E+06	5.04E+06	3.37E+09	3.37E+08	2.77E+05	3.71E+07	3.37E+05	0.00E+00	6.85E+10
Rh-106m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rh-106	0.00E+00	0.00E+00	3.37E+09	3.37E+08	0.00E+00	3.70E+07	3.37E+05	0.00E+00	6.85E+10
Ag-110m	4.18E+07	8.65E+07	5.76E+10	5.76E+09	4.76E+06	6.34E+08	5.76E+06	0.00E+00	1.17E+12
Ag-110	0.00E+00	0.00E+00	7.49E+08	7.49E+07	0.00E+00	8.24E+06	7.49E+04	0.00E+00	1.52E+10
Te-129m	1.92E+08	3.98E+08	2.46E+11	2.46E+10	2.19E+07	2.71E+09	2.46E+07	0.00E+00	5.00E+12
Te-129	2.05E+08	4.23E+08	1.59E+11	1.59E+10	2.33E+07	1.75E+09	1.59E+07	0.00E+00	3.23E+12

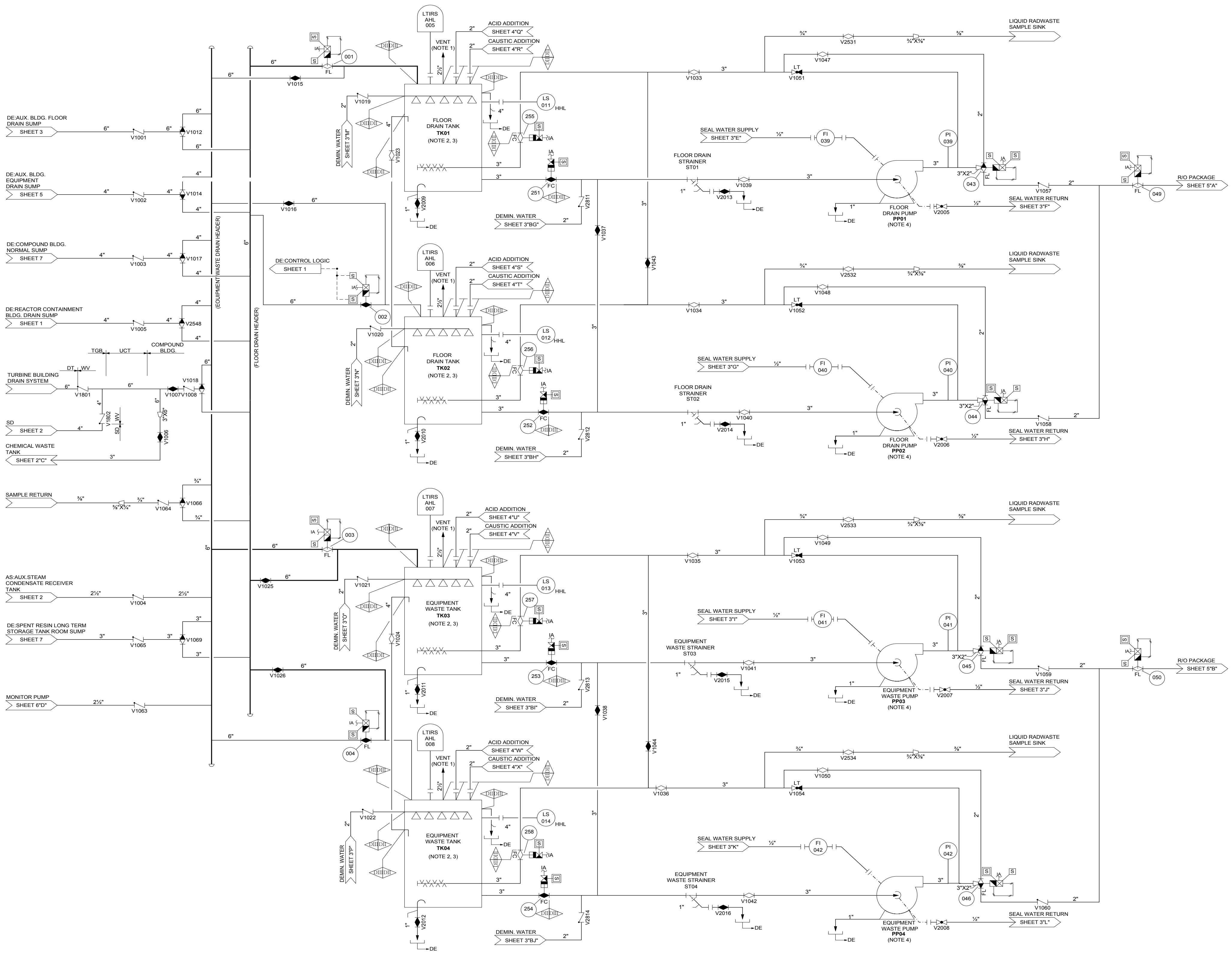
Table 11.2-14 (3 of 4)

Nuclide	MF Membrane	R/O Feed Tank	Reverse Osmosis	Cation Bed	IX Feed Tank	Mixed Bed 1	Mixed Bed 2	Detergent Waste Filter	Concentrate Holding Tank
I-129	0.00E+00	0.00E+00	1.23E+02	1.23E+01	0.00E+00	1.35E+00	1.23E-02	0.00E+00	2.50E+03
Te-131m	9.01E+08	1.87E+09	2.60E+11	2.60E+10	1.03E+08	2.86E+09	2.60E+07	0.00E+00	5.28E+12
Te-131	3.61E+08	7.46E+08	4.83E+10	4.84E+09	4.10E+07	5.32E+08	4.84E+06	0.00E+00	9.81E+11
I-131	8.11E+10	1.68E+11	7.99E+13	1.78E+10	9.23E+09	8.78E+12	7.98E+10	0.00E+00	1.62E+15
Xe-131m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-132	6.31E+09	1.31E+10	4.02E+12	4.02E+11	7.18E+08	4.42E+10	4.02E+08	0.00E+00	8.17E+13
I-132	2.16E+10	4.47E+10	4.48E+12	4.01E+11	2.46E+09	9.85E+10	8.97E+08	0.00E+00	9.10E+13
I-133	1.14E+11	2.37E+11	2.33E+13	8.87E+09	1.30E+10	2.57E+12	2.34E+10	0.00E+00	4.73E+14
Xe-133m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	1.36E+10	2.81E+10	1.19E+11	1.05E+09	1.54E+09	1.36E+10	1.24E+08	0.00E+00	2.42E+12
Cs-134	1.14E+10	2.37E+10	1.59E+13	1.59E+12	1.30E+09	8.84E+10	7.95E+10	0.00E+00	3.23E+14
I-135	6.31E+10	1.31E+11	4.12E+12	4.88E+09	7.18E+09	4.55E+11	4.14E+09	0.00E+00	8.37E+13
Xe-135m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-135	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-136	1.53E+09	3.18E+09	1.71E+12	1.71E+11	1.75E+08	9.53E+09	8.57E+09	0.00E+00	3.47E+13
Cs-137	1.32E+10	2.74E+10	1.84E+13	1.85E+12	1.51E+09	1.03E+11	9.23E+10	0.00E+00	3.76E+13
Ba-137m	1.26E+10	2.60E+10	1.73E+13	1.73E+12	1.43E+09	9.59E+10	8.62E+10	0.00E+00	3.52E+13

Table 11.2-14 (4 of 4)

Nuclide	MF Membrane	R/O Feed Tank	Reverse Osmosis	Cation Bed	IX Feed Tank	Mixed Bed 1	Mixed Bed 2	Detergent Waste Filter	Concentrate Holding Tank
Ba-140	1.29E+08	2.67E+08	1.44E+11	1.44E+10	1.47E+07	1.58E+09	1.44E+07	0.00E+00	2.93E+11
La-140	4.51E+07	9.33E+07	1.26E+11	1.26E+10	5.13E+06	1.39E+09	1.26E+07	0.00E+00	2.56E+11
Ce-141	4.81E+06	9.95E+06	6.13E+09	6.13E+08	5.47E+05	6.75E+07	6.13E+05	0.00E+00	1.25E+10
Ce-143	1.36E+07	2.81E+07	4.28E+09	4.29E+08	1.54E+06	4.72E+07	4.29E+05	0.00E+00	8.70E+10
Pr-143	0.00E+00	0.00E+00	1.24E+09	1.24E+08	0.00E+00	1.36E+07	1.24E+05	0.00E+00	2.52E+10
Ce-144	1.38E+07	2.86E+07	1.90E+10	1.91E+09	1.57E+06	2.10E+08	1.91E+06	0.00E+00	3.86E+11
Pr-144	0.00E+00	0.00E+00	1.90E+10	1.90E+09	0.00E+00	2.09E+08	1.90E+06	0.00E+00	3.86E+11
W-187	7.88E+07	1.63E+08	1.82E+10	1.82E+09	8.96E+06	2.00E+08	1.82E+06	0.00E+00	3.70E+11
Np-239	7.00E+07	1.45E+08	3.53E+10	3.53E+09	7.96E+06	3.89E+08	3.53E+06	0.00E+00	7.17E+11
Sum of Fractions									
$\sum A_i/A_{1i}$	4.50E-01	9.40E-01	1.20E+02	6.20E+00	5.20E-02	8.00E+00	2.50E-01	2.40E-05	2.50E+03
$\sum A_i/A_{2i}$	6.00E-01	1.20E+00	2.20E+02	9.20E+00	6.80E-02	1.80E+01	4.50E-01	2.40E-05	4.60E+03
Radwaste Classification									
	RW-IIc	RW-IIb (1)	RW-IIa	RW-IIa	RW-IIc (1)	RW-IIa	RW-IIc	RW-IIc	RW-IIa

(1) The radwaste calssifiactions for the R/O Feed Pump and the IX Feed Pump are conservatively considered to be same as those for the corresponding tanks since the failure of the pumps would cause the same effect as that for the tanks.

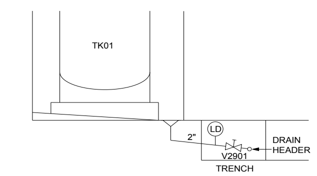


NOTES

1. RADWASTE TANK VENT LINES SHALL TERMINATE NEAR THE INLET TO EXHAUST OPENINGS THAT ARE PART OF THE EXHAUST DUCTWORK OF THE COMPOUND BUILDING HVAC SYSTEM (VB). (COOR : B7, D7, F7, H7)

2. FLOOR DRAIN TANKS AND EQUIPMENT WASTE TANKS INCLUDING DRAIN LINES AND INSTRUMENTATION ARE SEISMIC CATEGORY Ⅱ.
(COOR : A7, C7,E7, G7)

3. LEAK DETECTION PROVISION IS PROVIDED IN THE TANK ROOM AS BELOWS.



TK NO.	VLV NO.
TK01	V2901
TK02	V2902
TK03	V2903
TK04	V2904

4. PUMP SEAL WATER PROVISIONS ARE PUMP-SPECIFIC.

Figure 11.2-1 Liquid Radwaste System Flow Diagram (1 of 7)

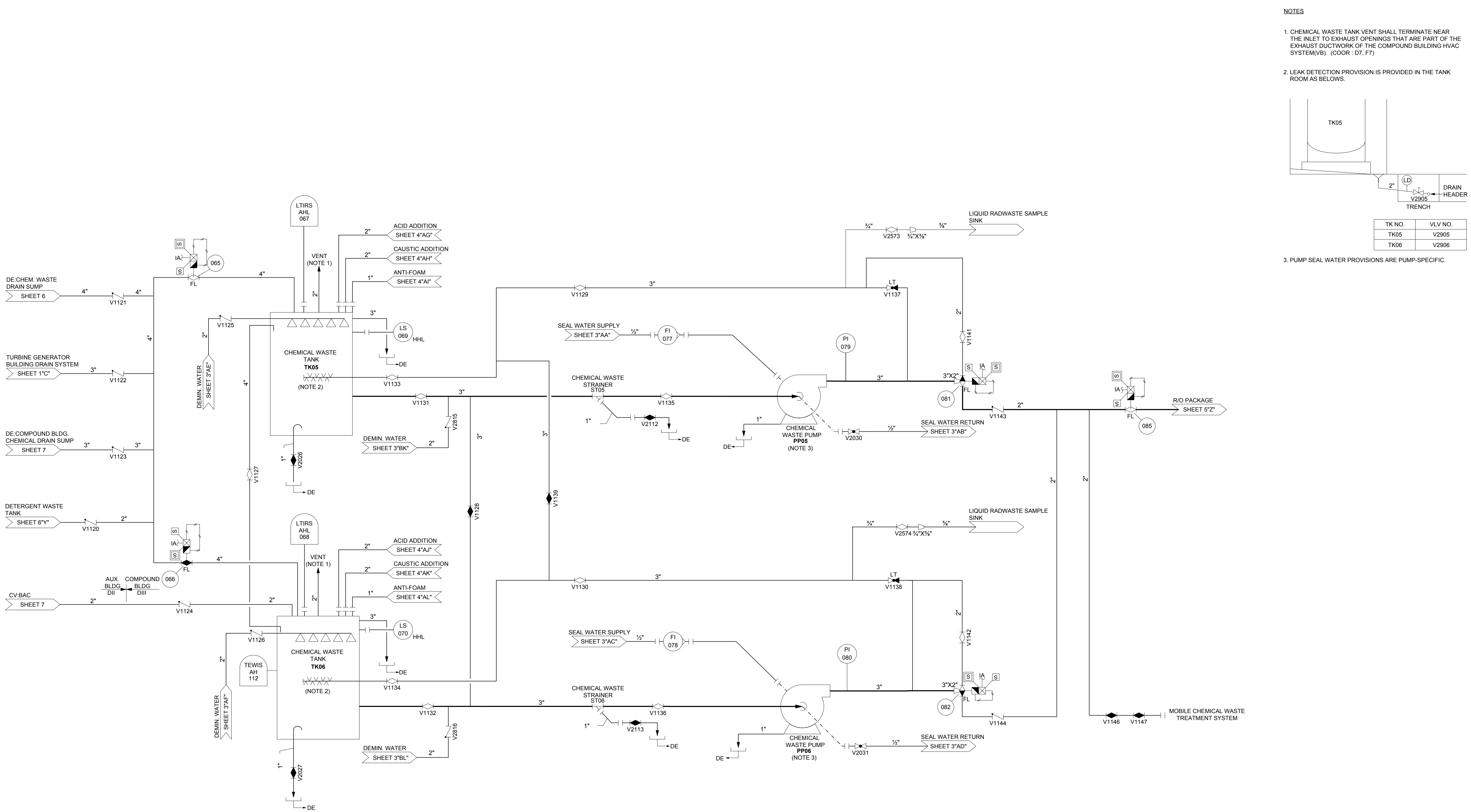


Figure 11.2-1 Liquid Radwaste System Flow Diagram (2 of 7)

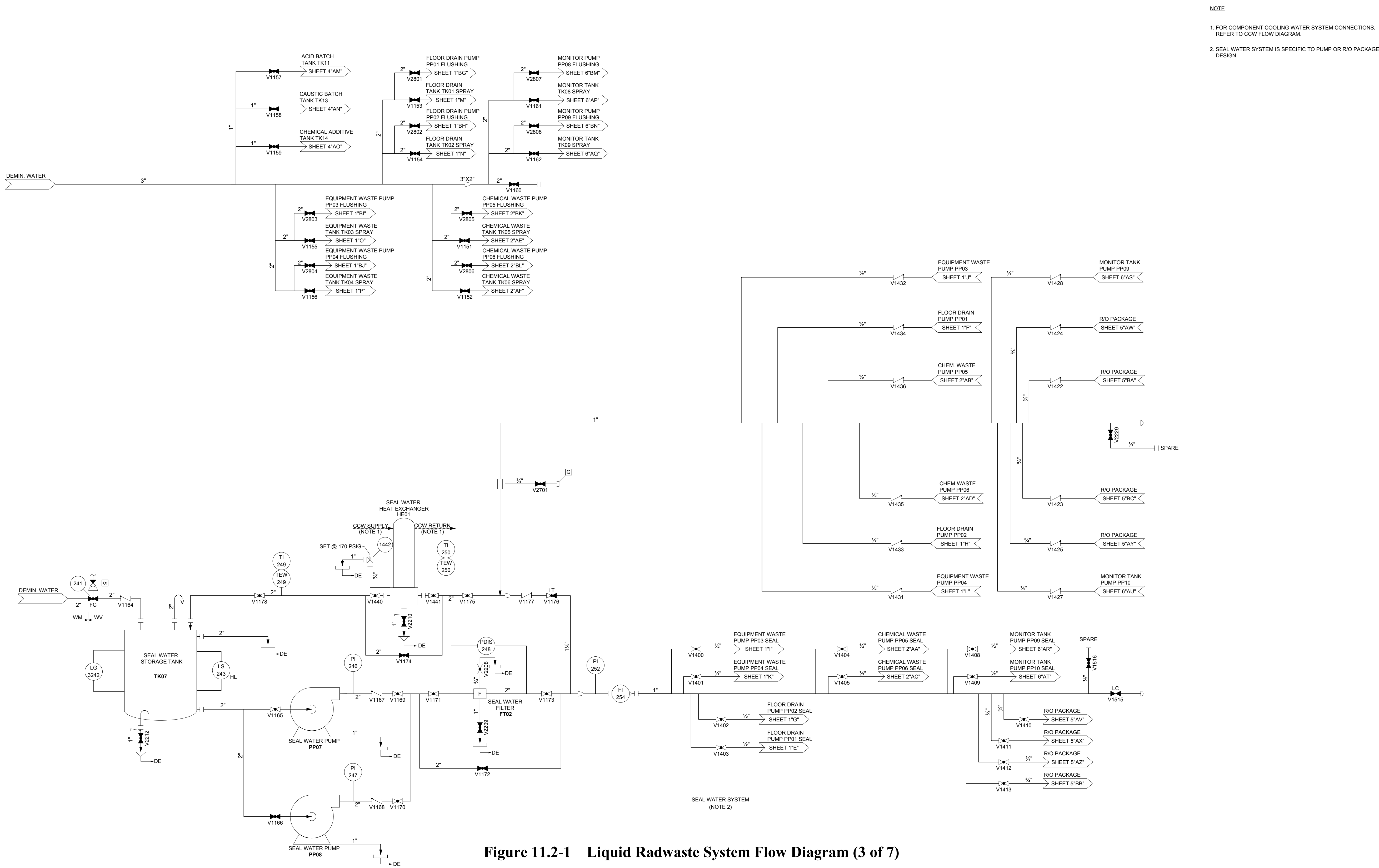


Figure 11.2-1   Liquid Radwaste System Flow Diagram (3 of 7)

NOTES

1. ANNUNCIATOR WILL BE MOUNTED NEAR TRUCK FILL AREA.

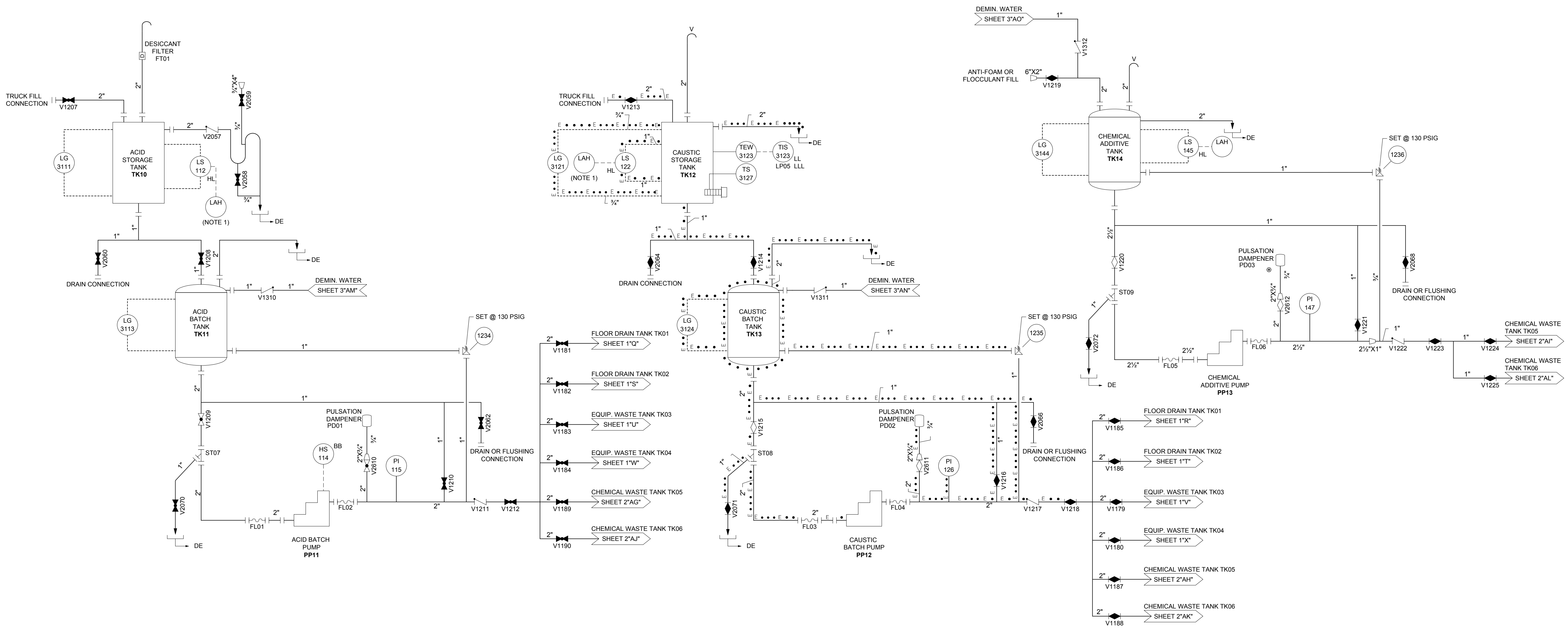


Figure 11.2-1   Liquid Radwaste System Flow Diagram (4 of 7)

NOTES

1. R/O MODULE,CATION BED AND MIXED BEDS ARE SEISMIC CATEGORY Ⅲ UNLESS OTHERWISE SHOWN. (COOR : C4, F4, C5, F5, C6, F6, C7, F7)

2. PUMP SEAL WATER PROVISIONS ARE SPECIFIC TO R/O PACKAGE DESIGN.

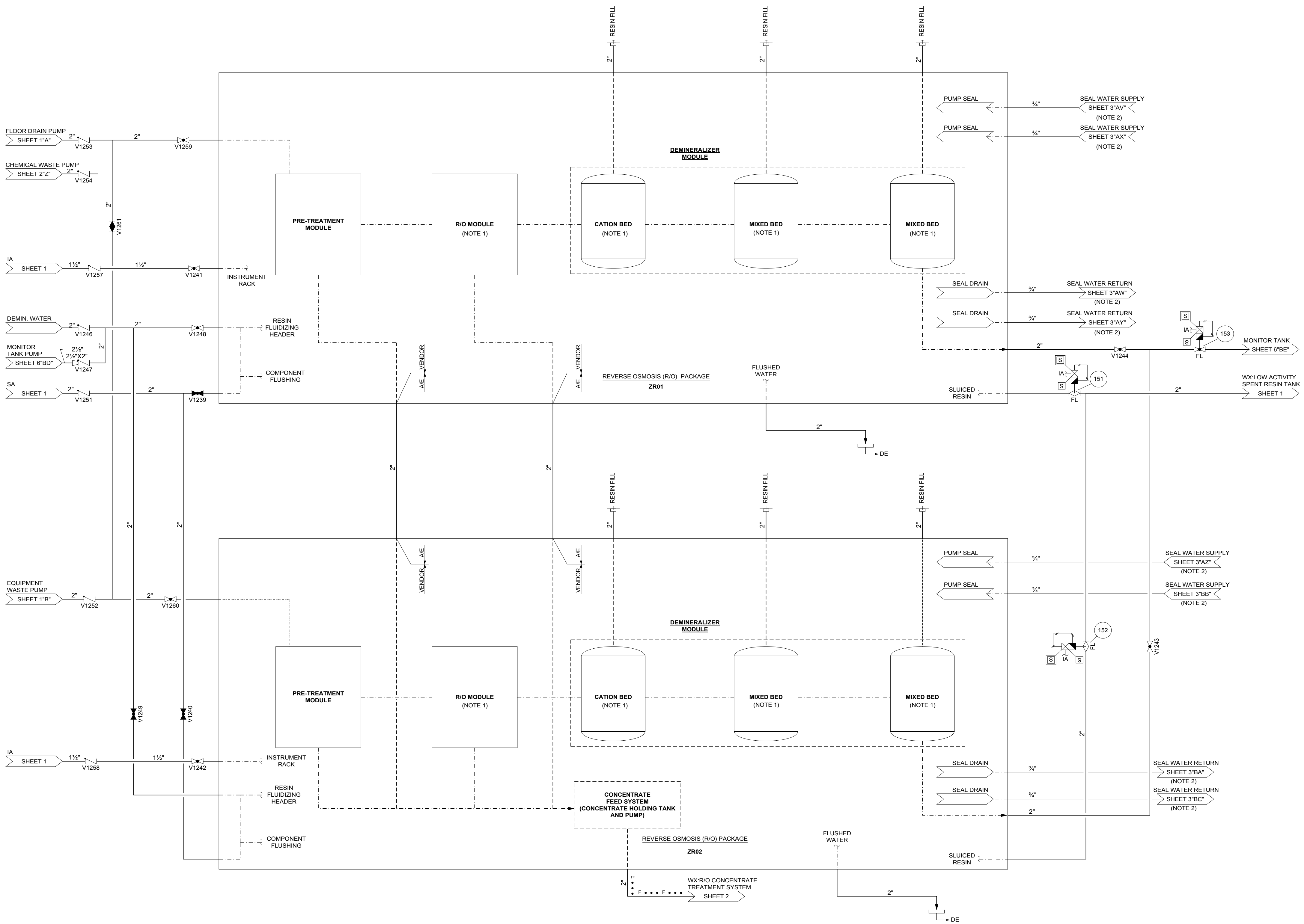


Figure 11.2-1 Liquid Radwaste System Flow Diagram (5 of 7)

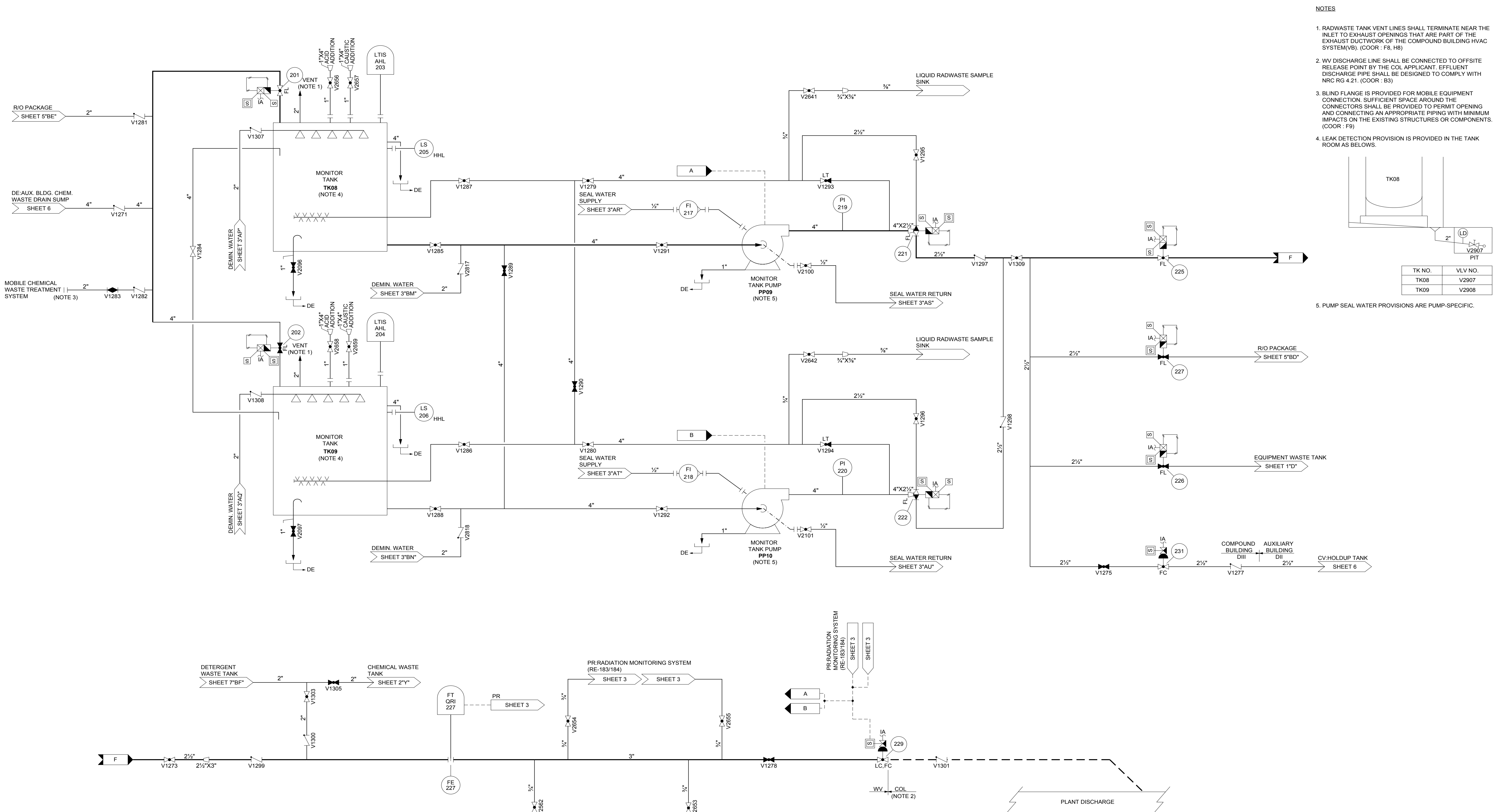


Figure 11.2-1 Liquid Radwaste System Flow Diagram (6 of 7)

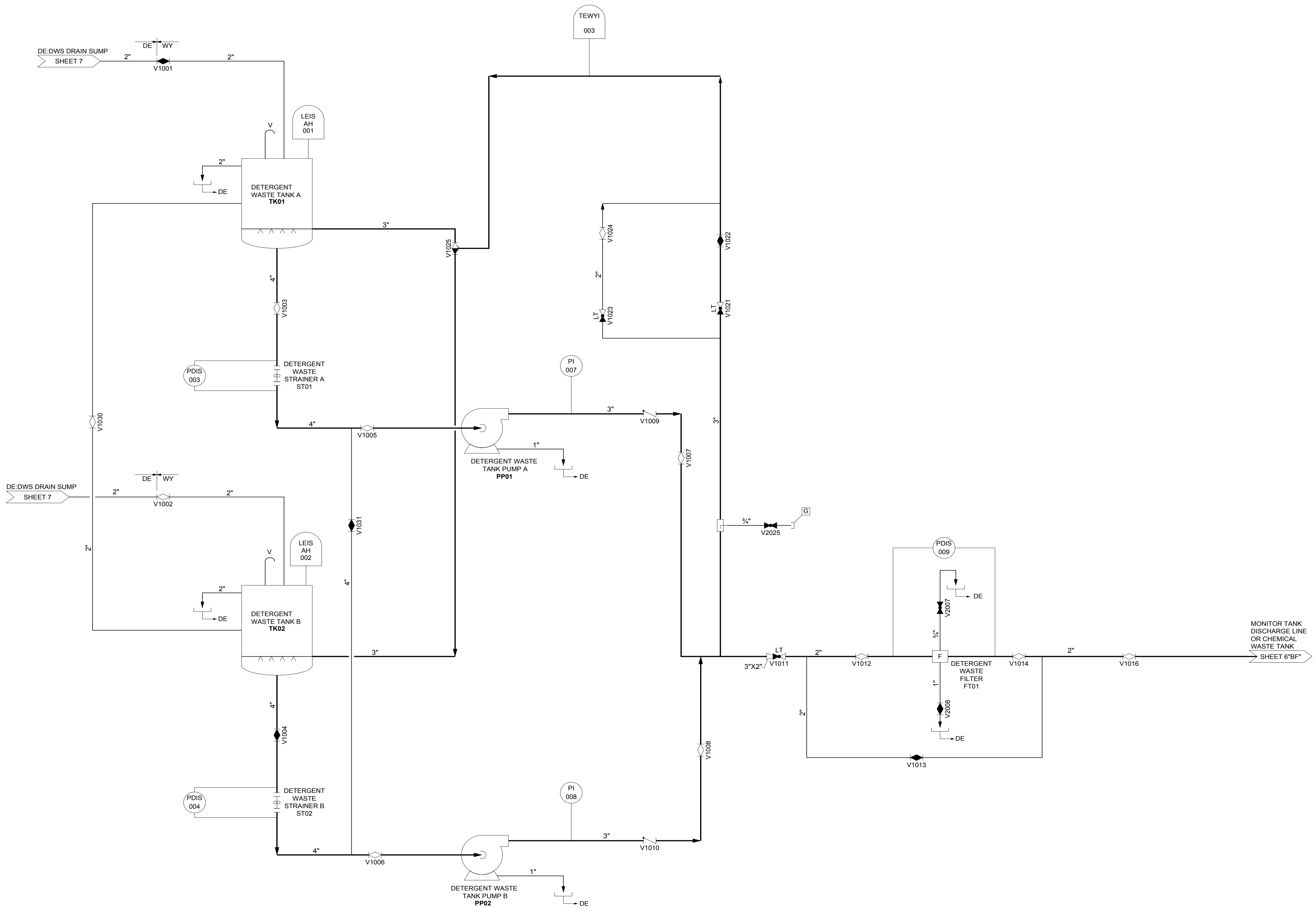


Figure 11.2-1 Liquid Radwaste System Flow Diagram (7 of 7)

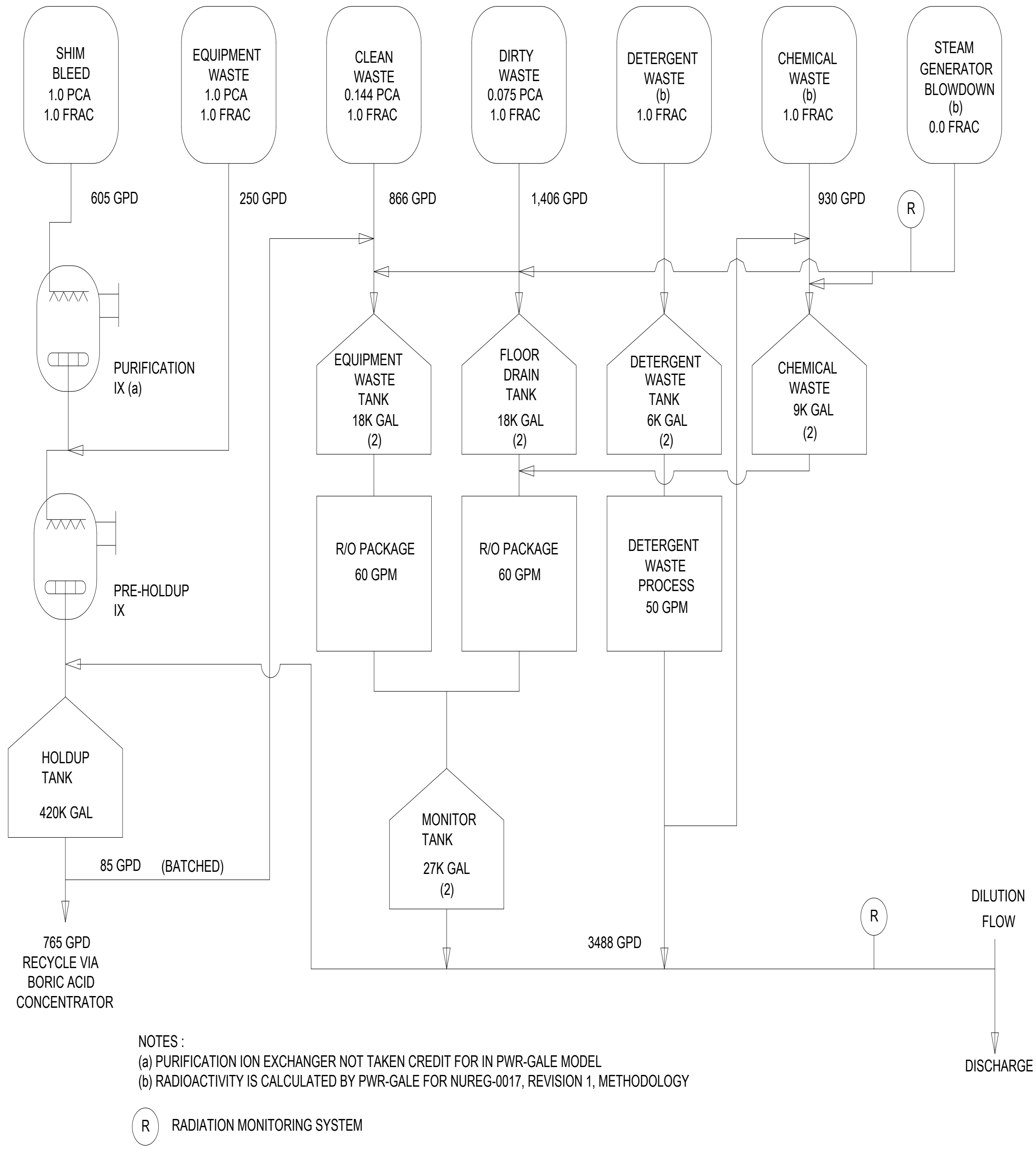


Figure 11.2-2 Simplified Liquid Process Model

11.3 Gaseous Waste Management System

As stipulated in U.S. NRC Standard Review Plan Section 11.3 (Reference 1), the gaseous waste management system (GWMS) covers the gaseous radwaste subsystem (GRS) and the building ventilation subsystems. The design and methods of treatment for the building ventilation subsystems are covered in Section 9.4 of this DCD, and the condenser vacuum system is covered in Section 10.4. Section 11.3 covers the design of the GRS, which handles and processes reactor offgas from the chemical and volume control system (CVCS). This section also includes a summary of the expected gaseous releases from the GRS and the various building ventilation release streams that are used to determine the offsite individual doses resulting from normal plant gaseous releases.

The GRS is designed to monitor, control, collect, process, handle, store, release to ventilation exhaust, and dispose of gaseous radioactive waste generated as a result of normal operation, including anticipated operational occurrences (AOOs), following the guidance of NRC RG 1.143 (Reference 2), NUREG-0017 (Reference 3), and other applicable codes and regulations delineated in this section.

The GRS handles and processes the radioactive offgases coming from the CVCS tank vents, the gas stripper offgases containing radioactive noble gases, halogens, hydrogen, and oxygen from reactor operation, and nitrogen used as a cover gas for these tanks. The noble gases and the halogens are removed by adsorption from the gaseous stream for decay in the charcoal beds to reduce the quantity of radioactive materials prior to release to the environment. The treated gases are routed into the compound building HVAC ventilation exhaust for release. The radionuclide concentrations of gaseous effluent releases at the site during normal operation, including AOOs, are treated to below the radionuclide concentration limits in 10 CFR 20, Appendix B (Reference 4), and conform with the ALARA criteria of 10 CFR 50, Appendix I (Reference 5), based on the use of the industry-proven technology of charcoal adsorption incorporated into the design.

The radiation level in the processed gases is verified by sampling and analysis and by radiation monitoring prior to release to the environment. Process and effluent radiation monitoring and sampling systems are described in Section 11.5.

The GRS design is supplemented with operating procedures, programs, and operator actions to provide assurance that the SSC integrity and functions are maintained, and the releases are within the limits in 10 CFR 20, Appendix B (Reference 4).

The lessons learned program provides guidance on the integration of industry, operating, and construction experience into the APR1400 design. Under this program, NRC generic communications, and industry operating and construction experience, are maintained in a database that is reviewed, assessed, and integrated into the design as appropriate. The construction and operating experience of nuclear power plants has been incorporated into the database for design improvement.

11.3.1 Design Bases

11.3.1.1 Design Objectives

The design objectives of the GRS are as follows:

a. Provide the capability to control, collect, process, handle, store, monitor, and dispose of radioactive gaseous waste generated as the result of normal operation including AOOs to meet release radionuclide concentration limits stipulated in 10 CFR 20, Appendix B (Reference 4), prior to discharge to the environment. The radioactive gaseous inputs include processed gases from the GRS and the building ventilation exhausts from the facilities that house components containing radioactive materials.

b. Provide reasonable assurance that the radiation doses resulting from the releases of radioactive materials in gaseous effluents are kept ALARA.

c. Remove and reduce radioactive materials to the environment to meet the requirements of 10 CFR 50, Appendix I (Reference 5).

The GRS is designed for individual unit operation, and no subsystems or components are shared with other radwaste systems or other systems.

11.3.1.2 Design Criteria

The GRS is designed with the set of design criteria below to ensure safe and uninterrupted operation. These criteria generally follow the guidance in the NRC Standard Revew Plan, Section 11.3 (Reference 1) as discussed below.

a. The GRS shall be designed to process effluents normally released to unrestricted areas to meet the concentration limits of 10 CFR 20, Appendix B (Reference 4), during normal operation, including AOOs.

   The GRS uses charcoal adsorber systems with guard beds, a waste gas dryer, a HEPA filter, and associated controls to process offgas. The charcoal adsorption method has been proven in industrial operation for its effectiveness for noble gases adsorption for delaying release and providing decay of radioactivity from noble gases. The GRS operates and discharges gaseous effluents continuously.

   Table 11.3-1 provides an estimate of the annual airborne effluent releases using the PWR-GALE Code, which is based on NRC RG 1.112 (Reference 6), NUREG-0017 (Reference 3), and ANSI/ANS 18.1 (Reference 7) methodology. Assumptions used to calculate the annual release rate are addressed in Subsection 11.3.3.1, and the results are listed in Table 11.3-6. Based on the design, the analysis provides reasonable assurance that effluents from normal operation and AOOs meet the concentration limits of 10 CFR 20, Appendix B (Reference 4) and the design objectives.

b. The system is designed to meet the design objective of maintaining operation and maintenance exposure ALARA.

   The GRS is designed in accordance with guidance provided in ANSI/ANS 55.4 (Reference 8) and conforms with NRC RGs 1.143, 1.52, 1.140, and 8.8 (References 2, 9, 10, and 11, respectively). The components are located in shielded cubicles to keep doses to personnel ALARA and instrumentation is located in low radiation areas. Operational procedures are to be used to provide assurance that occupational exposures for inspection, instrument calibration, and maintenance activities are maintained ALARA. The system meets the design objectives of performance without interference with normal operation, including AOOs.

c. The GRS is a non-safety-related system with the exception of the containment penetration isolation valves and piping (which are safety related) and has no accident mitigation function. The GRS is designed in accordance with 10 CFR 50, Appendix A, GDC 3 (Reference 12) and ANSI/ANS 55.4 (Reference 8) and conforms with NRC RGs 1.52 (Reference 9), 1.140 (Reference 10), and 1.143 (Reference 2).

The GRS design includes the following features:

1) The GRS is designed with a gas header to collect the gas inlet streams and a drain tank, which provides a means to collect condensed moisture in the process pipe. The process gas collected from the gas header is cooled by the gas dryer to reduce and condense the moisture. The gas is heated and routed into the guard beds, which are designed to protect delay beds from the residual moisture in the gas streams. The inlet gas mixture is monitored for oxygen content and can be diluted with nitrogen gas to preclude the buildup of an explosive mixture of hydrogen and oxygen, which could affect operation of the plant. This design approach meets the design objectives for radioactive gas handling and processing.

2) The GRS components are located in cubicles that provide shielding. The system is designed with automatic operations with minimum operator actions. This design reduces worker exposure to meet 10 CFR 50, Appendix I (Reference 5) and is in keeping with the ALARA principle.

3) The GRS design is based on the maximum gas flow rate and design basis source terms. The system is designed with redundancy to accommodate an increase in demand during normal operation of the plant and to allow component isolation for maintenance. This design approach provides sufficient capacity and flexibility for various modes of operation with minimum interruption of normal operation.

d. Releases of radioactive materials to the environment are controlled and monitored in accordance with 10 CFR 50, Appendix A, General Design Criteria (GDC) 60, 61, and 64 (References 13, 14, and 15, respectively).

   The GRS is provided with a radiation monitor at the discharge line from the charcoal delay beds to the compound building HVAC system. The discharge of the GRS is automatically isolated if the preset trip setpoint is exceeded. Section 11.5 provides a detailed description of radiation monitoring for the GRS.

e. The GRS is designed such that accidental releases of radioactive materials from the failure of a single component of the GRS are not to result in offsite doses that exceed the guidelines of NRC Branch Technical Position (BTP) 11-5 (Reference 16).

   Subsection 11.3.3.2 provides a description of the analysis of a single component failure of the GRS. The methodology used in this analysis is in accordance with BTP 11-5 (Reference 16) using the design basis source term. The results of these analyses confirm that the consequence of a single component failure of the GRS is within the guideline dose limits of BTP 11-5 (Reference 16) (1 mSv total effective dose equivalent).

f. The GRS design includes two gas analyzers with automatic control functions to preclude the buildup of an explosive mixture of hydrogen and oxygen in accordance with the NRC Standard Review Plan, Section 11.3 (Reference 1). Gaseous waste is sampled from various process points for analyzing oxygen and hydrogen concentration within the GRS. Two hydrogen analyzers and two oxygen analyzers are used to monitor hydrogen and oxygen gas concentrations within the GRS. One hydrogen and one oxygen analyzer is provided to continuously monitor hydrogen and oxygen gas concentrations in the gas surge header of the GRS. The other hydrogen and oxygen analyzer in the gaseous radwaste sample panel is used to analyze samples from process points within the GRS package. Alarms are provided in the radwaste control room of the compound building and the main control room (MCR) and annunciate on high and high-high levels of oxygen concentrations.

g. The GRS components are designed to be hosued in shielded cubicles that are provided with an epoxy coating to provide smooth and nonporous surfaces to minimize crevices and cracks that could lead to the buildup of contamination and to facilitate cleaning and decontamination. Floors are lined with Service Level II epoxy coatings as defined in NRC RG 1.54 (Reference 17). The COL applicant is to prepare and implement the epoxy inspection and maintenance program in the GRS (COL 11.3(1)). This design approach is implemented following the objectives of NRC RG 4.21 (Reference 18) for the minimization of cross-contamination and decommissioning planning.

h. Interconnections between the GRS and other plant systems are designed with double isolation devices so that the potential of contamination of nonradioactive systems is minimized and the potential for uncontrolled and unmonitored releases of radiation to the environment from a single failure is minimized. This feature meets the requirements of IE Bulletin 80-10 (Reference 19).

   The GRS is designed with hard piping between radioactive and nonradioactive systems in accordance with the IE Bulletin 80-10 (Reference 19). Nitrogen gas is provided for purging the pipe after each transfer of contaminated fluid. The nitrogen system operates at a higher pressure, the interface connections are hard pipes, and the interfaces are equipped with double isolation devices to prevent unintended contamination in accordance with NRC RG 4.21 (Reference 18).

i. In accordance with NRC RG 1.143 (Reference 2) and ANSI/ANS-55.4 (Reference 8), the GRS is designed to withstand the effects of an OBE.

11.3.1.3 Other Design Considerations

The GRS design conforms with NRC RG 1.143 (Reference 2) from the applicable Regulatory Positions (C.2, C.4, C.5, C.6, and C.7). The Regulatory Positions include the following:

a. The GRS is designed and tested in accordance with Regulatory Position C.2 of NRC RG 1.143 (Reference 2).

1) The GRS is designed and tested according to the codes and standards that are listed in Table 11.3-2, and conforms with Regulatory Positions C.2.2 and C.4 of NRC RG 1.143 (Reference 2).

2) Materials used for pressure-retaining portions of components in the GRS are designed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section II (Reference 20). Materials used in the GRS are compatible with the chemical, physical, and radioactive environments during normal and AOO conditions. Malleable, wrought or cast irons, and plastics cannot be used in the GRS.

3) A high oxygen concentration alarm (for a concentration greater than 2 percent) from any incoming source is annunciated in the radwaste control room of the compound building. Operating personnel can mitigate the situation by closing the source of the oxygen or via nitrogen dilution or purge. A high-high oxygen concentration alarm (for a concentration greater than 4 percent) from any incoming sources is annunciated in the MCR and the radwaste control room of the compound building. Under this condition, nitrogen is automatically injected into the GRS to reduce the oxygen concentration in order to prevent the buildup of an explosive mixture.

b. The GRS design and testing requirements conform with Regulatory Position C.4 of NRC RG 1.143 (Reference 2).

1) The GRS is housed in the compound building and is designed to minimize leakage through the use of welded construction, following the quality assurance requirements stipulated in Regulatory Position 7, and is to be fully tested in accordance with Table 1 in NRC RG 1.143 (Reference 2).

2) The GRS is constructed in accordance with Regulatory Position C.4.3 of NRC RG 1.143 (Reference 2). In addition, sufficient spaces are provided to facilitate access, operation, periodic inspection and testing, and maintenance to maintain personnel radiation exposures ALARA in accordance with NRC RG 8.8 (Reference 11) guidance.

3) The GRS is pneumatically pressure tested in conformance with Regulatory Position C.4.4 of NRC RG 1.143 (Reference 2). Testing of piping systems is performed in accordance with applicable codes and standards as described in Table 11.3-2. For the APR1400, pipes less than 25 mm (1 in) are exempt from pressure testing provided the original system was pressure tested.

4) The GRS is designed to permit periodic testing of active components to evaluate the operability in accordance with Regulatory Position C.4.5 of NRC RG 1.143 (Reference 2).

c. The GRS components are classified as RW-IIa, RW-IIb, or RW-IIc as described in Regulatory Position C.5 and are designed to the natural phenomena and man-induced hazards criteria in Regulatory Position C.6 of NRC RG 1.143 (Reference 2), as applicable. The compound building is designed to Radwaste Safety Classification RW-IIa.

d. The quality assurance (QA) program for the design, installation, procurement, and fabrication of GRS components conforms with Regulatory Position C.7 of NRC RG 1.143 (Reference 2) and NRC RG 1.33 (Reference 21). Table 3.2-1 identifies seismic category, quality, and safety class for each of the respective components in the GRS.

e. The Radwaste Safety Classification applies to the GRS equipment up to and including the nearest isolation valves. The Radwaste Safety Classification for piping is determined by the inventory based on pipe sizes, lengths, and the fluid concentrations in accordance with NRC RG 1.143 (Reference 2), except for the containment isolation valves and penetration piping. The components and the associated piping are designed, fabricated, and tested in accordance with NRC RG 1.143 (Reference 2) as specified in Table 11.3-2.

f. The GRS is designed to operate at slightly above atmospheric pressure and with a low pressure drop across the charcoal adsorber beds. This operating condition, and the component welded construction, minimizes the potential for leakage and also minimizes the entrainment of fines. Additionally, a HEPA filter is provided downstream of the charcoal adsorbers to filter and retain the fines to minimize airborne releases.

g. To prevent the premature exhaustion of the charcoal, the GRS is designed with a waste gas dryer and guard beds, including the associated control and alarm functions to protect the charcoal delay beds from damages due to excessive moisture. Charcoal guard beds also remove the residual moisture. Humidity sensors are located in upstream and downstream of charcoal guard beds. If excessive moisture is detected to enter the charcoal guard beds, the operator can isolate the charcoal guard bed train for regeneration, drying with a nitrogen purge, or the operator replaces the charcoal.

11.3.1.4 Method of Treatment

The GRS uses charcoal at ambient temperature to delay the passage of radioactive gases for decay. When operating at design conditions, the mass of charcoal provided in the adsorber beds is sufficient to provide a delay of 45 days for xenon and a delay of 3.5 days for krypton. The waste gas dryer controls the inlet gas moisture and temperature to achieve the desired performance of the charcoal delay beds.

Streams in the GRS are monitored for both hydrogen and oxygen content so that a flammable mixture does not accumulate. An explosive mixture of hydrogen and oxygen in the GRS is prevented by maintaining an oxygen concentration of less than 4 percent by volume. The treated gases are then routed to mix with the compound building ventilation flow before it is discharged to the environment. A sampling connection and a radiation monitor are provided with the GRS effluent gas for confirmation that the effluent release concentrations are below the prescribed limits. Another radiation monitor is provided on the ventilation flow before it is released. This feature conforms with 10 CFR 50, Appendix A, GDC 3 (Reference 12) and the guidance in NRC RG 1.189 (Reference 22).

The design parameters for the GRS are listed in Tables 11.3-3 and 11.3-4. The GRS has the capability to process gases associated with the design basis source term and the design basis flow rate.

The GRS is designed with the industry proven technology of charcoal adsorption so that releases of radioactive gases are below the concentration limits in 10 CFR 20, Appendix B (Reference 4). The GRS design allows the Technical Specifications for the release of gaseous effluents to be met and keeps offsite doses to the public within the specified limits.

The GRS uses equipment that is commonly used in the nuclear power industry, whose performance is proven and documented. The equipment is sized to process waste gases using design basis source term and design conditions that bound normal operation including AOOs. The equipment is also housed in the compound building with sufficient shielding. Charcoal guard beds reduce the concentration of radioactive iodine in the effluent stream. Noble gases are delayed in the charcoal beds to facilitate decay prior to release.

GRS equipment is designed, located, and shielded to conform with the guidance of NRC RG 8.8 (Reference 11), thus maintaining occupational doses ALARA.

The GRS includes radiation monitoring to continuously measure the radioactivity in the effluent stream prior to release into the environment to conform with the requirements of GDC 60 (Reference 13) and 64 (Reference 15). Additional and redundant radiation monitors are provided in the building ventilation system to verify the radiation level. Upon detection of radiation levels above the setpoint, the monitor activates an alarm and sends signals to close the GRS discharge valves. Hence, the GRS design precludes the unmonitored and uncontrolled releases of radioactivity to the environment to meet the requirements of IE Bulletin 80-10 (Reference 19).

The GRS is designed with at least two isolation valves between the clean and contaminated systems to minimize the potential for contamination of clean systems. This feature meets the requirements of 10 CFR 20.1406 (Reference 23) and RG 4.21 (Reference 18).

11.3.1.5 Radioactive Source Terms in GRS

As shown in Figure 11.3-1, the input sources to the GRS are the vent gases from the reactor drain tank (RDT), volume control tank (VCT), equipment drain tank (EDT), and gas stripper. The radioactive sources for each component of the GRS are calculated using the radioactive concentrations of the inflows to the GRS from the CVCS components shown in Table 11.1-8, which are determined based on the reactor coolant radionuclide concentrations provided in Table 11.1-2.

The mixed specific activities of sources to the GRS are then calculated by weighting each source contribution corresponding to its partial flow fractions. Activity buildup on the charcoal beds is modeled, assuming the holdup times for noble gases that are presented in Table 11.2-2.

Buildup activity in the charcoal guard bed is calculated assuming that the inflow activities are adsorbed in the charcoal bed entirely. For the buildup activities in i^{th} charcoal delay bed, the following differential equation is used:

$$\frac{dA_i}{dt} = Q_{i-1} - Q_i - \lambda A_i$$

$$Q_i = Q_{i-1} \cdot e^{-\lambda T_H}$$

Where:

A_i = buildup activity in the i^{th} charcoal delay bed (Bq)

Q_i = flow rate of radioactivity (Bq/hr)

T_H = holdup time per each charcoal bed (hr)

λ = decay constant (hr^{-1})

Tables 11.3-10 and 11.3-11 provide the expected and design basis (1 percent fuel defect) radioactive inventories of each GRS component.

The method to determine radwaste classification for GRS components is the same as the method described for the LWMS (see Subsection 11.2.1.4).

11.3.1.6 Site-Specific Cost-Benefit Analysis

The GRS is designed to be used for any site.

A cost-benefit analysis is required by 10 CFR 50, Appendix I (Reference 5), Section II, Paragraph D, to demonstrate that the addition of treatment units to the existing design is not cost beneficial. NRC RG 1.110 (Reference 24) provides numerical guidelines for offsite radiation doses as a result of gaseous or airborne radioactive effluents during normal operations including AOOs for conformance with 10 CFR 50 Appendix I (Reference 5).

The COL applicant is to perform a site-specific cost-benefit analysis following the guidance in NRC RG 1.110 (Reference 24) for conformance with 10 CFR 50 Appendix I (Reference 5) (COL 11.3(2)).

11.3.1.7 Mobile or Temporary Equipment

The GRS is designed with permanently installed equipment. The GRS does not include the use of mobile or temporary equipment.

11.3.2 GRS Description

The process flow diagram (PFD) of the GRS is provided in Figure 11.3-1, and an equipment list for the GRS is provided in Table 11.3-4. The COL applicant is to prepare and provide the piping and instrumentation diagram (P&ID) for the combined operating license application (COL 11.3(3)).

Gaseous waste contains radioactive krypton and xenon, which are fission products that originate from fuel and tramp uranium on fuel surfaces. The GRS receives fission gases through the gas surge header and uses charcoal delay beds to selectively delay the discharge of xenon and krypton gases to facilitate decay prior to release. The primary input sources to the gas surge header are the gas stripper, VCT, RDT, EDT in the CVCS. The gases consist primarily of hydrogen with a small amount of nitrogen, oxygen, and trace quantities of fission gases. Most of the hydrogen comes from the VCT, which use hydrogen as a cover gas. A small amount is generated from the hydrazine injected into the reactor coolant for pH control. The removal of these gases, including the trace amount of fission gases occurs in the VCT from depressurization, and in the gas stripper, which is used to remove the dissolved gases in the reactor coolant. This minimizes the buildup and escape of radioactive gases during maintenance on the reactor coolant system (RCS) and minimizes releases resulting from leakage of reactor coolant.

The GRS consists of one header drain tank, two waste gas dryers, one standby chiller, two charcoal guard beds, four charcoal delay beds, one high-efficiency particulate air (HEPA) filter and the associated piping, valves, and instrumentation.

The GRS is designed with a gas flow rate of 22 scfm. This design flow rate exceeds the normal letdown flow rate of 80 gpm, which converts to a displaced cover gas flow of 11 scfm. During a pressurizer relief operation, the cover gas flow could increase for a short period of time. The surge cover gas is directed to the gas surge header for moderation and buffering of flow. The gas surge results in a sligtly higher gas velocity through the charcoal bed, because of the enlarged charcoal bed diameter and the void spaces within the bed. The gas velocity inside the charcoal beds is still slow and does not cause any significant entrainment of charcoal. The screen installed at the bottom of each GRS delay bed prevents the blowout of charcoal fines. The HEPA filter is used to remove any charcoal fines escaped from the delay beds. There is no buffering function in the GRS header drain tank.

The GRS uses charcoal at ambient temperature to delay the passage of radioactive gases. When operating at design conditions, the mass of charcoal in the absorber beds is sufficient to provide a delay of 45 days for xenon and a delay of 3.5 days for krypton. The waste gas dryer controls the inlet gas moisture and temperature to achieve the desired performance of the charcoal delay beds.

The condensed liquid in the gas surge header in the auxiliary building and in the GRS inlet piping in the compound building is collected in the GRS header drain tank. The tank is also used to collect condensate from the waste gas dryer.

Downstream of the gas surge header, two 100 percent capacity trains, each comprising one waste gas dryer and one charcoal guard bed, are used to reduce the gas moisture to protect the charcoal in the main delay beds as the performance of the charcoal delay beds can be degraded by moisture.

The waste gas dryer cools the waste gases to below 7.8 °C (46 °F) and removes the condensate before the gas enters the guard beds. The GRS chiller provides the cooling water when the plant chilled water system is unavailable. Humidity sensors downstream of the waste gas dryer and charcoal guard beds are provided to monitor the moisture content and alarm for operator actions if the moisture content is at an unacceptable level.

Two charcoal guard beds are provided upstream of the charcoal delay beds. Only one is normally operating; the other one is in the standby mode or regeneration mode. The guard bed further protects the main charcoal delay bed from moisture. Humidity sensors are installed upstream and downstream of the charcoal guard bed to monitor the charcoal wetting condition. Temperature sensors are installed at the guard beds and delay beds. Iodine is held up for decay in the charcoal guard beds.

The four delay beds are normally operating in series. The leading delay beds can be isolated for regeneration or replacement, if needed. This mode of operation is temporary and the delay beds can be switched back in when they are ready. During this mode of operation, the gas velocity remains unchanged, but the adsorption rate of the xenon and krypton gases is temporarily increased to compensate for the beds in maintenance mode. Nitrogen purge is available to dry the charcoal beds in the event of excessive moisture contamination. The four charcoal delay beds, containing a total of 9,525 kg (21,000 lb) of charcoal, are used for xenon and krypton delay. All GRS components are located in a shielded cubicles.

After passing through the charcoal delay beds, the waste gas flows through a HEPA filter where particulates, including charcoal fines, are removed, and then it is vented to the compound building HVAC system.

The GRS operates at pressures slightly above atmospheric to provide the necessary pressure to route the gas flow into the HVAC ventilation exhaust. Operating at this slightly pressurized condition also minimizes the potential for oxygen inleakage. Leakage from the GRS is further limited through the use of welded connections wherever the connections are not restricted for maintenance purposes. Control valves are provided with bellow seals to minimize leakage through the valve stem.

The GRS is designed to prevent the formation or buildup of explosive mixtures of hydrogen and oxygen by monitoring the concentrations of hydrogen and oxygen through one of the two gas analyzers (continuous monitoring). The concentrations are confirmed by periodic sampling and analysis at several routing locations. When the oxygen concentration is detected to be higher than the predetermined setpoint (high-high setpoint), nitrogen is injected to dilute the concentration to below the lower flammable limit of 4 percent. Along the gas flow paths, there are process vessels (VCT, RDT, EDT, gas stripper, GRS header drain tanks, and associated piping) that are designed in accordance with ASME VIII (Reference 25) for pressure vessels. Accordingly, design pressures are assigned to contain significant margins above the normal operating pressure, and relief valves are provided for each vessel to protect against surges in pressure. A loop seal is provided downstream of the rupture disc for the GRS header drain tank vent to limit the leakage of radioactive gases. The loop seal is provided with a connection for demineralized water to allow filling of the loop seal to prevent continuous venting of radioactive gas.

The system is designed to alarm locally and in the MCR for operator action.

One of the two gas analyzers takes continuous samples from various process points and from input sources to the system (e.g., gas stripper, volume control tank, reactor drain tank). The gas analyzer is set at a high alarm of 2 percent and high-high alarm at 4 percent oxygen concentrations. The alarm at the high setpoint (2 percent) provides the operating personnel with sufficient time for mitigation actions to lower the concentration of hydrogen and oxygen. Mitigation actions include investigation and eliminating or isolating the source of oxygen infiltration to the system or adding nitrogen gas as needed to stabilize and reduce oxygen concentrations within the system to less than the alarm level (2 percent). At the high-high setpoint, nitrogen is automatically injected to lower the oxygen concentration to below the 2 percent "high" setpoint. The APR1400 also includes design features such as chemistry control to minimize hydrolysis, which produces oxygen, and welded fabrication, which reduces or eliminates sources of oxygen leakage. The GRS is designed with the capability to isolate system and component inputs and the capability to use nitrogen purges to prevent the buildup of explosive mixtures.

11.3.2.1 Component Description

The radioactive safety classifications of the GRS components are determined in accordance with the guidance provided in RG 1.143 (Reference 2). The component safety classification is summarized in Tables 11.3-4. Accordingly, the GRS is classified as RW-IIa, based on the highest safety classification for the components within the system boundary. The GRS components are housed within the compound building, which has been determined to be RW-IIa.

The safety classification for the GRS component applies to the components, up to and including the nearest isolation valves, fittings, and/or welded/flanged nozzle connections. The piping classification is determined based on the inventory and the composition of the gas for the corresponding segments of piping.

11.3.2.1.1 Charcoal Delay Beds

The holdup of radioactive gases for decay is accomplished by the selective adsorption of xenon and krypton gases in the charcoal delay beds. Charcoal has been used in the nuclear industry for adsorption and delay for radioactive decay of noble gases. The charcoal beds are designed with a sufficient amount of charcoal to provide the necessary retention time to facilitate decay of these gases to reduce the radioactivity for release. Analyses using the method described in ANSI/ANS 55.4 (Reference 8), the design basis source terms, and the maximum gas flow rate provides a conservative design and the operating margins for normal operations, including AOOs. The results of the analyses are summarized in Tables 11.3-10 and 11.3-11 for the expected and design basis conditions.

The GRS is designed to operate close to the ambient conditions. The operating pressure is slightly higher than the atmospheric pressure to provide the motive forces for the gas stream to flow through the process components into the compound building exhaust. The gas temperature is controlled by the dryer and chilled water. The charcoal bed performs more efficiently and effectively at lower temperatures through the use of the chilled water system or the standby chiller unit when plant chilled water is not available. The vessels are designed to prevent charcoal carryover by charcoal support screens. The charcoal delay beds are designed to allow the replacement of the charcoal. Piping connections are arranged to make it possible to bypass any charcoal delay bed for maintenance reasons. In addition, nitrogen gas can be introduced to each charcoal delay bed to flush or dry the charcoal.

The charcoal delay bed vessels are pressure vessels designed in accordance with NRC RG 1.143 (Reference 2).

11.3.2.1.2 Waste Gas Dryer

One of two waste gas dryers is normally in service to reduce the moisture content of the gases by cooling with chilled water. The cooling water is supplied from the plant chilled water system or from the standby GRS chiller. The waste gas dryer is designed to take an inlet gas flow of 623 L/min (22 scfm) with an inlet temperature of 48.9 °C (120 °F) and an outlet temperature of 7.8 °C (46 °F).

The waste gas dryer is a condensing shell-and-tube heat exchanger designed in accordance with NRC RG 1.143 (Reference 2).

11.3.2.1.3 GRS Header Drain Tank

All of the condensed liquid in the gas surge header is collected in the GRS header drain tank. The condensed liquid is drained into the compound building normal sump, from which the drains are routed to the LWMS for processing and release. The GRS header drain tank is provided with a level control. Tank water level is interlocked with the drain line isolation valve.

The GRS header drain tank is a pressure vessel designed in accordance with NRC RG 1.143 (Reference 2).

11.3.2.1.4 Charcoal Guard Bed

The charcoal guard beds are provided at the upstream of the charcoal delay beds. The guard beds protect the main charcoal delay bed from moisture and the delay of iodine for decay.

The charcoal guard bed vessels are pressure vessels designed in accordance with NRC RG 1.143 (Reference 2).

11.3.2.1.5 HEPA Filter

After passing through charcoal delay beds, the waste gas flows through a HEPA filter where particulates, including charcoal fines, are removed. The waste gas then flows through a check valve to prevent backflow. Efficiency of the HEPA filter is not credited in the PWR-GALE Code calculation. The filter has test ports for in-place testing. Through the check valve, the waste gas flows to the environment after it is diluted with building ventilation air.

The HEPA filter vessel is a vessel designed in accordance with NRC RG 1.143 (Reference 2).

11.3.2.2 Design Features

11.3.2.2.1 General Design Features

The following features are provided for the GRS to meet the design criteria and design objectives:

a. The high-activity gaseous waste stream is processed through charcoal delay beds to facilitate decay of radioactive fission gases prior to release. Decay of fission product gases (xenon and krypton) prior to release significantly reduces radioactivity levels and lowers offsite radiation exposure doses. Additionally, filtration by the air cleaning unit of the compound building HVAC reduces offsite exposure by reducing radioactive particulates and iodine in plant effluents. The use of charcoal beds for delay and decay of fission gases is industry proven. The GRS is designed with a sufficient quantity of charcoal to provide the necessary delay and decay for compliance with 10 CFR 20, Appendix B (Reference 4) and 10 CFR 50, Appendix I (Reference 5) for all modes of operation. Radiation monitoring is provided for continuous monitoring of the gaseous activity to ensure that the effluent specification is met. Sampling and analysis are provided for additional measurements that can be used to confirm the radiation monitor readings. This design approach meets the design objectives and the design criteria set forth in Subsection 11.3.1 for the GRS.

b. The header drain tank is used to collect all condensates from the gaseous waste piping and the waste gas dryer. The header drain tank provides the supporting function to collect the moisture for the protection of the delay beds and directs the drainage to the LWMS for treatment and release, thus meeting the NRC RG 4.21 (Reference 18) design objective for minimization of the spread of contamination.

c. The charcoal delay process consists only of passive components. A charcoal guard bed is provided upstream of the charcoal delay beds to protect the beds in the unlikely event of excessive moisture input. The moisturized charcoal guard bed can be dried by nitrogen injection. This design approach minimizes the consumption of charcoal, supports the performance objective of the charcoal bed treatment, and additionally meets the NRC RG 4.21 (Reference 18) design objective for minimization of waste generation.

d. The radioactivity of the processed gaseous waste is monitored prior to discharge to the environment via the compound building ventilation exhaust, and the discharge flow is automatically isolated if the preset limit is exceeded. If the limit is exceeded, an isolation signal is triggered by the radiation monitor to close the effluent discharge valve. Operator actions are needed to reset the flow after clearance of the radioactivity alarm. This approach precludes an accidental release of processed gaseous waste that could result in offsite exposures.

e. The GRS is also designed with a ventilation exhaust flow measurement instrument with alarm signals and interlocks to the GRS efflucent discharge valve to preclude accidental releases when there is insufficient or no ventilation flow.

f. Continuous gas analysis and alarms are provided to preclude the buildup of explosive mixtures of hydrogen and oxygen in accordance with ANSI/ANS 55.4 (Reference 8)..

g. Drain lines and valves are sized and sloped to minimize the potential for plugging. Valves are the packless metal diaphragm type and have bellowed sealed stems to minimize leakage. This design approach minimizes waste generation and is in compliance with NRC RG 4.21 (Reference 18).

The GRS is designed, constructed, and tested to be as leak-tight as practicable. In order to minimize maintenance and corresponding personnel dose, in compliance with NRC RG 8.8 (Reference 11), during maintenance, the following design features are implemented:

a. Components are installed in separately shielded cubicles to minimize doses to workers.

b. Only proven and qualified equipment from the nuclear industry is used.

c. Steel piping with butt-welded construction is used to minimize crud traps. Only qualified welders are used.

d. GRS components are located in cubicles at a higher elevation than the basemat floor. The cubicle floors are epoxy coated to provide smooth surfaces that minimize cracks and crevices, and facilitate cleaning and decontamination in accordance with NRC RG 4.21 (Reference 18).

e. Nonradioactive auxiliary subsystems are provided with double isolation devices to prevent cross-contamination from the radioactive process streams in accordance with NRC RG 4.21 (Reference 18).

f. Equipment, piping, and instruments are subjected to strict leak rate testing and inspections.

11.3.2.2.2 Design Features for Minimization of Contamination

The APR1400 is designed with features that meet the requirements of 10 CFR 20.1406 (Reference 23) and NRC RG 4.21 (Reference 18). The basic principles of NRC RG 4.21 (Reference 18) and the methods of control suggested in the regulations are delineated in four design objectives and two operational objectives, which are addressed in Subsection 12.4.2. The following description summarizes the primary features of the design and operational objectives for the GRS.

The GRS SSCs, including the facility that houses the components, are designed to limit leakage and/or control the spread of contamination. In accordance with NRC RG 4.21 (Reference 18), the GRS has been evaluated for leakage identification from the SSCs that contain radioactive or potentially radioactive materials, the areas and pathways where probable leakage may occur, and the methods of leakage control incorporated into the design of the system. The leak identification evaluation indicated that the GRS is designed to facilitate early leak detection and has the capability to assess collected fluids and respond quickly to manage the collected fluids. Thus, unintended contamination to the facility and the environment is minimized by the SSC design, and by operational procedures and programs for inspection and maintenance activities.

Prevention/Minimization of Unintended Contamination

a. The system contains sufficient charcoal material to hold the noble gas nuclides for a period of decay to reduce the release of radioactivity, thus minimizing contamination of the facility and the environment.

b. The system design, including the waste gas dryers and guard beds, is configured in two parallel trains, one operating and one standing by, each with sufficient capacity to remove the moisture to protect the charcoal beds.

c. A HEPA filter is provided downstream of the charcoal beds to prevent the spread of contaminated charcoal fines.

d. Cubicles in which contaminated materials are stored and processed are epoxy-coated to facilitate cleaning. The GRS header drain tank is equipped with level instrumentation to detect fluid accumulation and drain the fluid to the radioactive drain system.

e. The system is designed with above-ground piping to the extent practicable. Buried and embedded piping is minimized. Piping is sloped to facilitate drainage of condensate to the header drain tank.

f. The system uses valves with leak-tight characteristics, such as the bellows or metal diaphragm types, to minimize leakage.

g. The system uses welded construction to the maximum practicable extent to minimize leakage.

<u>Adequate and Early Leak Detection</u>

a. The system is designed with gas analyzers and a radiation monitor to provide reasonable assurance of the integrity of the SSCs, including piping, and to provide alarms to warn operators of the potential for explosive gas concentrations.

b. The system is designed with adequate space around all components to enable prompt evaluation and response to leakage detection.

<u>Reduction of Cross-Contamination, Decontamination, and Waste Generation</u>

a. The SSCs are designed with life-cycle planning through the use of nuclear–industry-proven equipment and materials that are compatible with the chemical, physical, and radiological environment, thus minimizing cross-contamination and waste generation.

b. The process piping containing contaminated fluid is sloped to facilitate flow and reduce fluid traps, thus reducing decontamination and waste generation. Decontamination fluid is collected and routed to the LWMS for processing and release.

c. Utility connections are designed with a minimum of two barriers to prevent contamination of nonradioactive systems from radioactive systems.

Decommissioning Planning

a. The SSCs are designed for the full service life and are fabricated, to the maximum extent practicable, as individual assemblies for easy removal.

b. The SSCs are designed with decontamination capabilities using low-pressure nitrogen. Design features such as welding techniques and surface finishes are included to minimize the need for decontamination and minimize waste generation.

c. The GRS is designed with minimal embedded or buried piping. The drain gas header between buildings is equipped with piping sleeves with leakage directed back to the compound building for collection, thus preventing the spread of contamination.

Operations and Documentation

a. The GRS is designed for remote and automated operations. The system is equipped with instruments to actuate the drain valve from the header drain tank.

b. The COL applicant is to prepare the operational procedures and maintenance programs related to leak detection and contamination control (COL 11.3(4)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning.

c. The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations (COL 11.3(5)). Documentation requirements are included as a COL information item.

Site Radiological Environmental Monitoring

The GRS is part of the plant and is included in the site process control program and the site radiological environmental monitoring program for monitoring of facility and environmental contamination. The site radiological environmental monitoring program includes sampling and analysis of effluent to be released, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants. The COL applicant is to prepare the site process control program and the site radiological environmental monitoring program (COL 11.3(6)).

11.3.3 Radioactive Effluent Releases

11.3.3.1 Radioactive Effluent Releases and Dose Calculation in Normal Operation

Radioactive gaseous effluents generated from normal operation, including AOOs, are treated and released through the compound building ventilation exhaust. The GRS is designed to treat radioactive gaseous effluents to meet the concentration limits of 10 CFR 20, Appendix B (Reference 4), and dose limits of 10 CFR 50, Appendix I (Reference 5). The treated gaseous effluents are released through the compound building. In addition, other building ventilation exhausts are released from the auxiliary building, reactor containment building, and turbine generator building HVAC vents. Figure 11.3-2 provides information on the release points of gaseous effluents including the height, dimensions, effluent temperature, effluent flow rates, and exit velocity.

During normal operation, there are nine release points from various locations from the top of the auxiliary building, reactor containment building, turbine generator building, and compound building. The releases are monitored and the release concentrations are collected and analyzed for 10 CFR 20, Appendix B (Reference 4) compliance evaluation. The individual release fractions are calculated and summed to obtain the total release fraction for all effluent streams and verify that the total fraction is below 1.0. Table 11.3-1 summarizes the expected annual release activities from the various streams: GRS waste gas, fuel handling area, RCB, AB, TGB, and condenser offgas air ejector exhaust. The normalized design basis gaseous effluent concentrations at the site boundary, and a comparison to the 10 CFR 20, Appendix B limits, are tabulated in Table 11.3-6. The design basis gaseous effluent release fraction is 0.16, indicating that the release concentrations are lower than the release limits by a significant margin.

An offline radiation monitor is provided in the GRS to measure the activities of the gross gas gamma. The gas sample is taken at the downstream of the charcoal delay beds but upstream of the treated gas effluent isolation valve for analysis of the activities. The gas sample is returned back to a point downstream of the discharge check valve, which is located close to the connection to the suction side of the compound building HVAC exhaust blowers. The analysis is used to provide reasonable assurance that the short-lived nuclides in the gaseous effluent are treated adequately for the delayed removal of noble gases and to monitor the performance of the charcoal delay beds.

The low (WARN) and high (ALARM) setpoints of the radiation monitors are to be determined by the COL applicant when the site-specific environmental information is available so that the sum of the radioactive releases from all vents does not exceed the concentration limits of 10 CFR 20 Appendix B and dose limits of 10 CFR 50 Appendix I (COL 11.5(8)).

The gaseous releases from plant sources during normal operation, including AOOs, are calculated by using the PWR-GALE Code, which conforms with the methodology of NUREG-0017 (Reference 3). The input data for calculating gaseous releases are presented in Table 11.2-2. The χ/Q value at the exclusion area boundary (EAB) is assumed to be 2.0×10^{-5} sec/m^3 for the calculation of gamma dose in air, beta dose in air, dose to total body, dose from ground, and dose due to inhalation. The χ/Q value at the offsite food production area is assumed to be 1.0×10^{-5} sec/m^3 for the calculation of dose from food intake. The D/Q value at the site boundary is 2.0×10^{-7}/m^2. Expected annual gaseous effluent releases are presented in Table 11.3-1. The design basis effluent concentrations are calculated using Eq. 11.3-1 and are then compared against the concentrations of 10 CFR 20, Appendix B (Reference 4). The sum of ratios of concentrations for the design basis fuel defect is 0.162 as presented in Table 11.3-6. This value is less than 1.0, which indicates that the releases meet the regulatory limit.

The equation for calculating design basis radionuclide concentration of the gaseous effluent is as follows:

$$C_i = CF \cdot Q_i \cdot MF_i \cdot \frac{\chi}{Q} \quad \text{(Eq. 11.3-1)}$$

Where:

C_i = design basis gaseous effluent concentration for the i^{th} isotope, Bq/L

CF = conversion factor (= 3.17×10^{-8} yr/sec)

Q_i = annual expected release rate of nuclide i evaluated by PWR-GALE Code (Bq/yr)

MF_i = multiplication factor for the i^{th} isotope $\left(= \frac{RCS(i)_{DAMSAM}}{RCS(i)_{ANSI/ANS18.1}}\right)$
(ratio of 1 percent fuel defect design basis nuclide concentration to ANSI/ANS-18.1-1999 expected concentration)

χ/Q = maximum directional annual average atmospheric dispersion factor at restricted area boundary, sec/m^3

The maximum individual doses at the exclusion area boundary are calculated using the GASPAR II Code (Reference 26). Parameters used in the GASPAR II Code (Reference 26) are presented in Table 11.3-5. Calculated doses are shown in Table 11.3-7. The annual beta air dose is 0.0794 mGy, and the annual gamma air dose is 0.00613 mGy, which are less than the limits of 0.2 mGy and 0.1 mGy, respectively, as presented in 10 CFR 50, Appendix I. The dose to the total body is 0.00379 mSv/yr, dose to skin is 0.0557 mSv/yr, and the maximum dose to any organ (child bone) is 0.145 mSv/yr. These doses are less than the limitations of 0.05 mSv/yr, 0.15 mSv/yr, and 0.15 mSv/yr, respectively, as presented in 10 CFR 50, Appendix I (Reference 5).

The COL applicant is to perform a site-specific dose calculation following NRC RG 1.109 (Reference 27) and NRC RG 1.111 (Reference 28) and compare the doses from the gaseous effluents with the numerical design objectives of 10 CFR 50, Appendix I (Reference 5) and the conformance requirements of 10 CFR 20.1302 (Reference 29). The COL applicant is also to perform the dose calculation using the total gaseous effluents from the site for comparison with the requirements of 40 CFR 190 (Reference 30) (COL 11.3(7)).

11.3.3.2 Radioactive Effluent Releases and Dose Calculation due to Gaseous Radwaste System Leak or Failure

The GRS, as described in Subsection 11.3.1, is designed to collect, monitor, and process radioactive waste gases that originate in the RCS and are processed by holdup for decay prior to release. The GRS uses ambient-temperature charcoal adsorption beds to provide sufficient holdup for decay of noble gases.

The GRS leak or failure event is described as an unexpected and uncontrolled release of radioactive xenon and krypton gases from the GRS resulting from an inadvertent bypass of the main decay portion of the charcoal delay beds. Isolating or terminating the release is assumed to take as long as 2 hours.

The bases for calculating the maximum offsite concentration of the gaseous effluent resulting from a leak or failure of the GRS are as follows:

a. The design basis gaseous effluent source term is based on the design basis RCS equilibrium concentration resulting from fission product leakage into the RCS based on 1 percent fuel defect in accordance with the NRC Standard Review Plan BTP 11-5 (Reference 16). The BTP 11-5 method adds the accidentally induced charcoal unit bypass leakage to the source term for normal operation. The accidental release source contributions are calculated based on the design basis RCS equilibrium concentration.

b. The short-term, 2-hour accident atmospheric dispersion factor is assumed to be 1.0×10^{-3} sec/m^3. This is consistent with the dispersion factors provided in Section 2.3.

c. The annual gaseous effluent releases and the accidental event gaseous effluent releases without decay are calculated using the PWR-GALE Code, and the calculated values are multiplied by isotope-specific multiplication factors. This multiplication factor is determined by dividing the design basis RCS equilibrium concentration calculated using the DAMSAM Code (Reference 31), which is presented in Table 11.1-2, by the RCS expected concentration, which is presented in Table 11.1-9 for each isotope.

d. For isotopes whose design basis RCS equilibrium concentration, calculated using the DAMSAM Code, is less than the expected concentration, the expected concentration is used for conservatism.

e. Particulates and iodines are assumed to be removed by pretreatment equipment. Therefore, only the total effective dose equivalent (TEDE) from external exposure to noble gases is calculated in this analysis.

The equation used to calculate the dose consequences for failures in the GRS, which is consistent with BTP 11-5 (Reference 16), is as follows:

$$D = \sum_{i} K(i) \cdot Q(i) \cdot \chi/Q$$

Where:

D = total effective dose equivalent, mSv

K(i) = dose conversion factor given in U.S. Environmental Protection Agency (EPA) Federal Guidance Report No. 12 (Reference 32) for the ith isotope, $mSv \cdot m^3/Bq \cdot sec$

χ/Q = short-term accident atmospheric dispersion factor for 2 hours at EAB, sec/m^3

Q(i) = noble gas release rate of the ith isotope for 2 hours calculated using the following equation, Bq

$$Q(i) = [R(i)_n + R(i)_a] \cdot MF(i)$$

$R(i)_n$ = gaseous effluent release rate due to normal operation for 2 hours, Bq

$R(i)_a$ = gaseous effluent release rate due to GRS failure for 2 hours, Bq

Dose consequence is calculated using the RADTRAD Code (Reference 33). As presented in Table 11.3-9, the calculated TEDE at the EAB is 0.0316 mSv, which is less than the acceptance criterion of 1 mSv specified in the Standard Review Plan, Section 11.3 (Reference 1).

The COL applicant is to perform an analysis using site-specific meteorological data to demonstrate that the potential airborne concentration resulting from GRS failure meets the requirements of 10 CFR 20, Appendix B, Table 2 (Reference 4) (COL 11.3(8)).

11.3.3.3 Offsite Dose Calculation Manual

The offsite dose calculation manual (ODCM) is prepared to describe the control and the calculation methods for the determination of offsite doses resulting from the gaseous effluent releases, and the responses and mitigation if the setpoints are exceeded. The COL applicant is to prepare an ODCM following the guidance in NEI 07-09A template (Reference 34) (COL 11.3(9)).

11.3.4 Testing and Inspection Requirements

The GRS is pneumatically pressure tested in conformance with Regulatory Position C.4.4 of NRC RG 1.143 (Reference 2). Testing of piping systems is performed in accordance with the applicable codes and standards as described in Table 11.3-2.

Inspections and tests are conducted for the periodic evaluation of system operability and performance in accordance with NRC RG 1.143 (Reference 2).

The GRS does not have a safety-related function and inservice inspection of the components is, therefore, not required.

11.3.4.1 Instrumentation Testing Requirements

Periodic tests and calibrations are performed on the analyzers and radiation monitor during normal operation. These tests and calibrations provide reasonable assurance that the instruments provide accurate data for safe operation, confidence that the radiation doses from the projected gaseous effluent releases are ALARA, and assurance that the explosive gas mixture concentration is less than the flammability limit. Sampling and analysis data can also be used to confirm the accuracy of the gas analyzers and the radiation monitor readings.

The discharge isolation valves are designed to close upon the receipt of a high-radiation signal and low or no flow for ventilation exhaust. This function can be verified by simulating the high-radiation signal and no-flow signal to the discharge valve so it closes automatically. Simulating the high-radiation alarm signal and no-flow signal confirms the proper operation of the discharge valve closure.

11.3.4.2 Preoperational Inspection

After installation, but prior to initial system operation, the GRS is tested to verify pressure integrity, design and normal flow conditions, instrumentation accuracy, and control operability. Gauges and instrumentation are to be calibrated for accurate readings.

11.3.5 Instrumentation Requirements

Table 11.3-8 provides a list of instrumentation for the GRS, including instrumentation that is used to measure oxygen and hydrogen concentrations and provide alarm functions to minimize the potential for explosion. Manual override capability of automatic controls is provided where necessary to maintain system operability. For the equipment operated manually, remote manual hand switches with status lights are provided for all frequently operated valves and components. The description of the radiation monitoring system interfaces with the MCR is provided in Section 11.5.

11.3.6 Combined License Information

COL 11.3(1) The COL applicant is to prepare and implement the epoxy inspection and maintenance program in the GRS.

COL 11.3(2) The COL applicant is to perform a site-specific cost-benefit analysis following the guidance in NRC RG 1.110 for conformance with 10 CFR 50 Appendix I.

COL 11.3(3) The COL applicant is to prepare and provide the piping and instrumentation diagram (P&ID) for the combined operating license application.

COL 11.3(4) The COL applicant is to prepare the operational procedures and maintenance programs related to leak detection and contamination control

COL 11.3(5) The COL applicant is to maintain complete documentation of system design, construction, design modifications, field changes, and operations for decommissioning planning.

COL 11.3(6) The COL applicant is to prepare the site process control program and the site radiological environmental monitoring program.

COL 11.3(7) The COL applicant is also to perform the dose calculation using the total gaseous effluents from the site for comparison with the requirements of 40 CFR 190.

COL 11.3(8) The COL applicant is to perform an analysis using site-specific meteorological data to demonstrate that the potential airborne concentration resulting from GRS failure meets the requirements of 10 CFR 20, Appendix B, Table 2.

COL 11.3(9) The COL applicant is to prepare an ODCM following the guidance in NEI 07-09A template.

11.3.7 References

1. NUREG-0800, Standard Review Plan, Section 11.3, “Gaseous Waste Management System,” Rev. 3, U.S. Nuclear Regulatory Commission, March 2007.

2. Regulatory Guide 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” Rev. 2, U.S. Nuclear Regulatory Commission, November 2001.

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

4. 10 CFR Part 20, Appendix B, “Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage,” U.S. Nuclear Regulatory Commission.

5. 10 CFR Part 50, Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low as is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents,” U.S. Nuclear Regulatory Commission.

6. Regulatory Guide 1.112, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors,” Rev. 1, U.S. Nuclear Regulatory Commission, March 2007.

7. ANSI/ANS 18.1-1999, “Radioactive Source Term for Normal Operation of Light-Water Reactors,” American Nuclear Society, 1999.

8. ANSI/ANS 55.4, “Gaseous Radioactive Waste Processing System for Light Water Reactor Plants,” American Nuclear Society, 1993.

9. Regulatory Guide 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Rev. 4, U.S. Nuclear Regulatory Commission, September 2012.

10. Regulatory Guide 1.140, “Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants,” Rev. 2, U.S. Nuclear Regulatory Commission, June 2001.

11. Regulatory Guide 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable,” Rev. 3, U.S. Nuclear Regulatory Commission, June 1978.

12. 10 CFR Part 50, Appendix A, General Design Criterion 3, “Fire Protection,” U.S. Nuclear Regulatory Commission.

13. 10 CFR Part 50, Appendix A, General Design Criterion 60, “Control of Releases of Radioactive Materials to the Environment,” U.S. Nuclear Regulatory Commission.

14. 10 CFR Part 50, Appendix A, General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," U.S. Nuclear Regulatory Commission.

15. 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases," U.S. Nuclear Regulatory Commission.

16. NUREG-0800, Standard Review Plan, BTP 11-5, "Postulated Radioactive Release Due to Waste System Leak or Failure," U.S. Nuclear Regulatory Commission, March 2007.

17. Regulatory Guide 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," Rev. 2, U.S. Nuclear Regulatory Commission, October 2010.

18. Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," U.S. Nuclear Regulatory Commission, June 2008.

19. IE Bulletin 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment," U.S. Nuclear Regulatory Commission, May 1980.

20. ASME Section II, "Material Specification," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

21. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Rev. 3, U.S. Nuclear Regulatory Commission, June 2013.

22. Regulatory Guide 1.189, "Fire Protection for Nuclear Power Plants," Rev. 2, U.S. Nuclear Regulatory Commission, October 2009.

23. 10 CFR 20.1406, "Minimization of Contamination," U.S. Nuclear Regulatory Commission.

24. Regulatory Guide 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors," Rev. 1, U.S. Nuclear Regulatory Commission, October 2013.

25. ASME Section VIII, "Rules for Construction of Pressure Vessels," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

26. NUREG/CR-4653, “GASPAR II - Technical Reference and User Guide,” U.S. Nuclear Regulatory Commission, March 1987.

27. Regulatory Guide 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I,” Rev. 1, U.S. Nuclear Regulatory Commission, October 1977.

28. Regulatory Guide 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors,” Rev. 1, U.S. Nuclear Regulatory Commission, July 1977.

29. 10 CFR 20.1302, “Compliance with Dose Limits for Individual Members of the Public,” U.S. Nuclear Regulatory Commission.

30. 40 CFR Part 190, “Environmental Radiation Protection Standards for Nuclear Power Operations,” Environmental Protection Agency.

31. DAMSAM, “A Digital Computer Program to Calculate Primary and Secondary Activity Transients,” Combustion Engineering, Inc.

32. USEPA Federal Guidance Report No. 12, “External Exposure to Radionuclides in Air, Water, and Soil,” 1993.

33. NUREG/CR-6604, “RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation,” U.S. Nuclear Regulatory Commission, June 1999.

34. NEI 07-09A, “Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description,” Rev. 0, Nuclear Energy Institute, March 2009.

35. NEI 08-08A, “Generic FSAR Template Guidance for Life Cycle Minimization of Contamination,” Rev. 0, Nuclear Energy Institute, October 2009.

Table 11.3-1 (1 of 2)

Expected Gaseous Radioactive Effluents During Normal Operation Including AOOs (Bq/yr)

Nuclide	Waste Gas System	Building Ventilation				Air Ejector Exhaust	Total
		Fuel Handling	Reactor	Auxiliary	Turbine		
I-131	0.00E+00	1.07E+07	3.15E+07	2.59E+07	0.00E+00	0.00E+00	6.66E+07
I-132	0.00E+00	3.03E+08	7.40E+08	7.40E+08	0.00E+00	0.00E+00	1.78E+09
I-133	0.00E+00	1.37E+08	3.70E+08	3.29E+08	0.00E+00	0.00E+00	8.51E+08
I-134	0.00E+00	4.81E+08	1.22E+09	1.22E+09	0.00E+00	0.00E+00	2.92E+09
I-135	0.00E+00	2.81E+08	7.40E+08	6.66E+08	0.00E+00	0.00E+00	1.70E+09
Kr-85m	0.00E+00	0.00E+00	2.59E+11	0.00E+00	0.00E+00	0.00E+00	2.59E+11
Kr-85	6.66E+13	0.00E+00	1.15E+14	9.25E+11	0.00E+00	4.44E+11	1.81E+14
Kr-87	0.00E+00	0.00E+00	7.40E+10	0.00E+00	0.00E+00	0.00E+00	7.40E+10
Kr-88	0.00E+00	0.00E+00	1.85E+11	0.00E+00	0.00E+00	0.00E+00	1.85E+11
Xe-131m	3.63E+12	0.00E+00	7.77E+13	7.03E+11	0.00E+00	3.33E+11	8.14E+13
Xe-133m	0.00E+00	0.00E+00	4.81E+12	7.40E+10	0.00E+00	0.00E+00	4.81E+12
Xe-133	0.00E+00	0.00E+00	2.59E+12	0.00E+00	0.00E+00	0.00E+00	2.59E+12
Xe-135m	0.00E+00	0.00E+00	1.48E+11	1.11E+11	0.00E+00	3.70E+10	2.96E+11
Xe-135	0.00E+00	0.00E+00	1.85E+12	3.70E+10	0.00E+00	0.00E+00	1.89E+12
Xe-137	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.11E+11	1.11E+11
Xe-138	0.00E+00	0.00E+00	7.40E+10	3.70E+10	0.00E+00	0.00E+00	1.11E+11
Cr-51	5.18E+03	6.66E+04	3.40E+06	1.18E+05	0.00E+00	0.00E+00	3.59E+06
Mn-54	7.77E+02	1.11E+05	1.96E+06	2.89E+04	0.00E+00	0.00E+00	2.11E+06
Co-57	0.00E+00	0.00E+00	3.03E+05	0.00E+00	0.00E+00	0.00E+00	3.03E+05
Co-58	3.22E+03	7.77E+06	9.25E+06	7.03E+05	0.00E+00	0.00E+00	1.78E+07
Co-60	5.18E+03	3.03E+06	9.62E+05	1.89E+05	0.00E+00	0.00E+00	4.07E+06
Fe-59	6.66E+02	0.00E+00	9.99E+05	1.85E+04	0.00E+00	0.00E+00	1.04E+06
Sr-89	1.63E+04	7.77E+05	4.81E+06	2.78E+05	0.00E+00	0.00E+00	5.92E+06
Sr-90	6.29E+03	2.96E+05	1.92E+06	1.07E+05	0.00E+00	0.00E+00	2.33E+06
Zr-95	1.78E+03	1.33E+03	0.00E+00	3.70E+05	0.00E+00	0.00E+00	3.70E+05

Table 11.3-1 (2 of 2)

Nuclide	Waste Gas System	Building Ventilation				Air Ejector Exhaust	Total
		Fuel Handling	Reactor	Auxiliary	Turbine		
Nb-95	1.37E+03	8.88E+05	6.66E+05	1.11E+04	0.00E+00	0.00E+00	1.55E+06
Ru-103	1.18E+03	1.41E+04	5.92E+05	8.51E+03	0.00E+00	0.00E+00	6.29E+05
Ru-106	9.99E+02	2.55E+04	0.00E+00	2.22E+03	0.00E+00	0.00E+00	2.89E+04
Sb-125	0.00E+00	2.11E+04	0.00E+00	1.44E+03	0.00E+00	0.00E+00	2.26E+04
Cs-134	1.22E+04	6.29E+05	9.25E+05	2.00E+05	0.00E+00	0.00E+00	1.78E+06
Cs-136	1.96E+03	0.00E+00	1.18E+06	1.78E+04	0.00E+00	0.00E+00	1.22E+06
Cs-137	2.85E+04	9.99E+05	2.04E+06	2.66E+05	0.00E+00	0.00E+00	3.33E+06
Ba-140	8.51E+03	0.00E+00	0.00E+00	1.48E+05	0.00E+00	0.00E+00	1.55E+05
Ce-141	8.14E+02	1.63E+02	4.81E+05	9.62E+03	0.00E+00	0.00E+00	4.81E+05
H-3							5.92E+12
C-14							2.70E+11
Ar-41							1.26E+12

Table 11.3-2

Equipment Codes and Standards for Radwaste Equipment
(from NRC RG 1.143, Table 1)

Equipment	Design and Fabrication	Material	Welder Qualifications and Procedures	Inspection and Testing
Pressure Vessels	ASME Sec. VIII, Div. 1 or Div. 2	ASME Sec. II	ASME Sec. IX	ASME Sec. VIII, Div. 1 or Div. 2
Piping and Valves	ASME B31.3	ASME Sec. II	ASME Sec. IX	ASME B31.3
Filters	ASME Sec. VIII, Div. 1	ASME Sec. II	ASME Sec. IX	ASME Sec. VIII, Div. 1
Heat Exchanger	TEMA STD, 8th Edition; ASME Sec. VIII, Div. 1 or Div. 2	ASTM B359-98 or ASME Sec. II	ASME Sec. IX	ASME Sec. VIII, Div. 1 or 2

Table 11.3-3

System Design Parameters

Design Parameter	Design Value
Xenon delay	45 days
Dynamic adsorption coefficient, Kd for Xe	263 cc/g
Dynamic adsorption coefficient, Kd for Kr	18.7 cc/g
Maximum gaseous waste stream temp, °C (°F)	60 (140)
Charcoal temperature, °C (°F)	10 to 40 (50 to 104)
Plant chilled water temperature, °C (°F)	5.8 (42.5)
Component cooling water (CCW) temperature, °C (°F)	43.3 (110)
Minimum activated charcoal ignition temperature, °C (°F)	156.1 (313)
Gas flow range, L/min (scfm)	0 to 28.3 (0 to 1.2)
Charcoal bed vault temperature range, °C (°F)	10 to 40 (50 to 104)
Charcoal particle size	6-12 mesh (USS) with 90 to 100 percent retention
Charcoal moisture content	2.0 percent maximum

Table 11.3-4 (1 of 2)

GRS Major Equipment Design Information

Tanks	
Equipment name Quantity (each) Design capacity, L (ft^3) Design pressure, kg/cm^2 (psig) Design temperature, °C (°F) [[Material]] Radwaste safety class	Header drain tank 1 566 (20) 10.5 (150) 93.3 (200) Stainless steel RW-IIa
Equipment name Quantity (each) Total mass of charcoal, kg (lbm) Design flow, L/min (scfm) Design pressure, kg/cm^2 (psig) Design temperature, °C (°F) [[Material]] Radwaste safety class	Charcoal guard bed (GRS package) 2 272 (600) 623 (22) 10.5 (150) 93.3 (200) Stainless steel RW-IIc
Equipment name Quantity (each) Total mass of charcoal, kg (lbm) Design flow, L/min (scfm) Design pressure, kg/cm^2 (psig) Design temperature, °C (°F) [[Material]] Radwaste safety class	Charcoal delay bed (GRS package) 4 9,525 (21,000) 57 (2) 10.5 (150) 93.3 (200) Carbon steel RW-IIa

Table 11.3-4 (2 of 2)

Heat Exchanger	
Equipment name	HEPA filter (GRS package)
Quantity (each)	1
Size, µm	0.3
Efficiency, %	99.97 (1)
Design flow, L/min (scfm)	2,548 (90)
Design pressure, kg/cm^2 (psig)	10.5 (150)
Design temperature, °C (°F)	93.3 (200)
[[Material]]	Stainless steel
Radwaste safety class	RW-IIa
Equipment name	Waste gas dryer (GRS package)
Quantity (each)	2
Design flow, L/min (scfm)	623 (22)
Outlet temperature, °C (°F)	7.8 (46)
Design pressure, kg/cm^2 (psig)	10.5 (150)
Design temperature, °C (°F)	93.3 (200)
[[Material]]	Stainless steel
Radwaste safety class	RW-IIa

(1) Efficiency of this HEPA filter is not credited in the PWR-GALE Code calculation.

Table 11.3-5

Input Parameters Used for GASPAR II Code

Parameter	Value	Basis
Midpoint of plant life (years)	30	Half of the APR1400 design life of 60 years
Distance to site boundary (m)	700	Assumed. For DC application, no specific distance to site boundary is defined. Doses are calculated using the χ/Q values given below.
Fraction of the year that leafy vegetables are grown	1.0	Since there is no site-specific information for DC application, default values in RG 1.109 are used.
Fraction of the year that milk cows are grown	1.0	
Fraction of the maximum individual's vegetable intake that is from his own garden	0.76	
Fraction of milk-cow feed intake that is from pasture while on pasture.	1.0	
Average absolute humidity over the growing season (g/m^3)	8.0	
Fraction of the year that beef cattle are on pasture	1.0	
Fraction of beef-cattle feed intake that is from pasture while the cattle are on pasture	1.0	
Source term multiplier to be applied to each radionuclide release	1.0	
Source terms	See Table 11.3-1	Based on PWR-GALE code calculation
Milk pathway considered	Goat	Assumed
χ/Q at EAB (sec/m^3)	2.0E-05	See Table 2.0-1
χ/Q at food production area (sec/m^3)	1.0E-05	Assumed that in food production area, χ/Q will be reduced by half due to the distance from the site boundary
D/Q (m^{-2})	2.0E-07	See Table 2.0-1
Other parameters	Default values in RG 1.109	Since there is no site-specific information for DC application, default values in RG 1.109 are used.

Table 11.3-6 (1 of 2)

Design Basis Gaseous Effluent Concentration at the Site Boundary

Nuclide	Design Basis Release[1] (Bq/yr)	Effluent Concentration (Bq/m^3)	10 CFR 20 Appendix B Limits (Bq/m^3)	Ratio
H-3	2.08E+13	1.32E+01	3.70E+03	3.56E-03
C-14	2.70E+11	1.71E-01	1.11E+04	1.54E-05
Ar-41	1.26E+12	7.98E-01	3.70E+02	2.16E-03
I-131	3.62E+09	5.15E-02	7.40E+00	6.96E-03
I-132	6.00E+09	1.30E-02	7.40E+02	1.76E-05
I-133	2.17E+10	7.32E-02	3.70E+01	1.98E-03
I-134	3.87E+09	8.14E-03	2.22E+03	3.67E-06
I-135	1.33E+10	3.90E-02	2.22E+02	1.76E-04
Kr-85m	1.32E+12	8.11E+00	3.70E+03	2.19E-03
Kr-85	3.04E+12	1.15E+02	2.59E+04	4.44E-03
Kr-87	3.91E+11	2.13E+00	7.40E+02	2.88E-03
Kr-88	1.33E+12	1.27E+01	3.33E+02	3.82E-02
Xe-131m	8.14E+13	1.15E+01	7.40E+04	1.55E-04
Xe-133m	4.81E+12	3.05E+00	2.22E+04	1.37E-04
Xe-133	2.37E+13	1.31E+03	1.85E+04	7.09E-02
Xe-135m	1.42E+12	8.74E-01	1.48E+03	5.91E-04
Xe-135	7.79E+12	6.09E+01	2.59E+03	2.35E-02
Xe-137	4.93E+11	3.04E-01	-	-
Xe-138	4.93E+11	5.98E-01	7.40E+02	8.08E-04
Cr-51	1.55E+07	9.85E-06	1.11E+03	8.88E-09
Mn-54	2.11E+06	1.34E-06	3.70E+01	3.61E-08
Co-57	3.03E+05	1.92E-07	3.33E+01	5.78E-09
Co-58	1.78E+07	1.13E-05	3.70E+01	3.04E-07
Co-60	4.07E+06	2.58E-06	1.85E+00	1.39E-06
Fe-59	1.04E+06	6.57E-07	1.85E+01	3.55E-08

Table 11.3-6 (2 of 2)

Nuclide	Design Basis Release[1] (Bq/yr)	Effluent Concentration (Bq/m^3)	10 CFR 20 Appendix B Limits (Bq/m^3)	Ratio
Sr-89	1.34E+08	8.51E-05	7.40E+00	1.15E-05
Sr-90	4.21E+07	2.67E-05	2.22E-01	1.20E-04
Zr-95	5.57E+05	3.53E-07	1.48E+01	2.39E-08
Nb-95	2.77E+06	1.76E-06	7.40E+01	2.38E-08
Ru-103	6.29E+05	3.99E-07	3.33E+01	1.20E-08
Ru-106	2.89E+04	1.83E-08	7.40E-01	2.47E-08
Sb-125	2.26E+04	1.43E-08	2.59E+01	5.52E-10
Cs-134	7.89E+07	9.60E-03	7.40E+00	1.30E-03
Cs-136	6.00E+07	3.80E-05	3.33E+01	1.14E-06
Cs-137	1.29E+08	1.45E-02	7.40E+00	1.96E-03
Ba-140	1.55E+05	9.85E-08	7.40E+01	1.33E-09
Ce-141	4.81E+05	3.05E-07	2.96E+01	1.03E-08
			SUM	1.62E-01

(1) The design basis release rates are adjusted from the expected values in Table 11.3-1 using multiplication factors that are the ratios of design basis RCS concentrations to expected concentrations. If a multiplication factor is less than 1, a value of 1 is used for conservatism.

Table 11.3-7 (1 of 3)

Maximum Offsite Individual Dose Resulting
from Normal Plant Gaseous Releases

Type of Dose	Estimated Dose
Annual beta air dose (mGy/yr)	7.94E-02
Annual gamma air dose (mGy/yr)	6.13E-03
Dose to total body (mSv/yr)	3.79E-03
Dose to skin (mSv/yr)	5.57E-02

Table 11.3-7 (2 of 3)

	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Plume	3.47E-03							5.53E-02
Ground	3.19E-04	3.19E-04	3.19E-04	3.19E-04	3.19E-04	3.19E-04	3.19E-04	3.74E-04
Vegetable								
Adult		5.03E-03	2.46E-02	4.97E-03	4.92E-03	1.15E-02	4.86E-03	
Teen		7.75E-03	3.88E-02	7.72E-03	7.64E-03	1.57E-02	7.56E-03	
Child		1.77E-02	9.19E-02	1.78E-02	1.77E-02	3.28E-02	1.76E-02	
Meat								
Adult		1.70E-03	7.84E-03	1.66E-03	1.66E-03	1.93E-03	1.65E-03	
Teen		1.40E-03	6.61E-03	1.38E-03	1.37E-03	1.57E-03	1.37E-03	
Child		2.55E-03	1.24E-02	2.55E-03	2.54E-03	2.85E-03	2.54E-03	
Goat milk								
Adult		2.18E-03	9.07E-03	2.43E-03	2.29E-03	1.23E-02	2.18E-03	
Teen		3.77E-03	1.66E-02	4.21E-03	3.97E-03	1.99E-02	3.78E-03	
Child		8.67E-03	4.06E-02	9.46E-03	9.04E-03	4.11E-02	8.71E-03	
Infant		1.75E-02	7.82E-02	1.92E-02	1.82E-02	9.64E-02	1.77E-02	
Inhalation								
Adult		7.54E-04	7.54E-04	7.54E-04	7.54E-04	7.54E-04	7.54E-04	
Teen		7.54E-04	7.54E-04	7.54E-04	7.54E-04	7.54E-04	7.54E-04	

Table 11.3-7 (3 of 3)

	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Child		6.53E-04	2.05E-05	6.58E-04	6.62E-04	1.64E-03	6.66E-04	
Infant		3.75E-04	1.06E-05	3.81E-04	3.81E-04	1.27E-03	3.85E-04	
Total								
Adult	3.79E-03	9.96E-03	4.18E-02	1.01E-02	9.93E-03	2.74E-02	9.75E-03	5.57E-02
Teen	3.79E-03	1.40E-02	6.23E-02	1.44E-02	1.40E-02	3.90E-02	1.38E-02	5.57E-02
Child	3.79E-03	2.99E-02	1.45E-01	3.08E-02	3.03E-02	7.87E-02	2.98E-02	5.57E-02
Infant	3.79E-03	1.82E-02	7.85E-02	1.99E-02	1.89E-02	9.80E-02	1.84E-02	5.57E-02

Table 11.3-8

Instrument Indication and Alarm Information Page

Equipment	Parameter	Record	Indicate	Alarm		Auto Control
				High	Low	
Waste gas dryer	Outlet cooling water					
	Temperature		×			
	Effluent gas temperature		×	×		
Gas drying or moisture removal	Outlet gas moisture content	×		×		
Charcoal guard bed	Bed temperature		×	×		
Charcoal delay beds	Inlet gas moisture		×	×		
	Inlet gas temperature		×	×		
	Outlet gas temperature		×	×		
System gas analyzers	H_2 concentration (% volume)	×	×			
	O_2 concentration (% volume)	×	×	×		×
System discharge line	Radiation	×	×	×		×
System	Gas flow rate – inlet	×	×			
	Gas flow rate – outlet	×	×			

× = required

Table 11.3-9

Gaseous Radwaste System Failure Doses

Radioactivity Release Path	Dose (mSv)	
	EAB	LPZ
Tank Release to Environment	3.16E-02	6.950E-03
Allowable Dose Limit	1.00E+00	1.00E+00

Table 11.3-10

Expected Radioactive Source Terms for GRS Components

Nuclide	Inlet (Bq/cm^3)	Buildup Activity on Charcoal Bed (Bq) 1st Delay Bed	2nd Delay Bed	3rd Delay Bed	4th Delay Bed	Outlet (Bq/cm^3)
Kr-85m	1.83E+04	2.09E+11	8.12E+09	3.15E+08	1.22E+07	4.17E-02
Kr-85	1.38E+06	5.30E+13	5.30E+13	5.30E+13	5.30E+13	1.38E+06
Kr-87	1.92E+04	6.47E+10	6.92E+05	7.39E+00	7.90E-05	2.51E-16
Kr-88	2.02E+04	1.51E+11	8.94E+08	5.30E+06	3.15E+04	2.50E-05
Xe-131m	1.00E+06	3.65E+14	1.90E+14	9.85E+13	5.11E+13	7.31E+04
Xe-133m	8.31E04	1.12E+13	3.19E+11	9.07E+09	2.58E+08	5.42E-02
Xe-133	3.66E+04	9.43E+12	2.13E+12	4.82E+11	1.09E+11	9.56E+01
Xe-135m	1.46E+05	9.83E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	7.67E+04	1.85E+12	2.11E+03	2.41E-06	2.75E-15	1.30E-31
Xe-137	3.82E+04	6.45E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-138	6.92E+04	4.33E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-84	2.21E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-131	3.08E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-132	8.39E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	3.86E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-134	1.43E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-135	7.87E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

Table 11.3-11 (1 of 2)

Design Basis Radioactive Source Terms for GRS Components (1 % Fuel Defect)

Nuclide	At Inlet (Bq/cm^3)	Buildup Activity on Charcoal Bed (Bq)					At Outlet (Bq/cm^3)	Header Drain Tank (Bq)	Waste Gas Dryer (Bq)
		Guard Bed	1st Delay Bed	2nd Delay Bed	3rd Delay Bed	4th Delay Bed			
Kr-85m	9.08E+05	2.34E+10	1.04E+13	4.02E+11	1.56E+10	6.06E+08	2.06E+00	5.14E+11	4.69E+12
Kr-85	2.27E+04	5.84E+08	8.75E+11	8.75E+11	8.75E+11	8.75E+11	2.27E+04	1.29E+10	1.17E+11
Kr-87	8.90E+05	2.29E+10	3.00E+12	3.20E+07	3.42E+02	3.66E-03	1.16E-14	5.04E+11	4.59E+12
Kr-88	2.27E+06	5.84E+10	1.70E+13	1.01E+11	5.97E+08	3.54E+06	2.81E-03	1.29E+12	1.17E+13
Xe-131m	2.27E+05	5.84E+09	8.25E+13	4.29E+13	2.23E+13	1.16E+13	1.65E+04	1.29E+11	1.17E+12
Xe-133m	5.81E+04	1.50E+09	7.86E+12	2.23E+11	6.34E+09	1.80E+08	3.79E-02	3.29E+10	3.00E+11
Xe-133	2.91E+07	7.48E+11	7.50E+15	1.70E+15	3.83E+14	8.67E+13	7.60E+04	1.65E+13	1.50E+14
Xe-135m	6.99E+05	1.80E+10	4.72E+11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.96E+11	3.61E+12
Xe-135	4.00E+06	1.03E+11	9.62E+13	1.10E+05	1.25E-04	1.43E-13	6.78E-30	2.26E+12	2.06E+13
Xe-137	1.73E+05	4.44E+09	2.92E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	9.77E+10	8.93E+11
Xe-138	5.99E+05	1.54E+10	3.75E+11	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.40E+11	3.09E+12
Br-84	2.88E-01	4.04E+05	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.63E+05	1.49E+06
I-131	3.66E+01	1.87E+10	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.07E+07	1.89E+08
I-132	9.84E+00	5.99E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	5.58E+06	5.08E+07
I-133	5.22E+01	2.87E+09	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.95E+07	2.69E+08
I-134	6.17E+00	1.43E+07	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.49E+06	3.18E+07
I-135	2.88E+01	5.03E+08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.63E+07	1.49E+08

Table 11.3-11 (2 of 2)

Nuclide	At Inlet (Bq/cm^3)	Buildup Activity on Charcoal Bed (Bq)					At Outlet (Bq/cm^3)	Header Drain Tank (Bq)	Waste Gas Dryer (Bq)
		Guard Bed	1st Delay Bed	2nd Delay Bed	3rd Delay Bed	4th Delay Bed			
Sum of Fractions									
$\sum A_i/A_{1i}$	-	2.01E-01	4.26E+02	8.63E+01	1.98E+01	4.17E+00	-	4.17E+00	3.80E+01
$\sum A_i/A_{2i}$	-	2.82E-01	8.19E+02	1.71E+02	3.90E+01	9.05E+00	-	5.48E+00	5.00e+01
Radwaste Classification									
	-	RW-IIc	RW-IIa	RW-IIa	RW-IIa	RW-IIa	-	RW-IIa	RW-IIa

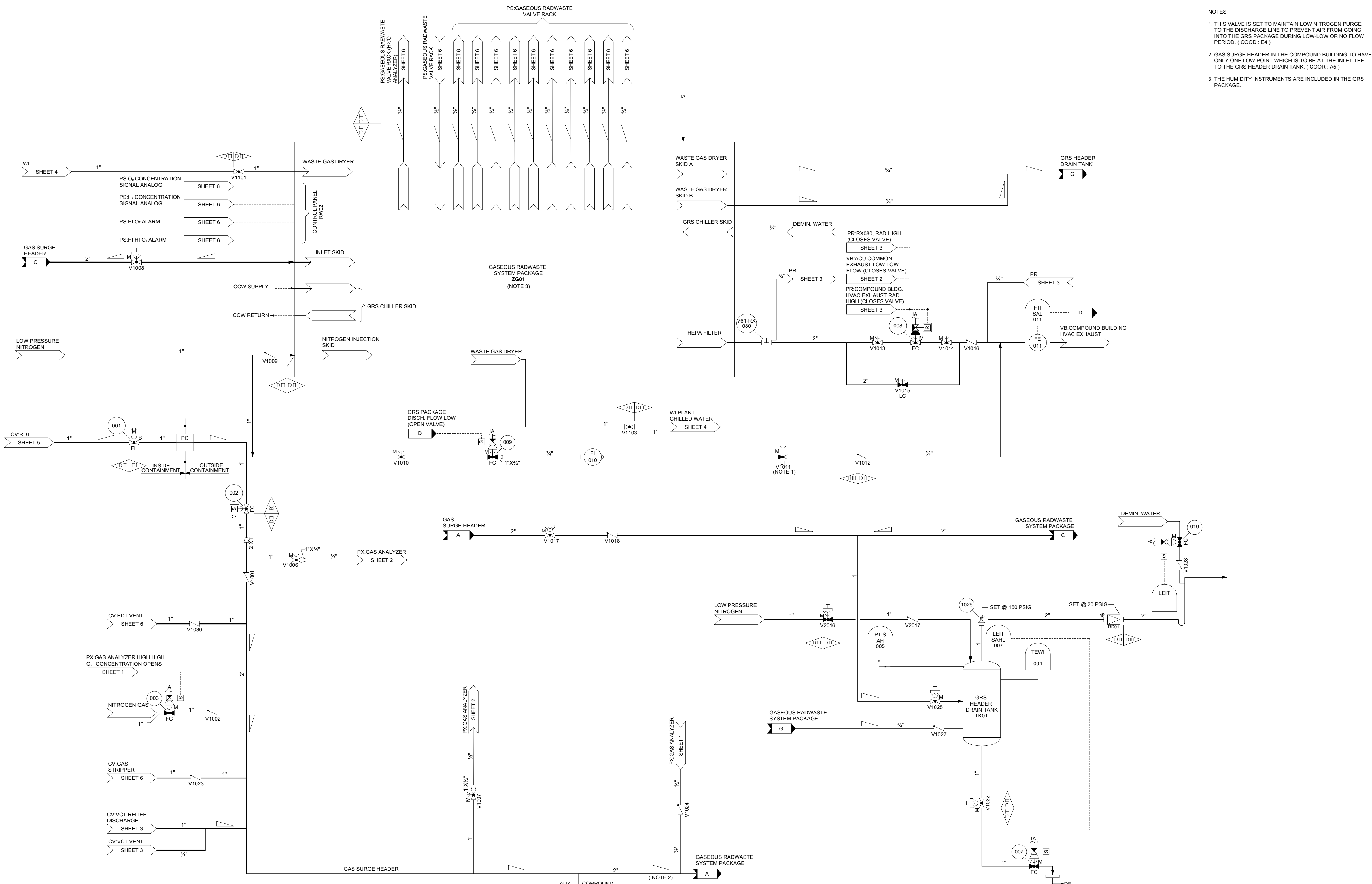


Figure 11.3-1 Gaseous Radwaste System Flow Diagram

EXHAUST NO.	INSIDE DIA.(Shape) (INCH)	FLOW (CFM)	VELOCITY (FPM)	TEMP (°F)	REMARK
1	60X48 (Rectangular)	1,500	75	60~135	LOW VOLUME
	60X48 (Rectangular)	54,000	2,700	50~104	HIGH VOLUME
2	24X20 (Rectangular)	9,000	2,700	50~113	HELB AREA
3	54X36 (Rectangular)	38,000	2,815	50~113	CONTROLLED AREA I NORMAL
4	120X72 (Rectangular)	36,150	679	50~113	CONTROLLED AREA II NORMAL
5	36X20 (Rectangular)	3,000	600	50~113	CONTROLLED AREA I EMERGENCY
6	36X20 (Rectangular)	3,000	600	50~113	CONTROLLED AREA II EMERGENCY
7	42X32 (Rectangular)	28,450	3,048	50~104	FUEL HANDLING AREA NORMAL
	42X32 (Rectangular)	5,000	536	50~136	FUEL HANDLING AREA EMERGENCY
8	80X54 (Rectangular)	68,540	2,285	50~104	COMPOUND BLDG HEPA FILTER EXHAUST
	80X54 (Rectangular)	68,540	2,285	50~108	COMPOUND BLDG CARBON ADSORBER EXHAUST
9	21.25Ø (Circular)	13,039 (lb/hr)	1,522	71~291	

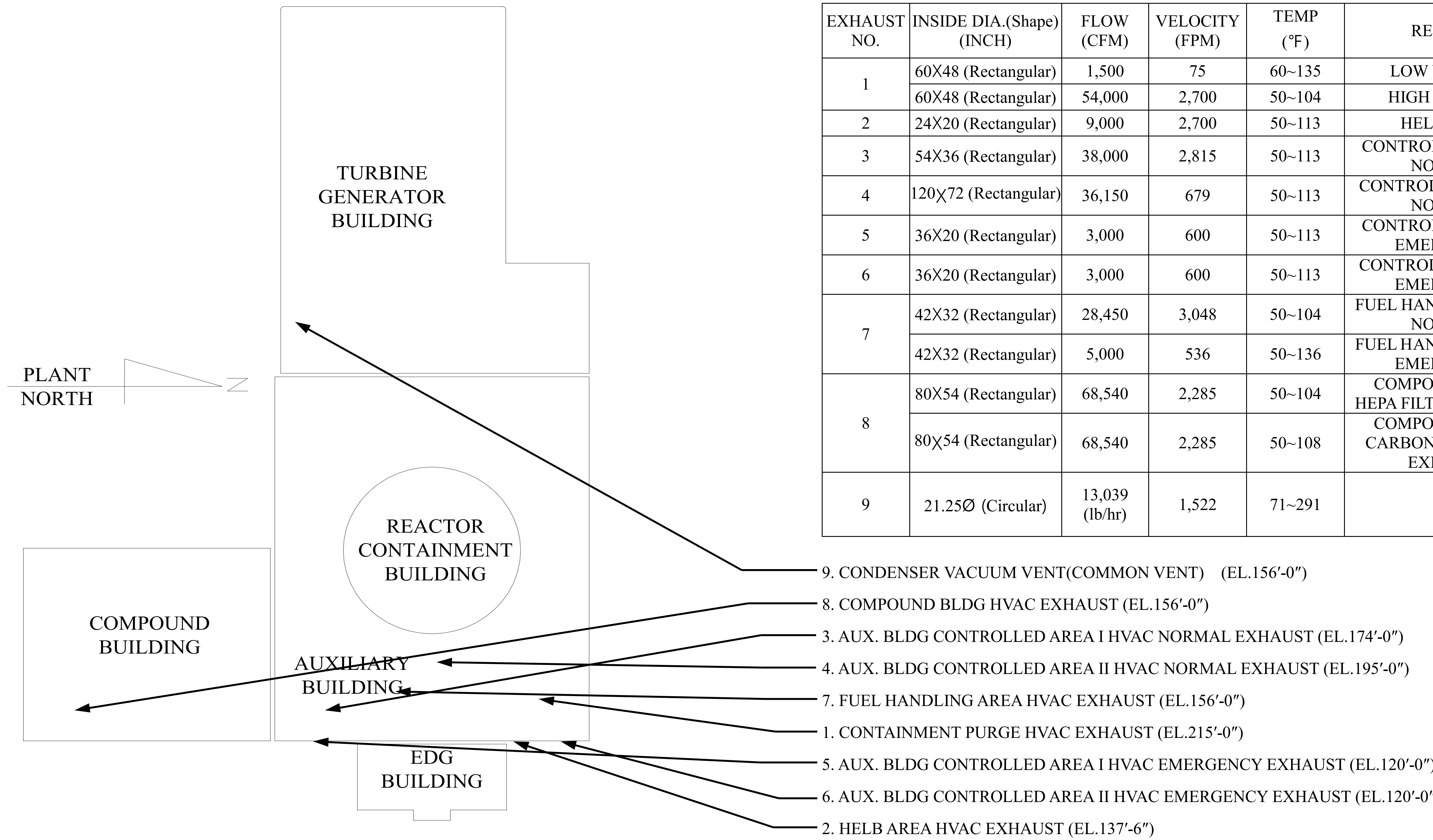


Figure 11.3-2 Gaseous Effluent Release Points and Exhaust Parameters

11.4 Solid Waste Management System

The solid waste management system (SWMS) is designed to provide the means to collect, segregate, store, process, and pack the various types of solid radioactive waste, and prepare it for offsite shipment. The SWMS processes both wet solid waste and dry active waste (DAW) for onsite temporary storage and shipment to the offsite disposal facility.

The SWMS design is supplemented with operating procedures, programs, and operator actions to provide assurance that the SSC integrity and functions are maintained and that the packaged wastes meet transportation requirements and waste acceptance criteria from offsite disposal sites.

The SWMS is designed with hard piping between radioactive and nonradioactive systems in accordance with IE Bulletin 80-10 (Reference 1). Demineralized water is used for flushing the lines after each transfer of contaminated fluid. Such connections are hard pipes and are equipped with double barriers to prevent unintended contamination in accordance with NRC RG 4.21 (Reference 2).

The lessons learned program provides guidance on the integration of industry, operating, and construction experience into the APR1400 design. Under this program, NRC generic communications and industry operating and construction experience are maintained in a database that is reviewed, assessed, and integrated into the design as appropriate. The construction and operating experience of nuclear power plants has been incorporated into the database for design improvement.

11.4.1 Design Bases

11.4.1.1 Design Objectives

The design objectives of the SWMS are:

a. To collect, segregate, process, package, and store various solid radioactive wastes generated from the normal operation, maintenance, refueling, and anticipated operational occurrences (AOOs) in accordance with 10 CFR 50, Appendix A, GDC 60 (Reference 3).

b. To store, process, and package the radioactive spent resins transferred from the liquid waste management system (LWMS), the chemical and volume control system (CVCS), the spent fuel pool cooling and cleanup system (SFPCCS), and the steam generator blowdown system (SGBDS).

c. To temporarily store the high- and low-activity waste, and to retrieve and ship wastes.

d. To process and package wastes into drums, high-integrity containers (HICs), or other containers that satisfy the regulations of the U.S. Department of Transportation (DOT) and the requirements from the disposal facility.

e. To satisfy federal regulations and protect the workers and the general public by maintaining radiation exposures ALARA.

11.4.1.2 Design Criteria

The SWMS design criteria are as follows:

a. The SWMS is designed to meet the requirements of 10 CFR 50, Appendix A, GDC 61 (Reference 4), to provide reasonable assurance of adequate safety under normal and postulated accident conditions by providing adequate shielding and appropriate containment and confinement features, and GDC 63 (Reference 5), such that the SWMS has the ability to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. The radiation monitoring system is described in Section 12.3.

   Spent resin is sampled for analysis, and the volume to be transferred into the HIC is predetermined. After the filling operation, the radiation level of the container is monitored prior to offsite shipment, providing reasonable assurance that the containers meet regulatory radiation limit and waste acceptance criteria, achieving conformance with 10 CFR 50, Appendix A, GDC 64 (Reference 6).

b. Liquid and gaseous effluents arising from the operation of the SWMS are within the effluent concentration limits of 10 CFR 20, Appendix B (Reference 7). To comply with GDC 60 (Reference 3), the liquid and gaseous wastes from the SWMS during normal operation and AOOs are processed through the LWMS, which is addressed in Section 11.2, and the compound building heating, ventilation, and air conditioning (HVAC) system, which is addressed in Section 9.4. Sections 11.2 and 11.3 provide estimates of releases from the LWMS and the GWMS based on the methodology in NUREG-0017 (Reference 8), which includes contributions from the SWMS.

c. Solid waste is classified, processed, and disposed of in accordance with the requirements of 10 CFR 61 (Reference 9) and packaged in accordance with the requirements of 10 CFR 71 (Reference 10). For the transportation and disposal of waste, the SWMS conforms with the transportation requirements of 49 CFR 173, Subpart I (Reference 11).

d. Spent resins in disposal containers are sampled and surveyed to verify that the dewatering process is completed in accordance with the guidance of NRC Branch Technical Position (BTP) 11-3 (Reference 12).

e. The SWMS is designed to stabilize solid waste into a suitable form for disposal or package the solid waste in drums or HICs approved for disposal. The drums and HICs meet the requirement of 49 CFR 171 (Reference 13).

f. Spent resin storage tanks (SRSTs) are sized to collect waste inputs to allow satisfactory operation of the solid waste processing system. The size of the components of the spent resin subsystem allows at least a 30-day decay of short-lived radionuclides prior to processing in accordance with ANSI/ANS 55.1 (Reference 14). Spent resin tanks have the capacity to hold at least two batches of spent resin from the source of the greatest volumetric input (see Tables 11.4-1 and 11.4-4).

g. The SWMS is designed to operate independently without interfering with the normal operation, including AOOs, of other plant systems.

h. The SWMS design conforms with NRC RG 1.143 (Reference 15). The compound building where radioactive waste management equipment is located is designed as Seismic Category II, and the design of radioactive waste management SSCs follows the guidance in NRC RG 8.8 (Reference 16). Demonstration of conformance with NRC RG 8.8 (Reference 16) is provided in Chapter 12.

i. To provide reasonable assurance of adequate safety per GDC 61 (Reference 4), the design of the shielding of the solid waste areas is based on the design basis source terms presented in Section 11.1. The spent filters and spent resins are assumed to be fully loaded with suspended solids and dissolved solids using the design basis source term and 1 year of loading operation, which is also used to determine the thicknesses of the shield walls.

j. Per NRC RG 4.21 (Reference 2), cubicles containing radioactive liquid are lined with an epoxy coating to minimize the potential for contamination of the groundwater system and to facilitate maintenance and decontamination. Epoxy coatings in the SWMS component cubicles are Service Level II coatings as defined in NRC RG 1.54 (Reference 17).

11.4.1.3 Other Design Considerations

The following features provide additional design considerations:

a. The SWMS provides adequate space for the packaging of wastes. Spaces are available to accommodate additional future waste packaging equipment to meet site-specific needs and the requirements for waste disposal.

b. Active and replaceable components are installed in locations that are accessible by crane or monorail hoist to facilitate removal, repair, and replacement.

c. The SWMS is designed, manufactured, and tested in accordance with the codes and standards listed in Table 1 of NRC RG 1.143 (Reference 15), as shown in Table 11.4-5.

d. In accordance with Regulatory Position C.3.2 of NRC RG 1.143 (Reference 15), materials used for pressure-retaining portions in the SWMS are designed in accordance with requirements in ASME Section II (Reference 18). Materials used in the SWMS are compatible with the chemical, physical, and radioactive environment during normal and anticipated operating conditions. Malleable wrought or cast iron and plastics are not used in the SWMS.

e. The permanently installed components in the SWMS that are used to store, collect, and process wet solid waste and DAW are designed to conform with the seismic criteria specified in Regulatory Position C.6 of NRC RG 1.143 (Reference 15). The two SRSTs are categorized as RW-IIa (See Table 11.4-3) and are designed accordingly. The NRC RG 1.143 seismic criteria are not applicable for mobile equipment.

f. To avoid any uncontrolled or unmonitored releases, the SWMS is designed to prevent liquid and gases generated during processing the wet solid waste and DAW from direct discharge to the environment. The SWMS is designed to route liquids removed during the dewatering process of wet solid waste to the LWMS for processing. Vents collected in the SWMS areas are filtered, monitored, and discharged to the environment via the compound building HVAC system.

g. Sufficient space to facilitate access, operation, inspection, testing, and maintenance is provided in the compound building to maintain personnel exposures ALARA in accordance with NRC RG 8.8 (Reference 16) guidelines.

h. The quality assurance (QA) program for the design, installation, procurement, and fabrication of components conforms with Regulatory Position C.7 of NRC RG 1.143 (Reference 15) and NRC RG 1.33 (Reference 19). In addition, the seismic category and quality group classifications applicable to the design of the SWMS are described in Section 3.2.

i. Piping in the SWMS is hydrostatically pressure tested in accordance with Regulatory Position C.4.4 of NRC RG 1.143 (Reference 15). Testing of piping systems during the operation phase is performed in accordance with ASME B31.3 (Reference 20) or system piping specifications.

j. The SWMS is designed to allow periodic testing to evaluate operability of components in accordance with Regulatory Position C.4.5 of NRC RG 1.143 (Reference 15).

k. The SWMS is designed to conform with GDC 60, GDC 61 (Reference 3 and 4), 10 CFR 50.34a (Reference 21), 10 CFR 50, Appendix I (Reference 22), and NRC RG 1.110 (Reference 23) by maintaining control over the release of liquid and gaseous radioactive materials during the processing of solid wastes through the use of appropriate technologies.

l. The SWMS is designed to meet the applicable requirements of NUREG-0800 11.4, and Branch Technical Position (BTP) 11-3 (Reference 12).

m. To further reduce the potential radiation exposure to operators, many of the SWMS functions are designed to be remotely controlled from the radwaste control room of the compound building, which also allows operators to more effectively coordinate operating activities.

n. The SWMS is designed to reduce the volume of DAW by implementing appropriate sorting, segregation, and compressing of waste prior to shipping or storing in the disposal containers.

o. Sufficient temporary storage is provided to store packaged solid waste for at least 30 days in accordance with ANSI/ANS 55.1 (Reference 14).

p. The current design provides for the collection and packaging of potentially contaminated clothing for offsite shipment and/or processing. Depending on site-specific requirements, the COL applicant can incorporate an onsite laundry facility for processing of contaminated clothing (COL 11.4(1)).

11.4.1.4 Method of Treatment

To treat and handle the different waste types including spent resins, spent filter cartridges, reverse osmosis (R/O) membranes, R/O concentrates, mixed wastes, and DAW, the SWMS provides separate treatment and handling methods.

To reduce the volume of DAW, the waste is compressed and packaged in a disposal container. Potentially contaminated air generated during DAW operations is filtered, monitored, and discharged through the compound building HVAC system. The dry active waste includes the following:

a. Contaminated clothing, gloves, rags, and shoe coverings

b. Compressible materials such as HVAC filters

c. Contaminated metallic materials and incompressible solid objects such as contaminated wood, small tools, and equipment or subcomponents

To meet 10 CFR 61.56 (Reference 24), wet solid wastes are stabilized or dewatered and packaged in a 200 L (DOT-17H 55 gal) drum or HIC. Wet solid wastes are collected based on the radiation level of waste and the radiation is monitored before processing. The major wet solid wastes are spent resins, spent filters, R/O membranes, R/O concentrates, and mixed wastes. The methods of treatment of wet wastes are as follows:

a. Two SRSTs are used to collect and store the spent resins generated during operation, one for high-activity spent resin from the CVCS purification demineralizers, and the other for low-activity spent resin from the LWMS and SFP. Spent resin from the SGBD treatment demineralizers is also transferred to the low-activity spent resin storage tank when the resin is determined to be contaminated. The high-activity spent resin storage tank is designed to provide sufficient storage for 10 years but to facilitate long-term decay (from 0 to 9 years) before the resin is packaged and shipped. The spent resins collected in the SRSTs are packaged in HICs and dewatered by the dewatering subsystem.

b. Wet spent filters are dewatered and packaged in a 200 L (55 gal) drum (or HIC) with other DAW acting as absorbent materials. The disposal container is then placed in the shielded temporary waste storage area prior to shipment to an offsite disposal facility.

c. R/O membranes are dewatered and packaged in a 200 L (55 gal) drum.

d. R/O concentrates are dried to reduce the volume and converted to a granular or bead form. The dried concentrates are packaged in a 200 L (55 gal) drum or HIC.

The generation of mixed waste is minimized by prohibiting the use of hazardous materials. However, if mixed waste cannot be avoided, the mixed wastes are collected separately in a 200 L (55 gal) drums and stored prior to shipment to an appropriately licensed processor.

The SWMS is designed to provide 30-day storage of packaged wastes in accordance with the guidance of ANSI/ANS 55.1 (Reference 14). The storage facility is designed with adequate shielding to minimize the radiation dose to the operators, as described in Sections 12.3 and 12.4.

11.4.1.5 Radioactive Source Terms in SWMS

Source terms for solid radwaste are calculated for high-activity spent resin, low-activity spent resin, spent filter, and R/O concentrate. Tables 11.4-2 and 11.4-3 present the expected and design basis (1 percent fuel defect) radionuclide quantities in various solid radioactive wastes.

The spent resin long-term storage tank (SRLST) in the SWMS is designed to accumulate and contain high-activity spent resins from the purification, deborating, pre-holdup, and boric acid condensate ion exchangers of the CVCS for 10 years. The source terms for the SRLST are calculated by summing the source terms for each CVCS ion exchanger resin bed considering decay of up to 10 years.

The low-activity spent resin tank (LASRT) contains spent resins from the LWMS, SFPCCS, and SGBDS. The source term in the LASRT is determined based on spent resin generation assuming a one-time replacement of each ion exchanger bed. Because the buildup activities of SGBD and SFPCC spent resins are low, it is conservatively assumed that the LASRT is filled only with spent resins generated from the LWMS.

The high-activity spent filters generated from the CVCS, SFPCCS, and SGBDS are removed by means of a shielded plug and a shielded cask and transferred to the filter capping area in the compound building via the filter transporter and capped in a 200 L (55 gal) drum for offsite storage. The activity level of spent filters is determined by summing the buildup of radionuclide activities on each filter for 1 year.

The R/O concentrate is collected, drummed, dried, and moved to the temporary waste storage area in the compound building. The activities in the concentrate are determined using the inflow activities in each liquid waste stream to an R/O package and the corresponding decontamination factor of the R/O. The expected radioactive concentrations of the inflows to the equipment waste tank, chemical waste tank, and floor drain tanks and their flow rates are used to calculate the annual buildup activities in the R/O concentrate.

The methodology that is used to determine SWMS component safety classification follows NRC RG 1.143 (Reference 15), and is described in Subsection 11.2.1.4. The results are provided in Table 11.4-4.

11.4.1.6 Site-Specific Cost-Benefit Analysis

The cost-benefit numerical analysis as required by 10 CFR 50, Appendix I (Reference 22), Section II, Paragraph D, demonstrates that the addition of items of reasonably demonstrated technology would not provide a favorable cost-benefit. The COL applicant is to perform a site-specific cost-benefit analysis, following the guidance in NRC RG 1.110 (Reference 23) (COL 11.4(2)).

11.4.1.7 Mobile Equipment

The spent resin dewatering system is designed as a modular and mobile system. The mobile design facilitates equipment replacement when advanced treatment technologies are developed or when the equipment is broken, or both. The mobile system includes an exhaust fan and HEPA filter to control airborne dust, in accordance with BTP 11-3 (Reference 12). The exhaust air is discharged to the compound building HVAC system. The mobile system is designed to meet the provisions and conformance requirements of ANSI/ANS-40.37-2009, “Mobile Radioactive Waste Processing Systems” (Reference 25). The COL applicant is to provide reasonable assurance that the provisions and requirements of ANSI/ANS-40.37-2009 (Reference 25) are met (COL 11.4(3)).

The COL applicant is to provide reasonable assurance that mobile and temporary solid radwaste processing equipment and its interconnections to plant systems conform with regulatory requirements and guidance such as 10 CFR 50.34a (Reference 21), 10 CFR 20.1406 (Reference 26), and NRC RG 1.143 (Reference 15) (COL 11.4(3)). The COL applicant is to prepare a plan to develop and use operating procedures so the guidance and information in IE Bulletin 80-10 (Reference 1) are followed (COL 11.4(3)).

11.4.2 System Description

The primary functions of the SWMS are to process, package, and store the dry and wet solid wastes generated from the plant in accordance with regulatory guidelines, to handle and store dry and low-activity wastes prior to shipment to the offsite disposal facility, and to provide reasonable assurance that plant personnel and public radiation exposure is ALARA.

The SWMS handles wet and dry solid wastes, prepares for the waste transportation and offsite disposal as described in Subsection 11.4.1.4, and is divided into the following subsystems:

a. Spent resin transfer and packaging subsystem

b. Spent filter handling subsystem

c. Dry active waste subsystem

d. R/O concentrate treatment subsystem

e. Temporary waste storage subsystem

The expected and maximum annual waste volumes are shown in Table 11.4-1. The expected and design basis (1 percent fuel defect) activity levels of solid waste in the SWMS are presented in Tables 11.4-2 and 11.4-3.

The SWMS boundary starts at the receipt of waste from various waste generation components and ends where the packaged and dewatered waste is loaded onto a truck for shipment to the onsite interim storage facility or offsite disposal facility. The boundaries and descriptions for SWMS subsystems are presented as follows:

Spent Resin Transfer and Packaging Subsystem

The boundary of the spent resin transfer and packaging subsystem starts at the spent resin discharge isolation valve from each of the demineralizers and terminates at the temporary waste storage area for packaged spent resins.

The spent resin transfer and packaging subsystem is designed to transfer spent radioactive resins from the demineralizer vessels to the spent resin tanks where the resin is held up prior to being processed. The major components of this subsystem are the low-activity spent resin tank and the spent resin long-term storage tank. The process flow diagram for the spent resin transfer and storage subsystem is shown in Figure 11.4-1. The COL applicant is to provide piping and instrumentation diagrams (P&IDs) (COL 11.4(4)).

The spent resin tanks provide a settling capacity for radioactive resins transferred from various demineralizers. The spent resin long-term storage tank and low-activity spent resin tank are provided to hold up the spent resin for decay prior to processing. The spent resins in the CVCS demineralizers are transferred to the spent resin long-term storage tank hydraulically using demineralized water for sluicing. The sluice water in the SRST is then removed and routed to the LWMS for processing and release. The relatively low-activity spent resins from the LWMS, SFPCCS, and SGBDS are transferred to the low-activity spent resin tank via a similar method. Each spent resin tank has a connection for the transfer of spent resin to the mobile dewatering system.

Spent Filter Handling Subsystem

The boundary of the spent filter handling subsystem starts at the point of removal of the filter media from the filter housing and terminates at the temporary waste storage area.

The filter handling subsystem provides the capability to replace normally radioactive filters with a minimum of personnel radiation exposure. Following the detection of a pressure drop across the filter at a predetermined level, spent filters are removed from the filter vessel through a shield plug and a shielded transfer cask. Spent filters are transferred to the compound building via the filter transporter and then placed and capped in a 200 L (55 gal) drum. R/O membranes are dewatered and packaged in a 200 L (55 gal) drum. Low-activity filters, such as drain filters of the detergent waste subsystem and the HEPA filters of the GRS, are removed manually and disposed of in drums. Absorbent materials may be put into the filter disposal drums to minimize any standing water to meet transportation and disposal requirements.

Dry Active Waste Subsystem

The dry active waste subsystem boundary starts at the collection point of dry active wastes from the various generation areas in the plant and terminates at the temporary waste storage area.

The dry active wastes are collected and sorted at the generation area by plant personnel. Non-contaminated wastes are not processed in the SWMS and are handled separately. A space is also provided to sort miscellaneous contaminated dry solids for appropriate and cost-effective packaging and temporary storage. Miscellaneous solid waste consisting of contaminated rags, paper, clothing, glass, and other small items is collected at the DAW sorting area located in the compound building. Wastes are compacted and/or packaged into approved disposal containers and transferred to and stored in a temporary waste storage area in the compound building prior to shipment to the offsite disposal facility.

Charcoal used in the GRS is not expected to be replaced. Therefore, spent charcoal waste is not generated routinely. If spent charcoal waste is generated from the GRS, it is processed in accordance with the process control program provided by the COL applicant (COL 11.4(5)).

R/O Concentrate Treatment Subsystem

The R/O concentrate treatment subsystem boundary starts at the R/O concentrate discharge isolation valve from the R/O concentrate feed module of the LWMS and terminates at the temporary waste storage area.

Concentrate waste generated from the R/O membrane separation process is dried by the concentrate treatment system, and then dried concentrate waste is packaged in a 200 L (55 gal) drum or HIC.

Temporary Waste Storage Subsystem

The temporary waste storage subsystem boundary starts at the receipt point of packaged waste from the SWMS subsystems and ends at the truck bay for shipment of waste to the onsite interim storage facility or offsite disposal facility.

A shielded temporary waste storage area in the compound building is provided to facilitate the interim storage of higher-activity packaged wastes. The temporary waste storage area is sized to accommodate the number of drums and HICs generated in a 6-month period of normal operation. The expected and maximum generation volumes, and their shipped volumes, are summarized and presented in Table 11.4-1. This satisfies the 30-day criteria of ANSI/ANS-55.1 (Reference 14). The processing and temporary waste storage areas include a dedicated overhead crane with direct access to adjacent truck bays with sufficient overhead clearance to facilitate direct trailer loading of waste packages. Crane operation may be performed remotely with the aid of crane-mounted video cameras or locally to provide additional flexibility.

11.4.2.1 Dry Solid Waste

11.4.2.1.1 Dry Active Waste

Dry active waste (DAW) is classified as contaminated and non-contaminated waste at the point of generation by plant personnel. Contaminated waste is sent to a sorting and staging space provided in the SWMS area of the compound building to further sort DAW material for efficient packaging. Filtered hoods are provided to remove airborne contamination during the sorting of DAW.

DAW items such as rags, contaminated clothing, sweepings, and other items are compressed into a waste container by a solid waste compactor. During compactor operation, a fan is used to pull gases through the HEPA filter and exhaust them to the compound building HVAC system. When the container is full, it is manually sealed and moved to the low-level temporary waste storage area for shipment. The overhead crane with a capacity of 15 tons is provided to move the containers to and from the temporary waste storage area. The containers are surveyed prior to shipment. Table 11.4-1 provides an estimate of annual dry solid wastes generated. The dry active waste handling and storage operation is outlined in Figure 11.4-1.

11.4.2.1.2 HVAC Filters Handling

The HVAC filters are placed directly into the drums without disassembly to reduce personnel exposure. Filter hoods are also provided for handling filters where airborne contamination may occur.

11.4.2.2 Wet Solid Waste

11.4.2.2.1 Spent Resin Storage and Handling

The various spent resins generated from demineralizers or ion exchangers are sluiced to spent resin tanks in the compound building where they are allowed to settle prior to processing. Spent resins are segregated based on level of activity. High-activity spent resins from demineralizers used to process the reactor coolant, such as the purification and pre-holdup ion exchangers in the CVCS, are sluiced to the spent resin long-term storage tank located in the compound building. Low-activity spent resins from the demineralizers in the LWMS and the SFPCCS are sluiced to the low-activity spent resin tank. Figure 11.4-1 is a process flow diagram of the spent resin handling subsystem.

The spent resin of demineralizers in the SGBDS is sluiced to the low-activity spent resin tank only if the resin is radioactively contaminated. Otherwise, the resin is directly transferred to HICs for transport to an offsite facility for treatment and disposal.

Service air or water injected through the resin outlet line at the bottom of each demineralizer vessel is used to agitate the resins prior to transfer to the spent resin tank. The spent resins are transferred to the spent resin tank by opening the resin discharge valve on the resin outlet line after pressurization by demineralized water or the reactor makeup pump.

Low-activity resin is sluiced to the low-activity spent resin tank to allow for settling and holdup prior to processing. Spent resin is then transferred to a disposal container and dewatered by the dewatering system. The dewatering of the spent resins results in less than one percent of the waste volume as free-standing water in the HIC in order to meet the requirements of 10 CFR 61.56 (Reference 24). Packaged spent resin containers are transferred to the temporary waste storage area. The spent resins stored in the long-term storage tanks are processed in the same manner as the low-activity spent resins after sufficient radioactive decay.

Water in the spent resin is removed by the dewatering subsystem. The water is routed to the LWMS for processing prior to release to the environment. The dewatered spent resin is in compliance with the storage or disposal requirements of BTP 11-3 (Reference 12). Non-clogging screens are provided to prevent the carryover of spent resin beads or fines to the LWMS during the dewatering process.

11.4.2.2.2 Spent Filter Storage and Handling

An area is provided in the compound building for the storage of process filters used throughout the plant. The services are summarized as follows:

a. High-activity cartridge filters

   When a cartridge filter is to be replaced, it is first taken out of service. The filter is then vented and allowed to drain the liquid to the equipment drain tank (EDT). An overhead hoist is used to remove the shield hatch above the filter. The spent filter handling cask is lowered to rest on top of the plug opening. The filter head is then removed with extended tools and the spent filter is removed from the filter housing and placed into the spent filter handling cask. The spent filter-handling cask is then transferred to the filter handling area by the use of a hoist and a cask transporter. The cartridge is then lowered into a 200 L (55 gal) drum and capped. The drum is then placed in the shielded temporary waste storage area prior to final processing and shipment to an offsite disposal facility.

b. Low-activity LWMS cartridge filters

Cartridge filters are replaced based on the measured differential pressure or radiation level. During replacement, water is drained from the filter housing and the spent filter media is removed following the same process used for the high-activity spent filters. The cartridge is then lifted from the housing and placed in a drum. The drum is then moved to the shielded temporary waste storage area for shipment and disposal.

11.4.2.2.3 R/O Concentrate Processing

The concentrate generated from the R/O system is dried by the concentrate treatment subsystem. The dried waste is packaged in drums and stored in a temporary waste storage area in the compound building prior to shipment to the offsite disposal facility.

11.4.2.2.4 Mixed Waste

Mixed waste, which contains radioactive and hazardous contaminants, may be generated during normal operation. The COL applicant is responsible for the collection, temporary storage, and shipment of mixed waste for offsite treatment and disposal (COL 11.4(6)).

11.4.2.2.5 Interim Radwaste Storage Facility (IRSF)

The provision of an IRSF is site-specific and may be common to other nuclear power generating units at the same site. The COL applicant is responsible for the provision of a site-wide IRSF (COL 11.4(7)).

11.4.2.3 Packaging, Storage, and Shipping

The SWMS is designed to package wastes in a 200 L (55 gal) drum, a HIC, or other DOT-approved container, for temporary storage and offsite shipment. The HIC is typically used to package spent resin. Radioactive wastes such as concentrates, cartridge filters, and miscellaneous solid wastes are typically packaged in a 200 L (55 gal) drum or boxes. Large components that have been contaminated and cannot be compacted are prepared appropriately for packaging in shipping casks.

Waste is classified as Class A, B, C, or greater than Class C in accordance with 10 CFR 61.55 (Reference 27). Estimates of the expected and maximum volume of solid radwaste and its classification to be shipped offsite are provided in Table 11.4-1.

Packaged solid radioactive waste is transported to a shielded temporary waste storage area in the compound building via the bridge crane. The bridge crane is also used for waste onsite movement and offsite shipment.

In the event of the malfunctioning of the bridge crane, the COL applicant is to provide a mobile crane to retrieve a waste package that becomes stuck in the lifted condition or is dropped (COL 11.4(8)). The COL applicant is also to provide operational procedures to properly ship low-level wastes to external sites in accordance with U.S. NRC and Department of Transportation (DOT) regulations (COL 11.4(9)).

The storage facilities are designed in accordance with NRC Standard Review Plan, Section 11.4 (Reference 28), Electric Power Research Institute (EPRI) Utility Requirements Document Chapter 12.5 (Reference 29), and NRC RG 1.143 (Reference 15) as follows:

a. Potential release pathways are controlled and monitored in accordance with 10 CFR 50, Appendix A, General Design Criteria 60 and 64 (Reference 3 and 6), by the following:

    1) Furnishing curb-stones or dikes to retain spills of waste, such as dewatered resin or sludge

    2) Furnishing floor drains to collect and route spills and leaks to the LWMS for processing

    3) Furnishing area, airborne, and process radiation monitors

b. Sufficient shielding is provided to maintain radiation exposure ALARA and to limit the radiation streaming to adjacent areas in keeping with the designated radiation zones.

c. High-activity drums are retrievable on a row-by-row basis.

d. Video monitoring is used for remote viewing of high-radiation areas.

e. Automatic fire detection and suppression are provided.

A truck bay in the compound building is provided with a 15-ton crane used for loading packaged waste onto a transport vehicle for offsite transportation. The crane is manually operated and controlled remotely.

11.4.2.4 Operation and Personnel Doses

In order to reduce occupational radiation exposure, operations to process and transfer low- and high-activity radioactive wastes are conducted remotely. Operator access is provided to the low-level radioactive waste area only. The SRSTs are located in individually shielded cubicles in the compound building. The cubicles are coated with an epoxy liner, up to the cubicle wall height equivalent to the full tank volume, to facilitate decontamination of the facility in the event of tank failure. The SRSTs are remotely operated with no contact during handling.

Dewatering of spent resin is performed in a separately shielded cubicle and is controlled from the radwaste control room. This approach minimizes personnel radiation exposures and conforms with the ALARA principle implementation guidelines of NRC RG 8.8 (Reference 16) and NRC RG 8.10 (Reference 30). Water from the dewatering operation is collected and routed to the LWMS for processing and release, in accordance with BTP 11-6 (Reference 31), 10 CFR 20.1302 (Reference 32), and 10 CFR 20.1406 (Reference 26).

11.4.2.5 Design Features for Minimization of Contamination

The APR1400 is designed with features to meet the requirements of 10 CFR 20.1406 (Reference 26) and NRC RG 4.21 (Reference 2). The basic principles of NRC RG 4.21 (Reference 2), and the methods of control suggested in the regulations are delineated into four design objectives and two operational objectives, which are defined in Subsection 12.4.2.

The primary features that address the design and operational objectives for the SWMS are summarized below.

The SWMS SSCs, including the facility that houses the components, are designed to limit leakage and/or control the spread of contamination. In accordance with NRC RG 4.21 (Reference 2), the SWMS has been evaluated for leakage identification from the SSCs that contain radioactive or potentially radioactive materials, the areas and pathways where probable leakage may occur, and the methods of leakage control incorporated into the design of the system. The leak identification evaluation indicated that the SWMS is designed to facilitate early leak detection and provide the capability to assess collected fluid and respond to manage the collected fluids quickly. Thus, unintended contamination of the facility and the environment is minimized or prevented by the SSC design, and by operational procedures and programs for inspection and maintenance activities.

Prevention/Minimization of Unintended Contamination

a. The system components, including the low-activity spent resin tank and the spent resin long-term storage tank, are designed with stainless steel material and welded construction for life-cycle planning. The tanks are designed to have sufficient capacity to contain the spent resins for a decay period to reduce radioactivity.

b. The concentrate treatment subsystem and the spent resin dewatering subsystem are designed as skid packages with self-containing drip pans to contain leakage. The drains connected to the drip pan are routed to a floor drain sump for collection and then pumped to the LWMS for treatment and release.

c. The temporary waste storage area is designed with epoxy-coated floors and a drainage system to direct drainage to a floor drain sump for collection and subsequent pumping to the LWMS for treatment and release.

d. Cubicles in which contaminated materials are handled and stored have floors that are sloped and epoxy-coated to facilitate cleaning and facilitate drainage to early leak detection piping. Sump tanks are equipped with level switches to detect liquid accumulation, and pumps are provided to transfer the fluid for proper treatment.

e. The SWMS is designed with above-ground piping to the extent practicable. Buried and embedded piping is minimized. In lieu of embedded piping, a drainage pipe is provided in a covered concrete trench to collect any component overflow or leakage and route it to the building sump. In the event that buried or embedded piping cannot be avoided, double-walled piping and leak detection instruments are considered and evaluated based on the risks and the radiological consequences associated with the contamination of the facility and the environment.

Adequate and Early Leak Detection

a. The low-activity spent resin tank and the spent resin long-term storage tank are designed with level instrumentation to provide reasonable assurance of the integrity of the SSCs including the associated piping and to provide alarms to warn the operators of leakages.

b. The temporary waste storage area is designed with remote cameras for waste handling operation as well as for periodic surveillance.

c. Other components are designed for batch operation and are provided with adequate space to enable prompt assessment and response when required.

Reduction of Cross-Contamination, Decontamination, and Waste Generation

a. The SSCs are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radiological environment, thus minimizing cross-contamination and waste generation.

b. The process piping containing contaminated slurry is sized to facilitate flow and provide velocities that are sufficient to prevent solids settling in accordance with ANSI/ANS-55.6 (Reference 33). The piping is designed to reduce fluid traps, thus reducing decontamination and waste generation per 10 CFR 20.1406 (Reference 26). Pipe flushing fluid is collected and routed to the LWMS for processing and release, per ANSI/ANS-55.1 (Reference 14).

c. Utility connections are designed with a minimum of two barriers to prevent contamination of nonradioactive systems from potentially radioactive systems.

Decommissioning Planning

a. The SSCs are designed for the full service life and are fabricated, to the maximum extent practicable, as individual assemblies for easy removal.

b. The SSCs are designed with decontamination capabilities. Design features such as welding techniques and surface finishes are included to minimize the need for decontamination and minimize waste generation.

Operations and Documentation

a. Spent resin removal and packaging are designed for remote and automated operations. The system is equipped with instruments to provide alarms for operator actions in the event leakage is detected.

b. Operational procedures and maintenance programs related to leak detection and contamination control are to be prepared by the COL applicant (COL 11.4(10)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning.

c. A RG 4.21 program, Life Cycle Planning of Minimization of Contamination, is to be developed by the COL applicant to identify plant-wide components that contain radioactive materials, buried piping, embedded piping, leak detection methods and capabilities, and the methods utilized for the prevention of unnecessary contamination of clean components, facility areas, and the environment following the guidance of NEI 08-08A (Reference 34) (COL 11.4(11)). The leak identification program for the SWMS is to be integrated into this plant-wide program to facilitate the timely identification of leaks, prompt assessment, and appropriate responses to isolate and mitigate leakage.

d. Complete documentation of the system design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL information item (COL 11.4(12)).

Site Radiological Environmental Monitoring

The SWMS is part of the plant and is included in the site process control program and the site radiological environmental monitoring program for monitoring of facility and environmental contamination. The site radiological environmental monitoring program includes sampling and analysis of effluent to be released, meteorological conditions, hydrogeological parameters, and potential migration pathways of radioactive contaminants. Both programs are included as COL information items (COL 11.4(5)).

11.4.3 Radioactive Effluent Releases

Liquid radioactive wastes that leak from components in the SWMS are collected and transferred to the LWMS through the radioactive drain system. This liquid waste is collected and processed in the LWMS prior to discharge to the environment. Radioactive liquid effluents released from the LWMS are evaluated in Subsection 11.2.3.

To handle the airborne radioactive materials generated during SWMS operation, a fan is used to pull potentially contaminated air through a HEPA filter and exhaust it to the compound building HVAC system to monitor the releases to the environment. Potentially contaminated air in the compound building where the SWMS components are installed is exhausted though the air cleaning unit and radiation monitors of the compound building HVAC system. Radioactive ventilation exhaust from the compound building is evaluated in Subsection 9.4.

Liquid and gaseous radioactive waste generated during the SWMS operation is processed by the LWMS and the HVAC system. There is no uncontrolled and/or unmonitored release from the SWMS.

11.4.4 Process Control Program

Solid waste is processed in accordance with the process control program, which contains site-specific requirements. The COL applicant is to provide the process control program (COL 11.4(5)). The process control program provides reasonable assurance of the production of a solid waste matrix in accordance with 10 CFR 71 (Reference 10) and the guidance of NRC Branch Technical Position (BTP) 11-3 (Reference 12).

The process control program contains the planned effluent discharge flow rates and addresses the numerical guidelines in 10 CFR 50 Appendix I (Reference 22), as decribed in above Subsection 11.4.3. The program is prepared in accordance with the requirements of 10 CFR 71 (Reference 10) and the guidance of NUREG-1301 (Reference 35), NUREG-0133 (Reference 36), NRC RG 1.109 (Reference 37), NRC RG 1.111 (Reference 38), or NRC RG 1.113 (Reference 39). The program includes a description of how the NUREGs, NRC RGs, and alternative methods are implemented.

11.4.5 Component Descriptions

A summary of the SWMS components, including the design of the tanks and pumps, is shown in Table 11.4-4. The capacities, materials of construction, and applicable codes are included.

The SWMS components are determined for the radioactive safety classification in accordance with the guidance provided in RG 1.143 (Reference 15). The component safety classification is summarized in Table 11.4-4. Accordingly, the SWMS is classified as RW-IIa, based on the highest safety classification for the components within the system boundary. The SWMS components are housed within the compound building, which has been determined to be RW-IIa.

The SWMS safety classification applies to the components, up to and including the nearest valves, fittings, and/or welded/flanged nozzle connections.

11.4.5.1 Tanks

11.4.5.1.1 Spent Resin Storage Tank

Each of the two SRSTs is a cylindrical vertical tank with a hemispherical head and bottom and has a connection for transfer of spent resins to the dewatering system.

The SRST for low-activity spent resin is sized to hold at least two batches of spent resin from the source of the greatest input. The SRST for high-activity spent resin is sized to collect a volume of spent resin for 10 years of generation, which satisfies the 30-day criterion of ANSI/ANS-55.1 (Reference 14).

The SRSTs have a thermal dispersion-type level instrument to monitor the water and spent resin levels. A high-water-level alarm is provided for the SRSTs to prevent water overflow to the drain sump, as presented in Table 11.4-6.

The SRST vent nozzles are equipped with screens to prevent solids from being discharged to the compound building HVAC system, thus minimizing cross-contamination.

11.4.5.2 Piping and Valves

Piping used for the hydraulic transport of ion-exchange resins is designed with long-radius elbows to minimize the obstruction of spent resin slurry flow. Pipe-flow velocities are maintained in a turbulent flow regime to prevent settling when the slurry is being transported. Appropriate valves and pipe fittings are used to maintain unhindered flow. An adequate water/solids ratio is maintained throughout the transfer. Slurry piping is provided with a washing and flushing capability with sufficient water to flush the pipe after each use (e.g., at least two pipe volumes).

Ball or plug valves are selected for use within the SWMS to minimize flow resistance and pockets for trapping solids. A demineralized water connection is provided to flush the piping after each transfer.

11.4.5.3 Solid Waste Compactor

The solid waste compactor is used to reduce the volume of compactable DAW. The wastes are packed into 200 L (55 gal) drums, which are then further compacted by the solid waste compactor equipped with a hydraulic ram, hooded exhaust fan, and HEPA filter to control airborne dust. The compacted drums are then put into oversized drums for temporary storage and final disposal. The exhaust air from compaction is discharged to the compound building HVAC system. Sorting and staging space is available in the low-level waste handling and packing area.

11.4.5.4 Sorting Facility

During normal operation and maintenance, clean solid wastes are collected separately and are not brought into the SWMS sorting area. Contaminated DAW is collected at the point of generation, and is brought into the sorting area for further segregation for compactibility. DAW is sorted and segregated to appropriately and cost-effectively package, ship, and/or temporarily store the waste. The exhaust air is discharged to the compound building HVAC system.

11.4.5.5 Traveling Bridge Crane

The traveling bridge crane is remotely operated in the radwaste control room of the compound building or on local control panel for transporting drums or HICs. The crane moves waste drums and HICs from the processing area to the temporary waste storage area. It also moves waste drums and HICs from the temporary waste storage area to the shipping area. It is equipped with closed-circuit television cameras to facilitate remote handling.

11.4.5.6 R/O Concentrate Treatment System

The concentrate generated from the R/O system is dried by the concentrate treatment system. Packaged waste is stored in a temporary waste storage area in the compound building prior to shipment to the onsite interim storage facility or the offsite disposal facility. The subsystem is designed to include exhaust fan and HEPA filter to control airborne dust. The exhaust air is discharged to the compound building HVAC system.

11.4.6 Malfunction Analysis

There are no requirements to design the systems against a single failure criterion or multiple component train separations. The following internal hazards, however, are considered in the system design:

a. Safety interlock

   The spent resin handling subsystem is protected from component failure and operator error through a series of safety interlocks such as the level and temperature alarms.

b. Drum holding

In case of loss of electric power during handling a waste drum with the bridge crane, the bridge crane has a function to hold a waste drum during the operation in spite of loss of electric power.

11.4.7 Testing and Inspection Requirements

The SWMS is operated intermittently during normal plant operation. Therefore, periodic visual inspection and preventive maintenance are performed for the SWMS in accordance with industrial standards.

Epoxy coatings in cubicles that contain significant quantities of radioactive material, including the SRST cubicles, are Service Level II coatings as defined in NRC RG 1.54 (Reference 17), and are subject to the limited QA provisions, selection, qualification, application, testing, maintenance, and inspection provisions of NRC RG 1.54 (Reference 17) and standards referenced therein, as applicable to Service Level II coatings. Post-construction initial inspection is performed by personnel qualified using ASTM D 4537 (Reference 40) using the inspection plan guidance of ASTM D 5163 (Reference 41).

11.4.8 Instrumentation Requirements

The SWMS is operated and monitored from the radwaste control room in the compound building. Major system components of the SWMS instrumentation and indications are as follows:

a. Level indicators

   High-level alarms are provided to prevent the overflow of tanks during filling and resin transfer/sluice operations. These indicators are provided in the radwaste control room in the compound building.

b. Radiation monitoring

   Area radiation monitors are described in Subsection 12.3.4.

Instruments, including backflushing provisions, are located in low-radiation areas when possible for accessibility and fulfillment of the ALARA provisions. A list of alarm instruments and location of readouts is presented in Table 11.4-6.

11.4.9 Combined License Information

COL 11.4(1) The COL applicant can incorporate an onsite laundry facility for processing of contaminated clothing.

COL 11.4(2) The COL applicant is to perform a site-specific cost-benefit analysis following the guidance in NRC RG 1.110.

COL 11.4(3) The COL applicant is to provide reasonable assurance that the provisions and requirements of ANSI/ANS-40.37-2009 are met. The COL applicant is to provide reasonable assurance that mobile and temporary solid radwaste processing and its interconnection to plant systems conform with regulatory requirements and guidance such as 10 CFR 50.34a, 10 CFR 20.1406, and NRC RG 1.143. The COL applicant is to prepare a plan to develop and use operating procedures so the guidance and information in IE Bulletin 80-10 are followed.

COL 11.4(4) The COL applicant is to provide P&IDs.

COL 11.4(5) The COL applicant is to prepare the site process control program and the site radiological environmental monitoring program.

COL 11.4(6) The COL applicant is responsible for the collection, temporary storage, and shipment of mixed waste for offsite treatment and disposal.

COL 11.4(7) The COL applicant is responsible for the provision of a site-wide IRSF for interim storage of radioactive wastes.

COL 11.4(8) The COL applicant is to provide a mobile crane to retrieve a waste package that becomes stuck in the lifted condition or that is dropped.

COL 11.4(9) The COL applicant is also to provide operational procedures to properly ship low-level wastes to external sites in accordance with US NRC and US Department of Transportation (DOT) regulations.

COL 11.4(10) The COL applicant is to prepare the operational procedures and maintenance programs for the SWMS as related to leak detection and contamination control.

COL 11.4(11) The COL applicant is to develop plant-wide RG 4.21 life-cycle planning for minimization of contamination program following the guidance in NEI 08-08A, in which the SWMS procedures and programs are to be integrated.

COL 11.4(12) The COL applicant is to maintain the complete documentation of system design, construction, design modifications, field changes, and operations.

11.4.10 References

1. IE Bulletin 80-10, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment," U.S. Nuclear Regulatory Commission, May 1980.

2. Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," U.S. Nuclear Regulatory Commission, June 2008.

3. 10 CFR Part 50, Appendix A, General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," U.S. Nuclear Regulatory Commission.

4. 10 CFR Part 50, Appendix A, General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," U.S. Nuclear Regulatory Commission.

5. 10 CFR Part 50, Appendix A, General Design Criterion 63, "Monitoring Fuel and Waste Storage," U.S. Nuclear Regulatory Commission.

6 10 CFR Part 50, Appendix A, General Design Criterion 64, "Monitoring Radioactivity Releases," U.S. Nuclear Regulatory Commission.

7. 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage," U.S. Nuclear Regulatory Commission.

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

9. 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," U.S. Nuclear Regulatory Commission.

10. 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," U.S. Nuclear Regulatory Commission.

11. 49 CFR Part 173, "Shippers – General Requirements for Shipments and Packagings," U.S. Department of Transportation.

12. NUREG-0800, Standard Review Plan, BTP 11-3, "Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants," Rev. 3, U.S. Nuclear Regulatory Commission, March 2007.

13. 49 CFR Part 171, "General Information, Regulations, and Definitions," U.S. Department of Transportation.

14. ANSI/ANS-55.1, "Solid Radioactive Waste Processing System for Light-Water-Cooled Reactor Plants," American Nuclear Society, 1992 (Reaffirmed 2009).

15. Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Rev. 2, U.S. Nuclear Regulatory Commission, November 2001.

16. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," Rev. 3, U.S. Nuclear Regulatory Commission, June 1978.

17. Regulatory Guide 1.54, "Service Level I, II, and III Protective Coating Applied to Nuclear Power Plants," Rev. 2, U.S. Nuclear Regulatory Commission, October 2000.

18. ASME Section II, "Material Specification," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

19. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," Rev. 3, U.S. Nuclear Regulatory Commission, June 2013.

20. ASME B31.3, "Process Piping," The American Society of Mechanical Engineers, the 2007 Edition with the 2008 Addenda.

21. 10 CFR 50.34a, "Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents-Nuclear Power Reactors," U.S. Nuclear Regulatory Commission.

22. 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation To Meet the Criterion 'As Low as Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," U.S. Nuclear Regulatory Commission.

23. Regulatory Guide 1.110, "Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors," Rev. 1, U.S. Nuclear Regulatory Commission, October 2013.

24. 10 CFR 61.56, "Waste Characteristics" U.S. Nuclear Regulatory Commission.

25. ANSI/ANS-40.37-2009, "Mobile Low-level Radioactive Waste Processing System," American Nuclear Society, 2009.

26. 10 CFR 20.1406, "Minimization of Contamination," U.S. Nuclear Regulatory Commission.

27. 10 CFR 61.55, "Waste Classification," U.S. Nuclear Regulatory Commission.

28. NUREG-0800, Standard Review Plan, Section 11.4, "Solid Waste Management System," Rev.3, U.S. Nuclear Regulatory Commission, March 2007.

29. EPRI URD Section 12.5, "Solid Radioactive Waste Processing System," Electric Power Research Institute.

30. Regulatory Guide 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures as low as Is Reasonably Achievable,” Rev. 1, U.S. Nuclear Regulatory Commission, May 1977.

31. NUREG-0800, Standard Review Plan, BTP 11-6, “Postulated Radioactive Releases Due to Liquid-Containing Tank Failures,” Rev. 3, U.S. Nuclear Regulatory Commission, March 2007.

32. 10 CFR 20.1302, “Compliance with Dose Limits for Individual Members of the Public,” U.S. Nuclear Regulatory Commission.

33. ANS/ANSI-55.6, “Liquid Radioactive Waste Processing System for Light Water Reactor Plants,” American Nuclear Society, July 1993 (Reaffirmed 2007).

34. NEI 08-08A, “Generic FSAR Template Guidance for Life Cycle Minimization of Contamination,” Rev 0, Nuclear Energy Institute, October 2009.

35. NUREG-1301, “Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors,” U.S. Nuclear Regulatory Commission, November 1990.

36. NUREG-0133, “Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, October 1987.

37. Regulatory Guide 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I,” Rev. 1, U.S. Nuclear Regulatory Commission, October 1977.

38. Regulatory Guide 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors,” Rev. 1, U.S. Nuclear Regulatory Commission, July 1977.

39. Regulatory Guide 1.113, “Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I,” Rev. 1, U.S. Nuclear Regulatory Commission, April 1977.

40. ASTM D 4537-04a, “Standard Guide for Establishing Procedures to Qualify and Certify Personnel Performing Coating Work Inspection in Nuclear Facilities,” American Society for Testing and Materials, 2004.

41. ASTM D 5163-08, "Standard Guide for Establishing a Program for Condition Assessment of Coating Service Level I Coating Systems in Nuclear Power Plants," American Society for Testing and Materials, 2008.

Table 11.4-1

Estimated Annual Solid Waste Generation

(Unit: m^3/yr-unit)

Waste Stream		Expected Generation	Expected Shipped Volume(1)	Maximum Generation	Maximum Shipped Volume(1)	Waste Classification(2)
Spent Filter	High Activity	0.19	0.21	0.38	0.42	B
	Low Activity	0.15	0.17	0.29	0.32	A
Spent Resin	High Activity	2.72	-	5.43	-	B
	Low Activity	8.64	8.64	17.28	17.28	A
R/O Membrane		3.24	3.6	3.24	3.6	A
R/O Concentrate		4.2	4.2	12	12	A
Dry Active Waste		-	50.19	-	141.68	A
Total		-	67.01	-	175.30	-

(1) Shipped volume is estimated based upon the following:
- Spent filters are packed in a 200 L (55 gal) drum or HIC. Packing efficiency of 90% is considered.
- Spent resin is packed in HIC.
- High-activity spent resins generated from CVCS are stored in the spent resin long-term storage tank for 10 years. The high-activity spent resin will be shipped after sufficient decay.
- R/O membranes are packed in a 200 L (55 gal) drum. Packing efficiency of 90% is considered.
- Volume of DAWs is estimated using the 1000 MWe plant's average and maximum packaged volume during 10 years. The factor for the increment of electric power generation (1400/1000) is reflected.

(2) Waste classification per 10 CFR 61.55

(3) Generation of mixed waste is prevented and minimized by prohibiting use of hazardous material.

(4) GRS delay bed charcoal is expected to be essentially permanent.

Table 11.4-2 (1 of 2)

Expected Radioactive Source Terms for SWMS (Bq)

Nuclide	Spent Resin Long-Term Storage Tank	Low-Activity Spent Resin Tank	High-Activity Spent Filter	R/O Concentrate
Na-24	7.00E+11	7.16E+10	2.93E+07	1.02E-02
Cr-51	2.12E+12	4.17E+10	1.82E+12	9.22E+10
Mn-54	1.45E+13	2.37E+10	2.67E+12	9.42E+10
Fe-55	3.40E+13	1.79E+10	2.17E+12	7.38E+10
Co-58	8.28E+12	6.60E+10	5.17E+12	2.15E+11
Fe-59	3.34E+11	4.20E+09	2.63E+11	1.19E+10
Co-60	2.38E+13	7.90E+09	1.01E+12	3.30E+10
Zn-65	3.57E+12	7.53E+09	1.36E+09	2.95E+10
Br-84	8.28E+09	1.05E+09	9.05E+06	0.00E+00
Rb-88	3.64E+10	6.99E+09	1.22E+08	0.00E+00
Sr-89	1.72E+11	1.98E+09	1.48E+05	5.83E+09
Y-89m	0.00E+00	1.97E+05	0.00E+00	0.00E+00
Sr-90	8.65E+11	1.79E+08	1.38E+04	7.53E+08
Y-90	0.00E+00	1.08E+08	0.00E+00	0.00E+00
Sr-91	9.20E+09	9.37E+08	5.92E+05	8.92E-13
Y-91m	5.50E+07	5.87E+08	3.50E+05	6.75E-252
Y-91	6.99E+05	1.62E+08	9.40E+03	2.29E+08
Y-93	5.20E+08	4.33E+09	3.72E+06	8.87E-11
Zr-93	0.00E+00	4.40E+01	0.00E+00	0.00E+00
Zr-95	6.33E+11	5.56E+09	4.85E+08	1.77E+10
Nb-95m	0.00E+00	5.63E+07	0.00E+00	0.00E+00
Nb-95	2.41E+11	4.30E+09	2.92E+05	9.74E+09
Mo-99	4.30E+11	3.90E+10	4.78E+06	2.08E+08
Tc-99m	2.80E+10	3.63E+10	2.81E+06	2.82E-25
Tc-99	0.00E+00	2.12E+03	0.00E+00	0.00E+00
Ru-103	7.33E+12	1.04E+11	7.86E+06	2.78E+11
Rh-103m	0.00E+00	1.03E+11	0.00E+00	0.00E+00

Table 11.4-2 (2 of 2)

Nuclide	Spent Resin Long-Term Storage Tank	Low-Activity Spent Resin Tank	High-Activity Spent Filter	R/O Concentrate
Ru-106	9.67E+14	1.34E+12	1.02E+08	5.36E+12
Rh-106	0.00E+00	1.34E+12	0.00E+00	0.00E+00
Ag-110m	9.29E+12	1.92E+10	1.45E+06	7.52E+10
Ag-110	0.00E+00	2.50E+08	0.00E+00	0.00E+00
Te-129m	1.61E+11	2.60E+09	1.91E+05	6.43E+09
Te-129	2.62E+10	4.53E+09	1.41E+07	2.03E-175
I-129	0.00E+00	3.87E+00	0.00E+00	0.00E+00
Te-131m	4.44E+10	4.57E+09	9.94E+05	5.53E+03
Te-131	3.13E+09	1.19E+09	4.32E+06	0.00E+00
I-131	4.14E+11	2.41E+10	1.79E+06	9.84E+09
Te-132	1.41E+11	1.15E+10	1.29E+06	1.79E+08
I-132	1.31E+11	2.60E+10	3.46E+07	2.10E-82
I-133	5.65E+11	5.74E+10	1.76E+07	6.27E+01
I-134	8.58E+10	9.91E+09	5.75E+07	3.47E-235
Cs-134	5.50E+11	5.94E+08	5.05E+04	2.46E+09
I-135	3.73E+11	3.82E+10	3.33E+07	5.49E-21
Cs-136	1.83E+11	1.11E+10	1.01E+06	1.21E+10
Cs-137	2.57E+12	8.53E+08	7.24E+04	3.60E+09
Ba-137m	2.57E+12	7.96E+08	7.24E+04	3.60E+09
Ba-140	4.11E+12	1.55E+11	1.22E+07	1.61E+11
La-140	1.00E+12	2.18E+11	1.77E+07	6.34E+06
Ce-141	1.20E+11	2.05E+09	1.54E+05	4.97E+09
Ce-143	9.31E+10	9.33E+09	1.96E+06	4.68E+04
Pr-143	0.00E+00	2.70E+09	0.00E+00	0.00E+00
Ce-144	3.23E+13	5.90E+10	4.50E+06	2.33E+11
Pr-144	0.00E+00	5.90E+10	0.00E+00	0.00E+00
W-187	6.00E+10	6.07E+09	1.67E+06	1.24E+02
Np-239	1.30E+11	1.18E+10	1.60E+06	2.00E+07

Table 11.4-3 (1 of 3)

Design Basis Radioactive Source Terms for SWMS (Bq) (1 % Fuel Defect)

Nuclide	Spent Resin Long-Term Storage Tank	Low-Activity Spent Resin Tank	High-Activity Spent Filter	R/O Concentrate
Na-24	7.01E+11	7.16E+10	4.03E+07	1.02E-02
Cr-51	9.49E+12	1.86E+11	8.73E+12	4.11E+11
Mn-54	1.45E+13	2.37E+10	7.49E+12	9.42E+10
Fe-55	3.40E+13	1.79E+10	7.69E+12	7.38E+10
Co-58	8.28E+12	6.60E+10	7.40E+12	2.15E+11
Fe-59	3.34E+11	4.20E+09	3.05E+11	1.19E+10
Co-60	2.38E+13	7.90E+09	3.64E+12	3.30E+10
Zn-65	3.57E+12	7.53E+09	7.63E+09	2.95E+10
Br-84	1.11E+10	1.37E+09	1.70E+07	0.00E+00
Rb-88	3.87E+11	7.33E+10	1.79E+09	0.00E+00
Sr-89	4.14E+12	4.63E+10	3.64E+06	1.37E+11
Y-89m	0.00E+00	4.63E+06	0.00E+00	0.00E+00
Sr-90	1.62E+13	3.37E+09	2.49E+05	1.41E+10
Y-90	0.00E+00	2.02E+09	0.00E+00	0.00E+00
Sr-91	4.81E+10	4.90E+09	4.23E+06	4.67E-12
Y-91m	3.60E+08	3.12E+09	3.16E+06	4.35E-251
Y-91	6.39E+07	7.23E+09	8.90E+05	2.10E+10
Y-93	1.47E+07	1.20E+08	1.31E+05	2.45E-12
Zr-93	0.00E+00	1.21E+00	0.00E+00	0.00E+00
Zr-95	9.91E+11	8.67E+09	4.97E+09	2.75E+10
Nb-95m	0.00E+00	8.77E+07	0.00E+00	0.00E+00
Nb-95	4.41E+11	7.77E+09	5.65E+05	1.79E+10
Mo-99	1.90E+13	1.72E+12	2.71E+08	9.19E+09
Tc-99m	1.00E+12	1.59E+12	1.46E+08	1.05E-23
Tc-99	0.00E+00	9.23E+04	0.00E+00	0.00E+00
Ru-103	1.71E+11	2.46E+09	1.98E+05	6.58E+09

Table 11.4-3 (2 of 3)

Nuclide	Spent Resin Long-Term Storage Tank	Low-Activity Spent Resin Tank	High-Activity Spent Filter	R/O Concentrate
Rh-103m	0.00E+00	2.45E+09	0.00E+00	0.00E+00
Ru-106	8.26E+11	1.12E+09	8.52E+04	4.50E+09
Rh-106	0.00E+00	1.12E+09	0.00E+00	0.00E+00
Ag-110m	9.29E+12	1.92E+10	1.47E+06	7.52E+10
Ag-110	0.00E+00	2.50E+08	0.00E+00	0.00E+00
Te-129m	5.03E+12	8.20E+10	6.53E+06	2.03E+11
Te-129	7.48E+09	5.30E+10	5.42E+06	5.70E-176
I-129	0.00E+00	4.10E+01	0.00E+00	0.00E+00
Te-131m	8.39E+11	8.67E+10	2.57E+07	1.05E+05
Te-131	4.85E+09	1.61E+10	9.63E+06	0.00E+00
I-131	5.05E+14	2.66E+13	2.61E+09	1.19E+13
Te-132	1.62E+13	1.34E+12	1.92E+08	2.09E+10
I-132	1.62E+12	1.50E+12	5.72E+08	2.46E-81
I-133	7.68E+13	7.81E+12	3.21E+09	8.51E+03
I-134	3.84E+11	4.43E+10	3.61E+08	1.55E-234
Cs-134	4.82E+15	5.27E+12	4.47E+08	2.18E+13
I-135	1.41E+13	1.39E+12	1.74E+09	2.00E-19
Cs-136	9.25E+12	5.67E+11	5.71E+07	6.18E+11
Cs-137	1.84E+16	6.14E+12	5.28E+08	2.58E+13
Ba-137m	1.84E+16	5.74E+12	5.28E+08	2.46E+13
Ba-140	1.30E+12	4.80E+10	4.23E+06	4.97E+10
La-140	5.81E+10	4.20E+10	1.30E+06	3.60E+05
Ce-141	1.20E+11	2.04E+09	1.63E+05	4.96E+09
Ce-143	1.40E+10	1.43E+09	3.83E+05	7.17E+03
Pr-143	0.00E+00	4.13E+08	0.00E+00	0.00E+00
Ce-144	3.57E+12	6.37E+09	4.92E+05	2.51E+10
Pr-144	0.00E+00	6.33E+09	0.00E+00	0.00E+00

Table 11.4-3 (3 of 3)

Nuclide	Spent Resin Long-Term Storage Tank	Low-Activity Spent Resin Tank	High-Activity Spent Filter	R/O Concentrate
W-187	6.01E+10	6.07E+09	2.26E+06	1.24E+02
Np-239	1.30E+11	1.18E+10	2.06E+06	2.00E+07
Sum of Fractions				
$\sum A_i/A_{1i}$	1.70E+04	4.40E+01	N/A	5.10E+01
$\sum A_i/A_{2i}$	3.90E+04	8.40E+01	N/A	9.50E+01
Radwaste Classification				
	RW-IIa	RW-IIa	N/A (1)	RW-IIa (2)

(1) Radwaste classification for High Activity Spent Filter is not applicable since this is considered as radioactive waste, not the SSC.

(2) The R/O Concentrate is transferred to the Concentrate Treatment System for processing. Therefore, the classification for the R/O Concentrate is applied to the Concentrate Treatment System.

Table 11.4-4 (1 of 2)

Equipment List for the SWMS

Tanks	
Equipment Quantity Design Capacity, L (ft^3) Material Radwaste Safety Class	Low-activity spent resin tank 1 22,654 (800) Stainless steel RW-IIa
Equipment Quantity Design Capacity, L (ft^3) Material Radwaste Safety Class	Spent resin long-term storage tank 1 90,189 (3,185) Stainless steel RW-IIa
Equipment Quantity Design Capacity, L (ft^3) Material Radwaste Safety Class	New resin tank 1 5,678 (200) Stainless steel N/A(1)
Miscellaneous	
Equipment Quantity Design Capacity, Ton Operation/Control Material Radwaste Safety Class	Traveling bridge crane 1 15 Remote, televideo Carbon steel N/A(1)
Equipment Quantity Radwaste Safety Class	Concentrate treatment system 1 RW-IIa
Equipment Quantity Radwaste Safety Class	Solid waste compactor 1 N/A(1)

Table 11.4-4 (2 of 2)

Miscellaneous (cont.)	
Equipment	Sorting table
Quantity	1
Material	Stainless steel
Radwaste Safety Class	N/A[1]

(1) The equipment classified as N/A is non-radwaste component.

Table 11.4-5

Codes and Standards for SWMS Equipment

Equipment	Design and Fabrication	Material	Welder Qualifications and Procedures	Inspection and Testing
Atmospheric Tank	API 650	ASME Sec. II	ASME Sec. IX	API 650 (atmospheric)
Pressure Vessels	ASME Sec. VIII, Div. 1 or Div. 2	ASME Sec. II	ASME Sec. IX	ASME Sec. VIII, Div. 1 or Div. 2
Piping and Valves	ASME B31.3	ASME Sec. II	ASME Sec. IX	ASME B31.3

Table 11.4-6

Instrument Indication and Alarm Information

Equipment	Parameter	Record	Indication	Alarm		Location
				High	Low	
Low-Activity Spent Resin Tank	Tank level		X[(1)]	X		Radwaste control room
	Tank pressure		X			Radwaste control room
	Demin. water inlet flow rate		X			Radwaste control room
Spent Resin Long-Term Storage Tank	Tank level		X	X		Radwaste control room

(1) X: provided

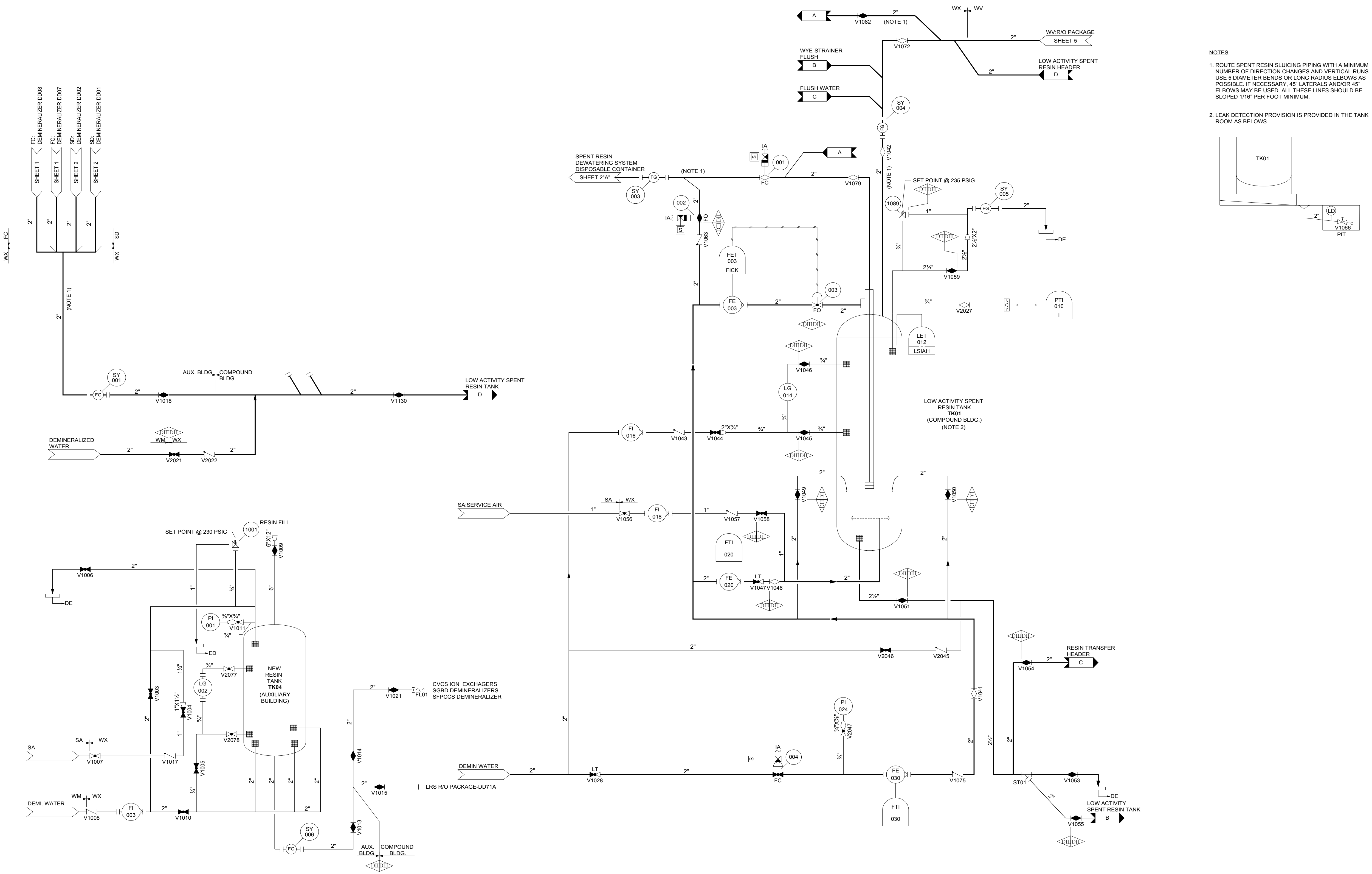


Figure 11.4-1 Solid Radwaste System Flow Diagram (1 of 2)

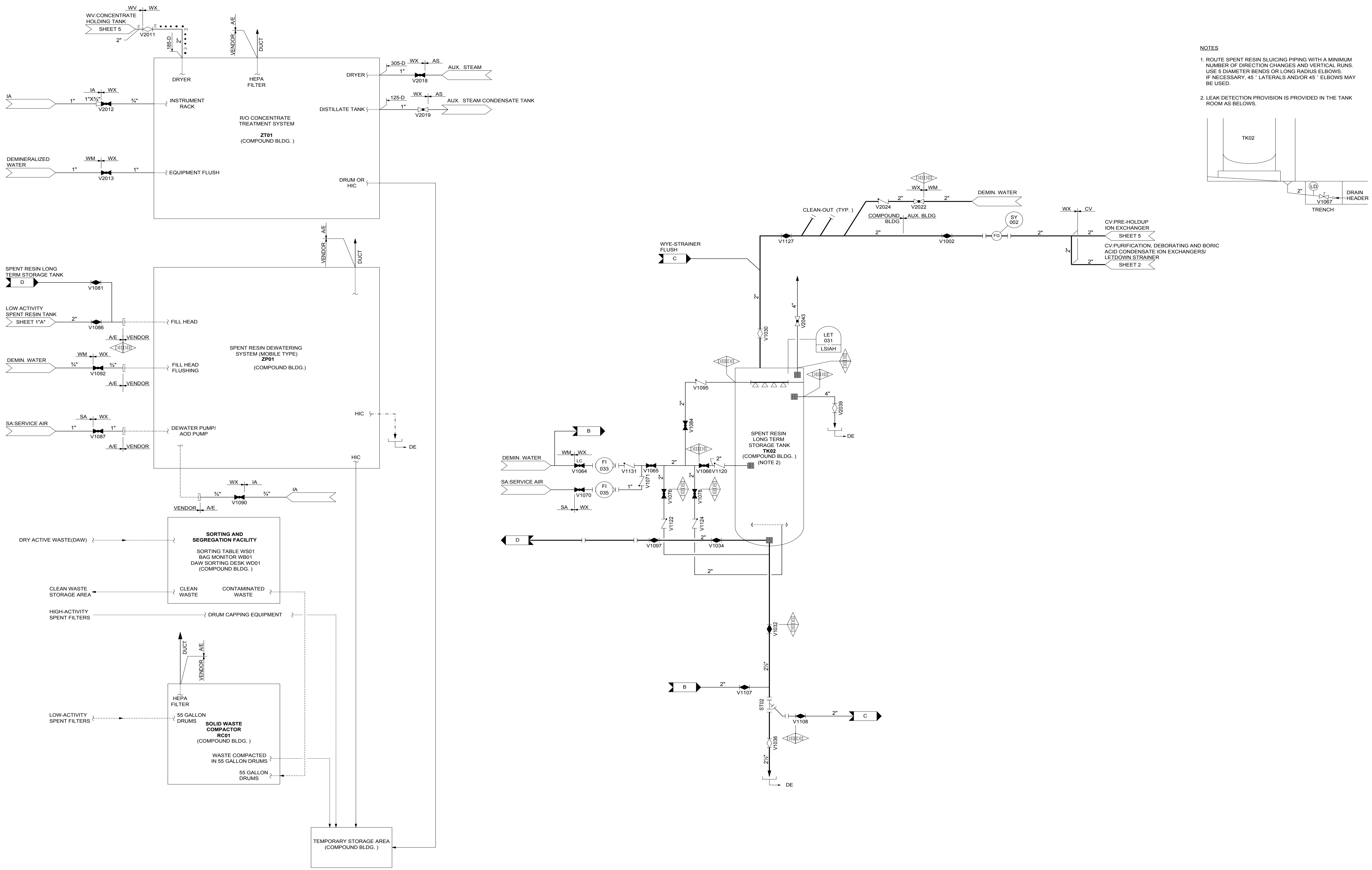


Figure 11.4-1   Solid Radwaste System Flow Diagram (2 of 2)

11.5 Process and Effluent Radiation Monitoring and Sampling Systems

Process and effluent radiation monitoring and sampling systems (PERMSS) are used to measure, record, and sample to control releases of radioactive materials of the liquid and gaseous process streams and effluents from the liquid waste management system (LWMS), gaseous waste management system (GWMS), and solid waste management system (SWMS) as a result of normal operations, including AOOs, and during postulated accidents.

Radiation monitoring and sampling systems consist of the PERMSS and an area radiation monitoring system (ARMS). The PERMSS is divided into a gaseous and liquid process and an effluent monitoring system, which are described in Subsections 11.5.2.2 and 11.5.2.3, respectively. The locations of the monitoring and sampling points, types of instruments, and locations of the alarms are provided in Table 11.5-1 and Figure 11.5-2. The monitoring and sampling systems for area radiation are described in Chapter 12.

The sampling system that collects representative samples of liquids and gases and delivers them to a facility for chemical and radiological analysis is described in Subsection 9.3.2. The subsection includes sampling locations, methodology, and frequency of continuous and grab sampling.

Flow diagrams of all radiation monitors for PERMSS and ARMS are presented in Figure 11.5-1.

11.5.1 Design Bases

11.5.1.1 Design Objective

The radiation monitoring system (RMS) is designed in accordance with the acceptance criteria of SRP Section 11.5 (Reference 1).

Continuous monitoring equipment is located in selected airborne, gaseous, and liquid process and effluent streams to detect activity generated during normal operations, including AOOs, and during postulated accidents.

The RMS assists plant operators in evaluating and controlling the radiological consequences of potential equipment failures, system malfunctions, or system misoperations. Tables 11.5-1 and 11.5-2 list the gaseous and liquid process and effluent radiation monitors. The tables also show the monitoring parameters.

11.5.1.2 Design Criteria

The RMS is designed to perform the following functions:

a. Provide early warning to station personnel of malfunction or misoperation of the systems or potential radiological hazards according to 10 CFR 20 (Reference 2) and 10 CFR 50, Appendix I (Reference 3).

b. Provide continuous monitoring of radioactive liquid and airborne releases according to 10 CFR 20; 10 CFR 50, GDC 13, 60, 61, 63, and 64 (Reference 4); and the guidelines of NRC RG 1.21 (Reference 5).

c. Provide monitoring of liquid and airborne activity in selected locations and effluent paths for postulated accidents in accordance with the requirements of 10 CFR 50, Appendix I (Reference 3), NUREG-0737 (Reference 6), and the guidance of NRC RGs 1.45 (Reference 7) and 1.97 (Reference 8).

d. The design of the process and effluent radiation monitoring and sampling systems provides instrumentation to measure, record, and indicate in the MCR as well as to control releases of radioactive materials in plant process systems and effluent streams.

e. These systems are designed to provide continuous sampling and monitoring of radioactive iodine, particulates, and gases as well as the capability to obtain grab samples in gaseous process or effluent streams in all potential accident release points.

f. The capability to take grab samples from HVAC system exhausts is provided at radiation monitor locations. Grab samples are used for analysis during normal operating and post-accident conditions.

g. Continuous sampling of all potential post-accident release points is provided by vent samplers. These samplers are designed to be used during normal operating and post-accident conditions to meet the sampling requirements in 10 CFR 50.34(f)(2)(xvii) and 50.34(f)(2)(xxvii) (Reference 9) and the guidance in NUREG-0718 (Reference 10), NUREG-0737, Appendix 2, Section II.F.1.

h. The process and effluent monitoring and sampling system is designed with a continuous MCR interface via the information processing system (IPS) and qualified indication and alarm system (QIAS) systems.

i. Guidance on the selection of instrumentation for use of continuous radiation monitoring equipment and performance requirements are in accordance with ANSI N42.18-2004 (Reference 11). The qualification is also in accordance with NRC RG 1.143 (Reference 12).

j. Instrument setpoints are determined in accordance with NUREG-1301 (Reference 13) and NUREG-0133 (Reference 14). The effluent concentration limits in Table 2 of Appendix B to 10 CFR 20 (Reference 15) are used so that the release to unrestricted areas is not exceeded. The COL applicant is to determine the WARN and ALARM setpoints of the PERMSS based on the site-specific conditions and operational requirements (COL 11.5(1)).

k. PERMSSs are designed to conform with 10 CFR 20.1301 (Reference 16); 10 CFR 20.1302 (Reference 17); 40 CFR 190 (Reference 18); 10 CFR 20, Appendix B; and the numerical guides in 10 CFR 50, Appendix I. An annual report that specifies the quantity of each principal radionuclide released to unrestricted areas in liquid and in gaseous effluents is provided to conform with 10 CFR 50.36a (Reference 19). The COL applicant is to develop an annual report that specifies the quantity of each principal radionuclide released to unrestricted areas in liquid and in gaseous effluents (COL 11.5 (2)) and site-specific procedures on equipment inspection, calibration, and maintenance and regulated record keeping. The COL applicant is to provide site-specific procedures that conform with the numerical guides of 10 CFR 50.34a (Reference 20) and 10 CFR 50, Appendix I (COL 11.5 (3)).

l. The COL applicant is to prepare an offsite dose calculation manual (ODCM) that contains a description of the methodology and parameters for calculation of the offsite doses for the gaseous and liquid effluents (COL 11.5 (5)). NEI 07-09A (Reference 21) is an alternative and provides a radiological and environmental monitoring program. The ODCM is to be in accordance with NRC RGs 1.109 (Reference 22), 1.111 (Reference 23), and 1.113 (Reference 24). The COL applicant is to provide analytical procedures and sensitivity for selected radio-analytical methods and type of sampling media for site-specific matter (COL 11.5 (5)).

m. The COL applicant is to develop a radiological and environmental monitoring program in accordance with NUREG-1301 and also NUREG-0133, which describes the scope of the program, taking into account local and land use census data in identifying all potential radiation exposure pathways, associated radioactive materials present in liquid and gaseous effluent, and direct external radiation from SSC. The COL applicant is also to develop calibration procedures in accordance with NRC RG 1.33 (Reference 25) and NRC RG 4.15 (Reference 26) (COL 11.5 (6)). The COL applicant is to develop detailed location and tubing installation and provide the sampling method including the sampling time to acquire representative sampling (COL 11.5 (7)).

The RMS monitors normal and potential paths for release of radioactive materials to provide continuous indication and recording of radioactivity levels of the gaseous and liquid waste leaving the plant.

Continuous representative sampling is provided for airborne particulate and iodine radioactivity in discharge paths. The gaseous PERMSS is designed in accordance with ANSI/HPS N13.1 (Reference 27). The RMS also initiates control actions as shown in Tables 11.5-1 and 11.5-2 to control or reduce continuous effluent releases or to terminate releases.

The RMS is installed in the HVAC systems to monitor the airborne radioactivity resulting from system malfunction or misoperation, or from maintenance activities that could cause radioactivity to reach unacceptable levels. Portable airborne radiation monitors are available for use in areas where work activities or surveillance poses an unacceptable risk to plant personnel of exposure to airborne radioactive material.

Radiation monitoring equipment is provided to detect primary-to-secondary system leakage and leakage from the primary system to the containment atmosphere. These monitoring systems are designed to be consistent with the recommendations of NRC RG 1.45 for detection of primary system leakage. Leakage detection methods, instrumentation, relevant systems, sensitivity, response time, testing, and calibration are described in Subsection 5.2.5.

Other systems that interface via heat exchangers with the primary system or other normally radioactive systems are also monitored to detect leakage between the systems so that appropriate actions can be taken to mitigate any potential consequences. The reactor coolant gross activity levels are also monitored or sampled during normal operation to maintain reactor coolant system (RCS) activity within acceptable levels.

This provides reasonable assurance that activity levels in other normally radioactive auxiliary systems are also maintained at acceptable levels. Under accident conditions, the RMS provides indication to plant operators if a breach of a fission product barrier has occurred and provides information to evaluate the magnitude of actual or potential releases of radioactive materials in order that appropriate emergency actions are taken to protect plant personnel and the health and safety of the public. The RMS monitors gross radioactivity to detect and evaluate a breach of the fuel cladding or potential core melt conditions for the RCS.

The containment atmosphere is monitored for particulate, iodine, and gaseous activity resulting from a breach of the reactor coolant pressure boundary (RCPB), a fuel handling accident, or other equipment failures that could release significant activity. Indication of high containment activity automatically initiates containment purge isolation.

In order to maintain MCR habitability, the outside air supply to the MCR is monitored by automatically isolating one or both intakes if radioactivity is detected in the MCR intake plena. The exhaust air in the fuel handling area is monitored in the unlikely event of a radiological release to divert the exhaust through a filter train before being released to the environment.

The ranges and sensitivities of the monitors are based on the maximum and minimum expected concentrations of radioactive material for normal plant operation including AOOs and postulated accidents in accordance with 10 CFR 20 limits and regulatory guidance.

11.5.2 System Description

11.5.2.1 Monitor Design and Configuration

Process, effluent, and airborne radiation monitors typically consist of components such as a microprocessor, one or more detectors, a shielded detection chamber, a sample pump, flow instrumentation, and associated tubing and cabling.

Each process, effluent, and airborne radiation monitor is located in an easily accessible area and has sufficient shielding to provide reasonable assurance that the required sensitivity is achieved at the design background radiation level for the area. This approach is consistent with NRC RG 8.8 (Reference 28) and NRC RG 8.10 (Reference 29). Instrumentation and sensors are provided to detect component failures such as loss of power, loss of sample flow, check source response failure, and loss of detector signal.

Radiation level signals, alarms, and operation status alarms are generated by each monitor microprocessor and are transmitted to IPS, QIAS, and other interfacing systems. Alarm relay contacts are provided for alert-radiation, high-radiation, and operation status alarms.

For some monitors, the high-radiation alarm contacts are used to initiate control functions to terminate batch releases or to divert flow from one location to another. The operation status alarm is initiated by the microprocessor if conditions indicate that the monitor is not operating properly.

Radiation monitoring equipment is designed for service based on expected environmental conditions during normal operation and AOOs. These conditions include temperature, pressure, humidity, chemical spray (where applicable), and radiation exposure. Post-accident radiation monitors conform with NRC RG 1.97 including equipment qualification, redundancy, power source, channel availability, quality assurance, display and recording, range, interfaces, testing, calibration, and human factors engineering recommendations. Further description of conformance with NRC RG 1.97 is contained in Subsections 7.1.2.44 and 7.5.2.1.

The RMS has an integral activated check source similar to the sample isotope to be detected to monitor proper system response automatically.

The RMS has provisions at the monitor/sampler skid for maintenance and repair. The RMS subcomponents are designed to be quickly and easily dismantled for corrective or routine maintenance such as cleaning. The sample chamber is designed so that a cleaning solution may be introduced into it and be drained.

Safety-related monitors are powered from a Class 1E 120 Vac distribution panel in the instrument power system. Information on instrumentation and control power is provided in Subsection 8.3.2.

11.5.2.2 Gaseous PERMSS

In accordance with ANSI/HPS N13.1, sample tubing for gaseous monitors is installed and routed to minimize interference with sample integrity.

The following paragraphs contain descriptions of the monitors in the gaseous process and effluent radiation monitoring system. Each monitor is listed along with associated parameters (also see Table 11.5-1).

a. High-energy line break area HVAC effluent monitors (RE-006, 007)

   A monitor at the exhaust air cleaning unit (ACU) inlet in the auxiliary building detects particulate, iodine and noble gas activities, and the other monitor at the outlet of the exhaust ACU has a particulate and iodine sampler.

b. Auxiliary building controlled area I, II HVAC normal/emergency exhaust ACU inlet effluent monitors (RE-013, 014, 017, and 018)

   Four monitors with particulate, gas, and iodine channels are provided to monitor HVAC effluent from the auxiliary building.

c. Auxiliary building controlled area HVAC normal/emergency exhaust effluent monitors (RE-015, 016, 019, and 020)

   Four samplers for particulate and iodine are provided to monitor HVAC effluent from the auxiliary building controlled area HVAC filter discharge.

d. Containment purge effluent monitor (RE-037)

A monitor with air particulate, gas, and iodine channels is provided to monitor containment purge effluent.

e. Containment air monitors (RE-039A and 040B)

Two monitors with air particulate, gas, and iodine channels are provided to monitor the radiation level in the containment. The wet parts of the detectors maintain pressure boundary integrity during normal conditions. The containment air monitors continuously measure, indicate, and record the radioactivity of particulate, iodine, and noble gas in a sample of air extracted from the containment. The sample lines of these offline monitors are provided with heat tracing to prevent dew condensation and are purged before sampling to provide reasonable assurance that samples are representative. The purge gas is routed back to the containment atmosphere.

f. Fuel handling area HVAC effluent monitor (RE-043)

A monitor with air particulate, gas, and iodine channels is provided to monitor the fuel handling area HVAC effluent.

g. Condenser vacuum pump vent effluent monitor (RE-063)

A monitor with a gas channel, air particulate, and iodine sampler is provided to monitor the condenser vacuum system effluent.

h. MCR air intake monitors (RE-071A, 072B, 073B, and 074B)

Two monitors per division (a total of four monitors) are provided with gas channels to monitor each of the intakes. The monitors are interlocked with the makeup air cleaning unit and MCR air intake dampers. On a high-radiation emergency signal, the outside air intake damper, which is open for normal operation, automatically closes and the air is routed through the makeup air-cleaning unit. The channels used for monitoring are Class 1E.

i. Gaseous radwaste system exhaust monitor (RE-080)

   A monitor with gas channels is provided to monitor the radiation level of gaseous radwaste system exhaust to the compound building vent stack.

j. Compound building HVAC effluent monitor (RE-082)

   One monitor with air particulate and iodine sampler is provided to monitor compound building HVAC effluent.

k. Compound building exhaust ACU inlet monitor (RE-083)

   A monitor with air particulate, iodine, and gas channels is provided to monitor the compound building plant area.

l. Compound building hot machine shop monitor (RE-084)

   One monitor with air particulate, iodine, and gas channels is provided to monitor the compound building hot machine shop.

m. Main steam line area and N-16 radiation monitors (RE-217, 218, 219, and 220)

   These monitors are located near the main steam safety valves and main steam atmospheric dump valves. Alarms are provided in the MCR to alert the operator when these monitors detect the PTS leakage due to a steam generator leakage. The method of detecting the SG leak rate is described in Appendix 11B.

The RMS for the release point in the high-energy line break (HELB) area, auxiliary building, and compound building is described in Subsection 9.4.

11.5.2.3 Liquid PERMSS

Each liquid process and effluent monitor is described in the following paragraphs. A list of each monitor and associated parameters is given in Table 11.5-2.

a. Component cooling water supply header monitors (RE-111 and 112)

Homogeneous isokinetic sampling is considered in the sample location and flow rate in accordance with ANSI/HPS N13.1.

Component cooling water is sampled downstream of the component cooling water pumps and is continuously monitored by a gamma scintillation detector mounted in a shielded liquid sampler. After passing through the monitor, the sample is returned to the component cooling water system.

Activity detected above a predetermined setpoint is indicative of a leak into the component cooling water system from the RCS or one of the other systems containing radioactive fluids.

b. Liquid radwaste system effluent monitors (RE-183 and 184)

Two radiation monitors for waste monitor tank effluent, are installed. In the event that radioactivity in excess of a preset limit is detected in the waste liquid discharge flow, the liquid radwaste system effluent monitors actuate an alarm in the MCR/radwaste control room and terminate the discharge.

The LWMS is designed with dual radiation monitors on the treated effluent discharge line. A radiation level in any portion of the flow that exceeds the predetermined setpoint will trigger alarms in the MCR and the remote shutdown room for operator actions, simultaneously turn off the monitor tank pump, and close the effluent discharge valve that is under supervisory control. The status of the pumps and the position of the valve are indicated in the MCR and remote shutdown room for verification. The LWMS is designed with no release bypass.

In the event of failure of one or both radiation monitors, a failure (inoperable) signal will generate an alarm in the MCR. The discharge of the treated effluent will be terminated through operator action until the radiation monitor(s) is/are repaired or replaced.

A set of inspections, tests, analyses, and acceptance criteria is included in the Tier 1 Table 2.7.6.1-2, sub-item 6, to inspect, test, and verify the installation of the as-built dual radiation monitors and isolation valve on the sole LWMS discharge line. A report is to be prepared to confirm the installation, functionality, operability testing, and calibration of the dual radiation monitors and isolation valve.

c. Steam generator blowdown and downcomer monitors (RE-104, 185, and 186)

These offline monitors sample the SG blowdown and downcomer for radioactivity, which is indicative of PTS leakage. Samples from each of the SGs are continuously monitored by a detector mounted in a shielded liquid sampler. Samples are cooled down through secondary sample cooler rack, which is a part of the secondary sampling system described in Subsection 9.3.2.2.3, before being transferred to a local unit. After being monitored, the sample is passed back to the SG blowdown and downcomer systems.

d. Condensate polishing area sump water monitor (RE-164)

This monitor is an offline monitor that continuously monitors the condensate polishing area sump water for gross gamma activity. Upon receipt of a high-radiation signal, the discharge flow is automatically diverted to the liquid waste management system (LWMS) prior to release to the environment.

e. Condensate receiver tank monitor (RE-103)

This monitor uses an offline shielded liquid sampler and a gamma scintillation detection system to continuously monitor the effluent from the condensate receiver tank in the auxiliary steam system. Detection of high activity automatically terminates releases from the system; the effluent is routed to the equipment waste tank in the LWMS and initiates alarms to plant operators.

f. Collective sewage treatment sump area monitor (RE-109)

This is an offline monitor to detect gross gamma activity at the outlet of the collective sewage treatment sump area. Upon receipt of a high-radiation signal, the discharge flow is automatically diverted to the LWMS prior to release to the environment, and the high-radiation alarm is provided to the operator. This monitor is also used to detect gross gamma activity at the outlet of the condenser pit sumps.

g. Miscellaneous process liquid monitors (RE-204 and 265)

   Two monitors are provided to continuously monitor the radiation levels of the letdown and gas stripper outlets in the CVCS. Detailed information for these monitors is presented in Subsection 9.3.4.5.5.

h. Essential service water pump discharge header monitors (RE-113 and 114)

   Two monitors for essential service water discharge header are installed. The essential service water is sampled downstream of the component cooling water heat exchangers and is continuously monitored by a gamma scintillation detector mounted in a shielded liquid sampler. After passing through the monitor, the sample is returned to the essential service water system. Activity detected above background is indicative of a leak into the essential service water system from the ultimate heat sink basins or one of the other systems containing radioactive fluids.

The sample lines for gaseous PERMSS are sloped down toward the monitor skid, and the use of sample line fittings such as unions, elbows, and tees are avoided to the extent practical. Setpoints, the calibration method, and the frequency for safety-related monitors are described in Subsections 12.3.4.1.6 and 12.3.4.1.1.

11.5.2.4 Design Features for Minimization of Contamination

The APR1400 is designed with specific features to meet the requirements of 10 CFR 20.1406 and Regulatory Guide 4.21. The basic principles of RG 4.21, and the methods of control suggested in the regulations, are specifically delineated in four design objectives and two operational objectives described in Subsection 12.4.2 of this DCD. The following evaluation summarizes the primary features to address the design and operational objectives for the RMS.

The RMS has been evaluated for leak identification from the SSCs that contain radioactive or potentially radioactive materials, the areas and pathways where probable leak may occur, and methods of control incorporated in the design of the system. The leak identification evaluation indicated that the RMS is designed to facilitate the identification of leaks, provide prompt assessment and evaluation, and initiate responses to isolate and mitigate leaked areas. Thus unintended contamination of the facility and the environment is minimized or prevented by the SSC design, supplemented by operational procedures and programs for inspection and maintenance activities.

Prevention/Minimization of Unintended Contamination

a. PERMS uses both offline and inline type detectors. Process and effluent radiation monitors that come in contact with process fluids are fabricated with stainless steel to prevent unintended contamination. Readout and alarm units are installed in low-radiation areas.

b. The piping is stainless steel material and of welded construction, thus minimizing leakage and unintended contamination of the facility and the environment.

c. Area radiation monitors are installed in normally accessible locations. In the event that a detector is located in a high-radiation area, the local readout and alarm unit are located in low-radiation areas for remote readout from the detector.

Adequate and Early Leak Detection

The RMS is designed for automated operation to provide early alarms for process and area monitoring. Radiation levels and alarms are displayed in the MCR, with the exception of the compound building radiation levels and alarms which are displayed in the radwaste control room.

Reduction of Cross-Contamination, Decontamination, and Waste Generation

a. The monitors are designed with life-cycle planning through the use of nuclear industry-proven materials compatible with the chemical, physical, and radioactive environment, thus minimizing waste generation.

b. PERMS monitors are provided with means for the purging or flushing the detector assemblies using nitrogen gas or demineralized water.

c. PERMS returns process fluids back to the system to minimize waste generation.

Decommissioning Planning

a. The SSCs are designed for the full service life and are fabricated as individual assemblies for easy removal.

b. The SSCs are designed with decontamination capabilities. Design features such as welding techniques used and surface finishes are intended to minimize the need for decontamination, and hence reduce waste generation.

c. The RMS system is designed with no embedded or buried piping, thus preventing unintended contamination due to the leaking of buried or embedded piping.

Operations and documentation

a. The RMS maintains a continuous record of radiation levels for documentation purposes. Plant personnel can access all RMS information from the MCR operator console.

b. The RMS piping and components are located in the reactor containment building, compound building, and turbine generator building. Adequate ingress and egress spaces are provided for prompt assessment and appropriate responses, when required.

c. Operational procedures and maintenance programs are to be prepared by the COL applicant (COL 11.5 (8)). Procedures and maintenance programs are to be completed before fuel is loaded for commissioning.

d. Complete documentation of design, construction, design modifications, field changes, and operations is to be maintained by the COL applicant. Documentation requirements are included as a COL information item.

Site Radiological Environmental Monitoring

The RMS is designed to provide data for plant operation. The RMS components do not process or generate wastes that can impact contamination of facility and the environment. The RMS is designed to provide data for plant operation and the data can be used to support the radiological environmental monitoring program. The program is included as a COL information item (COL 11.5 (9)).

11.5.3 Effluent Monitoring and Sampling

Effluent monitoring and sampling instruments are provided for monitoring the containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident (LOCA) fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including AOOs and from postulated accidents, as shown in Tables 11.5-1 and 11.5-2.

Periodic sampling is performed to supplement the function of the process and effluent radiation monitors. The sampling programs conform with NRC RG 1.21 and the sampling requirements defined in the Technical Specifications.

Special provisions are made for post-accident sampling of effluent pathways in accordance with the guidelines of NUREG-0737, 10 CFR 50, Appendix I, and NRC RG 1.97. See Subsection 9.3.2 for additional post-accident sampling details.

Sampling locations conform with NRC RG 1.21 and NUREG-0800. Post-accident sampling points and equipment conform with NUREG-0737 and NRC RG 1.97.

The containment atmosphere and the liquid radioactive waste tanks are sampled prior to release to the environment. An analysis is performed to determine constituent radionuclides in the liquid radioactive waste tanks, to set the proper release rates and isolation setpoints in accordance with 10 CFR 20 limits. The containment purge isolation setpoint is determined so that there is reasonable assurance that 10 CFR 20 limits will be met.

Sampling equipment is designed to prevent the spread of contaminants and radiation exposure to operating personnel while taking grab samples and to include provisions for a rapid change-out of filter elements to limit possible radiation exposure to operating personnel. Sampling equipment is modularized to the maximum extent possible for quick component change-out and calibration.

Design requirements regarding the ALARA provisions of NRC RG 8.8 and 8.10 provide reasonable assurance of conformance with the occupational dose limits of 10 CFR 20.1201, 10 CFR 20.1202, and the occupational limits (ALI and DAC) in Table 1 of Appendix B to 10 CFR 20. Sufficient shielding is provided to all equipment of the PERMSS, and any equipment that requires frequent maintenance, inspections, testing, and calibration is designed so that radiation exposures to operating and maintenance personnel are maintained ALARA. In addition, instrument locations provide sufficient space for easy access, operation, inspections, testing, and maintenance to maintain personnel exposures ALARA. High-radiation alarms and interlock signals are provided to operating and maintenance personnel to meet ALARA requirements.

11.5.4 Process Monitoring and Sampling

Process and effluent monitoring and sampling instruments include the means to control the release of radioactive materials to meet the requirements of 10 CFR 50, Appendix A, GDC 60 during normal operation and AOOs. Automatic isolation functions are also implemented to conform with 10 CFR 50, Appendix A, GDC 60.

Radiological monitoring and sampling instruments for fuel and waste storage are provided to detect conditions that may result in the loss of residual heat removal capability and excessive radiation levels and to initiate appropriate safety actions. These are to meet the requirements of 10 CFR 50, Appendix A, GDC 63.

11.5.5 Combined License Information

COL 11.5(1) The COL applicant is to determine the WARN and ALARM setpoints of the PERMSS based on the site-specific conditions and operational requirements.

COL 11.5(2) The COL applicant is to develop an annual report that specifies the quantity of each principal radionuclide released to unrestricted areas in liquid and gaseous effluents.

COL 11.5(3) The COL applicant is to provide site-specific procedures that conform with the numerical guides of 10 CFR 50.34a and 10 CFR 50, Appendix I.

COL 11.5(4) The COL applicant is to prepare an ODCM that contains a description of the methodology and parameters for calculation of the offsite doses for the gaseous and liquid effluents. The COL applicant is to follow NEI 07-09A as an alternative to providing an offsite dose calculation manual.

COL 11.5(5) The COL applicant is to provide analytical procedures and sensitivity for selected radioanalytical methods and types of sampling media for site-specific matter.

COL 11.5(6) The COL applicant is to develop the calibration procedures in accordance with NRC RG 1.33 and 4.15.

COL 11.5(7) The COL applicant is to develop detailed location and tubing installation and provide the sampling method including the sampling time to acquire representative sampling.

COL 11.5(8) The COL applicant is to provide operational procedures and maintenance programs related to leak detection and contamination control.

COL 11.5(9) The COL applicant is to develop a radiological and environmental monitoring program, taking into consideration local land use and census data in identifying all potential radiation exposure pathways. The COL applicant is to follow NEI 07-09A as an alternative to providing a radiological and environmental monitoring program.

11.5.6 References

1. NUREG-0800, Standard Review Plan, Section 11.5, "Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems," Rev. 5, U.S. Nuclear Regulatory Commission, May 2010.

2. 10 CFR Part 20, “Standard for Protection Against Radiation,” U.S. Nuclear Regulatory Commission.

3. 10 CFR Part 50, Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion,” U.S. Nuclear Regulatory Commission.

4. 10 CFR Part 50, Appendix A, General Design Criterion 13, “Instrumentation and Control,” U.S. Nuclear Regulatory Commission.

   10 CFR Part 50, Appendix A, General Design Criterion 60, “Control of Releases of Radioactive Material to the Environment,” U.S. Nuclear Regulatory Commission.

   10 CFR Part 50, Appendix A, General Design Criterion 61, “Fuel Storage and Handling and Radioactivity Control,” U.S. Nuclear Regulatory Commission.

   10 CFR Part 50, Appendix A, General Design Criterion 63, “Monitoring Fuel and Waste Storage,” U.S. Nuclear Regulatory Commission.

   10 CFR Part 50, Appendix A, General Design Criterion 64, “Monitoring Radioactivity Releases,” U.S. Nuclear Regulatory Commission.

5. Regulatory Guide 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” U.S. Nuclear Regulatory Commission, June 2009.

6. NUREG-0737, “Clarification of TMI Action Plan Requirements,” U.S. Nuclear Regulatory Commission, November 1980.

7. Regulatory Guide 1.45, “Reactor Coolant Pressure Boundary Leakage Detection Systems,” Rev. 1, U.S. Nuclear Regulatory Commission, May 2008.

8. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Rev. 4, U.S. Nuclear Regulatory Commission, June 2006.

9. 10 CFR 50.34, "Contents of applications; technical information, Domestic Licensing of Production and Utilization Facilities" U.S. Nuclear Regulatory Commission.

10. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses," U.S. Nuclear Regulatory Commission, July 1981.

11. ANSI N42.18, "Specification and Performance of On-site Instrumentation for Continuously Monitoring Radioactivity in Effluents," American National Standards Institute, 2004.

12. Regulatory Guide 1.143, "Design Guide for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Rev. 2, U.S. Nuclear Regulatory Commission, November 2001.

13. NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, April 1991.

14. NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1978.

15. 10 CFR Part 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Concentrations for Release to Sewerage," U.S. Nuclear Regulatory Commission.

16. 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," U.S. Nuclear Regulatory Commission.

17. 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public," U.S. Nuclear Regulatory Commission.

18. 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations," Environmental Protection Agency.

19. 10 CFR 50.36a, “Technical Specifications on Effluents from Nuclear Power Reactors,” U.S. Nuclear Regulatory Commission.

20. 10 CFR 50.34a, “Design Objectives for Equipment to Control Releases of Radioactive Material in Effluents-Nuclear Power Reactors,” U.S. Nuclear Regulatory Commission.

21. NEI 07-09A, “Generic Template Guidance for Offsite Dose Calculation Manual Program Description,” Nuclear Energy Institute, March 2009.

22. Regulatory Guide 1.109, “Calculation of Annual Doses to Man from Routine Release of Reactor Effluents for the Purpose of Evaluating Compliance with 10 Code of Federal Regulations Part 50, Appendix I,” Rev. 1, U.S. Nuclear Regulatory Commission, October 1977.

23. Regulatory Guide 1.111, “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors,” Rev. 1, U.S. Nuclear Regulatory Commission, July 1977.

24. Regulatory Guide 1.113, “Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I,” Rev. 1, U.S. Nuclear Regulatory Commission, April 1997.

25. Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operation),” U.S. Nuclear Regulatory Commission, February 1978.

26. Regulatory Guide 4.15, “Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) - Effluent Streams and the Environment,” Rev. 2, U.S. Nuclear Regulatory Commission, July 2007.

27. ANSI/HPS N13.1, “Sampling and Monitoring Releases of Airborne Radioactive Substance from the Stack and Ducts of Nuclear Facilities,” Health Physics Society, January 1999.

28. Regulatory Guide 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable,” Rev. 3, U.S. Nuclear Regulatory Commission, June 1978.

29. Regulatory Guide 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures as low as Is Reasonably Achievable,” Rev. 1, U.S. Nuclear Regulatory Commission, May 1977.

30. 10 CFR 20.1406, “Minimization of Contamination,” U.S. Nuclear Regulatory Commission.

31. Regulatory Guide 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” U.S. Nuclear Regulatory Commission, June 2008.

32. ANSI/ANS 51.1, “American National Standard Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants,” American Nuclear Society, 1983.

Table 11.5-1 (1 of 3)

<u>Gaseous Process and Effluent Radiation Monitors</u>

Location	Tag No.	Class(1)				Range (Bq/cc)(2)					Function and Remarks
		S	SE	Q	E	Particulate Gross β	I-131 γ	Gas Gross β	Liquid Gross γ	Area	
High-energy line break area HVAC effluent (offline)	RE-006	N	III	A	N	Sampler	Sampler	N/A	N/A	N/A	Analysis
High-energy line break area exhaust ACU inlet (offline)	RE-007	N	III	A	N	3.7×10^{-7} to 3.7×10^{-1}	3.7×10^{-7} to 3.7×10^{-1}	3.7×10^{-2} to 3.7×10^{7}	N/A	N/A	Alarm (MCR)
Auxiliary building controlled area (I, II) HVAC normal/emergency exhaust ACU inlet (offline)	RE-013 RE-014 RE-017 RE-018	N	II	A	N	3.7×10^{-7} to 3.7×10^{6}	3.7×10^{-7} to 3.7×10^{6}	3.7×10^{-2} to 3.7×10^{7}	N/A	N/A	Alarm (MCR)
Auxiliary building controlled area (I, II) HVAC normal/emergency exhaust ACU effluent (offline)	RE-015 RE-016 RE-019 RE-020	N	II	A	N	Sampler	Sampler	N/A	N/A	N/A	Analysis
Containment purge effluent (offline)	RE-037	N	II	A	N	3.7×10^{-7} to 3.7×10^{6}	3.7×10^{-7} to 3.7×10^{6}	3.7×10^{-2} to 3.7×10^{9}	N/A	N/A	Alarm (MCR), containment building purge stop

Table 11.5-1 (2 of 3)

Location	Tag No.	Class(1)				Range (Bq/cc)(2)					Function and Remarks
		S	SE	Q	E	Particulate Gross β	I-131 γ	Gas Gross β	Liquid Gross γ	Area	
Containment air (offline)	RE-039A RE-040B	3	I	Q	B	3.7×10^{-5} to 3.7×10^{1}	3.7×10^{-5} to 3.7×10^{1}	3.7×10^{-2} to 3.7×10^{5}	N/A	N/A	Alarm (MCR), leak detection
Fuel handling area HVAC effluent (offline)	RE-043	N	II	A	N	3.7×10^{-7} to 3.7×10^{-1}	3.7×10^{-7} to 3.7×10^{-1}	3.7×10^{-2} to 3.7×10^{7}	N/A	N/A	Alarm (MCR) isolation interlock diversion interlock
Condenser vacuum pump vent effluent (offline)	RE-063	N	III	A	N	Sampler	Sampler	3.7×10^{-2} to 3.7×10^{3}	N/A	N/A	Alarm (MCR), diversion interlock analysis
MCR air intake (inline)	RE-071A RE-072B RE-073A RE-074B	3	I	Q	A B A B	N/A	N/A	3.7×10^{-2} to 3.7×10^{3}	N/A	N/A	Alarm (MCR), CREVAS
Gaseous radwaste system exhaust (offline)	RE-080	N	III	A	N	N/A	N/A	3.7×10^{1} to 3.7×10^{6}	N/A	N/A	Alarm (MCR) isolation interlock
Compound building HVAC effluent (offline)	RE-082	N	III	A	N	Sampler	Sampler	N/A	N/A	N/A	Analysis
Main steam line	RE-217 RE-218 RE-219 RE-220	N	II	T	N	N/A	N/A	N/A	N/A	$10^{-4} \sim 10^{2}$ (Note 3)	Alarm (MCR, Local)

Table 11.5-1 (3 of 3)

Location	Tag No.	Class(1)				Range (Bq/cc)(2)					Function and Remarks
		S	SE	Q	E	Particulate Gross β	I-131 γ	Gas Gross β	Liquid Gross γ	Area	
Compound building exhaust ACU inlet (offline)	RE-083	N	III	A	N	3.7×10^{-7} to 3.7×10^{-1}	3.7×10^{-7} to 3.7×10^{-1}	3.7×10^{-2} to 3.7×10^{6}	N/A	N/A	Alarm (MCR), isolation interlock, diversion interlock from normal to emergency ventilation
Compound building hot machine shop	RE-084	N	III	A	N	3.7×10^{-7} to 3.7×10^{-1}	3.7×10^{-7} to 3.7×10^{-1}	3.7×10^{-2} to 3.7×10^{3}	N/A	N/A	Alarm (MCR)

(1) S = Safety Class per ANSI/ANS 51.1 (Reference 32): 1 = SC-1, 2 = SC-2, 3 = SC-3, N = NNS
SE = Seismic Category: I, II, III
E = Electrical Class: A, B, C, D=Class 1E Separation Division, N = Non-Class 1E
Q = Quality Class: Q, A, S

(2) Detector type and calibration nuclide for each measurement:
Particulate Gross β = β scintillator with Cs-137
Gas Gross β = β scintillator with Kr-85
Liquid Gross γ = γ scintillator with Cs-137
Iodine γ = γ scintillator with Ba-133

(3) Detector type for area radiation monitor is GM tube or ionization chamber.

Table 11.5-2 (1 of 2)

Liquid Process and Effluent Radiation Monitors

Location	Tag No.	Class (1)				Range (Bq/cc) (2)					Function and Remarks
		S	SE	Q	E	Particulate Gross β	I-131 γ	Gas Gross β	Liquid Gross γ	Area	
CVCS letdown	CV-RE-204	N	II	A	N	N/A	N/A	N/A	3.7×10^{0} to 3.7×10^{6}	N/A	Alarm (MCR)
CVCS gas stripper effluent	CV-RE-265	N	II	A	N	N/A	N/A	N/A	3.7×10^{0} to 3.7×10^{5}	N/A	Alarm (MCR)
Condensate receiver tank	RE-103	N	III	S	N	N/A	N/A	N/A	3.7×10^{-2} to 3.7×10^{3}	N/A	Alarm (MCR), diversion interlock
Steam generator blowdown and downcomer	RE-104 RE-185 RE-186	N	II III III	A	N	N/A	N/A	N/A	3.7×10^{-2} to 3.7×10^{3}	N/A	Alarm (MCR), leak detection isolation interlock
CCW supply header	RE-111 RE-112	N	II	A	N	N/A	N/A	N/A	3.7×10^{-2} to 3.7×10^{3}	N/A	Alarm (MCR), leak detection isolation of inlet/outlet valve of heat exchanger
Essential service water (ESW) pump discharge headers	RE-113 RE-114	N	II	A	N	N/A	N/A	N/A	3.7×10^{-2} to 3.7×10^{3}	N/A	Alarm (MCR), leak detection

Table 11.5-2 (2 of 2)

Location	Tag No.	Class(1)				Range (Bq/cc)(2)					Function and Remarks
		S	SE	Q	E	Particulate Gross β	I-131 γ	Gas Gross β	Liquid Gross γ	Area	
CPP area sump water	RE-164	N	III	S	N	N/A	N/A	N/A	3.7×10^{-2} to 3.7×10^{3}	N/A	Alarm (MCR), pump stop signal
Liquid radwaste system effluent	RE-183 RE-184	N	III	A	N	N/A	N/A	N/A	3.7×10^{-2} to 3.7×10^{3}	N/A	Alarm (MCR), isolation interlock
[Collective sewage treatment sump]	RE-190	N	III	A	N	N/A	N/A	N/A	3.7×10^{-3} to 3.7×10^{3}	N/A	Alarm, pump stop signal

(1) S = Safety Class per ANSI/ANS 51.1 (Reference 32): 1 = SC-1, 2 = SC-2, 3 = SC-3, N = NNS
SE = Seismic Category: I, II, III
E = Electrical Class: A, B, C, D = Class 1E Separation Division, N = Non-Class 1E
Q = Quality Class: Q, A, S

(2) Detector type and calibration nuclide for each measurement:
Particulate Gross β = β scintillator with Cs-137
Gas Gross β = β scintillator with Kr-85
Liquid Gross γ = γ scintillator with Cs-137
Iodine γ = γ scintillator with Ba-133

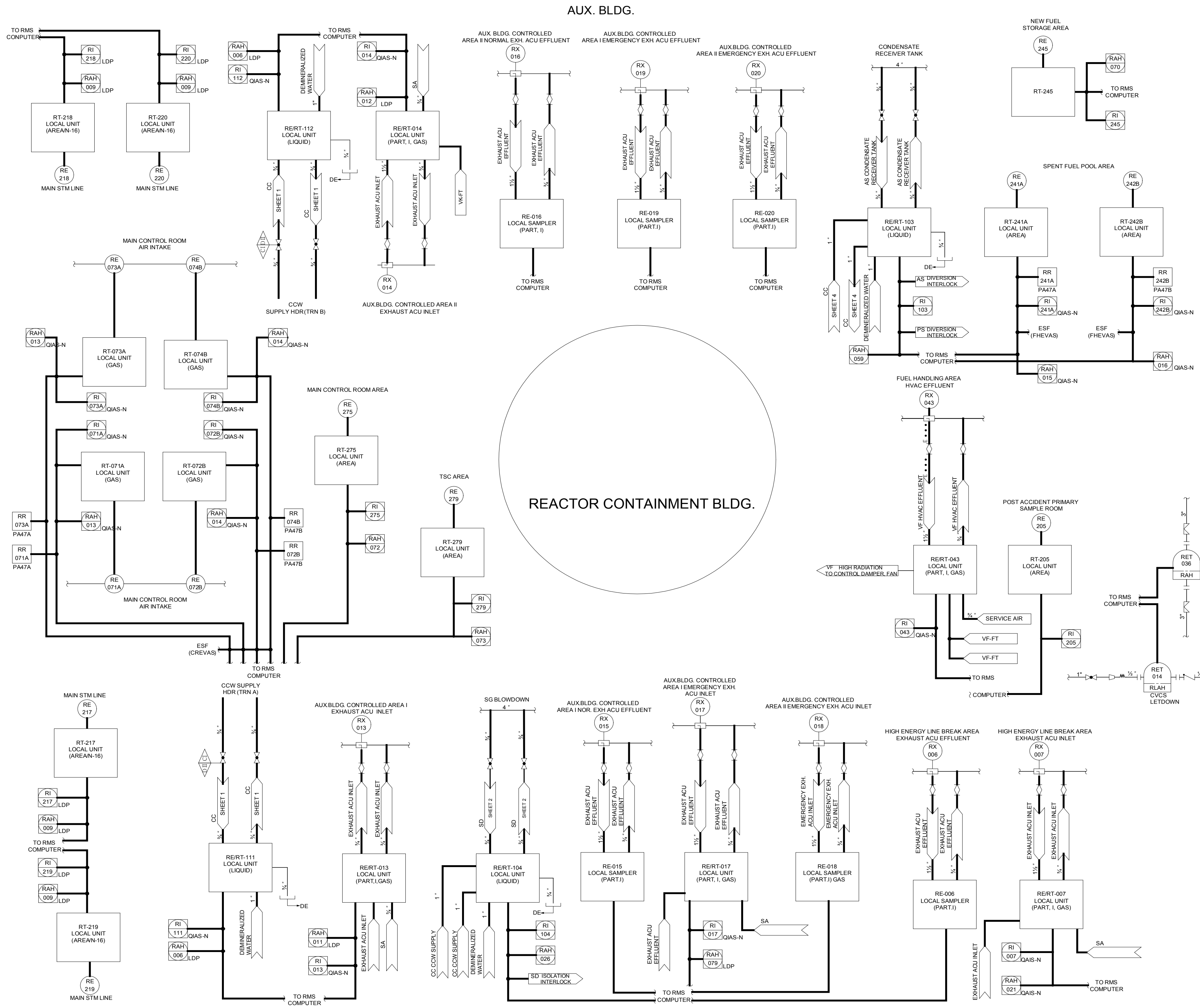


Figure 11.5-1 Radiation Monitoring System (PR) (1 of 3)

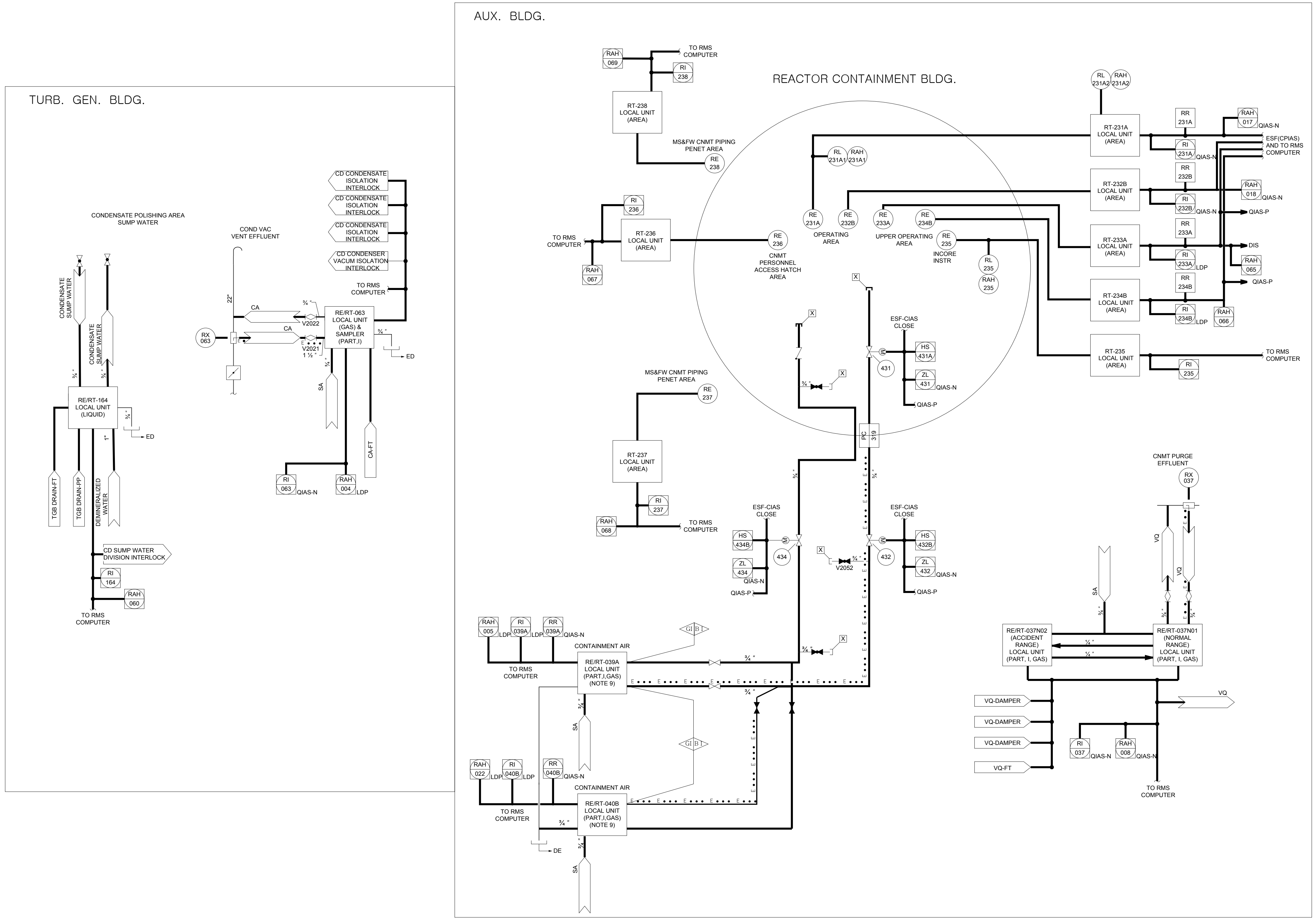


Figure 11.5-1 Radiation Monitoring System (PR) (2 of 3)

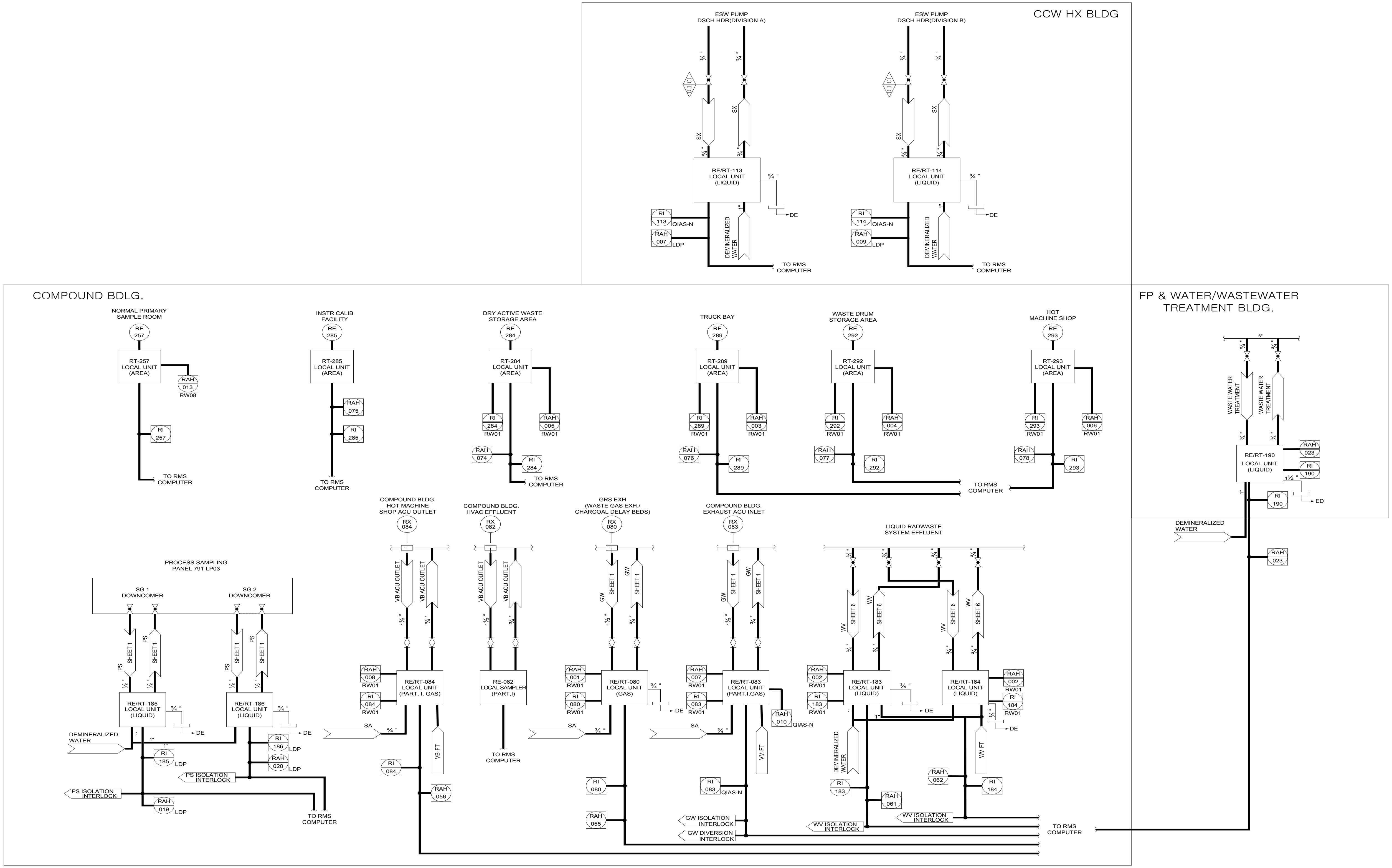


Figure 11.5-1 Radiation Monitoring System (PR) (3 of 3)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2A Location of Radiation Monitors at Plant (Reactor Containment Building El. 156′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2B Location of Radiation Monitors at Plant (Auxiliary Building El. 55′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2C Location of Radiation Monitors at Plant (Auxiliary Building El. 55′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2D   Location of Radiation Monitors at Plant (Auxiliary Building El. 55′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2E Location of Radiation Monitors at Plant (Auxiliary Building El. 78′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2F Location of Radiation Monitors at Plant (Auxiliary Building El. 100′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2G Location of Radiation Monitors at Plant (Auxiliary Building El. 120′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2H Location of Radiation Monitors at Plant (Auxiliary Building El. 120′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2I Location of Radiation Monitors at Plant (Auxiliary Building El. 120′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2J   Location of Radiation Monitors at Plant (Auxiliary Building El. 120′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2K Location of Radiation Monitors at Plant (Auxiliary Building El. 137′-6″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2L Location of Radiation Monitors at Plant (Auxiliary Building El. 137′-6″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2M Location of Radiation Monitors at Plant (Auxiliary Building El. 137′-6″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2N Location of Radiation Monitors at Plant (Auxiliary Building El. 156′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2O Location of Radiation Monitors at Plant (Auxiliary Building El. 156′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2P Location of Radiation Monitors at Plant (Auxiliary Building El. 174′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2Q Location of Radiation Monitors at Plant (Auxiliary Building El. 174′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2R Location of Radiation Monitors at Plant (Compound Building El. 63′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2S Location of Radiation Monitors at Plant (Compound Building El. 85′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2T Location of Radiation Monitors at Plant (Compound Building El. 100′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2U   Location of Radiation Monitors at Plant (Compound Building El. 100′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2V Location of Radiation Monitors at Plant (Compound Building El. 120′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2W   Location of Radiation Monitors at Plant (Compound Building El. 139′-6″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2X Location of Radiation Monitors at Plant (CCW HX Building El. 113′-0″ and 127′-5″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2Y Location of Radiation Monitors at Plant (Turbine Generator Building El. 73′-0″)

Security-Related Information – Withhold Under 10 CFR 2.390

Figure 11.5-2Z Location of Radiation Monitors at Plant (Turbine Generator Building El. 170′-0″)

APPENDIX 11A

CORE RESIDENCE TIMES

APPENDIX 11A – CORE RESIDENCE TIMES

The derivation of the core residence times for circulating crud, as shown in Subsection 11.1.3, is as follows:

Circulating Crud

The number of radioactive atoms (N_f) in the crud film on in-core surfaces at any time is:

$$\frac{dN_f}{dt} = \Sigma_i \phi - \lambda_i N_f \qquad \text{(Eq. 11A-1)}$$

Solving for N_f yields the following:

$$N_f = \frac{\Sigma_i \phi}{\lambda_i}\left(1 - e^{-\lambda_i t_{res}}\right), \text{atoms/g} \qquad \text{(Eq. 11A-2)}$$

Where:

$\Sigma_i\phi$ = activation rate for each isotope i, d/g-sec

λ_i = decay constant for each isotope i, sec^{-1}

t_{res} = desired core residence time, seconds

The number of radioactive atoms (N_c) released to the reactor coolant at any time is:

$$\frac{dN_c}{dt} = N_f\{ER\}A_c - (\alpha + \beta + \lambda_i)\, N_c, \text{atoms/sec}$$

Solving for N_c yields the following:

$$N_c = \frac{N_f\{ER\}A_c}{(\alpha + \beta) + \lambda_i}\left(1 - e^{-(\alpha+\beta+\lambda_i)t}\right) \qquad \text{(Eq. 11A-3)}$$

Where:

{ER} = erosion rate, g/cm^2-sec

A_C = core surface area, cm^2

α = plateout rate, sec^{-1}

β = purification cleanup rate, sec^{-1}

λ_i = decay constant, sec^{-1}

Total amount of crud (M_c) released to the reactor coolant at any time is:

$$\frac{dM_c}{dt} = \{ER\}A_T - (\alpha + \beta)M_c \qquad \text{(Eq. 11A-4)}$$

Where M_c includes both radioactive and nonradioactive material.

Solving for M_c yields:

$$M_c = \frac{\{ER\}A_T}{\alpha+\beta}\left(1 - e^{-(\alpha+\beta)t}\right), \text{grams} \qquad \text{(Eq. 11A-5)}$$

Where:

$\{ER\}$ = erosion rate, g/cm^2-sec

A_T = total system area, cm^2

α = plateout rate, sec^{-1}

β = purification cleanup rate, sec^{-1}

The activity (A_i) of the crud released to the reactor coolant is:

$$A_i = \frac{\lambda_i N_c}{M_c}, \text{Bq/g-crud} \qquad \text{(Eq. 11A-6)}$$

Substituting the values of Nc and Mc into the above expression and assuming λ_i is small compared to α and β, the activity of the crud is as follows:

$$A_i = \sum_i \phi\left(1 - e^{-\lambda_i t_{res}}\right)\frac{A_c}{A_T}, \text{Bq/g-crud} \qquad \text{(Eq. 11A-7)}$$

This activity (A_i) is also assumed to be the activity of the crud that plates out on out-of-core surfaces.

Solving Equation (11A-7) for t_{res} yields Eq. 11.1-8.

Deposited Crud

The activity (A_j) of the deposited crud is:

$$A_j = \lambda_j N_f = \sum_j \phi\left(1 - e^{-\lambda_j t_{res}}\right) \qquad \text{(Eq. 11A-8)}$$

Solving Eq. 11A-8 for t_{res} yields Eq. 11.1-9.

APPENDIX 11B

PRIMARY-TO-SECONDARY LEAKAGE DETECTION

APPENDIX 11B – PRIMARY-TO-SECONDARY LEAKAGE DETECTION

The methods used to monitor steam generator (SG) leakage are as follows: (1) the SG blowdown monitor, which samples the secondary side liquid of SG, (2) the main steam line N-16 monitor, which detects N-16 activity in the main steam, and (3) the condenser vacuum vent effluent monitor, which monitors non-condensable gases released from the condenser.

This appendix describes the capability of the N-16 monitor to detect a SG leak rate of 4.73 L/hr (30 gal/day), which is required by Nuclear Energy Institute (NEI) 97-06.

The main steam line N-16 monitors (RE-217 ~ RE-220), which are designed to detect N-16 concentration in the main steam, are usually operated in a multichannel analyzer mode with a certain energy window or in a gross counting mode with a lower-level discriminator that is set to detect only high-energy gamma radiation. The effectiveness of leakage monitoring via the N-16 detectors varies depending on the primary-to-secondary (PTS) leak rate, reactor power level, transit time from the core to the detector, and amount of uranium depletion over the life of the core.

In order to correlate the PTS leak rate to the N-16 activity concentration in the main steam line at the detector location, the following assumptions and models are used:

a. A neutron energy level of 10.2 MeV is used as the threshold energy for the O-16 (n, p) N-16 reaction.

b. The O-16 (n, p) N-16 cross section for the neutron flux spectrum above 10.2 MeV is determined based on a weighted average cross section for the reaction.

c. N-16 activity in the primary coolant at the location of the leakage is calculated using the method described in Subsection 12.2.1.1.2.

d. N-16 activity concentration decreases linearly as the reactor power decreases.

e. The primary coolant transient time is obtained from Subsection 12.2.1.1.2.

f. The length of the main steam lines inside containment is 32.04 m (105.13 ft).

g. The N-16 monitors are assumed to be installed at 0.61 m (2 ft) away from the containment wall.

The count rate of the N-16 monitor is determined using the following equation:

$$\text{Count rate} = A_V \cdot k$$

Where:

count rate = detector count rate in count per second (cps)

k = detection efficiency in (cps)/(Bq/cm^3), which represents the conversion factor between the detector count rate in cps and the N-16 concentration in Bq/cm^3

A_v = N-16 concentration in Bq/cm^3 corresponding to an SG leak rate

Parameter k is dependent on the configuration of the detection such as thickness of the pipe, types of material in the pipe and insulation, and distance between the probe and the pipe. The value A_v is determined as follows:

$$A_V = 0.2778 \frac{L \cdot A_p}{Q_p \cdot A_s} \exp\left(-\lambda t\right)$$

Where:

0.2778 = unit conversion factor. (cm^3/sec) / (L/hr)

L = SG leakage rate, L/hr

A_P = N-16 activity concentration at the leakage location, which is proportional to the core power, Bq/cm^3

Q_P = steaming rate for a corresponding power level, which is the flow rate of the steam in the secondary loop, cm/sec

A_S = surface area of main steam line, cm^2

λ = decay constant of N-16, sec^{-1} (=0.097 sec^{-1})

t = transit time ($t = t_1+t_2$) of N-16 from the leakage location to the N-16 detector, sec.

- t_1 = transit time between the leak location and the SG outlet nozzle in the SG
- t_2 = transit time from the SG outlet to detector

N-16 activity concentrations at the detector location calculated at a leak rate of 4.73 L/hr (30 gpd) with varying power levels are given in Table 11B-1 and Figure 11B-1. According to the calculation, the SG leakage rate of 4.73 L/hr (30 gal/day) can be detected within the N-16 concentrations range from 3.68 x 10^{-4} Bq/cm^3 (10% power) to 1.68 x 10^{-1} Bq/cm^3 (100% power). Figure 11B-2 presents the N-16 concentration in the main steam line as a function of SG leakage rate for different power levels. For the SG leakage rate of 23.66 L/hr (150 gal/day), which is the limiting condition of operation in the Technical Specifications, the N-16 concentrations range from 1.84 x 10^{-3} Bq/cm^3 (10 % power) to 8.38 x 10^{-1} Bq/cm^3 (100 % power).

Therefore, the N-16 detector, which is capable of monitoring N-16 concentrations from 1.0×10^{-4} to 1.0×10^{2} Bq/cm^3, satisfies the minimum SG leakage rate of 4.73 L/hr (30 gal/day) as well as the Technical Specification limit of 23.66 L/hr (150 gal/day).

Table 11B-1

N-16 Concentrations with Power Level at SG Leakage of 4.73 L/hr (30 gal/day)

Percent Power	N-16 Activity at Leakage Location, A_P (Bq/cm^3)	Steaming Rate, Q_P (cm/sec)	Surface Area of Main Steam Line, A_S (cm^2)	Transit Time, t (sec)	N-16 Activity at Detector Location, A_V (Bq/cm^3)
100	4.04E+06	3,784.397	4,146.694	7.24	1.68E-01
80	3.23E+06	2,905.354	4,146.694	9.07	1.46E-01
60	2.42E+06	2,057.705	4,146.694	12.19	1.14E-01
40	1.62E+06	1,291.133	4,146.694	18.22	6.79E-02
20	8.08E+05	585.8256	4,146.694	36.35	1.29E-02
10	4.04E+05	266.3952	4,146.694	73.96	3.68E-04

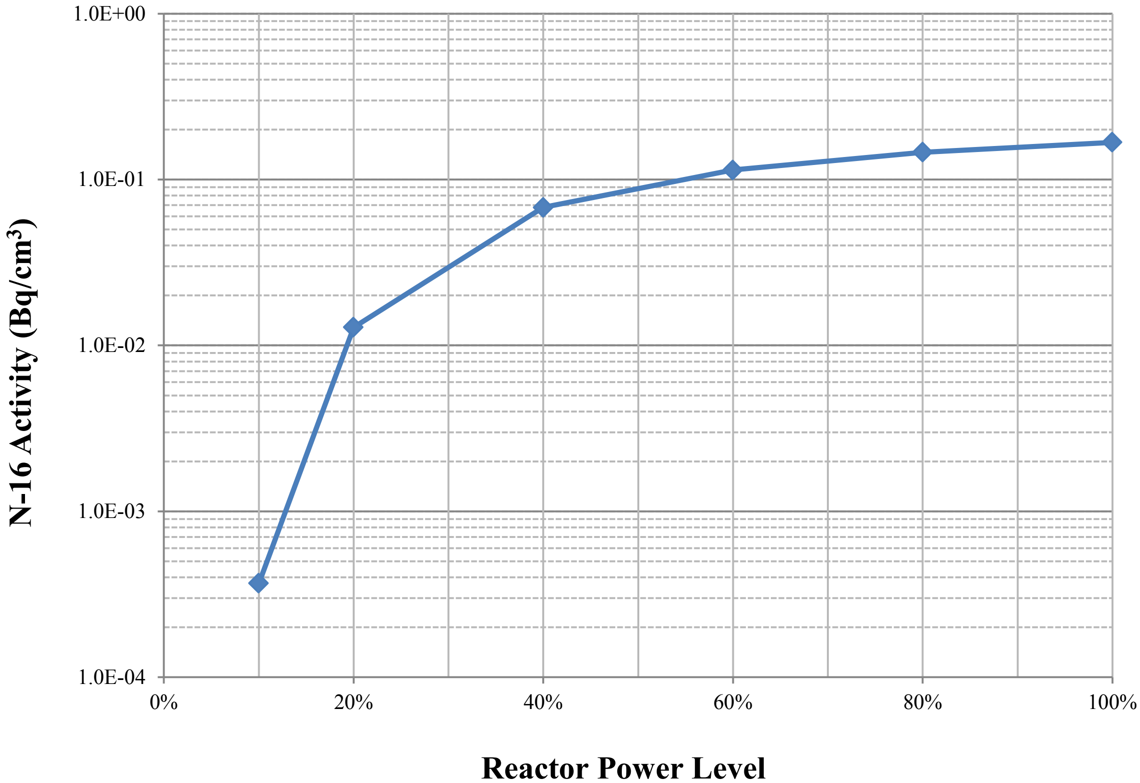


Figure 11B-1 N-16 Concentrations with Power at SG Leakage of 4.73 L/hr (30 gpd)

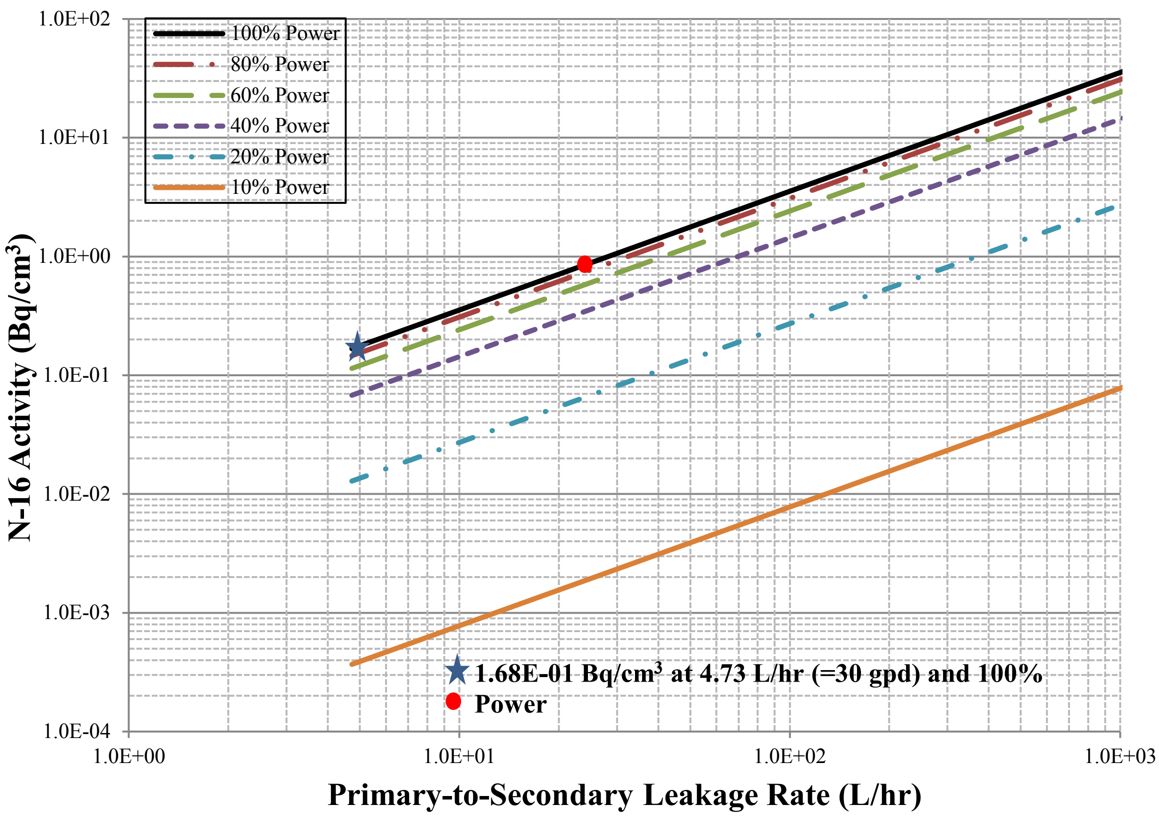


Figure 11B-2 N-16 Concentrations vs. Primary-to-Secondary Leakage Rate For Various Power Levels