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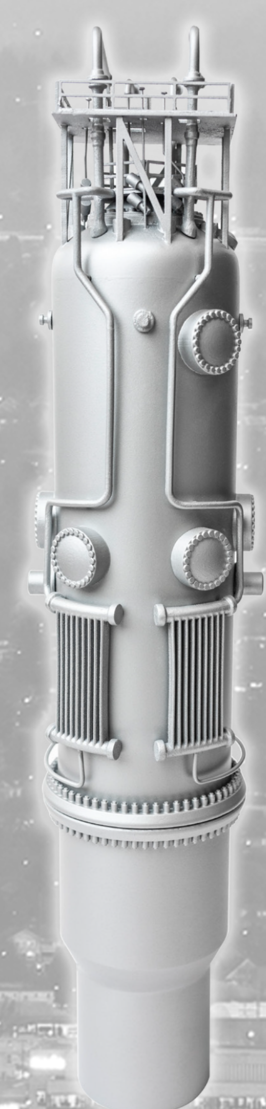
NuScale Standard Plant
Design Certification Application



Chapter Eleven

Radioactive Waste Management

PART 2 - TIER 2

Revision 3
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CHAPTER 11 RADIOACTIVE WASTE MANAGEMENT

11.1 Source Terms

Sources of radioactivity are created by fission and activation processes in the reactor. Fission products generated within the fuel pins during reactor operations have the potential to leak into the primary coolant through fuel pin cladding defects that may develop. Neutron activation of various isotopes, and corrosion and wear products found in the primary coolant form activation products. These activation products are independent of the failed fuel fraction. The secondary coolant may become contaminated by primary-to-secondary leakage through the steam generator. There is also the potential for the secondary coolant to be activated due to its proximity to the reactor core.

This section discusses two source terms for the primary and secondary coolants: a design basis source term and a realistic source term. The design basis source term is calculated using conservative assumptions and provides a basis for:

- design capacities of waste management components and performance of the waste management systems
- design of radiological monitoring equipment

The design basis source term is also used for the evaluation of shielding (General Design Criteria 61). The coolant source terms used for dose consequences of design basis events are found in Section 15.0. Equipment qualification is discussed in Section 3.11.

The second source term is a realistic source term and is derived by using realistic yet conservative input assumptions that are representative of normal operation. However, the realistic source term still maintains conservatism based on industry experience. The realistic source term is used to calculate the quantity of radioactive materials released annually in liquid and gaseous effluents during normal plant operations, including anticipated operational occurrences, to demonstrate compliance with effluent limits of 10 CFR 20 Appendix B, Table 2, and the "as low as reasonably achievable" objectives of 10 CFR 50 Appendix I. In addition, the realistic source term is used by the combined license applicant in a site-specific demonstration of compliance with the public exposure limits in 10 CFR 20.1301, 10 CFR 20.1302.

This section describes the methodology used to develop the primary and secondary coolant realistic source terms, which is also described in TR-1116-52065, (Reference 11.1-1). The primary and secondary coolant design basis source terms are developed using the same methodology, but using a higher failed fuel fraction. The development of an alternate methodology is necessary since the existing PWR-GALE code (NUREG-0017) was originally developed in the 1980s for the existing fleet of reactors and is not representative of the NuScale Power, LLC design.

The plant is designed with up to 12 NuScale Power Modules (NPMs), including individual containment vessels, partially immersed in a single pool of water, called the reactor pool. Because of this design, there is a potential for neutron activation of the reactor pool water. Additionally, the design is an integrated pressurized water reactor (PWR) design in which the reactor core, steam generators, and pressurizer are contained in a single reactor vessel. Given the relative proximity of the secondary coolant to the reactor core, there is the potential for neutron activation of the secondary coolant. However, the production of radionuclides in the secondary coolant is several orders of magnitude less than that in the primary coolant and is considered negligible. The production of radionuclides in the reactor pool water is evaluated independently of the activity within the primary or secondary coolant loops, as discussed in Section 12.2.1.

The resulting impact of these coolant source terms on the design of radioactive waste systems and normal effluents is described in Section 11.2 and Section 11.3. The resulting shielding design and occupational dose assessments are in Chapter 12.

The source terms presented herein demonstrate that the facility design complies with General Design Criteria 60 by suitably controlling the release of radioactive material.

11.1.1 Design Basis Reactor Coolant Activity

During reactor operations, radioactive fission, activation, and corrosion products are generated. The fission process generates both fission products in the fuel pins and neutrons in the reactor core. The neutrons also generate radionuclides by activating the coolant and corrosion products present in the primary coolant. The majority of the reactor core's radioactive source term is confined within the fuel pins. However, radioactive fission products have the potential to leak through fuel pin cladding defects and enter the primary coolant. This material can also travel from the primary coolant to the secondary coolant in the event of a steam generator tube leak.

The design basis source term assumes a conservative value of equivalent fuel defects an order-of-magnitude greater than the realistic coolant source term. This results in a design basis failed fuel fraction that is ten times greater than the realistic failed fuel fraction (see Table 11.1-2). These defects are assumed to be uniformly distributed throughout the reactor core. The primary coolant design basis source term is provided in Table 11.1-4.

11.1.1.1 Fission Products

The isotopic inventory is developed for a single fuel assembly conservatively irradiated to the value presented in Table 11.1-2. The quantity of each nuclide is calculated by ORIGEN-S. The ORIGEN-ARP calculation sequence was developed to speed the depletion calculations by interpolating previously defined sets of cross sections, thereby eliminating the necessity of transport calculations for every depletion case. The isotopic inventory values are developed for the end of the last irradiation time step and the end of each decay time step. An irradiation and decay calculation is performed for each enrichment, both with and without gadolinium-containing pins. The maximum inventory values for the modeled fuel enrichments are used. The resultant bounding reactor core isotopic inventory is provided in Table 11.1-1.

The parameters used in the calculation of the coolant source terms, including values for the fission product escape rate coefficients, coolant cleanup rate, and demineralizer effectiveness, are listed in Table 11.1-2. The fission product escape rate coefficients are based on average fuel temperature for the typical large light-water reactor. Because NuScale's average fuel temperature is lower than the typical large light-water reactor, these escape rate coefficients are conservative. The quantity of fission products in the fuel pins and the release to the primary coolant are calculated using the following equations:

$$N_{Fp} = \frac{A_{fp}}{\lambda_p} \tag{Eq. 11.1-1}$$

$$P = R_p \times f \times N_{fp} \tag{Eq. 11.1-2}$$

where,

A_{fp} = The activity of parent isotopes in the fuel (Ci),

N_{fp} = The quantity of isotopes in the fuel rods (atoms),

λ_p = The decay constant of the parent isotope (s^{-1}),

R_p = The fission product escape coefficient (s^{-1}),

f = The fraction of fuel rods with defective cladding, and

P = The production rate, also known as release rate of isotopes from failed fuel (atoms/s).

The final equilibrium activity of isotopes that do not have contributions beyond the failed fuel rods (an isotope is not a daughter product) is calculated by the following equation:

$$A_{cp} = \frac{A_{fp} \times f \times R_p}{\lambda_p + \lambda_L + \lambda_U} \tag{Eq. 11.1-3}$$

where,

A_{cp} = The activity of parent isotopes in the primary coolant (Ci),

λ_L = The letdown removal coefficient through degasifier (s^{-1}), and

λ_U = The removal coefficient for purification (s^{-1}).

The activity of the daughter fission products in the primary coolant is dependent upon two factors: 1) the release of daughter fission products already present in the fuel rods and, 2) the generation of daughter fission products in the primary coolant from parent isotopes. The resultant equation used to calculate the activity of the daughter fission products in the primary coolant is,

$$A_{cd} = \frac{A_{fd} \times f \times R_p + A_{cp} \times \lambda_d \times f_p}{\lambda_d + \lambda_L + \lambda_U} \quad \text{Eq. 11.1-4}$$

where,

A_{cd} = The activity of the daughter nuclide in the primary coolant (Ci),

A_{fd} = The activity of the daughter nuclide in the fuel (Ci),

f_p = The branching fraction of the parent nuclide that decays to the daughter isotope, and

λ_d = The decay constant of the daughter isotope (s^{-1}).

11.1.1.2 Activation Products

The tritium in the RCS originating from the fuel is handled differently. Instead of using a core inventory and escape rate coefficient, the permeation of tritium from the fuel to the primary coolant is modeled using the EPRI Tritium Management Model (Reference 11.1-4). The tritium permeation rate of 0.0003 Curies per minute is linearly scaled from 4100 MWt to 160 MWt, and adjusted for a 95 percent unit capacity factor.

In the primary coolant system, neutron activation of various constituents in the water forms activation products. These activation products are independent of the failed fuel fraction. The neutron activation products include N-16, H-3, Ar-41, and C-14.

Nitrogen-16 results from neutron reaction of oxygen-16 in the primary water. Nitrogen-16 has an energetic gamma, but a very short half-life. Because of its short half-life (7.1 seconds), N-16 is not of concern for offsite dose considerations. Table 12.2-5 lists the N-16 concentration at various locations in the primary coolant loop.

Tritium is produced in the primary coolant by neutron activation from several different reactions. The predominant tritium production reactions are high-energy neutron interactions with lithium and boron isotopes. Tritium is also produced by ternary fission of U-235, but only a small fraction of the total tritium produced in the fuel diffuses through the cladding into the coolant. Once the tritium is generated, it will move throughout the plant systems, until ultimately being released via both the liquid and gaseous effluent pathways.

The concentration of tritium in the coolant streams will vary depending on whether the primary coolant letdown to LRWS is recycled to the reactor pool, recycled back to CVCS makeup, or discharged through LRWS. The various tritium concentrations are presented in Table 11.1-8.

Argon-41 results from the neutron activation of Ar-40, which is a natural component of air. The primary reactor coolant is degassed early in the operating cycle to minimize the content of dissolved air. There is a small amount of Ar-41produced as a fission product that may leak into the primary coolant through an assumed defect in the fuel cladding. Argon-41 could also be produced in the space surrounding the reactor vessel, however this potential is significantly reduced due to the reduced air environment resulting from the vacuum maintained in this area during operations. In the absence of significant N-16 in the primary coolant near the steam generators, natural argon can be injected into the primary coolant to improve the sensitivity of primary-to-secondary leak rate calculations. When the injected argon is activated in the reactor core, Ar-41 is produced, which can be used to detect leakage from the primary system. Operators may inject argon into the primary system to maintain a consistent level of Ar-41 in the primary coolant.

Carbon-14 results from the activation of carbon, nitrogen, or oxygen. The predominant reactions in a light water reactor are:

- $^{14}N(n,p)\ ^{14}C$, and
- $^{17}O(n,\alpha)\ ^{14}C$.

The equilibrium amount of C-14 activity in the primary coolant is dependent on the fraction of C-14 retained in the coolant, which is a function of the removal rate by the chemical and volume control system (CVCS) demineralizers, and the letdown removal rate through the degasifier.

It is conservatively assumed that one percent of the C-14 produced is retained in the primary coolant. As an additional conservatism, it is also assumed that the removal of C-14 by the CVCS demineralizers is zero.

The majority of C-14 produced is from the activation of O-17 in the primary coolant. The C-14 primary coolant equilibrium activity is calculated using the following equation:

$$A_{C14} = P_{C14} \times f \qquad \text{Eq. 11.1-5}$$

where,

A_{C14} = The equilibrium activity of C-14 in the primary coolant (Ci),

P_{C14} = The total production rate of C-14 from N-14 and O-17 reactions (Ci/s), and

f = The fraction of C-14 retained in the primary coolant.

11.1.1.3 Corrosion Products

Radioactive corrosion and wear products in the reactor coolant primarily result from neutron activation of non-radioactive corrosion and wear products that are circulated in the primary coolant. The values were developed using guidance from

ANSI/ANS 18.1-1999 (Reference 11.1-2). Corrosion product concentrations are presented for steady state operation, are based on industry operating experience, and are independent of failed fuel. The specific parameters for adjusting the ANSI 18.1 reference values are listed in Table 11.1-3.

11.1.2 Design Basis Secondary Coolant Activity

Activity present in the secondary coolant is assumed to be introduced through primary-to-secondary leakage as well as neutron activation of the secondary coolant. The neutron activation in the secondary coolant was calculated to be negligible due to the small neutron flux at the bottom of the steam generator, which is the closest to the active core. The flux at the bottom of the steam generator is several orders of magnitude less than the average core flux. Therefore, the amount of neutron activation of the secondary coolant is negligible compared to the activity from radionuclides entering the secondary coolant from an assumed design basis primary-to-secondary leak. The design basis secondary coolant activity is determined from an assumed primary-to-secondary leak rate of 75 pounds per day assuming a design basis primary coolant activity concentration (Table 11.1-4). The secondary coolant design basis source term is listed in Table 11.1-5.

11.1.2.1 Steam Generator Leakage

The source of activity in the secondary coolant is assumed to be 75 pounds per day leakage of primary coolant (at standard pressure and temperature). This conservative value of 75 pounds per day for steam generator leakage is consistent with NUREG-0017 and is assumed to exist for each NPM in operation. Assuming there are 12 operating NPMs, this results in a total of 900 pounds per day. This is conservative in comparison to typical large PWRs that assume 75 pounds per day for a 3400 MWth nuclear power plant. In addition, the NuScale helical coil steam generators are of a different design than the typical PWR steam generator design, from which the NUREG-0017 empirical data was derived, and is configured such that the secondary coolant is inside the steam generator tubes and the pressure differential from the primary coolant induces a compressive load on the tubes during operation, which tends to close tube cracks. For the radionuclides that enter the secondary coolant, various removal mechanisms are also incorporated that affect the equilibrium concentration in the secondary coolant. The removal mechanisms include steam leaks in the Turbine Building, condensate polishers, and radioactive decay.

11.1.2.2 Noble Gases in Secondary Coolant Activity Source Term

Noble gases are removed in the secondary coolant by the condenser air removal system. Therefore, only pass-through concentrations of noble gases are assumed to be present in the steam generators. The concentration of noble gases in the secondary coolant is calculated by multiplying the concentration of the noble gas in the primary coolant by the primary-to-secondary leak rate, and dividing by the sum of the secondary flow rate and primary-to-secondary leak rate. The secondary coolant noble gas concentration after passing through the condenser is negligible.

11.1.2.3 Other Isotopes in Secondary Coolant Activity Source Term

The secondary steady state activity is calculated using the following equation (Reference 11.1-3):

$$A_{eq} = P / K \qquad \text{Eq. 11.1-6}$$

where,

A_{eq} = secondary coolant equilibrium activity (Ci),

P = production term for secondary coolant, which is equivalent to the in-leakage rate of primary reactor coolant activity into the secondary coolant (Ci/sec),

K = removal coefficient (s^{-1}) = ($\lambda_d + \lambda_{dm}$),

λ_d = radioactive decay constant (s^{-1}),

λ_{dm} = condensate demineralizer removal coefficient = (F_{dm} x e / M_{sc}),

F_{dm} = flow through condensate demineralizer,

M_{sc} = mass of the secondary coolant, and

e = removal efficiency by demineralizer for the nuclide.

For radioisotopes other than tritium, it is assumed that there are no steam leaks. This conservatively maximizes the calculated radionuclide concentration in the secondary coolant. The condensate system is designed such that 100 percent of the secondary coolant flow passes through the condensate demineralizers. The parameters used to calculate the secondary design basis source term are listed in Table 11.1-2 and Table 11.1-4.

11.1.3 Realistic Reactor Coolant and Secondary Coolant Activity Source Terms

A realistic source term, which is significantly lower than the design basis source term, is used to evaluate normal expected effluent releases as described in Section 11.2 and Section 11.3. The same methodology used to calculate the design basis source terms for the primary and secondary coolant is used to calculate the realistic source terms, with the exceptions that the failed fuel fraction is lower by an order-of-magnitude from the conservatively high value assumed for design purposes, an average value for argon injection concentration is used, and a more realistic primary-to-secondary leak rate is assumed. The NUREG-0017 value of 75 pounds per day is not considered appropriate for a realistic source term for the NuScale design based on the significant differences in thermal power output. Additionally, the NUREG-0017 value is based on empirical data from the 1970's that included typical large PWR U-tube steam generators. The NuScale helical coil steam generator is different in design, as previously discussed in Section 11.1.2.1. A realistic steam generator tube leak rate for the NuScale design was determined by scaling the

NUREG-0017 value of 75 pounds per day based on thermal power. A single NPM produces 160 megawatts (thermal), compared to the typical PWR, which produces approximately 3,400 megawatts (thermal); therefore, the NPM and associated steam generator components are similarly smaller in scale.

Parameters used in the model are included in Table 11.1-2. The realistic source term values for the primary and secondary coolant are provided in Table 11.1-6 and Table 11.1-7.

Details of the release modeling are presented in Section 11.2 and Section 11.3.

The resultant airborne concentrations are presented in Section 12.2.

11.1.4 References

11.1-1 NuScale Power, LLC, “Effluent Release (GALE Replacement) Methodology and Results,” TR-1116-52065, Rev. 1.

11.1-2 American National Standards Institute/American Nuclear Society, "Radioactive Source Term for Normal Operation of Light Water Reactors," ANSI/ANS 18.1-1999, LaGrange Park, IL.

11.1-3 Bevelacqua, J.J., "Basic Health Physics: Problems and Solutions," Wiley-VCH Publishing, Weinheim, Germany, 2004.

11.1-4 Electric Power Research Institute, Inc., "EPRI Tritium Management Model," EPRI #1009903, Palo Alto, CA, 2005.

Table 11.1-1: Maximum Core Isotopic Inventory

Nuclide	Core Inventory (Ci)	Nuclide	Core Inventory (Ci)
Noble Gases		Other Fission Products	
Kr83m	4.7E+05	Y92	4.9E+06
Kr85m	9.7E+05	Y93	5.8E+06
Kr85	1.3E+05	Zr97	7.2E+06
Kr87	1.9E+06	Nb95	6.8E+06
Kr88	2.4E+06	Mo99	8.1E+06
Kr89	3.0E+06	Mo101	7.9E+06
Xe131m	6.0E+04	Tc99m	7.2E+06
Xe133m	2.9E+05	Tc99	2.1E+02
Xe133	9.0E+06	Ru103	8.9E+06
Xe135m	2.1E+06	Ru105	7.2E+06
Xe135	3.5E+06	Ru106	5.7E+06
Xe137	7.9E+06	Rh103m	8.8E+06
Xe138	7.4E+06	Rh105	6.8E+06
Halogens		Rh106	6.1E+06
Br82	2.6E+04	Ag110	2.3E+06
Br83	4.6E+05	Sb124	1.3E+04
Br84	7.7E+05	Sb125	1.1E+05
Br85	9.6E+05	Sb127	5.3E+05
I129	5.4E-01	Sb129	1.5E+06
I130	2.7E+05	Te125m	2.7E+04
I131	4.6E+06	Te127m	8.7E+04
I132	6.6E+06	Te127	5.3E+05
I133	8.9E+06	Te129m	2.5E+05
I134	9.9E+06	Te129	1.5E+06
I135	8.5E+06	Te131m	9.9E+05
Rubidium, Cesium		Te131	3.9E+06
Rb86m	2.0E+03	Te132	6.4E+06
Rb86	1.6E+04	Te133m	4.1E+06
Rb88	2.5E+06	Te134	7.6E+06
Rb89	3.3E+06	Ba137m	1.6E+06
Cs132	3.2E+02	Ba139	7.7E+06
Cs134	2.7E+06	Ba140	7.4E+06
Cs135m	3.2E+04	La140	7.8E+06
Cs136	5.9E+05	La141	7.0E+06
Cs137	1.6E+06	La142	6.6E+06
Cs138	8.1E+06	Ce141	7.0E+06
Other Fission Products		Ce143	6.4E+06
P32	7.2E+02	Ce144	5.9E+06
Co57	5.3E+00	Pr143	6.2E+06
Sr89	3.4E+06	Pr144	5.9E+06
Sr90	1.1E+06	Np239	1.2E+08
Sr91	4.4E+06	C14	2.1E+01
Sr92	4.9E+06	H3	2.0E+04
Y90	1.2E+06		
Y91m	2.6E+06		
Y91	4.5E+06		

Table 11.1-2: Parameters Used to Calculate Coolant Source Terms

Parameter	Value
Reactor core thermal power (MWt)	160 + 3.2 = 163.2 MWt (102%)
Number of fuel assemblies in one core	37
Range of U-235 fuel enrichment (0.5% increments)	1.5% - 5.0%
UO_2 mass in one fuel assembly (with no gadolinium)	282.8 kg
Range of gadolinium within burnable poison rods (0, 4 or 8 rods per fuel assembly)	2% - 8%
U-235 enrichment cutback for gadolinium-containing rods	5% for each 1% Gd_2O_3 (minimum 15% cutback)
Maximum fuel assembly burnup	60,000 MWD/MTU
Failed fuel fractions: Realistic source term Design basis source term	 0.0066% 0.066%
Escape rate coefficients: Xe, Kr gases I, Br, Cs, Rb Mo, Tc, Ag Te Sr, Ba Others	 6.5E-08 s^{-1} 1.3E-08 s^{-1} 2.0E-09 s^{-1} 1.0E-09 s^{-1} 1.0E-11 s^{-1} 1.6E-12 s^{-1}
Average density of reactor coolant	0.724 gram/cm^3
Reactor coolant system mass	1.03E+05 lb
Argon injection concentration: Design Basis Realistic	 0.15 µCi/cm^3 0.10 µCi/cm^3
CVCS flow rate (purification)	22 gpm (180 lb/min)
Secondary coolant mass	5.6E+04 lb
Secondary steam leak rate	80 lb/hr/unit x 12 units =960 lb/hr
Secondary coolant flow rate	5.3E+05 lb/hr
Decontamination factors for CVCS mixed bed demineralizers: Halogens Cs, Rb Other	 100 2 50
Decontamination factors for condensate demineralizers: Halogens Cs, Rb Other	 100 10 100
Primary-to-secondary leak rate: Design Basis Realistic	 75 lb/day/unit x 12 units = 900 lb/day 3.5 lb/day/unit x 12 units = 42 lb/day

Table 11.1-3: Specific Parameters for Crud

Parameter	Symbol	Units	Value
Thermal power	P	MW_{th}	160
Steam flow rate	FS	kg/s	6.7E+01
Weight of water in steam generators	WS	kg	7.5E+02
Weight of water in reactor coolant system	WP	kg	4.7E+04
Letdown flow rate for purification	FD	kg/s	1.3
Letdown flow rate for boron control	FB	kg/s	2.6E-03
Flow through cation demineralizer	FA	kg/s	0
Ratio of condensate demineralizer flow to total steam flow	NC	---	1
Fraction of material removed by cation demineralizer	NA	---	0.9
Fraction of material removed by purification demineralizer	NB	---	0.98
Ratio of concentration in steam to that in steam generator (once-through steam generator)	NS	---	1
Fraction of activity removed by condensate demineralizers	NX	---	0.99

Table 11.1-4: Primary Coolant Design Basis Source Term

Nuclide	Primary Coolant Concentrations (Ci/g)	Nuclide	Primary Coolant Concentrations (Ci/g)
Noble Gases		Other FPs (continued)	
Kr83m	4.9E-09	Mo101	2.7E-10
Kr85m	2.1E-08	Tc99m	6.7E-09
Kr85	6.1E-06	Tc99	2.1E-13
Kr87	1.1E-08	Ru103	6.9E-12
Kr88	3.3E-08	Ru105	2.3E-12
Kr89	7.5E-10	Ru106	4.5E-12
Xe131m	8.0E-08	Rh103m	6.8E-12
Xe133m	7.3E-08	Rh105	4.8E-12
Xe133	5.4E-06	Rh106	4.5E-12
Xe135m	7.0E-09	Ag110	5.2E-12
Xe135	1.8E-07	Sb124	1.0E-14
Xe137	2.4E-09	Sb125	8.9E-14
Xe138	8.3E-09	Sb127	3.9E-13
Halogens		Sb129	4.8E-13
Br82	1.4E-10	Te125m	1.3E-11
Br83	7.8E-10	Te127m	4.2E-11
Br84	3.6E-10	Te127	1.7E-10
Br85	4.4E-11	Te129m	1.2E-10
I129	3.4E-15	Te129	1.7E-10
I130	1.1E-09	Te131m	4.0E-10
I131	2.8E-08	Te131	1.9E-10
I132	1.3E-08	Te132	2.9E-09
I133	4.3E-08	Te133m	2.5E-10
I134	7.6E-09	Te134	3.5E-10
I135	2.7E-08	Ba137m	1.9E-08
Rubidium, Cesium		Ba139	6.5E-12
Rb86m	3.2E-14	Ba140	3.5E-11
Rb86	1.9E-10	La140	1.0E-11
Rb88	3.3E-08	La141	2.0E-12
Rb89	1.5E-09	La142	9.6E-13
Cs132	3.7E-12	Ce141	5.5E-12
Cs134	3.3E-08	Ce143	4.1E-12
Cs135m	2.5E-11	Ce144	4.6E-12
Cs136	7.1E-09	Pr143	4.9E-12
Cs137	2.0E-08	Pr144	4.5E-12
Cs138	1.2E-08	Np239	8.7E-11
Other FPs		Corrosion/Activation Products - Crud	
P32	5.5E-16	Na24	9.1E-09
Co57	4.1E-18	Cr51	5.2E-10
Sr89	2.5E-11	Mn54	2.7E-10
Sr90	5.5E-12	Fe55	2.0E-10
Sr91	1.3E-11	Fe59	5.0E-11
Sr92	6.8E-12	Co58	7.7E-10

Table 11.1-4: Primary Coolant Design Basis Source Term (Continued)

Nuclide	Primary Coolant Concentrations (Ci/g)	Nuclide	Primary Coolant Concentrations (Ci/g)
Y90	1.3E-12	Co60	8.8E-11
Y91m	6.8E-12	Ni63	4.4E-11
Y91	3.6E-12	Zn65	8.5E-11
Y92	5.8E-12	Zr95	6.5E-11
Y93	2.7E-12	Ag110m	2.2E-10
Zr97	4.0E-12	W187	4.7E-10
Nb95	5.8E-12	Water Activation Products	
Mo99	7.2E-09	C14	2.2E-10
		Ar41	2.1E-07

Table 11.1-5: Secondary Coolant Design Basis Source Term

Nuclide	Secondary Coolant Concentrations (Ci/g)	Nuclide	Secondary Coolant Concentrations (Ci/g)
Noble Gases		Other FPs (continued)	
Kr83m	2.9E-14	Mo101	1.2E-15
Kr85m	1.2E-13	Tc99m	3.9E-14
Kr85	3.6E-11	Tc99	1.2E-18
Kr87	6.6E-14	Ru103	4.1E-17
Kr88	1.9E-13	Ru105	1.3E-17
Kr89	4.4E-15	Ru106	2.6E-17
Xe131m	4.7E-13	Rh103m	3.7E-17
Xe133m	4.3E-13	Rh105	2.8E-17
Xe133	3.2E-11	Rh106	2.7E-18
Xe135m	4.1E-14	Ag110	2.6E-18
Xe135	1.1E-12	Sb124	6.0E-20
Xe137	1.4E-14	Sb125	5.3E-19
Xe138	4.8E-14	Sb127	2.3E-18
Halogens		Sb129	2.8E-18
Br82	8.1E-16	Te125m	7.8E-17
Br83	4.5E-15	Te127m	2.5E-16
Br84	1.9E-15	Te127	9.9E-16
Br85	1.0E-16	Te129m	7.2E-16
I129	2.0E-20	Te129	9.6E-16
I130	6.5E-15	Te131m	2.3E-15
I131	1.7E-13	Te131	9.8E-16
I132	7.4E-14	Te132	1.7E-14
I133	2.5E-13	Te133m	1.4E-15
I134	4.2E-14	Te134	1.9E-15
I135	1.6E-13	Ba137m	4.2E-14
Rubidium, Cesium		Ba139	3.7E-17
Rb86m	3.7E-20	Ba140	2.1E-16
Rb86	1.3E-15	La140	6.1E-17
Rb88	1.7E-13	La141	1.2E-17
Rb89	7.5E-15	La142	5.5E-18
Cs132	2.4E-17	Ce141	3.2E-17
Cs134	2.2E-13	Ce143	2.4E-17
Cs135m	1.5E-16	Ce144	2.7E-17
Cs136	4.6E-14	Pr143	2.9E-17
Cs137	1.3E-13	Pr144	2.2E-17
Cs138	6.8E-14	Np239	5.1E-16
Other FPs		Corrosion/Activation Products - Crud	
P32	3.3E-21	Na24	5.4E-14
Co57	2.4E-23	Cr51	3.1E-15
Sr89	1.5E-16	Mn54	1.6E-15
Sr90	3.3E-17	Fe55	1.2E-15
Sr91	7.5E-17	Fe59	3.0E-16
Sr92	3.9E-17	Co58	4.6E-15

Table 11.1-5: Secondary Coolant Design Basis Source Term (Continued)

Nuclide	Secondary Coolant Concentrations (Ci/g)	Nuclide	Secondary Coolant Concentrations (Ci/g)
Y90	7.9E-18	Co60	5.2E-16
Y91m	3.7E-17	Ni63	2.6E-16
Y91	2.1E-17	Zn65	5.0E-16
Y92	3.4E-17	Zr95	3.9E-16
Y93	1.6E-17	Ag110m	1.3E-15
Zr97	2.4E-17	W187	2.7E-15
Nb95	3.4E-17	Water Activation Products	
Mo99	4.3E-14	C14	1.3E-15
		Ar41	1.2E-12

Table 11.1-6: Primary Coolant Realistic Source Term

Nuclide	Primary Coolant Concentrations (Ci/g)	Nuclide	Primary Coolant Concentrations (Ci/g)
Noble Gases		Other FPs (continued)	
Kr83m	4.9E-10	Mo101	2.7E-11
Kr85m	2.1E-09	Tc99m	6.7E-10
Kr85	1.8E-07	Tc99	2.1E-14
Kr87	1.1E-09	Ru103	6.9E-13
Kr88	3.3E-09	Ru105	2.3E-13
Kr89	7.5E-11	Ru106	4.5E-13
Xe131m	7.4E-09	Rh103m	6.8E-13
Xe133m	7.2E-09	Rh105	4.8E-13
Xe133	5.3E-07	Rh106	4.5E-13
Xe135m	7.0E-10	Ag110[1]	3.2E-12
Xe135	1.8E-08	Sb124	1.0E-15
Xe137	2.4E-10	Sb125	8.9E-15
Xe138	8.3E-10	Sb127	3.9E-14
Halogens		Sb129	4.8E-14
Br82	1.4E-11	Te125m	1.3E-12
Br83	7.8E-11	Te127m	4.2E-12
Br84	3.6E-11	Te127	1.7E-11
Br85	4.4E-12	Te129m	1.2E-11
I129	3.4E-16	Te129	1.7E-11
I130	1.1E-10	Te131m	4.0E-11
I131	2.8E-09	Te131	1.9E-11
I132	1.3E-09	Te132	2.9E-10
I133	4.3E-09	Te133m	2.5E-11
I134	7.6E-10	Te134	3.5E-11
I135	2.7E-09	Ba137m	1.9E-09
Rubidium, Cesium		Ba139	6.5E-13
Rb86m	3.2E-15	Ba140	3.5E-12
Rb86	1.9E-11	La140	1.0E-12
Rb88	3.3E-09	La141	2.0E-13
Rb89	1.5E-10	La142	9.6E-14
Cs132	3.7E-13	Ce141	5.5E-13
Cs134	3.3E-09	Ce143	4.1E-13
Cs135m	2.5E-12	Ce144	4.6E-13
Cs136	7.0E-10	Pr143	4.9E-13
Cs137	2.0E-09	Pr144	4.5E-13
Cs138	1.2E-09	Np239	8.7E-12
Other FPs		Corrosion/Activation Products - Crud	
P32	5.5E-17	Na24	9.1E-09
Co57	4.1E-19	Cr51	5.2E-10
Sr89	2.5E-12	Mn54	2.7E-10
Sr90	5.5E-13	Fe55	2.0E-10
Sr91	1.3E-12	Fe59	5.0E-11
Sr92	6.8E-13	Co58	7.7E-10

Table 11.1-6: Primary Coolant Realistic Source Term (Continued)

Nuclide	Primary Coolant Concentrations (Ci/g)	Nuclide	Primary Coolant Concentrations (Ci/g)
Y90	1.3E-13	Co60	8.8E-11
Y91m	6.8E-13	Ni63	4.4E-11
Y91	3.6E-13	Zn65	8.5E-11
Y92	5.8E-13	Zr95	6.5E-11
Y93	2.7E-13	Ag110m	2.2E-10
Zr97	4.0E-13	W187	4.7E-10
Nb95[1]	1.0E-12	Water Activation Products	
Mo99	7.2E-10	C14	2.2E-10
		Ar41	1.4E-07

Note 1: Ag110 and Nb95 are both a fission product and a crud daughter product.

Table 11.1-7: Secondary Coolant Realistic Source Term

Nuclide	Secondary Coolant Concentrations (Ci/g)	Nuclide	Secondary Coolant Concentrations (Ci/g)
Noble Gases		Other FPs (continued)	
Kr83m	1.4E-16	Mo101	5.8E-18
Kr85m	5.7E-16	Tc99m	1.8E-16
Kr85	5.0E-14	Tc99	5.7E-21
Kr87	3.1E-16	Ru103	1.9E-19
Kr88	9.1E-16	Ru105	6.2E-20
Kr89	2.1E-17	Ru106	1.2E-19
Xe131m	2.1E-15	Rh103m	1.8E-19
Xe133m	2.0E-15	Rh105	1.3E-19
Xe133	1.5E-13	Rh106	1.3E-20
Xe135m	1.9E-16	Ag110	7.6E-20
Xe135	5.1E-15	Sb124	2.8E-22
Xe137	6.7E-17	Sb125	2.5E-21
Xe138	2.3E-16	Sb127	1.1E-20
Halogens		Sb129	1.3E-20
Br82	3.8E-18	Te125m	3.7E-19
Br83	2.1E-17	Te127m	1.2E-18
Br84	8.9E-18	Te127	4.6E-18
Br85	4.9E-19	Te129m	3.4E-18
I129	9.4E-23	Te129	4.5E-18
I130	3.0E-17	Te131m	1.1E-17
I131	7.9E-16	Te131	4.6E-18
I132	3.5E-16	Te132	8.0E-17
I133	1.2E-15	Te133m	6.4E-18
I134	2.0E-16	Te134	8.8E-18
I135	7.4E-16	Ba137m	2.0E-16
Rubidium, Cesium		Ba139	1.7E-19
Rb86m	1.7E-22	Ba140	9.9E-19
Rb86	5.9E-18	La140	2.9E-19
Rb88	7.9E-16	La141	5.5E-20
Rb89	3.5E-17	La142	2.6E-20
Cs132	1.1E-19	Ce141	1.5E-19
Cs134	1.0E-15	Ce143	1.2E-19
Cs135m	7.1E-19	Ce144	1.3E-19
Cs136	2.2E-16	Pr143	1.4E-19
Cs137	6.2E-16	Pr144	1.0E-19
Cs138	3.2E-16	Np239	2.4E-18
Other FPs		Corrosion/Activation Products - Crud	
P32	1.5E-23	Na24	2.5E-15
Co57	1.1E-25	Cr51	1.4E-16
Sr89	6.8E-19	Mn54	7.5E-17
Sr90	1.5E-19	Fe55	5.6E-17
Sr91	3.5E-19	Fe59	1.4E-17
Sr92	1.9E-19	Co58	2.1E-16

Table 11.1-7: Secondary Coolant Realistic Source Term (Continued)

Nuclide	Secondary Coolant Concentrations (Ci/g)	Nuclide	Secondary Coolant Concentrations (Ci/g)
Y90	3.7E-20	Co60	2.5E-17
Y91m	1.8E-19	Ni63	1.2E-17
Y91	9.9E-20	Zn65	2.4E-17
Y92	1.6E-19	Zr95	1.8E-17
Y93	7.5E-20	Ag110m	6.1E-17
Zr97	1.1E-19	W187	1.3E-16
Nb95	2.9E-19	Water Activation Products	
Mo99	2.0E-16	C14	6.2E-17
		Ar41	3.8E-14

Table 11.1-8: Tritium Concentrations versus Primary Coolant Recycling Modes

Recycle Mode	Primary Coolant Average Concentration (Ci/g)	RCS Letdown / CVCS Outlet (Ci/g)	Realistic Secondary Coolant Concentration (Ci/g)	Design Basis Secondary Coolant Concentration (Ci/g)
No recycle (discharge)	9.6E-07	7.2E-07	1.8E-09	---
Recycle to reactor pool makeup	9.9E-07	7.4E-07	1.8E-09	---
Recycle back to CVCS makeup	2.8E-06	2.8E-06	---	5.2E-09

Note: The maximum calculated peak primary coolant tritium concentration is 3.5μCi/g.

11.2 Liquid Waste Management System

For the NuScale design, the liquid waste management system (LWMS) is called the liquid radioactive waste system. The liquid radioactive waste system (LRWS) is a system that is designed to collect, hold, and process liquid radioactive waste generated from normal operations and anticipated operational occurrences (AOOs). After processing and satisfactory sampling, liquids may be recycled or discharged. The LRWS is operated in a batch mode by an operator located in the waste management control room (WMCR).

The LRWS receives radioactive fluids from the chemical and volume control system (CVCS), the solid radioactive waste system (SRWS), the containment evacuation system (CES), the reactor component cooling water system (RCCWS), and the radioactive waste drain system (RWDS). The LRWS components are located in the Reactor Building (RXB) and in the Radioactive Waste Building (RWB).

The RXB is a Seismic Category I structure and the RWB is a Seismic Category II structure that also meets the requirements of Regulatory Guide (RG) 1.143 for a RW-IIa safety classification as discussed in Section 3.2.1.

11.2.1 Design Bases

The LRWS has no safety-related function. The LRWS is not credited for mitigation of design basis accidents and has no safe shutdown functions. General Design Criteria (GDC) 2, 3, 60, and 61 were considered in the design of the LRWS.

Consistent with GDC 2, the LRWS is designed using the guidance of RG 1.143. Consistent with GDC 3, the generation of explosive gas mixtures and exothermic reactions is avoided. Consistent with GDC 60 as it relates to the LRWS, releases of radioactive materials to the environment are controlled. Consistent with GDC 61 as it relates to the LRWS, the design ensures adequate safety under normal and postulated accident conditions.

The LRWS is designed to comply with the as low as reasonably achievable (ALARA) philosophy of 10 CFR 20.1101(b) and the dose limits of 10 CFR 20.1301, 10 CFR 20.1302, and 10 CFR 50 Appendix I ALARA design objectives, including the effluent concentration limits of 10 CFR 20 Appendix B, Table 2 and 40 CFR 190 as implemented under 10 CFR 20.1301(e).

Consistent with 10 CFR 20.1406, the design of the LRWS includes provisions to minimize the contamination of the facility and the environment, facilitate eventual decommissioning, and minimize the generation of radioactive waste.

COL Item 11.2-1: A COL applicant that references the NuScale Power Plant design certification will ensure mobile equipment used and connected to plant systems is in accordance with American National Standards Institute / American Nuclear Society (ANSI/ANS)-40.37, Regulatory Guide (RG) 1.143, 10 CFR 20.1406, NRC IE Bulletin 80-10 and 10 CFR 50.34a.

11.2.2 System Description

The LRWS includes tanks, pumps, filters, and ion exchangers to receive, store, process, and monitor liquid radioactive waste to be recycled or released to the environment in accordance with regulations.

The inputs to the LRWS are segregated as follows:

- low conductivity: reactor coolant grade waste water collected from the primary coolant system letdown through the CVCS and RWDS equipment drains
- high conductivity: RWDS floor drains, SRWS decant water, RCCWS, and other liquid wastes with potentially high dissolved and suspended solids content collected from various drains and sumps
- detergent waste: detergent wastes from hand decontamination processes and personnel decontamination showers
- chemical waste: collected by the RWDS and not introduced into the LRWS (Section 9.3.3)

The liquid wastes from the various sources are temporarily stored in collection tanks located in the RWB. System equipment and components are located in stainless-steel-lined, shielded cubicles as necessary to contain leaks and for radiation shielding. Other equipment areas, located outside of steel-lined cubicles, have concrete surfaces that are sealed with a qualified coating meeting the requirements of RG 1.54. The system operates on a batch basis, using skid-based processing equipment that includes filters, ion exchangers, and reverse osmosis components. Subsequent to processing, the liquid is routed to sample tanks to monitor the quality of the liquid prior to recycling or release. If the water quality is not acceptable, the water is returned to a collection tank for further treatment.

The LRWS includes cross-tie connections to allow for processing options depending on the waste characteristics and redundancy of equipment.

The LRWS is designed with sufficient capacity to process liquid wastes during periods of equipment maintenance or failures and during periods of abnormal waste generation. To meet these processing demands, interconnections between LRWS components, redundant equipment, skid-based equipment, liquid holdup storage, and treatment capacity are provided in the design.

The LRWS is designed to control leakage and facilitate access, operation, inspection, testing, and maintenance to maintain radiation exposures to operating and maintenance personnel as low as is reasonably achievable and to minimize contamination of the facility.

The LRWS design includes the following maintenance considerations:

- location of redundant permanent plant equipment in separate shielded cubicles
- clean-in-place (CIP) provisions to reduce the radiation source term prior to maintenance

- redundant components allow uninterrupted waste operation and flexibility in maintenance scheduling

11.2.2.1 System Operation

The LRWS processing is manually operated in a batch mode by an operator selecting appropriate tank, valve, pump, and processing line-ups for each batch. If power is lost, the radioactive waste is held in tanks, pipes, and vessels until power can be restored.

The principal functions of the LRWS are:

- collecting and processing radioactive fluids for recycle or release in accordance with release limits.
- providing sufficient capacity for the accumulation and processing of liquid radioactive wastes produced in the plant during normal operation and during AOOs (including shutdown, refueling, and maintenance) for up to 12 power modules for seven days without affecting plant availability.
- providing a high level of system configuration flexibility to ensure plant availability and continuous waste receiving capability.
- providing cross connections between installed plant collection, skid-based processing systems, and sample tanks.

11.2.2.1.1 Liquid Radioactive Waste System Input Streams

Separate collection tanks are provided for high-conductivity wastes (HCW), low-conductivity wastes (LCW), and detergent wastes. Oily waste is processed in oil separators before being sent to the high-conductivity waste collection tank. The oil is separated from the waste stream and collected in drums and sent to the SRWS for eventual shipment offsite. Chemical wastes are collected as part of the RWDS. Mixed wastes are collected locally in drums and sent offsite. Table 11.2-3 provides a tabulation of expected values for the inputs. Figure 11.2-1a through Figure 11.2-1j provide flow diagrams of the LRWS.

<u>Chemical and Volume Control System</u>

There are two CVCS streams that are processed by the LRWS in the skid-based degasifier: (1) letdown during normal operation and reactor heatup and (2) pressurizer vent before and during reactor shutdown. Letdown during normal operations is conducted to dilute the primary coolant boron concentration to account for fuel burnup. This dilution is performed in discrete activities, using letdown flow for a short duration (few minutes) multiple times per day. During reactor heatup, excess coolant inventory due to thermal expansion is removed from the primary system by the CVCS letdown line to LRWS through the degasifier.

The other CVCS stream processed by the LRWS through the degasifier is the pressurizer vent flow path. The pressurizer vent line is sized assuming that three or fewer vents will be opened simultaneously. Because the flow stream properties from the letdown are significantly different than the pressurizer vent, two different headers are provided, appropriately sized for the expected flow conditions.

The degasifier is sized for maximum letdown from a reactor heatup and the simultaneous deboration letdown from three other reactors. Each of the two degasifiers has a liquid ring vacuum pump and a vent condenser. Each degasifier also has a liquid transfer pump with full recovery of sealed liquid.

Floor and Equipment Drains

As described in Section 9.3.3, building floor drains, equipment drains, and pool leakage are collected by the RWDS and routed to the LRWS if determined to be radioactive.

Containment Evacuation System

Containment evacuation liquid waste includes liquid leakage from the reactor coolant system, the RCCWS, the main steam system, the feedwater system, the CVCS, and potentially the reactor pool and in-leakage to the containment evacuation system itself.

RCCW Drain Tank

The reactor component cooling water drain tank collects water drained from various RCCWS components. The RCCWS is normally returned to the RCCWS as makeup water. If the RCCWS water becomes cross-contaminated, there are design provisions to handle this situation, as described in Section 9.2.2.

Solid Radioactive Waste System Dewatering Skid

Contaminated water from the dewatering process in the SRWS is directed to the LRWS for processing.

Clean-in-Place Flushing Water

The LRWS includes a CIP system that provides clean demineralized water for flushing CVCS and LRWS resin sluice lines to the spent resin storage tank or pool cleanup system (PCUS) resin sluice lines to the phase separator tank. Use of clean water keeps the spent resin pipe line as clean as possible for ALARA purposes. Connections are provided to allow use of the CIP system prior to maintenance (e.g. flushing tanks and transfer lines).

Boric Addition System

As a contingency, in the case of a chemically contaminated batch of boric acid, the boron addition system can send the contaminated batch to the LRWS.

Pool Surge Control System Storage Tank Catch Basin

Water from the pool surge control system (PSCS) storage tank is collected in the drain sump of the PSCS storage tank catch basin and sent to the LRWS for processing, if found to be radioactive.

Recycled LRWS Sample Tank contents

If the LRWS sample tank results are out of specification, the contents can be recycled for further processing.

Detergent Wastes

Waste water that enters the detergent waste collection tank comes through the RWDS from personnel decontamination showers and small component cleaning in the decontamination sink.

11.2.2.1.2 Low Conductivity Waste System

Degasifier

There are two skid-based degasifiers located in the RXB. One degasifier is normally used for operation while the other degasifier is in standby. The degasifier is designed to strip the reactor coolant letdown or pressurizer vapor space of non-condensable gases. The stripped gases are drawn through a vent condenser to remove moisture and then are transferred to the gaseous radioactive waste system while the liquid in the degasifier is pumped to a LCW collection tank by a degasifier liquid transfer pump.

The non-condensable gases are primarily hydrogen with trace amounts of noble gases. The removal of the non-condensable gases from the degasifier is facilitated by a liquid ring vacuum pump, which transfers the gases to the gaseous radioactive waste system. The removal of the remaining liquid is accomplished using degasifier liquid transfer pumps, which are air-operated, double-diaphragm pumps.

Backpressure regulators on the letdown and pressurizer headers control the pressure at the degasifier nozzle to ensure consistent performance with either intermittent or continuous design flow and pressure. The degasifier vacuum pump operates to maintain a constant degasifier pressure. The degasifier portion of the LRWS is shown in Figure 11.2-1a.

Low-Conductivity Waste Collection Tanks

There are two LCW collection tanks located in the RWB. The use of two collection tanks allows one tank to receive waste input and hold it for a time while the other tank is in standby mode or is being recirculated, sampled, or emptied for processing. Each tank receives wastes from the following sources:

- degasified reactor coolant (normal letdown and letdown during heatup)
- RXB and RWB equipment drains from RWDS
- recycled water from LCW or HCW sample tanks
- spent fuel pool cooling system
- out-of-specification boric acid batch from boron addition system

The LCW collection tank design parameters are listed in Table 11.2-1.

The LCW collection tanks are designed with the discharge and drain line at the lowest point of the tank. The on/off bottom valve is a minimum distance from the tank bottom to optimize drainage and cleaning capability. The pump recirculation line is equipped with an in-tank mixing eductor to keep solids in suspension for sampling and to facilitate transfer of tank contents without leaving sludge behind. The LCW portion of the LRWS is shown in Figure 11.2-1b, Figure 11.2-1e and Figure 11.2-1g.

Low-Conductivity Waste Collection Tank Transfer Pump

The two LCW collection tank transfer pumps are air-operated, double-diaphragm pumps designed to deliver LCW wastes to the processing equipment skid. Both LCW collection tank transfer pumps are variable speed pumps, which provide the flexibility to accept different waste inputs and transfer rates. The transfer pumps are located in the RWB. Each LCW collection tank has a dedicated transfer pump and each is cross-connected to provide flexibility and redundancy for transferring waste content from either collection tank.

Low-Conductivity Waste Processing Equipment Skid

Space in the RWB and process connections has been allocated for processing equipment. The LCW processing equipment skid typically includes

- granulated activated carbon - The granulated activated carbon filter consists of granulated activated carbon contained in a stainless steel vessel. The filter is designed to remove suspended solids and total organic carbon from the influent stream. When needed, the used granulated carbon is removed from the granulated activated carbon vessel by sluicing to a high-integrity container.
- tubular ultrafiltration (TUF) recirculation tank - The TUF recirculation tank provides a means to concentrate solids from the TUF membrane reject and serves as a settling tank before it is pumped to the drum dryer.
- tubular ultrafiltration - The TUF unit is located upstream of the reverse osmosis unit and removes suspended solids and other metal complexes. A feed pump provides the necessary motive force to push the stream through the TUF membranes. A permeate pump forwards the permeate to the reverse osmosis (RO) unit. The TUF reject is sent back to the TUF recirculation tank.
- reverse osmosis - The RO skid is a two-pass RO unit that receives the filtered waste stream from the TUF unit and forwards the RO permeate to the demineralizers. The RO reject from the second pass is returned to the front of the first pass. The RO reject from the first pass is sent back to the TUF recirculation tank.
- demineralizers - There are five demineralizers on the LCW processing skid: cation, anion, mixed, antimony specific, and cesium specific. Flexibility allows a combination of demineralizers to be used.

Low-Conductivity Waste Sample Tank

There are two cross-tied LCW sample tanks located in the RWB. The sample tanks receive liquid wastes from either the HCW processing skid or the LCW processing skid. Prior to discharging or recycling, the content of the tank is recirculated and sampled. The recirculation line is equipped with an in-tank eductor to minimize the recirculation time prior to sampling. The tank design parameters are listed in Table 11.2-1.

Low-Conductivity Waste Sample Tank Transfer Pump

The two LCW sample tank transfer pumps are air-operated, double-diaphragm pumps designed to deliver processed and sampled water to the following destinations:

- CVCS makeup
- spent fuel pool cooling system makeup
- discharge to the utility water discharge basin

If the LCW sample tank sample results do not meet specified requirements, the LCW waste can be returned to the HCW or LCW collection tanks for reprocessing. The pumps are located in the RWB in a room adjacent to their associated sample tank. The transfer pump of each sample tank is cross-connected to provide flexibility and redundancy for transferring waste content from either sample tank.

11.2.2.1.3 High-Conductivity Waste System

Oil Separators

There are two oil separators to segregate the oily portion of liquid waste inputs before transferring the non-oily portion to the HCW collection tanks. The oily waste collected in the oil separators is accumulated and placed in drums and stored in the RWB until shipped offsite. One of the oil separators is normally in use to receive waste while the other is in standby mode. The HCW portion of LRWS is illustrated in Figure 11.2-1c, Figure 11.2-1f, and Figure 11.2-1h.

High-Conductivity Waste Collection Tanks

There are two HCW collection tanks located in the RWB. The use of two collection tanks allows one tank to receive waste input while the other is in standby mode or is being recirculated, sampled, or emptied. The tank receives wastes from the following sources:

- RXB floor drains
- RWB floor drains
- SRWS dewatering skid
- ANB decontamination room sumps
- north and south chemical waste collection tank sumps

- SRWS spent resin storage tank and phase separator
- pool surge control system
- HCW and LCW sample tanks
- detergent waste collection tank
- RXB RCCW drain tank
- RXB chemical drain tank

The HCW process and equipment are redundant to the LCW system and are designed to allow flexibility in processing sequences. When the collection tank that is receiving waste is full, waste collection is automatically transferred to the other collection tank. After samples indicate satisfactory results, the contents of the tank are pumped to a processing skid.

<u>High-Conductivity Waste Collection Tank Transfer Pump</u>

The two HCW collection transfer pumps are air-operated, double-diaphragm pumps designed to deliver HCW wastes to the processing equipment skid. Each HCW collection tank has an associated transfer pump and each are cross-connected to the other HCW collection tank to provide operational flexibility. If operational considerations warrant, the liquid waste in the HCW collection tank can be transferred to the LCW processing skid.

<u>High-Conductivity Waste Processing Equipment Skid</u>

The typical HCW processing equipment skid is similar to the LCW processing equipment skid, except for the demineralizers. The LCW demineralizers are shared, and if operational conditions warrant, can be used to treat the waste stream from HCW. The HCW processing equipment skid typically includes:

- granulated activated carbon
- TUF recirculation tank
- TUF unit
- RO unit

Additional details on the skid components can be found in the LCW processing equipment skid section.

<u>High-Conductivity Waste Sample Tank</u>

There are two cross-tied HCW sample tanks located in the RWB. The sample tanks receive liquid wastes from either the HCW processing skid or the LCW processing skid. Prior to discharging or recycling, the content of the tank is recirculated and sampled. The recirculation line is equipped with an in-tank eductor to minimize the recirculation time prior to sampling. The tank design parameters are listed in Table 11.2-1.

High-Conductivity Waste Sample Tank Transfer Pump

The two HCW sample tank transfer pumps are air-operated, double-diaphragm pumps designed to deliver processed and sampled water, meeting specified requirements, to the LCW sample tank transfer pump discharge. From there, the water can be directed to the following destinations:

- CVCS makeup
- spent fuel pool cooling system makeup
- discharge header to the utility water discharge basin

If the HCW sample tank sample results do not meet specified requirements, the HCW waste can be returned to the HCW or LCW collection tanks for re-processing. The pumps are located in the RWB in a room adjacent to their associated sample tank. The transfer pump of each sample tank is cross-connected to provide flexibility and redundancy for transferring waste content from either sample tank.

11.2.2.1.4 Chemical Waste

The chemical waste is collected in the RWDS and is described in Section 9.3.3. The chemical waste volume is small and, if contaminated, is forwarded to the HCW collection tank. The RCCWS drains are collected separately in RWDS to prevent the introduction of nitrite into LRWS demineralizer resins to avoid the potential for exothermic reactions. The RCCWS drains are collected and either returned to the RCCWS as makeup, routed to the drum dryer skid, or discharged. Administrative controls prohibit the mixing of nitrite containing fluids with demineralizer resins (COL Item 9.2-1).

11.2.2.1.5 Detergent Waste System

Detergent waste from personnel decontamination showers and small component decontamination sinks is collected in a dedicated collection tank and sampled, then discharged through a cartridge filter if the sample results indicate the specified requirements are satisfied. If the detergent sample result is not within discharge limits, the contents can be sent to the HCW collection tank or to the drum dryer skid. There is a single train of detergent waste due to the low volume of waste generation expected. The detergent waste system is located in the RWB and includes a collection tank, a transfer pump and a filter. The detergent waste portion of the LRWS is illustrated in Figure 11.2-1i.

Detergent Waste Collection Tank

The detergent waste collection tank is a stainless-steel, vertical, cylindrical tank with a conical bottom located in the RWB.

Detergent Waste Collection Tank Transfer Pump

The detergent waste collection tank transfer pump is an air-operated, double-diaphragm pump designed to deliver detergent waste to the HCW collection tank, the drum dryer skid, or the discharge header through the suction side of the LCW sample tank transfer pump.

Detergent Waste Drain Filter

The detergent waste drain filter is a cartridge filter located downstream of the transfer pump.

11.2.2.1.6 System Support Equipment

Demineralized Water Break Tank

There is one demineralized water break tank in the RWB that supplies the CIP skids and provide a backup source of sluice water to transfer spent resins from CVCS and PCUS demineralizers (Figure 11.2-1d).

Clean-in-Place Skid

The LRWS design includes two CIP skids to clean and flush LRWS components and piping prior to maintenance operations to reduce personnel exposures. The CIP skids are located in the RWB, one on each of the two building levels. The demineralized water system supplies water to the CIP skids through a demineralized water break tank and two transfer pumps.

Demineralized Water Break Tank Transfer Pump

There are two transfer pumps associated with the demineralized water break tank. Each of these two pumps is a centrifugal pump with variable frequency drives.

Neutralization Skid

The neutralization skid is a vendor supplied package that provides pH adjustment capability to the LCW and HCW processing equipment skids. A typical neutralization skid consists of an influent equalization tank, chemical metering pumps, treatment tank, and a main control panel.

Drum Dryer

The drum dryer skid is located in the RWB and is used to concentrate TUF reject and detergent waste (Figure 11.2-1j). A 55-gallon drum is heated while a vacuum pump evacuates the evaporated liquid through a condenser, then directs the liquid discharge to RWDS and the gaseous discharge to the Radioactive Waste Building HVAC system. When the contents of the drum are dried, the 55-gallon drum is prepared for storage or shipping.

11.2.2.2 System Component Details

11.2.2.2.1 Tanks

Tanks that may contain a significant quantity of radioactive liquids are located in individual stainless-steel-lined, shielded cubicles. The lined volume is sufficient to retain a spill of the tank volume to prevent an uncontrolled release. The cubicle floors are sloped to a RWDS sump. Tank overflows are routed to the sump which is served by the RWDS (Section 9.3.3).

Vents are provided on atmospheric tanks and are designed for maximum influent and effluent flow rates to prevent tank overpressure or vacuum conditions from occurring under maximum filling and draining conditions. Tanks that have a potential for a buildup of solids have bottoms that are conical with a central drain connection.

The LRWS tanks use inline automatic samplers. These samplers are located in accessible areas and do not require sample drain lines. This allows automatic or manual sampling to be done with reduced exposure to personnel and reduced waste generation.

Liquid radioactive waste system tanks are indoors.

11.2.2.2.2 Pumps

The LCW and HCW collection and sample tank transfer pumps are located in separate shielded cubicles. Cubicle wall thicknesses and ventilation airflows are designed to reduce personnel exposures. The LRWS transfer pumps have catch pans to contain pump leakages. The components inside the pump cubicles are remotely operated for ALARA purposes.

The CIP skids provide flushing and cleaning capability of contaminated pumps to reduce radiation levels before entering component cubicles for maintenance purposes. Wetted surfaces of pumps are stainless steel. Pumps are of a low-leakage type appropriate for the service such as double-diaphragm or mechanically sealed pumps.

A list of LRWS pumps, with design parameters, is provided in Table 11.2-1.

11.2.2.2.3 Valves and Piping

Double isolation valves, backflow check valves, or air breaks are provided at LRWS connections with non-radioactive systems to prevent cross contamination. LRWS valves are either full-ported ball valves or diaphragm valves.

Pressure retaining piping is seamless stainless steel and uses large radius bends and butt welds to reduce crud traps and leaks. The use of flanged connections is limited to certain components, such as pumps. Process piping containing contaminated solids are sized with sufficient velocities and sloped to facilitate flow and prevent the settling of solids.

11.2.2.3 Off-Normal Operations

The LRWS is designed and LRWS components are sized to accommodate anticipated surge volumes of liquid waste associated with normal operations and refueling outages. Flexibility and redundancy are also incorporated to enable operators to successfully manage off-normal operational conditions.

The system is designed to accommodate unusually high influents from peak reactor coolant system letdown from three closely spaced reactor shutdowns for refueling in addition to normal liquid waste flow rates due to the operation of the other reactors.

The LRWS provides the necessary remote operation capability and indications to operate the system safely during off-normal conditions that would make LRWS equipment inaccessible to operating personnel.

The LRWS is designed to be tolerant of failures and abnormal conditions as summarized in Table 11.2-2.

Section 11.2.5 contains additional information regarding automatic control features as a result of upset conditions.

11.2.2.4 Safety Evaluation

The LRWS provides for the collection of liquid radioactive waste and storage for a seven-day period without processing. This permits processing at times when staff and normal electrical power are available. The design objectives of the LRWS are

- to receive, hold, process, monitor and either recycle or release the maximum expected volume of wastewater arising from plant operations, including anticipated operational occurrences.
- to provide the capability for sampling to ensure that liquid releases of radioactive material in liquid effluents are ALARA.
- to incorporate sufficient system flexibility to ensure adequate system performance in the event of equipment outages, assuming operation with design basis fuel leakage.
- to maintain worker radiation exposures ALARA during normal operation, inspection, testing and maintenance, in accordance with RG 8.8.
- to ensure adequate water quality prior to recycling or releasing the processed liquid.

Process and effluent monitoring and sampling systems are discussed in Section 11.5.

The LRWS complies with the following General Design Criteria found in 10 CFR 50, Appendix A:

- GDC 2 as it relates to structures and components of the LRWS, by using the guidance of RG 1.143 for the seismic, safety and quality classifications

- GDC 3 as it relates to protecting the LRWS from the effects of fires or explosions by avoiding the generation of explosive gas mixtures and exothermic reactions with ion exchange resins
- GDC 60 as it relates to the design of the LRWS to control releases of radioactive liquid effluents generated during normal reactor operations, including AOOs (Section 11.2.3)
- GDC 61 as it relates to radioactive waste systems being designed to provide for adequate safety under normal and postulated accident conditions, and designed with suitable shielding for radiation protection and with appropriate containment, confinement, and filtering systems

The LRWS components are evaluated and classified as RW-IIa, RW-IIb or RW-IIc, as described in RG 1.143, by comparing the radioisotopic content of the component with the A_1 and A_2 quantities listed in Appendix A of 10 CFR 71. The safety classification for the LRWS components applies to components up to and including the nearest isolation device. The resulting safety classifications for LRWS components are listed in Table 11.2-1. The applicable standards from RG 1.143 Table 1 are used in the design, construction, and testing of the LRWS components (see Table 11.2-10). The applicable design criteria from RG 1.143 Table 2, Table 3, and Table 4 are used in the design analysis of the LRWS components.

Design features are provided in accordance with the requirements of 10 CFR 20.1406, following the guidance of RG 4.21 to the extent practicable, to reduce contamination of the facility and the environment, facilitate eventual decommissioning, and reduce the generation of radioactive waste. Additional details are provided in Section 12.3.6.

The principle components, piping, and valves that contain radioactive fluids are designed to the seismic and quality requirements of RG 1.143. The design of the LRWS utilizes and conforms to the guidance provided in RG 1.143, including Branch Technical Position 11-6.

The RWB safety classification is RW-IIa as described in Section 3.2.1.

11.2.3 Radioactive Effluent Releases

11.2.3.1 Radioactive Releases

The facility design reduces liquid effluent discharges from the LRWS to the environment by adequately processing liquid wastes and monitoring releases. The design employs the use of a single point of discharge for liquid effluents to the environment through the LRWS discharge header, which is sent to the utility water system (UWS) discharge basin. The calculated annual activity discharged through liquid effluents is tabulated by isotope and provided in Table 11.2-5.

An alternate methodology to replace PWR-GALE was developed because PWR-GALE was written for large PWRs and some unique NuScale design features that were inadequately modeled in PWR-GALE. This alternate methodology uses first principles based calculations, combined with more recent nuclear industry experience.

The calculation of liquid effluent releases is consistent with RG 1.112, as modified by Technical Report TR-1116-52065 (Reference 11.2-2). The calculation of off-site dose consequences from normal liquid effluents is consistent with RG 1.109.

The radioisotopes that are assumed to be available for offsite release originate in the reactor core and the primary coolant. These radioisotopes are assumed to propagate through plant systems by various pathways and processes, and eventually are released. The calculation of this propagation through pathways and processes is performed using appropriate flow rates, decontamination factors and radioactive decay characteristics.

Normal liquid effluents are routed through the LRWS using two pathways: low-conductivity waste, and high-conductivity waste. The influent quantities for these are tabulated in Table 11.2-3. The expected activity of the influent streams is based on the realistic coolant activities from Section 11.1.3. For the purposes of calculating liquid effluents, liquid waste processing is assumed to be confined to a single pathway. Therefore, the liquid waste that is input into the LCW collection tank will be processed only by the LCW processing equipment, the HCW collection tank contents will be processed only by the HCW processing equipment and the detergent collection tank contents are expected to contain a negligible amount of radionuclides. The input assumptions related to decontamination factors for the LRWS processing equipment are listed in Table 12.2-12.

The influent streams to the LCW and HCW collection tanks include liquid waste that contains expected activity levels as indicated in Table 11.2-3. The activity concentrations listed in Table 11.2-3 that are based on primary coolant activity (PCA) are the indicated fraction of the radionuclide concentrations listed in Table 11.1-6. The activity concentrations that are labeled as 'CVCS outlet' has a radionuclide concentration based on primary coolant that has been processed by the CVCS processing equipment assuming the input values and decontamination factors listed in Table 11.2-4.

The activity indicated as the pool source term in Table 11.2-3 is a time weighted average of the reactor pool water activity concentration based on radionuclides being added to the pool water during NPM disassembly and being removed by the PCUS demineralizers. Prior to NPM disassembly during a refueling outage, the primary coolant is cleaned up by the CVCS demineralizers. Afterwards, the NPM is disconnected, relocated to the refueling pool and disassembled. Between refueling outages, the reactor pool water is cleaned by the PCUS demineralizers and filters.

The activity indicated as CES liquid in Table 11.2-3 is primary coolant that leaves the primary system, through leaks into the containment vessel. This is treated as primary coolant through an evaporator, with the associated partition fractions listed in Table 11.2-4. These decontamination factors credit the evaporation process within containment, resulting in the majority of radioactivity remaining behind. The radioactivity that does get out is divided between gaseous and liquid streams in the CES condenser.

The activities indicated as secondary coolant in Table 11.2-3 are the radionuclide activities listed in Table 11.1-7.

The effluents from the LRWS waste processing pathways are discharged through a single discharge header to the utility water system (UWS) discharge basin, where it is diluted, monitored and released. The assumed dilution factor is described in Section 11.2.3.3 and is listed in Table 11.2-4. An additional adjustment is added to the total non-tritium liquid release to account for unidentified AOOs. This adjustment is a reactor thermal power scaled value from NUREG-0017, Section 2.2.23. This value is also provided in Table 11.2-4. The total resultant liquid release concentrations are provided in Table 11.2-8, and demonstrate compliance with 10 CFR 20 Appendix B, Table 2.

The maximum individual doses are calculated using the LADTAP II Code, using the input parameters listed in Table 11.2-6. The resultant doses are presented in Table 11.2-7 and demonstrate compliance with the limits of 10 CFR 50 Appendix I.

COL Item 11.2-2: A COL applicant that references the NuScale Power Plant design certification will calculate doses to members of the public using the site-specific parameters, compare those liquid effluent doses to the numerical design objectives of 10 CFR 50, Appendix I, and comply with the requirements of 10 CFR 20.1302 and 40 CFR 190.

11.2.3.2 Compliance with Branch Technical Position 11-6

The only outdoor tank expected to contain radioactive liquids is the PSCS storage tank, described in FSAR Section 9.1.3.2.4. The PSCS storage tank catch basin has sufficient volume to store the contents of the PSCS storage tank plus the contents of related piping. The radionuclide inventory of the PSCS storage tank is provided in Table 12.2-10.

An analysis of an accidental release of radioactive liquid effluents in groundwater and surface water is site-specific and addressed by COL Item 2.4-1 and COL Item 11.2-3.

COL Item 11.2-3: A COL applicant that references the NuScale Power Plant design certification will perform a site-specific evaluation of the consequences of an accidental release of radioactive liquid from the pool surge control system storage tank in accordance with NRC Branch Technical Position 11-6.

11.2.3.3 Dilution Factors

The liquid effluent from LRWS is discharged through the discharge header and ties into the UWS as shown in Figure 11.2-1g. The UWS receives discharge water from multiple sources that provides dilution for the LRWS discharge in the discharge basin (Section 9.2.9). The UWS also provides a signal to LRWS in the event that dilution flow reduces to an unacceptable level to automatically close the LRWS discharge header isolation valves (Section 11.5.2.1).

A dilution factor of 5.34 cfs of the LRWS discharge is assumed in the calculation of the discharge concentrations, as shown in Table 11.2-4. This ensures that the discharge concentrations are within 10 CFR 20 Appendix B, Table 2, limits. The unrestricted area doses are calculated using an additional dilution factor of 270 cfs (e.g., river), which results in the unrestricted area doses being within 10 CFR 50, Appendix I, limits.

COL Item 11.2-4: A COL applicant that references the NuScale Power Plant design certification will perform a site-specific evaluation using the site-specific dilution flow.

11.2.3.4 Site-Specific Cost-Benefit Analysis

COL Item 11.2-5: A COL applicant that references the NuScale Power Plant design certification will perform a cost-benefit analysis as required by 10 CFR 50.34a and 10 CFR 50, Appendix I, to demonstrate conformance with regulatory requirements. This cost-benefit analysis is to be performed using the guidance of Regulatory Guide 1.110.

11.2.4 Testing and Inspection Requirements

The LRWS preoperational tests are described in Section 14.2 and include the applicable testing and inspection requirements from RG 1.143.

Inspection and testing provisions are incorporated to enable periodic evaluation of the operability and required functional performance of active components of the system.

11.2.5 Instrumentation and Controls

The LRWS waste collection is operated in the automatic mode and LRWS processing is operated in a batch-type mode. For normal operation, automated and manual valves are aligned to collect the waste from other systems, hold it until processed, and discharge or recycle treated waste.

The LRWS processing equipment and processing functions are controlled and monitored from the WMCR by an operator.

The permanent plant controls and indications for filling waste collection tanks are automatic and are controlled by the plant control system with indication in the WMCR. The atmospheric tanks in LRWS include high-level alarms and controls to prevent overflow. If a collection tank high-level alarm is received, system valves will automatically realign to direct the incoming waste flows toward the collection tank that was in the standby mode.

The liquid radioactive waste effluent discharge line is a double-walled pipe that has dual radiation monitors, dual automated isolation valves, a flow-indicating transmitter with totalizer, and a pressure transmitter that monitors the pipe's annulus. The double-walled pipe's annulus is pressurized to be greater than the process or groundwater pressure and is alarmed to stop the discharge flow upon an indication of low pressure.

A liquid radioactive waste discharge is automatically isolated upon an alarm due to a low dilution flow indication, a low pressure indication in the discharge pipe annulus, or a high-radiation alarm in a discharge line radiation monitor.

Table 11.2-9 contains a listing of the major LRWS instruments.

11.2.6 References

11.2-1 American National Standards Institute/American Nuclear Society, "Mobile Low-Level Radioactive Waste Processing Systems," ANSI/ANS 40.37-2009, La Grange Park, IL.

11.2-2 NuScale Power, LLC, “Effluent Release (GALE Replacement) Methodology and Results,” TR-1116-52065, Rev. 1.

Table 11.2-1: Major Component Design Parameters

Component (Quantity)	RG 1.143 Safety Classification	Type	Capacity	Design Pressure (psig)	Design Temperature (°F)	Material	Table for Assumed Radioactive Content
Degasifier (2)	RW-IIa	Pressure vessel	12,500 gallons	150	225	Stainless steel	Table 11.2-11
LCW collection tank (2)	RW-IIb	Vertical, cylindrical	16,000 gallons	15	150	Stainless steel	Table 12.2-13a
HCW collection tank (2)	RW-IIc	Vertical, cylindrical	16,000 gallons	15	150	Stainless steel	Table 12.2-13a
LCW sample tank (2)	RW-IIc	Vertical, cylindrical	16,000 gallons	15	150	Stainless steel	Table 12.2-13b
HCW sample tank (2)	RW-IIc	Vertical, cylindrical	16,000 gallons	15	150	Stainless steel	Table 12.2-13b
Oil separator (2)	RW-IIc		900 gallons	Atmospheric	150	Stainless steel	Table 12.2-13a
Detergent waste collection tank (1)	RW-IIc	Vertical, cylindrical	500 gallons	15	150	Stainless steel	-----
Detergent waste drain filter (1)	RW-IIc	Cartridge	25 gpm	15	150	Stainless steel (housing)	-----
Demineralized water break tank (1)	RW-IIc	Vertical, cylindrical	10,000 gallons	15	100	Stainless steel	-----
Granulated activated carbon filter (2)	RW-IIc	Vertical	57 gpm	90	150	Stainless steel	LCW: Table 12.2-13a HCW: Table 12.2-13b
Tubular ultrafiltration skid (2)	RW-IIc	Vertical	50 gpm	60	150	Stainless steel	LCW: Table 12.2-13a HCW: Table 12.2-13b
Reverse osmosis skid (2)	RW-IIc	Vertical	50 gpm	60	150	Stainless steel	LCW: Table 12.2-13a HCW: Table 12.2-13b
Cation demineralizer (1)	RW-IIc	Vertical, cylindrical	20 ft^3	150	175	Stainless steel	Table 12.2-13a
Anion demineralizer (1)	RW-IIc	Vertical, cylindrical	20 ft^3	150	175	Stainless steel	Table 12.2-13a
Mixed bed demineralizer (1)	RW-IIc	Vertical, cylindrical	16 ft^3	150	175	Stainless steel	Table 12.2-13b
Cesium demineralizer (1)	RW-IIc	Vertical, cylindrical	20 ft^3	150	175	Stainless steel	Table 12.2-13b
Antimony demineralizer (1)	RW-IIc	Vertical, cylindrical	20 ft^3	150	175	Stainless steel	Table 11.2-13

Table 11.2-1: Major Component Design Parameters (Continued)

Component (Quantity)	RG 1.143 Safety Classification	Type	Capacity	Design Pressure (psig)	Design Temperature (°F)	Material	Table for Assumed Radioactive Content
Liquid waste transfer pump (13)	RW-IIc	Double diaphragm	25 gpm	150	150	Stainless steel	-----
Demineralized water break tank transfer pump (2)	RW-IIc	Centrifugal	200 gpm	150	100	Stainless steel	-----

Table 11.2-2: Off-Normal Operation and Anticipated Operational Occurrence Consequences

Off Normal Operation/AOO		Consequences	
Event	**Indication**	**System Response**	**Corrective Action**
High level or loss of vacuum in degasifier	Alarm in main control room (MCR) and WMCR	Lineup automatically shifts to second degasifier	Corrective maintenance of idle equipment in <72 hr
Degasifier transfer pump trips	Alarm in MCR and WMCR	Lineup automatically shifts to second degasifier	Corrective maintenance of idle equipment in <72 hr
Waste collection tank transfer pump failure	Alarm in WMCR	Align redundant pump	Corrective maintenance of idle equipment in <7 days
High-high level in waste collection or sample tanks	Alarm in MCR and WMCR	Flow automatically diverts to standby tank on high level or stops on high-high level	Investigate cause and procedures to prevent challenges to control system
Processing skid equipment high-differential pressure or contact radiation	Operator surveillance or alarm in WMCR	Shutdown process if excessive or plan cleaning or replacement prior to next batch	Membrane cleaning / resin replacement
One train of processing equipment inoperable	Operator surveillance	Use one train to alternately process LCW and HCW	Repair or replace components and restore operability with 14 days
Sample tank shows sample of high radioactivity	Review of sample results	Pump to waste collection tank or processing equipment	Diagnose cause and repair or replace component
High radiation in LRWS discharge header	High radiation alarm	Discharge valves (two in series) automatically close	Diagnose cause, flush the lines to collection tank, and reprocess
Low guard pipe pressure on buried LRWS discharge pipe	MCR and WMCR alarm	Discharge valves automatically close	Diagnose cause and repair buried LRWS discharge pipe
Low dilution flow at LRWS discharge	MCR and WMCR alarm	Discharge valves automatically close	Diagnose cause and suspend discharge operations until dilution flow is adequate
Area radiation alarm, leak, or spill	MCR and WMCR alarm or operator surveillance	Suspend processing, prevent the spread of contamination	Investigate cause and initiate cleanup and corrective maintenance
Loss of air pressure	MCR and WMCR alarm	Air operated pumps and components stop in safe mode	Investigate and resume operation when air pressure is restored
Loss of nitrogen pressure	MCR and WMCR alarm	Secure flow to gaseous radioactive waste system and use degasifier vacuum for gas waste holdup	Investigate, restore nitrogen pressure within seven days, and resume operation
Loss of alternating current or instrumentation and control power	MCR and WMCR alarm	Verify suspension of processing	Investigate and resume processing when power is available

Table 11.2-3: Expected Liquid Waste Inputs

LRWS Input Source	Expected Input Rate (12 NPMs)	Expected Activity
LCW collection tank		
RXB/RWB equipment drains	2.9E+04 gpy 80 gpd	0.001 primary coolant activity (PCA)
Pool Leak detection	2.6E+05 gpy 700 gpd	Pool water source term
Other equipment drains	1.1E+04 gpy 29 gpd	0.093 PCA
Normal letdown (12 operating units)	2.7E+05 gpy 730 gpd	CVCS outlet
Letdown from ≤120 °F to normal operating temperature (9 events per year)	3.5E+04 gpy	CVCS outlet
Letdown from 420 °F to normal operating temperature (2 events per year)	2.8E+03 gpy	CVCS outlet
Degasification prior to shutdown (12 events per year)	2.0E+02 gpy	1.0 PCA
Fresh resin rinse mid-cycle (1 event per year)	1.8E+03 gpy	CVCS outlet
LCW Total	**6.0E+05 gpy**	
HCW collection tank		
RXB/RWB floor drains (via oil separator)	7.3E+04 gpy 200 gpd	0.1 PCA
RXB RCCW drain tank (via oil separator)	3.6E+01 gpy	0.001 PCA
ANB hot machine shop, decontamination room sump (via oil separator)	9.0E+04 gpy	0.01 PCA
RXB chemical drain tank (Hot lab sink) (via oil separator)	8.8E+03 gpy 24 gpd	0.05 PCA
RXB chemical drain tank (CES sample tank) (via oil separator)	4.4E+04 gpy 120 gpd	CES liquid
Pump seal leaks (via oil separator)	1.1E+04 gpy 30 gpd	0.1 PCA
Valve packing leaks (via oil separator)	6.6E+03 gpy 18 gpd	0.1 PCA
Groundwater / Condensation (via oil separator)	2.5E+05 gpy 680 gpd	0.001 PCA
Equipment area decontamination (outside hot machine shop) (via oil separator)	1.5E+04 gpy 40 gpd	0.01 PCA
CVCS demineralizer sluice water (19 events per year)	2.9E+03 gpy	CVCS outlet
PCUS demineralizer sluice water (1.2 events per year)	3.6E+03 gpy	Pool water source term
LRWS demineralizer sluice water (except mixed bed; 4 events per year)	1.0E+03 gpy	CVCS outlet
LRWS mixed bed demineralizer sluice water (1 event per year)	4.5E+02 gpy	CVCS outlet
GAC filter sluice water (0.2 events per year)	7.5E+01 gpy	CVCS outlet
SRST transfer water	3.4E+03 gpy	CVCS outlet
PST transfer water	1.4E+03 gpy	CVCS outlet
PSC storage tank dike water	9.1E+03 gpy	Pool water source term
Miscellaneous CIP water	2.0E+04 gpy	CVCS outlet

Table 11.2-3: Expected Liquid Waste Inputs (Continued)

LRWS Input Source	Expected Input Rate (12 NPMs)	Expected Activity
Secondary coolant sampling drains	4.2E+03 gpy	Secondary coolant
Condensate polisher rinse and transfer	3.6E+04 gpy	Secondary coolant
Condensate polisher regeneration solutions	1.0E+04 gpy	Secondary coolant
Turbine Generator Building floor drains	2.2E+04 gpy	Secondary coolant
Pool boron adjustment	1.6E+04 gpy	Pool water source term
HCW Total	**6.2E+05 gpy**	

Note: Assumes 12 NPMs operating on a two-year refueling cycle.

Table 11.2-4: Liquid Effluent Release Calculation Inputs

NuScale Effluent Source Term Model Assumption	Value
Primary coolant source term	Table 11.1-6
CVCS demineralizer decontamination factors:	
- Halogens	100
- Cs, Rb	2
- Others	50
PCUS demineralizer decontamination factors:	
- Halogens	100
- Cs, Rb	2
- Others	50
PCUS filter efficiency	0%
PCUS flow rate	594 gpm
Pool water mass	2.87E+10 grams
Time between refueling outages/NPM disassembly	2 months
CES liquid partition fractions:	
- Noble gases	1
- Halogens	100
- Others	1000
Secondary coolant source term	Table 11.1-7
AOO adjustment	0.09 Ci/year
UWS dilution factor	5.34 ft^3/sec

Table 11.2-5: Estimated Annual Releases to Liquid Radioactive Waste System Discharge Header

Nuclide	LRWS LCW Sample Tank Release (Ci/yr)	LRWS HCW Sample Tank Release (Ci/yr)	Plant Liquid Release without AOO Adjustment (Ci/yr)	Total Liquid Release with AOO Adjustment (Ci/yr)
Br82	4.36E-11	1.86E-06	1.86E-06	2.37E-06
Br83	1.73E-25	7.54E-21	7.54E-21	9.60E-21
I129	1.41E-14	5.87E-10	5.87E-10	7.48E-10
I130	3.08E-12	1.34E-07	1.34E-07	1.70E-07
I131	3.42E-07	3.40E-03	3.40E-03	4.33E-03
I132	1.03E-08	1.59E-04	1.60E-04	2.03E-04
I133	2.97E-09	9.97E-05	9.97E-05	1.27E-04
I135	1.24E-13	5.39E-09	5.39E-09	6.86E-09
Rb86	1.83E-06	6.05E-05	6.23E-05	7.94E-05
Cs132	2.42E-08	7.99E-07	8.24E-07	1.05E-06
Cs134	3.85E-04	1.27E-02	1.31E-02	1.66E-02
Cs136	6.17E-05	2.03E-03	2.10E-03	2.67E-03
Cs137	2.37E-04	7.82E-03	8.05E-03	1.03E-02
P32	4.76E-15	7.31E-11	7.31E-11	9.31E-11
Co57	4.55E-17	6.99E-13	6.99E-13	8.90E-13
Sr89	3.65E-09	4.05E-06	4.05E-06	5.16E-06
Sr90	6.21E-11	9.53E-07	9.53E-07	1.21E-06
Sr91	1.26E-14	1.96E-10	1.96E-10	2.49E-10
Sr92	1.63E-25	2.54E-21	2.54E-21	3.23E-21
Y90	5.06E-11	7.75E-07	7.75E-07	9.87E-07
Y91m	8.01E-15	1.25E-10	1.25E-10	1.59E-10
Y91	3.80E-11	5.84E-07	5.84E-07	7.44E-07
Y92	2.79E-21	4.35E-17	4.35E-17	5.53E-17
Y93	4.44E-15	6.91E-11	6.91E-11	8.80E-11
Zr97	2.08E-13	3.23E-09	3.24E-09	4.12E-09
Nb95	1.20E-08	2.36E-05	2.36E-05	3.01E-05
Mo99	2.06E-08	3.18E-04	3.18E-04	4.05E-04
Tc99m	1.99E-08	3.07E-04	3.07E-04	3.91E-04
Tc99	2.32E-12	3.56E-08	3.56E-08	4.53E-08
Ru103	7.03E-11	1.08E-06	1.08E-06	1.38E-06
Ru105	4.09E-20	6.38E-16	6.38E-16	8.12E-16
Ru106	4.97E-11	7.63E-07	7.63E-07	9.71E-07
Rh103m	6.95E-11	1.07E-06	1.07E-06	1.36E-06
Rh105	4.51E-12	6.98E-08	6.98E-08	8.89E-08
Rh106	4.97E-11	7.63E-07	7.63E-07	9.71E-07
Ag110	1.71E-12	3.36E-09	3.36E-09	4.28E-09
Sb124	3.36E-17	2.32E-10	2.32E-10	2.95E-10
Sb125	3.14E-16	2.17E-09	2.17E-09	2.76E-09
Sb127	5.14E-16	3.56E-09	3.56E-09	4.53E-09
Sb129	2.24E-24	1.57E-17	1.57E-17	2.00E-17
Te125m	1.38E-10	2.12E-06	2.12E-06	2.70E-06
Te127m	4.59E-10	7.05E-06	7.05E-06	8.97E-06
Te127	4.50E-10	6.91E-06	6.91E-06	8.80E-06

Table 11.2-5: Estimated Annual Releases to Liquid Radioactive Waste System Discharge Header (Continued)

Nuclide	LRWS LCW Sample Tank Release (Ci/yr)	LRWS HCW Sample Tank Release (Ci/yr)	Plant Liquid Release without AOO Adjustment (Ci/yr)	Total Liquid Release with AOO Adjustment (Ci/yr)
Te129m	1.22E-09	1.87E-05	1.87E-05	2.38E-05
Te129	7.68E-10	1.18E-05	1.18E-05	1.50E-05
Te131m	2.21E-10	3.42E-06	3.42E-06	4.35E-06
Te131	4.97E-11	7.70E-07	7.70E-07	9.80E-07
Te132	1.00E-08	1.55E-04	1.55E-04	1.97E-04
Ba137m	2.24E-04	7.38E-03	7.60E-03	9.68E-03
Ba140	2.97E-10	4.57E-06	4.57E-06	5.82E-06
La140	3.06E-10	4.69E-06	4.69E-06	5.97E-06
La141	2.50E-21	3.89E-17	3.89E-17	4.95E-17
Ce141	5.49E-11	8.42E-07	8.42E-07	1.07E-06
Ce143	3.04E-12	4.71E-08	4.71E-08	6.00E-08
Ce144	5.10E-11	7.82E-07	7.82E-07	9.96E-07
Pr143	4.51E-11	6.93E-07	6.93E-07	8.82E-07
Pr144	5.05E-11	7.75E-07	7.75E-07	9.86E-07
Np239	1.98E-10	3.06E-06	3.06E-06	3.90E-06
Na24	2.50E-09	3.88E-05	3.88E-05	4.94E-05
Cr51	3.26E-09	6.50E-06	6.51E-06	8.28E-06
Mn54	4.72E-09	9.26E-06	9.26E-06	1.18E-05
Fe55	3.82E-07	7.49E-04	7.50E-04	9.55E-04
Fe59	8.62E-08	1.71E-04	1.71E-04	2.17E-04
Co58	9.88E-07	1.20E-03	1.20E-03	1.53E-03
Co60	2.53E-08	4.95E-05	4.95E-05	6.30E-05
Ni63	8.47E-08	1.66E-04	1.66E-04	2.11E-04
Zn65	1.60E-07	3.14E-04	3.14E-04	4.00E-04
Zr95	1.16E-07	2.29E-04	2.29E-04	2.91E-04
Ag110m	1.26E-10	2.47E-07	2.47E-07	3.14E-07
W187	6.44E-09	2.41E-05	2.41E-05	3.07E-05
H3	8.35E+02	2.78E+02	1.11E+03	1.11E+03
C14	2.59E-01	3.37E-02	2.92E-01	3.72E-01
Total	**8.35E+02**	**2.78E+02**	**1.11E+03**	**1.11E+03**

Table 11.2-6: LADTAP II Inputs

Parameter	Value
Source term	Table 11.2-5
Shore-width factor	1.0
Discharge flow rate	270 ft^3/sec
Impoundment reconcentration model	None
Irrigation rate	100 liters/m^2-month
Dilution factor for aquatic food, boating, shoreline, swimming and drinking water	1
Dilution factor for irrigation water usage location	1
Site type	Freshwater
Exposure Pathway:	
• Transit time - aquatic food	0
• Transit time - boating	0
• Transit time - swimming	0
• Transit time - shoreline	0
• Transit time - drinking water	0
• Transit time - irrigated crops	0
• Transit time - milk/meat animal water usage	0
Fraction of crops irrigated using non-contaminated water	0
Fraction of milk/meat animal feed irrigated using non-contaminated water	0
Fraction of milk/meat animal drinking water from non-contaminated water	0

Table 11.2-7: Liquid Effluent Dose Results for 10 CFR 50 Appendix I

Type of Dose	Calculated Dose (mrem/yr)	10 CFR 50, Appendix I ALARA Design Objective (mrem/yr)
Total Body	2.8	3
Individual Organ	9.8 (child bone)	10

Table 11.2-8: Liquid Release Concentrations Compared to 10 CFR 20 Appendix B Limits

Nuclide	Discharge Concentration (µCi/ml)	Concentration Limit (µCi/ml)	Fraction of Limit
Br82	4.97E-13	4.00E-05	1.24E-08
I129	1.57E-16	2.00E-07	7.83E-10
I130	3.56E-14	2.00E-05	1.78E-09
I131	9.06E-10	1.00E-06	9.06E-04
I132	4.25E-11	1.00E-04	4.25E-07
I133	2.66E-11	7.00E-06	3.80E-06
I135	1.44E-15	3.00E-05	4.79E-11
Rb86	1.66E-11	7.00E-06	2.37E-06
Cs132	2.20E-13	4.00E-05	5.49E-09
Cs134	3.49E-09	9.00E-07	3.87E-03
Cs136	5.59E-10	6.00E-06	9.32E-05
Cs137	2.15E-09	1.00E-06	2.15E-03
Co57	1.86E-19	6.00E-05	3.11E-15
Sr89	1.08E-12	8.00E-06	1.35E-07
Sr90	2.54E-13	5.00E-07	5.08E-07
Sr91	5.22E-17	2.00E-05	2.61E-12
Y90	2.07E-13	7.00E-06	2.95E-08
Y91m	3.33E-17	2.00E-03	1.66E-14
Y91	1.56E-13	8.00E-06	1.95E-08
Y92	1.16E-23	4.00E-05	2.90E-19
Y93	1.84E-17	2.00E-05	9.21E-13
Zr97	8.63E-16	9.00E-06	9.59E-11
Nb95	6.30E-12	3.00E-05	2.10E-07
Mo99	8.48E-11	2.00E-05	4.24E-06
Tc99m	8.19E-11	1.00E-03	8.19E-08
Tc99	9.49E-15	6.00E-05	1.58E-10
Ru103	2.88E-13	3.00E-05	9.60E-09
Ru105	1.70E-22	7.00E-05	2.43E-18
Ru106	2.03E-13	3.00E-06	6.78E-08
Rh103m	2.85E-13	6.00E-03	4.75E-11
Rh105	1.86E-14	5.00E-05	3.72E-10
Sb124	6.18E-17	7.00E-06	8.83E-12
Sb125	5.77E-16	3.00E-05	1.92E-11
Sb127	9.50E-16	1.00E-05	9.50E-11
Sb129	4.20E-24	4.00E-05	1.05E-19
Te125m	5.66E-13	2.00E-05	2.83E-08
Te127m	1.88E-12	9.00E-06	2.09E-07
Te127	1.84E-12	1.00E-04	1.84E-08
Te129m	4.99E-12	7.00E-06	7.13E-07
Te129	3.15E-12	4.00E-04	7.87E-09
Te131m	9.12E-13	8.00E-06	1.14E-07
Te131	2.05E-13	8.00E-05	2.57E-09
Te132	4.13E-11	9.00E-06	4.58E-06
Ba140	1.22E-12	8.00E-06	1.52E-07

Table 11.2-8: Liquid Release Concentrations Compared to 10 CFR 20 Appendix B Limits (Continued)

Nuclide	Discharge Concentration (µCi/ml)	Concentration Limit (µCi/ml)	Fraction of Limit
La140	1.25E-12	9.00E-06	1.39E-07
La141	1.04E-23	5.00E-05	2.07E-19
Ce141	2.25E-13	3.00E-05	7.49E-09
Ce143	1.26E-14	2.00E-05	6.28E-10
Ce144	2.09E-13	3.00E-06	6.95E-08
Pr143	1.85E-13	2.00E-05	9.24E-09
Pr144	2.07E-13	6.00E-04	3.44E-10
Np239	8.17E-13	2.00E-05	4.08E-08
Na24	1.04E-11	5.00E-05	2.07E-07
Cr51	1.74E-12	5.00E-04	3.47E-09
Mn54	2.47E-12	3.00E-05	8.23E-08
Fe55	2.00E-10	1.00E-04	2.00E-06
Fe59	4.55E-11	1.00E-05	4.55E-06
Co58	3.20E-10	2.00E-05	1.60E-05
Co60	1.32E-11	3.00E-06	4.40E-06
Ni63	4.43E-11	1.00E-04	4.43E-07
Zn65	8.38E-11	5.00E-06	1.68E-05
Zr95	6.10E-11	2.00E-05	3.05E-06
Ag110m	6.59E-14	6.00E-06	1.10E-08
W187	6.44E-12	3.00E-05	2.15E-07
H3	2.33E-04	1.00E-03	2.33E-01
C14	7.80E-08	3.00E-05	2.60E-03
Total	**2.33E-04**	–	**2.43E-01**

Table 11.2-9: Instrument List

Type	Location	Indication	
		Local	Control Room
Level	HCW collection	Yes	WMCR
Level	HCW sample tanks	Yes	WMCR
Level	LCW collection tanks	Yes	WMCR
Level	LCW sample tanks	Yes	WMCR
Level	Detergent waste collection tank	Yes	WMCR
Level	Degasifiers	Yes	WMCR
Level	Oil separators	Yes	WMCR
Level	Demineralized water break tank	Yes	WMCR
Pressure regulator	Degasifier	No	WMCR
Pressure	Degasifier vent condenser outlet	Yes	WMCR
Pressure	Degasifier liquid transfer pump discharge	Yes	WMCR
Pressure	HCW collection tank transfer pump outlet	Yes	WMCR
Pressure	HCW sample tank transfer pump outlet	Yes	WMCR
Pressure	LCW collection tank transfer pump outlet	Yes	WMCR
Pressure	LCW sample tank transfer pump outlet	Yes	WMCR
Pressure	Demineralized water break tank transfer pumps outlet	Yes	WMCR
Pressure	HCW and LCW processing skid transfer pump outlet	Yes	WMCR
Pressure	HCW and LCW processing skid transfer pump outlet	Yes	WMCR
Pressure	LCW processing tubular ultrafiltration skid	Yes	WMCR
Pressure	LCW processing reverse osmosis skid	Yes	WMCR
Pressure	LCW processing skid demineralizers	Yes	WMCR
Pressure	LCW processing granular activated charcoal filter	Yes	WMCR
Pressure	HCW processing granular activated charcoal filter	Yes	WMCR
Pressure	HCW processing tubular ultrafiltration skid	Yes	WMCR
Pressure	HCW processing reverse osmosis skid	Yes	WMCR
Pressure	Detergent waste collection tank transfer pump outlet	Yes	WMCR
Radioactivity	Treated liquid effluent discharge line - 2 redundant radioactivity-indicating transmitter	Yes	WMCR, MCR
Flow with totalizer	Treated liquid effluent discharge line	Yes	WMCR, MCR
Hydrogen analyzer	Degasifier area	No	WMCR, MCR

Table 11.2-10: Codes and Standards from Regulatory Guide 1.143, Table 1

Component	Design and Construction	Materials	Welding	Inspection and Testing
Piping and valves	ANSI/ASME B31.3[1,2]	ASME Section II[3]	ASME Section IX	ANSI/ASME B31.3
Atmospheric tanks	API-650	ASME Section II	ASME Section IX	API-650
Tanks (0-15 psig)	API-620	ASME Section II	ASME Section IX	API-620
Pressure vessels and tanks (>15 psig)	ASME BPVC Section VIII, Division 1 or Division 2	ASME Section II	ASME Section IX	ASME Section VIII, Division 1 or Division 2
Pumps	API-610; API-674; API-675; ASME BPVC Section VIII, Division 1 or Division 2	ASTM A571- 84(1997) or ASME Section II	ASME Section IX	ASME BPVC Code Section III, Class 3[4]
Heat exchangers	TEMA STD, 8th Edition; ASME BPVC Section VIII Division 1 or Division 2	ASTM B359-98 or ASME Section II	ASME Section IX	ASME Section VIII, Division 1 or Division 2
Flexible hoses and hose connections for MRWP[5]	ANSI/ANS-40.37	ANSI/ANS-40.37	ANSI/ANS-40.37	ANSI/ANS-40.37

Notes:

1. Class RW-IIa and RW-IIb piping systems are to be designed as category "M" systems.
2. Classes RW-IIa, RW-IIb, and RW-IIc are discussed in Regulatory Position 5 of RG 1.143.
3. ASME BPVC Section II required for pressure-retaining components.
4. ASME Code Stamp, material traceability, and the quality assurance criteria of ASME BPVC, Section III, Div. 1, Article NCA are not required. Therefore, these components are not classified as ASME Code Section III, Class 3.
5. Flexible hoses should only be used in conjunction with mobile radwaste processing systems.

Table 11.2-11: Degasifier Radiological Content

Isotope	Activity (Ci)
Kr83m	2.3E-01
Kr85m	9.8E-01
Kr85	2.9E+02
Kr87	5.3E-01
Kr88	1.6E+00
Kr89	3.6E-02
Xe131m	3.8E+00
Xe133m	3.5E+00
Xe133	2.6E+02
Xe135m	3.3E-01
Xe135	8.7E+00
Xe137	1.1E-01
Xe138	3.9E-01
Br82	6.5E-03
Br83	3.7E-02
Br84	1.7E-02
Br85	2.1E-03
I129	1.6E-07
I130	5.2E-02
I131	1.3E+00
I132	6.1E-01
I133	2.0E+00
I134	3.6E-01
I135	1.3E+00
Rb86m	1.5E-06
Rb86	9.1E-03
Rb88	1.6E+00
Rb89	7.1E-02
Cs132	1.8E-04
Cs134	1.6E+00
Cs135m	1.2E-03
Cs136	3.3E-01
Cs137	9.6E-01
Cs138	5.7E-01
P32	2.6E-08
Co57	1.9E-10
Sr89	1.2E-03
Sr90	2.6E-04
Sr91	6.0E-04
Sr92	3.2E-04
Y90	6.3E-05
Y91m	3.2E-04
Y91	1.7E-04
Y92	2.7E-04
Y93	1.3E-04
Zr97	1.9E-04

Table 11.2-11: Degasifier Radiological Content (Continued)

Isotope	Activity (Ci)
Nb95	2.7E-04
Mo99	3.4E-01
Mo101	1.3E-02
Tc99m	3.2E-01
Tc99	9.7E-06
Ru103	3.3E-04
Ru105	1.1E-04
Ru106	2.1E-04
Rh103m	3.2E-04
Rh105	2.3E-04
Rh106	2.1E-04
Ag110	2.5E-04
Sb124	4.8E-07
Sb125	4.2E-06
Sb127	1.8E-05
Sb129	2.3E-05
Te125m	6.2E-04
Te127m	2.0E-03
Te127	7.9E-03
Te129m	5.7E-03
Te129	8.1E-03
Te131m	1.9E-02
Te131	9.2E-03
Te132	1.4E-01
Te133m	1.2E-02
Te134	1.7E-02
Ba137m	9.1E-01
Ba139	3.1E-04
Ba140	1.7E-03
La140	4.9E-04
La141	9.6E-05
La142	4.6E-05
Ce141	2.6E-04
Ce143	2.0E-04
Ce144	2.2E-04
Pr143	2.3E-04
Pr144	2.2E-04
Np239	4.1E-03
Na24	4.3E-01
Cr51	2.5E-02
Mn54	1.3E-02
Fe55	9.5E-03
Fe59	2.4E-03
Co58	3.6E-02
Co60	4.2E-03
Ni63	2.1E-03
Zn65	4.0E-03

Table 11.2-11: Degasifier Radiological Content (Continued)

Isotope	Activity (Ci)
Zr95	3.1E-03
Ag110m	1.0E-02
W187	2.2E-02
H3	1.3E+02
C14	1.1E-02
Ar41	9.8E+00

Note: The radiological content of the liquid in the degasifier in Table 11.2-14 is primary coolant that has been processed through the CVCS demineralizers. The CVCS demineralizer decontamination factors are provided in Table 11.1-2.

Table 11.2-12: LCW Antimony Bed Radiological Content

Isotope	(Ci)
Sb124	2.67E-13
Sb125	8.87E-12
Sb127	6.59E-13
Sb129	3.82E-14
Te125m	1.63E-12
Te127m	1.05E-13
Te127	5.42E-13
Te129m	8.63E-15
Te129	2.95E-14

Figure 11.2-1a: Liquid Radioactive Waste System Diagram

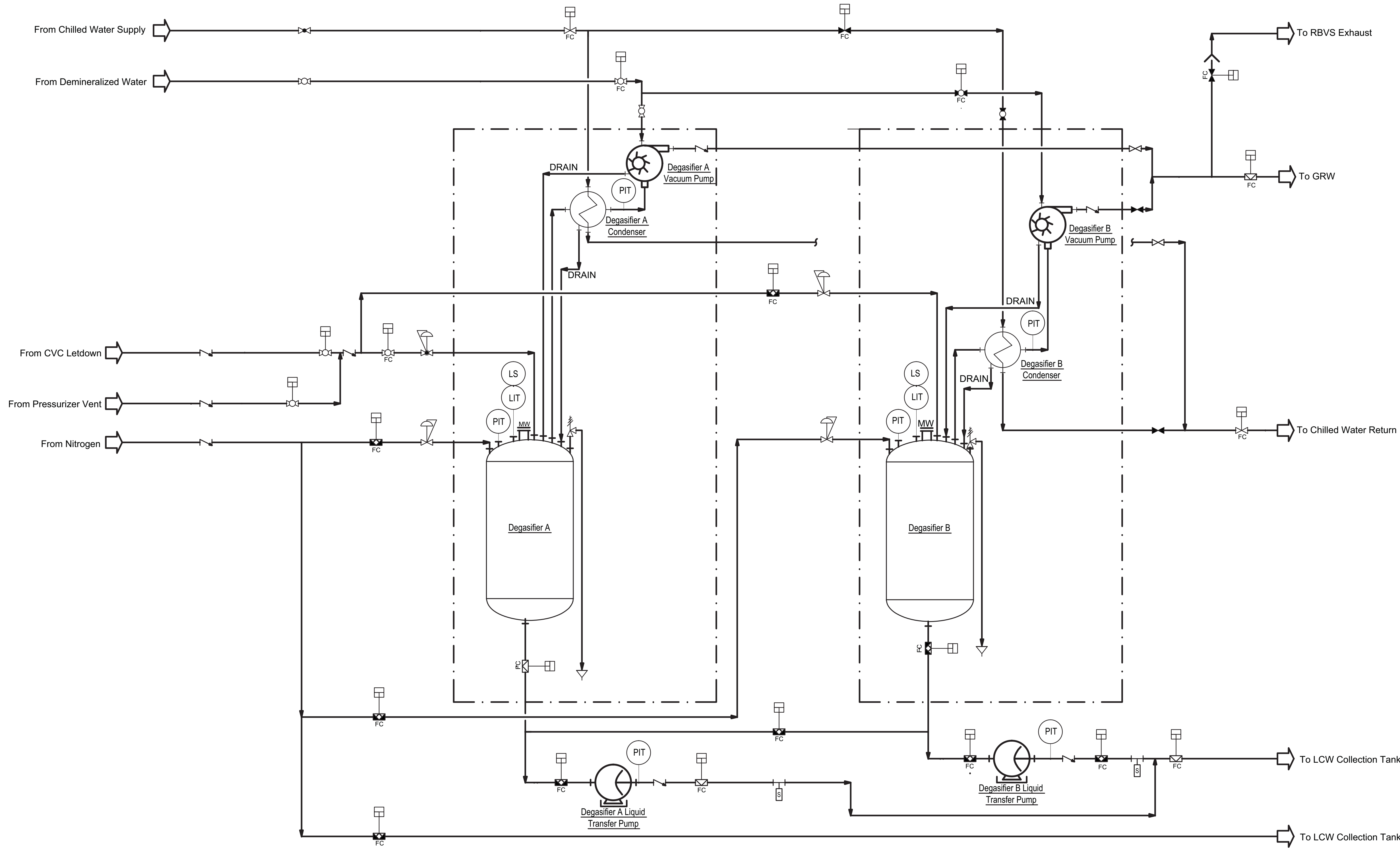


Figure 11.2-1b: Liquid Radioactive Waste System Diagram

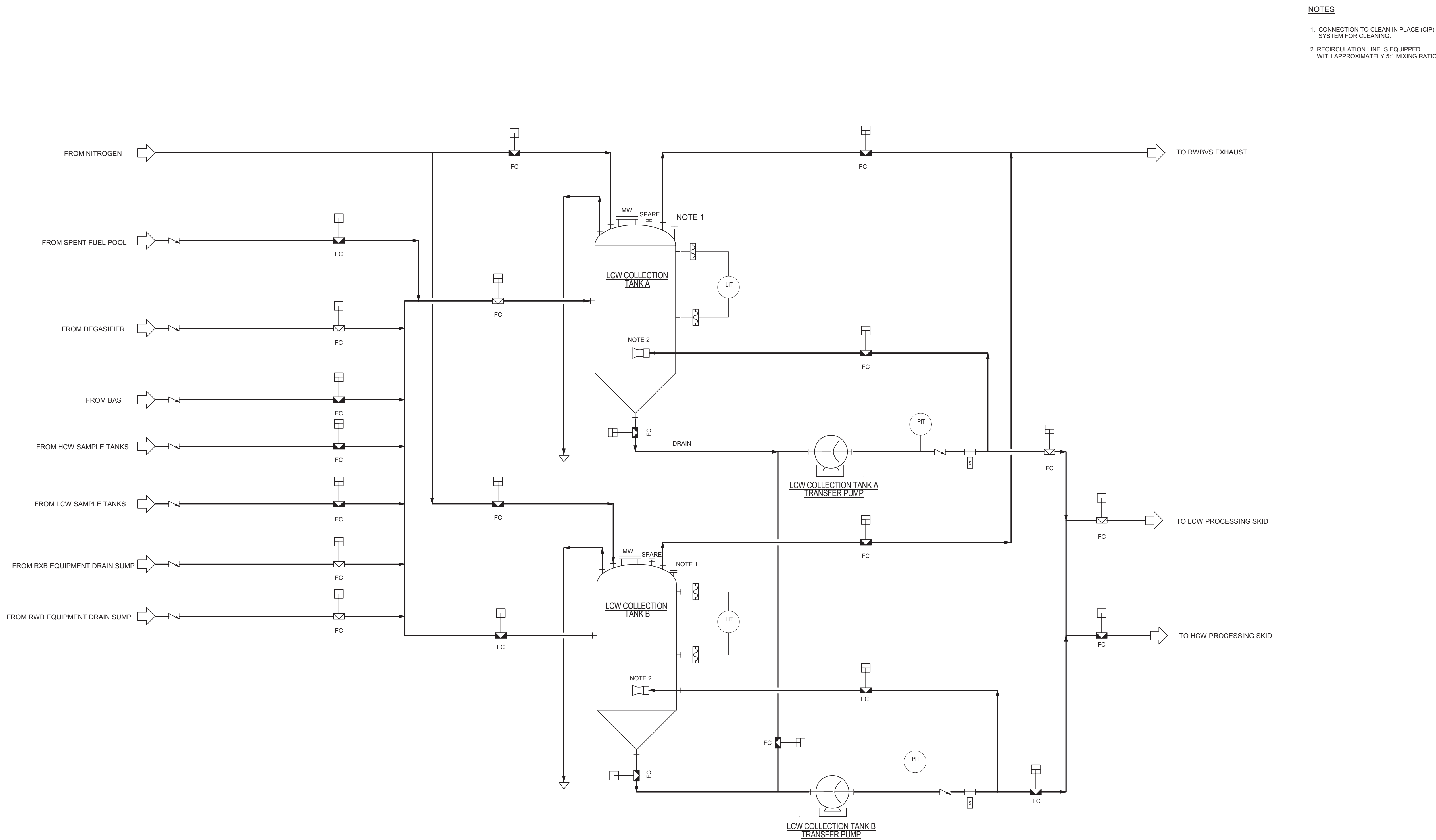


Figure 11.2-1c: Liquid Radioactive Waste System Diagram

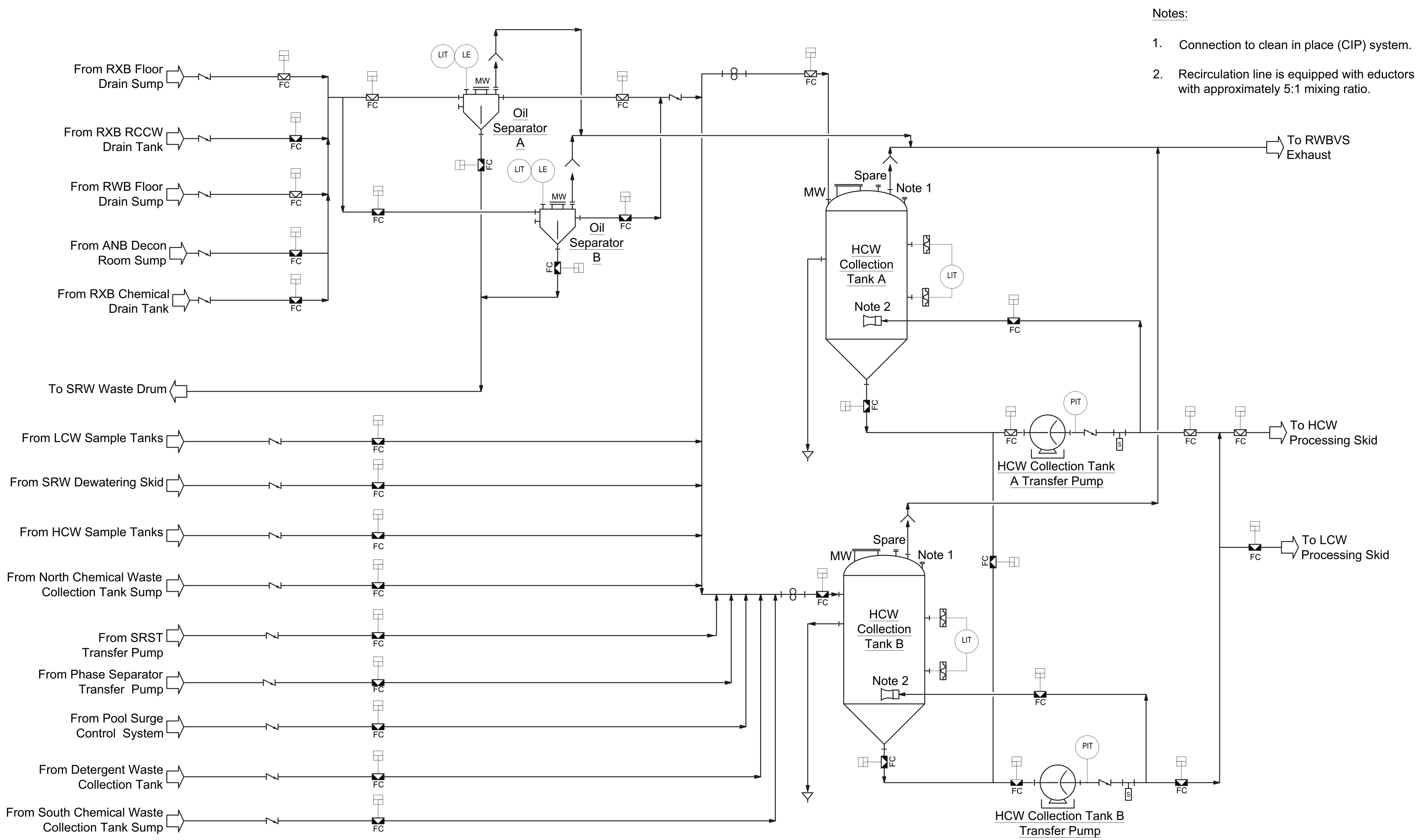


Figure 11.2-1d: Liquid Radioactive Waste System Diagram

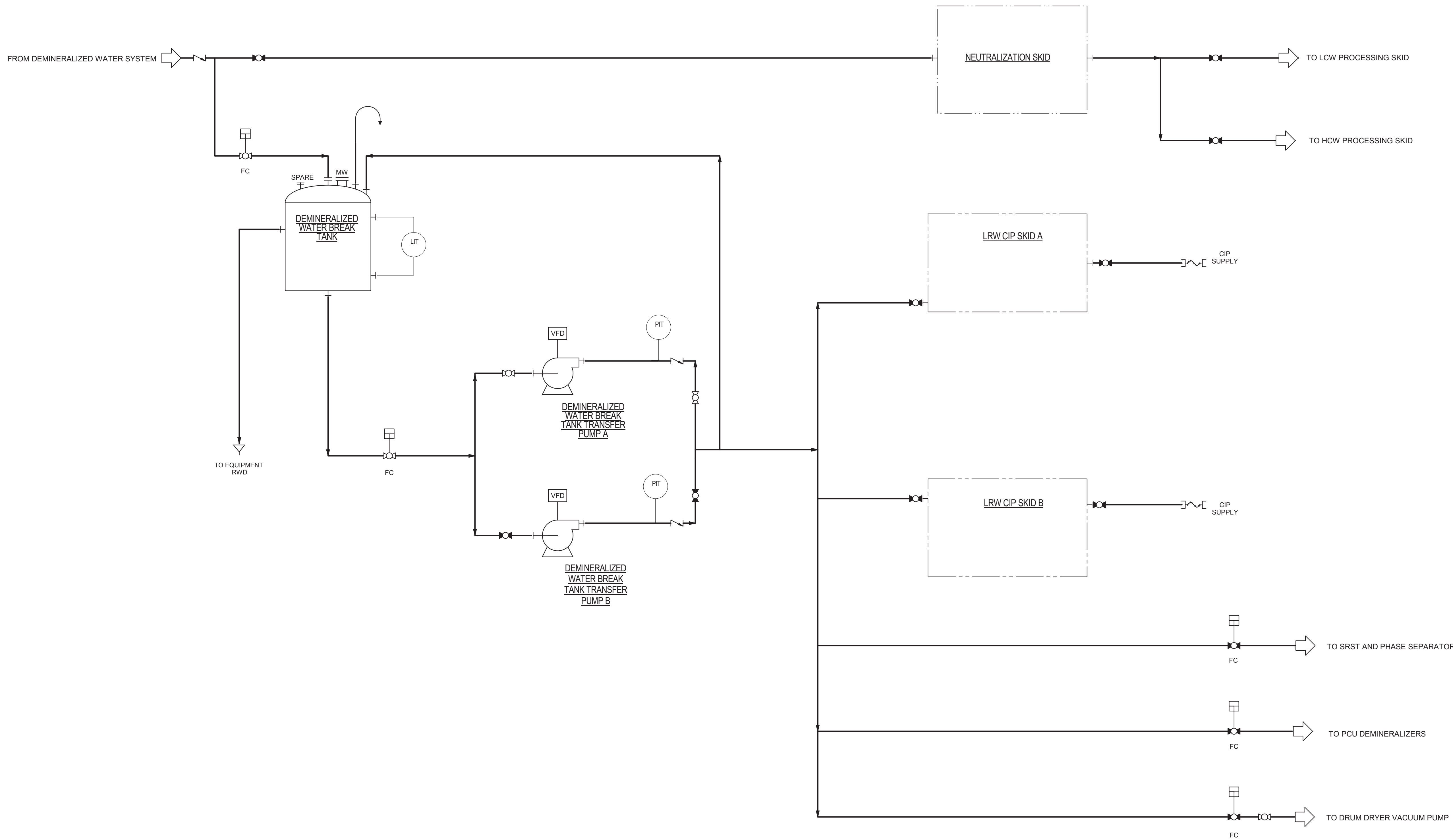


Figure 11.2-1e: Liquid Radioactive Waste System Diagram

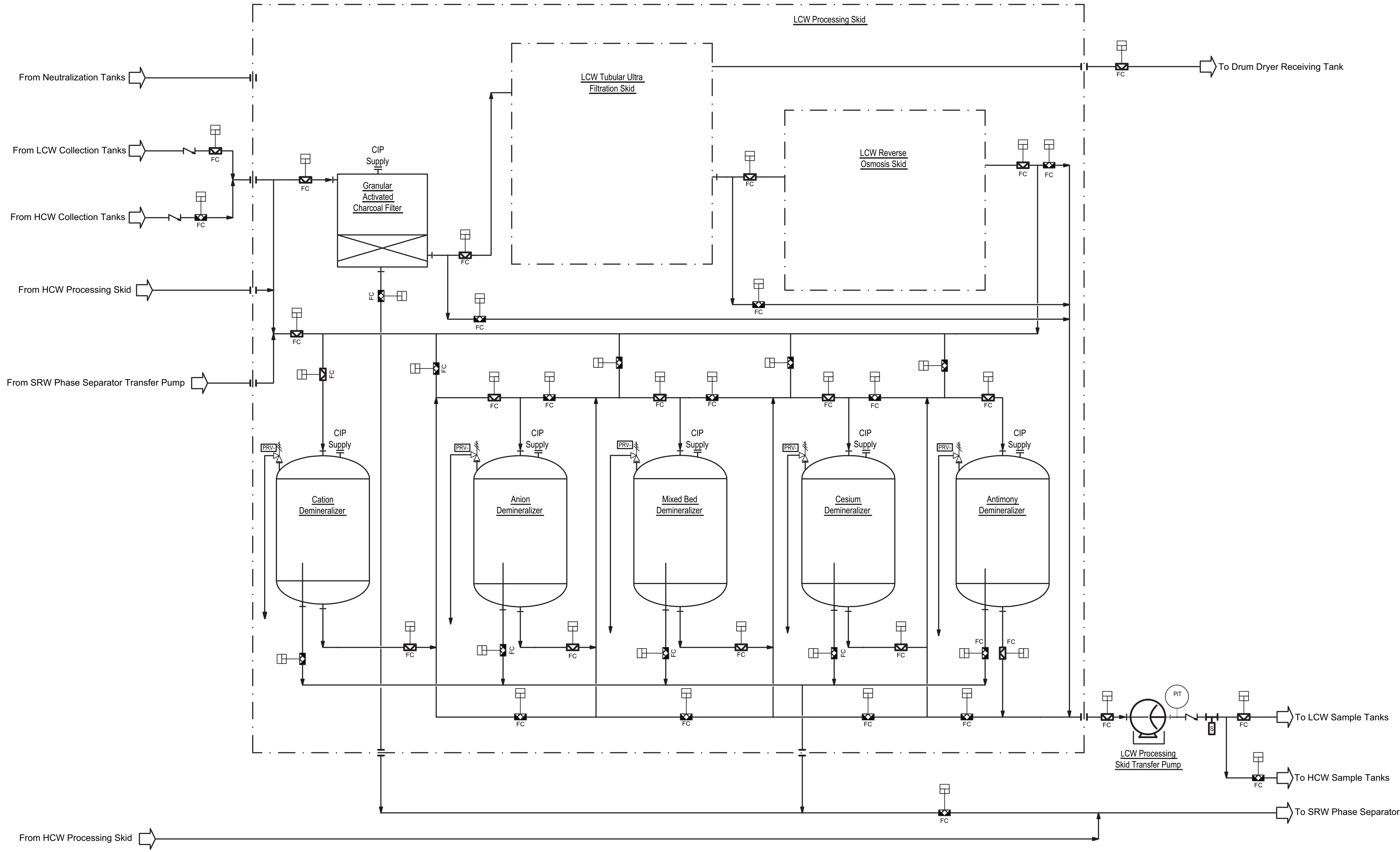


Figure 11.2-1f: Liquid Radioactive Waste System Diagram

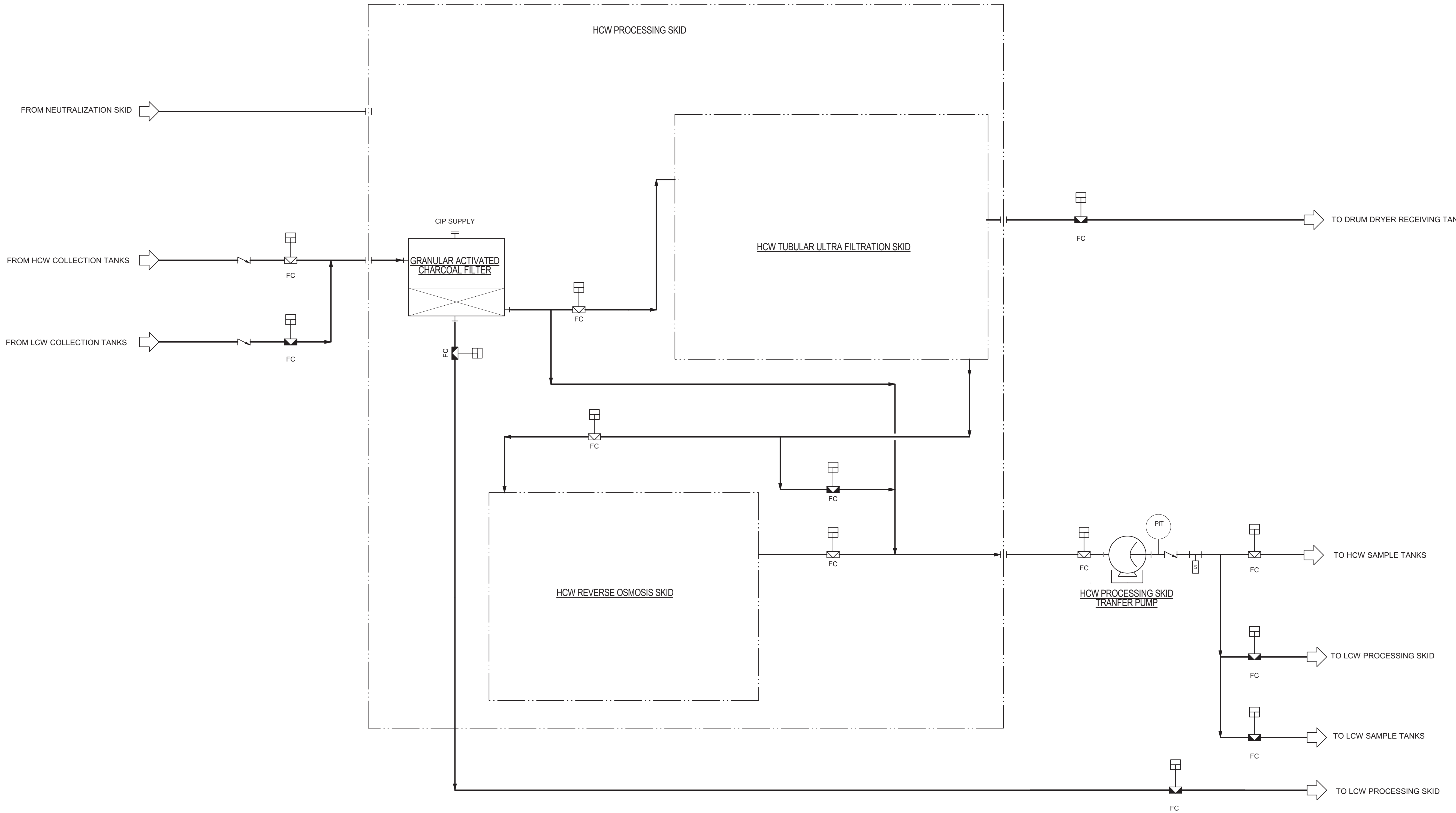


Figure 11.2-1g: Liquid Radioactive Waste System Diagram

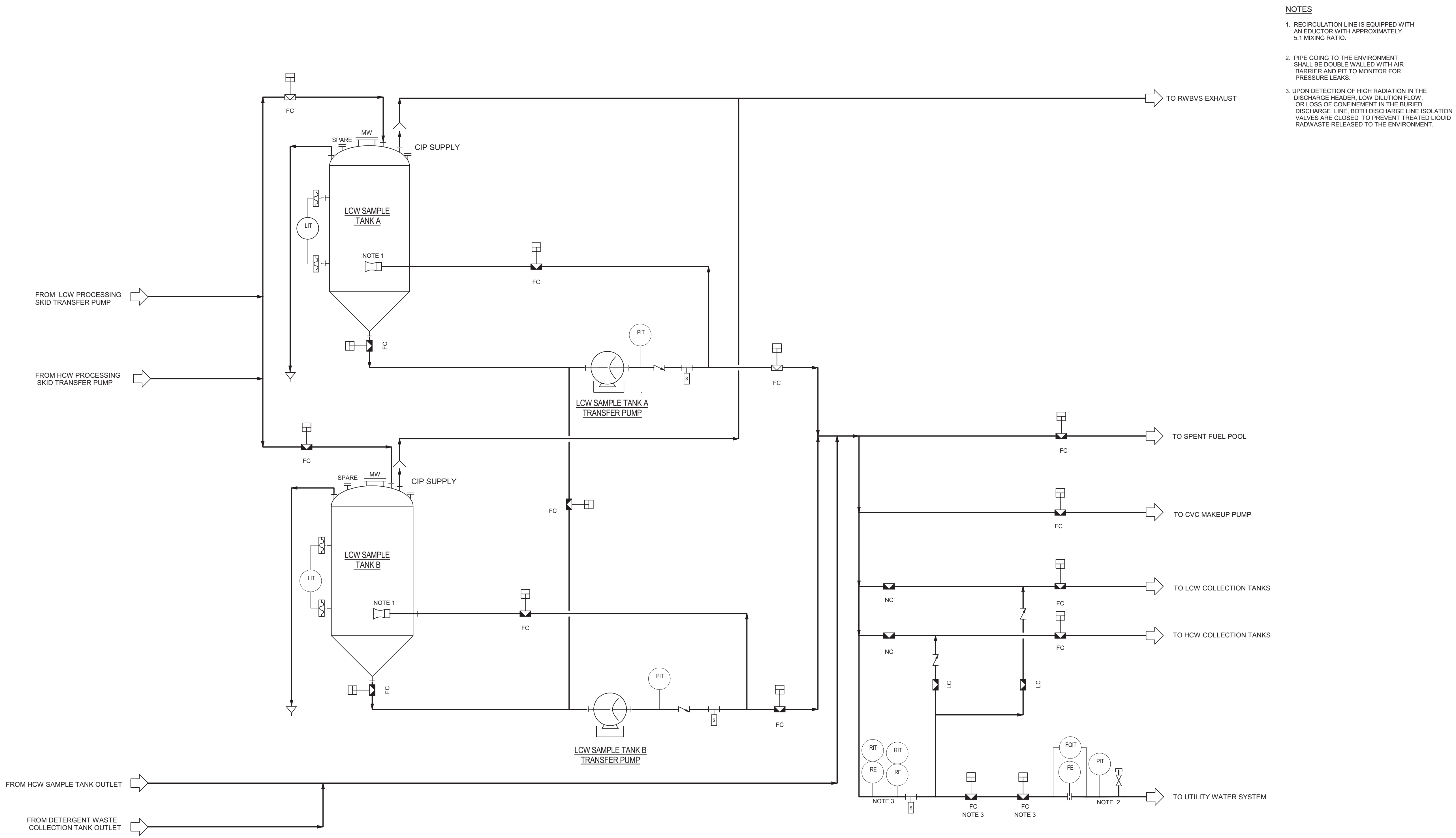


Figure 11.2-1h: Liquid Radioactive Waste System Diagram

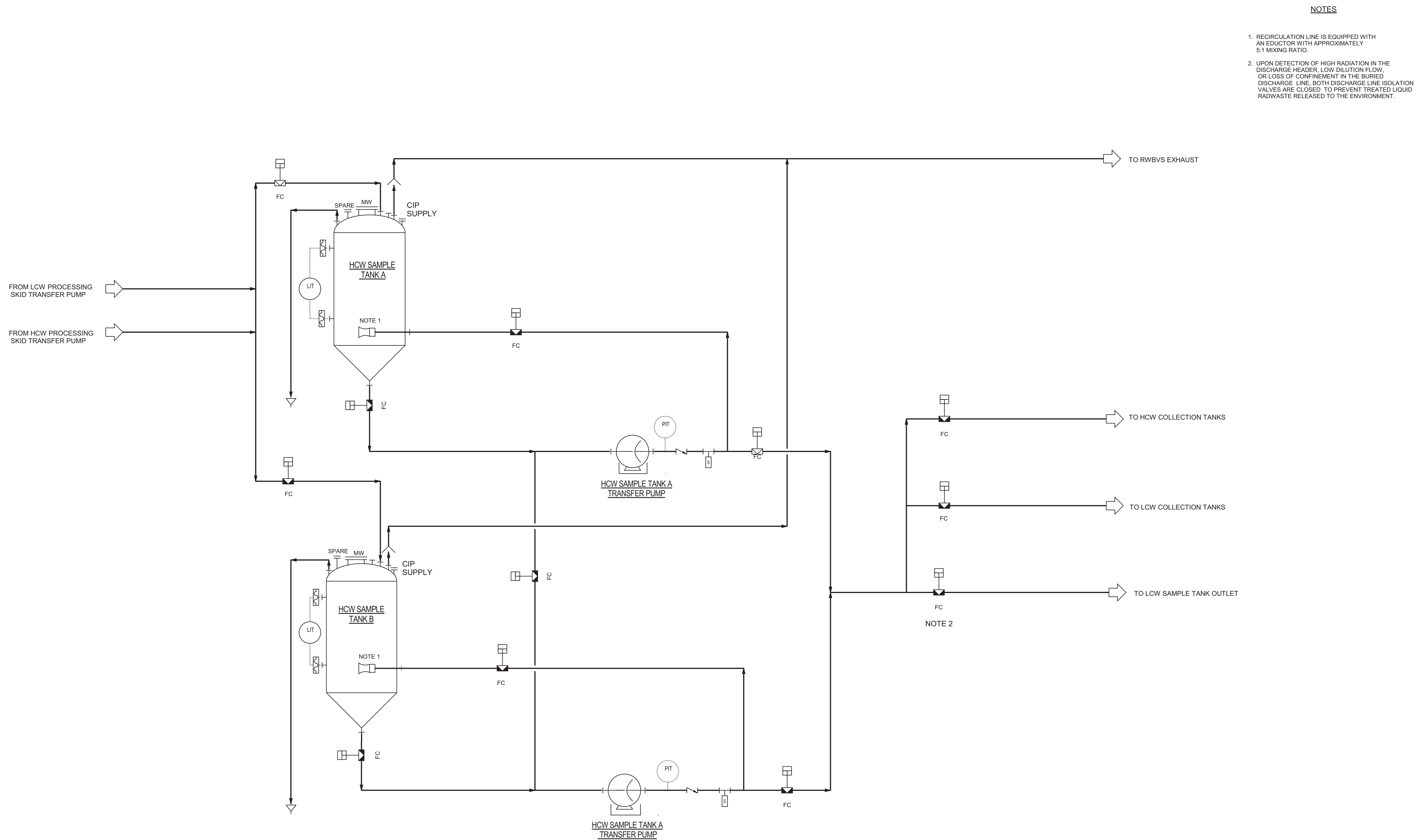


Figure 11.2-1i: Liquid Radioactive Waste System Diagram

NOTES

1. RECIRCULATION LINE IS EQUIPPED WITH AN EDUCTOR WITH APPROXIMATELY 5:1 MIXING RATIO.
2. UPON DETECTION OF HIGH RADIATION IN THE DISCHARGE HEADER, LOW DILUTION FLOW, OR LOSS OF CONFINEMENT IN THE BURIED DISCHARGE LINE, BOTH DISCHARGE LINE ISOLATION VALVES ARE CLOSED TO PREVENT TREATED LIQUID RADWASTE RELEASED TO THE ENVIRONMENT.

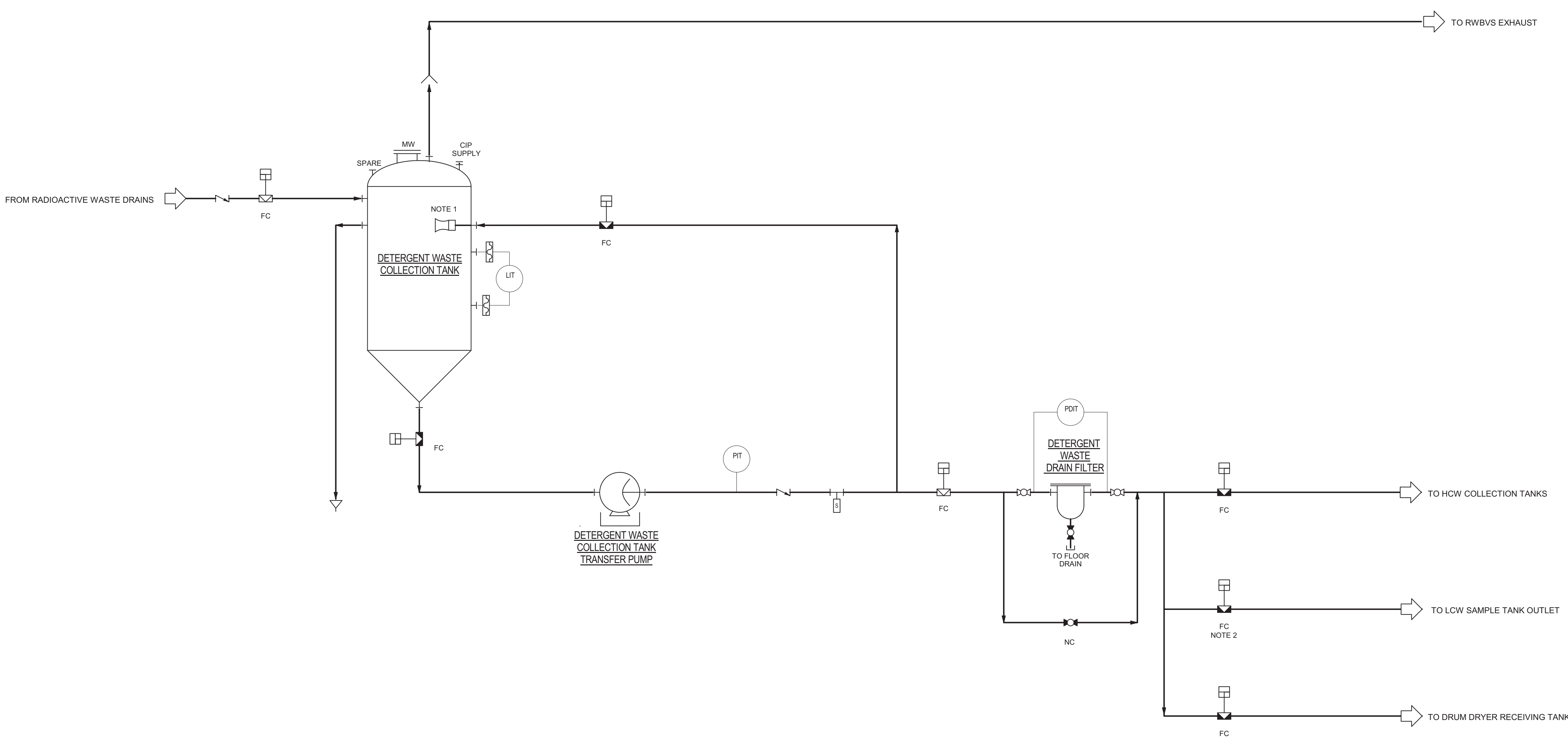


Figure 11.2-1j: Liquid Radioactive Waste System Diagram

11.3 Gaseous Waste Management System

For the NuScale design, the gaseous waste management system is called the gaseous radioactive waste system (GRWS). The GRWS is designed to process the gaseous waste stream from the liquid radioactive waste system (LRWS) degasifier and the containment evacuation system (CES), provide holdup for radioactive decay of xenon and krypton, and convey the gaseous effluent to the Radioactive Waste Building HVAC system (RWBVS), which transports the effluent to the Reactor Building HVAC system (RBVS) for monitoring and release. The GRWS filters out particulate carryover and delays the noble gases through activated charcoal beds until they have decayed sufficiently to allow release to the environment. Design and performance of the charcoal delay system is in accordance with NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code)," April 1985, as modified by TR-1116-52065 (Reference 11.3-1).

Discharge from the GRWS is initially diluted by nitrogen flow in the GRWS, and then diluted by the air flow through the plant exhaust stack in the RBVS. The gaseous effluents meet the objectives of 10 CFR 50 Appendix I and 10 CFR 20 Appendix B. Small amounts of radioactive or hydrogen-bearing gaseous wastes not directly collected by the GRWS are captured by the plant heating ventilation and air conditioning (HVAC) systems. Exhaust flow from the RWBVS and RBVS are combined and monitored by the RBVS exhaust stack radiation effluent monitor prior to release to the environment (Section 11.5). Primary gaseous effluent sources, besides gaseous radioactive waste, include the CES (Section 9.3.6), RWBVS (Section 9.4.3) and other sources exhausted by the RBVS (Section 9.4.2). In addition, small releases that occur in the Turbine Generator Building from the main condenser air removal system (CARS) (Section 10.4.2) and turbine gland sealing system (Section 10.4.3) are monitored, but directly released to the environment as illustrated in Figure 11.5-1.

11.3.1 Design Bases

The GRWS serves no safety function. The GRWS is not credited for mitigation of design basis accidents and has no safe shutdown functions. General Design Criteria (GDC) 2, 3, 60, and 61 were considered in the design of the GRWS.

Consistent with GDC 2, the GRWS is designed using the guidance of RG 1.143. Consistent with GDC 3, the detonation from a hydrogen-oxygen mixture is precluded by preventing such mixtures from occurring. Consistent with GDC 60, as it relates to the GRWS, the releases of radioactive materials to the environment is controlled. Consistent with GDC 61, as it relates to the GRWS, adequate safety is ensured under normal and postulated accident conditions.

The GRWS is designed to comply with the “as low as reasonably achievable” (ALARA) philosophy of 10 CFR 20.1101(b) and the dose limits of 10 CFR 20.1301, 10 CFR 20.1302, and 10 CFR 50 Appendix I ALARA design objectives, including the effluent concentration limits of 10 CFR 20 Appendix B, Table 2 and 40 CFR 190 as implemented under 10 CFR 20.1301(e).

Consistent with 10 CFR 20.1406, the design of the GRWS includes provisions to reduce contamination of the facility and the environment, facilitate eventual decommissioning, and reduce the generation of radioactive waste.

11.3.2 System Description

The GRWS is located in the RWB and is a passive, once-through, ambient temperature charcoal delay system that receives hydrogen-bearing gas containing fission gases from the LRWS degasifier. The GRWS also receives gaseous waste inputs from the individual NuScale Power Modules (NPMs) via the CES, if high radiation is detected in the CES exhaust. The GRWS filters particulate carryover, removes moisture, delays the gas to allow radioactive decay, and conveys it to the RBVS via the RWBVS for release to the environment through the plant exhaust stack as a monitored release (See Section 11.5).

The waste gas input from the liquid radioactive waste degasifier (and potentially CES) is diluted automatically with nitrogen to maintain a hydrogen concentration of less than 1 percent. Because the waste gas input flow is not constant, nitrogen is supplied to maintain a positive GRWS pressure and a constant flow. The waste gas input into the GRWS passes through a vapor condenser package assembly that contains a waste gas cooler (cooled by chilled water) and a moisture separator. The moisture separator includes level control drain valves piped to the equipment drain sump in the radioactive waste drain system (RWDS). The drain line passes through a drain trap to prevent radioactive gas from passing to the RWDS in the event of a system failure. After the vapor condenser, the waste gas stream passes through two redundant oxygen analyzers, two hydrogen analyzers, and a manual sample port. If high oxygen levels are detected, the inlet stream to the GRWS is automatically isolated and a nitrogen purge flushes the GRWS. Termination of nitrogen flushing and restart of normal operations are manually initiated.

The waste gas then passes through a charcoal guard bed located in an ambient temperature-controlled shielded cubicle. Because the guard bed is at ambient room temperature, it warms the gas from the gas cooler (lowering its relative humidity) to improve fission gas capture efficiency in the decay beds. The guard bed also acts as a backup moisture-removal device. The guard bed includes a safety relief valve, differential pressure instrumentation, and a means to dry or replace charcoal. Charcoal drying is manually initiated by remotely-operated valves and a normally deenergized charcoal drying heater, which provide a heated nitrogen flow to the guard bed. The heated, moisture-laden nitrogen is recycled back to the inlet of the vapor condenser. The guard bed also contains a fire detector that automatically activates a nitrogen purge upon detecting a fire.

The conditioned waste gas then flows into either one of two charcoal decay beds, each decay bed consisting of four charcoal vessels connected in series. Entrance into the first vessel and exit from the last vessel is through the top of the vessel to minimize the potential of charcoal loss. Each decay bed contains activated charcoal optimized for xenon and krypton retention. Like the guard bed, the decay beds contain differential pressure instrumentation, fire detection instrumentation, safety relief valves, and the ability to either dry or replace charcoal. In addition, the decay beds contain radiation monitors that automatically isolate flow in the event of a high radiation indication.

Finally, the processed waste gas is released to the RWBVS which interfaces with the RBVS that provides the monitored effluent path to the environment. The GRWS outlet also has an offline radiation monitor with the capability to take samples prior to being sent to the ventilation systems.

The gaseous radioactive waste process design is illustrated in Figure 11.3-1a and Figure 11.3-1b.

11.3.2.1 Component Description

This section describes the key GRWS equipment. Specific component design parameters are summarized in Table 11.3-2. Design codes, standards, and materials for construction of these components are consistent with RG 1.143, Table 1.

11.3.2.1.1 Waste Gas Cooler

The waste gas cooler is a stainless steel, double-pipe heat exchanger that is designed to cool the incoming waste gas stream from the LRWS and CES. The waste gas stream (tube side) is cooled by chilled water (shell side) to condense water vapor from the gas stream to protect the charcoal beds from moisture. The shell side of the cooler is protected by a pressure relief valve.

11.3.2.1.2 Moisture Separator

The condensed water from the waste gas cooler is collected in the moisture separator, which is a stainless steel tank with level instrumentation that controls the outlet drain valve. The condensate is routed to the equipment drain waste sump in the RWDS. The drain line passes through a drain trap to prevent radioactive gases from passing to the RWDS.

11.3.2.1.3 Charcoal Guard Bed

The charcoal guard bed is an ASME Section VIII stainless steel vessel located in an ambient temperature-controlled cubicle which warms the waste gas stream, thus reducing its relative humidity. The guard bed also removes additional moisture in the waste gas stream to improve fission gas capture efficiency and protect the charcoal decay beds. The charcoal guard bed includes a safety relief valve, differential pressure instrumentation, a fire detector, and a means to dry (see charcoal drying heater) or replace the charcoal, if needed. A moisture sensor is provided downstream of the guard bed with both local and waste management control room (WMCR) indication.

11.3.2.1.4 Charcoal Decay Beds

The two charcoal decay beds each consist of four ASME Section VIII stainless steel decay vessels connected in series. The vessels contain activated charcoal to allow the waste gas radionuclides to decay sufficiently before being released. Each decay bed train is provided with a pressure relief valve, differential pressure instrumentation, and a fire detector that automatically activates a nitrogen purge. The exit of each of the two decay beds has a radiation monitor that automatically isolates the bed in the event of a high radiation signal.

11.3.2.1.5 Charcoal Drying Heater

The charcoal drying heater is a manually initiated, stainless steel electric heater that heats nitrogen gas from the nitrogen distribution system to flow through the charcoal guard bed to dry the charcoal, if needed. The heated nitrogen can also be routed to the charcoal decay beds. After exiting the guard bed or decay beds, the nitrogen is routed back to the inlet of the waste gas cooler to remove the moisture. The charcoal drying heater has a temperature controller with a high temperature cutoff. If a fire is detected in a charcoal bed, the heater is automatically deenergized.

11.3.2.1.6 Oxygen and Hydrogen Analyzers

There are a total of three independent oxygen analyzers and two hydrogen analyzers that continuously monitor the GRWS, which follows the guidance from Section 4.7 of ANSI/ANS-55.4 (Reference 11.3-2). Two redundant oxygen analyzers and two redundant hydrogen analyzers are located downstream of the moisture separator, upstream of the charcoal guard bed, and indicate and annunciate locally, in the main control room (MCR), and in the WMCR. In the event that high oxygen levels are detected (>1 percent), the system initiates an alarm locally and in both the WMCR and MCR. If the oxygen level reaches 2 percent, the inlet stream to the GRWS is automatically isolated and a nitrogen purge valve automatically opens so that the GRWS is purged with nitrogen. The hydrogen monitor ensures detection of a maximum concentration of 4 percent with notification of a high-high alarm. The notification of a high alarm at approximately one-half of the maximum concentration also ensures a local, WMCR and MCR notification. The GRWS does not automatically shutdown or purge on a high hydrogen concentration because hydrogen is expected to be in the system.

The gas analyzer instruments are designed to be non-sparking. Gas analyzers have sensor checks, functional checks, and calibrations performed in accordance with vendor recommendations.

11.3.2.2 Malfunction Analysis

A malfunction analysis is provided and summarized in Table 11.3-3. The analysis demonstrates the GRWS tolerance to various failures.

11.3.2.3 Design Safety Evaluation

The GRWS is designed for maintenance accessibility and ALARA with the following features:

- as practicable, controls and transmitters are located outside of high-radiation areas
- charcoal beds are located in separate shielded cubicles
- stainless steel process piping is butt-welded to minimize crud traps
- cubicles contain continuous airborne monitors to detect leaks from the GRWS (Section 12.3.4)

- to minimize leakage, packless metal diaphragm valves are used in the system when in contact with the waste gas stream
- pressure-retaining components comply with RG 1.143, Position 4.3

The GRWS complies with the following General Design Criteria found in 10 CFR Part 50, Appendix A:

- GDC 2 as it relates to structures and components of the GRWS using the guidance of RG 1.143 for the seismic, safety, and quality classifications
- GDC 3 as it relates to protecting the GRWS from the effects of a detonation of a hydrogen-oxygen mixture by preventing such mixtures from occurring
- GDC 60 as it relates to the design of the GRWS to control releases of radioactive gaseous effluents generated during normal reactor operations, including AOOs
- GDC 61 as it relates to radioactive waste systems being designed to provide for adequate safety under normal and postulated accident conditions, and designed with suitable shielding for radiation protection and with appropriate containment, confinement, and filtering systems

Design features are provided in accordance with the requirements of 10 CFR 20.1406 following the guidance of RG 4.21, to minimize contamination of the facility and the environment, facilitate eventual decommissioning, and minimize the generation of radioactive waste. Additional details are provided in Section 12.3.6. The principle components, piping, and valves that contain radioactive gases are designed to the seismic and quality requirements of Regulatory Guide (RG) 1.143. The design of the GRWS utilizes and conforms to the guidance provided in RG 1.143, including Branch Technical Position 11-5. The design objectives of the GRWS are:

- receive radioactive and hydrogen-bearing waste gases that originate in the reactor coolant system, and process by filtration and holdup for decay prior to releasing to the environment
- provide the capability for sampling to ensure that gaseous releases of radioactive material in gaseous effluents are ALARA
- provide the capability to collect, process, and dispose of radioactive gases generated as the result of normal operations and AOOs
- maintain a non-flammable, non-explosive gas environment
- condition gases to provide moisture and temperature conditions necessary for desired performance of charcoal decay beds
- retain off-gases for a sufficient time to allow decay of short-lived, inert fission products (xenon and krypton)
- provide monitoring and alarm functions to protect against the release of significant quantities of gaseous and particulate radioactive material to the environment
- ensure in-plant occupational exposures due to operation and maintenance of the off-gas systems are ALARA

The RBVS and RWBVS are also designed to comply with RG 1.140 as it pertains to the design, testing, and maintenance of normal ventilation exhaust system air filtration and adsorption units (see Section 9.4.2 and Section 9.4.3, respectively).

11.3.2.4 Method of Treatment

The GRWS utilizes ambient temperature charcoal beds to delay the release of radioactive gases generated from plant operations. The delay characteristics are provided in Table 11.3-1 for normal operating conditions.

The vapor condenser package reduces the moisture level of the waste gas stream prior to entering the charcoal beds. A charcoal guard bed provides another means of removing moisture from the waste gas stream to ensure proper performance of the charcoal decay beds.

The gaseous radioactive waste inlet stream is monitored for hydrogen and oxygen content to ensure a flammable mixture does not accumulate. An explosive hydrogen and oxygen mixture is prevented by maintaining hydrogen and oxygen gas concentrations below 4 percent by volume. This is accomplished by allowing only non-aerated gaseous inputs, maintaining a positive pressure with respect to the surroundings, using nitrogen as a carrier gas, and monitoring the gas stream for the presence of oxygen and hydrogen.

Once the waste gas stream exits the charcoal decay beds, the stream is conveyed to the RWBVS, where it is mixed and diluted with the normal RWBVS ventilation flow. The RWBVS outlet flow is monitored and sent to the RBVS. Releases to the environment through the plant exhaust stack are monitored.

The annual average airborne releases of radionuclides from the plant exhaust stack are determined using the methodology described in RG 1.112, as modified by TR-1116-52065 (Reference 11.3-1). The expected annual quantities of radioactive material released and expected doses to members of the public in unrestricted areas are calculated and provided in Table 11.3-5 and Table 11.3-8, respectively.

The GRWS equipment is designed to accommodate gases using the design basis source term (Section 11.1) and operating conditions that include normal operation and anticipated operational occurrences (AOOs). The system equipment is contained within the Radioactive Waste Building (RWB) with sufficient shielding to protect workers in accordance with RG 8.8. Charcoal decay beds remove radioactive iodine in the effluent stream and hold up noble gases to sufficiently reduce the activity level in the effluent stream prior to release to comply with regulatory limits.

The gaseous radioactive waste structures, systems, and components are designed in accordance with the codes and standards provided in RG 1.143, Table 1 through 4 (see Table 11.3-10). The applicable design criteria from RG 1.143, Table 2, Table 3 and Table 4 are used in the design analysis of the GRWS components. The safety classification for the GRWS components applies to components, up to and including the nearest isolation device. Design parameters of major components, including safety classification and operating conditions, are provided in Table 11.3-2.

11.3.2.5 Site-Specific Cost-Benefit Analysis

Regulatory Guide 1.110 provides guidance for complying with 10 CFR 50, Appendix I, Section II, Paragraph D, to demonstrate that the addition of items of reasonably demonstrated technology is not favorable or cost-beneficial.

COL Item 11.3-1: A COL applicant that references the NuScale Power Plant design certification will perform a site-specific cost-benefit analysis.

11.3.2.6 Mobile or Temporary Equipment

The GRWS does not employ the use of mobile or temporary equipment in the design.

11.3.2.7 Seismic Design

The gaseous radioactive waste equipment and piping are classified in accordance with RG 1.143. The RWB seismic design is described in Section 3.7.2. The structures, systems, and component classifications for the GRWS components are listed in Table 3.2-1 and Table 11.3-2. The component activity contents are shown in Section 12.2.1.

11.3.3 Radioactive Effluent Releases

The GRWS processes and releases waste gas from normal reactor operations and AOOs to the RWBVS, and the waste gas is monitored and released to the environment through the RBVS exhaust stack. Section 9.4.2 provides additional information on the plant exhaust stack. Other normal gaseous discharge pathways include the condenser air removal system and secondary system steam leaks as illustrated in Figure 11.5-1. The discharge of gaseous effluents is tabulated by isotope, pathway, and annual released activity in Table 11.3-5.

As described in Section 11.2.3, an alternate methodology to replace PWR-GALE was developed that uses first principles based calculations, combined with more recent nuclear industry experience. The calculation of gaseous effluent offsite dose consequences is consistent with methodologies presented in RG 1.112 and RG 1.109. A description of the methodology used to develop the primary and secondary coolant source terms is provided in Section 11.1. For normal effluents, the realistic coolant source terms are used and propagated through the plant systems. The major assumptions and inputs for the gaseous release methodology are listed in Table 11.3-4. From the component and airborne source terms, the normal gaseous effluent source term is determined and presented in Table 11.3-5. From the gaseous effluent source term, the offsite consequences are calculated using GASPAR II from the input values presented in Table 11.3-6. The released gaseous radioactive effluent meets the concentration limits of 10 CFR 20.1302 and the dose limits of 10 CFR 50 Appendix I. A more thorough description of this PWR-GALE replacement methodology is presented in Reference 11.3-1.

<u>Gaseous Radioactive Waste System</u>

The releases from the GRWS are determined by the release rate from the last decay bed vessel. Gaseous radioactivity from the degasifier is routed to the GRWS through the guard bed and one train of decay bed vessels. As the decayed gaseous waste exits the GRWS, it enters the RWBVS, and then the RBVS, and out the plant exhaust stack.

The other releases from the plant exhaust stack are determined by combining the calculated gaseous effluent contributions from reactor pool evaporation, primary coolant system leaks and containment vessel (CNV) leakage during an AOO.

Reactor Pool Evaporation

The concentration of radionuclides in the reactor pool water spikes during refueling events and then decreases as the water is processed by the PCUS. As a result, the airborne concentration in the airspace above the reactor pool water exhibits a similar behavior. While the peak activity concentrations are used for Reactor Building (RXB) ventilation design purposes, the gaseous effluent from reactor pool evaporation is determined based on an annual time-weighted average reactor pool water source term, pool water evaporation rate, airspace ventilation rate, and ventilation system filter efficiencies.

Primary System Coolant Leaks

Another contribution to the RXB atmosphere radionuclide concentration is primary coolant leaks into the RXB originating from chemical and volume control system (CVCS) components. The portion of primary coolant leakage into the RXB that flashes to steam is captured by RBVS, filtered, and exhausted to the plant exhaust stack. The major input parameters are found in Table 11.3-4.

Containment Leakage AOO

The input values from NUREG-0017, many of which account for AOOs during plant operations, are used whenever appropriate (Section 4.5 of Reference 11.3-1). In addition to these AOOs, a design-specific AOO is included that results from an inadvertent actuation of the emergency core cooling system. This event will pressurize the CNV, assuming to result in a CNV leak of noble gases at the CNV design leak rate for the time it takes to depressurize the primary system. For purposes of determining the normal annual effluents, this AOO is assumed to occur once per year. The other major parameter assumptions are in Table 11.3-4.

Turbine Generator Building

Each NPM has its own secondary coolant system with independent condenser air removal systems. The releases from the CARS assume that the noble gases in the main steam system are completely removed by the CARS. The release of halogens from the main steam system through the CARS is calculated using a normalized effective release rate multiplied by the secondary coolant realistic activity concentration found in Table 11.1-13. The condenser air removal system's exhaust releases halogens at a thermal power scaled release rate, shown in Table 11.3-4.

Gaseous releases from secondary steam leaks are also assumed to occur. The parameters used to calculate the gaseous effluents from secondary side leaks are found in Table 11.3-4.

Pool Surge Control System Storage Tank Vent

It is assumed, for the normal effluent releases through reactor pool water evaporation discussed above, the total radionuclide inventory in the UHS pool water is there for the entire year, thus maximizing the pool evaporation gaseous releases. During the short periods of time, during refueling outages, a portion of the pool water from the dry dock is moved to the enclosed and vented pool surge control system storage tank, reducing the releases by evaporation. Therefore, gaseous releases from the pool surge control system storage tank vent are not explicitly modeled for annual effluents, but are conservatively accounted for in the pool water evaporation releases.

The results of the radioactive effluent calculation are tabulated in Table 11.3-5 and demonstrate compliance with the limits from 10 CFR 20, Appendix B, Table 2. The comparison demonstrates that the overall expected gaseous releases are within the release limits.

The maximum individual doses at the exclusion area boundary are calculated using the GASPAR II Code. The input parameters for the calculation are tabulated in Table 11.3-6. The resultant doses are tabulated in Table 11.3-8 and demonstrates compliance with the limits of 10 CFR 50 Appendix I.

COL Item 11.3-2: A COL applicant that references the NuScale Power Plant design certification will calculate doses to members of the public using the site-specific parameters, compare those gaseous effluent doses to the numerical design objectives of 10 CFR 50, Appendix I, and comply with the requirements of 10 CFR 20.1302 and 40 CFR 190.

11.3.3.1 Radioactive Effluent Releases and Dose Calculation due to Gaseous Radioactive Waste System Leak or Failure

The GRWS is designed to minimize the potential for both air in-leakage and system process gas out-leakage. However, failure of the GRWS is postulated, which results in a release of gaseous radionuclides. The analysis of a GRWS leak or failure follows the guidance of Branch Technical Position 11-5 and demonstrates compliance with regulatory limits. The dose consequence analysis evaluates a postulated event in which the GRWS fails, and the charcoal guard bed and charcoal decay beds are bypassed for one hour. The analysis used in determining the radionuclide content of the effluents assumes that one percent of the operating fission product inventory in the core is released to the primary coolant. This event releases the inventory of gaseous radionuclides that are transferred from 12 operating NPMs to a degasifier as a ground-level release. The release source term is found in Table 11.3-9. The dose consequences are calculated using the Radionuclide Transport and Removal and Dose (RADTRAD) code using the two-hour exclusion area boundary atmospheric dispersion factor from Section 2.3.4. The resultant offsite doses are also presented in Table 11.3-9.

COL Item 11.3-3: A COL applicant that references the NuScale Power Plant design certification will perform an analysis in accordance with Branch Technical Position 11-5 using the site-specific parameters.

11.3.4 Ventilation Systems

Radioactive gases are potentially present in the RXB and RWB due to evaporation or leakages. The design of the ventilation systems for normal operation are in accordance with RG 1.140, and are described in Section 9.4.2 and Section 9.4.3. Airborne radioactivity levels are discussed in Section 12.2.2.

The RWBVS and RBVS ventilation flow provides dilution for GRWS releases in the plant exhaust stack, where the releases are monitored per GDC 64 as detailed in Section 11.5. The GRWS discharge isolation valve upstream of the discharge radiation monitor closes upon a loss of RBVS or RWBVS flow. This prevents the release of concentrated gaseous waste and the potential accumulation of hydrogen gas. Equipment cubicles are ventilated to reduce the accumulation of airborne radioactive materials transported to the atmosphere from equipment in a radiological controlled area, with provisions for radiation monitoring and hydrogen monitoring, where necessary.

11.3.5 Testing and Inspection Requirements

The GRWS and individual structures, systems, and components, are tested in conformance with RG 1.143 as described in Section 14.2.

During normal operation, the hydrogen and oxygen analyzers, process equipment, and instrument monitor channels are periodically tested and calibrated to ensure that the flammable gas mixture is below the flammability limit.

11.3.6 Instrumentation and Controls

Table 11.3-7 provides a listing of the major instrumentation in the GRWS. The instruments that provide automated functions in the GRWS include the following.

Waste Gas Cooler Moisture Separator Level

Monitors the water level in the drain tank and opens the tank's drain valve to route the water to the RWDS.

Hydrogen and Oxygen Gas Analyzers

These analyzers are described in Section 11.3.2.1.6.

Fire Detectors

Fire detectors are provided for each of the charcoal beds to indicate the presence of a fire. If a fire is detected in a guard or decay bed, the GRWS waste gas inlet valve is automatically closed and the nitrogen supply valve is automatically opened to the associated charcoal bed.

Waste Gas Flow Instrument

The flow of the waste gas stream is measured downstream of the moisture separator and downstream of the decay beds in the outlet line. Nitrogen is used to maintain a minimum flow through the charcoal beds.

Moisture Instrument

A moisture level instrument is provided for the waste gas stream at the outlet of the guard bed. If high moisture is detected, the waste gas inlet valve to the GRWS is closed to stop the system flow.

Charcoal Bed Process Radiation Monitors

Process radiation monitors are provided at the outlet of each of the two charcoal decay beds. If high radiation is detected, the charcoal bed outlet valve is automatically closed.

Gaseous Radioactive Waste System Outlet Process Radiation Monitor

A process radiation monitor is provided at the outlet of the GRWS. If high radiation is detected, the GRWS outlet valve is closed to stop system flow to the RWBVS.

Cubicle Area Airborne Radiation Detectors

Area airborne radiation detectors are located in each of the charcoal bed cubicles. If high radiation is detected, the waste gas inlet valve to the GRWS is closed and the nitrogen purge valve is opened.

11.3.7 References

11.3-1 NuScale Power, LLC, "Effluent Release (GALE Replacement) Methodology and Results," TR-1116-52065, Rev. 1.

11.3-2 American National Standards Institute/American Nuclear Society, "Gaseous Radioactive Waste Processing Systems for Light Water Reactors," ANSI/ANS 55.4-1993, LaGrange Park, IL.

Table 11.3-1: Gaseous Radioactive Waste System Design Parameters

Parameter	Nominal Value
Xenon delay	45 days
Krypton delay	1.9 days
Dynamic adsorption coefficient (K_d) for xenon	1400 cm^3/gram
Dynamic adsorption coefficient (K_d) for krypton	60 cm^3/gram
Maximum gas waste stream temperature	212 °F
Activated carbon operating temperature	77 °F
Activated carbon operating dewpoint	45 °F
Gas flow rate	1.56 scfm
Charcoal particle size	6-12 mesh with 90-100% retention

Table 11.3-2: Major Equipment Design Parameters

Equipment / Parameter	Description / Value
Gas Cooler	
Quantity	2
Type	Double pipe
Design pressure (tube / shell)	15 / 100 psig
Design temperature (tube / shell)	250 / 200°F
Design flow rate (tube / shell)	1.56 scfm / 0.29 gpm
Temperature inlet (tube / shell)	212 / 40 °F
Temperature outlet (tube / shell)	45 / 45 °F
Material	Stainless steel
RG 1.143 safety classification	RW-IIc
Table for Assumed Radioactive Content	Table 11.3-11
Charcoal Guard Bed	
Quantity	1
Type	Cylindrical pressure vessel
Nominal volume	10.2 ft^3
Design pressure	15 psig
Design temperature	250 °F
Design flow rate	1.56 scfm
Material	Stainless steel
RG 1.143 safety classification	RW-IIa
Table for Assumed Radioactive Content	Table 12.2-16
Charcoal Decay Bed Vessel	
Quantity	8 (4 in each of two trains)
Type	Cylindrical pressure vessel
Nominal volume	147 ft^3 per train
Mass of charcoal	4,600 lb per train
Design pressure	15 psig
Design temperature	250 °F
Design flow rate	1.56 scfm
Material	Stainless steel
RG 1.143 safety classification	RW-IIa
Table for Assumed Radioactive Content	Table 12.2-16
Moisture Separator	
Quantity	2
Type	Vertical cylinder
Nominal volume	1 gallon
Design pressure	15 psig
Design temperature	250 °F
Material	Stainless steel
RG 1.143 safety classification	RW-IIc
Table for Assumed Radioactive Content	Table 11.3-11
Charcoal Drying Heater	
Quantity	1
Type	Electric
Duty	0.5 KW
Design flow rate	76 lb/hr / 16 scfm (nitrogen)
Temperature inlet	60 °F
Temperature outlet	150 °F

Table 11.3-3: Gaseous Radioactive Waste System Equipment Malfunction Analysis

Equipment Item	Malfunction	Results (Consequences)	Mitigating or Alternate Action
Gas cooler	Cooler failure	The gas cooler is part of the vapor condenser package assembly and there are two vapor condenser package assemblies in the GRWS. Failure of a gas cooler causes ineffective removal of moisture in the gas.	If one of the gas coolers fails, the gas from the degasifier can be manually diverted to the other gas cooler. This allows processing to continue.
Moisture separator	Separator failure	The moisture separator is part of the vapor condenser package and located downstream of the gas cooler. Failure of a moisture separator causes ineffective removal of moisture in the gas.	Manual valve realignment can divert the saturated gas to the other moisture separator.
Charcoal drying heater	Heater failure	The purpose of the heater is to heat nitrogen used to periodically dry the charcoal in the charcoal guard bed by allowing hot nitrogen to flow through the charcoal. There is no immediate impact to the GRWS operation if the heater fails. The downstream charcoal decay bed efficiency may be lowered.	The charcoal decay beds may be aligned in series to improve the decontamination factor if the guard bed is saturated with moisture.
Charcoal guard bed	Guard bed failure	There is only one charcoal guard bed in the system. If the guard bed fails, the fission gas removal efficiency is potentially lowered.	Operation continues by sending the gas to be treated in one of the two charcoal decay beds, which is located downstream of the guard bed. The charcoal decay beds may be aligned in series to improve the decontamination factor.
Charcoal decay beds	Decay bed failure	There are two sets of redundant charcoal decay beds. Each decay bed consists of four decay vessels connected in series. Failure of one set of decay beds decreases removal efficiency of xenon and krypton.	If one set of decay beds fails, the gas can be diverted to the second set of decay beds for continued processing.
Pressure boundary	Gas leaks	Waste gas is released to the RWB.	Very small gas leaks can be detected by the area airborne monitors. In addition, flange joint color coded telltale leak indicators are provided on mechanical joints that are identifiable leak sensors.
Oxygen monitor	Monitor failure	Monitoring capability is lost for detecting oxygen concentration.	The redundant oxygen analyzer monitors the oxygen concentration.
Hydrogen monitor	Monitor failure	Monitoring capability is lost for detecting hydrogen concentration.	The redundant hydrogen analyzer monitors the hydrogen concentration.
Radiation monitor	Monitor failure	Capability is lost for monitoring waste gas in the GRWS discharge header to the RWBVS.	Waste gas stream is monitored for radiation at each of the decay bed outlets (2 sets of decay beds). Gaseous effluents are monitored as they exit the RBVS exhaust stack to the environment.

Table 11.3-4: Gaseous Effluent Release Calculation Inputs

NuScale Effluent Source Term Model Assumption	Value (1 NPM)	Value (12 NPMs)
Degasifier partition fractions: - Noble gases - Halogens	 1 0.5	 1 0.5
Reactor pool evaporation rate	-	2100 lb/hour
Pool evaporation partition fractions: - Halogens (except iodine) - Iodine - Cs, Rb, particulates - Gases and tritium	 0.01 0.0005 0.005 1	 0.01 0.0005 0.005 1
Steam generator partition coefficient	1	1
HEPA filter particulate efficiency	99%	99%
Primary coolant system leakrate	7.5 lb/day	90 lb/day
Primary coolant leak flashing fraction	0.4	0.4
Primary coolant leak partition fractions: - Halogens - Cs, Rb, particulates - Gases and tritium	 0.01 0.005 1	 0.01 0.005 1
Secondary coolant system steam leakrate	80 lb/day	960 lb/day
Condenser air removal normalized iodine release rate	80 Ci/yr/μCi/gm	960 Ci/yr/μCi/gm
Containment vessel design leakrate	0.2 weight%/day	0.2 weight%/day
Containment depressurization time	30 hours	30 hours

Table 11.3-5: Gaseous Estimated Discharge for Normal Effluents

Nuclide	GRWS (Ci/yr)	Pool Evaporation (Ci/yr)	AOO Gas Leakage (Ci/yr)	Primary Coolant Leaks (Ci/yr)	Plant Exhaust Stack Total (Ci/yr)	Secondary Steam Leaks (Ci/yr)	Condenser Air Removal System (Ci/yr)	Total TGB Releases (Ci/yr)	Total Gaseous Effluent Concentration at Site Boundary (μCi/ml)	10 CFR 20 Appendix B Limits (μCi/ml)	Fraction of Limit
Kr83m	6.94E-07	7.52E-06	5.73E-05	7.35E-03	7.41E-03	5.18E-07	3.44E-03	3.44E-03	4.95E-15	5.00E-05	9.91E-11
Kr85m	2.10E-03	-	2.41E-04	3.09E-02	3.33E-02	2.18E-06	1.45E-02	1.45E-02	2.18E-14	1.00E-07	2.18E-07
Kr85	2.11E+02	-	2.09E-02	2.68E+00	2.14E+02	1.89E-04	1.26E+00	1.26E+00	9.83E-11	7.00E-07	1.40E-04
Kr87	2.14E-11	-	1.32E-04	1.69E-02	1.70E-02	1.19E-06	7.92E-03	7.92E-03	1.14E-14	2.00E-08	5.69E-07
Kr88	5.69E-05	-	3.84E-04	4.92E-02	4.96E-02	3.47E-06	2.31E-02	2.31E-02	3.32E-14	9.00E-09	3.69E-06
Kr89	-	-	8.78E-06	1.13E-03	1.13E-03	7.93E-08	5.27E-04	5.27E-04	7.58E-16	-	
Xe131m	6.29E-01	8.02E-01	8.69E-04	1.11E-01	1.54E+00	7.84E-06	5.22E-02	5.22E-02	7.28E-13	2.00E-06	3.64E-07
Xe133m	5.92E-06	1.29E+00	8.40E-04	1.08E-01	1.40E+00	7.58E-06	5.04E-02	5.05E-02	6.60E-13	6.00E-07	1.10E-06
Xe133	1.66E+00	1.82E+01	6.15E-02	7.88E+00	2.78E+01	5.55E-04	3.69E+00	3.69E+00	1.44E-11	5.00E-07	2.88E-05
Xe135m	6.58E-05	3.64E-01	8.16E-05	1.05E-02	3.75E-01	7.37E-07	4.90E-03	4.90E-03	1.73E-13	4.00E-08	4.33E-06
Xe135	2.93E-05	2.02E-01	2.14E-03	2.74E-01	4.78E-01	1.93E-05	1.28E-01	1.28E-01	2.77E-13	7.00E-08	3.95E-06
Xe137	-	-	2.82E-05	3.61E-03	3.64E-03	2.54E-07	1.69E-03	1.69E-03	2.43E-15	-	
Xe138	-	-	9.64E-05	1.24E-02	1.25E-02	8.71E-07	5.79E-03	5.79E-03	8.33E-15	2.00E-08	4.16E-07
Br82	8.68E-09	1.08E-08	-	8.15E-07	8.34E-07	1.45E-08	3.64E-09	1.81E-08	3.89E-19	5.00E-09	7.78E-11
Br83	4.97E-08	3.52E-13	-	4.67E-06	4.72E-06	8.05E-08	2.03E-08	1.01E-07	2.20E-18	9.00E-08	2.44E-11
Br84	2.31E-08	-	-	2.17E-06	2.19E-06	3.39E-08	8.53E-09	4.25E-08	1.02E-18	8.00E-08	1.28E-11
Br85	2.79E-09	-	-	2.62E-07	2.65E-07	1.85E-09	4.66E-10	2.32E-09	1.22E-19	-	
I129	2.15E-13	6.46E-13	-	2.01E-11	2.10E-11	3.58E-13	9.01E-14	4.49E-13	9.79E-24	4.00E-11	2.45E-13
I130	7.01E-08	1.69E-08	-	6.58E-06	6.67E-06	1.16E-07	2.93E-08	1.46E-07	3.11E-18	3.00E-09	1.04E-09
I131	1.80E-06	4.61E-04	-	1.69E-04	6.32E-04	3.01E-06	7.57E-07	3.77E-06	2.90E-16	2.00E-10	1.45E-06
I132	8.23E-07	1.16E-06	-	7.72E-05	7.92E-05	1.33E-06	3.35E-07	1.67E-06	3.69E-17	2.00E-08	1.85E-09
I133	2.72E-06	5.48E-05	-	2.55E-04	3.13E-04	4.52E-06	1.14E-06	5.66E-06	1.45E-16	1.00E-09	1.45E-07
I134	4.84E-07	7.00E-22	-	4.54E-05	4.59E-05	7.46E-07	1.88E-07	9.33E-07	2.14E-17	6.00E-08	3.56E-10
I135	1.71E-06	4.49E-08	-	1.61E-04	1.62E-04	2.83E-06	7.11E-07	3.54E-06	7.57E-17	6.00E-09	1.26E-08
Rb86m	-	-	-	9.72E-13	9.72E-13	6.61E-13	-	6.61E-13	7.45E-25	-	
Rb86	-	6.42E-09	-	5.75E-09	1.22E-08	2.25E-08	-	2.25E-08	1.58E-20	1.00E-09	1.58E-11
Rb88	-	-	-	9.83E-07	9.83E-07	3.02E-06	-	3.02E-06	1.83E-18	9.00E-08	2.03E-11
Rb89	-	-	-	4.51E-08	4.51E-08	1.34E-07	-	1.34E-07	8.17E-20	2.00E-07	4.09E-13

Table 11.3-5: Gaseous Estimated Discharge for Normal Effluents (Continued)

Nuclide	GRWS (Ci/yr)	Pool Evaporation (Ci/yr)	AOO Gas Leakage (Ci/yr)	Primary Coolant Leaks (Ci/yr)	Plant Exhaust Stack Total (Ci/yr)	Secondary Steam Leaks (Ci/yr)	Condenser Air Removal System (Ci/yr)	Total TGB Releases (Ci/yr)	Total Gaseous Effluent Concentration at Site Boundary (µCi/ml)	10 CFR 20 Appendix B Limits (µCi/ml)	Fraction of Limit
Cs132	-	1.09E-10	-	1.11E-10	2.19E-10	4.34E-10	-	4.34E-10	2.98E-22	6.00E-09	4.97E-14
Cs134	-	1.19E-06	-	9.93E-07	2.18E-06	3.88E-06	-	3.88E-06	2.77E-18	2.00E-10	1.38E-08
Cs135m	-	5.12E-25	-	7.54E-10	7.54E-10	2.70E-09	-	2.70E-09	1.58E-21	3.00E-07	5.26E-15
Cs136	-	2.28E-07	-	2.10E-07	4.39E-07	8.23E-07	-	8.23E-07	5.76E-19	9.00E-10	6.40E-10
Cs137	-	7.28E-07	-	6.08E-07	1.34E-06	2.38E-06	-	2.38E-06	1.70E-18	2.00E-10	8.48E-09
Cs138	-	-	-	3.61E-07	3.61E-07	1.23E-06	-	1.23E-06	7.27E-19	8.00E-08	9.09E-12
P32	-	9.76E-15	-	1.65E-14	2.62E-14	5.85E-14	-	5.85E-14	-	5.00E-10	7.73E-17
Co57	-	7.93E-17	-	1.23E-16	2.02E-16	4.37E-16	-	4.37E-16	-	9.00E-10	3.24E-19
Sr89	-	4.71E-10	-	7.34E-10	1.20E-09	2.61E-09	-	2.61E-09	1.74E-21	2.00E-10	8.70E-12
Sr90	-	1.07E-10	-	1.65E-10	2.72E-10	5.87E-10	-	5.87E-10	3.92E-22	6.00E-12	6.53E-11
Sr91	-	9.74E-12	-	3.82E-10	3.91E-10	1.35E-09	-	1.35E-09	7.93E-22	5.00E-09	1.59E-13
Sr92	-	1.10E-15	-	2.04E-10	2.04E-10	7.07E-10	-	7.07E-10	4.15E-22	9.00E-09	4.62E-14
Y90	-	5.73E-11	-	4.00E-11	9.73E-11	1.42E-10	-	1.42E-10	1.09E-22	9.00E-10	1.21E-13
Y91m	-	6.21E-12	-	2.04E-10	2.11E-10	6.68E-10	-	6.68E-10	4.01E-22	2.00E-07	2.00E-15
Y91	-	6.85E-11	-	1.07E-10	1.75E-10	3.79E-10	-	3.79E-10	2.53E-22	2.00E-10	1.26E-12
Y92	-	7.45E-14	-	1.73E-10	1.73E-10	6.04E-10	-	6.04E-10	3.55E-22	1.00E-08	3.55E-14
Y93	-	2.47E-12	-	8.14E-11	8.39E-11	2.87E-10	-	2.87E-10	1.69E-22	3.00E-09	5.65E-14
Zr97	-	1.21E-11	-	1.20E-10	1.32E-10	4.25E-10	-	4.25E-10	2.54E-22	2.00E-09	1.27E-13
Nb95	-	4.52E-07	-	3.10E-10	4.53E-07	1.10E-09	-	1.10E-09	2.07E-19	2.00E-09	1.04E-10
Mo99	-	8.71E-08	-	2.15E-07	3.02E-07	7.65E-07	-	7.65E-07	4.87E-19	2.00E-09	2.43E-10
Mo101	-	-	-	8.13E-09	8.13E-09	2.22E-08	-	2.22E-08	1.38E-20	2.00E-07	6.92E-14
Tc99m	-	8.40E-08	-	1.99E-07	2.83E-07	7.00E-07	-	7.00E-07	4.49E-19	2.00E-07	2.24E-12
Tc99	-	4.00E-12	-	6.16E-12	1.02E-11	2.19E-11	-	2.19E-11	1.46E-23	8.00E-09	1.83E-15
Ru103	-	1.29E-10	-	2.06E-10	3.35E-10	7.32E-10	-	7.32E-10	4.87E-22	9.00E-10	5.41E-13
Ru105	-	3.95E-14	-	6.76E-11	6.77E-11	2.37E-10	-	2.37E-10	1.39E-22	2.00E-08	6.95E-15
Ru106	-	8.63E-11	-	1.33E-10	2.20E-10	4.75E-10	-	4.75E-10	3.17E-22	2.00E-11	1.58E-11
Rh103m	-	1.28E-10	-	2.03E-10	3.31E-10	6.71E-10	-	6.71E-10	4.57E-22	2.00E-06	2.29E-16
Rh105	-	4.13E-11	-	1.44E-10	1.85E-10	5.10E-10	-	5.10E-10	3.17E-22	8.00E-09	3.97E-14
Rh106	-	8.63E-11	-	1.33E-10	2.20E-10	4.81E-11	-	4.81E-11	1.22E-22	-	

Table 11.3-5: Gaseous Estimated Discharge for Normal Effluents (Continued)

Nuclide	GRWS (Ci/yr)	Pool Evaporation (Ci/yr)	AOO Gas Leakage (Ci/yr)	Primary Coolant Leaks (Ci/yr)	Plant Exhaust Stack Total (Ci/yr)	Secondary Steam Leaks (Ci/yr)	Condenser Air Removal System (Ci/yr)	Total TGB Releases (Ci/yr)	Total Gaseous Effluent Concentration at Site Boundary (μCi/ml)	10 CFR 20 Appendix B Limits (μCi/ml)	Fraction of Limit
Ag110	-	5.71E-07	-	9.50E-10	5.72E-07	2.88E-10	-	2.88E-10	2.61E-19	-	
Sb124	-	1.93E-13	-	3.03E-13	4.96E-13	1.08E-12	-	1.08E-12	7.19E-25	3.00E-10	2.40E-15
Sb125	-	1.73E-12	-	2.67E-12	4.40E-12	9.50E-12	-	9.50E-12	6.35E-24	7.00E-10	9.07E-15
Sb127	-	5.37E-12	-	1.16E-11	1.69E-11	4.12E-11	-	4.12E-11	2.65E-23	1.00E-09	2.65E-14
Sb129	-	7.81E-15	-	1.42E-11	1.43E-11	4.99E-11	-	4.99E-11	2.93E-23	1.00E-08	2.93E-15
Te125m	-	2.49E-10	-	3.92E-10	6.41E-10	1.40E-09	-	1.40E-09	9.29E-22	1.00E-09	9.29E-13
Te127m	-	8.11E-10	-	1.26E-09	2.07E-09	4.49E-09	-	4.49E-09	3.00E-21	4.00E-10	7.49E-12
Te127	-	8.86E-10	-	5.01E-09	5.90E-09	1.77E-08	-	1.77E-08	1.08E-20	2.00E-08	5.38E-13
Te129m	-	2.26E-09	-	3.62E-09	5.88E-09	1.29E-08	-	1.29E-08	8.56E-21	3.00E-10	2.85E-11
Te129	-	1.43E-09	-	5.12E-09	6.55E-09	1.71E-08	-	1.71E-08	1.08E-20	9.00E-08	1.20E-13
Te131m	-	2.72E-09	-	1.18E-08	1.46E-08	4.20E-08	-	4.20E-08	2.58E-20	1.00E-09	2.58E-11
Te131	-	6.12E-10	-	5.83E-09	6.44E-09	1.76E-08	-	1.76E-08	1.10E-20	2.00E-08	5.49E-13
Te132	-	3.73E-08	-	8.62E-08	1.24E-07	3.06E-07	-	3.06E-07	1.96E-19	9.00E-10	2.18E-10
Te133m	-	1.23E-23	-	7.35E-09	7.35E-09	2.42E-08	-	2.42E-08	1.44E-20	2.00E-08	7.21E-13
Te134	-	-	-	1.05E-08	1.05E-08	3.37E-08	-	3.37E-08	2.02E-20	7.00E-08	2.88E-13
Ba137m	-	6.87E-07	-	5.71E-07	1.26E-06	7.45E-07	-	7.45E-07	9.14E-19	-	
Ba139	-	2.28E-20	-	1.95E-10	1.95E-10	6.58E-10	-	6.58E-10	3.89E-22	4.00E-08	9.73E-15
Ba140	-	6.22E-10	-	1.06E-09	1.68E-09	3.77E-09	-	3.77E-09	2.49E-21	2.00E-09	1.24E-12
La140	-	4.43E-10	-	3.07E-10	7.50E-10	1.09E-09	-	1.09E-09	8.41E-22	2.00E-09	4.20E-13
La141	-	1.39E-14	-	6.04E-11	6.04E-11	2.11E-10	-	2.11E-10	1.24E-22	1.00E-08	1.24E-14
La142	-	2.43E-20	-	2.88E-11	2.88E-11	9.79E-11	-	9.79E-11	5.78E-23	3.00E-08	1.93E-15
Ce141	-	1.02E-10	-	1.63E-10	2.65E-10	5.81E-10	-	5.81E-10	3.86E-22	8.00E-10	4.83E-13
Ce143	-	3.13E-11	-	1.24E-10	1.55E-10	4.39E-10	-	4.39E-10	2.71E-22	2.00E-09	1.36E-13
Ce144	-	8.87E-11	-	1.37E-10	2.26E-10	4.88E-10	-	4.88E-10	3.26E-22	2.00E-11	1.63E-11
Pr143	-	9.05E-11	-	1.45E-10	2.36E-10	5.17E-10	-	5.17E-10	3.43E-22	9.00E-10	3.82E-13
Pr144	-	8.79E-11	-	1.36E-10	2.24E-10	3.86E-10	-	3.86E-10	2.78E-22	2.00E-07	1.39E-15
Np239	-	9.72E-10	-	2.60E-09	3.57E-09	9.23E-09	-	9.23E-09	5.84E-21	3.00E-09	1.95E-12
Na24	-	2.20E-07	-	2.73E-06	2.95E-06	9.66E-06	-	9.66E-06	5.75E-18	7.00E-09	8.22E-10
Cr51	-	9.63E-05	-	1.55E-07	9.65E-05	5.53E-07	-	5.53E-07	4.43E-17	3.00E-08	1.48E-09

Table 11.3-5: Gaseous Estimated Discharge for Normal Effluents (Continued)

Nuclide	GRWS (Ci/yr)	Pool Evaporation (Ci/yr)	AOO Gas Leakage (Ci/yr)	Primary Coolant Leaks (Ci/yr)	Plant Exhaust Stack Total (Ci/yr)	Secondary Steam Leaks (Ci/yr)	Condenser Air Removal System (Ci/yr)	Total TGB Releases (Ci/yr)	Total Gaseous Effluent Concentration at Site Boundary (µCi/ml)	10 CFR 20 Appendix B Limits (µCi/ml)	Fraction of Limit
Mn54	-	5.17E-05	-	7.99E-08	5.18E-05	2.84E-07	-	2.84E-07	2.38E-17	1.00E-09	2.38E-08
Fe55	-	3.88E-05	-	5.99E-08	3.89E-05	2.13E-07	-	2.13E-07	1.79E-17	3.00E-09	5.95E-09
Fe59	-	9.46E-06	-	1.50E-08	9.48E-06	5.34E-08	-	5.34E-08	4.35E-18	5.00E-10	8.70E-09
Co58	-	1.47E-03	-	2.30E-07	1.47E-03	8.18E-07	-	8.18E-07	6.70E-16	1.00E-09	6.70E-07
Co60	-	1.72E-05	-	2.64E-08	1.72E-05	9.41E-08	-	9.41E-08	7.89E-18	5.00E-11	1.58E-07
Ni63	-	8.58E-06	-	1.32E-08	8.59E-06	4.70E-08	-	4.70E-08	3.94E-18	2.00E-09	1.97E-09
Zn65	-	1.64E-05	-	2.54E-08	1.65E-05	9.05E-08	-	9.05E-08	7.56E-18	4.00E-10	1.89E-08
Zr95	-	1.24E-05	-	1.95E-08	1.24E-05	6.93E-08	-	6.93E-08	5.70E-18	4.00E-10	1.43E-08
Ag110m	-	4.20E-05	-	6.50E-08	4.21E-05	2.31E-07	-	2.31E-07	1.93E-17	1.00E-10	1.93E-07
W187	-	2.43E-05	-	1.39E-07	2.44E-05	4.94E-07	-	4.94E-07	1.14E-17	1.00E-08	1.14E-09
H3	-	6.96E+02	-	5.93E+00	7.01E+02	6.92E+00	-	6.92E+00	3.23E-10	1.00E-07	3.23E-03
C14	2.63E-01	3.01E-03	-	1.33E-03	2.67E-01	2.37E-07	-	2.37E-07	1.22E-13	3.00E-09	4.06E-05
Ar41	1.19E+01	-	1.61E-02	2.07E+00	1.39E+01	1.46E-04	9.69E-01	9.69E-01	6.81E-12	1.00E-08	6.81E-04
Total	**2.26E+02**	**7.16E+02**	**1.03E-01**	**1.92E+01**	**9.62E+02**	**6.92E+00**	**6.21E+00**	**1.31E+01**	**4.45E-10**	**5.84E-05**	**4.14E-03**

Note: The X/Q that was used to calculate the site boundary concentrations is provided in Table 11.3-6

Table 11.3-6: GASPAR Code Input Parameter Values

Parameter	Value
Routine release χ/Q (undepleted/no decay)	Table 2.0-1
Routine release D/Q	Table 2.0-1
Milk animal	Goat
Midpoint of plant life	20 yrs
Fraction of year that leafy vegetables are grown	1.0
Fraction of year that milk cows are in pasture	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	8.0 gram/m^3
Fraction of year that beef cattle are in pasture	1.0
Fraction of beef cattle feed intake that is from pasture while the cattle are on pasture	1.0
Source term	Table 11.3-5

Table 11.3-7: Instrument List

Type	Location	Indication	
		Local	Control Room
Temperature	Outlet from vapor condenser package	No	WMCR
Temperature	Outlet from the three charcoal beds	No	WMCR
Temperature	Outlet from the charcoal drying heater	No	WMCR
Pressure	Inlet to the system upstream of the vapor condenser packages	Yes	WMCR
Pressure	Inlet to the charcoal guard bed	Yes	WMCR
Pressure	Inlet to the charcoal decay beds	Yes	WMCR
Pressure	Outlet from the charcoal decay beds	Yes	WMCR
Pressure differential	Across the charcoal guard bed	Yes	WMCR
Pressure differential	Across charcoal decay beds	Yes	WMCR
Level	Moisture separators	Yes	WMCR
Flow	Inlet to charcoal guard bed	Yes	WMCR
Flow	Inlet from nitrogen distribution system	Yes	WMCR
Flow	Outlet from system to RWBVS	Yes	WMCR
Flow	Outlet from the charcoal drying heater	Yes	WMCR
Oxygen	Inlet to charcoal guard bed	Yes	WMCR MCR
Oxygen	Outlet from system to RWBVS	Yes	WMCR
Hydrogen	Inlet to charcoal guard bed	Yes	WMCR MCR
Moisture	Outlet from charcoal guard bed	Yes	WMCR
Radiation	Outlet from the charcoal decay beds	Yes	WMCR MCR
Radiation	Outlet from system to RWBVS	Yes	WMCR MCR
Fire detection	All GRW charcoal beds	No	WMCR MCR

Table 11.3-8: Gaseous Effluent Dose Results for 10 CFR 50 Appendix I

Type of Dose	Dose Estimate
Beta Dose Air (mrad/yr)	0.2
Gamma Dose Air (mrad/yr)	0.07

PATHWAY	T.BODY	GI-TRACT	BONE	LIVER	KIDNEY	THYROID	LUNG	SKIN
Plume/ Ground	0.05							0.22
VEGETABLE								
ADULT		4.51E-01	1.11E-01	4.44E-01	4.44E-01	6.03E-01	4.43E-01	
TEEN		5.25E-01	1.81E-01	5.19E-01	5.19E-01	7.18E-01	5.17E-01	
CHILD		8.39E-01	4.35E-01	8.37E-01	8.36E-01	1.21E+00	8.34E-01	
MEAT								
ADULT		7.15E-02	4.11E-02	6.90E-02	6.88E-02	7.59E-02	6.87E-02	
TEEN		4.44E-02	3.47E-02	4.32E-02	4.30E-02	4.82E-02	4.30E-02	
CHILD		5.74E-02	6.53E-02	5.70E-02	5.68E-02	6.45E-02	5.67E-02	
GOAT MILK								
ADULT		2.99E-01	4.59E-02	3.01E-01	3.00E-01	5.37E-01	2.99E-01	
TEEN		3.94E-01	8.46E-02	3.97E-01	3.97E-01	7.71E-01	3.94E-01	
CHILD		6.39E-01	2.08E-01	6.44E-01	6.43E-01	1.38E+00	6.38E-01	
INFANT		9.92E-01	4.06E-01	1.00E+00	1.00E+00	2.80E+00	9.92E-01	
INHALATION								
ADULT		2.32E-01	1.40E-05	2.32E-01	2.32E-01	2.36E-01	2.33E-01	
TEEN		2.34E-01	1.94E-05	2.34E-01	2.34E-01	2.39E-01	2.36E-01	
CHILD		2.07E-01	2.62E-05	2.07E-01	2.07E-01	2.12E-01	2.08E-01	
INFANT		1.19E-01	1.82E-05	1.19E-01	1.19E-01	1.24E-01	1.20E-01	
TOTAL								
ADULT	5.34E-02	1.05E+00	1.98E-01	1.05E+00	1.04E+00	1.45E+00	1.04E+00	2.19E-01
TEEN	5.34E-02	1.20E+00	3.00E-01	1.19E+00	1.19E+00	1.78E+00	1.19E+00	2.19E-01
CHILD	5.34E-02	1.74E+00	7.08E-01	1.75E+00	1.74E+00	2.87E+00	1.74E+00	2.19E-01
INFANT	5.34E-02	1.11E+00	4.06E-01	1.12E+00	1.12E+00	2.92E+00	1.11E+00	2.19E-01

Table 11.3-9: Gaseous Effluent Dose Evaluation for Gaseous Radioactive Waste System Failure

Parameter	Value
Release Source Term:	
I-131	9.33E-04 Ci
I-132	3.71E-04 Ci
I-133	1.33E-03 Ci
I-134	2.03E-04 Ci
I-135	7.80E-04 Ci
Xe-133	2.80E+01 Ci
Xe-135	9.63E-01 Ci
Kr-85m	1.07E-01 Ci
Kr-85	3.32E+01 Ci
Kr-87	5.82E-02 Ci
Kr-88	1.69E-01 Ci
Dispersion factor (0-2 hour exclusion area boundary)	6.22E-04 sec/m^3
Offsite dose consequence	< 10 mrem
Allowable dose limit	100 mrem

Table 11.3-10: Codes and Standards from Regulatory Guide 1.143, Table 1

Component	Design and Construction	Materials	Welding	Inspection and Testing
Piping and Valves	ANSI/ASME B31.3[1,2]	ASME Section II[3]	ASME Section IX	ANSI/ASME B31.3
Atmospheric Tanks	API-650	ASME Section II	ASME Section IX	API-650
Tanks (0-15 psig)	API-620	ASME Section II	ASME Section IX	API-620
Pressure Vessels and Tanks (>15 psig)	ASME BPVC Section VIII, Division 1 or Division 2	ASME Section II	ASME Section IX	ASME Section VIII, Division 1 or Division 2
Pumps	API-610; API-674; API-675; ASME BPVC Section VIII, Division 1 or Division 2	ASTM A571- 84(1997) or ASME Section II	ASME Section IX	ASME BPVC Code Section III, Class 3[4]
Heat Exchangers	TEMA STD, 8th Edition; ASME BPVC Section VIII Division 1 or Division 2	ASTM B359-98 or ASME Section II	ASME Section IX	ASME Section VIII, Division 1 or Division 2
Flexible Hoses and Hose Connections for MRWP[5]	ANSI/ANS-40.37	ANSI/ANS-40.37	ANSI/ANS-40.37	ANSI/ANS-40.37

Notes:
1. Class RW-IIa and RW-IIb Piping Systems are to be designed as category "M" systems.
2. Classes RW-IIa, RW-IIb, and RW-IIc are discussed in Regulatory Position 5 of RG 1.143.
3. ASME BPVC Section II required for Pressure Retaining Components.
4. ASME Code Stamp, material traceability, and the quality assurance criteria of ASME BPVC, Section III, Div. 1, Article NCA are not required. Therefore, these components are not classified as ASME Code Section III, Class 3.
5. Flexible hoses should only be used in conjunction with Mobile Radwaste Processing Systems (MRWP).

Table 11.3-11: GRWS Moisture Separator and Gas Cooler Radiological Content

Isotope	Activity (Ci/cm^3)
Kr83m	2.8E-10
Kr85m	1.2E-09
Kr85	3.5E-07
Kr87	6.4E-10
Kr88	1.9E-09
Kr89	4.3E-11
Xe131m	4.5E-09
Xe133m	4.1E-09
Xe133	3.1E-07
Xe135m	4.0E-10
Xe135	1.0E-08
Xe137	1.4E-10
Xe138	4.7E-10
Br82	4.6E-14
Br83	2.6E-13
Br84	1.2E-13
Br85	1.5E-14
I129	1.1E-18
I130	3.7E-13
I131	9.5E-12
I132	4.3E-12
I133	1.4E-11
I134	2.6E-12
I135	9.0E-12
C14	1.3E-11
Ar41	1.2E-08

Table 11.3-12: Not Used

Figure 11.3-1a: Gaseous Radioactive Waste System Diagram

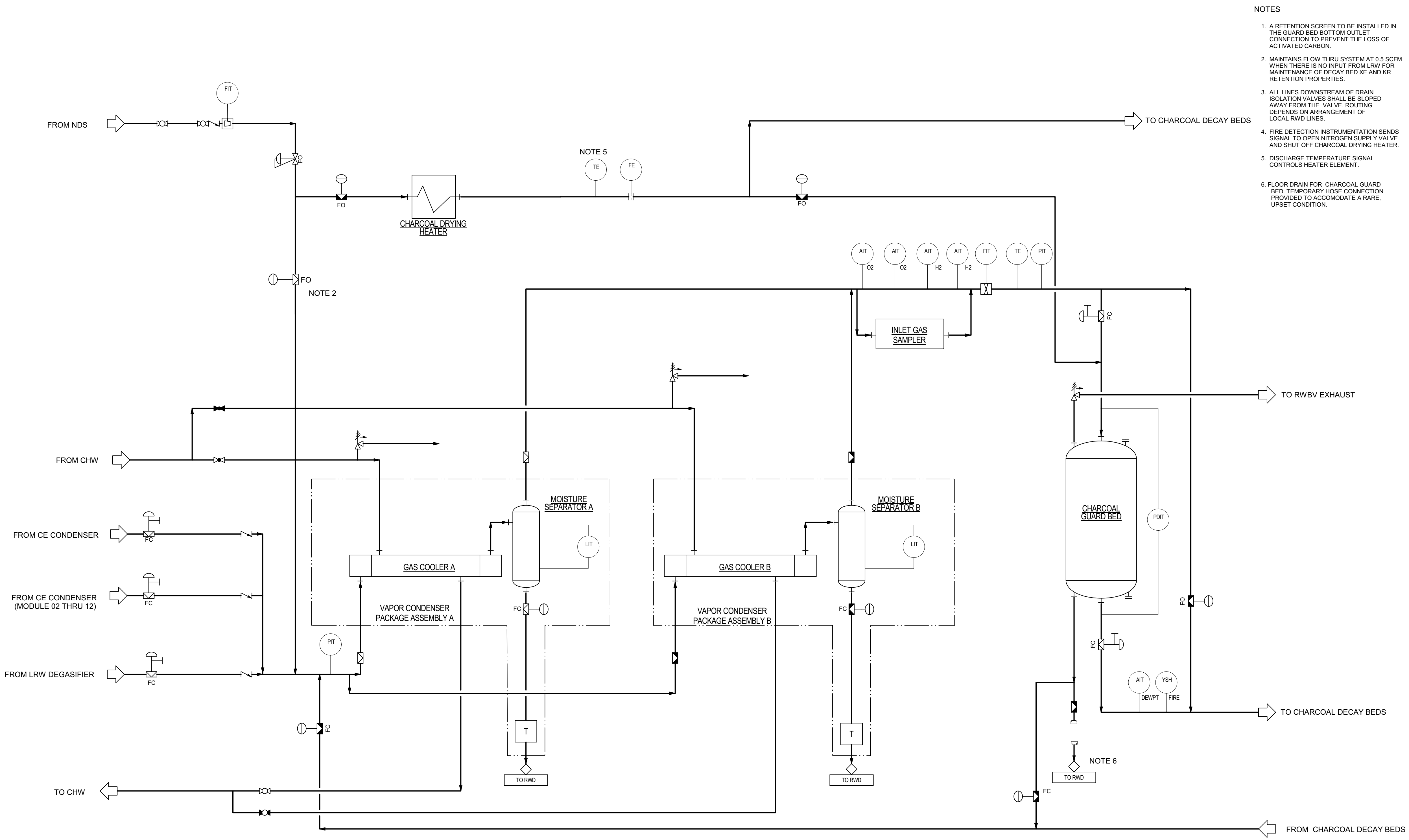
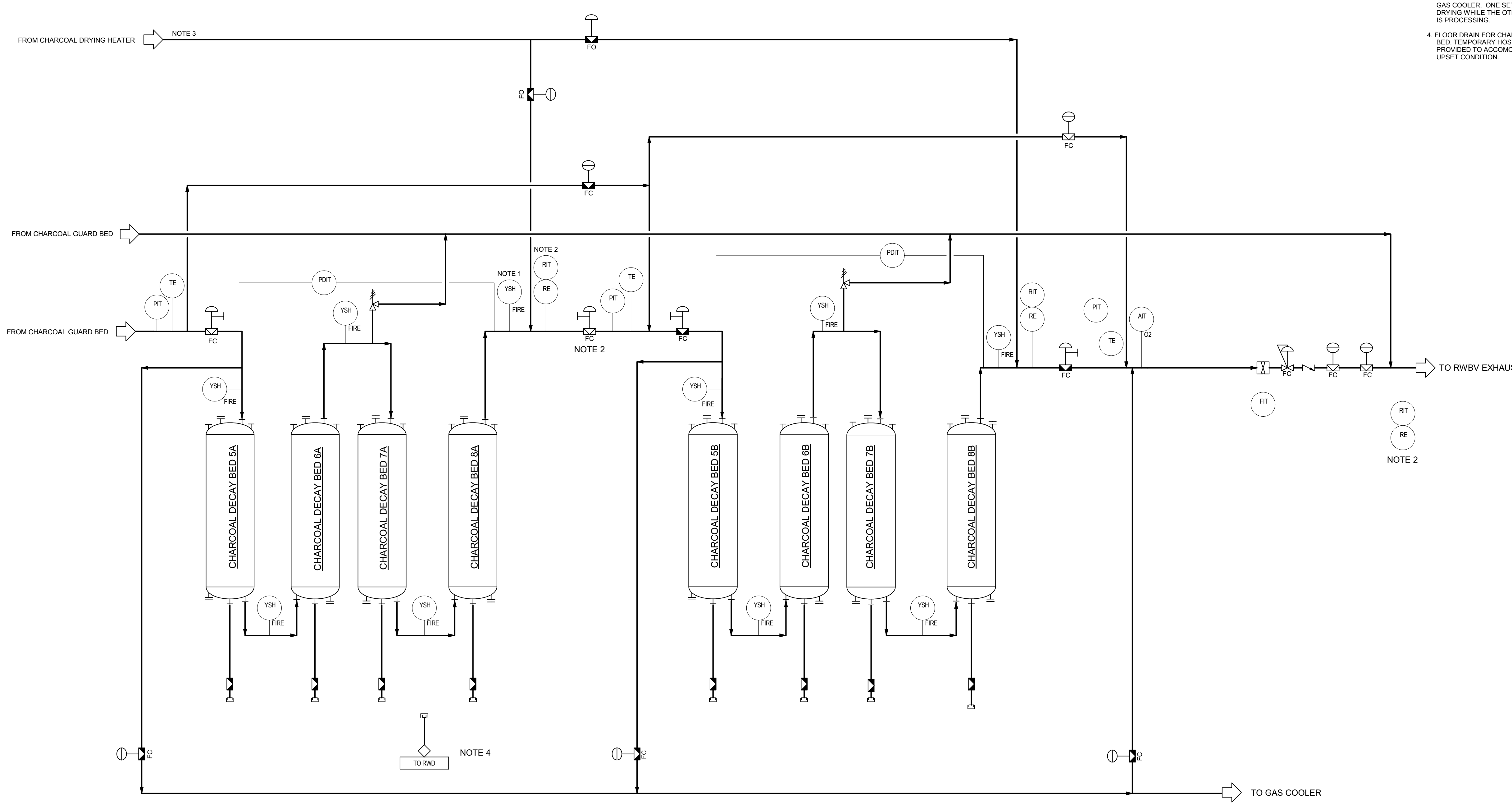


Figure 11.3-1b: Gaseous Radioactive Waste System Diagram



NOTES

1. FIRE DETECTION INSTRUMENTATION SENDS SIGNAL TO OPEN NITROGEN SUPPLY VALVE AND SHUT OFF CHARCOAL DRYING HEATER.
2. THIS RADIATION MONITOR REPRESENTS AN OFF-LINE RADIATION MONITOR . ON HIGH RADIATION THE GRWS IS ISOLATED.
3. THE NITROGEN TIES IN AT THE PROCESS OUTLET OF EACH SET OF BEDS, FLOWS IN REVERSE DIRECTION TO NORMAL PROCESS FLOW AND THEN RECYCLES BACK TO THE GAS COOLER. ONE SET OF BEDS MAY BE DRYING WHILE THE OTHER SET OF BEDS IS PROCESSING.
4. FLOOR DRAIN FOR CHARCOAL GUARD BED. TEMPORARY HOSE CONNECTION PROVIDED TO ACCOMODATE A RARE, UPSET CONDITION.

11.4 Solid Waste Management System

For the NuScale design, the solid waste management system is called the solid radioactive waste system (SRWS). The SRWS is designed to process both wet and dry solid waste from various plant systems produced during normal operation and anticipated operational occurrences, including startup, shutdown, and refueling operations. The SRWS processes solid waste by dewatering, decontaminating, surveying, sorting, and classifying solid waste for storage and eventual shipment to licensed offsite facilities. The Radioactive Waste Building (RWB) is classified as a RW-IIa building (Section 3.7.2) and has adequate space for onsite storage for various solid waste containers plus space for mobile processing equipment. The SRWS includes the wet solid waste (WSW) system, dry solid waste (DSW) system, mixed waste system, and an onsite storage area.

11.4.1 Design Bases

The following represent the design bases for the SRWS:

- The SRWS complies with guidance of Regulatory Guide (RG) 1.143, Revision 2, Branch Technical Position (BTP) 11-3, Revision 4, and ANSI/ANS-55.1-1992 (Reference 11.4-1). The RWB meets the seismic criteria of RG 1.143, as described in Section 3.7.2. Table 11.4-1 lists the SRWS equipment and compliance with RG 1.143.
- Regulatory Guide 1.143, Section 4.1, states that the design of the radioactive waste management structures, systems, and components (SSC) should follow the direction in RG 8.8, Revision 3 for maintaining personnel exposures ALARA during normal operation and maintenance activities, as required by 10 CFR 20.1101(b). Compliance with RG 8.8 and RG 8.10 is explained in Section 12.1 and Section 12.3. Design features incorporated to maintain exposure ALARA include remote system operation and remotely actuated flushing equipment, with equipment layout that shields plant personnel from components containing radioactive materials.
- The SRWS is designed to prevent the release of significant quantities of radioactive materials to the environment and keep the overall exposure to the public and plant personnel within the limits specified in 10 CFR 20 and 10 CFR 50, Appendix I.
- Shielding is designed based upon the conservative design basis source term identified in Section 11.1. The spent filters and spent resins are assumed to be fully loaded using the design basis source term in evaluating the shield wall thickness. Additional details on the shielding design are discussed in Section 12.3.
- The SRWS is designed such that wet and dry radioactive solid waste packaged for offsite shipment and disposal complies with the requirements of 10 CFR 61.55, 10 CFR 61.56, 10 CFR 71 and 49 CFR 171-180, as applicable.
- Tank cubicles that contain the spent resin storage tanks (SRSTs) and the phase separator tanks (PSTs) are stainless steel-lined up to the cubicle wall height equivalent to the full tank volume to provide containment and facilitate decontamination in the event of tank leakage or failure. This design feature, in conjunction with drainage and transfer capabilities, provides passive design features to prevent the release of radioactive liquid to groundwater and the environment in accordance with the requirements of 10 CFR 20.1406 and guidance in RG 4.21 (Section 12.3).

- Interconnections between the SRWS and other plant systems are designed so that contamination of non-radioactive systems is precluded, and the potential for uncontrolled and unmonitored release of radiation to the environment from a single failure are minimized. This feature meets the guidance provided by IE Bulletin 80-10.
- Onsite storage allows for radioactive decay with adequate storage in case of processing, maintenance or transportation delays. Onsite storage is provided to hold solid waste for at least 30 days in accordance with ANSI/ANS-55.1-1992 (Reference 11.4-1) and BTP 11-3.
- General Design Criteria 60 was considered in the design of the SRWS. The SRWS is designed to control the release of radioactive materials and to handle solid wastes generated during normal operation and during anticipated operational occurrences.
- General Design Criteria 61 was considered in the design of the SRWS. The SRWS is designed with a capability to permit appropriate periodic inspection and testing of components with suitable shielding for radiation protection and with appropriate waste shipping containers.
- General Design Criteria 63 was considered in the design of the SRWS. The SRWS and associated handling areas have area radiation monitoring equipment to detect excessive radiation or airborne levels and initiate appropriate alarms and procedural actions to maintain radiation exposure ALARA. This information is provided in Section 12.3.
- General Design Criteria 3 was considered in the design of the SRWS. The SRWS and components avoid the generation of explosive gas mixtures and exothermic reactions.
- The SRWS is designed to meet the dose limits for individual members of the public (10 CFR 20.1301(e), 40 CFR 190, and 10 CFR 20.1302), the disposal and manifest requirements of 10 CFR 20.2006, the environmental and health regulations of 10 CFR 20.2007, and the waste disposal records requirements of 10 CFR 20.2108.
- The SRWS has permanently installed connections for mobile equipment that process waste directly into mobile equipment or to a high integrity container (HIC). These connections meet RG 1.143, ANSI/ANS-40.37-2009 (Reference 11.4-2), and IE Bulletin 80-10.
- Mobile or temporary solid waste processing equipment is designed in accordance with ANSI/ANS-40.37-2009 (Reference 11.4-2). The equipment and its interconnection to plant systems will conform to 10 CFR 20.1406, Branch Technical Position 11-3, and Regulatory Guide 1.143.

11.4.2 System Description

The SRWS is a nonsafety-related system and serves no safety-related functions. The SRWS is designed to:

- collect, process, sample, package, and store WSW generated from the chemical and volume control system (CVCS), pool cleanup system (PCUS), and liquid radioactive waste system (LRWS), using both permanently installed and mobile equipment in the SRWS. Wastes from these systems are mainly WSW and consist of spent resins, spent charcoal, cartridge filters, filter membranes, and sludge.

- collect, segregate, sample, package, and store compactible and non-compactible DSW, including heating ventilation and air conditioning (HVAC) filters, failed tools and equipment parts, used personnel protective equipment, rags, paper, plastic, rubber, scrap wood, concrete, and metal.
- collect, sample, segregate, package, and ship mixed and oily wastes.
- provide sufficient storage space for packaged solid wastes.
- process and package waste into disposal containers that are approved by the Department of Transportation and are acceptable to licensed waste disposal facilities for offsite shipment and burial.
- meet federal regulations and protect the worker and the general public from radiation by maintaining dose levels ALARA.

The SRWS is designed primarily to handle three types of generated wastes: wet solid wastes, dry solid wastes, and miscellaneous wastes. The generated solid waste varies in characteristics and contamination level, and is further divided into the following waste streams:

- wet solid waste, such as spent resin, spent process filter cartridges, tubular ultra-filtration (TUF) and reverse osmosis (RO) filter membranes, and granular activated charcoal (GAC)
- dry solid waste such as ventilation filters, activated charcoal (bulk and filter elements), rags, paper, plastic, rubber, scrap wood, glass, concrete, metal, and failed equipment parts and tools
- miscellaneous waste includes mixed waste and oily sludge

The boundaries of the SRWS begin at the connection to a particular waste stream source and end at the packaged waste container offsite shipment. For WSW, these connections usually involve flanged joints, and boundary valves at the system inlets. For dry solid waste, the boundaries are not always physical because much of dry solid waste is collected from a variety of locations and transported through corridors to the solid radioactive waste sorting area.

For spent resins and GAC, the SRWS starts downstream of the boundary valve from each demineralizer and carbon bed. Spent resin is sluiced into the SRSTs or PSTs for decay and eventually sent to waste containers. The containers are processed (dewatered, sealed) and placed in the storage area until shipped offsite for further processing or disposal. Spent carbon is normally sent directly to waste containers to avoid mixing with resins.

For spent cartridge filters, the SRWS starts at the filter extraction point. With the assistance of a monorail hoist, the spent filter is removed from the filter housing and placed in a shielded spent filter transfer cask. The cask is transferred to a cart and then taken to the RWB. The spent filter is placed into a waste container. Once the waste container is full, it is dewatered, sealed, and stored for eventual offsite processing and disposal.

Mixed wastes and filter membranes are collected separately at the point of generation during maintenance activities. Oil sludge from the oil separators is manually collected in drums. Low-contamination wastes, such as used personnel protective equipment and maintenance parts, are collected in bags or boxes. These wastes are typically stored in 55-gallon drums or B-25 boxes in the low-activity storage area until transported offsite for processing and disposal.

11.4.2.1 Dry Solid Waste

The DSW system is used to process waste by:

- segregating waste types at the source of generation, including nonradioactive waste, in controlled access areas of the plant as much as practical, including bagging and tagging where appropriate.
- collecting and transporting contaminated DSW to a sorting area within the RWB.
- sorting contaminated DSW by waste type.
- decontaminating DSW, if possible, and providing storage prior to processing.
- compacting or shredding DSW, as appropriate, or preparing DSW for shipment to an offsite volume reduction processing facility.
- packaging both compactable and non-compactable DSW into storage or shipping containers.
- monitoring the waste for classification per 10 CFR 61.55.
- surveying containers and decontaminating, as required to meet 49 CFR 173.
- transporting filled waste containers to the truck bay area and loading onto vehicle.

Dry solid waste includes HVAC filters, tools and equipment, used personnel protective equipment, rags, paper, wood and miscellaneous cleaning supplies. Non-contaminated wastes are normally separated from contaminated DSW as close as possible to the point of generation. Additional sorting is done to segregate waste types to facilitate efficient packaging. The DSW handling and storage operation is summarized in Figure 11.4-1.

Table 11.4-2 provides an estimate of expected annual DSW and the anticipated waste classification based upon operating experience and industry practices. During some anticipated operational occurrences, such as refueling, the rate of DSW generation is higher than during normal operations. Major equipment items, such as core components and containment vessel components, are not processed in the SRWS.

11.4.2.2 Wet Solid Waste

The WSW processing system receives and processes three major waste streams:

- radioactive spent resin and spent charcoal
- spent cartridge filters
- filter membranes from TUF and RO

The WSW is homogenized, sampled, and analyzed to classify the waste in accordance with 10 CFR 61. Spent resin and spent charcoal are transferred to HICs that are connected to a dewatering system located inside a confined enclosure.

Dewatering fluids are transferred back to the LRWS for processing. Containers are capped and sealed after dewatering, and the container is surveyed and decontaminated, as necessary, to meet 49 CFR 173 requirements.

The sources of spent resins to SRWS come from the CVCS, the PCUS, and the LRWS.

The CVCS resins originate from the following demineralizers: mixed bed ion exchangers A and B, cation bed ion exchanger, and an auxiliary mixed bed ion exchanger for each NuScale Power Module, for a total of up to 48 demineralizers. The CVCS spent resins are considered Class B/C waste, except for the CVCS deborating demineralizer resins, which are classified as Class A waste.

The PCUS resins originate from three demineralizers and are considered Class B/C waste.

The other source of spent resins originates from the LRWS in which there are five demineralizers (one cation, one anion, one mixed bed, one cesium, and one antimony) on the low-conductivity waste processing skid. The spent resin generated in the low-conductivity waste processing skid is considered Class A waste.

If operational conditions develop such that condensate polisher demineralizer resins require removal as contaminated waste, local transfer of the resins to HICs or other suitable containers is performed and the container is transferred to the SRWS area for processing and storage.

There is also media in the granular activated charcoal filter present in both the low-conductivity waste and high-conductivity waste processing skids that are classified as Class A waste.

Other sources of wet solid waste include spent cartridge filters, oily wastes, TUF and RO filter membranes and their associated rejects from the liquid radioactive waste system, and waste from the CVCS and pool cleanup system. RO and TUF reject material includes dissolved and suspended solids, respectively (Section 11.2.2.1.2).

Table 11.4-3 provides an estimate of expected annual WSW generated based on the expected operation of 12 NuScale Power Modules.

In accordance with BTP 11-3, components and piping that contain slurries have flushing capabilities via the LRW clean-in-place skid or directly from the demineralized water break tank. The spent resin storage and PSTs are ASME Section VIII tanks that can use compressed service air to pressurize the tanks and pneumatically transport resin to a HIC. The associated pressure relief valves on the spent resin storage and PSTs are vented to the tank's cubicle, which are vented to the RWBVS. These tanks are also designed with break pots that provide a vent path during non-pressurized operation to prevent carryover from the tank into the RWBVS ductwork. The dewatering system also provides a vent path that is directed to the RWBVS. This vent path is provided with a high-efficiency particulate air (HEPA) filtration unit to prevent spreading contamination to the ventilation system.

11.4.2.2.1 Spent Resin Handling

Spent resins from the CVCS and PCUS demineralizers are normally sluiced to one of the two redundant SRSTs. Each tank has sufficient capacity to receive and store spent resins from at least two years of operation of the PCUS and CVCS, plus the associated flush water used to transfer the resins from the demineralizer to the SRST. The SRSTs provide staging for decay and transfer capability into approved disposal containers for offsite disposal. Figure 11.4-2a and Figure 11.4-2b are process flow diagrams of the spent resin handling system.

Each of the two PSTs is sized to receive spent resins from the LRW demineralizers for at least two years of plant operation, plus the associated flush water. The carbon from the GAC bed is normally transferred directly to a HIC instead of a PST to prevent mixing the carbon with spent resins, maximizing flexibility of offsite disposal options.

There are two spent resin transfer pumps that are used to take suction from the decant portion of the SRST to provide sluicing water for transferring spent resins from the CVCS or PCUS demineralizers to the SRST. Three decant connections are provided on the side of each SRST to ensure adequate suction pressure is available for the pumps, while minimizing waste generation.

Two phase separator transfer pumps are used to take suction from the decant portion of the PST to provide sluicing water for transferring spent resins from the LRW demineralizers to the PST. Similar to the SRSTs, three decant connections are provided on the side of each PST.

If necessary, service air can be used to fluff the resins through the sparging nozzles at the bottom of the SRST and PST. Air is not used to backwash the tank decant screens.

Turbidity instruments are provided on the resin transfer lines upstream of the SRST, PST, and dewatering system to start the flushing process automatically when the solid content of the vessel has been fully transferred.

After the spent resins have been allowed to decay within the SRST or PST, resins are transferred to a HIC using compressed air as the motive force. An automatic inline sampler located upstream of the dewatering system provides the capability to sample the resin slurry during the filling process to classify the waste. The inline sampler is programmed to obtain multiple samples at pre-set intervals during the resin transfer to obtain a representative sample.

The dewatering system is a vendor-supplied package that allows dewatering spent resins and filter cartridges in a HIC before it is shipped for offsite disposal.

The dewatering system removes standing water in the HIC to meet 10 CFR 61 and the transportation requirements of 49 CFR 173, Subpart I. Gaseous exhaust during transfer and dewatering activities is vented through a vent port connection on the fillhead to the RWBVS.

To avoid the generation of explosive gas mixtures and exothermic reactions, the upstream systems (LRWS, PCUS, CVCS) that transfer resins to the SRST or PST do not use chemicals (e.g., nitrates, nitrites) that can generate exothermic reactions with resins (COL Items 9.2-1 and 13.5-3). Flush water is also available to respond to exothermic reactions in the SRST, phase separator tank, and a HIC during the filling and dewatering operations. The SRWS has monitoring capabilities, including temperature monitoring during the HIC filling and dewatering operations and closed-circuit television monitoring in the storage areas.

11.4.2.2.2 Spent Charcoal Handling

Granular activated charcoal filter is utilized in both the low-conductivity waste and high-conductivity waste in the LRW processing skids. The spent activated carbon from the LRWS is removed using the same process as the spent resin but is normally transferred directly to a HIC and placed in separate disposal containers. The process flow diagram of the spent charcoal handling system is shown on Figure 11.4-1.

11.4.2.2.3 Spent Cartridge Filters

Spent process cartridge filters are generated from the chemical and volume control system, pool cleanup system and LRWS. The CVCS consists of two reactor coolant filters downstream of the demineralizers. One of these filters is expected to be replaced once per year, resulting in an annual total of 12 spent filter cartridges replaced for a 12 NPM plant. The PCUS contains two filters upstream of the demineralizers with each filter assembly consisting of seven cartridges. One PCUS filter is expected to be replaced each year. The CVCS and PCUS cartridge filters are processed as high activity filters (Class B/C).

The LRWS also contains a cartridge filter located downstream of the detergent waste collection tank transfer pump and is expected to be replaced twice per year. The cartridge filters generated from LRWS are processed as low-activity spent filters (Class A).

Spent cartridge filters are removed from their filter housings using a shielded filter cask and a crane or monorail hoist. The shielded filter cask is transported by a cart to the RWB and placed in a HIC. See Figure 11.4-1 for an overview of the cartridge filter handling process.

11.4.2.2.4 Contaminated Oil Handling

The generation of contaminated oil is expected to be very low. The main source of oily waste is expected to come from floor drains. The oil is directed to the SRWS from the LRWS oil separators and is manually collected in drums. The drums of contaminated oil are sent to an offsite treatment facility.

11.4.2.3 Mixed Waste Handling

Mixed waste is a combination of radioactive waste mixed with Resource Conservation and Recovery Act-listed hazardous waste as defined in 40 CFR 261 Subpart D. The generation of mixed waste volume is expected to be extremely low. Mixed waste can only be disposed of in a permitted mixed waste disposal facility. Mixed waste is collected near the source and transferred in 55-gallon drums to a permitted facility.

11.4.2.4 Packaging, Storage, and Shipping

The SRWS is designed to utilize Department of Transportation-approved containers for packaging, storing, and disposal of solid radioactive wastes. High-integrity containers are used as the final waste containers for spent resins, granulated active carbon, and spent filters. B-25 boxes are mainly used for DSW, such as HVAC filters and Class A DSW. Type 7A drums are also used for items such as oily waste, mixed waste, dried filter membranes, sludge, chemical waste, and dried detergent waste.

Waste is classified as Class A, Class B, Class C, or greater than Class C in accordance with 10 CFR 61.55 and 10 CFR 61.56 per the site process control program. The expected annual volumes of solid waste and shipment offsite estimates are provided in Table 11.4-2 and Table 11.4-3. The packaging and shipment of radioactive solid waste for disposal complies with 10 CFR 20, Appendix G, 10 CFR 61.56, and 49 CFR 173, Subpart I.

The module trolley bay is located near the Class A package waste storage area to provide an enclosed area to bring a shipping container and load packaged waste onto the truck for offsite burial or processing. Class A and Class B/C HIC storage is located on the 71' elevation. A HIC filling room is also located on the 71' elevation adjacent to the HIC storage room. Both spaces are accessible by removable shield plugs in the floor.

A permanently installed overhead 50-ton crane is also provided for moving packaged waste into and out of the storage areas, and to load the waste onto shipping trucks. This crane is also used to move waste containers into and out of the HIC fill station and dewatering areas, and to move the floor shield plugs that provide access to the HIC storage room and HIC filling room on the 71' elevation. The crane has a closed-circuit television camera to allow remote crane operations during waste package handling.

Space is provided in the RWB for both Class A and Class B/C waste storage. At the expected waste generation rates, there is storage capacity for at least 30 days.

The design does not provide for a temporary onsite low-level radioactive waste storage facility.

11.4.2.5 Component Description

The SRWS components are designed and constructed using the codes and standards provided in RG 1.143. Each component is classified as RW-IIa, RW-IIb or RW-IIc based on the radionuclide content compared against the A_1 and A_2 values tabulated in 10 CFR 71, Appendix A. The safety classification for the SRWS components applies to components, up to and including the nearest isolation device. The radionuclide source terms for the SRWS components are provided in Section 12.2. Table 11.4-1 provides design parameters for each of the major components.

11.4.2.5.1 Tanks

Spent Resin Storage Tanks

There are two permanently installed SRSTs that are ASME Section VIII pressure vessels constructed of welded stainless steel with conical bottoms. The SRST pressure relief valves are vented back to the tank cubicle.

The SRSTs are cross-connected so that the failure or maintenance of one tank does not impair the system or plant operation. The SRSTs are housed in individual cubicles, each with a shield wall thickness commensurate with the projected maximum dose rate of its contents. The SRSTs are housed in separate cubicles which are lined with stainless steel and have a sloped floor to a drain. The drain system in the RWB is described in Section 9.3.3. As listed in Table 11.4-3, it is anticipated that the Class B/C spent resins from the CVCS (except the deborating demineralizers) and PCUS demineralizers are sent to one of the SRSTs for approximately two years and then allowed to decay as the second SRST is utilized. After approximately two years of decay, the first SRST resins are sluiced to disposal containers for storage or offsite shipment.

Phase Separator Tanks

There are two permanently installed PSTs that are provided to receive spent resins from the LRW demineralizers and the CVCS deborating demineralizers. Similar to the SRSTs, the phase separator tanks are cross-connected for operational flexibility and are ASME Section VIII pressure vessels housed within separate cubicles lined with stainless steel and having a sloped floor to a drain. The phase separator tanks are smaller than the SRSTs and are anticipated to collect Class A resins from the LRWS demineralizers and the CVCS deborating demineralizers.

Break Pot Tank

There are four ASME Section VIII break pot tanks, one for each of the SRSTs and phase separator tanks. The break pot tank is designed with level transmitters to provide an indication of tank overflow. The break pot tank also provides a volume to capture overflow fluid and prevent the fluid from reaching the HVAC ductwork.

11.4.2.5.2 Pumps

Two spent resin storage tank transfer pumps with variable frequency drives are used to take suction from the decant portion of the resin storage tanks and provide water to sluice spent resins from the PCUS and CVCS demineralizers to an SRST. This provides the motive force to sluice resins, while minimizing the generation of radioactive waste. These pumps can also be used to fluff the spent resins inside the SRST by recirculating decant water prior to transferring spent resins to a HIC.

Two phase separator transfer pumps are used to take suction from the decant portion of the PST to provide sluice water as a motive force to transfer resins from LRWS demineralizers, or CVCS deborating demineralizers, to a PST. The PST transfer pump can also be used to fluff the spent resins inside the PST prior to transferring spent resins to a HIC.

11.4.2.5.3 Piping and Valves

The SRWS piping material is stainless steel and is butt-welded to minimize crud traps. Backing rings are not allowed in SRWS piping. Slurry transport lines are sized to maintain a flow velocity to prevent the slurry from settling and utilize bends of five pipe diameter radius. Slurry lines are also sloped to promote complete drainage and are connected to the clean-in-place skid and directly to the demineralized water break tank to allow flushing and cleaning of SRWS piping and components after batch operations. Piping is also arranged to minimize tees, pipe branches, and dead legs. The SRWS valves are stainless steel, remote air-operated valves. Valves in slurry transfer lines are full-ported ball valves and liquid process valves are diaphragm valves.

11.4.2.5.4 Waste Compactor

The waste compactor is a vendor supplied package located in the RWB and is used to compact DSW that is compactible to reduce the volume of contaminated items. The waste compactor is equipped with a HEPA filter to remove airborne contamination from the exhausted air prior to entering the RWBVS.

11.4.2.5.5 Dewatering System

The dewatering system is a skid-based, vendor-supplied package that removes free-standing water from waste packages to meet transportation and disposal requirements. The fillhead portion of the dewatering system includes an exhaust vent with HEPA filtration routed to the RWBVS to control airborne contamination. Liquid removed by the dewatering system is routed to the LRWS high-conductivity waste collection tank. The dewatering system and associated connections to permanent plant equipment, including non-contaminated utilities, complies with IE Bulletin 80-10, Regulatory Guide 1.143, and ANSI/ANS-40.37-2009 (Reference 11.4-2).

11.4.2.6 Effluent Controls

The SRWS waste streams are shown in Figure 11.4-1. The flow diagram for the spent resin handling system is shown in Figure 11.4-2a and Figure 11.4-2b. Table 11.4-1 provides the classifications of the SRWS components.

Expected waste classifications of solid wastes for influents processed and shipped are provided in Table 11.4-2 and Table 11.4-3.

The SRWS does not release effluents directly to the environment. Liquids removed from solid waste processing are transferred to the LRWS for further processing.

During the operation of the SRWS, such as processing and packaging solid waste, the expelled air is captured by the RWBVS to prevent unmonitored contamination being released to the environment.

11.4.2.7 Operation and Personnel Exposure

The SRWS is designed with features in accordance with RG 8.8, including shielded cubicles and remotely-operated equipment, to meet the occupational dose limits provided in 10 CFR 20.1201 and 20.1202. Sections 12.3 and 12.4 further describe these design features and the occupational doses associated with operation of the SRWS.

11.4.2.8 Site-Specific Cost-Benefit Analysis

Because the SRWS does not release effluents to the environment, a cost-benefit analysis is not performed separately from the evaluations in Sections 11.2 and 11.3.

11.4.2.9 Mobile or Temporary Equipment

The SRWS is designed with modular equipment (spent resin dewatering system) and options for additional mobile equipment (shredders, laundry unit, etc.). The purpose of modular and mobile equipment is to provide ease of equipment replacement due to either advances in treatment technologies or equipment problems. Modular equipment is permanent plant equipment that is designed to facilitate easier replacement (e.g., skid-based equipment), while mobile equipment is non-permanent or temporary equipment that is added to the facility.

COL Item 11.4-1: A COL applicant that references the NuScale Power Plant design certification will describe mobile equipment used and connected to plant systems in accordance with American National Standards Institute / American Nuclear Society (ANSI/ANS) 40.37, Regulatory Guide 1.143, 10 CFR 20.1406, NRC IE Bulletin 80-10, and 10 CFR 50.34a.

11.4.3 Radioactive Effluent Releases

Solid wastes are packaged in disposal containers and shipped offsite for disposal. The annual estimates for total solid radioactive waste shipped offsite for burial are listed in Table 11.4-2 and Table 11.4-3.

The SRWS is designed to send liquid and gaseous effluents to the LRWS and RWBVS, respectively. As a result, other than solid waste shipments offsite, the SRWS is not designed to release effluents directly to the environment. The contributions to the offsite dose consequences from SRWS are included in the evaluations for LRW and gaseous radioactive waste systems in Sections 11.2 and 11.3.

The SRWS is designed in accordance with the requirements of 10 CFR 20.1406. The SRWS design features to prevent the spread of contamination, facilitate decommissioning, and reduce the generation of radioactive waste are discussed in Section 12.3.

COL Item 11.4-2: A COL applicant that references the NuScale Power Plant design certification will develop a site-specific process control program following the guidance of Nuclear Energy Institute (NEI) 07-10A (Reference 11.4-3).

11.4.4 Malfunction Analysis

The SRWS is not safety-related and does not perform a safety-related function; thus there is no requirement for the system to be single-failure proof. However, to demonstrate the design's resistance to failures, a malfunction analysis was performed. This malfunction analysis is summarized in Table 11.4-4.

11.4.5 Testing and Inspection Requirements

The SRWS is tested during plant pre-operations to ensure operation of components and processes as discussed in Section 14.2. During plant operations, the periodic testing and inspection requirements of RG 1.143 are performed to support continued proper operation of components.

11.4.6 Instrumentation and Controls

The SRWS is operated as batch operations for both dry and wet wastes. The dry waste operations are largely manual. The batch operations for wet wastes are controlled from the waste management control room (WMCR). Significant SRWS parameters are displayed and alarmed to provide process information. A listing of instruments, alarms, and indication locations is presented in Table 11.4-5.

SRWS instrumentation readouts, controls and alarms are located in the WMCR, with the exception of the HIC dewatering skid, which has its own local control panel.

The SRWS does not have automatic control features using radiation detectors.

11.4.7 References

11.4-1 American National Standards Institute/American Nuclear Society, "Solid Radioactive Waste Processing System for Light-Water-Cooled Reactor Plants," ANSI/ANS-55.1-1992, LaGrange Park, IL.

11.4-2 American National Standards Institute/American Nuclear Society, "Mobile Low-Level Radioactive Waste Processing Systems," ANSI/ANS-40.37-2009, LaGrange Park, IL.

11.4-3 Nuclear Energy Institute, "Generic FSAR Template Guidance for Process Control Program," NEI 07-10A, Rev. 0, Washington, DC, March 2009.

Table 11.4-1: List of Systems, Structures, and Components Design Parameters

Component (Quantity)	RG 1.143 Safety Classification	Standards	Type	Capacity	Design Pressure (psig)	Design Temperature (°F)	Material	Table for Assumed Radioactive Content
Spent resin storage tank (2)	RW-IIa	ASME Section VIII	pressure vessel	16,000 gal	150	180	stainless steel	Table 12.2-19
Phase separator tank (2)	RW-IIb	ASME Section VIII	pressure vessel	5,000 gal	150	180	stainless steel	Table 12.2-19
SRST break pot tank (2)	RW-IIc	ASME Section VIII	pressure vessel	12 gal	150	180	stainless steel	Table 11.4-6
PST break pot tank (2)	RW-IIc	ASME Section VIII	pressure vessel	12 gal	150	180	stainless steel	Table 11.4-6
SRST transfer pump (2)	RW-IIc	API-610	seal-less (variable frequency drive)	196 gpm (25 hp)	150	180	stainless steel	-----
PST transfer pump (2)	RW-IIc	API-610	seal-less	75 gpm (10 hp)	150	180	stainless steel	-----

Table 11.4-2: Estimated Annual Volumes of Dry Solid Waste

Waste Classification	Sources and Waste Classification (A or B/C)	Volume Generated (ft^3/yr)	Container Type	Container Volume (ft^3)	No. of Containers (rounded off)
Class A	HVAC HEPA filters	650	B-25 box	90	7 boxes
Class A	Used PPE, rags and wipes	3000	B-25 box	90	33 boxes
Class A	tools, miscellaneous	9	55-gal drum	7.3	2 drums
Class B/C	failed equipment	7	55-gal drum	7.3	1 drum

Table 11.4-3: Estimated Annual Volumes of Wet Solid Waste

Waste Classification	Sources	Volume Generated (ft^3/yr)	Container Type	Container Volume (ft^3)	No. of Containers (rounded up)
Class B/C	CVCS and PCUS spent resins	600	8-120 HIC	112	6 HICs
Class B/C	cartridge filters from CVCS and PCUS	52	8-120 HIC	112	1 HIC
Class A	LRWS spent resins and CVCS deborating spent resins	170	8-120 HIC	112	2 HICs
	LRWS filter cartridges, TUF filter rejects, RO rejects, membranes, and misc.	20	55 gal drum	7.3	3 drums
Class A	oily waste	14	55 gal drum	7.3	2 drums
	mixed waste	14	55 gal drum	7.3	2 drums
	charcoal (from GAC) (replaced every 5 years)	20 (avg)	8-120 HIC	112	1 HIC every five years

Table 11.4-4: Solid Radioactive Waste System Equipment Malfunction Analysis

Equipment Item	Malfunction	Results (Consequences)	Mitigating or Alternate Actions
Phase separator tank (PST)	tank failure	The demineralizers are not able to transfer spent resin to the PST. The PSTs are installed in steel-lined cubicles able to contain the total tank contents if there is a catastrophic failure.	There are two PSTs to collect spent resins. If one tank fails, the other tank continues to receive spent resins.
Phase separator transfer pump	pump failure	Water cannot be recirculated or sent downstream of the pump.	There are two PST transfer pumps, with one pump dedicated to each tank. The tank transfer line is cross-connected to the pump suction, allowing the process to continue if one pump fails.
Spent resin storage tank	tank failure	The demineralizers are not able to transfer spent resins to the SRST. The SRSTs are installed in steel-lined cubicles to contain the total tank contents if there is a catastrophic failure.	There are two SRSTs to collect spent resins. If one of the tanks fails, the other tank can receive the spent resins.
SRST Transfer Pump	pump failure	Water cannot be recirculated or sent downstream of the pump.	There are two SRST transfer pumps, with one pump dedicated to each tank. The tank transfer line is cross-connected to the pump suction, allowing the process to continue if one pump fails.
High integrity container (HIC)	drop container during transportation	HICs are transported between the fill station, storage area, and truck bay area. Dropping an HIC can cause local contamination. The floor drains in the area collects the liquid; however, the solid portion (resin or charcoal) is removed by a vacuum.	The grapple assembly has limit switches to ensure the legs are engaged prior to lifting the HIC. In addition, the crane has its own safety break system to ensure the HIC is not dropped during a power failure.
SRSTs and PSTs	service air supply failure	The tanks cannot be pressurized to perform pneumatic sluicing to a HIC. The demineralizers are not able to send spent resins to the HICs for shipping offsite.	• If one compressor fails, the backup compressor is used to pressurize the tanks to complete the sluicing. • If the service air supply fails during the resin transfer, the lines are flushed to the HIC and system is restored to standby position.
Break pot tank	tank failure	The SRST or PST may not be able to vent or detect overflow condition to protect the ventilation system.	Each resin storage tank is equipped with one break pot tank in the vent line. The tank or the level transmitter is repaired prior to filling. An alternative is to use the second tank until this tank is fixed.

Table 11.4-5: Instrument List

Type	Location	Local	Indication Location
Tank level - radar	Tank top - SRST and PST (measures slurry level not measurable by conventional d/p)	No	WMCR
Tank level - differential pressure (d/p)	Side of tank - water level in the SRST and PST (measures by conventional d/p)	No	WMCR
Valve position indication	Each automatic valve	No	WMCR
Turbidity or density sensor	Upstream of the tanks and HIC to start the automatic flushing when the resin transfer is completed	No	WMCR
Tank level - break pot	Side of tank - water level in the vent line of the SRST and PST (measures by conventional d/p)	No	WMCR
Pressure indicating transmitter	SRST and PST pressure	No	WMCR
Pressure indicating transmitter	SRST and PST transfer pump discharge pressure	Yes	WMCR
Flow transmitter	SRST and PST discharge flow	No	WMCR

Table 11.4-6: Spent Resin Storage Tank and Phase Separator Tank Breakpot Radiological Content

Isotope	SRST Breakpot Activity (Ci/cm^3)	PST Breakpot Activity (Ci/cm^3)
Kr83m	4.9E-09	4.9E-09
Kr85m	2.1E-08	2.1E-08
Kr85	6.1E-06	6.1E-06
Kr87	1.1E-08	1.1E-08
Kr88	3.3E-08	3.3E-08
Kr89	7.5E-10	7.5E-10
Xe131m	8.0E-08	8.0E-08
Xe133m	7.3E-08	7.3E-08
Xe133	5.4E-06	5.4E-06
Xe135m	7.0E-09	7.0E-09
Xe135	1.8E-07	1.8E-07
Xe137	2.4E-09	2.4E-09
Xe138	8.3E-09	8.3E-09
Br82	1.4E-10	1.4E-12
Br83	7.8E-10	7.8E-12
Br84	3.6E-10	3.6E-12
Br85	4.4E-11	4.4E-13
I129	3.4E-15	3.4E-17
I130	1.1E-09	1.1E-11
I131	2.8E-08	2.8E-10
I132	1.3E-08	1.3E-10
I133	4.3E-08	4.3E-10
I134	7.6E-09	7.6E-11
I135	2.7E-08	2.7E-10
Rb86m	3.3E-14	1.6E-14
Rb86	1.9E-10	9.6E-11
Rb88	3.3E-08	1.6E-08
Rb89	1.5E-09	7.5E-10
Cs132	3.7E-12	1.9E-12
Cs134	3.3E-08	1.7E-08
Cs135m	2.5E-11	1.3E-11
Cs136	7.1E-09	3.5E-09
Cs137	2.0E-08	1.0E-08
Cs138	1.2E-08	6.0E-09
P32	5.5E-16	1.1E-17
Co57	4.1E-18	8.2E-20
Sr89	2.5E-11	4.9E-13
Sr90	5.5E-12	1.1E-13
Sr91	1.3E-11	2.6E-13
Sr92	6.8E-12	1.4E-13
Y90	1.3E-12	2.7E-14
Y91m	6.8E-12	1.4E-13
Y91	3.6E-12	7.1E-14
Y92	5.8E-12	1.2E-13
Y93	2.7E-12	5.4E-14
Zr97	4.0E-12	8.0E-14
Nb95	5.8E-12	1.2E-13
Mo99	7.2E-09	1.4E-10
Mo101	2.7E-10	5.4E-12

Table 11.4-6: Spent Resin Storage Tank and Phase Separator Tank Breakpot Radiological Content (Continued)

Isotope	SRST Breakpot Activity (Ci/cm^3)	PST Breakpot Activity (Ci/cm^3)
Tc99m	6.7E-09	1.3E-10
Tc99	2.1E-13	4.1E-15
Ru103	6.9E-12	1.4E-13
Ru105	2.3E-12	4.5E-14
Ru106	4.5E-12	8.9E-14
Rh103m	6.8E-12	1.4E-13
Rh105	4.8E-12	9.6E-14
Rh106	4.5E-12	8.9E-14
Ag110	5.2E-12	1.0E-13
Sb124	1.0E-14	2.0E-16
Sb125	8.9E-14	1.8E-15
Sb127	3.9E-13	7.8E-15
Sb129	4.8E-13	9.5E-15
Te125m	1.3E-11	2.6E-13
Te127m	4.2E-11	8.5E-13
Te127	1.7E-10	3.4E-12
Te129m	1.2E-10	2.4E-12
Te129	1.7E-10	3.4E-12
Te131m	4.0E-10	7.9E-12
Te131	2.0E-10	3.9E-12
Te132	2.9E-09	5.8E-11
Te133m	2.5E-10	4.9E-12
Te134	3.5E-10	7.0E-12
Ba137m	1.9E-08	3.8E-10
Ba139	6.5E-12	1.3E-13
Ba140	3.6E-11	7.1E-13
La140	1.0E-11	2.1E-13
La141	2.0E-12	4.0E-14
La142	9.6E-13	1.9E-14
Ce141	5.5E-12	1.1E-13
Ce143	4.1E-12	8.3E-14
Ce144	4.6E-12	9.2E-14
Pr143	4.9E-12	9.7E-14
Pr144	4.6E-12	9.1E-14
Np239	8.7E-11	1.7E-12
Na24	9.1E-09	1.8E-10
Cr51	5.2E-10	1.0E-11
Mn54	2.7E-10	5.3E-12
Fe55	2.0E-10	4.0E-12
Fe59	5.0E-11	1.0E-12
Co58	7.7E-10	1.5E-11
Co60	8.8E-11	1.8E-12
Ni63	4.4E-11	8.8E-13
Zn65	8.5E-11	1.7E-12
Zr95	6.5E-11	1.3E-12
Ag110m	2.2E-10	4.3E-12
W187	4.7E-10	9.3E-12
H3	2.8E-06	2.8E-06
C14	2.2E-10	2.2E-10
N16	1.5E-51	5.1E-08

Table 11.4-6: Spent Resin Storage Tank and Phase Separator Tank Breakpot Radiological Content (Continued)

Isotope	SRST Breakpot Activity (Ci/cm^3)	PST Breakpot Activity (Ci/cm^3)
Ar41	2.1E-07	2.1E-07

NOTE: The radiological content of the liquid in the SRST breakpot is the same concentration as primary coolant (Table 11.1-4). The PST breakpot liquid is the same concentration as CVCS outlet.

Figure 11.4-1: Block Diagram of the Solid Radioactive Waste System

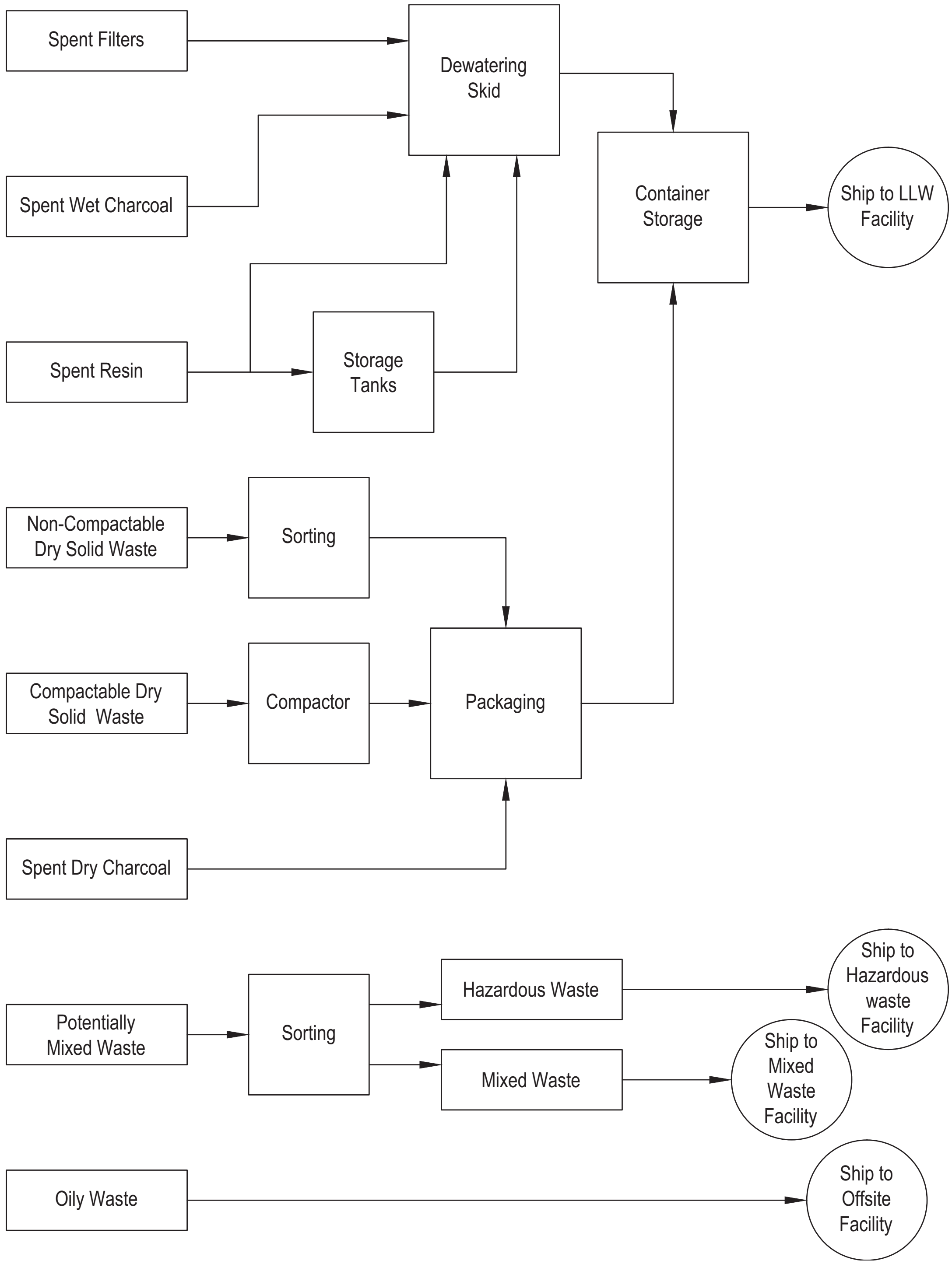


Figure 11.4-2a: Process Flow Diagram for Wet Solid Waste

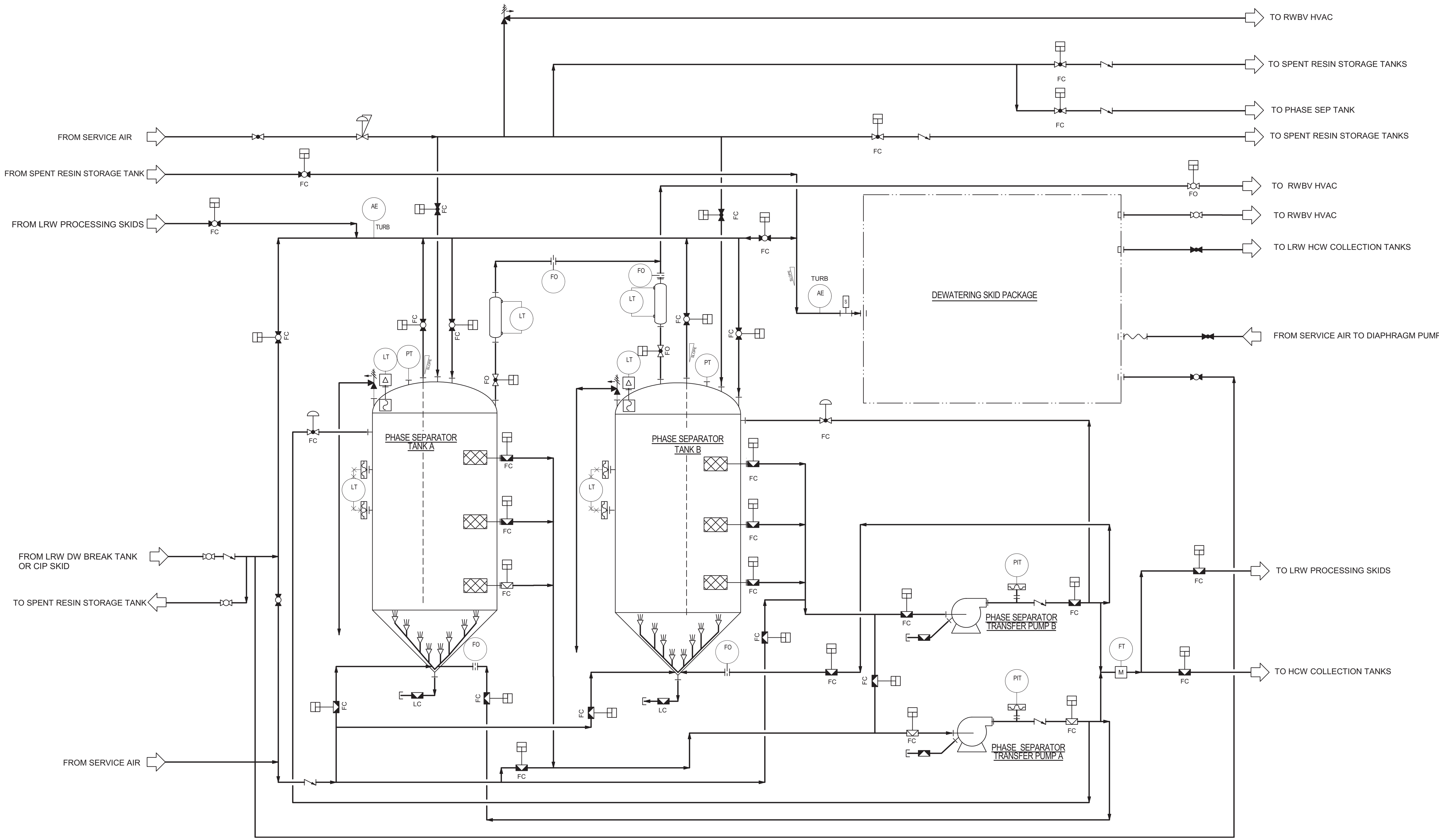
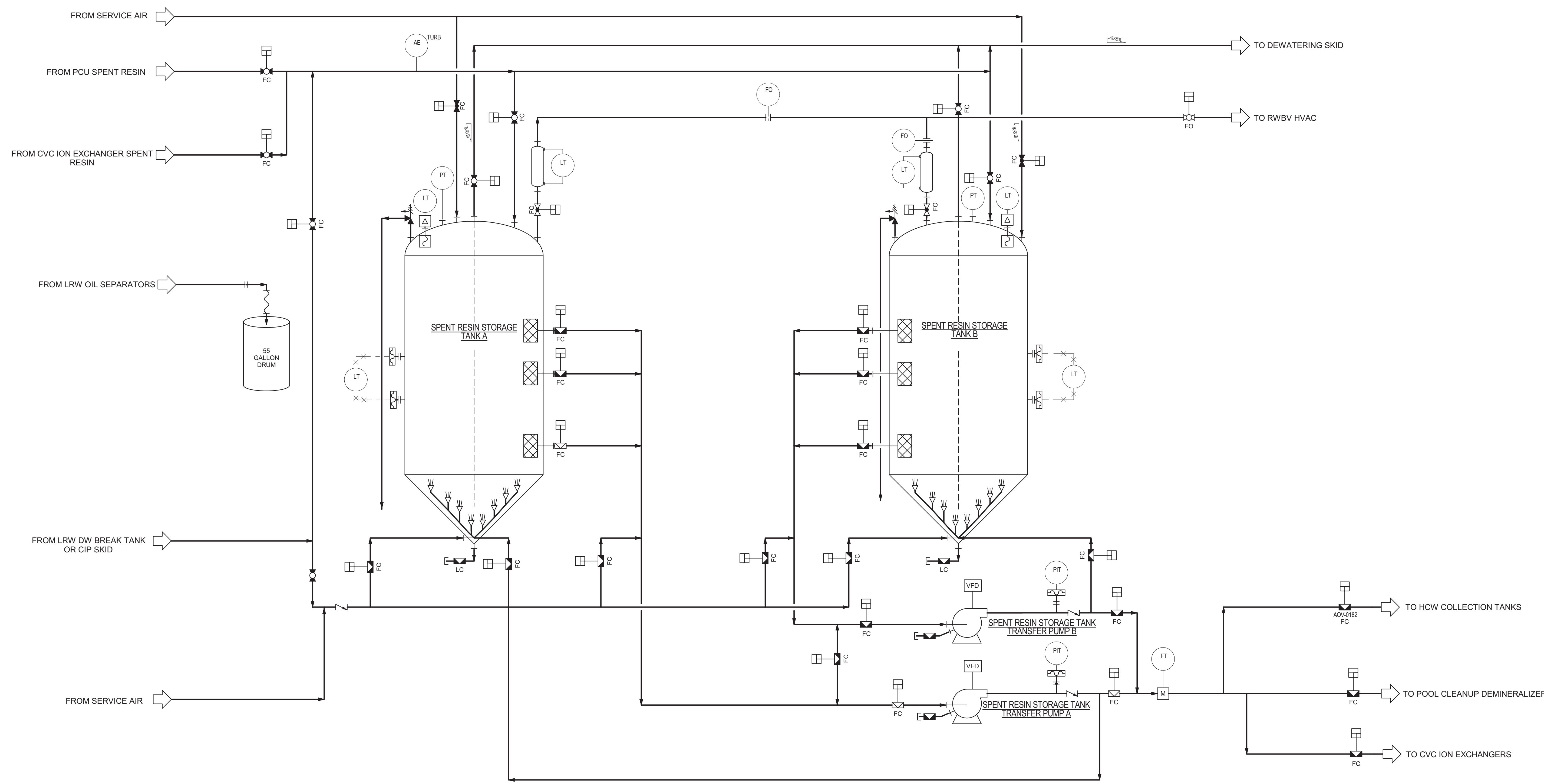


Figure 11.4-2b: Solid Radioactive Waste System Diagram



11.5 Process and Effluent Radiation Monitoring Instrumentation and Sampling System

The process and effluent radiological monitoring instrumentation and sampling design features provide the ability to detect and determine the content and, where required, the concentration and release rate of radioactive material in various gaseous and liquid process and effluent streams. The design features facilitate radiation monitoring and control, archiving, alarm functions and, where required, isolation and actuation functions to support the design objectives of the related system. The monitoring of in-plant radiation and airborne radioactivity is performed by the area radiation monitoring instrumentation described in Section 12.3.4.

The following functions are performed by process and effluent radiological monitoring instrumentation and sampling design:

- indicate and archive radioactive release rate levels and initiate alarms for effluent paths when system-specific thresholds are reached. Where required, provide signals to perform isolation functions to control releases of radioactivity to the environment. Where required, sampling capability is provided.
- indicate and archive process radiation levels and initiate alarms for process paths when system-specific thresholds are reached. Where required, provide signals to perform isolation or actuation functions to protect control room environmental conditions, isolate contamination sources, or protect process sample locations. Where required, sampling capability is provided.
- detect and determine the rate of reactor coolant system leakage into the containment vessel or secondary coolant.
- initiate system automated functions if predetermined process thresholds are exceeded to limit the release of radioactive materials to the environment, protect plant personnel, and support control room habitability.
- provide radiological data and design features to support the as low as reasonably achievable (ALARA) program and plant operations.

A description of the effluent and process radiological monitoring features, and sampling provisions is provided in Section 11.5.2 on a per-system basis.

11.5.1 Design Bases

The process and effluent radiological monitoring and sampling equipment is designed to meet the following design basis requirements.

- Monitor processes in plant areas during normal operations and anticipated operational occurrences (AOOs) such that the worker limits do not exceed the limits specified in 10 CFR 20.1201 and 10 CFR 20.1202.
- Monitor radiological effluents released to unrestricted areas during normal operations, AOOs and under accident and post-accident conditions so that the dose limits to the public do not exceed the limits specified in 10 CFR 20.1301, 10 CFR 20.1302 and 10 CFR 50, Appendix I.
- Provide monitoring and sampling and process controls such that doses in unrestricted areas from liquid and gaseous effluents are ALARA (10 CFR 50.34(a) and 10 CFR 50, Appendix I).

- Provide for data collection of liquid and gaseous effluents to unrestricted areas (10 CFR 50.36a).
- Provide design features to reduce the potential contamination of the facility and the environment, as well as reducing radioactive waste generation (10 CFR 20.1406).
- For the control room ventilation system, perform protective functions while withstanding the effects of natural phenomena without loss of function (GDC 2).
- Provide initiation signals for automated system functions to maintain control room habitability of the control room under accident conditions (PDC 19).
- Control the release of liquid and gaseous effluents from plant systems (GDC 60).
- Provide monitoring and sampling of fuel storage and handling areas, and radioactive waste management systems to detect excessive radiation levels, and initiate the appropriate automated system functions (GDC 61 and GDC 63).
- Provide monitoring of the effluent discharge paths, and the plant environs for radioactivity that may be released during normal operations, AOOs, and under post-accident conditions (GDC 64).
- Provide monitoring and sampling instrumentation for measuring and recording radiological data of noble gases at release points with continuous monitoring and sampling of radioactive iodine and particulates in gaseous effluents from accident release points in accordance with the requirements of 10 CFR 50.34(f)(2)(xvii).

11.5.2 System Description

The effluent and process radiation monitoring and sampling provisions are described on a per system basis within Section 11.5.2.1 and Section 11.5.2.2. The following system descriptions apply to the systems on a generic basis. Information that is unique to a specific effluent or process monitor including automated system actuations is provided within the individual system descriptions.

The radiation monitors provide a continuous indication and an archiving function to the main control room. When a specified setpoint is exceeded, the effluent and process radiation monitors provide a visual and audible alarm in the main control room, and where specified, locally and in the waste management control room. Alarms are designed such that they do not reset without operator action. The radiation monitors remain operable when the alarm setpoint is exceeded. The process and effluent radiation monitoring instrumentation provides self-monitoring to the extent that power failure or equipment failure causes an alarm in the main control room.

Mitigating actions for abnormal events where a potentially contaminated system is leaking into a normally non-radiologically contaminated system are specified in site procedures. If required, operators have the ability to manually isolate and sample the potentially contaminated system using isolation valves operated from the main control room or locally in the plant. The alarm setpoints, control room monitoring capability, system sampling capability, and operator response in accordance with site procedures conform with GDC 60 and RG 4.21 and ensure that the objectives of 10 CFR 20.1406 are met.

Sampling capability is provided for systems that may possess radioactive material content to

- determine process radionuclide content.
- validate radiological monitoring indication and alarms.
- serve as a redundant means of determining process or effluent radioactivity in the event that radiological monitoring is unavailable.
- provide information to determine the existence and rate of primary coolant leakage for selected systems.

Continuous monitoring and grab sampling site locations and design use applicable guidance from ANSI/HPS N13.1-2011, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities" (Reference 11.5-2), ANSI N42.18-2004, "Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity for Effluents" (Reference 11.5-1), Electric Power Research Institute (EPRI) 1022832, “Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines," Revision 4 (Reference 11.5-6), and RG 1.21 to ensure that effective and representative samples are obtained. The continuous and grab sampling provisions are designed to conform with Regulatory Guides 8.8 and 8.10 and enhance capability to meet ALARA goals. The provisions for sampling potentially contaminated fluid process systems include mechanical in-line samplers that draw directly from the process stream. Mechanical in-line samplers provide a representative sample, eliminate the need for sample lines, and reduce the possibility of personnel contamination. In addition, integrated sampling capability is provided within off-line radiation monitors that serve gaseous and fluid process systems. The off-line monitors and their sampling provisions return the sample streams to the process systems, and are designed to limit the possibility of personnel contamination. These design features, and the use of the radioactive waste drain system (RWDS) and liquid radioactive waste system for waste collection limit occupational exposure in accordance with 10 CFR 20.1201 and 10 CFR 20.1202, and limit contamination per 10 CFR 20.1406.

The off-line radiation detectors have taps for purging, flushing, or cleaning the sampling pathway within the detectors. The design of process and effluent radiation detectors permits removal without breaching the process system for repair, cleaning, calibration, and functional checks, as appropriate. The process and effluent radiation monitors contain built-in check sources for calibration and functional testing.

The process and effluent monitor calibration methods and frequency are in accordance with manufacturer recommendations and consider the rate at which instrument components age or become damaged. The calibrations are performed in a manner consistent with ALARA principles and follow the guidance of EPRI report TR-102644, “Calibration of Radiation Monitors at Nuclear Power Plants”(Reference 11.5-9). For effluent monitors, the guidance of RG 1.21 is used to determine the calibration requirements and the frequency of calibration is in accordance with manufacturer instructions. Recalibrations are performed on the detectors after maintenance or replacement of components that affect calibration.

Effluent radiation monitor setpoints are established for effluent streams containing multiple radionuclides considering whether instrumentation responses may change for radionuclide distributions that vary with the operating status of the plant or other factors. The methods used to establish setpoints follow the guidance of EPRI report TR-102644, (Reference 11.5-9).

The upper range of effluent radiation monitors is selected to ensure on-scale reading for exposure rates greater than the expected peak exposure rate and the low range is selected below the lower end of the anticipated measurement range but provides an on-scale value for the range of the instrumentation selected. Each effluent detector is designed with a minimum level of detectability for the nuclides being monitored in accordance with the guidance provided in ANSI N42.18-2004.

Radiation monitoring and archiving functions for effluent streams may contain hard to detect radionuclides that can be accounted for by use of easy to detect radionuclides. The ratio will be determined by the plant source term calculations and augmented as necessary by sample analysis. The methodology for calculation of radionuclide content and frequency of sampling will be determined by the Offsite Dose Calculation Manual (ODCM), which is described in Section 11.5.2.6.

The process and effluent radiation monitoring detectors are mounted in a shielded housing which decreases the influence of background radiation. Shielding the detector increases monitor sensitivity and optimizes the time response of the measuring system.

Radiation monitoring systems used to detect or quantify reactor coolant leakage into the secondary systems or the containment vessel monitor the Ar-41 concentration in process or effluent streams. To make the Ar-41 a more dependable isotope for primary-to-secondary leak rate monitoring, natural argon gas may be injected into the reactor coolant system to maintain the concentration of Ar-41 at a consistent level. The guidance of EPRI 1022832, (Reference 11.5-6), is used for determining primary coolant leak rates and establishing the baseline operating concentration of Ar-41 in the reactor coolant.

The radiological effluent and process monitoring data is capable of being supplied to the NRC operations center through the emergency response data system (ERDS) via a secure direct electronic data link in the event of an emergency. The ERDS connection is discussed in Section 7.2.13.

Except where otherwise noted, electrical power for the radiation monitors is provided from the normal DC power system (EDNS) electrical distribution system.

The quality and safety classification of the systems containing process and effluent radiation monitors is provided in Table 3.2-1.

The inspections, tests, analyses, and acceptance criteria associated with process and effluent radiation monitoring are described in Section 14.2 and Section 14.3.

The following tables and figures provide a summary of radiological monitoring.

- Detector information including number, type, location, and nominal range is provided in Table 11.5-1.
- Provisions for sampling are described in Table 11.5-2 and Table 11.5-3 for gaseous and liquid process streams, respectively.

- Estimated dynamic detection range, principle radionuclides measured, and basis for dynamic range are provided in Table 11.5-4.
- Figure 11.5-1 presents an integrated plant radiological monitoring drawing.
- Figure 11.5-2 provides a logic block diagram for radiation monitoring.
- Figure 11.5-3 provides an off-line radiation detection drawing.
- Figure 11.5-4 provides a process radiation adjacent-to-line detection drawing.
- Figure 11.5-5 provides a process radiation in-line detection drawing.
- Figure 11.5-6 provides a plant exhaust stack effluent radiation detection drawing.

COL Item 11.5-1: A COL applicant that references the NuScale Power Plant design certification will describe site-specific process and effluent monitoring and sampling system components and address the guidance provided in American National Standards Institute (ANSI) N13.1-2011, ANSI N42.18-2004, and Regulatory Guides 1.21, 1.33, and 4.15.

11.5.2.1 Effluent Radiation Monitoring

A description of effluent radiation monitoring and sampling equipment is provided below for systems with a potentially radioactive gaseous or liquid effluent stream. The applicable regulatory requirements are also addressed for each system.

11.5.2.1.1 Reactor Building HVAC System

The Reactor Building HVAC system (RBVS) exhaust fans remove air from the Reactor Building general area, Radioactive Waste Building, and Annex Building. The RBVS also conveys the process flow from the containment evacuation system (CES) to the plant exhaust stack, and gaseous radioactive waste system (GRWS) process flow via the Radioactive Waste Building HVAC system (RWBVS) to the plant exhaust stack. The RBVS plant exhaust stack is a continuously monitored gaseous radioactive effluent flow path that uses an off-line radiation monitor and integrated sampling system that measures and records exhaust stack flow, particulate, iodine and noble gases in three ranges. Stack flow is also continuously monitored to calculate and record radiological release rates. These design features ensure compliance with RG 1.21, 10 CFR 20.1301, and 10 CFR 20.1302. Stack flow measurement capability supports the consideration of atmospheric dispersion (χ/Q) and deposition (D/Q) factors when developing alarm setpoints.

The RBVS plant exhaust stack gaseous effluent radiation monitor provides continuous indication for effluent parameters and an alarm function via the plant control system (PCS) to the main control room when predetermined plant exhaust stack thresholds are exceeded. The alarms and indications alert operators to abnormal conditions to allow appropriate mitigating action. System monitoring and operator response in accordance with site procedures ensures that gaseous effluent content meets the objectives of 10 CFR 50 Appendix I and 10 CFR 20 prior to being released into the environment, and ensures compliance with GDC 60, 63, and 64.

The RBVS plant exhaust stack gaseous effluent radiation monitor setpoints are developed with sufficient margin to the operating effluent dose limits of 10 CFR 20 and 10 CFR 50, Appendix I. The actual setpoints are provided as part of the ODCM, which is described in Section 11.5.2.6.

The RBVS plant exhaust stack flow rate and noble gas, particulate, and halogen activity indications are post-accident monitoring system (PAMS) Type E variables as described in Table 7.1-7.

The RBVS system is described in Section 9.4.2.

11.5.2.1.2 Condenser Air Removal System

The exhaust of the condenser air removal system (CARS) vacuum pumps is monitored for radioactive effluents. For each condenser, there are two vacuum pumps each with its own separator tank from which gaseous process flow is discharged to the atmosphere via a common header. The common header for each condenser is provided with a single off-line particulate, iodine, and noble gas radiation monitor (PING) with grab sampling capability and a single adjacent-to-line Ar-41 monitor. The Ar-41 monitor is provided to detect and quantify reactor coolant system (RCS) leakage into the secondary systems. Both monitors provide continuous main control room and local monitoring capability. To provide early indication of primary to secondary leakage, a high radiation alarm setpoint is chosen that is as low as possible without causing spurious alarms in the main control room.

In addition to radiation monitoring, the CARS effluent flow rate is sensed and used to develop the effluent release rate. The CARS gaseous effluent radiation release rate is used to determine the rate of steam generator tube leakage using the guidance of EPRI 1022832, (Reference 11.5-6).

An additional high-high radiation alarm setpoint is provided in the main control room to detect increased degradation. Mitigating actions for steam generator tube leakage or failure events are contained in site procedures. If required, operators in the main control room have the ability to manually shut down and isolate the CARS in response to an abnormal plant condition. The alarm setpoints, control room monitoring capability, and operator response in accordance with site procedures ensures compliance with GDC 60 and 64 and ensures the objectives of 10 CFR 20 and 10 CFR 50 Appendix I are met.

The CARS gaseous flow and radiation monitors are classified as Post-Accident Monitoring System (PAMS) Type E variables as described in Table 7.1-7.

The CARS is described in Section 10.4.2.

11.5.2.1.3 Turbine Gland Sealing System

The exhaust of the turbine gland sealing condenser exhaust fans is monitored for radioactive effluents. For each turbine, there is a gland sealing steam condenser with two redundant exhaust fans that discharge to a common header. These headers combine to form a single discharge header for up to six turbines. A north and south Turbine Building turbine gland sealing system (TGSS) exhaust header is provided for the plant to direct the gaseous effluent from the TGSS to the atmosphere outside of the Turbine Building. Each discharge header is provided with a single off-line particulate, iodine, and noble gas radiation monitor (PING) with grab sampling capability and a single adjacent-to-line Ar-41 monitor. The Ar-41 monitor is provided to detect reactor coolant system leakage into the secondary systems. Both monitors provide continuous main control room and local monitoring capability. To provide early indication of primary to secondary leakage, a high radiation alarm setpoint is chosen that is as low as possible without causing spurious alarms in the main control room.

An additional high-high radiation alarm setpoint is provided in the main control room to detect increased degradation. Mitigating actions for steam generator tube leakage or failure events are contained in site procedures. If required, operators have the ability to manually shut down and isolate the TGSS in response to an abnormal plant condition. System monitoring and operator response in accordance with site procedures ensures that gaseous effluents meet the objectives of 10 CFR 20 and 10 CFR 50 Appendix I prior to being released into the environment, and ensures compliance with GDC 60 and 64.

The TGSS is described in Section 10.4.3.

11.5.2.1.4 Pool Surge Control System

The pool surge control system (PSCS) storage tank is provided with a single off-line radiation monitor with sampling capability at the vent of the pool surge control tank to monitor gaseous effluent released from the tank. Upon detection of a high radiation condition, the PSCS tank isolation valves to the dry dock and from the pool cleanup system are automatically closed, thereby limiting the release from the tank vent path due to changing water level. The radiation monitor output is sent to the plant control system (PCS) which supplies continuous display and alarm capability to the MCR. The high radiation alarm function provides the operators with a visual and audible indication to investigate and determine the cause of the alarm and take necessary corrective actions to rectify the condition. The setpoint for the high radiation alarm and automated function initiation is based on ensuring that the limitations of 10 CFR 20 and 10 CFR 50 are met for plant conditions. In this manner, compliance with the requirements of GDC 60, 61, and 64 are ensured.

The PSCS is described in Section 9.1.3.2.

11.5.2.1.5 Liquid Radioactive Waste System

The liquid radioactive waste system (LRWS) processes liquids that are contaminated with radioactive products. When liquid process releases from the LRWS are required, effluents from the low-conductivity, high conductivity, and detergent waste collection tanks are pumped through two redundant adjacent-to-line radiation monitors prior to being directed to the utility water system (UWS). The LRWS control logic sends an isolation signal to redundant isolation valves to terminate the LRWS effluent release and provides the main and waste management control rooms with an alarm of the condition in the event of:

- sensed radiation levels on either one of two LRWS effluent radiation monitors higher than the predetermined threshold permissible for planned liquid release
- loss of UWS flow or UWS flow below predetermined threshold permissible for the planned liquid release
- sensed leakage within a pipe section of the system

The LRWS effluent release isolation valves fail in the closed position upon loss of air or power.

The LRWS effluent radiation monitor and isolation functions are designed to meet the requirements of GDC 60, GDC 61, GDC 64 and 10 CFR 20, 10 CFR 50 Appendix I, and 10 CFR 50.34(a). The LRWS effluent radiation monitor reads and records the radiation levels of liquid materials being released to the environs. The UWS system flow to the environs is monitored to ensure that plant liquid effluent at the point of discharge remains below the specified site radiological concentration limitations for release. UWS flow monitoring permits the operator to establish appropriate setpoints for alarm and isolation functions. In addition, the UWS plant liquid effluent path is a monitored release liquid path. In this manner the operator can comply with RG 1.21, the requirements of 10 CFR 20.1302, and document compliance with 10 CFR 20.1301.

LRWS tanks are recirculated at least three full tank volumes prior to sampling to obtain a representative sample and the tanks are analyzed prior to commencement of the liquid release per RG 1.21. The sample site is located on the tank recirculation line and sampled while the pump is in service to ensure a representative sample. The LRWS sampling is performed by a mechanical sampler that takes a representative sample from the process stream directly from the tank recirculation line and does not require a cooling interface or purging function. This sampling device reduces process sample collection time, and the sampler seal design reduces potential for personnel exposure to radioactive materials. LRWS tank recirculation pump capacity is enhanced with in-tank recirculation line eductors that provide the capability to exchange five tank volumes within an eight-hour period.

The LRWS tank rooms are provided with area radiation monitoring that provides alarm functions to the MCR. This feature alerts the operators to off normal conditions allowing them to take corrective actions in a timely manner, thus satisfying in part the requirements of GDC 63.

The LRWS is described in Section 11.2.

11.5.2.1.6 Utility Water System

The utility water system (UWS) receives the effluent from the site cooling water system (SCWS) cooling tower basin, circulating water system (CWS) cooling tower basins, balance-of-plant drain system (BPDS), potable water system discharge, site drainage system and liquid radioactive waste system. The UWS site discharge path is continuously monitored for radiation by an off-line monitor with sampling capability. The discharge flow rate is measured to ensure sufficient flow for the LRWS effluent releases and to derive an overall release rate to the environs. The UWS effluent radiation monitor is designed to meet the requirements of GDC 64 and 10 CFR 50.34(a) and it reads and records the liquid materials being released to the environs. In this manner, the operator can comply with RG 1.21, 10 CFR 20.1301, and 10 CFR 20.1302.

The UWS liquid effluent radiation monitor provides continuous indication of effluent parameters. An alarm is provided in the main control room and the waste management control room via the plant control system when predetermined system thresholds are exceeded. The alarms and indications ensure the operators are alerted to abnormal conditions to allow appropriate mitigating actions. The numerical setpoints are generated during the development of the offsite dose calculation manual to ensure that the limitations of 10 CFR 20 and 10 CFR 50 are not exceeded. System manipulation and component isolation from a high radiation condition is a manual process and an operator action in response to an abnormal plant condition. System monitoring and operator response in accordance with site procedures ensures that liquid effluents meet the objectives of 10 CFR 50 Appendix I and 10 CFR 20 prior to being released into the environment, and ensures compliance with GDC 60 and 64.

The UWS is described in Section 9.2.9.

11.5.2.1.7 Circulating Water System

The liquid effluent from the circulating water system (CWS) is continuously monitored for radiation via the UWS prior to its exit to the environs. Effluent radiation monitoring for the UWS is described in Section 11.5.2.1.6.

The CWS is not expected to contain radioactive materials, and is not expected to become contaminated. Contamination of the main condenser could occur as a result of steam generator tube leakage, however, the CWS is at a higher pressure than the main condenser, thereby preventing the release of radioactive materials into the CWS. To ensure the system effluent is continuously monitored, the cooling tower blowdown line, cooling tower overflow line, and storm drains adjacent to the cooling towers are routed to the UWS where the liquid effluent is continuously monitored for radioactive materials at the point of release. Continuous effluent monitoring via the UWS and a programmatic approach to periodic sampling of the CWS liquid process ensure compliance with GDC 64.

Provisions for system sampling serve as a means of determining process radionuclide content. It is necessary to sample and analyze the system for radioactive material content on a periodic basis as this is considered a less-significant release point as characterized in RG 1.21 and to verify the effluent monitor readings for the utility water system. The system is sampled for radioactive material content on a periodic basis to verify the effluent monitor readings for the UWS. The frequency is determined by the offsite dose calculation manual (ODCM) described in Section 11.5.2.6. Samples of the CWS are taken periodically from the sample points located in the cooling tower basins where turbulent flow ensures a representative sample. Provisions for CWS sampling are described in Table 11.5-3.

The CWS is described in Section 10.4.5.

11.5.2.2 Process Radiation Monitoring

For each plant system that is potentially radiologically contaminated within its process liquid or gas, a description of its radiation monitoring and sampling equipment is provided. The applicable regulatory requirements for each system are considered, and a description of how these requirements are met is provided for each system.

11.5.2.2.1 Normal Control Room HVAC System

The normal control room HVAC system (CRVS) protects personnel from exposure to radiation during abnormal and accident conditions initially by removing radioactive contamination from outside air via charcoal filtration. If conditions degrade and radiation levels are detected that exceed the capability of the charcoal filtration units or if power is not available, the environment within the control room boundary is isolated and pressurized with a bottled air supply from the control room habitability system (CRHS).

Three in-line radiation monitors are located upstream of the CRVS filter unit and the outdoor air isolation dampers that are in parallel with the CRVS filter unit. Upon detection of a high radiation level in the outside air intake, the system is realigned so that 100 percent of the outside air passes through the CRVS filter unit, containing high-efficiency particulate air and charcoal filters to process outside air and minimize radiation exposure to personnel within the control room boundary.

If air conditions are further degraded as sensed by two off-line radiation monitors downstream of the CRVS filter unit or if power is not available, the CRVS filter unit is stopped, the outside air intake is automatically isolated, the operating supply air handling unit is stopped, the general exhaust fan is stopped, the CRB battery room exhaust fan is stopped, and the CRHS system is initiated to facilitate continued control room habitability consistent with PDC 19. Section 6.4.3.2 provides a complete discussion of the CRHS system initiation.

The radiation monitors that initiate the isolation for operation of the CRHS have augmented quality assurance requirements. The augmented quality requirements are specified in Table 3.2-1.

The CRVS radiation monitors provide continuous display and alarm capability to the MCR and provide the signal for the CRVS and CRHS automated functions. To ensure accurate and representative indication, these in-line and off-line monitors are designed to meet the guidance of ANSI/HPS N13.1-2011 (Reference 11.5-2).

Provisions for system sampling allow determination of process radionuclide content and serve as a redundant means of determining process radioactivity in the event that radiological monitoring is unavailable. Although not considered to be a contaminated system, grab sampling provisions via the off-line radiation monitoring system conform with RGs 8.8 and 8.10 and enhance plant staff capability to meet ALARA goals.

Electrical power for the radiation monitors downstream of the CRVS charcoal filters is provided by the highly reliable DC power system (EDSS). Electrical power for the radiation monitors upstream of the CRVS charcoal filters is provided by the normal DC power system (EDNS). If a loss of AC power occurs, the backup power supply system (BPSS) auxiliary AC power source (AAPS) supplies power to the CRVS loads.

11.5.2.2.2 Control Room Habitability System

The CRHS is initiated by two off-line control room ventilation system radiation monitors described in Section 11.5.2.2.1. Section 6.4.3.2 provides a complete discussion of the CRHS system initiation.

11.5.2.2.3 Reactor Building HVAC System

The RBVS provides an exhaust path for the Reactor Building heating ventilation and air conditioning (HVAC) general area, RWBVS, and the Annex Building HVAC system (ABVS). Each of these areas has a single main exhaust duct that feeds into a plenum served by the RBVS exhaust fans. The individual area exhaust ducts are provided with off-line process radiation monitors, each with built-in sampling capability. The monitors provide continuous monitoring capability and alarm functions both locally and in the MCR. These design functions ensure the operators detect abnormal plant ventilation radiological conditions within those areas and take mitigating actions per site procedures. In this manner compliance with GDC 63 and 64 and 10 CFR 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii) is ensured. The high radiation alarm setpoints are designed to alert plant personnel when radioactivity in the RBVS process flow reaches levels that have been determined to be greater than normal, and provide an alarm function below the thresholds of 10 CFR 20 and 10 CFR 50 limitations. The actual setpoints are provided in the ODCM described in Section 11.5.2.6.

The RBVS spent fuel pool exhaust radiation monitors provide continuous display and alarm capability to the MCR and provide the initiation signal for the RBVS spent fuel pool exhaust filter units charcoal filters and the isolation signal to the RBVS spent fuel pool area damper.

The spent fuel pool exhaust air is filtered by the RBVS spent fuel pool exhaust fan high-efficiency particulate air filters located in the Reactor Building and exhausted from the RBVS exhaust stack in the Radioactive Waste Building. Within the common suction duct of the RBVS spent fuel pool exhaust fans, three off-line radiation detectors monitor the process flow. Upon a detection of radiation above predetermined limits within the spent fuel pool area, the system initiates an alarm in the main control room, the spent fuel pool exhaust air is diverted to the charcoal adsorbers within the spent fuel pool exhaust fans, and the duct for the Reactor Building general exhaust for the reactor pool and dry dock area is isolated. This automated function ensures RBVS compliance with GDC 60, 61, 63, and 64. The

RBVS automated function initiation setpoint is consistent with ensuring system actuation prior exceeding the limitations of 10 CFR 20 and 10 CFR 50 Appendix I.

The isolation dampers located in the normal flow path of the Reactor Building HVAC spent fuel pool exhaust ductwork fail to the open position. The isolation dampers located in the ductwork of the secondary flow path, containing the charcoal filters, fail to the closed position. This allows a passive high-efficiency particulate air-filtered vent path for the atmosphere within the Reactor Building and provides a monitored release path to the environment for the potentially contaminated air.

The RBVS is described in Section 9.4.2.

11.5.2.2.4 Radioactive Waste Building HVAC System

The RWBVS is monitored for radiation at the exit of the process stream by the Reactor Building HVAC System process radiation monitor described in Section 11.5.2.2.3. The RWBVS exhaust duct to the RBVS exhaust fan suction plenum is provided with an off-line process radiation monitor with built-in sampling capability. The monitor provides continuous monitoring capability and alarm functions both locally and in the MCR. These design functions ensure the operators detect abnormal plant ventilation radiological conditions within the RWBVS and take mitigating actions per plant procedures. Provisions for sampling within the process radiation monitor allow determination of process radionuclide content and serve as a redundant means of determination of process radioactivity in the event that radiological monitoring is unavailable. In this manner compliance with GDC 63 and 64, 10 CFR 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii) is ensured.

The RWBVS effluent is monitored by the RBVS plant vent effluent radiation monitor described in Section 11.5.2.1.1.

The RWBVS is described in Section 9.4.3.

11.5.2.2.5 Annex Building HVAC System

The Annex Building HVAC system (ABVS) is monitored for radiation at the exit of the process stream by the Reactor Building HVAC system process radiation monitor described in Section 11.5.2.2.3. The ABVS exhaust duct to the RBVS exhaust fan suction plenum is provided with an off-line process radiation monitor with built-in sampling capability. The monitor provides continuous monitoring capability and alarm functions both locally and in the MCR. These design functions ensure the operators detect abnormal plant ventilation radiological conditions within the ABVS and take mitigating actions per site procedures. Provisions for sampling within the process radiation monitor allow determination of process radionuclide content and serve as a redundant means of determination of process radioactivity in the event that radiological monitoring is unavailable. In this manner compliance with GDC 63 and 64 and 10 CFR 50.34(f)(2)(xvii) and 10 CFR 50.34(f)(2)(xxvii) is ensured.

The ABVS effluent is monitored by the RBVS plant vent effluent radiation monitor described in Section 11.5.2.1.1.

11.5.2.2.6 Gaseous Radioactive Waste System

The GRWS treats the radioactive gases originating within the reactor vessel and released via the chemical and volume control system (CVCS) to the LRWS degasifier. The CES may also provide infrequent inputs to the GRWS during conditions of abnormally high process radiation levels.

The following system isolation functions are supported by radiation detection instrumentation and ensure system compliance with GDC 60, 63, 64, 10 CFR 20.1406, and 10 CFR 50.34 (f)(2)(xxvii).

- A single off-line radiation monitor is located at the exit of each of the GRWS decay beds. Upon detection of process radiation levels greater than the high alarm setpoint or loss of system power, the system isolates flow for the associated bed.
- A single off-line radiation monitor is located at the exit of the GRWS on the line to RWBVS. Upon detection of process radiation levels greater than the high alarm setpoint or loss of system power, the system isolates flow to RWBVS.
- Airborne radiation detection instrumentation is located in each GRWS charcoal bed cubicle for system leakage detection. The system isolates the GRWS and opens the nitrogen purge valve upon detection of high airborne concentrations or loss of system power. Area radiation and airborne radiation monitoring are described in Section 12.3.4.
- Loss of Reactor Building HVAC exhaust system flow isolates GRWS flow to the RWBVS. This isolation function is described in Section 11.3.

The automated functions of the GRWS and effluent flow through charcoal beds ensure that gaseous effluents meet the objectives of 10 CFR 50 Appendix I and 10 CFR 20 prior to being released into the environment. Radiation monitors of the GRWS send signals to the PCS providing display and alarm signals to the main and waste management control rooms.

The effluent from the GRWS is directed to the RWBVS and ultimately to the RBVS where it is monitored by the RBVS exhaust stack effluent radiation monitor prior to release, ensuring compliance with GDC 64 and 10 CFR 50.34(f)(2)(xvii). To ensure that GRWS releases are monitored, and that gaseous radioactive materials are controlled within the plant environs, the GRWS isolates upon loss of Reactor Building HVAC exhaust flow.

The GRWS is described in Section 11.3.

11.5.2.2.7 Containment Evacuation System

The containment evacuation system (CES) process stream is monitored to determine the integrity of the reactor coolant pressure boundary using two methods:

- radiological monitoring of the gaseous discharge of the vacuum pump and CES sample vessel
- measurement of condensed water vapor collected in the CES sample vessel

The off-line particulate, iodine and noble gas radiation monitor and the adjacent-to-line Ar-41 monitor installed on the gaseous process stream of the CES vacuum pump can be used to detect pressure boundary leakage or fuel failure. A high radiation condition from either of these monitors initiates an alarm in the main control room, which prompts the operators to investigate by using the CES sample vessel flow rate to validate the condition.

To provide an early indication of reactor coolant system leakage, an alarm setpoint is chosen that is set sufficiently low to detect anomalies without causing spurious alarms in the MCR. The use of Ar-41 for primary system leak detection provides a method that does not rely upon fuel leakage or reactor coolant system impurities, and is not affected by plate-out of particulate radionuclides on the internal surfaces of the containment vessel and other environmental factors.

Water vapor removed from the containment vessel using the CES vacuum pumps is condensed and collected in the CES sample vessel. An adjacent-to-line radiation monitor is installed on the sample vessel and in the event of a high radiation condition, the system provides an alarm in the MCR alerting the operators of a potential reactor coolant system leak or possible fuel clad degradation. The setpoint for this alarm is adjusted to values that are low enough for the minimum detectable activity anticipated and high enough not to give false alarms.

An installed mechanical liquid grab sampler located downstream of the CES sample vessel allows for samples to be taken and analyzed in the laboratory for a more finite definition of the radionuclide content of the condensate, and to serve as a redundant means of measuring liquid process radiation levels. The mechanical sampler is designed to conform with RGs 8.8 and 8.10 and enhance plant staff capability to meet ALARA goals and contamination control in accordance with10 CFR 20.1406. Compliance with Regulatory Guide 1.45 requirements and the capabilities of the CES sample vessel are discussed in Section 5.2.5.

The off-line particulate, iodine and noble gas radiation monitor used to detect radioactivity in the gaseous discharge from the CES vacuum pump to the RBVS also sends signals to the module control system (MCS) to initiate automatic system functions. Upon detection of a high radiation condition in the CES gaseous process stream, failure of the radiation monitor, or loss of system power, the CES vacuum pump discharge flow path transfers from the RBVS to the gaseous radioactive waste system and the CES air inputs isolate. In addition, the same signal isolates the CES from the plant sampling system sample stream. The radiation monitor output is sent to the module control system which provides continuous display and alarm capability to the MCR and a local alarm capability at the PSS sample panel to ensure personnel protection. The setpoint for the high radiation alarm and automated function initiation is based on ensuring that the limitations of 10 CFR 20 and 10 CFR 50 are met for plant conditions. These design features ensure compliance with GDC 60, 61, and 64, and the applicable 10 CFR 20 and 10 CFR 50 requirements and limitations.

In addition to local grab sampling capability, process sampling for the CES is also provided by the PSS. Section 9.3.2 describes the design and functions of the PSS.

The CES is described in Section 9.3.6.

11.5.2.2.8 Main Steam System

The two main steam lines for each module are each provided with two adjacent-to-line radiation monitors to detect Ar-41. The main steam system (MSS) radiation monitors provide continuous MCR indication and alarm capability. The MSS is normally not radioactive, and an increased level of radiation in the MSS could indicate steam generator tube leakage or failure. To provide an early indication of primary to secondary system leakage, the high radiation alarm setpoint is set sufficiently low to detect leakage without causing spurious alarms in the MCR.

An additional high-high radiation alarm setpoint is provided in the main control room to detect increased degradation. Mitigating actions for steam generator tube leakage or failure events are contained in site procedures. If required, operators in the main control room have the ability to isolate the MSS in response to an abnormal plant condition. The alarm setpoints, control room monitoring capability, and operator response in accordance with site procedures ensures compliance with GDC 60, and 64 and ensures that the objectives of 10 CFR 20 and 10 CFR 50 are met.

MSS sampling is provided by the PSS as a secondary means to determine MSS radiological content and to detect steam generator tube leakage. Section 9.3.2 describes the PSS interface for MSS sampling.

The MSS is described in Section 10.3.

11.5.2.2.9 Containment Flooding and Drain System

The containment flooding and drain system (CFDS) drain separator tank gaseous discharge to the Reactor Building ventilation system is monitored by an off-line radiation monitor with built-in sampling capability. Upon detection of a high radiation condition in the CFDS gaseous process stream, radiation monitor failure, or loss of power, the CFDS drain separator tank discharge line to the RBVS is automatically isolated. The radiation monitor output is sent to the module control system (MCS) which provides continuous display and alarm capability to the main control room. The setpoint for the high radiation alarm and automated function initiation is based on ensuring that the limitations of 10 CFR 20 and 10 CFR 50 are met for plant conditions. In this manner compliance with the requirements of GDC 60, 61, 63, and 64 are ensured.

The CFDS is described in Section 9.3.6.

11.5.2.2.10 Process Sampling System

There are no process radiation monitoring instruments within the process sampling system design. To enhance worker protection and ALARA objectives, the PSS is served by the radiological monitoring functions of the sampled systems for each module.

- A high radiation condition on the containment evacuation system gas discharge line will result in main control room and local process sampling system panel alarms. The containment evacuation system will automatically close the isolation valve to the process sampling system sample line. The radiation monitoring, high radiation alarm, and automated isolation function is provided by the instrument and control system of the containment evacuation system.
- A high radiation condition on the reactor coolant suction line to the CVCS will result in main control room and local process sampling system panel alarms. The CVCS will automatically close the isolation valve to the process sample system sampling line. The radiation monitoring, high radiation alarm, and automated isolation function is provided by the instrument and control system of the CVCS.
- A high radiation condition on the A or B main steam lines will result in main control room and local process sampling system panel alarms. The high radiation monitoring and alarm function is provided by the instrument and control system of the MSS.

The CVCS, containment evacuation system, and MSS radiation monitors provide continuous indication and alarm capability to the main control room via the MCS. In addition, a high radiation condition on these monitors will initiate an alarm at the applicable plant sample system panel to alert workers of the high radiation condition in the process system. The alarm set points, design features and automated isolation features ensure compliance with GDC 60, RG 8.8 and the principles and limits of 10 CFR 20.1201, 10 CFR 20.1202 and 10 CFR 20.1406.

The PSS is described in Section 9.3.2 including component quality group classification and seismic requirements. Sampling requirements for plant systems served by the process sample system are described in Table 11.5-2 and Table 11.5-3.

11.5.2.2.11 Chemical and Volume Control System

An independent CVCS is provided for each NPM, and each system has two reactor coolant filters. Area radiation monitors are located adjacent to each of the two filters. The monitors provide continuous indication to the MCR of the buildup of radioactive materials on the filter media. If the radiation level for the in-service filter exceeds the predetermined limit, an alarm is provided in the MCR to prompt the operators to direct the CVCS process flow to the redundant filter and to replace the filter media. This activity is a manual process and is directed by site procedures in response to the alarm condition. The alarm setpoint is consistent with ALARA occupational dose consequence, and limits the radiation dose to workers for the filter replacement task. The alarm setpoints, system design features, and manually performed actions ensure CVCS filter maintenance compliance with RG 8.8 and ensure compliance with the principles and limits of 10 CFR 20.1201, 10 CFR 20.1202 and 10 CFR 20.1406.

The CVCS filter area radiation detectors provide for monitoring in a broad range of operating and accident conditions, in part ensuring compliance with 10 CFR 50.34(f)(2)(xxvii). The range of the radiation monitor is chosen so that the upper end of the scale is high enough to ensure an on-scale reading for exposure rates far greater (approximately two decades) than the expected peak exposure rate and the low end is at the lower end of the expected exposure rate range but provides an on-scale value for the range of the instrumentation selected.

The CVCS filter area radiation monitors are located outside of the filter shielding and do not penetrate the process fluid boundary. The electronic controls for the monitors are placed in a low dose area. The area radiation monitor contains a built-in check source for calibration and functional testing and the monitor design ensures the source is returned to the non-test mode upon deactivation or loss of power to the monitor. The area radiation monitor design features and configuration facilitate ALARA considerations for both monitor maintenance and calibration.

Area radiation monitor calibrations are performed in accordance with manufacturer instructions and the guidance of Regulatory Guide 1.21. The energy and range accuracy is documented by the manufacturer for each area monitor. The method for determining calibration accuracy is defined in the calibration procedure. These calibration methods follow the guidance of EPRI report TR-102644, (Reference 11.5-9). The calibration is performed in a manner consistent with ALARA principles. The same documents provide guidance for the frequency of monitor calibration. In addition to written guidance, component age, performance, and condition are considered in determining the frequency and scope of calibrations. Functional tests of the CVCS area radiation monitors are performed in accordance with manufacturer requirements and use the guidance of EPRI Report TR-104862 (Reference 11.5-7).

Area radiation monitoring detailed design features, quality group, seismic requirements, and functions are described in Section 12.3.4. CVCS detailed design features, functions, and flow paths are described in Section 9.3.4.

To provide CVCS process monitoring and worker protection, a single adjacent-to-line radiation monitor is provided on the CVCS suction line from the reactor coolant system, upstream of the regenerative heat exchanger. A high radiation signal or loss of power to the monitor results in MCS automatically closing the normally open process sample system line isolation valve. The MCS also initiates an alarm in the MCR and at the applicable PSS panel.

The CVCS process radiation monitoring functions and automated functions enhance staff capability to meet ALARA goals. Additionally, this monitor provides continuous monitoring indication and may provide indication of fuel failure, as it is located directly on the CVCS suction line from the reactor coolant system. The sample line and module heatup isolation are discussed in Section 11.5.2.2.14, and alarm design features are consistent with GDC 60, 64, and 10 CFR 50.34(f)(2)(xxvii).

Sample connections are provided from the CVCS to the process sampling system in three locations for each module: downstream of the reactor coolant suction line to CVCS, downstream of the CVCS reactor coolant filters and downstream of the CVCS regenerative heat exchanger prior to the return to the reactor coolant system. Process sampling system features and functions are described in Section 9.3.2.

The CVCS is described in Section 9.3.4.

11.5.2.2.12 Reactor Component Cooling Water System

The reactor component cooling water system is a closed loop cooling system. It is an intermediate system between the radiologically contaminated systems in the Reactor Building and the normally nonradioactive SCWS. For each NuScale Power Module, a single adjacent-to-line radiation monitor is provided for the following loads:

- downstream of the containment evacuation system vacuum pumps A and B coolers, and the containment evacuation system condenser cooler
- downstream of the process sampling system analyzer cooler and cooling water temperature control units, and the chemical and volume control non-regenerative heat exchanger

The reactor component cooling water system radiation monitors provide continuous main control room and waste management control room indication and alarm capability. The reactor component cooling water system does not normally contain radioactive process fluid. An increased level of radiation in the RCCWS could indicate heat exchanger leakage or component cooler failure. To provide an early indication of leakage from radiologically contaminated loads to the clean reactor component cooling water system, the high radiation alarm setpoint is set as low as possible without causing spurious alarms in the main control room.

System provisions for process fluid grab sampling are located within the isolation boundary of each reactor component cooling water system load to allow the operators to determine the source and content of the radioactive contamination. The ability to isolate and sample potentially contaminated systems ensures compliance to the occupation exposure limits in accordance with 10 CFR 20.1201 and 10 CFR 20.1202, and limits the spread of contamination per 10 CFR 20.1406.

The RCCWS is described in Section 9.2.2.

11.5.2.2.13 Site Cooling Water System

The site cooling water system (SCWS) is a closed loop cooling system with external cooling towers that serves both radiologically contaminated and non-contaminated systems. An off-line process radiation monitor provided with built-in sampling capability is located downstream of each heat exchanger where there is a possibility of contaminated system leakage into the SCWS, in the outlet of the reactor pool cooling system heat exchangers, the spent fuel pool cooling system heat exchangers, and the reactor component cooling heat exchangers.

The discharge from the SCWS cooling tower basin is routed to the utility water system, which is described in Section 11.5.2.1.6.

To provide the control room with a means to determine if the SCWS is discharging radiologically contaminated water to the utility water system, the SCWS cooling tower basin blowdown line is provided with a single off-line radiation monitor with sampling capability and the cooling tower basin overflow lines is provided with a single adjacent-to-line process radiation monitor. The monitors provide continuous indication and a high radiation alarm function to the main control room.

To limit the release of radioactive materials from the SCWS to the utility water system the SCWS blowdown radiation monitor initiates an alarm in the main control room and isolates the cooling tower blowdown line to the utility water system upon detection of a high radiation condition in the process flow. The alarm and isolation setpoints ensure that the limitations of 10 CFR 20 and 10 CFR 50 Appendix I are not exceeded. The requirements of GDC 60 and 64, and 10 CFR 20.1406 are in part met by these design features.

The SCWS process radiation monitors provide continuous indication and alarm functions to the main control room and waste management control room. The SCWS is normally not radioactive. An increased level of radiation in the SCWS could indicate heat exchanger leakage or failure. To provide an early indication of leakage from radiologically contaminated loads to the clean SCWS, the high radiation alarm setpoint is set as low as possible without causing spurious alarms in the main control room. These system features ensure the operators are alerted to abnormal conditions to allow appropriate mitigating actions.

Mechanical sampling provisions are incorporated into the SCWS and are located downstream of potential contamination sources within their respective isolation boundaries. The samplers draw directly from the process stream and provide a representative sample. The mechanical samplers eliminate the need for sample lines and reduce the possibility of personnel contamination. For the sections of the system that could possibly become contaminated, the SCWS piping design facilitates safe draining of the contamination to the radioactive waste drains. To minimize the generation of radioactive waste, the process sample stream for continuous off-line radiation monitoring is returned to the process system. These design features in part ensure compliance for occupation exposure in accordance with 10 CFR 20.1201 and 10 CFR 20.1202, and limit contamination per 10 CFR 20.1406.

The SCWS is described in Section 9.2.7.

11.5.2.2.14 Auxiliary Boiler System

The auxiliary boiler system (ABS) provides two high pressure boilers that supply steam to the module heatup system (MHS). The associated ABS radiological monitoring equipment is designed to monitor and control radioactive releases to the environment and to detect and mitigate leakage from potentially radiologically contaminated systems to clean systems. The ABS radiation monitors provide the MCR continuous indication and alarm functions and isolation functions for specific process streams.

A single adjacent-to-line radiation monitor is located on each of the ABS exits from MHS heat exchangers 0A and 0B. If radiation is detected in the ABS greater than the high radiation alarm setpoint, the system initiates a MCR alarm notifying the operators to investigate and initiate mitigating actions. If radiation is detected in the ABS that is greater than the high-high radiation isolation setpoint or if system power is lost, the system initiates a control room alarm and the CVCS isolation valves for the affected MHS heat exchanger close.

A single adjacent-to-line radiation monitor is located on the vent for the high pressure ABS flash tank. If radiation is detected in the ABS greater than the high radiation alarm setpoint, the system initiates a MCR alarm notifying the operators to investigate and initiate mitigating actions. If radiation is detected in the ABS that is greater than the high-high radiation isolation or if system power is lost, the ABS flash tank pressure regulating valve and the steam supply valves from both boilers isolate.

A single adjacent-to-line radiation monitor is located on the cross-tie line from the high-pressure to low-pressure ABS. If radiation is detected in the ABS greater than the high radiation alarm setpoint, the system initiates a MCR alarm notifying the operators to investigate and initiate mitigating actions. If radiation is detected in the ABS that is greater than the high-high radiation isolation or if system power is lost, the high-pressure to low-pressure ABS cross-tie valves isolate.

For radiation monitors in the ABS, Ar-41 is the radionuclide selected for primary coolant leak detection.

The ABS radiation monitors have local indication and send signals to the MCS which provides display and alarm signals to the main control room and initiates isolation functions. To provide an early indication of primary to secondary leakage, the high alarm setpoint is chosen for the radiation monitor that is set as low as possible without causing spurious alarms in the main control room. The high-high alarm setpoint ensures that the limitations of 10 CFR 20 and 10 CFR 50 Appendix I are not exceeded. The requirements of GDC 60 and 64, and 10 CFR 20.1406 are met by these design features.

The ABS blowdown is directed to the balance of plant drain system (BPDS) north waste water sump tank. A radiation monitor is located on the line to monitor the ABS process flow to the BPDS. If a high radiation condition is detected, an alarm is initiated in the MCR, the north waste water sump pumps automatically shut down and the discharge flow path to the balance of plant drain system collection tanks automatically isolates. If required, the discharge flow path of the north waste water sump pumps can be opened to the liquid radioactive waste system high-conductivity waste tank and the sump pumps can be restored to automatic operation to facilitate processing radiologically contaminated water and system flushing. The effluent from both systems is ultimately monitored by the UWS liquid effluent radiation monitor prior to release. In this manner the operator can comply with RG 1.21, comply with the requirements of 10 CFR 20.1302, and document compliance with 10 CFR 20.1301. A detailed description of the BPDS process radiation monitoring system and the UWS effluent monitoring system can be found in Section 11.5.2.2.15 and in Section 11.5.2.1.6, respectively.

The ABS is described in Section 10.4.10.

11.5.2.2.15 Balance-of-Plant Drain System

The BPDS includes the collection, segregation, and storage of nonradioactive waste streams originating from structures, systems and components located outside of the radiologically controlled area. The BPDS does not serve the Reactor Building or the Radioactive Waste Building.

The following BPDS process radiation monitoring instrumentation is provided for system inputs that have the potential to be radiologically contaminated.

- A single adjacent-to-line radiation monitor is located on the line downstream of the ABS blowdown coolers for the high and low-pressure boilers prior to entering the north waste water sump tank.
- A single in-line radiation monitor is located on the north condensate regeneration skid drain line to the north chemical waste collection sump tank.
- A single in-line radiation monitor is located on the south condensate regeneration skid drain line to the south chemical waste collection sump tank.
- A single in-line radiation monitor is located on the north turbine generator building floor drain line to the north waste water sump tank.
- A single in-line radiation monitor is located on the south turbine generator building floor drain line to the south waste water sump tank.

The BPDS process radiation monitors provide continuous indication to the main and waste management control rooms. If a high radiation condition is detected an alarm initiates in the main and waste management control rooms, the associated waste water sump pumps automatically shut down and transfer to manual control, and the discharge flow path to the BPDS collection tanks automatically isolate.

If required, the discharge flow path of the associated chemical waste water sump pumps can be opened to the liquid radioactive waste system high-conductivity waste tank, and the sump pumps can be restored to automatic operation to facilitate processing radiologically contaminated water and system flushing.

To provide an early indication of primary to secondary leakage, the high alarm setpoint is chosen for the radiation monitor that is set sufficiently low to detect leakage without causing spurious alarms in the control room. In the event of loss of power or air, the sump pump discharge valves fail in their current positions.

System sampling provisions located on the discharge of the BPDS sump tank pumps allow the process fluid to be recirculated to ensure a representative sample. Process sampling permits the determination of process radionuclide content and serves as a redundant means of detecting process radioactivity in the event that radiological monitoring is unavailable. The BPDS design limits the amount of radioactive material entering the system and ultimately to the environs, ensuring compliance with GDC 60 and 64, and 10 CFR 20.1406.

The BPDS is described in Section 9.3.3.

11.5.2.2.16 Demineralized Water System

The demineralized water system (DWS) supplies high quality demineralized water to contaminated, potentially contaminated, and clean systems. A single adjacent-to-line radiation monitor is provided for each of the north and south Reactor Building DWS headers.

The DWS radiation monitors provide continuous MCR indication and alarm capability. To provide an early indication of leakage from radiologically contaminated systems to the clean DWS, the high radiation alarm setpoint is set sufficiently low to detect leakage without causing spurious alarms in the main control room.

The DWS is described in Section 9.2.3.

11.5.2.2.17 Condensate Polishing System

The condensate polishing system (CPS) is normally a non-radiologically contaminated system. In the event of steam generator tube leakage or failure, radioactive contamination products may be released within the condensate system process stream and captured in the condensate polisher resin beds. To ensure prompt detection of contamination of the CPS, a single adjacent-to-line radiation monitor is provided on the process inlet to the resin regeneration package.

The CPS radiation monitor provides continuous MCR indication and alarm functions. To provide an early indication of radiologically contaminated resins and water from the condensate system to the CPS, the high radiation alarm setpoint is set sufficiently low to detect leakage without causing spurious alarms in the MCR.

Operators have the ability to manually isolate and sample the CPS. In addition, the system process flow and drains are directed to the BPDS, which has the capability to detect radiation in the drain system process stream from the condensate polishing and regeneration system and transfer the water to the LRWS. Details of these design features are provided in Section 11.5.2.2.15. The alarm setpoints, control room monitoring capability, system sampling capability, use of the BPDS, and operator response in accordance with site procedures ensures compliance with GDC 60 and RG 4.21 and ensures that the objectives of 10 CFR 20.1406 are met.

System provisions for process fluid grab sampling are located within the isolation boundary of each condensate system and the CPS to allow the operators to determine the source and content of the radioactive contamination and to provide a secondary method of detection, if the related radiation monitor is not available. The condensate system and the CPS sampling provisions are designed to conform with RG 8.8 and 8.10 and enhance staff capability to meet ALARA goals. The ability to isolate potentially contaminated systems, provisions for sampling potentially contaminated systems, and the use of the BPDS as drain collection in the Turbine Building ensures compliance for occupation exposure in accordance with 10 CFR 20.1201 and 10 CFR 20.1202, and limits contamination per 10 CFR 20.1406.

The design of the detectors permits removal without breaching the process system for repair, cleaning, calibration, and functional checks, as appropriate. The electrical power to the radiation monitoring system is supplied by the normal DC power system.

11.5.2.2.18 Radioactive Waste Drain System

The RWDS receives both radiologically contaminated and non-contaminated liquids and transfers the liquids to the LRWS for processing. A single adjacent-to-line radiation monitor is provided on the normally non-contaminated RCCWS drain tank to ensure prompt identification of reactor component cooling water radiological contamination.

The RWDS Reactor Building RCCWS drain tank radiation monitor provides continuous main control room indication and alarm capability. To provide an early indication of leakage from radiologically contaminated systems to the clean RCCWS, the high radiation alarm setpoint is set sufficiently low to detect leakage without causing spurious alarms in the main control room.

The provision for Reactor Building RCCWS drain tank grab sampling is located on the Reactor Building RCCW drain tank pump minimum flow line to allow for tank recirculation to obtain a representative sample. Grab sampling provides the capability to analyze the Reactor Building RCCWS drain tank contents prior to return to the reactor component cooling system, and provides a secondary method of initial detection, if the related radiation monitor is not available. The Reactor Building RCCWS drain tank sampling provisions are designed to conform with RGs 8.8 and 8.10 and enhance staff capability to meet ALARA goals. The provisions for sampling potentially contaminated systems and the use of the RWDS and LRWS for waste collection in the Reactor Building ensure compliance to occupational exposure limits in accordance with 10 CFR 20.1201 and 10 CFR 20.1202, and limit contamination per 10 CFR 20.1406.

The RWDS is described in Section 9.3.3.

11.5.2.2.19 Solid Radioactive Waste System

Within the SRWS, liquid process sampling is performed by an in-line automatic sampler located outside of the resin storage tank room to minimize radiation exposure to operating personnel. The suction tap for the sample line is located upstream of the dewatering skid and high-integrity container. The sampler is programmed on a preset interval to extract sample results representative of the entire contents of the high-integrity container as it is being filled to ensure correct classification for shipping. The sampling time is adjustable to permit the use of different size vessels by the SRWS. The sampling unit is mechanical and retrieves the sample directly from the process flow. The sampling unit does not rely upon sample lines, purging or flushing to obtain a representative sample, precluding the possibility of clean system cross-contamination and conforming with 10 CFR 20.1406. Prevention of personnel contamination and dose is provided by remote operation of the sampler, shielded piping, and area radiation monitors with alarm function. This conforms with RGs 8.8 and 8.10 and enhances the capability to meet ALARA goals ensuring compliance with the occupational dose limits of 10 CFR 20.1201 and 10 CFR 20.1202 when the sampling system is in service.

The SRWS is described in Section 11.4.

11.5.2.3 Reliability and Quality Assurance

The quality assurance controls for digital computer software used in radiation monitoring and sampling equipment is described in Section 7.2.1.

Section 17.6 describes the program for monitoring the effectiveness of maintenance required by 10 CFR 50.65.

The process and effluent monitoring equipment functionality is confirmed by periodic channel comparisons and instrument calibrations in accordance with site procedures. Calibration of monitors is conducted through the use of known radionuclide source national standards.

Programs and procedures for the control of measuring and test equipment are administered per the quality assurance program described in Section 17.5.

11.5.2.4 Effluent Instrumentation Alarm Setpoints

Effluent alarm setpoints are determined in accordance with the guidance of NUREG-1301 and NUREG-0133 such that effluent releases to unrestricted areas do not exceed those in 10 CFR 20 Appendix B, Table 2. The bases for establishing the alarm and trip setpoints for the initiating actions are documented in the ODCM, with consideration given to site-specific liquid effluent dilution factors and gaseous effluent atmospheric dispersion conditions. The ODCM is addressed in Section 11.5.2.6.

11.5.2.5 Effluent Release Controls

The gaseous and liquid effluent control for the plant is described in the ODCM and includes a description of how effluent release rates are derived and parameters used in setting instrumentation alarm setpoints to control or terminate effluent releases in unrestricted areas that are above the effluent concentrations in Table 2 of Appendix B to 10 CFR Part 20. In addition, the ODCM describes how the guidance of NUREG-1301 and NUREG-0133 are used in developing the alarm setpoints.

11.5.2.6 Offsite Dose Calculation Manual

The offsite dose calculation manual (ODCM) contains a description of the methodology and parameters used for calculation of offsite doses for gaseous and liquid effluents. The ODCM also contains the planned effluent discharge flow rates and addresses the numerical requirements of 10 CFR 50, Appendix I.

COL Item 11.5-2: A COL applicant that references the NuScale Power design certification will develop an offsite dose calculation manual (ODCM) that contains a description of the methodology and parameters used for calculation of offsite doses for gaseous and liquid effluents, using the guidance of Nuclear Energy Institute (NEI) 07-09A (Reference 11.5-8).

11.5.2.7 Radiological Environmental Monitoring Program

The Radiological Environmental Monitoring Program (REMP) follows the guidance in NUREG-1301 and NUREG-0133. It considers local land use census data for the identification of potential radiation pathways and takes into account radioactive materials present in liquid and gaseous effluents and direct external radiation from plant systems, structures, and components.

COL Item 11.5-3: A COL applicant that references the NuScale Power design certification will develop a Radiological Environmental Monitoring Program (REMP), consistent with the guidance in NUREG-1301 and NUREG-0133, that considers local land use census data for the identification of potential radiation pathways radioactive materials present in liquid and gaseous effluents, and direct external radiation from systems, structures, and components.

11.5.2.8 Process and Effluent Monitor Ranges

The process and effluent radiation monitor instrument ranges are based on 10 CFR 20, Appendix B, and Regulatory Guides 1.21, 1.97, and 1.45. The individual process and effluent monitor dynamic instrument ranges and the basis for selection are described in Table 11.5-4.

11.5.3 References

11.5-1 American National Standards Institute, "Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity for Effluents," ANSI N42.18 - 2004, New York, NY.

11.5-2 American National Standards Institute/Health Physics Society, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities," ANSI/HPS N13.1-2011, Washington, DC.

11.5-3 U.S. Nuclear Regulatory Commission, "Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to the Environment," Bulletin 80-10, May 6, 1980.

11.5-4 Nuclear Energy Institute, "Generic FSAR Template Guidance for Life Cycle Minimization of Contamination," NEI 08-08, Rev. 3, Washington, DC, September 2008.

11.5-5 Nuclear Energy Institute, "Steam Generator Program Guidelines," NEI 97-06, Rev. 3, Washington, DC, January 2011.

11.5-6 Electric Power Research Institute, "Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines," EPRI #1022832, Rev. 4, Palo Alto, CA, 2011.

11.5-7 Electric Power Research Institute, "Area and Process Radiation Monitoring System Guide," Rev. 2, EPRI #104862, Palo Alto, CA, 2003.

11.5-8 Nuclear Energy Institute, "Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description," NEI 07-09A, Revision 0, Washington, DC, March 2009.

11.5-9 Electric Power Research Institute, "Calibration of Radiation Monitors at Nuclear Power Plants," Rev. 1, EPRI #102644, Palo Alto, CA, 2005.

Table 11.5-1: Process and Effluent Radiation Monitoring Instrumentation Characteristics

System	Quantity	Type	Service	Isotopes	Nominal Range	Location/Function	PAM	Safety-related	Media	Instrument type
ABS	2	γ	ABS return flow from MHS	Cs-137	1.0E-7 to 1.0E-2 μCi/ml	ABS return flow from MHS heat exchangers 0A and 0B	No	No	Liquid	Adjacent-to-line
ABS	2	γ	ABS return flow from MHS	Ar-41	1.0E-7 to 1.0E-1 μCi/cc	ABS return flow from MHS heat exchangers 0A and 0B	No	No	Gas	Adjacent-to-line
ABS	1	γ	ABS high pressure condensate tank vent	Ar-41	1.0E-7 to 1.0E-1 μCi/cc	ABS 1100 psi condensate tank vent	No	No	Gas	Adjacent-to-line
ABS	1	γ	ABS high pressure boiler to low pressure boiler cross feed	Ar-41	1.0E-7 to 1.0E-1 μCi/cc	High pressure boiler to low pressure boiler cross feed	No	No	Gas	Adjacent-to-line
ABVS	1	γ	Radioactive waste building ventilation system (particulate)	Cs-137	3.0E-10 to 1.0E-6 μCi/cc	Hot machine shop RWBVS exhaust air	No	No	Gas	Off-line
BPDS	4	γ	BOP drains	Cs-137	1.0E-7 to 1.0E-2 μCi/ml	Monitors condensate regeneration skid effluent and TGB drains.	No	No	Liquid	In-Line
BPDS	1	γ	BOP drains	Cs-137	1.0E-7 to1.0E-2 μCi/ml	Inlet to 0A-BPD-TNK-0001 from ABS blowdown	No	No	Liquid	Adjacent-to-line
CARS	12	γ	Condenser air removal Sys	Ar-41	1.0E-7 to 1.0E-1 μCi/cc	Condenser air removal skid - Ar-41	Yes	No	Gas	Adjacent-to-line
CARS	12	β	Condenser air removal sys (particulate)	Cs-137	3E-10 to 1E-6 μCi / cc	Condenser air removal skid air particulate	Yes	No	Gas	Off-line (PING)
CARS	12	γ	Condenser air removal sys (iodine)	I-131	3E-10 to 5E-8 μCi / cc	Condenser air removal skid supply air iodine	Yes	No	Gas	Off-line (PING)
CARS	12	β	Condenser air removal sys (noble gas)	Kr-85 Xe-133	3E-7 to 1.0E+5 μCi/cc	Condenser air removal skid (NG)	Yes	No	Gas	Off-line (PING)
CVCS	12	γ	CVCS suction from RCS sample line	Cs-137	1E-7 to 1E-2 μCi / ml	RCS discharge sample line	No	No	Liquid	Adjacent- to-line
CES	12	γ	Containment evacuation atmosphere gas	Ar-41	1.0E-7 to 1.0E-1 μCi/cc	CES vacuum pump discharge line	No	No	Gas	Adjacent- to-line
CES	12	β	Containment evacuation atmosphere gas (particulate)	Cs-137	3.E-10 to 1E-6 μCi/cc	CES vacuum pump discharge line particulate	No	No	Gas	Off-line (PING)

Table 11.5-1: Process and Effluent Radiation Monitoring Instrumentation Characteristics (Continued)

System	Quantity	Type	Service	Isotopes	Nominal Range	Location/Function	PAM	Safety-related	Media	Instrument type
CES	12	γ	Containment evacuation atmosphere gas (iodine)	I-131	3.E-10 to 5E-8 µCi/cc	CES vacuum pump discharge line iodine	No	No	Gas	Off-line (PING)
CES	12	β	Containment evacuation atmosphere gas (noble gas)	Kr-85 Xe-133	3.E-7 to 1E-2 µCi/cc	CES vacuum pump discharge line (NG)	No	No	Gas	Off-line (PING)
CES	12	γ	CES sample vessel	Cs-137	3E-10 to 1E-6 µCi / ml	CES sample vessel discharge line.	No	No	Liquid	Adjacent -to-line
CFDS	2	β	Containment drain separator tank gaseous discharge	Kr-85 Xe-133	3E-7 to 1E-2 µCi / cc	Containment drain separator gaseous discharge lines	No	No	Gas	Off-line
[[CPS	2	γ	Condensate polisher resin regeneration	Cs-137	1E-7 to 1E-2 µCi/cc	0A, 0B condensate polishing system resin regeneration skid	No	No	Liquid	Adjacent-to-line]]
CRVS	3	β	Normal control room HVAC system	Kr-85 Xe-133	1E-5 to 1E+1 Rad/hr	Normal control room HVAC system, outside air duct upstream of filter unit	No	No	Gas	In-line
CRVS	2	β	Normal control room HVAC system (particulate)	Cs-137	3E-10 to 1E-4 µCi / cc	Normal control room HVAC system, outside air duct downstream of filter unit	No	No	Gas	Off-line
CRVS	2	γ	MCR supply air duct (iodine)	I-131	1E-9 to 1E+2 µCi / cc	Normal control room HVAC system, outside air duct downstream of filter unit	No	No	Gas	Off-line
CRVS	2	β	MCR supply air duct (noble gas)	Kr-85 Xe-133	4E-5 to 1E+4 µCi/cc	Normal control room HVAC system, outside air duct downstream of filter unit	No	No	Gas	Off-line
DWS	1	γ	North Reactor Building services header	Cs-137	1E-7 to 1E-2 µCi/ml	North Reactor Building, DWS service lines	No	No	Liquid	Adjacent-to-line
DWS	1	γ	South Reactor Building services header	Cs-137	1E-7 to 1E-2 µCi/ml	North Reactor Building, DWS service lines	No	No	Liquid	Adjacent-to-line
GRW	1	β	GRW common discharge line to HVAC exhaust	Kr-85 Xe-133	3E-7 to 1E-2 µCi/cc	Common discharge line from GRW to HVAC exhaust	No	No	Gas	Off-line

Table 11.5-1: Process and Effluent Radiation Monitoring Instrumentation Characteristics (Continued)

System	Quantity	Type	Service	Isotopes	Nominal Range	Location/Function	PAM	Safety-related	Media	Instrument type
GRW	2	β	GRW decay beds discharge lines	Kr-85 Xe-133	3E-7 to 1E-2 μCi/cc	Downstream of decay beds discharge line to HVAC exhaust	No	No	Gas	Off-line
LRW	2	γ	LRW discharge to environment	Cs-137	1E-7 to 1E-2 μCi/ml	LRW effluent discharge line to environment	No	No	Liquid	Adjacent-to-line
MSS	24	γ	MSS main steam line A	Ar-41	1.0E-7 to 1.0E-1 μCi/cc	Main steam line A	No	No	Gas	Adjacent to-line
MSS	24	γ	MSS main steam line B	Ar-41	1.0E-7 to 1.0E-1 μCi/cc	Main steam line B	No	No	Gas	Adjacent to-line
PSCS	1	β	Pool surge control system	Kr-85 Xe-133	3E-7 to 1E-2 μCi / cc	Pool surge control storage tank vent line	No	No	Gas	Off-line
RBVS	3	β	Spent fuel pool ventilation exhaust (particulate)	Cs-137	3E-10 to 1E-6 μCi / cc	RBVS spent fuel pool and refuel dock exhaust air	No	No	Gas	Off-line (PING)
RBVS	3	γ	Spent fuel pool ventilation exhaust (iodine)	I-131	3E-10 to 5E-8 μCi / cc	RBVS spent fuel pool and refuel dock exhaust air	No	No	Gas	Off-line (PING)
RBVS	3	β	Spent fuel pool ventilation exhaust (noble gas)	Kr-85 Xe-133	3E-7 to 1.0E-2 μCi/cc	RBVS spent fuel pool and refuel dock exhaust air	No	No	Gas	Off-line (PING)
RBVS	1	γ	Reactor Building ventilation system (particulate)	Cs-137	3E-10 to 1E-6 μCi/cc	RBVS general area exhaust air	No	No	Gas	Off-line
RBVS	1	γ	Reactor Building ventilation system (iodine)	I-131	3E-10 to 5E-8 μCi / cc	RBVS general area exhaust air	No	No	Gas	Off-line
RBVS	1	β	Plant vent particulate	Cs-137	1E-7 to 1E-2 μCi/cc	Plant exhaust stack	Yes	No	Gas	Off-line (PING)
RBVS	1	γ	Plant vent iodine	I-131	3E-10 to 1E-6 μCi/cc	Plant exhaust stack	Yes	No	Gas	Off-line (PING)
RBVS	1	β	Plant vent noble gas (normal range)	Kr-85 Xe-133	3E-7 to 1E-2 μCi/cc	Plant exhaust stack	Yes	No	Gas	Off-line (PING)
RBVS	1	β/γ	Plant vent extended range gas (accident mid-range)	Kr-85 Xe-133	3E-7 to 1E+4 μCi/cc	Plant exhaust stack	Yes	No	Gas	Off-line (PING)
RBVS	1	β/γ	Plant vent extended range gas (accident high range)	Kr-85 Xe-133	3E-7 to 1E+4 μCi/cc	Plant exhaust stack	Yes	No	Gas	Off-line (PING)

Table 11.5-1: Process and Effluent Radiation Monitoring Instrumentation Characteristics (Continued)

System	Quantity	Type	Service	Isotopes	Nominal Range	Location/Function	PAM	Safety-related	Media	Instrument type
RWBVS	1	γ	Radioactive Waste Building ventilation system (particulate)	Cs-137	3E-10 to 1E-6 μCi/cc	RWBVS exhaust air particulate	No	No	Gas	Off-line
RWBVS	1	γ	Radioactive Waste Building ventilation system (iodine)	I-131	3E-10 to 5E-8 μCi / cc	RWBVS exhaust air iodine	No	No	Gas	Off-line
RCCW	12	γ	RCCW return lines from CES condensers	Cs-137	1E-7 to 1E-2 μCi/ml	RCCW return lines from CES condensers	No	No	Liquid	Adjacent-to-line
RCCW	24	γ	RCCW return lines from CVC NRHX and PSS coolers	Cs-137	1E-7 to 1E-2 μCi/ml	RCCW return lines from CVC NRHX and PSS coolers	No	No	Liquid	Adjacent-to-line
RWDS	1	γ	RCCWS water drained from the CVCS non-regenerative and RCCW HX	Cs-137	1E-7 to 1E-2 μCi/ml	RWDS tank-0020	No	No	Liquid	Adjacent-to-line
SCWS	1	γ	Site cooling water	Cs-137	1E-7 to 1E-2 μCi/ml	From cooling tower to UWS discharge basin blowdown line	No	No	Liquid	Off-line with sampling capability
SCWS	1	γ	Site cooling water	Cs-137	1E-7 to 1E-2 μCi/ml	From cooling tower to UWS discharge basin overflow line	No	No	Liquid	Adjacent-to-line
SCWS	3	γ	SCW reactor pool cooling HXs return lines	Cs-137	1E-7 to 1E-2 μCi/ml	SCW reactor pool cooling HXs return lines prior to entering the main header	No	No	Liquid	Off-line
SCWS	2	γ	SCW spent fuel pool cooling HX return lines	Cs-137	1E-7 to 1E-2 μCi/ml	SCW spent fuel pool cooling HX return lines	No	No	Liquid	Off-line
SCWS	4	γ	SCW RCCW return lines	Cs-137	1E-7 to 1E-2 μCi/ml	SCW RCCW return lines	No	No	Liquid	Off-line
TGSS	2	γ	Turbine gland sealing system skid exhaust common vent, Turbine Building	Ar-41	1.0E-7 to 1.0E-1 μCi/cc	Turbine generator skid common exhaust vent point Ar-41	No	No	Gas	Adjacent-to-line
TGSS	2	γ	Turbine gland sealing system skid exhaust common vent, Turbine Building	I-131	3E-10 to 5E-8 μCi / cc	Turbine generator skid common exhaust vent point Iodine	No	No	Gas	Off-line (PING)

Table 11.5-1: Process and Effluent Radiation Monitoring Instrumentation Characteristics (Continued)

System	Quantity	Type	Service	Isotopes	Nominal Range	Location/Function	PAM	Safety-related	Media	Instrument type
TGSS	2	β	Turbine gland sealing system skid exhaust common vent, Turbine Building	Kr-85 Xe-133	3E-7 to 1.0E-2 µCi/cc	Turbine generator skid common exhaust vent point skid (NG)	No	No	Gas	Off-line (PING)
TGSS	2	γ	Turbine gland sealing system skid, exhaust common vent Turbine Building	Cs-137	3E-10 to 1E-6 µCi/cc	Turbine generator skid common exhaust vent point particulate	No	No	Gas	Off-line
UWS	1	γ	Utility Water system	Cs-137	1E-7 to 1E-2 µCi/ml	Utility water system effluent path	No	No	Liquid	Off-line with sampling capability

Table 11.5-2: Provisions for Sampling Gaseous Process and Effluent Streams

No.	Gaseous Process or Waste System	Sample Provisions	
		Process	Effluent
		Grab (a)	Grab (a)
1	Gaseous radioactive waste system	I	_
2	RBVS plant vent	I	NG,H3
3	ABVS (RBVS)	I	_
4	RBVS (spent fuel pool area exhaust fan)	I	_
5	Radioactive Waste Building HVAC system (RWBVS)	I	_
6	Containment evacuation system	I, S&A	_
7	Turbine gland sealing system	I, S&A	_
8	Condenser air removal system	I, S&A	_
9	Main steam system	S&A	
10	Containment flood and drain system	I	_
11	Pool surge control system	I	_
12	Solid radioactive waste system	I	_

(a) - Sample point is available to obtain grab samples for laboratory analyses.
NG - Noble gas radioactivity
I - Iodine radioactivity
H3 - Tritium
S&A - Sampling and analysis of radionuclides, including gross radioactivity, identification and concentration of principal or significant radionuclides, and concentration of alpha emitters

Table 11.5-3: Provisions for Sampling Liquid Process and Effluent Streams

No.	Liquid Process or Effluent System	Sample Provisions	
		Process	Effluent
		Grab (a)	Grab (a)
1	Liquid radioactive waste system	S&A	S&A,H3
2	Circulating water system	S&A	S&A,H3
3	Site cooling water system	S&A	S&A,H3
4	Demineralized water system	S&A	S&A,H3
5	Utility water system	S&A	S&A,H3
6	Reactor component cooling water system	S&A	S&A,H3
7	Spent fuel cooling system	S&A, H3	_
8	Reactor pool cooling system	S&A, H3	_
9	Pool cleanup system	S&A, H3	_
10	Balance-of-plant drain system	S&A	_
11	Containment evacuation system	S&A	_
12	Radioactive waste drain system	S&A	_
13	Solid radioactive waste system	S&A	_
14	Auxiliary boiler system	S&A	_
15	Chemical and volume control system	S&A	_
16	Condensate polishing system	S&A	_

(a) - Sample point is available to obtain grab samples for laboratory analyses
NG - Noble gas radioactivity
I - Iodine radioactivity
H3 - Tritium
S&A - Sampling and analysis of radionuclides, including gross radioactivity, identification and concentration of principal or significant radionuclides, and concentration of alpha emitters

Table 11.5-4: Effluent and Process Radiation Monitoring System Dynamic Ranges

Radiation Monitor	Dynamic Detection Range	Principal Radionuclides Measured	Basis for Dynamic Range
A. Effluent Radiation Monitors			
RBVS	3E-7 to 1E+4 µCi /cc 1E-9 to 1E+2 µCi /cc 1E-9 to 1E+2 µCi /cc	Noble gas (Kr-85, Xe-133): ß Particulate (Cs-137): γ Iodine (I-131): γ	• Regulatory Guide 1.97, Revision 3, Table 3 • Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
CARS	1.0E-7 to 1.0E-1 µCi /cc	Ar-41: γ	• EPRI 1022832 Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines, Revision 4 • NEI 97-06, Steam Generator Program Guidelines, Revision 3. • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
CARS	3E-7 to 1E+5 µCi / cc 1E-9 to 1E+2 µCi / cc 1E-9 to 1E+2 µCi / cc	Noble gas (Kr-85, Xe-133): ß Particulate (Cs-137): γ Iodine (I-131): γ	• Regulatory Guide 1.97, Revision 3, Table 3 • Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
TGSS	1.0E-7 to 1.0E-1 µCi /cc	Ar-41: γ	• Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004 • EPRI 1022832 Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines, Revision 4 • NEI 97-06, Steam Generator Program Guidelines, Revision 3
TGSS	3E-7 to 1E-2 µCi / cc 3E-10 to 1E-6 µCi / cc 3E-10 to 5E-8 µCi / cc	Noble gas (Kr-85, Xe-133): ß Particulate (Cs-137): γ Iodine (I-131): γ	• Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
PSCS	3E-7 to 1E-2 µCi / cc	Noble gas (Kr-85, Xe-133): ß	• Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
LRWS	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation

Table 11.5-4: Effluent and Process Radiation Monitoring System Dynamic Ranges (Continued)

Radiation Monitor	Dynamic Detection Range	Principal Radionuclides Measured	Basis for Dynamic Range
UWS	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Rx Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
SCWS* * Effluent continuously monitored via the UWS	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
Circulating Water (CWS)* * Effluent continuously monitored via the UWS	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
B. Process Radiation Monitors			
CRVS Upstream of charcoal filters	1E-5 to 1E+1 Rad/hr	Noble gas (Kr-85, Xe-133)	• Regulatory Guide 1.97, Revision 3 guidance. • Control Room Atmospheric Dispersion Factors
CRVS Downstream of charcoal filters	4E-5 to 1E+4 µCi / cc 3E-10 to 1E-4 µCi / cc 1E-9 to 1E+2 µCi / cc	Noble gas (Kr-85, Xe-133) Particulate (Cs-137) Iodine (I-131)	• Regulatory Guide 1.97, Revision 3 guidance • Control Room Atmospheric Dispersion Factors • Radiological Consequences of Design Basis Source Term
RBVS	3E-7 to 1E-2 µCi / cc 3E-10 to 1E-6 µCi / cc3E-10 to 5E-8 µCi / cc	Noble gas (Kr-85, Xe-133): ß Particulate (Cs-137): γ Iodine (I-131): γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
RBVS	3E-7 to 1E-2 µCi / cc 3E-10 to 1E-6 µCi / cc 3E-10 to 5E-8 µCi / cc	Noble gas (Kr-85, Xe-133): ß Particulate (Cs-137): γ Iodine (I-131): γ	• Regulatory Guide 1.97, Revision 3 range values used as guidance • Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
RWBVS	3E-10 to 1E-6 µCi / cc 3E-10 to 5E-8 µCi / cc	Particulate (Cs-137): γ Iodine (I-131): γ	• Realistic Primary Coolant and Secondary Coolant Activity • RXB Dose Rates and Shielding Calculations • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004

Table 11.5-4: Effluent and Process Radiation Monitoring System Dynamic Ranges (Continued)

Radiation Monitor	Dynamic Detection Range	Principal Radionuclides Measured	Basis for Dynamic Range
ABVS	3E-10 to 1E-6 µCi / cc	Particulate (Cs-137): γ	• Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
GRWS Common Discharge to HVAC	3E-7 to 1E-2 µCi / cc	Noble gas (Kr-85, Xe-133)	• Normal Effluent Release Source Term Calculation
GRWS Decay Bed Discharge Lines	3E-7 to 1E-2 µCi / cc	Noble gas (Kr-85, Xe-133)	• Normal Effluent Release Source Term Calculation
CES Vacuum Pump Discharge	3E-7 to 1E-2 µCi / cc 3E-10 to 1E-6 µCi / cc 3E-10 to 1E-8 µCi / cc	Noble gas (Kr-85, Xe-133) Particulate (Cs-137) Iodine (I-131)	• Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004.
CES Vacuum Pump Discharge	1.0E-7 to 1.0E-1 µCi / cc	Ar-41: γ	• EPRI 1022832 Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines, Revision 4 • NEI 97-06, Steam Generator Program Guidelines, Revision 3
CES Liquid Radiation Monitor	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
MSS	1.0E-7 to 1.0E-1 µCi / cc	Ar-41: γ	• EPRI 1022832 Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines, Revision 4 • NEI 97-06, Steam Generator Program Guidelines, Revision 3
CFDS	3E-7 to 1E-2 µCi / cc	Noble gas (Kr-85, Xe-133): ß	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
CVCS* *Also area radiation monitors described in Section 12.3.4	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004

Table 11.5-4: Effluent and Process Radiation Monitoring System Dynamic Ranges (Continued)

Radiation Monitor	Dynamic Detection Range	Principal Radionuclides Measured	Basis for Dynamic Range
RCCWS	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
SCWS Heat Exchanger Outlets	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
SCWS Cooling Tower Basin Blowdown	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
SCWS Cooling Tower Basin Overflow	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
ABS Flash Tank Vent	1.0E-7 to 1.0E-1 µCi /cc	Ar-41 γ	• EPRI 1022832 Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines, Revision 4. • NEI 97-06, Steam Generator Program Guidelines, Revision 3
ABS HP to LP cross tie	1.0E-7 to 1.0E-1 µCi /cc	Ar-41 γ	• EPRI 1022832 Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines, Revision 4. • NEI 97-06, Steam Generator Program Guidelines, Revision 3
ABS MHS Return	1E-7 to 1E-2 µCi/cc	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Release of Radioactive Materials in Gaseous and Liquid Effluent • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
ABS MHS Return	1.0E-7 to 1.0E-1 µCi /cc	Ar-41	• EPRI 1022832 Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines, Revision 4. • NEI 97-06, Steam Generator Program Guidelines, Revision 3

Table 11.5-4: Effluent and Process Radiation Monitoring System Dynamic Ranges (Continued)

Radiation Monitor	Dynamic Detection Range	Principal Radionuclides Measured	Basis for Dynamic Range
BPDS	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
DWS	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
CPS	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004
RWDS	1E-7 to 1E-2 µCi / ml	Cs-137: γ	• Realistic Primary Coolant and Secondary Coolant Activity • Best Estimate Fuel Isotopic Inventory Calculation • Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents, ANSI N42.18-2004

Figure 11.5-1: Radioactive Effluent Flow Paths with Process and Effluent Radiation Monitors

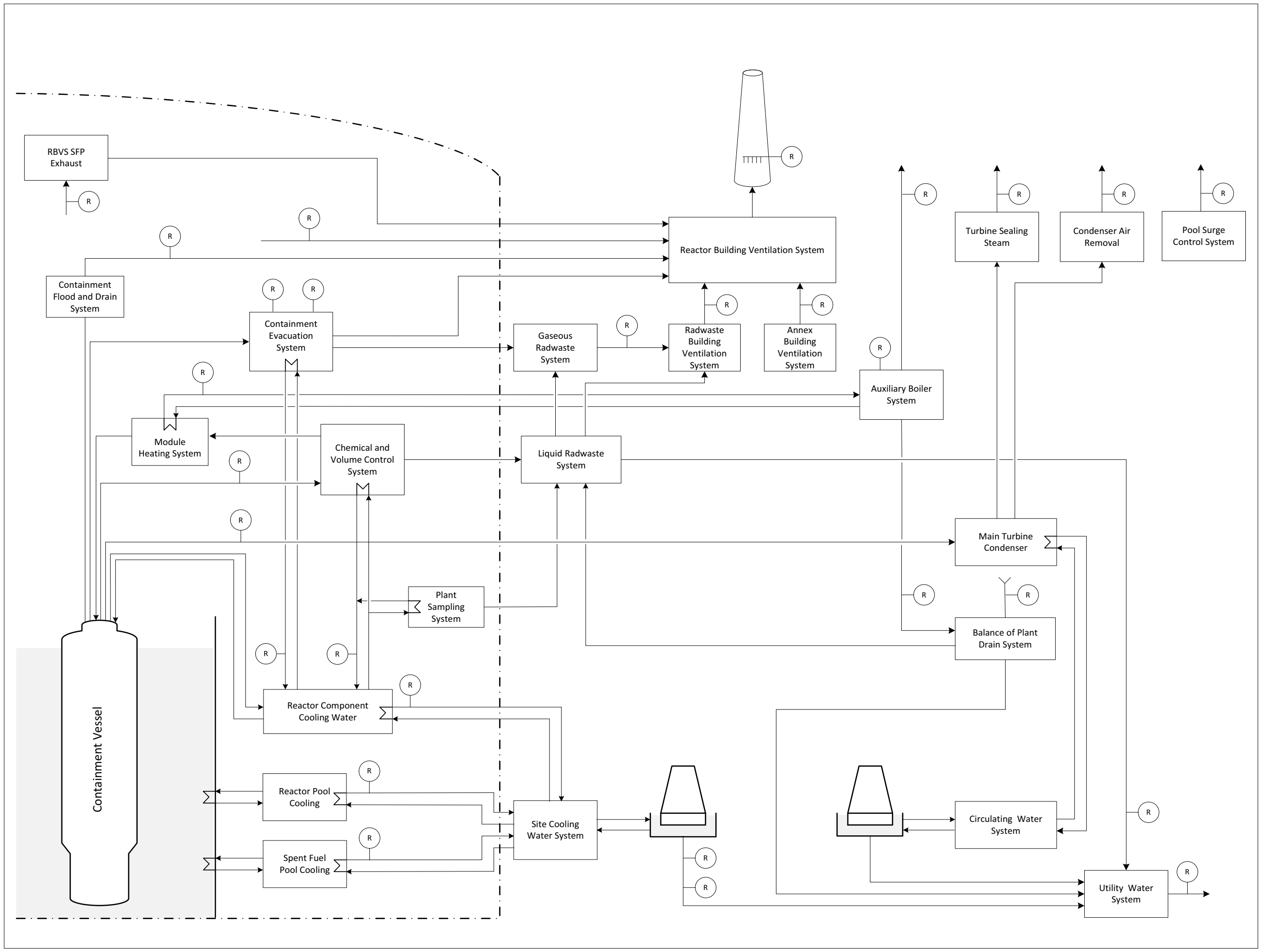


Figure 11.5-2: Process and Effluent Radiation Monitoring System I&C Configuration

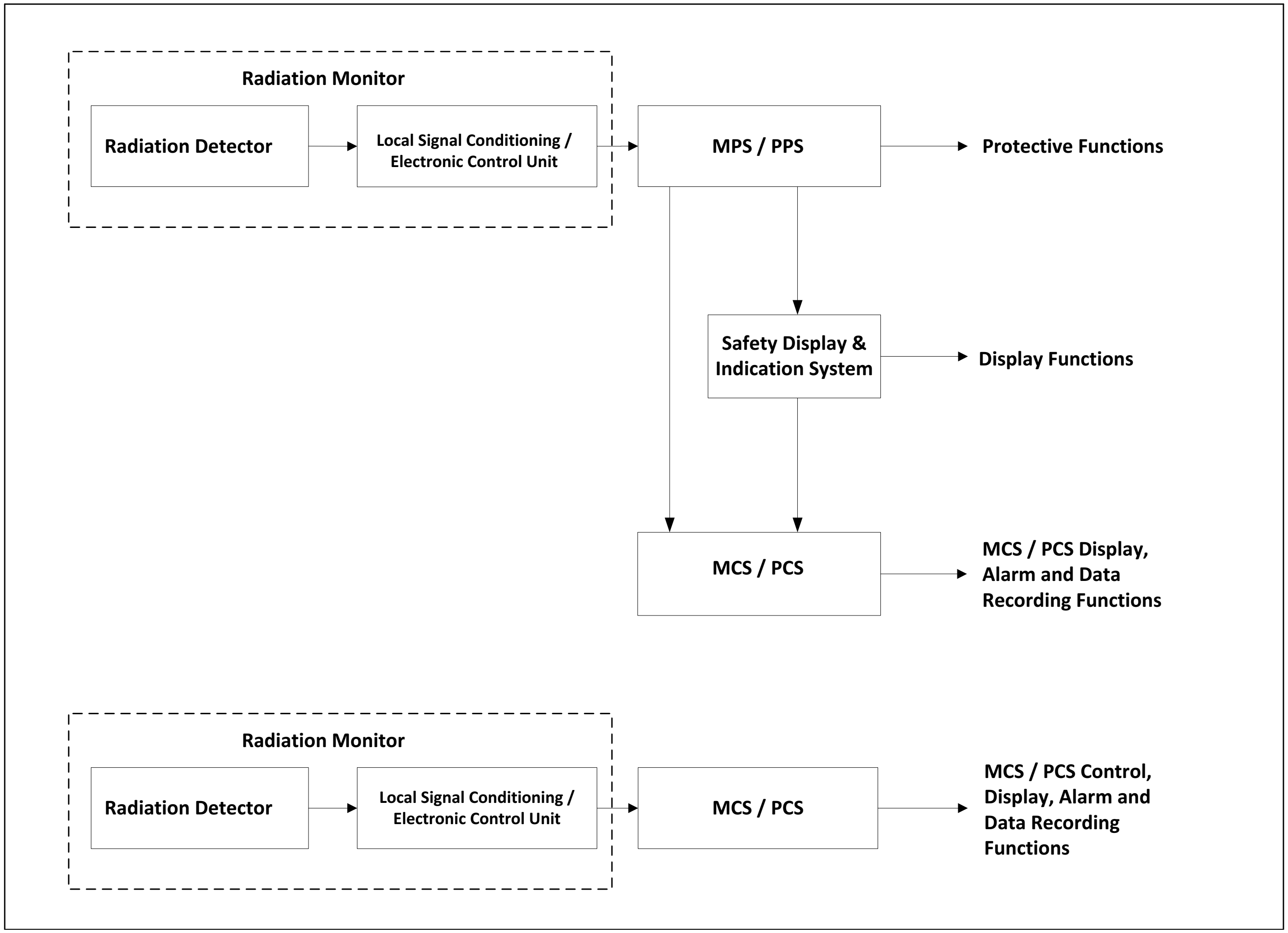


Figure 11.5-3: Off-Line Radiation Monitor

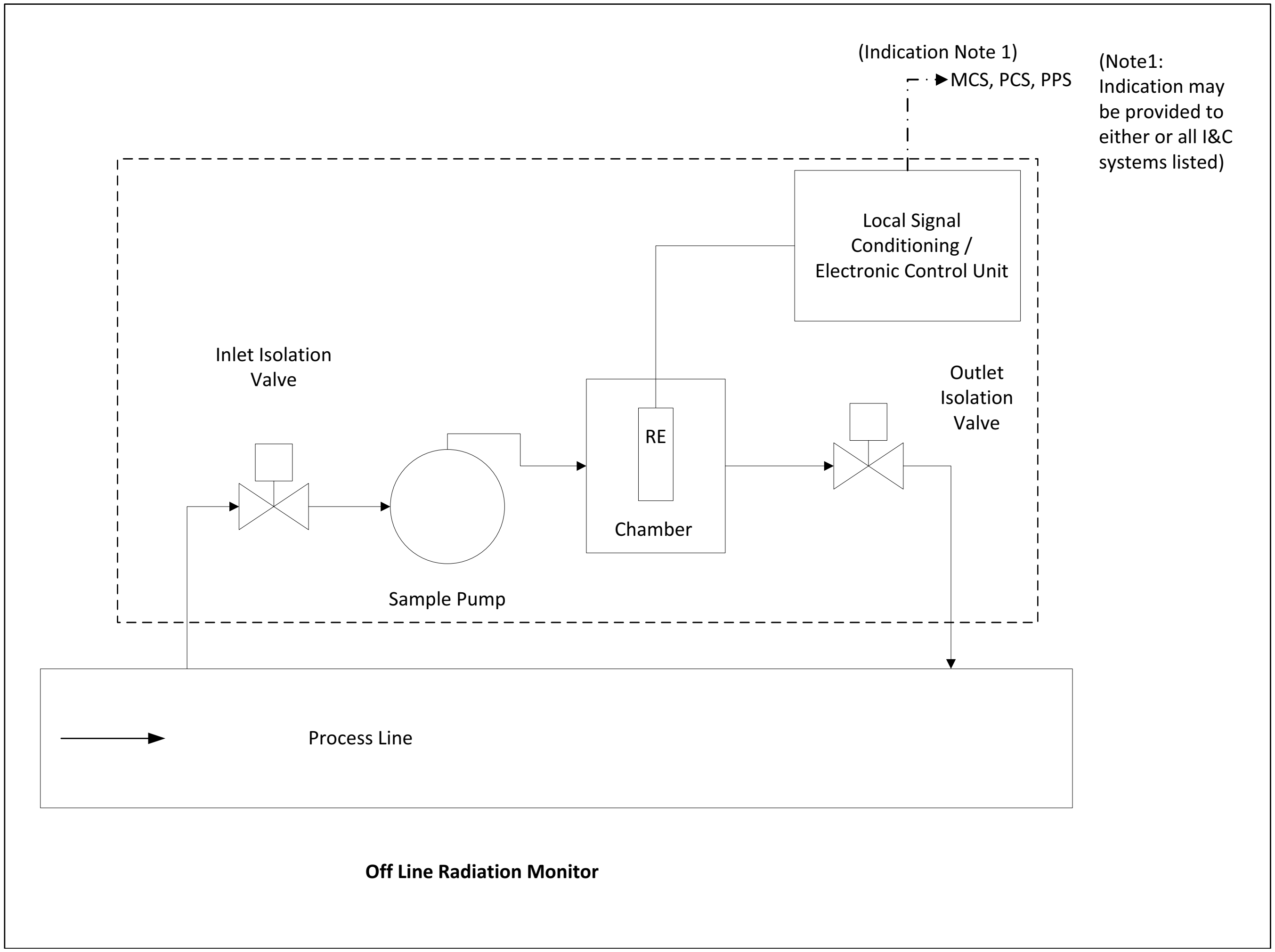


Figure 11.5-4: Adjacent-to-Line Radiation Monitor

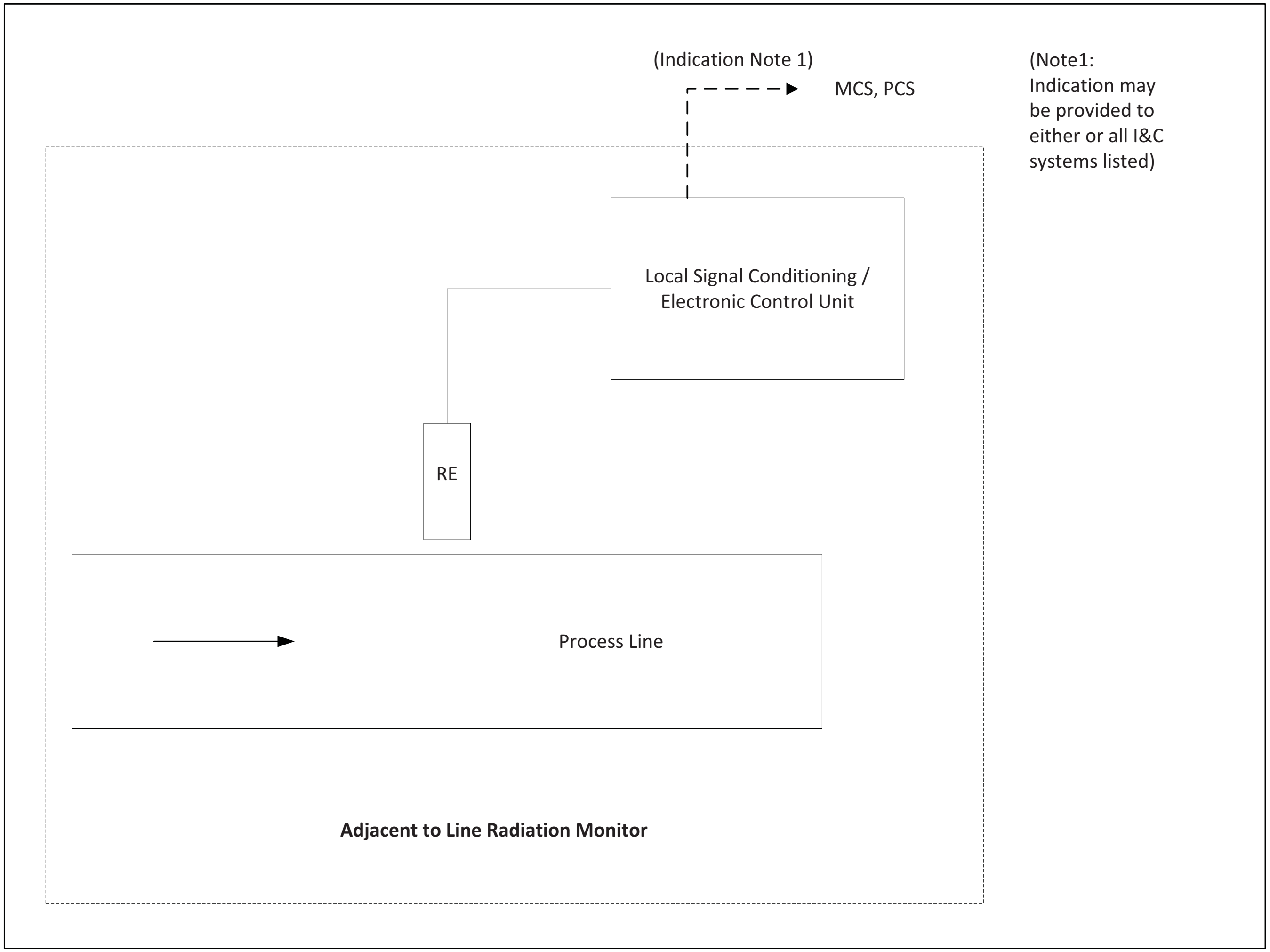


Figure 11.5-5: In-Line Radiation Monitor

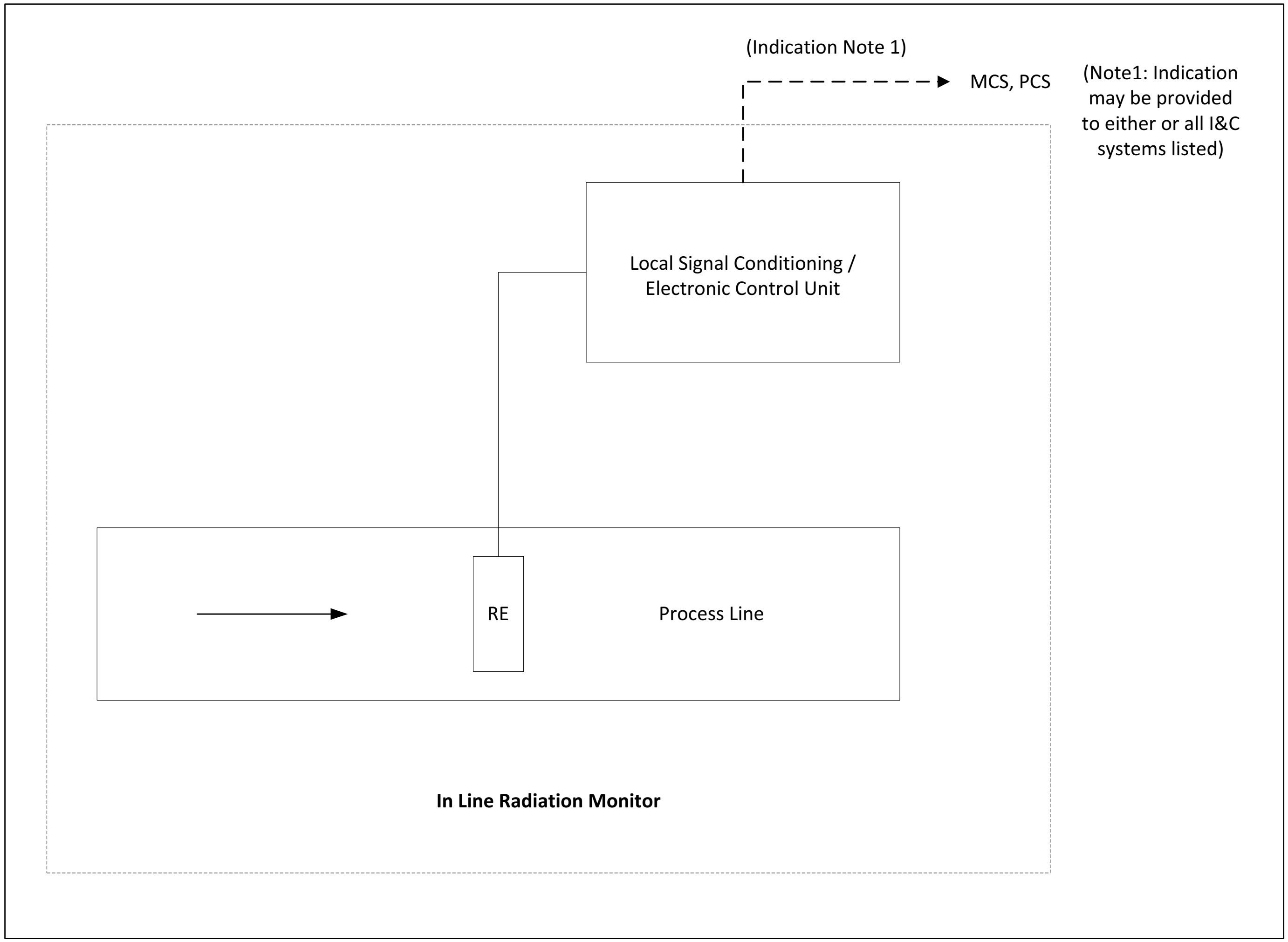
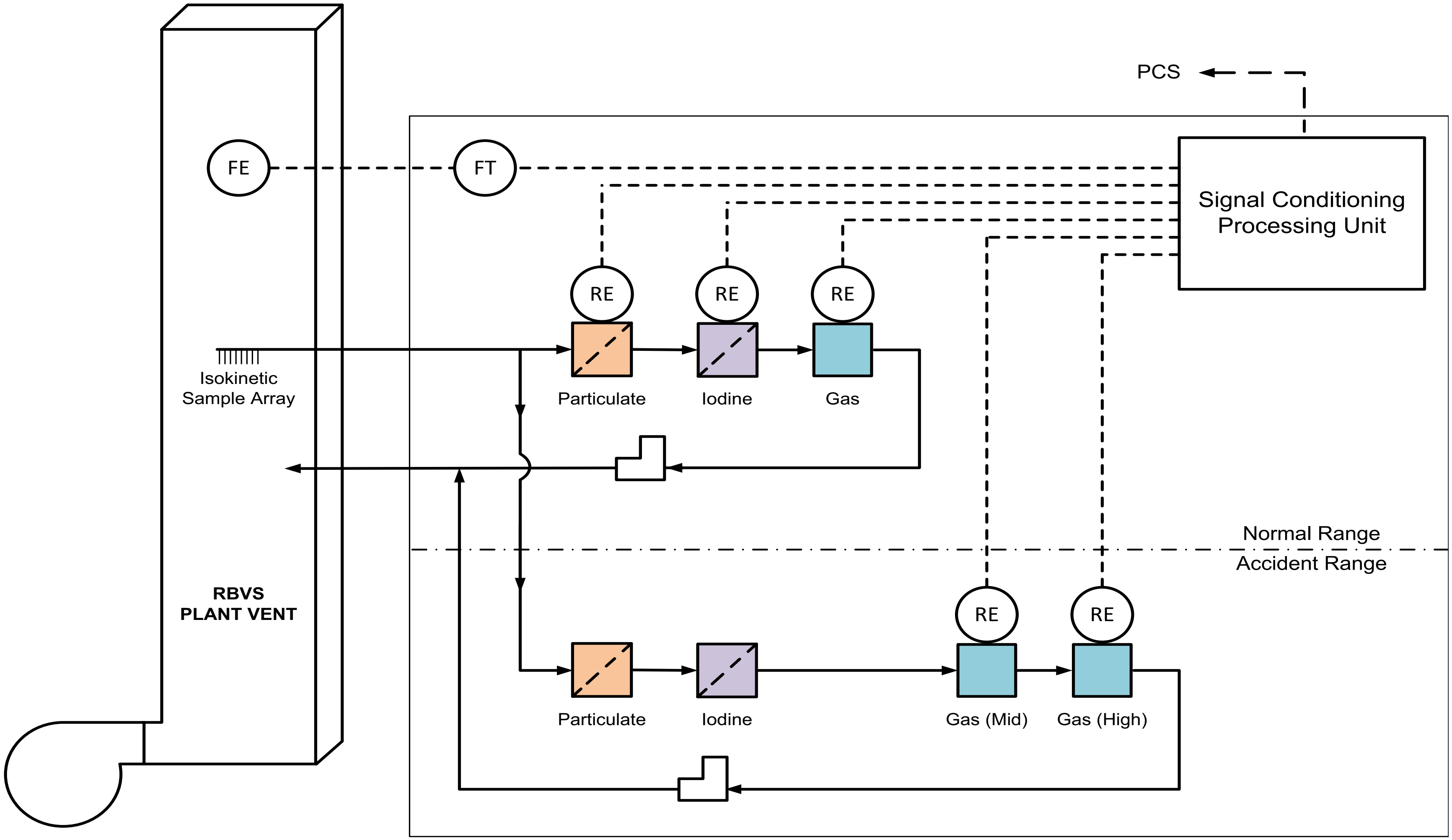


Figure 11.5-6: RBVS Plant Exhaust Stack Effluent Radiation Monitor



11.6 Instrumentation and Control Design Features for Process and Effluent Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring

Effluent and process radiation monitors are discussed in Section 11.5. This discussion contains the radiation monitoring (RM) design functions, features, and bases for the plant systems containing effluent or process radiation monitors and includes a discussion of the compliance with associated regulatory requirements and guidance documents.

Provisions for sampling in the systems containing effluent and process radiation monitors are included in the discussion in Section 11.5. For selected systems these provisions include functions provided by the process sampling system which is discussed in Section 9.3.2.

Area radiation and airborne contamination monitors are discussed in Section 12.3.4. This discussion contains the RM design functions, features and bases for plant area radiation and airborne contamination monitors and includes a discussion of the compliance with associated regulatory requirements and guidance documents.

Effluent and area RM provide input to the emergency response data system (ERDS). The electronic data communication interface is described in Section 7.2.13. Effluent and area radiological monitoring parameters and equipment are addressed in Sections 11.5 and 12.3.4.

Programs and procedures for the control of measuring and test equipment are administered under the applicable portions of the quality assurance program described in Section 17.5. Section 17.6 describes the program for monitoring the effectiveness of maintenance.