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December 30, 2022 Docket No. 52-050

U.S. Nuclear Regulatory Commission
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**SUBJECT:** NuScale Power, LLC Submittal of the NuScale Standard Design Approval Application Part 2 – Final Safety Analysis Report, Chapter 11, "Radioactive Waste Management," Revision 0

**REFERENCES:**

1. NuScale letter to NRC, "NuScale Power, LLC Submittal of Planned Standard Design Approval Application Content," dated February 24, 2020 (ML20055E565)
2. NuScale letter to NRC, "NuScale Power, LLC Requests the NRC staff to conduct a pre-application readiness assessment of the draft, 'NuScale Standard Design Approval Application (SDAA),'" dated May 25, 2022 (ML22145A460)
3. NRC letter to NuScale, "Preapplication Readiness Assessment Report of the NuScale Power, LLC Standard Design Approval Draft Application," Office of Nuclear Reactor Regulation dated November 15, 2022 (ML22305A518)
4. NuScale letter to NRC, "NuScale Power, LLC Staged Submittal of Planned Standard Design Approval Application," dated November 21, 2022 (ML22325A349)

NuScale Power, LLC (NuScale) is pleased to submit Chapter 11 of the Standard Design Approval Application, "Radioactive Waste Management," Revision 0. This chapter supports Part 2, "Final Safety Analysis Report," (FSAR) of the NuScale Standard Design Approval Application (SDAA) (Reference 1). NuScale submits the chapter in accordance with requirements of 10 CFR 52 Subpart E, Standard Design Approvals. As described in Reference 4, the enclosure is part of a staged SDAA submittal. NuScale requests NRC review, approval, and granting of standard design approval for the US460 standard plant design.

From July 25, 2022 to October 26, 2022, the NRC performed a pre-application readiness assessment of available portions of the draft NuScale FSAR to determine the FSAR's readiness for submittal and for subsequent review by NRC staff (References 2 and 3). The NRC staff reviewed draft Chapter 11. NuScale is enclosing information in this submittal that: 1) closes gaps identified between the draft SDAA Chapter 11 and technical content generally expected by the NRC; and 2) resolves identified technical issues that may have adversely impacted acceptance, docketing, or technical review of the application. Section B of the enclosures provide NuScale's responses to Reference 3 for Chapter 11 observations.

Enclosure 1 contains SDAA Part 2 Chapter 11, "Radioactive Waste Management," Revision 0, proprietary version. NuScale requests that the proprietary version (enclosure 1), be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 contains the nonproprietary version.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Mark Shaver at 541-360-0630 or at mshaver@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 30, 2022.

Sincerely,

Carrie Fosaaen
Senior Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Brian Smith, NRC
Michael Dudek, NRC
Getachew Tesfaye, NRC
Bruce Bavol, NRC
David Drucker, NRC

Enclosure 1: SDAA Part 2 Chapter 11, “Radioactive Waste Management,” Revision 0, (proprietary)
Enclosure 2: SDAA Part 2 Chapter 11, “Radioactive Waste Management,” Revision 0, (nonproprietary)
Enclosure 3: Affidavit of Carrie Fosaaen, AF-132449

**Enclosure 1:**

SDAA Part 2 Chapter 11, “Radioactive Waste Management,” Revision 0, (proprietary)

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**Enclosure 2:**

SDAA Part 2 Chapter 11, “Radioactive Waste Management,” Revision 0, (nonproprietary)

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# CHAPTER 11 RADIOACTIVE WASTE MANAGEMENT

## 11.1 Source Terms

Sources of radioactivity in the primary coolant are created by fission and activation processes in the reactor. The NuScale Power Plant US460 standard design uses a shared systems source term for systems that receive input from all six modules, such as radwaste systems. The shared systems source term is calculated as one module running at a design basis failed fuel fraction and five modules running at a realistic failed fuel fraction. The secondary coolant may become contaminated by primary-to-secondary leakage through the steam generator. 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 provides a basis for design capacities of waste management components, performance of the waste management systems and design of radiological monitoring equipment. The design basis source term is also used for the evaluation of shielding (General Design Criterion 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.

A 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. The methodology used to develop the primary and secondary coolant realistic source terms is described in TR-123242, (Reference 11.1-1).

The plant is designed with up to six NuScale Power Modules (NPMs) partially immersed in a pool of water, called the reactor pool. Because of this design, there is a potential for neutron activation of the reactor pool water. Additionally, 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 discussed in Section 12.2.1.

### 11.1.1 Design Basis Reactor Coolant Activity

The design basis source term uses the same methodology described in TR-123242, except it assumes that fuel defects are an order-of-magnitude greater than the realistic coolant source term (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 irradiated to the value presented in Table A-1 in TR-123242 (Reference 11.1-1). The quantity of each nuclide is calculated by ORIGEN-S. 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 quantity of fission products in the fuel pins and the release to the primary coolant are calculated using the methodology in TR-123242 (Reference 11.1-1).

#### 11.1.1.2 Activation Products

The permeation of tritium from the fuel to the primary coolant is modeled using the Electric Power Research Institute (EPRI) Tritium Management Model (Reference 11.1-3). The tritium permeation rate is linearly scaled from 4100 MWt to 250 MWt, and adjusted for a 95 percent capacity factor, as shown in Table 11.1-2.

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.

Because of its short half-life, N-16 is not of concern for offsite dose considerations. Table 12.2-4 lists the N-16 concentration at various locations in the primary coolant loop.

The predominant tritium production reactions are high-energy neutron interactions with lithium and boron isotopes.

The concentrations of tritium in the coolant streams vary depending on whether the primary coolant letdown to liquid radioactive waste system is recycled to the reactor pool, recycled back to chemical and volume control system (CVCS) makeup, or discharged through liquid radioactive waste system. The various tritium concentrations are presented in Table 11.1-8.

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.

The C-14 primary coolant equilibrium activity is calculated using the following equation:

$$A_{C14} = P_{C14} \times f \qquad \text{Eq. 11.1-1}$$

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

The concentration values of radioactive corrosion and wear products in the reactor coolant are developed using guidance from ANSI/ANS 18.1-1999 (Reference 11.1-2). The specific parameters for adjusting the ANSI/ANS 18.1-1999 reference values are listed in Table 11.1-3.

### 11.1.2 Design Basis Secondary Coolant Activity

The design basis secondary coolant activity is determined from an assumed primary-to-secondary leak rate (Table 11.1-2) 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

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

For radioisotopes other than tritium, it is assumed there are no steam leaks. The modeling process for secondary coolant concentrations is described in Section 4.2 of Reference 11.1-1. 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 is used to evaluate normal expected effluent releases as described in Section 11.2 and Section 11.3.

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-123242-P, Revision 0.

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 Electric Power Research Institute, Inc., "EPRI Tritium Management Model," EPRI #1009903, EPRI, 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 | 7.3E+05 | Y91 | 7.0E+06 |
| Kr85m | 1.5E+06 | Y92 | 7.6E+06 |
| Kr85 | 1.5E+05 | Y93 | 8.9E+06 |
| Kr87 | 2.8E+06 | Zr97 | 1.1E+07 |
| Kr88 | 3.7E+06 | Nb95 | 1.1E+07 |
| Kr89 | 4.6E+06 | Mo99 | 1.3E+07 |
| Xe131m | 1.0E+05 | Mo101 | 1.2E+07 |
| Xe133m | 4.5E+05 | Tc99m | 1.1E+07 |
| Xe133 | 1.4E+07 | Tc99 | 2.2E+02 |
| Xe135m | 3.3E+06 | Ru103 | 1.4E+07 |
| Xe135 | 4.1E+06 | Ru105 | 1.1E+07 |
| Xe137 | 1.2E+07 | Ru106 | 8.6E+06 |
| Xe138 | 1.1E+07 | Rh103m | 1.4E+07 |
| Halogens | | Rh105 | 1.0E+07 |
| Br82 | 4.0E+04 | Rh106 | 9.6E+06 |
| Br83 | 7.1E+05 | Ag110 | 3.6E+06 |
| Br84 | 1.2E+06 | Sb124 | 2.1E+04 |
| Br85 | 1.5E+06 | Sb125 | 1.5E+05 |
| I129 | 5.5E-01 | Sb127 | 8.3E+05 |
| I130 | 4.2E+05 | Sb129 | 2.4E+06 |
| I131 | 7.2E+06 | Te125m | 3.6E+04 |
| I132 | 1.0E+07 | Te127m | 1.3E+05 |
| I133 | 1.4E+07 | Te127 | 8.2E+05 |
| I134 | 1.5E+07 | Te129m | 3.9E+05 |
| I135 | 1.3E+07 | Te129 | 2.3E+06 |
| Rubidium, Cesium | | Te131m | 1.5E+06 |
| Rb86m | 3.1E+03 | Te131 | 6.1E+06 |
| Rb86 | 2.5E+04 | Te132 | 1.0E+07 |
| Rb88 | 3.8E+06 | Te133m | 6.4E+06 |
| Rb89 | 5.0E+06 | Te134 | 1.2E+07 |
| Cs132 | 5.2E+02 | Ba137m | 1.7E+06 |
| Cs134 | 3.4E+06 | Ba139 | 1.2E+07 |
| Cs135m | 4.4E+04 | Ba140 | 1.2E+07 |
| Cs136 | 7.8E+05 | La140 | 1.2E+07 |
| Cs137 | 1.8E+06 | La141 | 1.1E+07 |
| Cs138 | 1.3E+07 | La142 | 1.0E+07 |
| Other Fission Products | | Ce141 | 1.1E+07 |
| P32 | 1.1E+03 | Ce143 | 9.9E+06 |
| Co57 | 8.1E+00 | Ce144 | 9.2E+06 |
| Sr89 | 5.2E+06 | Pr143 | 9.6E+06 |
| Sr90 | 1.2E+06 | Pr144 | 9.3E+06 |
| Sr91 | 6.8E+06 | Np239 | 1.9E+08 |
| Sr92 | 7.5E+06 | C14 | 2.2E+01 |
| Y90 | 1.2E+06 | H3 | 2.1E+04 |
| Y91m | 4.0E+06 | | |

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

| Parameter | Value |
|---|---|
| Reactor core thermal power | 250 + 5 = 255 MWt (102%) |
| Number of fuel assemblies in one core | 37 |
| Range of U-235 fuel enrichment | 1.5% - 4.95% |
| Uranium mass in one fuel assembly | 250.6 kg |
| Maximum fuel assembly burnup | 62,GWD/MTU |
| Failed fuel fractions:<br>NPM Realistic source term<br>NPM Design basis source term | <br>0.0066%<br>0.066% |
| Escape rate coefficients:<br>Xe, Kr gases<br>I, Br, Cs, Rb<br>Mo, Tc, Ag<br>Te<br>Sr, Ba<br>Others | <br>6.5E-08 $s^{-1}$<br>1.3E-08 $s^{-1}$<br>2.0E-09 $s^{-1}$<br>1.0E-09 $s^{-1}$<br>1.0E-11 $s^{-1}$<br>1.6E-12 $s^{-1}$ |
| Average density of reactor coolant | 0.71 gram/$cm^3$ |
| RCS mass | 1.0E+05 lb |
| Argon injection concentration:<br>Design basis<br>Realistic | <br>0.15 μCi/$cm^3$<br>0.10 μCi/$cm^3$ |
| CVCS flow rate (purification) | 180 lb/min |
| Secondary coolant mass | 5.0E+04 lb |
| Secondary steam leak rate:<br>Design Basis<br>Realistic | <br>1700 lb/hr/NPM<br>125 lb/hr/NPM |
| Secondary coolant flow rate | 6.5E+05 lb/hr |
| Decontamination factors for CVCS mixed bed demineralizers:<br>Halogens<br>Cs, Rb<br>Other | <br>100<br>2<br>50 |
| Decontamination factors for condensate demineralizers:<br>Halogens<br>Cs, Rb<br>Other | <br>100<br>10<br>100 |
| Primary-to-secondary leak rate:<br>Design basis<br>Realistic | <br>75 lb/day/NPM<br>5.5 lb/day/NPM |
| Tritium permeation rate | 9 Ci/yr |
| Carbon-14 primary coolant retention rate | 0.01 |
| Carbon-14 removal rate by CVCS demineralizers | 0 |

## Table 11.1-3: Specific Parameters for CRUD

| Parameter | Symbol | Units | Value |
|---|---|---|---|
| Thermal power | P | MWt | 250 |
| Weight of water in RCS | WP | kg | 4.7E+04 |
| Letdown flow rate for purification | FD | kg/s | 1.4 |
| Letdown flow rate for boron control | FB | kg/s | 3.9E-03 |
| Flow through cation demineralizer | FA | kg/s | 0 |
| Fraction of material removed by cation demineralizer | NA | --- | 0.9 |
| Fraction of material removed by purification demineralizer | NB | --- | 0.98 |

**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 | 7.7E-09 | Mo101 | 4.3E-10 |
| Kr85m | 3.2E-08 | Tc99m | 1.1E-08 |
| Kr85 | 5.9E-06 | Tc99 | 2.1E-13 |
| Kr87 | 1.8E-08 | Ru103 | 1.1E-11 |
| Kr88 | 5.1E-08 | Ru105 | 3.6E-12 |
| Kr89 | 1.2E-09 | Ru106 | 6.8E-12 |
| Xe131m | 1.4E-07 | Rh103m | 1.1E-11 |
| Xe133m | 1.2E-07 | Rh105 | 7.4E-12 |
| Xe133 | 8.6E-06 | Rh106 | 6.8E-12 |
| Xe135m | 1.1E-08 | Ag110 | 8.1E-12 |
| Xe135 | 2.3E-07 | Sb124 | 1.6E-14 |
| Xe137 | 3.9E-09 | Sb125 | 1.2E-13 |
| Xe138 | 1.3E-08 | Sb127 | 6.1E-13 |
| Halogens | | Sb129 | 7.6E-13 |
| Br82 | 2.1E-10 | Te125m | 1.8E-11 |
| Br83 | 1.2E-09 | Te127m | 6.7E-11 |
| Br84 | 5.7E-10 | Te127 | 2.6E-10 |
| Br85 | 6.9E-11 | Te129m | 1.9E-10 |
| I129 | 3.5E-15 | Te129 | 2.7E-10 |
| I130 | 1.7E-09 | Te131m | 6.2E-10 |
| I131 | 4.4E-08 | Te131 | 3.1E-10 |
| I132 | 2.1E-08 | Te132 | 4.6E-09 |
| I133 | 6.7E-08 | Te133m | 3.9E-10 |
| I134 | 1.2E-08 | Te134 | 5.6E-10 |
| I135 | 4.3E-08 | Ba137m | 2.1E-08 |
| Rubidium, Cesium | | Ba139 | 1.0E-11 |
| Rb86m | 5.2E-14 | Ba140 | 5.6E-11 |
| Rb86 | 3.0E-10 | La140 | 1.6E-11 |
| Rb88 | 5.1E-08 | La141 | 3.2E-12 |
| Rb89 | 2.4E-09 | La142 | 1.5E-12 |
| Cs132 | 6.0E-12 | Ce141 | 8.6E-12 |
| Cs134 | 4.3E-08 | Ce143 | 6.5E-12 |
| Cs135m | 3.6E-11 | Ce144 | 7.3E-12 |
| Cs136 | 9.4E-09 | Pr143 | 7.6E-12 |
| Cs137 | 2.2E-08 | Pr144 | 7.2E-12 |
| Cs138 | 1.9E-08 | Np239 | 1.4E-10 |
| Other FPs | | Corrosion/Activation Products - CRUD | |
| P32 | 8.4E-16 | Na24 | 1.4E-08 |
| Co57 | 6.4E-18 | Cr51 | 7.7E-10 |
| Sr89 | 3.8E-11 | Mn54 | 4.0E-10 |
| Sr90 | 6.0E-12 | Fe55 | 3.0E-10 |
| Sr91 | 2.0E-11 | Fe59 | 7.5E-11 |
| Sr92 | 1.1E-11 | Co58 | 1.1E-09 |
| Y90 | 1.5E-12 | Co60 | 1.3E-10 |
| Y91m | 1.1E-11 | Ni63 | 6.6E-11 |
| Y91 | 5.6E-12 | Zn65 | 1.3E-10 |
| Y92 | 9.1E-12 | Zr95 | 9.7E-11 |
| Y93 | 4.3E-12 | Ag110m | 3.2E-10 |

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

| Nuclide | Primary Coolant Concentrations (Ci/g) | Nuclide | Primary Coolant Concentrations (Ci/g) |
|---|---|---|---|
| Zr97 | 6.3E-12 | W187 | 7.0E-10 |
| Nb95 | 9.1E-12 | Water Activation Products | |
| Mo99 | 1.1E-08 | C14 | 2.6E-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 | 3.7E-14 | Mo101 | 1.7E-15 |
| Kr85m | 1.6E-13 | Tc99m | 5.1E-14 |
| Kr85 | 2.8E-11 | Tc99 | 1.0E-18 |
| Kr87 | 8.5E-14 | Ru103 | 5.3E-17 |
| Kr88 | 2.5E-13 | Ru105 | 1.7E-17 |
| Kr89 | 5.6E-15 | Ru106 | 3.3E-17 |
| Xe131m | 6.6E-13 | Rh103m | 4.9E-17 |
| Xe133m | 5.6E-13 | Rh105 | 3.6E-17 |
| Xe133 | 4.2E-11 | Rh106 | 4.4E-18 |
| Xe135m | 5.3E-14 | Ag110 | 4.4E-18 |
| Xe135 | 1.1E-12 | Sb124 | 7.9E-20 |
| Xe137 | 1.9E-14 | Sb125 | 5.9E-19 |
| Xe138 | 6.3E-14 | Sb127 | 3.0E-18 |
| Halogens | | Sb129 | 3.6E-18 |
| Br82 | 1.0E-15 | Te125m | 8.5E-17 |
| Br83 | 5.8E-15 | Te127m | 3.2E-16 |
| Br84 | 2.5E-15 | Te127 | 1.3E-15 |
| Br85 | 1.6E-16 | Te129m | 9.3E-16 |
| I129 | 1.7E-20 | Te129 | 1.3E-15 |
| I130 | 8.4E-15 | Te131m | 3.0E-15 |
| I131 | 2.2E-13 | Te131 | 1.3E-15 |
| I132 | 9.8E-14 | Te132 | 2.2E-14 |
| I133 | 3.3E-13 | Te133m | 1.8E-15 |
| I134 | 5.5E-14 | Te134 | 2.5E-15 |
| I135 | 2.1E-13 | Ba137m | 4.5E-14 |
| Rubidium, Cesium | | Ba139 | 4.8E-17 |
| Rb86m | 6.1E-20 | Ba140 | 2.7E-16 |
| Rb86 | 1.6E-15 | La140 | 7.9E-17 |
| Rb88 | 2.3E-13 | La141 | 1.5E-17 |
| Rb89 | 1.0E-14 | La142 | 7.2E-18 |
| Cs132 | 3.2E-17 | Ce141 | 4.2E-17 |
| Cs134 | 2.3E-13 | Ce143 | 3.2E-17 |
| Cs135m | 1.8E-16 | Ce144 | 3.5E-17 |
| Cs136 | 5.0E-14 | Pr143 | 3.7E-17 |
| Cs137 | 1.2E-13 | Pr144 | 3.0E-17 |
| Cs138 | 9.3E-14 | Np239 | 6.6E-16 |
| Other FPs | | Corrosion/Activation Products - CRUD | |
| P32 | 4.1E-21 | Na24 | 6.7E-14 |
| Co57 | 3.1E-23 | Cr51 | 3.8E-15 |
| Sr89 | 1.9E-16 | Mn54 | 1.9E-15 |
| Sr90 | 2.9E-17 | Fe55 | 1.4E-15 |
| Sr91 | 9.6E-17 | Fe59 | 3.6E-16 |
| Sr92 | 5.1E-17 | Co58 | 5.6E-15 |
| Y90 | 7.0E-18 | Co60 | 6.4E-16 |
| Y91m | 4.9E-17 | Ni63 | 3.2E-16 |
| Y91 | 2.7E-17 | Zn65 | 6.1E-16 |
| Y92 | 4.3E-17 | Zr95 | 4.7E-16 |
| Y93 | 2.1E-17 | Ag110m | 1.6E-15 |

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

| Nuclide | Secondary Coolant Concentrations (Ci/g) | Nuclide | Secondary Coolant Concentrations (Ci/g) |
|---|---|---|---|
| Zr97 | 3.1E-17 | W187 | 3.4E-15 |
| Nb95 | 4.4E-17 | Water Activation Products | |
| Mo99 | 5.5E-14 | C14 | 1.3E-15 |
| | | Ar41 | 1.0E-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 | 7.7E-10 | Mo101 | 4.3E-11 |
| Kr85m | 3.2E-09 | Tc99m | 1.0E-09 |
| Kr85 | 1.6E-07 | Tc99 | 2.1E-14 |
| Kr87 | 1.8E-09 | Ru103 | 1.1E-12 |
| Kr88 | 5.1E-09 | Ru105 | 3.6E-13 |
| Kr89 | 1.2E-10 | Ru106 | 6.8E-13 |
| Xe131m | 1.3E-08 | Rh103m | 1.1E-12 |
| Xe133m | 1.1E-08 | Rh105 | 7.3E-13 |
| Xe133 | 8.3E-07 | Rh106 | 6.8E-13 |
| Xe135m | 1.1E-09 | Ag110 | 4.8E-12 |
| Xe135 | 2.3E-08 | Sb124 | 1.6E-15 |
| Xe137 | 3.9E-10 | Sb125 | 1.2E-14 |
| Xe138 | 1.3E-09 | Sb127 | 6.1E-14 |
| Halogens | | Sb129 | 7.6E-14 |
| Br82 | 2.1E-11 | Te125m | 1.7E-12 |
| Br83 | 1.2E-10 | Te127m | 6.6E-12 |
| Br84 | 5.7E-11 | Te127 | 2.6E-11 |
| Br85 | 6.9E-12 | Te129m | 1.9E-11 |
| I129 | 3.5E-16 | Te129 | 2.7E-11 |
| I130 | 1.7E-10 | Te131m | 6.2E-11 |
| I131 | 4.4E-09 | Te131 | 3.1E-11 |
| I132 | 2.1E-09 | Te132 | 4.6E-10 |
| I133 | 6.7E-09 | Te133m | 3.9E-11 |
| I134 | 1.2E-09 | Te134 | 5.6E-11 |
| I135 | 4.3E-09 | Ba137m | 2.1E-09 |
| Rubidium, Cesium | | Ba139 | 1.0E-12 |
| Rb86m | 5.2E-15 | Ba140 | 5.6E-12 |
| Rb86 | 3.0E-11 | La140 | 1.6E-12 |
| Rb88 | 5.1E-09 | La141 | 3.2E-13 |
| Rb89 | 2.4E-10 | La142 | 1.5E-13 |
| Cs132 | 6.0E-13 | Ce141 | 8.6E-13 |
| Cs134 | 4.3E-09 | Ce143 | 6.5E-13 |
| Cs135m | 3.6E-12 | Ce144 | 7.3E-13 |
| Cs136 | 9.4E-10 | Pr143 | 7.6E-13 |
| Cs137 | 2.2E-09 | Pr144 | 7.2E-13 |
| Cs138 | 1.9E-09 | Np239 | 1.4E-11 |
| Other FPs | | Corrosion/Activation Products - CRUD | |
| P32 | 8.4E-17 | Na24 | 1.4E-08 |
| Co57 | 6.4E-19 | Cr51 | 7.7E-10 |
| Sr89 | 3.8E-12 | Mn54 | 4.0E-10 |
| Sr90 | 5.9E-13 | Fe55 | 3.0E-10 |
| Sr91 | 2.0E-12 | Fe59 | 7.5E-11 |
| Sr92 | 1.1E-12 | Co58 | 1.1E-09 |
| Y90 | 1.4E-13 | Co60 | 1.3E-10 |
| Y91m | 1.1E-12 | Ni63 | 6.6E-11 |
| Y91 | 5.6E-13 | Zn65 | 1.3E-10 |
| Y92 | 9.0E-13 | Zr95 | 9.7E-11 |
| Y93 | 4.3E-13 | Ag110m | 3.2E-10 |

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

| Nuclide | Primary Coolant Concentrations (Ci/g) | Nuclide | Primary Coolant Concentrations (Ci/g) |
|---|---|---|---|
| Zr97 | 6.3E-13 | W187 | 7.0E-10 |
| Nb95 | 1.6E-12 | Water Activation Products | |
| Mo99 | 1.1E-09 | C14 | 2.6E-10 |
| | | Ar41 | 1.4E-07 |

## 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 | 2.7E-16 | Mo101 | 1.3E-17 |
| Kr85m | 1.1E-15 | Tc99m | 3.7E-16 |
| Kr85 | 5.7E-14 | Tc99 | 7.6E-21 |
| Kr87 | 6.2E-16 | Ru103 | 3.9E-19 |
| Kr88 | 1.8E-15 | Ru105 | 1.3E-19 |
| Kr89 | 4.2E-17 | Ru106 | 2.4E-19 |
| Xe131m | 4.5E-15 | Rh103m | 3.6E-19 |
| Xe133m | 4.0E-15 | Rh105 | 2.6E-19 |
| Xe133 | 2.9E-13 | Rh106 | 3.2E-20 |
| Xe135m | 3.9E-16 | Ag110 | 1.9E-19 |
| Xe135 | 8.1E-15 | Sb124 | 5.8E-22 |
| Xe137 | 1.4E-16 | Sb125 | 4.3E-21 |
| Xe138 | 4.7E-16 | Sb127 | 2.2E-20 |
| Halogens | | Sb129 | 2.7E-20 |
| Br82 | 7.6E-18 | Te125m | 6.2E-19 |
| Br83 | 4.3E-17 | Te127m | 2.4E-18 |
| Br84 | 1.8E-17 | Te127 | 9.4E-18 |
| Br85 | 1.2E-18 | Te129m | 6.8E-18 |
| I129 | 1.3E-22 | Te129 | 9.3E-18 |
| I130 | 6.2E-17 | Te131m | 2.2E-17 |
| I131 | 1.6E-15 | Te131 | 9.8E-18 |
| I132 | 7.2E-16 | Te132 | 1.6E-16 |
| I133 | 2.4E-15 | Te133m | 1.3E-17 |
| I134 | 4.1E-16 | Te134 | 1.8E-17 |
| I135 | 1.5E-15 | Ba137m | 3.3E-16 |
| Rubidium, Cesium | | Ba139 | 3.6E-19 |
| Rb86m | 4.5E-22 | Ba140 | 2.0E-18 |
| Rb86 | 1.2E-17 | La140 | 5.8E-19 |
| Rb88 | 1.7E-15 | La141 | 1.1E-19 |
| Rb89 | 7.5E-17 | La142 | 5.3E-20 |
| Cs132 | 2.3E-19 | Ce141 | 3.1E-19 |
| Cs134 | 1.7E-15 | Ce143 | 2.3E-19 |
| Cs135m | 1.3E-18 | Ce144 | 2.6E-19 |
| Cs136 | 3.7E-16 | Pr143 | 2.7E-19 |
| Cs137 | 8.7E-16 | Pr144 | 2.2E-19 |
| Cs138 | 6.8E-16 | Np239 | 4.9E-18 |
| Other FPs | | Corrosion/Activation Products - CRUD | |
| P32 | 3.0E-23 | Na24 | 4.9E-15 |
| Co57 | 2.3E-25 | Cr51 | 2.8E-16 |
| Sr89 | 1.4E-18 | Mn54 | 1.4E-16 |
| Sr90 | 2.1E-19 | Fe55 | 1.1E-16 |
| Sr91 | 7.1E-19 | Fe59 | 2.7E-17 |
| Sr92 | 3.7E-19 | Co58 | 4.1E-16 |
| Y90 | 5.2E-20 | Co60 | 4.7E-17 |
| Y91m | 3.6E-19 | Ni63 | 2.3E-17 |
| Y91 | 2.0E-19 | Zn65 | 4.5E-17 |
| Y92 | 3.2E-19 | Zr95 | 3.5E-17 |
| Y93 | 1.5E-19 | Ag110m | 1.2E-16 |

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

| Nuclide | Secondary Coolant Concentrations (Ci/g) | Nuclide | Secondary Coolant Concentrations (Ci/g) |
|---|---|---|---|
| Zr97 | 2.2E-19 | W187 | 2.5E-16 |
| Nb95 | 5.6E-19 | Water Activation Products | |
| Mo99 | 4.0E-16 | C14 | 9.4E-17 |
| | | Ar41 | 5.0E-14 |

**Table 11.1-8: Tritium Concentration versus Primary Coolant Recycling Modes**

| Recycle Mode | Primary Coolant Average Concentration (Ci/g) | Reactor Coolant System Letdown / CVCS Outlet (Ci/g) | Realistic Secondary Coolant Concentration (Ci/g) | Design Basis Secondary Coolant Concentration (Ci/g) |
|---|---|---|---|---|
| No recycle (discharge) | 1.3E-06 | 1.0E-06 | 2.4E-09 | --- |
| Recycle to reactor pool makeup | 1.3E-06 | 1.0E-06 | 2.5E-09 | --- |
| Recycle back to CVCS makeup | 2.6E-06 | 2.6E-06 | --- | 4.7E-09 |

Note: The maximum calculated peak primary coolant tritium concentration is 3.3 μCi/g.

## 11.2 Liquid Waste Management System

The liquid waste management system is called the liquid radioactive waste system (LRWS). The LRWS 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), mixtures from the boron addition system, waste water from pool cooling and cleanup system (PCWS), contaminated liquids from the balance-of-plant drain system, and the radioactive waste drain system (RWDS). The LRWS components are located in the Reactor Building (RXB) and in the Radioactive Waste Building (RWB).

### 11.2.1 Design Bases

The LRWS has no safety-related function and is not risk significant. A failure of the LRWS does not adversely affect safety-related systems or components. 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 are considered in the design of the LRWS. Section 11.2.2.6 contains further detail.

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). A design objective of the LRWS is to provide the capability for sampling to ensure that liquid releases of radioactive material in liquid effluents are ALARA. Section 12.1 contains more detail about ways ALARA is implemented into the design.

### 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 LRWS collects, processes, and releases radioactive and potentially radioactive liquid wastes produced by the plant during the plant lifecycle. Separate collection tanks are provided for low-conductivity waste subsystem (LCW), high-conductivity waste subsystem (HCW), and detergent wastes. Oily waste is removed by an oil separator, collected in drums and sent to the SRWS for eventual shipment offsite. The remaining fluid in the separator is sent to the HCW Collection Tanks. Chemical wastes are collected as a part of the RWDS. Mixed wastes are collected locally in drums and sent offsite. The LRWS processing equipment flow path consists of two preconditioning filter vessels, a solids collection filter, three accumulator vessels, five ion exchange vessels, a reverse osmosis skid, and four polisher vessels.

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. 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 before recycling or release. If the water quality is not acceptable, the water is returned to a collection tank for further treatment.

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 provisions to reduce the radiation source term before maintenance
- redundant components allow uninterrupted waste operation and flexibility in maintenance scheduling

When the sample results of the LRWS meet discharge limits, the utility water system (UWS) discharges the treated effluent to the environment. The UWS dilutes the liquid effluent further before discharge.

#### 11.2.2.1 Low Conductivity Waste Subsystem

The LCW consists of coolant-grade boron and hydrogen-containing wastes with high radioactivity concentrations. The CVCS letdown during normal operation and reactor heatup, along with the pressurizer vent letdown before and during reactor shutdown, are routed to the LRWS degasifiers for hydrogen and fission gas removal. The stripped gases are sent to the gaseous radioactive waste system (GRWS) for holdup and radioactive decay. The liquid waste is routed to the LCW collection tanks. The LCW collection tank fluid can be sent back to the degasifiers for additional gas removal if sample results determine it necessary. The LCW collection tanks have connections to be purged with nitrogen for inerting, as required.

The LCW collection tanks also receive RXB and RWB equipment drains from the RWDS, liquid from the SRWS spent resin storage tanks, liquid from the SRWS phase separator tanks, liquid from the SRWS dewatering process, and pool water from the PCWS and out-of-specification boric acid batches from the boron addition system.

The LCW collection tanks provide for sampling of the waste on a batch basis before sending to the LCW processing equipment. The LCW processing equipment is designed to handle filtering and removal of radioactive waste. The treated liquid waste is routed to the LCW sample tanks for sampling. The treated effluent is either recycled for use within the plant or discharged to the environment through the UWS. The UWS dilutes the liquid effluent further before discharge to the environment. If the LCW sample tank sample results do not meet specified requirements, the waste can be returned to the collection tanks for reprocessing. Reactor coolant letdown may be sent from the LCW collection tanks to the LCW sample tanks and then back to CVCS, bypassing the LCW processing equipment. Treated liquid waste may also be routed to the HCW sample tanks or the drum dryer skid.

#### 11.2.2.2 High Conductivity Waste Subsystem

The HCW consists of liquid radioactive waste containing a varying degree of suspended solids and low radioactivity concentration. The HCW collection tanks collect waste from the following sources:

- The RXB floor drains through RWDS
- The RWB floor drains through RWDS
- The balance-of-plant drain system chemical waste collection tank
- The PCWS pool surge control tank overflow
- The RWDS Reactor Building reactor component cooling water system drain tank
- The RWDS Reactor Building chemical drain tank

The HCW collection tank contents can be sampled before sending to the HCW processing equipment. The HCW processing equipment contains two carbon filter vessels. The HCW may be routed through the HCW processing equipment to the LCW processing equipment for additional treatment. Treated effluent is routed to the HCW sample tanks for sampling. When the sample result meets discharge limits, the treated effluent is discharged to the environment via UWS, or it is recycled for use within the plant. If needed, the waste can be returned to the HCW or LCW collection tanks for reprocessing.

#### 11.2.2.3 Chemical Waste Processing

Chemical waste is collected in the RWDS. If it is contaminated, operators send it to the HCW collection tanks. The RCCWS drains are collected separately in the RWDS to prevent the introduction of nitrite into resins that may be used in the LCW processing equipment to avoid the potential for exothermic reactions. The RCCWS drains are collected and either returned to the RCCWS as makeup, routed to the HCW processing equipment, or discharged.

#### 11.2.2.4 Detergent Waste Subsystem

Detergent wastes from personnel decontamination showers and small component decontamination sinks are collected in a dedicated collection tank and sampled. Wastes are then discharged through a cartridge filter if the sample results indicate that the specified requirements are satisfied. If the detergent sample result is not within the discharge limits, the contents go to the drum dryer skid. Detergent wastes are collected and processed separately to prevent degradation of the HCW and LCW processing equipment. There is a single train of detergent waste due to the low volume of waste generation expected, as there is no on-site laundry for the design.

#### 11.2.2.5 Off-Normal Operations

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

#### 11.2.2.6 Safety Evaluation

The LRWS complies with the following GDC:

- 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. 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.

Features are designed 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.

The design of the principle components, piping, and valves that contain radioactive fluids comply with 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 described in Section 3.2 and presented in Table 3.2-1.

## 11.2.3 Radioactive Effluent Releases

### 11.2.3.1 Radioactive Releases

The system 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 UWS discharge basin.

The calculation of liquid effluent releases is consistent with RG 1.112, as modified by Technical Report TR-123242 (Reference 11.2-1). The calculation of off-site dose consequences from normal liquid effluents is consistent with RG 1.109.

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 11.2-1: An applicant that references the NuScale Power Plant US460 standard design 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 PCWS pool surge control storage tank, described in FSAR Section 9.1. The PCWS pool surge control storage tank secondary containment tank has sufficient volume to store the contents of the PCWS pool surge control storage tank plus the contents of related piping. The radionuclide concentration water of the PCWS pool surge control storage tank is provided in Table 12.2-9 with Table 12.2-8 providing the water mass.

COL Item 11.2-2: An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific evaluation of the consequences of an accidental release of radioactive liquid from the pool surge control storage tank in accordance with NRC Branch Technical Position 11-6.

#### 11.2.3.3 Dilution Factors

The utility water discharge flow rate is credited in the calculation of the discharge concentrations, as described in Reference 11.2-1. The design ensures that the discharge concentrations are within 10 CFR 20 Appendix B, Table 2, limits. The unrestricted area doses are calculated using a dilution factor as shown in Table 11.2-6, which results in the unrestricted area doses being within 10 CFR 50, Appendix I limits.

COL Item 11.2-3: An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific evaluation using the site-specific source term and dilution flow for liquid effluent releases, and confirm that the discharge concentrations do not exceed the limits specified by 10 CFR 20, Appendix B, Table 2.

#### 11.2.3.4 Site-Specific Cost-Benefit Analysis

COL Item 11.2-4: An applicant that references the NuScale Power Plant US460 standard design 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

Section 14.2 describes the LRWS preoperational tests and includes the applicable testing and inspection requirements from RG 1.143.

The design incorporates inspection and testing provisions to enable periodic evaluation of the operability and requires functional performance of active components of the system.

### 11.2.5 Instrumentation and Controls

The 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 automatically realign to direct the incoming waste flows toward the collection tank that is 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 leak detection 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 automatically isolates 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.

### 11.2.6 Reference

11.2-1 NuScale Power, LLC, "Effluent Release (GALE Replacement) Methodology and Results," TR-123242, Revision 0.

## 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 | Vertical | 12,500 gallons | 150 | 550 | Stainless Steel | Table 12.2-15a |
| Degasifier Liquid Transfer Pumps (2) | RW-IIc | Sealless Centrifugal | 28 gpm | 150 | 210 | Stainless Steel | - |
| LCW collection tank (2) | RW-IIc | Vertical, conical | 16,000 gallons | 15 | 240 | Stainless Steel | Table 12.2-12a |
| LCW collection tank transfer pump (2) | RW-IIc | Sealless Centrifugal | 39 gpm | 290 | 155 | Stainless Steel | - |
| HCW collection tank (2) | RW-IIc | Vertical Conical | 16,000 gallons | 15 | 200 | Stainless Steel | Table 12.2-12a |
| HCW collection tank transfer pump (2) | RW-IIc | Sealless Centrifugal | 39 gpm | 230 | 155 | Stainless Steel | - |
| LCW sample tank (2) | RW-IIc | Vertical conical | 16,000 gallons | 15 | 155 | Stainless Steel | Table 12.2-12b |
| LCW sample tank transfer pump (2) | RW-IIc | Sealless Centrifugal | 28 gpm | 150 | 155 | Stainless Steel | - |
| HCW sample tank (2) | RW-IIc | Vertical conical | 16,000 gallons | 15 | 155 | Stainless Steel | Table 12.2-12b |
| HCW sample tank transfer pump (2) | RW-IIc | Sealless Centrifugal | 28 gpm | 150 | 155 | Stainless Steel | - |
| Oil separator (1) | RW-IIc | - | 240 gpm | 150 | 155 | Stainless Steel | Table 12.2-12a |
| Detergent waste collection tank (1) | RW-IIc | Vertical conical | 500 gallons | 15 | 200 | Stainless Steel | - |
| Detergent waste drain filter (1) | RW-IIc | Cartridge | 20 micron | 150 | 155 | Stainless Steel | - |
| Demineralized water break tank (1) | RW-IIc | Vertical | 10,000 gallons | 15 | 155 | Stainless Steel | - |
| HCW Processing charcoal filter (2) | RW-IIb | Vertical Vessel | 35 gpm | 230 | 155 | Stainless Steel | LCW-Table 12.2-12a<br>HCW- Table 12.2-12b |
| LCW Reverse Osmosis Skid (1) | RW-IIc | Vertical | 35 gpm | 290 | 155 | Stainless Steel | LCW-Table 12.2-12a<br>HCW- Table 12.2-12b |
| Clean-In-Place Skid (1) | RW-IIc | - | 55 gallons | 150 | 155 | Stainless Steel | - |
| Drum Dryer Skids (1) | RW-IIc | - | 55 gpd | 15 | 155 | Stainless Steel | - |
| LCW Pre-conditioning filter vessels (2) | RW-IIa | - | 35 gpm | 290 | 155 | Stainless Steel | |
| LCW accumulator vessels (3) | RW-IIa | - | 35 gpm | 290 | 155 | Stainless Steel | |
| LCW Ion Exchange vessel (5) | RW-IIa | - | 35 gpm | 290 | 155 | Stainless Steel | Table 12.2-12a |
| LCW solids collection filter (1) | RW-IIc | - | 35 gpm | 290 | 155 | Stainless Steel | |
| LCW Polishing Ion Exchange Vessel (4) | RW-IIc | - | 35 gpm | 290 | 155 | Stainless Steel | Table 12.2-12a |
| Demineralized water break tank transfer pump (1) | RW-IIc | Sealless centrifugal | 220 gpm | 150 | 155 | 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 when in use | Automatic | Switch to standby degasifier | Corrective maintenance of idle equipment |
| Degasifier transfer pump trips | Automatic | Switch to standby degasifier | Corrective maintenance of idle equipment |
| Collection Tank Transfer Pump failure | Automatic | Switch to standby Collection Tank Transfer Pump | Corrective maintenance of idle equipment |
| High-high level in Collection or Sample Tanks | Operator surveillance | Discharge valves automatically close and runnning pump stops | Investigate cause and procedures to prevent challenges to control system |
| One train of processing equipment inoperable | Operator surveillance | Based on sample results, use one train to alternately process LCW and HCW | Repair or replace components and restore operability |
| Sample Tank shows sample of high radioactivity | Review of sample results | Pump to Collection Tank for reprocessing | Diagnose cause and repair or replace as necessary |
| High radiation on single point LRW discharge | Automatic | Discharge valves automatically close and runnning pump stops | Diagnose cause, reprocess remaining sample tank contents |
| Low guard pipe pressure on buried LRW discharge pipe | Automatic | Discharge valves automatically close and running | Diagnose cause, repair buried LRW discharge pipe |
| Area radiation alarm | Local and MCR alarm | pump stops | Investigate cause and initiate cleanup and corrective maintenance |
| Leaks or spills | Operator surveillance (or leak detection alarm or area radiation alarm) | Suspend processing, prevent the spread of contamination | Investigate cause and initiate cleanup and corrective maintenance |
| Loss of Nitrogen pressure to Degasifiers | Indication from vendor control system | Vendor control system isolates degasifier skid | Investigate, restore Nitrogen pressure and resume operation |

## Table 11.2-3: Expected Liquid Waste Inputs

| LRWS Input Source | Expected Input Rate (6 NPMs) (gpy) | Expected Activity |
|---|---|---|
| LCW collection tank | | |
| RXB/RWB equipment drains | 2.9E+04 | 0.001 primary coolant activity (PCA) |
| Other equipment drains | 1.1E+04 | 0.093 PCA |
| Normal letdown (six operating units) | 1.9E+05 | CVCS outlet |
| Degasification prior to shutdown (six events per year) | 3.0E+03 | primary coolant through evaporator |
| Additional CVCS letdown streams | 3.8E+04 | CVCS outlet |
| **LCW Total** | **2.7E+05** | |
| HCW collection tank | | |
| RXB/RWB floor drains (via oil separator) | 7.3E+04 | 0.1 PCA |
| RXB RCCW drain tank (via oil separator) | 3.6E+01 | 0.001 PCA |
| Hot machine shop, decontamination room sump (via oil separator) | 9.0E+04 | 0.01 PCA |
| RXB chemical drain tank (Hot lab sink) (via oil separator) | 4.4E+03 | 0.05 PCA |
| RXB chemical drain tank (CES sample tank) (via oil separator) | 2.2E+04 | primary coolant through evaporator |
| Pump seal leaks (via oil separator) | 8.1E+03 | 0.1 PCA |
| Valve packing leaks (via oil separator) | 4.8E+03 | 0.1 PCA |
| Groundwater / Condensation (via oil separator) | 2.5E+05 | 0.001 PCA |
| Equipment area decontamination (outside hot machine shop) (via oil separator) | 1.5E+04 | 0.01 PCA |
| Secondary coolant sampling drains | 4.2E+03 | Secondary coolant |
| Condensate polisher rinse and transfer | 3.6E+04 | Secondary coolant |
| Condensate polisher regeneration solutions | 1.0E+04 | Secondary coolant |
| Turbine Generator Building floor drains | 2.2E+04 | Secondary coolant |
| Pool water source streams | 2.9E+05 | Pool water source term |
| CVCS outlet sources | 3.5E+04 | CVCS outlet |
| **HCW Total** | **8.6E+05** | |

Note: Assumes six NPMs operating on an 18-month refueling cycle.

## Table 11.2-4: Liquid Effluent Release Calculation Inputs

| NuScale Effluent Source Term Model Assumption | Value | Units |
|---|---|---|
| Primary coolant source term | Table 11.1-6 | |
| CVCS demineralizer decontamination factors- | | |
| - Halogens | 100 | -- |
| - Cs, Rb | 2 | -- |
| - Others | 50 | -- |
| Pool water source | Table 12.2-9 | |
| CES liquid partition fractions-and PCA through Evaporator | | |
| - Noble gases | 1 | -- |
| - Halogens | 100 | -- |
| - Others | 1000 | -- |
| Secondary coolant source term | Table 11.1-7 | |
| AOO adjustment | 0.07 | Ci/year |
| UWS dilution factor for 10 CFR 20 Appendix B | 700 | gpm |

## 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 | 1.4E-12 | 2.3E-09 | 2.3E-09 | 2.8E-09 |
| Br83 | - | 2.5E-30 | 2.5E-30 | 3.1E-30 |
| I129 | 4.9E-13 | 1.4E-12 | 1.9E-12 | 2.4E-12 |
| I130 | 1.2E-19 | 2.1E-11 | 2.1E-11 | 2.6E-11 |
| I131 | 1.0E-06 | 1.6E-05 | 1.7E-05 | 2.1E-05 |
| I132 | 1.6E-08 | 3.6E-07 | 3.8E-07 | 4.7E-07 |
| I133 | 4.5E-13 | 5.8E-08 | 5.8E-08 | 7.3E-08 |
| I135 | 4.1E-29 | 5.3E-14 | 5.3E-14 | 6.6E-14 |
| Rb86 | 3.6E-06 | 1.4E-06 | 5.0E-06 | 6.3E-06 |
| Cs132 | 1.6E-08 | 1.6E-08 | 3.3E-08 | 4.1E-08 |
| Cs134 | 1.1E-03 | 2.7E-04 | 1.4E-03 | 1.7E-03 |
| Cs136 | 8.1E-05 | 3.9E-05 | 1.2E-04 | 1.5E-04 |
| Cs137 | 5.8E-04 | 1.4E-04 | 7.2E-04 | 9.0E-04 |
| P32 | 9.8E-14 | 2.4E-13 | 3.4E-13 | 4.2E-13 |
| Co57 | 2.0E-15 | 2.6E-15 | 4.5E-15 | 5.7E-15 |
| Sr89 | 1.9E-08 | 1.7E-08 | 3.6E-08 | 4.5E-08 |
| Sr90 | 1.9E-09 | 2.5E-09 | 4.4E-09 | 5.5E-09 |
| Sr91 | 1.0E-24 | 1.3E-14 | 1.3E-14 | 1.6E-14 |
| Sr92 | - | 4.1E-30 | 4.1E-30 | 5.2E-30 |
| Y90 | 1.9E-09 | 2.2E-09 | 4.1E-09 | 5.2E-09 |
| Y91m | 6.5E-25 | 8.1E-15 | 8.1E-15 | 1.0E-14 |
| Y91 | 1.4E-09 | 2.1E-09 | 3.5E-09 | 4.4E-09 |
| Y92 | - | 2.7E-24 | 2.7E-24 | 3.4E-24 |
| Y93 | 1.6E-24 | 5.6E-15 | 5.6E-15 | 7.0E-15 |
| Zr97 | 1.7E-18 | 1.2E-12 | 1.2E-12 | 1.5E-12 |
| Nb95 | 9.7E-08 | 6.7E-07 | 7.7E-07 | 9.7E-07 |
| Mo99 | 1.8E-08 | 6.6E-07 | 6.8E-07 | 8.5E-07 |
| Tc99m | 1.7E-08 | 6.4E-07 | 6.6E-07 | 8.2E-07 |
| Tc99 | 6.9E-11 | 8.9E-11 | 1.6E-10 | 2.0E-10 |
| Ru103 | 2.4E-09 | 3.9E-09 | 6.3E-09 | 7.9E-09 |
| Ru105 | - | 3.7E-22 | 3.7E-22 | 4.7E-22 |
| Ru106 | 2.1E-09 | 2.8E-09 | 4.9E-09 | 6.1E-09 |
| Rh103m | 2.4E-09 | 3.9E-09 | 6.3E-09 | 7.8E-09 |
| Rh105 | 1.3E-13 | 8.5E-11 | 8.5E-11 | 1.1E-10 |
| Rh106 | 2.1E-09 | 2.8E-09 | 4.9E-09 | 6.1E-09 |
| Ag110 | 1.3E-08 | 1.8E-07 | 1.9E-07 | 2.4E-07 |
| Sb124 | 4.1E-12 | 6.1E-12 | 1.0E-11 | 1.3E-11 |
| Sb125 | 3.9E-11 | 5.0E-11 | 8.8E-11 | 1.1E-10 |
| Sb127 | 4.4E-12 | 6.2E-11 | 6.7E-11 | 8.4E-11 |
| Sb129 | - | 6.0E-23 | 6.0E-23 | 7.5E-23 |
| Te125m | 4.4E-09 | 6.6E-09 | 1.1E-08 | 1.4E-08 |
| Te127m | 1.9E-08 | 2.6E-08 | 4.5E-08 | 5.6E-08 |
| Te127 | 1.8E-08 | 2.6E-08 | 4.4E-08 | 5.5E-08 |

## 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 | 4.0E-08 | 6.7E-08 | 1.1E-07 | 1.3E-07 |
| Te129 | 2.5E-08 | 4.2E-08 | 6.8E-08 | 8.5E-08 |
| Te131m | 1.7E-12 | 3.5E-09 | 3.5E-09 | 4.4E-09 |
| Te131 | 3.9E-13 | 7.9E-10 | 7.9E-10 | 9.9E-10 |
| Te132 | 1.5E-08 | 3.5E-07 | 3.7E-07 | 4.6E-07 |
| Ba137m | 5.5E-04 | 1.3E-04 | 6.8E-04 | 8.5E-04 |
| Ba140 | 5.8E-09 | 1.5E-08 | 2.1E-08 | 2.6E-08 |
| La140 | 6.6E-09 | 1.7E-08 | 2.3E-08 | 2.9E-08 |
| La141 | - | 7.1E-24 | 7.1E-24 | 8.9E-24 |
| Ce141 | 1.8E-09 | 3.0E-09 | 4.8E-09 | 6.0E-09 |
| Ce143 | 5.2E-14 | 5.4E-11 | 5.4E-11 | 6.8E-11 |
| Ce144 | 2.2E-09 | 2.9E-09 | 5.2E-09 | 6.5E-09 |
| Pr143 | 9.2E-10 | 2.3E-09 | 3.2E-09 | 4.0E-09 |
| Pr144 | 2.2E-09 | 2.9E-09 | 5.1E-09 | 6.4E-09 |
| Np239 | 9.0E-11 | 5.8E-09 | 5.8E-09 | 7.3E-09 |
| Na24 | 3.0E-15 | 1.0E-08 | 1.0E-08 | 1.3E-08 |
| Cr51 | 1.5E-06 | 2.5E-05 | 2.7E-05 | 3.3E-05 |
| Mn54 | 1.2E-06 | 1.6E-05 | 1.7E-05 | 2.2E-05 |
| Fe55 | 9.5E-07 | 1.2E-05 | 1.3E-05 | 1.7E-05 |
| Fe59 | 1.7E-07 | 2.7E-06 | 2.9E-06 | 3.6E-06 |
| Co58 | 3.0E-06 | 3.9E-04 | 4.0E-04 | 5.0E-04 |
| Co60 | 4.2E-07 | 5.5E-06 | 5.9E-06 | 7.4E-06 |
| Ni63 | 2.1E-07 | 2.7E-06 | 3.0E-06 | 3.7E-06 |
| Zn65 | 3.9E-07 | 5.1E-06 | 5.5E-06 | 6.9E-06 |
| Zr95 | 2.5E-07 | 3.6E-06 | 3.9E-06 | 4.9E-06 |
| Ag110m | 9.9E-07 | 1.3E-05 | 1.4E-05 | 1.8E-05 |
| W187 | 8.7E-13 | 3.6E-08 | 3.6E-08 | 4.5E-08 |
| H3 | 8.6E+02 | 3.0E+02 | 1.2E+03 | 1.2E+03 |
| C14 | 2.3E-01 | 4.6E-02 | 2.8E-01 | 3.5E-01 |
| **Total** | **8.6E+02** | **3.0E+02** | **1.2E+03** | **1.2E+03** |

**Table 11.2-6: LADTAP II Inputs**

| Parameter | Value | Units |
|---|---|---|
| Source term | Table 11.2-5 | -- |
| Shore-width factor | 1.0 | -- |
| Discharge flow rate | 590 | gpm |
| 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 | -- |
| Minimum dilution flow rate | 240 | cfs |

### 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 | 2.9 | 3 |
| Individual Organ | 9.8 | 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 | 2.0E-15 | 4.0E-05 | 5.1E-11 |
| I129 | 1.7E-18 | 2.0E-07 | 8.6E-12 |
| I130 | 1.9E-17 | 2.0E-05 | 9.5E-13 |
| I131 | 1.5E-11 | 1.0E-06 | 1.5E-05 |
| I132 | 3.4E-13 | 1.0E-04 | 3.4E-09 |
| I133 | 5.2E-14 | 7.0E-06 | 7.5E-09 |
| I135 | 4.7E-20 | 3.0E-05 | 1.6E-15 |
| Rb86 | 4.5E-12 | 7.0E-06 | 6.4E-07 |
| Cs132 | 2.9E-14 | 4.0E-05 | 7.3E-10 |
| Cs134 | 1.2E-09 | 9.0E-07 | 1.4E-03 |
| Cs136 | 1.1E-10 | 6.0E-06 | 1.8E-05 |
| Cs137 | 6.4E-10 | 1.0E-06 | 6.4E-04 |
| Co57 | 4.1E-21 | 6.0E-05 | 6.8E-17 |
| Sr89 | 3.2E-14 | 8.0E-06 | 4.0E-09 |
| Sr90 | 3.9E-15 | 5.0E-07 | 7.9E-09 |
| Sr91 | 1.1E-20 | 2.0E-05 | 5.7E-16 |
| Y90 | 3.7E-15 | 7.0E-06 | 5.3E-10 |
| Y91m | 7.2E-21 | 2.0E-03 | 3.6E-18 |
| Y91 | 3.2E-15 | 8.0E-06 | 4.0E-10 |
| Y92 | 2.4E-30 | 4.0E-05 | 6.0E-26 |
| Y93 | 5.0E-21 | 2.0E-05 | 2.5E-16 |
| Zr97 | 1.1E-18 | 9.0E-06 | 1.2E-13 |
| Nb95 | 6.9E-13 | 3.0E-05 | 2.3E-08 |
| Mo99 | 6.1E-13 | 2.0E-05 | 3.1E-08 |
| Tc99m | 5.9E-13 | 1.0E-03 | 5.9E-10 |
| Tc99 | 1.4E-16 | 6.0E-05 | 2.4E-12 |
| Ru103 | 5.7E-15 | 3.0E-05 | 1.9E-10 |
| Ru105 | 3.3E-28 | 7.0E-05 | 4.8E-24 |
| Ru106 | 4.4E-15 | 3.0E-06 | 1.5E-09 |
| Rh103m | 5.6E-15 | 6.0E-03 | 9.4E-13 |
| Rh105 | 7.6E-17 | 5.0E-05 | 1.5E-12 |
| Sb124 | 9.2E-18 | 7.0E-06 | 1.3E-12 |
| Sb125 | 7.9E-17 | 3.0E-05 | 2.6E-12 |
| Sb127 | 6.0E-17 | 1.0E-05 | 6.0E-12 |
| Sb129 | 5.4E-29 | 4.0E-05 | 1.4E-24 |
| Te125m | 9.9E-15 | 2.0E-05 | 4.9E-10 |
| Te127m | 4.0E-14 | 9.0E-06 | 4.5E-09 |
| Te127 | 4.0E-14 | 1.0E-04 | 4.0E-10 |
| Te129m | 9.6E-14 | 7.0E-06 | 1.4E-08 |
| Te129 | 6.1E-14 | 4.0E-04 | 1.5E-10 |
| Te131m | 3.1E-15 | 8.0E-06 | 3.9E-10 |
| Te131 | 7.1E-16 | 8.0E-05 | 8.9E-12 |
| Te132 | 3.3E-13 | 9.0E-06 | 3.7E-08 |
| Ba140 | 1.9E-14 | 8.0E-06 | 2.3E-09 |
| La140 | 2.1E-14 | 9.0E-06 | 2.3E-09 |

### 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 |
|---|---|---|---|
| La141 | 6.4E-30 | 5.0E-05 | 1.3E-25 |
| Ce141 | 4.3E-15 | 3.0E-05 | 1.4E-10 |
| Ce143 | 4.9E-17 | 2.0E-05 | 2.4E-12 |
| Ce144 | 4.7E-15 | 3.0E-06 | 1.6E-09 |
| Pr143 | 2.9E-15 | 2.0E-05 | 1.4E-10 |
| Pr144 | 4.6E-15 | 6.0E-04 | 7.7E-12 |
| Np239 | 5.3E-15 | 2.0E-05 | 2.6E-10 |
| Na24 | 9.2E-15 | 5.0E-05 | 1.8E-10 |
| Cr51 | 2.4E-11 | 5.0E-04 | 4.8E-08 |
| Mn54 | 1.6E-11 | 3.0E-05 | 5.2E-07 |
| Fe55 | 1.2E-11 | 1.0E-04 | 1.2E-07 |
| Fe59 | 2.6E-12 | 1.0E-05 | 2.6E-07 |
| Co58 | 3.6E-10 | 2.0E-05 | 1.8E-05 |
| Co60 | 5.3E-12 | 3.0E-06 | 1.8E-06 |
| Ni63 | 2.7E-12 | 1.0E-04 | 2.7E-08 |
| Zn65 | 5.0E-12 | 5.0E-06 | 9.9E-07 |
| Zr95 | 3.5E-12 | 2.0E-05 | 1.7E-07 |
| Ag110m | 1.3E-11 | 6.0E-06 | 2.1E-06 |
| W187 | 3.2E-14 | 3.0E-05 | 1.1E-09 |
| H3 | 8.3E-04 | 1.0E-03 | 8.3E-01 |
| C14 | 2.5E-07 | 3.0E-05 | 8.3E-03 |
| **Total** | **8.3E-04** | | **8.4E-01** |

**Figure 11.2-1a: Liquid Radioactive Waste System Diagram**

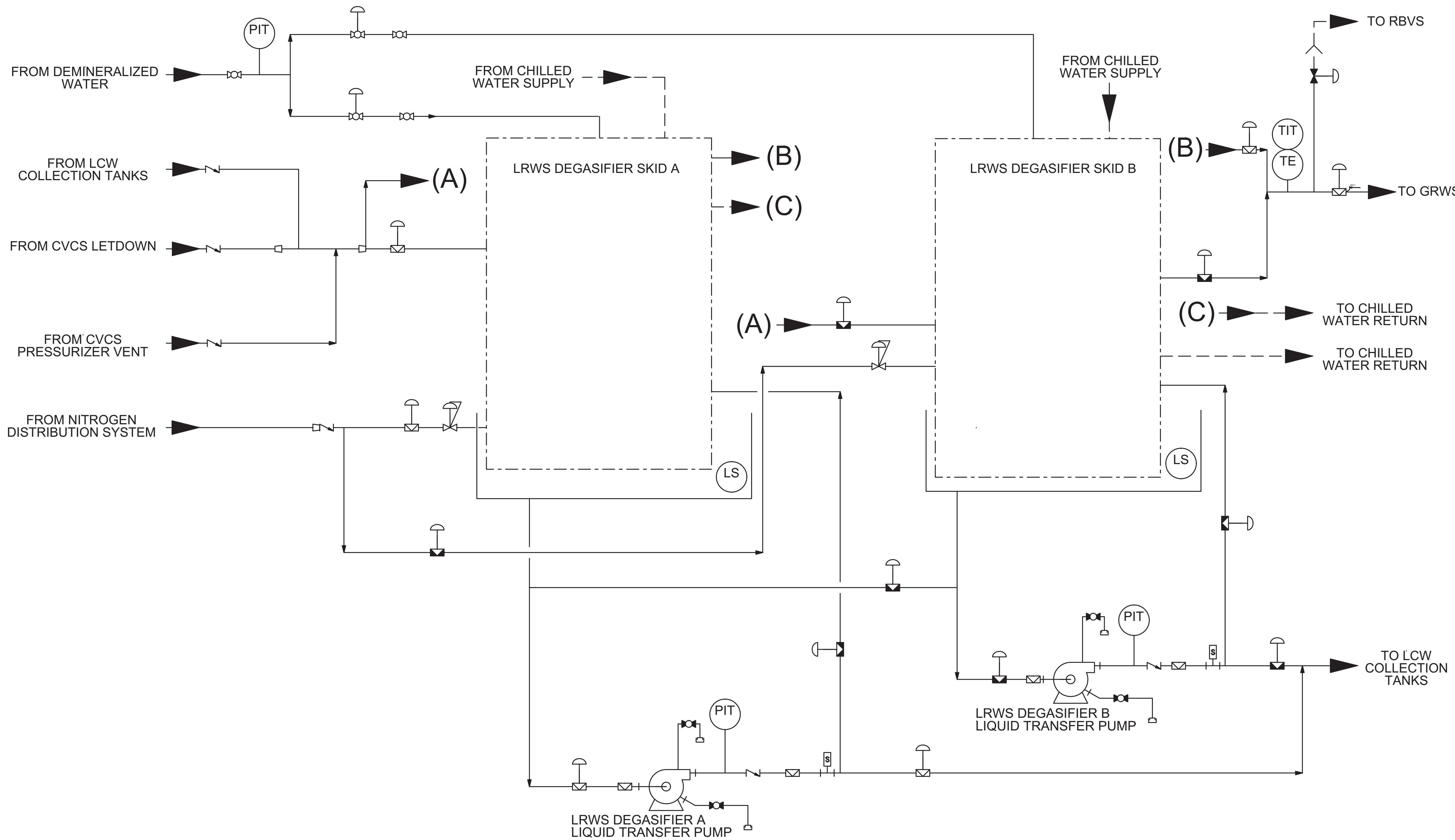


**Figure 11.2-1b: Liquid Radioactive Waste System Diagram**

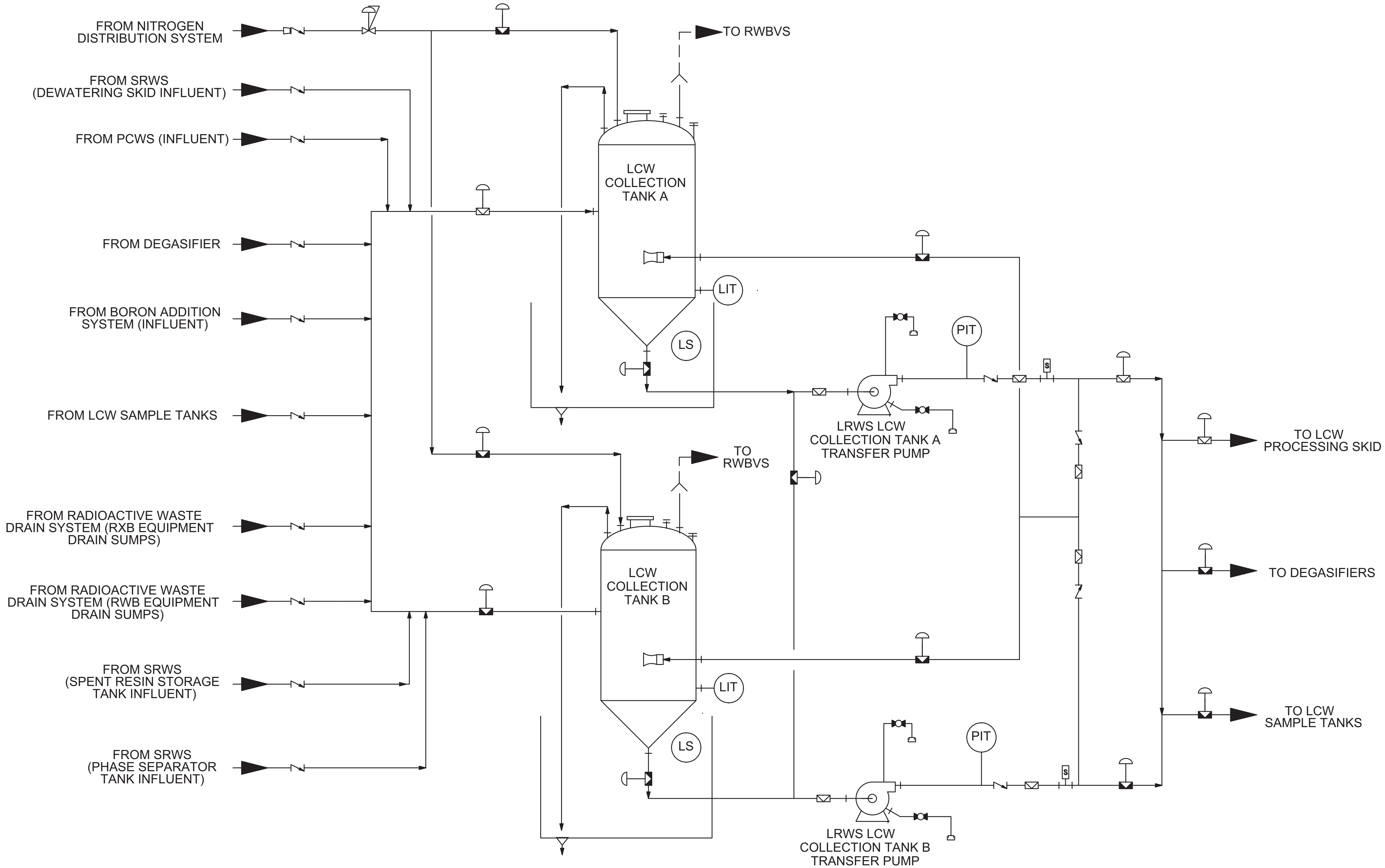


**Figure 11.2-1c: Liquid Radioactive Waste System Diagram**

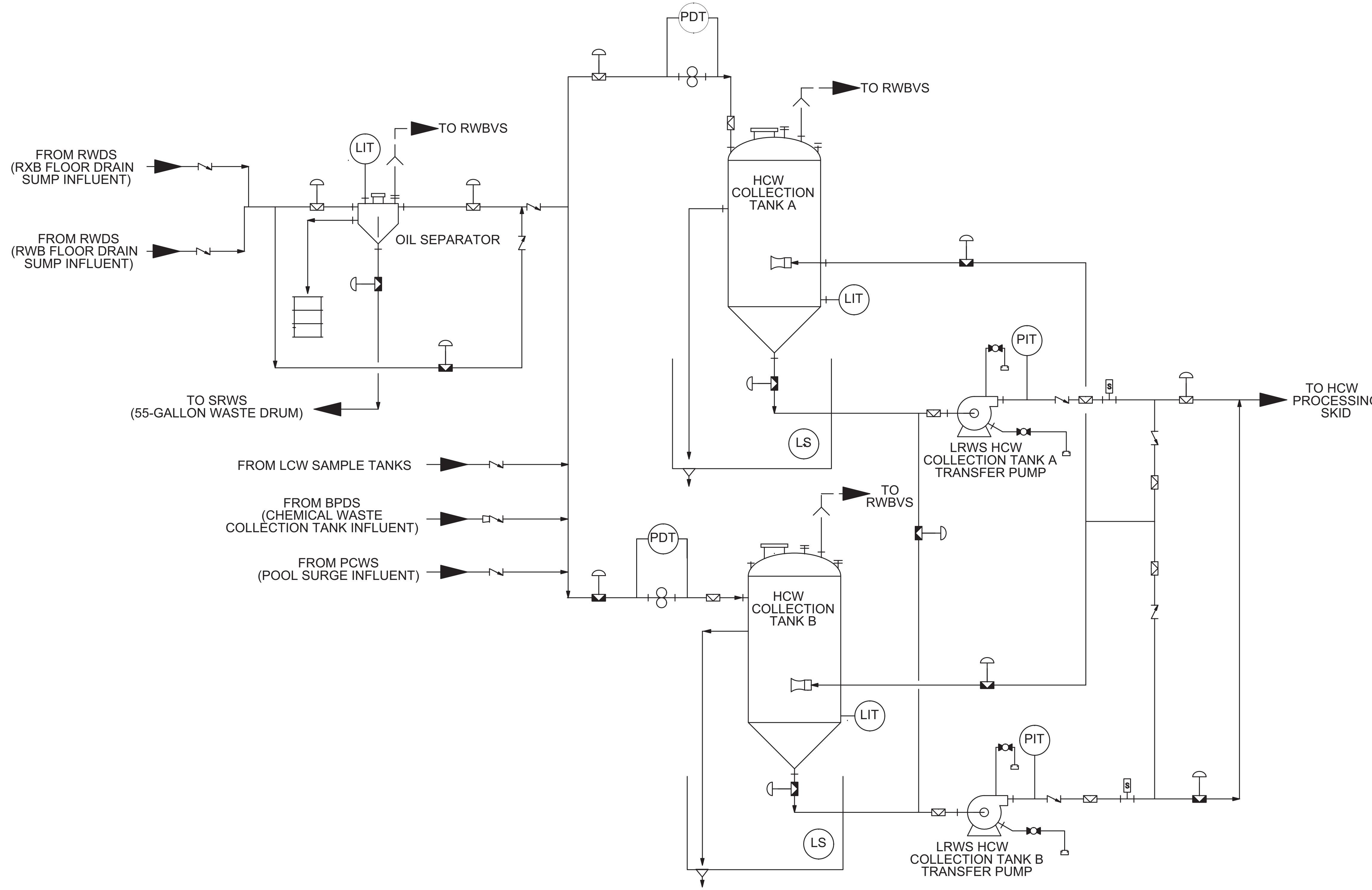


**Figure 11.2-1d: Liquid Radioactive Waste System Diagram**

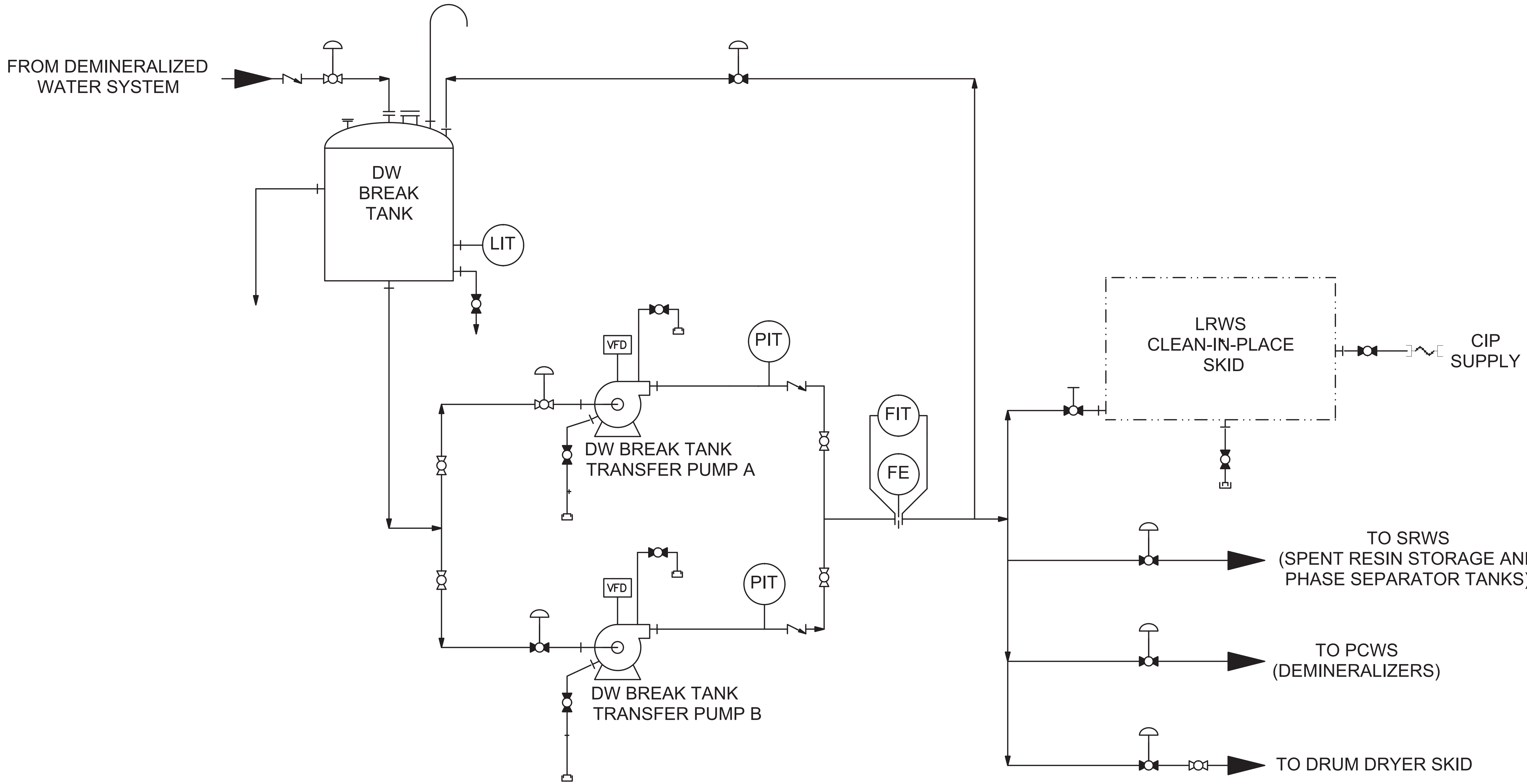


**Figure 11.2-1e: Liquid Radioactive Waste System Diagram**

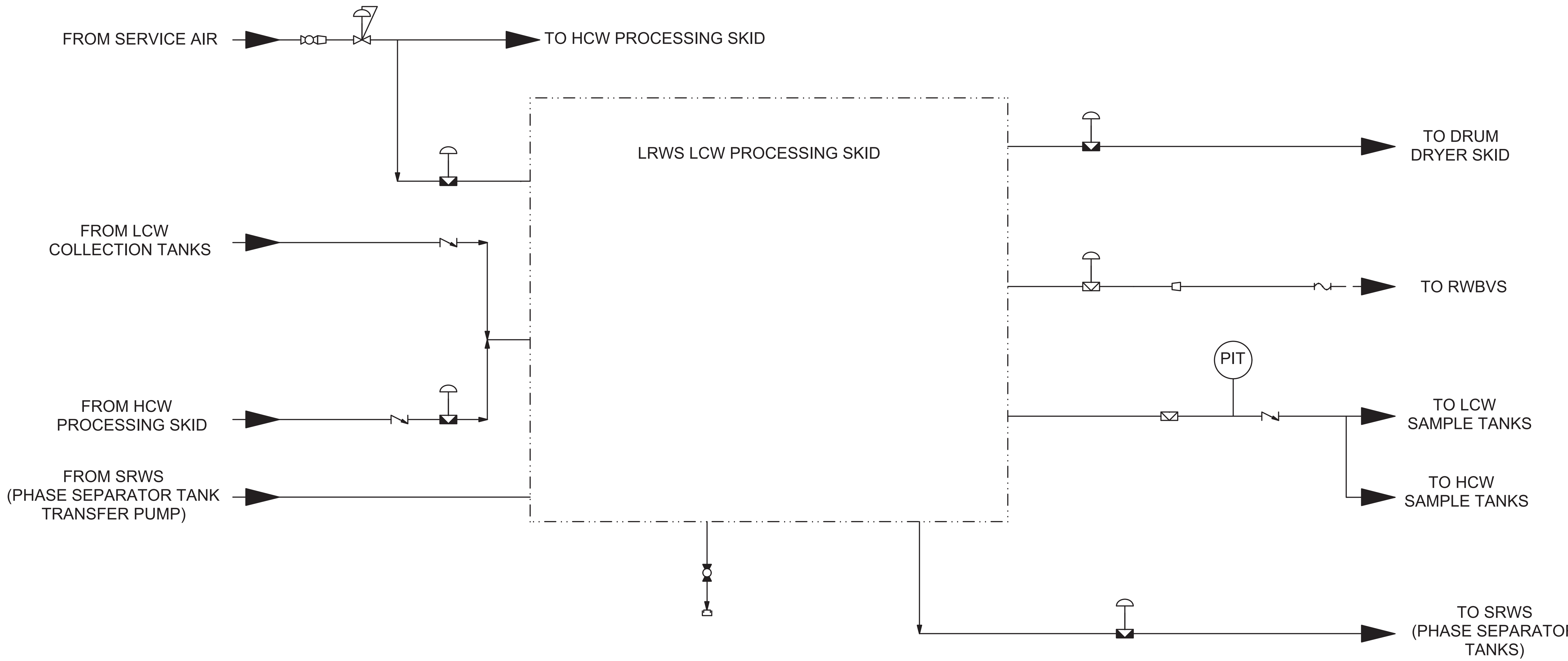


**Figure 11.2-1f: Liquid Radioactive Waste System Diagram**

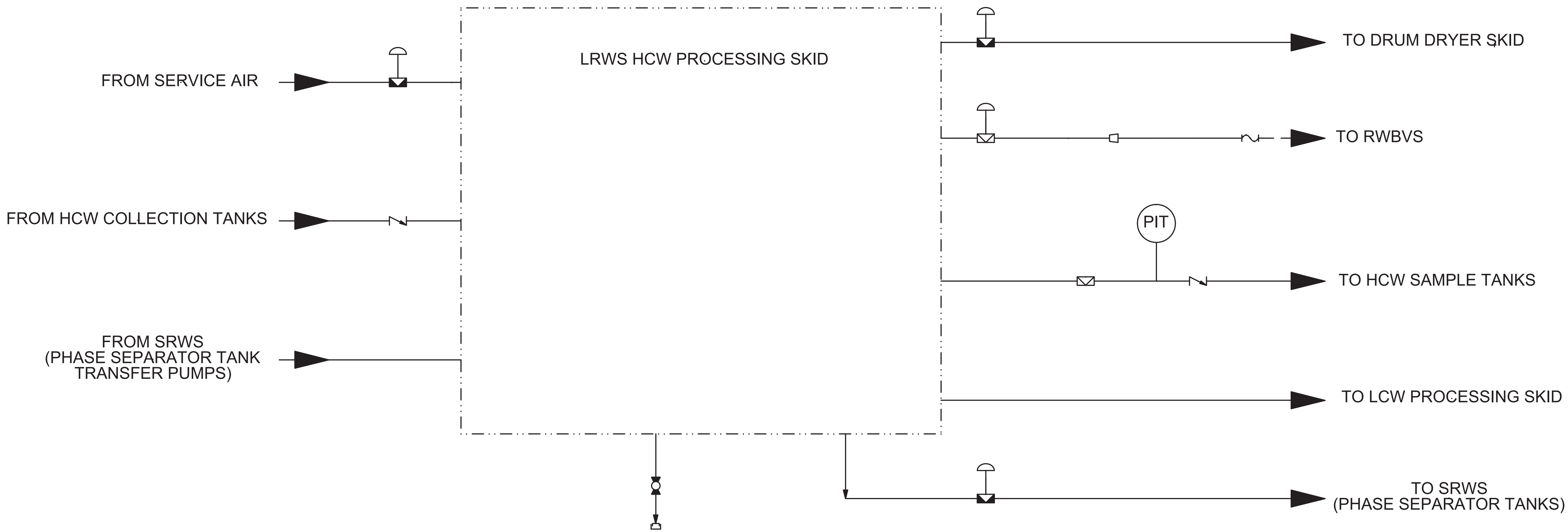


**Figure 11.2-1g: Liquid Radioactive Waste System Diagram**

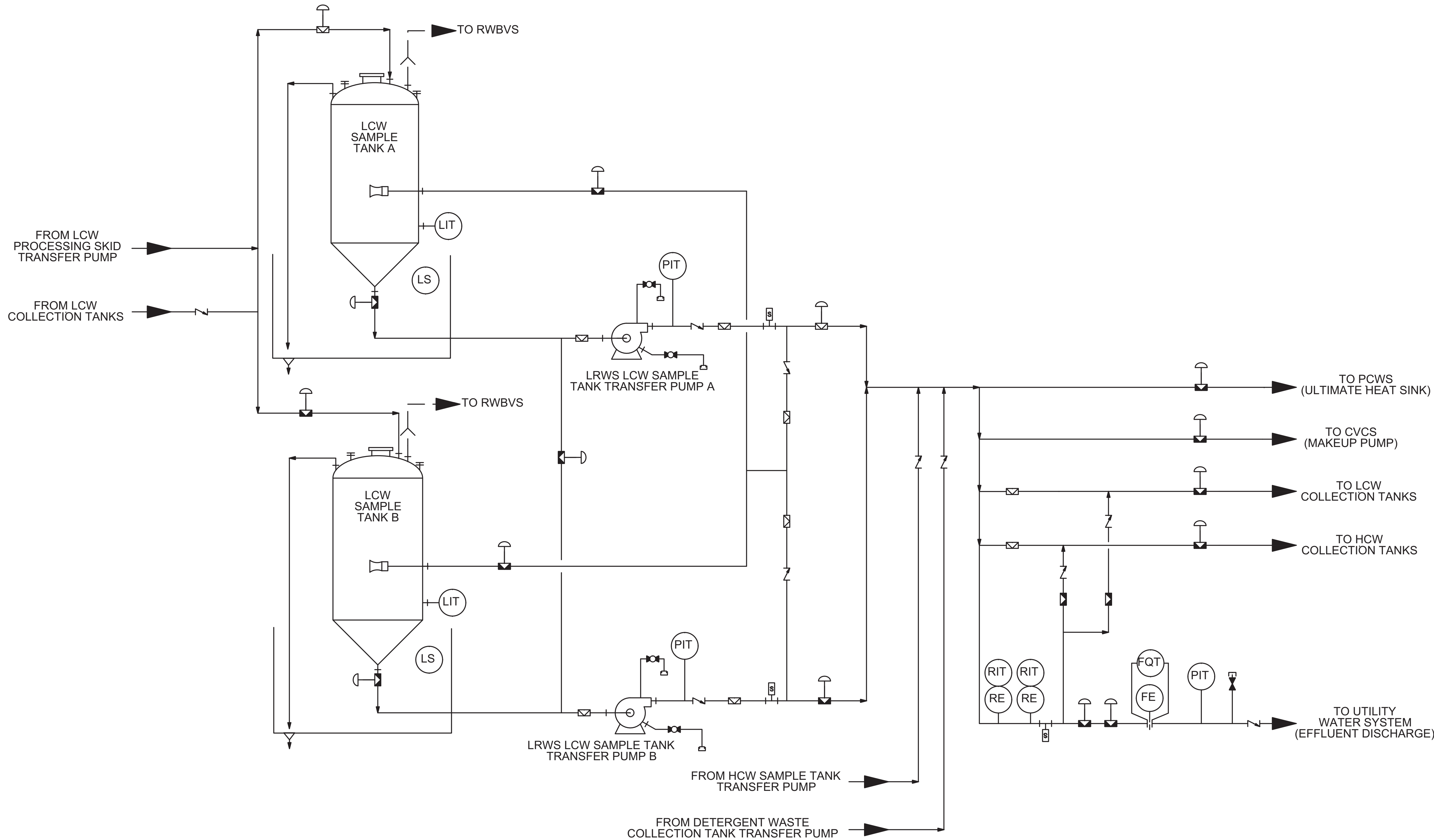


**Figure 11.2-1h: Liquid Radioactive Waste System Diagram**

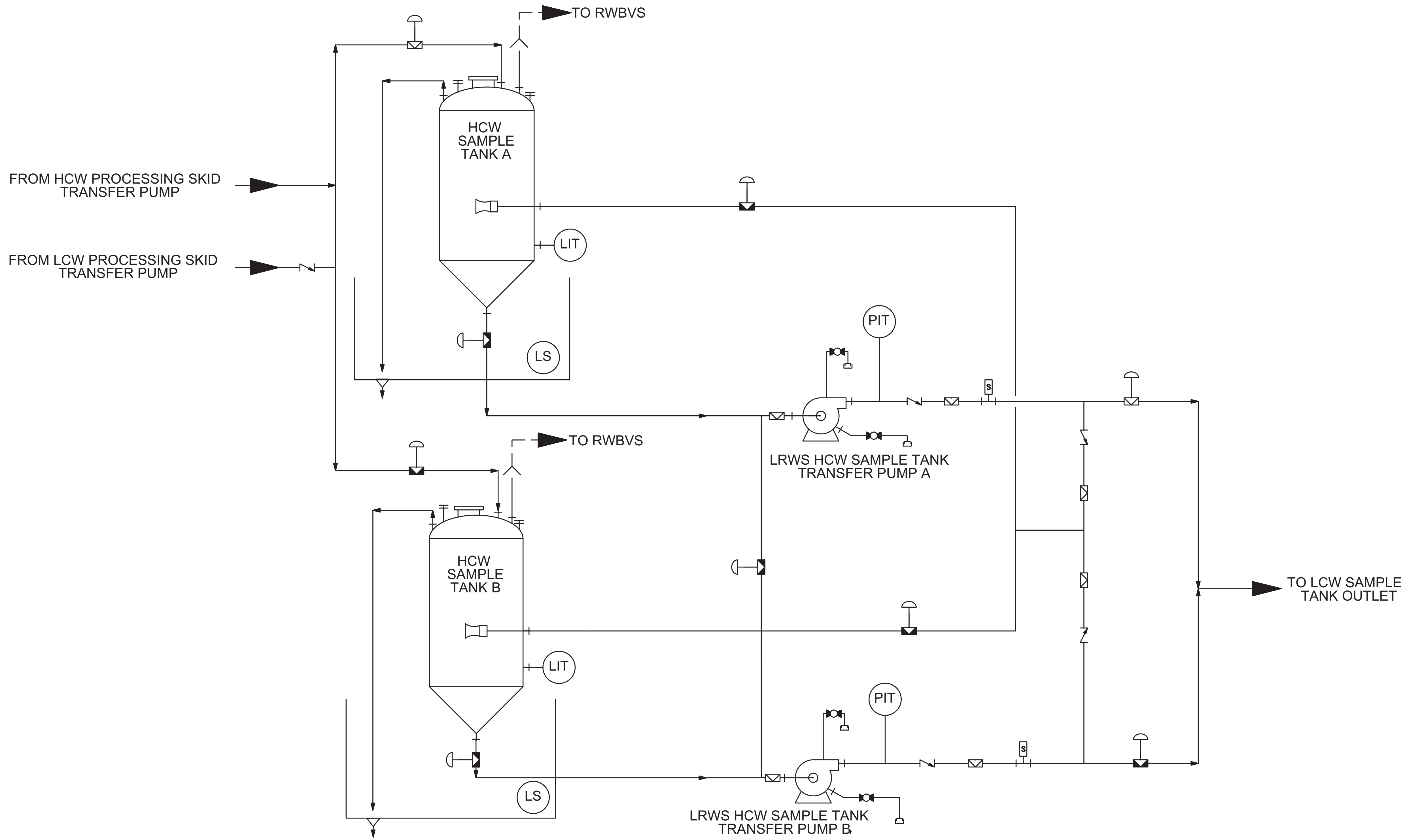


**Figure 11.2-1i: Liquid Radioactive Waste System Diagram**

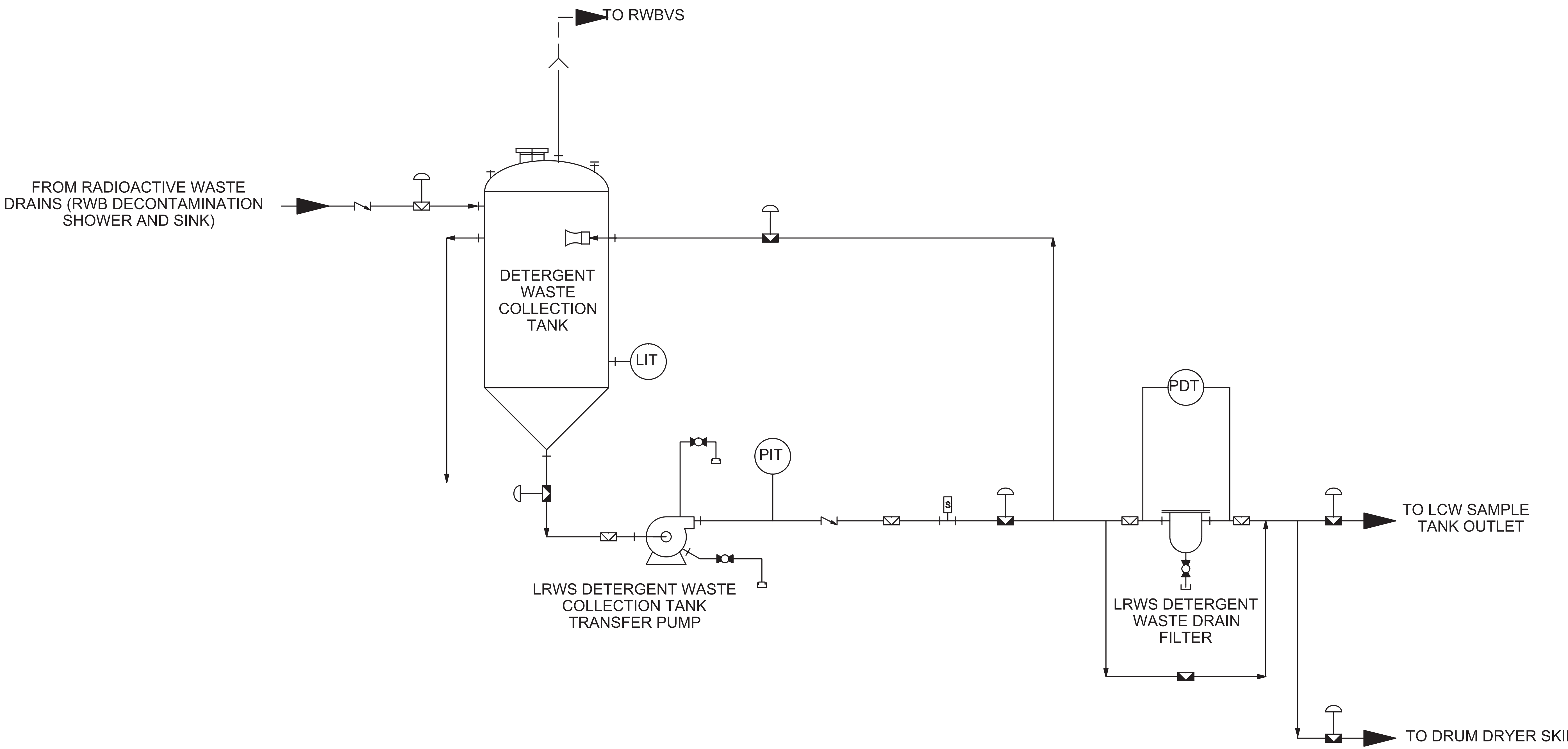
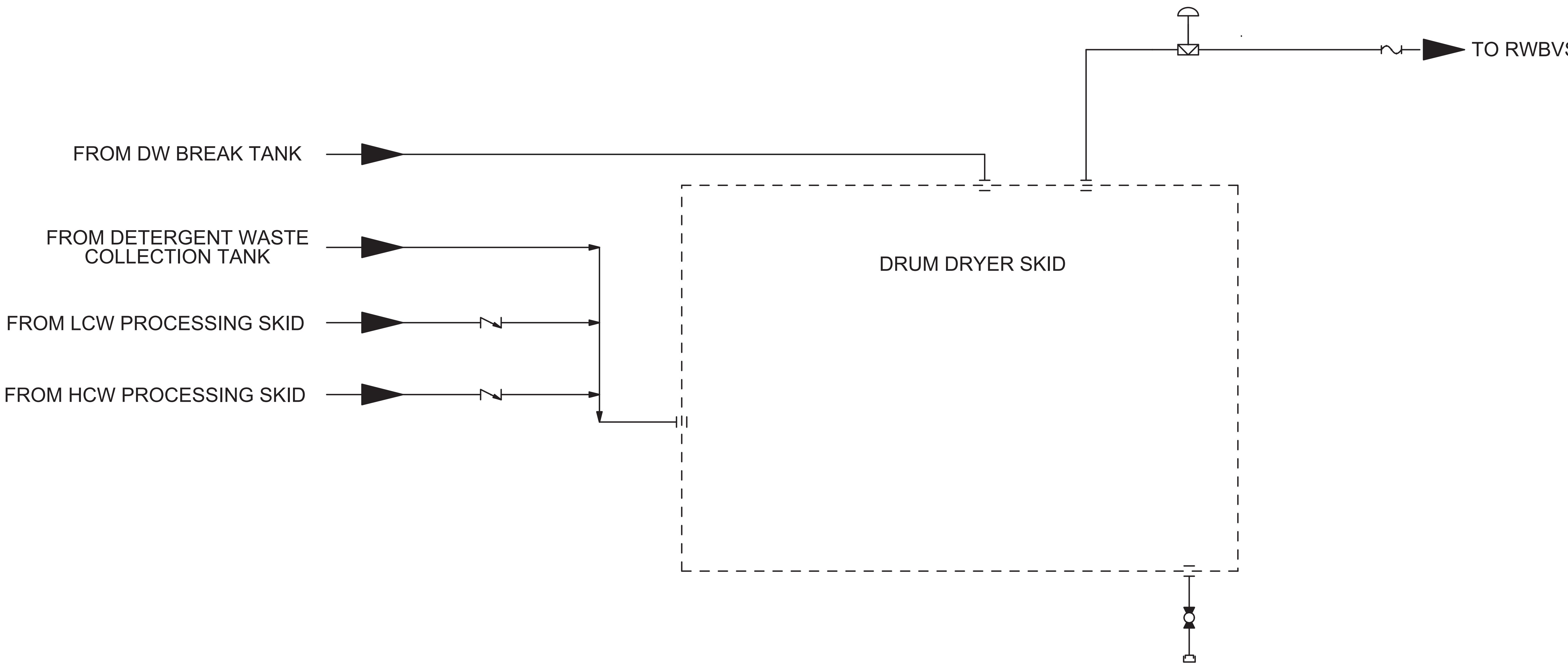


**Figure 11.2-1j: Liquid Radioactive Waste System Diagram**



## 11.3 Gaseous Waste Management System

The gaseous radioactive waste system (GRWS) design processes 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 heating ventilation and air conditioning (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-123242 (Reference 11.3-1).

Exhaust flow from the RWBVS and RBVS are combined and monitored by the RBVS exhaust stack radiation effluent monitor before 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 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.

### 11.3.1 Design Bases

The GRWS serves no safety function and is not risk-significant. The GRWS does not mitigate 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.

### 11.3.2 System Description

The GRWS is 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 (Section 11.5).

Nitrogen from the nitrogen distribution system dilutes the waste gas input from the liquid radioactive waste degasifier (and potentially CES) to maintain a hydrogen concentration of less than 4 percent. Because the waste gas input flow is not constant, the nitrogen supply maintains 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 automatically isolates and a nitrogen purge flushes the GRWS. Operators manually initiate termination of nitrogen flushing and restart of normal operations.

The waste gas passes through a charcoal guard bed located in an ambient temperature-controlled shielded cubicle. Because the guard bed is at ambient room temperature, the guard bed 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 contains a safety relief valve, differential pressure instrumentation, and a means to dry or replace charcoal. Operators manually initiate charcoal drying using 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 recycles 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.

The processed waste gas goes to the RWBVS, which interfaces with the RBVS that monitors the effluent path to the environment. The GRWS outlet also has an offline radiation monitor with the capability to take samples before being sent to the ventilation systems.

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

### 11.3.2.1 Component Description

This section describes the key GRWS equipment. Table 11.3-2 summarizes specific component design parameters. 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 cools the incoming waste gas stream from the LRWS and CES. Chilled water (shell side) cools the waste gas stream (tube side) to condense water vapor from the gas stream to protect the charcoal beds from moisture.

#### 11.3.2.1.2 Moisture Separator

The stainless steel moisture separator collects condensed water from the waste gas cooler. Level instrumentation controls the outlet drain valve. The condensate goes 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 American Society of Mechanical Engineers (ASME) Section VIII stainless steel vessel located in an ambient temperature-controlled cubicle that 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 with a charcoal drying heater or replace the charcoal, if needed.

#### 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 has a pressure relief valve, differential pressure instrumentation, and a fire detector that upon sensing a fire 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

If needed, 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. Operators can also send heated nitrogen 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. Two redundant oxygen analyzers and two redundant hydrogen analyzers are 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 exceed 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 automatically isolates and a nitrogen purge valve automatically opens to purge the GRWS with nitrogen. The hydrogen monitor ensures detection of a maximum concentration of 4 percent with notification of a high-high alarm. The high alarm at approximately one-half of the maximum oxygen concentration includes a local, WMCR and MCR notification.

The design of the gas analyzer instruments is 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

Table 11.3-3 provides a summary of a malfunction analysis of the GRWS.

#### 11.3.2.3 Design Safety Evaluation

The GRWS complies with the following GDC 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

There are design features that comply 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. Section 12.3.6 provides additional details.

The gaseous radioactive waste structures, systems, and components design complies with the codes and standards provided in RG 1.143, Table 1 through 4. The applicable design criteria from RG1.143, 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. Table 11.3-2 provides the design parameters of major components, including safety classification and operating conditions.

#### 11.3.2.4 Site-Specific Cost-Benefit Analysis

COL Item 11.3-1: An applicant that references the NuScale Power Plant US460 standard design will perform a site-specific cost-benefit analysis.

#### 11.3.2.5 Seismic Design

The gaseous radioactive waste equipment and piping classification complies with RG 1.143. Section 3.7 describes the RWB seismic design.

### 11.3.3 Radioactive Effluent Releases

Technical Report TR-123242 (Reference 11.3-1) describes the gaseous radioactive effluent release methodology, inputs, and results.

Table 11.3-5 tabulates the results of the radioactive effluent calculation 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 GASPAR II Code is used to calculate the maximum individual doses at the exclusion area boundary. Table 11.3-6 tabulates the input parameters. Table 11.3-7 tabulates the resultant doses and demonstrates compliance with the limits of 10 CFR 50 Appendix I.

COL Item 11.3-2: An applicant that references the NuScale Power Plant US460 standard design 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 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. The analysis used in determining the radionuclide content of the effluents assumes that 1 percent of the operating fission product inventory in the core is released to the primary coolant. Table 11.3-8 tabulates the release source term. 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 Table 2.0-1. Table 11.3-8 presents the resultant offsite doses.

COL Item 11.3-3: An applicant that references the NuScale Power Plant US460 standard design will perform an analysis in accordance with Branch Technical Position 11-5 using the site-specific parameters.

### 11.3.4 Ventilation Systems

The design of the ventilation systems for normal operation is in accordance with RG 1.140, and is described in Section 9.4.

### 11.3.5 Instrumentation and Controls

The instruments that provide automated functions in the GRWS include the following:

#### 11.3.5.1 Waste Gas Cooler Moisture Separator Level

The waste gas cooler moisture separator level instrument monitors the water level in the drain tank and opens the tank's drain valve to route the water to the RWDS.

#### 11.3.5.2 Hydrogen and Oxygen Gas Analyzers

Section 11.3.2.1.6 describes the hydrogen and oxygen gas analyzers.

#### 11.3.5.3 Fire Detectors

Each of the charcoal beds has fire detectors to indicate the presence of a fire. If a fire is detected in a guard or decay bed, the GRWS waste gas inlet valve automatically closes and the nitrogen supply valve automatically opens to the associated charcoal bed.

#### 11.3.5.4 Waste Gas Flow Instrument

The waste gas flow instrument is downstream of the moisture separator and downstream of the decay beds in the outlet line. Nitrogen flow maintains maintain a minimum flow through the charcoal beds.

#### 11.3.5.5 Moisture Instrument

The waste gas stream contains a moisture level instrument at the outlet of the guard bed. If high moisture is detected, the waste gas inlet valve to the GRWS closes to stop the system flow.

#### 11.3.5.6 Charcoal Bed Process Radiation Monitors

The outlet of each of the two charcoal decay beds has process radiation monitors. If high radiation is detected, the charcoal bed outlet valve closes.

#### 11.3.5.7 Gaseous Radioactive Waste System Outlet Process Radiation Monitor

The outlet of the GRWS also has a radiation monitor. If high radiation is detected, the GRWS outlet valve closes to stop system flow to the RWBVS.

#### 11.3.5.8 Cubicle Area Airborne Radiation Detectors

Each of the charcoal bed cubicles has area airborne radiation detectors. If high radiation is detected, the waste gas inlet valve to the GRWS closes and the nitrogen purge valve opens.

### 11.3.6 Reference

11.3-1 NuScale Power, LLC, "Effluent Release (GALE Replacement) Methodology and Results," TR-123242, Revision 0.

### Table 11.3-1: Gaseous Radioactive Waste System Design Parameters

| Parameter | Nominal Value |
|---|---|
| Xenon delay | 69 days (normal) |
| Krypton delay | 2.9 days (normal) |
| Dynamic adsorption coefficient (Kd) for xenon | 1400 cm3/g |
| Dynamic adsorption coefficient (Kd) for krypton | 60 cm3/g |
| Maximum gas waste stream temperature | 200 °F |
| Activated carbon operating temperature | 50-105 °F |
| Gas flow rate | 1.03 scfm (normal) |
| Charcoal particle size | 0.132 inch |

## Table 11.3-2: Major Equipment Design Parameters

| Equipment / Parameter | Description / Value |
|---|---|
| **Vapor Condenser Package Assembly** | |
| Quantity | 2 |
| Design pressure | 150 psig |
| Design temperature | 250 °F |
| Max gas design flow rate | 3.5 scfm |
| Max gas inlet temperature | 200 °F |
| Max chilled water inlet temperature | 40.5 °F |
| Material | Stainless Steel |
| RG 1.143 safety classification | RW-IIc |
| Table for Assumed Radioactive Content | Table 11.3-9 |
| **Charcoal Drying Heater** | |
| Quantity | 1 |
| Type | Electric |
| Flow | 2.28 scfm |
| Minimum Temperature Inlet | -10 °F |
| Temperature outlet | 140 °F |
| **Charcoal Guard Bed** | |
| Quantity | 1 |
| Type | cylindrical pressure vessel |
| Nominal volume | 10 $ft^3$ |
| Design pressure | 125 psig |
| Design temperature | 250 °F |
| Design flow rate | 3.5 scfm |
| Material | Stainless Steel |
| RG 1.143 safety classification | RW-IIa |
| Table for Assumed Radioactive Content | Table 12.2-15 |
| **Charcoal Decay Bed Vessel** | |
| Quantity | 2 |
| Type | cylindrical pressure vessel |
| Nominal volume | 147.5 $ft^3$/sec |
| Design pressure | 125 psig |
| Design temperature | 250 °F |
| Material | Stainless Steel |
| RG 1.143 safety classification | RW-IIa |
| Table for Assumed Radioactive Content | Table 12.2-15 |

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

| Equipment Item | Malfunction | Results (Consequences) | Mitigating or Alternate Action |
|---|---|---|---|
| Vapor Condenser Package Assembly | Skid Failure | Failure of the gas cooler causes ineffective removal of moisture in the gas. | If one of the condenser package assemblies fail, the influent gas stream can be diverted to the other vapor condenser package. This allows processing to continue. |
| 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 GRW operation if the heater fails. The downstream charcoal decay bed efficiency may be lowered. | The charcoal decay bed skids 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 be in the system. If the guard bed fails, the fission gas removal efficiency may be lowered. | Operation may continue by sending the gas to be treated in one of the two charcoal decay beds, which is located downstream of the bed. |
| | | | Pressure differential transmitter monitors different pressures across the guard bed. |
| | | | Detect moisture content of the gaseous stream through the moisture monitor located downstream of the guard bed. |
| | | | The charcoal decay bed skids may be aligned in series to improve the decontamination factor if the guard bed is saturated with moisture. |
| Charcoal Decay Bed Skids | Decay Bed Failure | There is one set of redundant charcoal decay bed skids. Failure of one decay bed skid decreases removal efficiency of radioactive noble gases. | If one decay bed skid fails, the gas can be switched to the redundant decay bed skid for continued operation. |
| | | | Radiation monitors downstream of the charcoal decay beds alarm when high radiation level is detected in the effluent gas. A high radiation alarm also triggers automatic closure of the isolation valves to the RWBV. Operators should review indications and determine whether to isolate influent streams to the GRW as well. |
| Pressure Boundary | Gas Leaks | Waste gas is released to the RWB. | Very small gas leaks in the GRW Charcoal Beds room can be detected by the area airborne radiation monitors. |
| | | | The system can be purged with nitrogen before repair of replacement of the leaking component. |
| Oxygen Monitor | Fail to monitor | Monitoring capability is lost in detecting oxygen concentration. | The redundant oxygen analyzer monitors the oxygen concentration downstream of the vapor condensers. A single oxygen analyzer is placed just before the discharge to RWBV. |
| | | | The oxygen analyzers are set to alarm at 1% and 2%. |
| | | | The high-high oxygen alarm with a set point of 2% isolates the input streams from CE and LRW. Operators allow the continuous nitrogen flow to purge the system. |

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

| Equipment Item | Malfunction | Results (Consequences) | Mitigating or Alternate Action |
|---|---|---|---|
| Hydrogen Monitor | Fail to monitor | Monitoring capability is lost in detecting hydrogen concentration. | The redundant hydrogen analyzer monitors the hydrogen concentration downstream of the vapor condensers. |
| | | | Hydrogen analyzers are set to alarm at 2% and 4%. |
| Radiation Monitor | Fail to monitor/Loss of power | There are two types of radiation monitors: one is an area airborne monitor and the other is an in-line process radiation monitor. In both cases, monitoring capability is lost in detecting leakage to the GRW rooms housing the equipment and system and any waste gas released through the doors to the RWB. | In-line process radiation is monitored at each of the decay skid outlets and at the system discharge providing redundancy for the process. Therefore, failure of one does not impact operation of the other unless there is a loss of power. |
| | | | Upon loss of power to the radiation monitors the inlet and outlet valves are closed to isolate the GRW. |

## Table 11.3-4: Gaseous Effluent Release Calculation Inputs

| NuScale Effluent Source Term Model Assumption | Value (1 NPM) | Value (6 NPMs) |
|---|---|---|
| Degasifier partition fractions: | | |
| - Noble gases | 1 | 1 |
| - Halogens | 0.5 | 0.5 |
| Reactor pool evaporation rate | - | 1300 lb/hour |
| Pool evaporation partition fractions: | | |
| - Halogens (except iodine) | 0.01 | 0.01 |
| - Iodine | 0.0005 | 0.0005 |
| - Cs, Rb, particulates | 0.005 | 0.005 |
| - Gases and tritium | 1 | 1 |
| Steam generator partition coefficient | 1 | 1 |
| High-efficiency particulate air filter particulate efficiency | 0 | 0 |
| Primary coolant system leakrate | 11.8 lb/day | 70.6 lb/day |
| Primary coolant leak flashing fraction | 0.4 | 0.4 |
| Primary coolant leak partition fractions: | | |
| - Halogens | 0.01 | 0.01 |
| - Cs, Rb, particulates | 0.005 | 0.005 |
| - Gases and tritium | 1 | 1 |
| Secondary coolant system steam leakrate | 125 lb/day | 750 lb/day |
| Condenser air removal normalized iodine release rate | 125 Ci/yr/μCi/gm | 750 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 | 6.5E-07 | 1.5E-07 | 8.8E-05 | 9.1E-03 | 9.1E-03 | 8.2E-07 | 4.2E-03 | 4.2E-03 | 4.2E-15 | 5.0E-05 | 8.5E-11 |
| Kr85m | 6.0E-05 | - | 3.7E-04 | 3.8E-02 | 3.8E-02 | 3.4E-06 | 1.8E-02 | 1.8E-02 | 1.8E-14 | 1.0E-07 | 1.8E-07 |
| Kr85 | 1.4E+02 | - | 1.8E-02 | 1.9E+00 | 1.4E+02 | 1.7E-04 | 8.9E-01 | 8.9E-01 | 4.6E-11 | 7.0E-07 | 6.6E-05 |
| Kr87 | 5.2E-17 | - | 2.0E-04 | 2.1E-02 | 2.1E-02 | 1.9E-06 | 9.7E-03 | 9.7E-03 | 9.7E-15 | 2.0E-08 | 4.8E-07 |
| Kr88 | 1.9E-07 | - | 5.9E-04 | 6.0E-02 | 6.1E-02 | 5.4E-06 | 2.8E-02 | 2.8E-02 | 2.8E-14 | 9.0E-09 | 3.1E-06 |
| Kr89 | - | - | 1.3E-05 | 1.4E-03 | 1.4E-03 | 1.2E-07 | 6.4E-04 | 6.4E-04 | 6.4E-16 | - | - |
| Xe131m | 2.0E-01 | 3.6E-01 | 1.4E-03 | 1.5E-01 | 7.0E-01 | 1.3E-05 | 6.9E-02 | 6.9E-02 | 2.4E-13 | 2.0E-06 | 1.2E-07 |
| Xe133m | 3.2E-07 | 4.2E-01 | 1.3E-03 | 1.3E-01 | 5.6E-01 | 1.2E-05 | 6.3E-02 | 6.3E-02 | 2.0E-13 | 6.0E-07 | 3.3E-07 |
| Xe133 | 8.4E-02 | 6.0E+00 | 9.5E-02 | 9.7E+00 | 1.6E+01 | 8.8E-04 | 4.6E+00 | 4.6E+00 | 6.5E-12 | 5.0E-07 | 1.3E-05 |
| Xe135m | 7.3E-05 | 5.4E-02 | 1.3E-04 | 1.3E-02 | 6.7E-02 | 1.2E-06 | 6.1E-03 | 6.1E-03 | 2.3E-14 | 4.0E-08 | 5.8E-07 |
| Xe135 | 3.2E-05 | 3.0E-02 | 2.6E-03 | 2.7E-01 | 3.0E-01 | 2.4E-05 | 1.3E-01 | 1.3E-01 | 1.4E-13 | 7.0E-08 | 1.9E-06 |
| Xe137 | - | - | 4.4E-05 | 4.5E-03 | 4.6E-03 | 4.1E-07 | 2.1E-03 | 2.1E-03 | 2.1E-15 | - | - |
| Xe138 | - | - | 1.5E-04 | 1.5E-02 | 1.6E-02 | 1.4E-06 | 7.2E-03 | 7.2E-03 | 7.2E-15 | 2.0E-08 | 3.6E-07 |
| Br82 | 9.5E-09 | 6.4E-09 | - | 1.0E-06 | 1.0E-06 | 2.3E-08 | 5.7E-09 | 2.8E-08 | 3.3E-19 | 5.0E-09 | 6.6E-11 |
| Br83 | 5.4E-08 | 1.1E-14 | - | 5.7E-06 | 5.8E-06 | 1.3E-07 | 3.2E-08 | 1.6E-07 | 1.9E-18 | 9.0E-08 | 2.1E-11 |
| Br84 | 2.5E-08 | - | - | 2.7E-06 | 2.7E-06 | 5.5E-08 | 1.4E-08 | 6.9E-08 | 8.7E-19 | 8.0E-08 | 1.1E-11 |
| Br85 | 3.1E-09 | - | - | 3.2E-07 | 3.2E-07 | 3.5E-09 | 8.7E-10 | 4.3E-09 | 1.0E-19 | - | - |
| I129 | 1.6E-13 | 3.1E-13 | - | 1.6E-11 | 1.7E-11 | 3.7E-13 | 9.4E-14 | 4.7E-13 | 5.5E-24 | 4.0E-11 | 1.4E-13 |
| I130 | 7.7E-08 | 6.8E-09 | - | 8.2E-06 | 8.2E-06 | 1.8E-07 | 4.7E-08 | 2.3E-07 | 2.7E-18 | 3.0E-09 | 8.9E-10 |
| I131 | 2.0E-06 | 3.2E-04 | - | 2.1E-04 | 5.3E-04 | 4.7E-06 | 1.2E-06 | 5.9E-06 | 1.7E-16 | 2.0E-10 | 8.5E-07 |
| I132 | 9.1E-07 | 7.9E-07 | - | 9.6E-05 | 9.8E-05 | 2.1E-06 | 5.4E-07 | 2.7E-06 | 3.2E-17 | 2.0E-08 | 1.6E-09 |
| I133 | 3.0E-06 | 2.8E-05 | - | 3.2E-04 | 3.5E-04 | 7.2E-06 | 1.8E-06 | 9.0E-06 | 1.1E-16 | 1.0E-09 | 1.1E-07 |
| I134 | 5.4E-07 | 8.5E-26 | - | 5.7E-05 | 5.7E-05 | 1.2E-06 | 3.1E-07 | 1.5E-06 | 1.9E-17 | 6.0E-08 | 3.1E-10 |
| I135 | 1.9E-06 | 1.0E-08 | - | 2.0E-04 | 2.0E-04 | 4.5E-06 | 1.1E-06 | 5.6E-06 | 6.5E-17 | 6.0E-09 | 1.1E-08 |
| Rb86m | - | - | - | 1.2E-10 | 1.2E-10 | 1.3E-12 | - | 1.3E-12 | 3.9E-23 | - | - |
| Rb86 | - | 8.3E-07 | - | 7.0E-07 | 1.5E-06 | 3.5E-08 | - | 3.5E-08 | 5.0E-19 | 1.0E-09 | 5.0E-10 |
| Rb88 | - | - | - | 1.2E-04 | 1.2E-04 | 5.0E-06 | - | 5.0E-06 | 4.0E-17 | 9.0E-08 | 4.4E-10 |
| Rb89 | - | - | - | 5.5E-06 | 5.5E-06 | 2.2E-07 | - | 2.2E-07 | 1.8E-18 | 2.0E-07 | 9.1E-12 |

## 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 | - | 1.4E-08 | - | 1.4E-08 | 2.8E-08 | 7.0E-10 | - | 7.0E-10 | 9.1E-21 | 6.0E-09 | 1.5E-12 |
| Cs134 | - | 1.3E-04 | - | 1.0E-04 | 2.3E-04 | 5.0E-06 | - | 5.0E-06 | 7.4E-17 | 2.0E-10 | 3.7E-07 |
| Cs135m | - | 1.1E-26 | - | 8.4E-08 | 8.4E-08 | 3.9E-09 | - | 3.9E-09 | 2.8E-20 | 3.0E-07 | 9.3E-14 |
| Cs136 | - | 2.5E-05 | - | 2.2E-05 | 4.7E-05 | 1.1E-06 | - | 1.1E-06 | 1.5E-17 | 9.0E-10 | 1.7E-08 |
| Cs137 | - | 6.6E-05 | - | 5.2E-05 | 1.2E-04 | 2.6E-06 | - | 2.6E-06 | 3.8E-17 | 2.0E-10 | 1.9E-07 |
| Cs138 | - | - | - | 4.5E-05 | 4.5E-05 | 2.0E-06 | - | 2.0E-06 | 1.5E-17 | 8.0E-08 | 1.9E-10 |
| P32 | - | 6.9E-13 | - | 2.0E-12 | 2.7E-12 | 8.9E-14 | - | 8.9E-14 | 8.7E-25 | 5.0E-10 | 1.7E-15 |
| Co57 | - | 5.8E-15 | - | 1.5E-14 | 2.1E-14 | 6.8E-16 | - | 6.8E-16 | 6.8E-27 | 9.0E-10 | 7.5E-18 |
| Sr89 | - | 3.4E-08 | - | 8.9E-08 | 1.2E-07 | 4.1E-09 | - | 4.1E-09 | 4.0E-20 | 2.0E-10 | 2.0E-10 |
| Sr90 | - | 5.4E-09 | - | 1.4E-08 | 1.9E-08 | 6.3E-10 | - | 6.3E-10 | 6.3E-21 | 6.0E-12 | 1.1E-09 |
| Sr91 | - | 3.2E-10 | - | 4.6E-08 | 4.7E-08 | 2.1E-09 | - | 2.1E-09 | 1.5E-20 | 5.0E-09 | 3.1E-12 |
| Sr92 | - | 4.6E-15 | - | 2.5E-08 | 2.5E-08 | 1.1E-09 | - | 1.1E-09 | 8.3E-21 | 9.0E-09 | 9.2E-13 |
| Y90 | - | 3.2E-09 | - | 3.4E-09 | 6.6E-09 | 1.5E-10 | - | 1.5E-10 | 2.1E-21 | 9.0E-10 | 2.4E-12 |
| Y91m | - | 2.1E-10 | - | 2.5E-08 | 2.5E-08 | 1.1E-09 | - | 1.1E-09 | 8.3E-21 | 2.0E-07 | 4.1E-14 |
| Y91 | - | 5.0E-09 | - | 1.3E-08 | 1.8E-08 | 5.9E-10 | - | 5.9E-10 | 5.9E-21 | 2.0E-10 | 2.9E-11 |
| Y92 | - | 6.5E-13 | - | 2.1E-08 | 2.1E-08 | 9.5E-10 | - | 9.5E-10 | 7.0E-21 | 1.0E-08 | 7.0E-13 |
| Y93 | - | 8.6E-11 | - | 1.0E-08 | 1.0E-08 | 4.5E-10 | - | 4.5E-10 | 3.3E-21 | 3.0E-09 | 1.1E-12 |
| Zr97 | - | 5.7E-10 | - | 1.5E-08 | 1.5E-08 | 6.7E-10 | - | 6.7E-10 | 5.1E-21 | 2.0E-09 | 2.5E-12 |
| Nb95 | - | 3.9E-05 | - | 3.7E-08 | 3.9E-05 | 1.7E-09 | - | 1.7E-09 | 1.2E-17 | 2.0E-09 | 6.2E-09 |
| Mo99 | - | 5.8E-06 | - | 2.7E-05 | 3.2E-05 | 1.2E-06 | - | 1.2E-06 | 1.1E-17 | 2.0E-09 | 5.3E-09 |
| Mo101 | - | - | - | 1.0E-06 | 1.0E-06 | 3.8E-08 | - | 3.8E-08 | 3.3E-19 | 2.0E-07 | 1.7E-12 |
| Tc99m | - | 5.6E-06 | - | 2.5E-05 | 3.0E-05 | 1.1E-06 | - | 1.1E-06 | 9.9E-18 | 2.0E-07 | 4.9E-11 |
| Tc99 | - | 2.0E-10 | - | 5.0E-10 | 7.0E-10 | 2.3E-11 | - | 2.3E-11 | 2.3E-22 | 8.0E-09 | 2.8E-14 |
| Ru103 | - | 9.5E-09 | - | 2.5E-08 | 3.5E-08 | 1.2E-09 | - | 1.2E-09 | 1.1E-20 | 9.0E-10 | 1.3E-11 |
| Ru105 | - | 5.3E-13 | - | 8.4E-09 | 8.4E-09 | 3.8E-10 | - | 3.8E-10 | 2.8E-21 | 2.0E-08 | 1.4E-13 |
| Ru106 | - | 6.2E-09 | - | 1.6E-08 | 2.2E-08 | 7.2E-10 | - | 7.2E-10 | 7.2E-21 | 2.0E-11 | 3.6E-10 |
| Rh103m | - | 9.4E-09 | - | 2.5E-08 | 3.4E-08 | 1.1E-09 | - | 1.1E-09 | 1.1E-20 | 2.0E-06 | 5.6E-15 |
| Rh105 | - | 2.4E-09 | - | 1.7E-08 | 2.0E-08 | 7.8E-10 | - | 7.8E-10 | 6.4E-21 | 8.0E-09 | 8.1E-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 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Rh106 | - | 6.2E-09 | - | 1.6E-08 | 2.2E-08 | 9.7E-11 | - | 9.7E-11 | 7.0E-21 | - | - |
| Ag110 | - | 4.0E-05 | - | 1.1E-07 | 4.0E-05 | 5.7E-10 | - | 5.7E-10 | 1.3E-17 | - | - |
| Sb124 | - | 1.4E-11 | - | 3.8E-11 | 5.2E-11 | 1.7E-12 | - | 1.7E-12 | 1.7E-23 | 3.0E-10 | 5.7E-14 |
| Sb125 | - | 1.1E-10 | - | 2.8E-10 | 3.9E-10 | 1.3E-11 | - | 1.3E-11 | 1.3E-22 | 7.0E-10 | 1.8E-13 |
| Sb127 | - | 3.7E-10 | - | 1.4E-09 | 1.8E-09 | 6.5E-11 | - | 6.5E-11 | 5.9E-22 | 1.0E-09 | 5.9E-13 |
| Sb129 | - | 1.0E-13 | - | 1.8E-09 | 1.8E-09 | 7.9E-11 | - | 7.9E-11 | 5.8E-22 | 1.0E-08 | 5.8E-14 |
| Te125m | - | 1.6E-08 | - | 4.1E-08 | 5.6E-08 | 1.9E-09 | - | 1.9E-09 | 1.8E-20 | 1.0E-09 | 1.8E-11 |
| Te127m | - | 6.0E-08 | - | 1.6E-07 | 2.2E-07 | 7.1E-09 | - | 7.1E-09 | 7.0E-20 | 4.0E-10 | 1.8E-10 |
| Te127 | - | 6.2E-08 | - | 6.2E-07 | 6.8E-07 | 2.8E-08 | - | 2.8E-08 | 2.2E-19 | 2.0E-08 | 1.1E-11 |
| Te129m | - | 1.7E-07 | - | 4.5E-07 | 6.1E-07 | 2.0E-08 | - | 2.0E-08 | 2.0E-19 | 3.0E-10 | 6.7E-10 |
| Te129 | - | 1.0E-07 | - | 6.3E-07 | 7.4E-07 | 2.8E-08 | - | 2.8E-08 | 2.4E-19 | 9.0E-08 | 2.7E-12 |
| Te131m | - | 1.6E-07 | - | 1.5E-06 | 1.6E-06 | 6.6E-08 | - | 6.6E-08 | 5.3E-19 | 1.0E-09 | 5.3E-10 |
| Te131 | - | 3.5E-08 | - | 7.2E-07 | 7.6E-07 | 2.9E-08 | - | 2.9E-08 | 2.5E-19 | 2.0E-08 | 1.2E-11 |
| Te132 | - | 2.5E-06 | - | 1.1E-05 | 1.3E-05 | 4.8E-07 | - | 4.8E-07 | 4.3E-18 | 9.0E-10 | 4.8E-09 |
| Te133m | - | 2.4E-25 | - | 9.2E-07 | 9.2E-07 | 3.9E-08 | - | 3.9E-08 | 3.0E-19 | 2.0E-08 | 1.5E-11 |
| Te134 | - | - | - | 1.3E-06 | 1.3E-06 | 5.5E-08 | - | 5.5E-08 | 4.3E-19 | 7.0E-08 | 6.1E-12 |
| Ba137m | - | 6.2E-05 | - | 4.8E-05 | 1.1E-04 | 9.7E-07 | - | 9.7E-07 | 3.5E-17 | - | - |
| Ba139 | - | 6.9E-21 | - | 2.4E-08 | 2.4E-08 | 1.1E-09 | - | 1.1E-09 | 8.0E-21 | 4.0E-08 | 2.0E-13 |
| Ba140 | - | 4.5E-08 | - | 1.3E-07 | 1.8E-07 | 5.9E-09 | - | 5.9E-09 | 5.7E-20 | 2.0E-09 | 2.9E-11 |
| La140 | - | 3.5E-08 | - | 3.8E-08 | 7.3E-08 | 1.7E-09 | - | 1.7E-09 | 2.4E-20 | 2.0E-09 | 1.2E-11 |
| La141 | - | 1.5E-13 | - | 7.5E-09 | 7.5E-09 | 3.4E-10 | - | 3.4E-10 | 2.5E-21 | 1.0E-08 | 2.5E-13 |
| La142 | - | 1.2E-20 | - | 3.6E-09 | 3.6E-09 | 1.6E-10 | - | 1.6E-10 | 1.2E-21 | 3.0E-08 | 3.9E-14 |
| Ce141 | - | 7.5E-09 | - | 2.0E-08 | 2.8E-08 | 9.1E-10 | - | 9.1E-10 | 9.0E-21 | 8.0E-10 | 1.1E-11 |
| Ce143 | - | 1.8E-09 | - | 1.5E-08 | 1.7E-08 | 6.9E-10 | - | 6.9E-10 | 5.6E-21 | 2.0E-09 | 2.8E-12 |
| Ce144 | - | 6.6E-09 | - | 1.7E-08 | 2.4E-08 | 7.7E-10 | - | 7.7E-10 | 7.7E-21 | 2.0E-11 | 3.9E-10 |
| Pr143 | - | 6.5E-09 | - | 1.8E-08 | 2.4E-08 | 8.1E-10 | - | 8.1E-10 | 7.9E-21 | 9.0E-10 | 8.8E-12 |
| Pr144 | - | 6.5E-09 | - | 1.7E-08 | 2.3E-08 | 6.5E-10 | - | 6.5E-10 | 7.6E-21 | 2.0E-07 | 3.8E-14 |
| Np239 | - | 6.3E-08 | - | 3.2E-07 | 3.8E-07 | 1.5E-08 | - | 1.5E-08 | 1.3E-19 | 3.0E-09 | 4.2E-11 |

# 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 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Na24 | - | 9.4E-06 | - | 3.2E-04 | 3.3E-04 | 1.5E-05 | - | 1.5E-05 | 1.1E-16 | 7.0E-09 | 1.6E-08 |
| Cr51 | - | 6.7E-03 | - | 1.8E-05 | 6.7E-03 | 8.2E-07 | - | 8.2E-07 | 2.1E-15 | 3.0E-08 | 7.1E-08 |
| Mn54 | - | 3.6E-03 | - | 9.3E-06 | 3.6E-03 | 4.2E-07 | - | 4.2E-07 | 1.1E-15 | 1.0E-09 | 1.1E-06 |
| Fe55 | - | 2.7E-03 | - | 7.0E-06 | 2.7E-03 | 3.2E-07 | - | 3.2E-07 | 8.6E-16 | 3.0E-09 | 2.9E-07 |
| Fe59 | - | 6.6E-04 | - | 1.7E-06 | 6.6E-04 | 8.0E-08 | - | 8.0E-08 | 2.1E-16 | 5.0E-10 | 4.2E-07 |
| Co58 | - | 1.0E-01 | - | 2.7E-05 | 1.0E-01 | 1.2E-06 | - | 1.2E-06 | 3.2E-14 | 1.0E-09 | 3.2E-05 |
| Co60 | - | 1.2E-03 | - | 3.1E-06 | 1.2E-03 | 1.4E-07 | - | 1.4E-07 | 3.8E-16 | 5.0E-11 | 7.6E-06 |
| Ni63 | - | 6.0E-04 | - | 1.5E-06 | 6.0E-04 | 7.0E-08 | - | 7.0E-08 | 1.9E-16 | 2.0E-09 | 9.5E-08 |
| Zn65 | - | 1.2E-03 | - | 3.0E-06 | 1.2E-03 | 1.3E-07 | - | 1.3E-07 | 3.7E-16 | 4.0E-10 | 9.1E-07 |
| Zr95 | - | 8.7E-04 | - | 2.3E-06 | 8.7E-04 | 1.0E-07 | - | 1.0E-07 | 2.7E-16 | 4.0E-10 | 6.9E-07 |
| Ag110m | - | 2.9E-03 | - | 7.5E-06 | 2.9E-03 | 3.4E-07 | - | 3.4E-07 | 9.3E-16 | 1.0E-10 | 9.3E-06 |
| W187 | - | 1.2E-03 | - | 1.6E-05 | 1.3E-03 | 7.4E-07 | - | 7.4E-07 | 4.0E-16 | 1.0E-08 | 4.0E-08 |
| H3 | - | 6.6E+02 | - | 6.3E+00 | 6.7E+02 | 7.4E+00 | - | 7.4E+00 | 2.1E-10 | 1.0E-07 | 2.1E-03 |
| C14 | 2.3E-01 | 5.4E-03 | - | 1.2E-03 | 2.4E-01 | 2.8E-07 | - | 2.8E-07 | 7.5E-14 | 3.0E-09 | 2.5E-05 |
| Ar41 | 2.3E+00 | - | 1.6E-02 | 1.6E+00 | 4.0E+00 | 1.5E-04 | 7.7E-01 | 7.7E-01 | 1.5E-12 | 1.0E-08 | 1.5E-04 |
| **Total** | **1.5E+02** | **6.7E+02** | **1.4E-01** | **2.0E+01** | **8.4E+02** | **7.4E+00** | **6.5E+00** | **1.4E+01** | **2.7E-10** | **5.8E-05** | **2.5E-03** |

Note- The X/Q 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 X/Q (undepleted/no decay) | Table 2.0-1 |
| Routine release D/Q | Table 2.0-1 |
| Milk animal | Greater of goat or cow |
| 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: Gaseous Effluent Dose Results for 10 CFR 50 Appendix I

| Type of Dose | Dose Estimate |
|---|---|
| Beta Dose Air (mrad/yr) | 0.10 |
| Gamma Dose Air (mrad/yr) | 0.02 |

| PATHWAY | T.BODY | GI-TRACT | BONE | LIVER | KIDNEY | THYROID | LUNG | SKIN |
|---|---|---|---|---|---|---|---|---|
| Plume | 1.2E-02 | 1.2E-02 | 1.2E-02 | 1.2E-02 | 1.2E-02 | 1.2E-02 | 1.3E-02 | 8.2E-02 |
| Ground | 2.7E-01 | 2.7E-01 | 2.7E-01 | 2.7E-01 | 2.7E-01 | 2.7E-01 | 2.7E-01 | 3.2E-01 |
| VEGETABLE | | | | | | | | |
| ADULT | 3.4E-01 | 6.6E-01 | 1.0E-01 | 3.3E-01 | 3.0E-01 | 3.9E-01 | 3.0E-01 | 2.9E-01 |
| TEEN | 4.1E-01 | 7.1E-01 | 1.6E-01 | 4.0E-01 | 3.6E-01 | 4.6E-01 | 3.5E-01 | 3.4E-01 |
| CHILD | 6.7E-01 | 7.8E-01 | 3.9E-01 | 6.4E-01 | 5.7E-01 | 7.7E-01 | 5.6E-01 | 5.5E-01 |
| MEAT | | | | | | | | |
| ADULT | 6.0E-02 | 1.8E-01 | 3.3E-02 | 5.8E-02 | 4.8E-02 | 4.9E-02 | 4.6E-02 | 4.5E-02 |
| TEEN | 4.0E-02 | 1.0E-01 | 2.7E-02 | 3.8E-02 | 3.0E-02 | 3.1E-02 | 2.9E-02 | 2.8E-02 |
| CHILD | 5.5E-02 | 7.3E-02 | 5.1E-02 | 4.9E-02 | 4.0E-02 | 4.2E-02 | 3.8E-02 | 3.7E-02 |
| COW MILK | | | | | | | | |
| ADULT | 1.2E-01 | 3.1E-01 | 5.0E-02 | 1.3E-01 | 1.1E-01 | 2.2E-01 | 1.0E-01 | 1.0E-01 |
| TEEN | 1.6E-01 | 3.7E-01 | 8.9E-02 | 1.8E-01 | 1.6E-01 | 3.2E-01 | 1.4E-01 | 1.3E-01 |
| CHILD | 2.6E-01 | 3.8E-01 | 2.2E-01 | 3.0E-01 | 2.6E-01 | 5.8E-01 | 2.2E-01 | 2.2E-01 |
| INFANT | 4.0E-01 | 5.3E-01 | 3.7E-01 | 4.9E-01 | 4.0E-01 | 1.2E+00 | 3.5E-01 | 3.5E-01 |
| GOAT MILK | | | | | | | | |
| ADULT | 2.3E-01 | 2.2E-01 | 4.7E-02 | 2.3E-01 | 2.1E-01 | 3.4E-01 | 2.0E-01 | 2.0E-01 |
| TEEN | 2.9E-01 | 2.9E-01 | 8.5E-02 | 3.2E-01 | 2.8E-01 | 4.8E-01 | 2.7E-01 | 2.6E-01 |
| CHILD | 4.5E-01 | 4.4E-01 | 2.0E-01 | 5.2E-01 | 4.6E-01 | 8.6E-01 | 4.3E-01 | 4.2E-01 |
| INFANT | 6.8E-01 | 6.8E-01 | 3.7E-01 | 8.4E-01 | 7.1E-01 | 1.7E+00 | 6.7E-01 | 6.5E-01 |
| INHALATION | | | | | | | | |
| ADULT | 1.5E-01 | 1.6E-01 | 2.1E-04 | 1.5E-01 | 1.5E-01 | 1.6E-01 | 1.9E-01 | 1.5E-01 |
| TEEN | 1.6E-01 | 1.6E-01 | 2.8E-04 | 1.6E-01 | 1.6E-01 | 1.6E-01 | 2.1E-01 | 1.6E-01 |
| CHILD | 1.4E-01 | 1.4E-01 | 3.8E-04 | 1.4E-01 | 1.4E-01 | 1.4E-01 | 1.8E-01 | 1.4E-01 |
| INFANT | 7.9E-02 | 7.9E-02 | 1.9E-04 | 7.9E-02 | 7.9E-02 | 8.2E-02 | 1.1E-01 | 7.9E-02 |
| **TOTAL** | | | | | | | | |
| **ADULT** | 2.8E-01 | 1.3E+00 | 1.9E-01 | 7.8E-01 | 7.2E-01 | 9.3E-01 | 7.4E-01 | 4.0E-01 |
| **TEEN** | 2.8E-01 | 1.3E+00 | 2.8E-01 | 9.2E-01 | 8.2E-01 | 1.1E+00 | 8.5E-01 | 4.0E-01 |
| **CHILD** | 2.8E-01 | 1.4E+00 | 6.6E-01 | 1.4E+00 | 1.2E+00 | 1.8E+00 | 1.2E+00 | 4.0E-01 |
| **INFANT** | 2.8E-01 | 7.6E-01 | 3.7E-01 | 9.2E-01 | 7.9E-01 | 1.8E+00 | 7.8E-01 | 4.0E-01 |

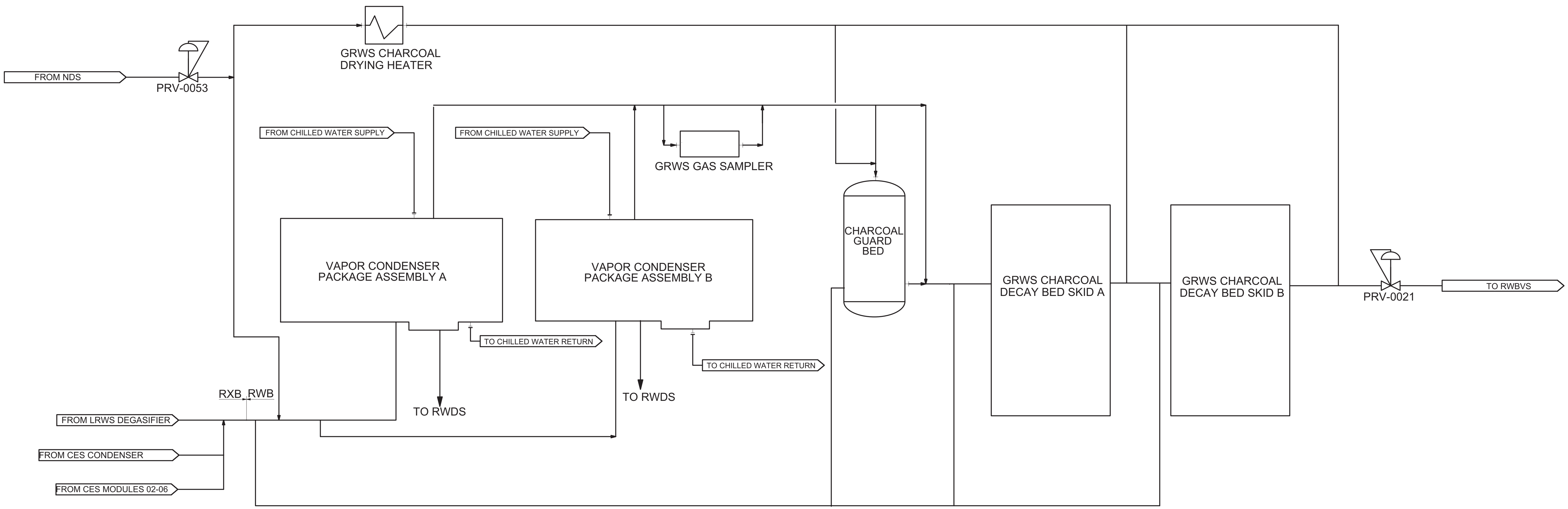
**Table 11.3-8: Gaseous Effluent Dose Evaluation for Gaseous Radioactive Waste System Failure**

| Parameter | Value |
|---|---|
| Release Source Term: | |
| I-131 | 4.1E-04 Ci |
| I-132 | 1.9E-04 Ci |
| I-133 | 6.2E-04 Ci |
| I-134 | 1.1E-04 Ci |
| I-135 | 3.9E-04 Ci |
| Xe-133 | 5.3E-02 Ci |
| Xe-135 | 9.7E+00 Ci |
| Kr-85m | 2.9E-02 Ci |
| Kr-85 | 8.4E-02 Ci |
| Kr-87 | 1.4E+01 Ci |
| Kr-88 | 3.8E-01 Ci |
| | |
| Dispersion factor (0-2 hour exclusion area boundary) | Table 2.0-1 |
| | |
| Offsite dose consequence | < 10 mrem |
| Allowable dose limit | 100 mrem |

## Table 11.3-9: Vapor Condenser Package Assembly Radiological Content

| Isotope | Activity ($Ci/cm^3$) |
|---|---|
| Kr83m | 1.2E-10 |
| Kr85m | 5.1E-10 |
| Kr85 | 9.2E-08 |
| Kr87 | 2.8E-10 |
| Kr88 | 8.0E-10 |
| Kr89 | 1.8E-11 |
| Xe131m | 2.1E-09 |
| Xe133m | 1.8E-09 |
| Xe133 | 1.3E-07 |
| Xe135m | 1.7E-10 |
| Xe135 | 3.6E-09 |
| Xe137 | 6.0E-11 |
| Xe138 | 2.1E-10 |
| Br82 | 1.8E-14 |
| Br83 | 1.0E-13 |
| Br84 | 4.8E-14 |
| Br85 | 5.8E-15 |
| I129 | 3.0E-19 |
| I130 | 1.5E-13 |
| I131 | 3.8E-12 |
| I132 | 1.7E-12 |
| I133 | 5.7E-12 |
| I134 | 1.0E-12 |
| I135 | 3.6E-12 |
| C14 | 1.6E-11 |
| Ar41 | 1.3E-08 |

**Figure 11.3-1: Gaseous Radioactive Waste System Diagram**



## 11.4 Solid Waste Management System

The solid waste management system is called the solid radioactive waste system (SRWS). The SRWS is designed to process both wet solid waste (WSW) and dry solid waste (DSW) from various plant systems produced during normal operation and anticipated operational occurrences, including startup, shutdown, and refueling operations. The Radioactive Waste Building (RWB) has adequate space for onsite storage for various solid waste containers plus space for mobile processing equipment. The SRWS includes the WSW system, DSW system, mixed waste system, and an onsite storage area.

The design basis source term identified in Section 11.1 forms the basis for the shielding design. The shield wall thickness evaluation assumes that the spent filters and spent resins fully loaded using the design basis source term. Section 12.3 discusses additional details on the shielding design.

The 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.

Onsite storage allows for radioactive decay with adequate storage in case of processing, maintenance or transportation delays. Onsite storage is adequate 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. The SRWS meets the design recommendations of BTP 11-3.

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 as low as reasonably achievable (ALARA). Section 12.3 provides additional information on area radiation monitors.

### 11.4.1 System Description

The SRWS is a nonsafety-related system, serves no safety-related functions, and is not risk-significant. The SRWS is designed to

- collect, process, sample, package, and store WSW generated from the chemical and volume control system (CVCS), pool cooling and cleanup system, and liquid radioactive waste system (LRWS), using both permanently installed and mobile equipment in the SRWS.
- collect, segregate, sample, package, and store compactible and non-compactible DSW.
- 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.

- transfer liquid wastes to the RWDS or LRWS.

The SRWS design handles three types of generated wastes: WSWs, DSWs, and miscellaneous wastes.

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 DSW, the boundaries are not always physical because much of DSW is collected from a variety of locations and transported through corridors to the solid radioactive waste sorting area.

For spent resins and granular activated charcoal, the SRWS starts downstream of the boundary valve from each demineralizer and carbon bed. Operators sluice spent resin into the SRSTs or PSTs for decay, and to waste containers.

For spent cartridge filters, the SRWS starts at the filter extraction point. Operators remove the spent filter from the filter housing and place it in a shielded spent filter transfer cask.

#### 11.4.1.1 Dry Solid Waste

Dry solid waste includes heating ventilation and air conditioning filters, tools and equipment, used personnel protective equipment, rags, paper, wood and miscellaneous cleaning supplies. Figure 11.4-1 summarizes the DSW handling and storage operation.

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.

#### 11.4.1.2 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 and reverse osmosis

The WSW is homogenized, sampled, and analyzed to classify the waste in accordance with 10 CFR 61. Operators transfer spent resin and spent charcoal to high integrity containers (HICs) that are connected to a dewatering system located inside a confined enclosure.

Operators cap and seal containers after dewatering, and survey and decontaminate the containers, as necessary, to meet 49 CFR 173 requirements.

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

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.

Figure 11.4-2a and Figure 11.4-2b are process flow diagrams of the spent resin handling system.

To avoid the generation of explosive gas mixtures and exothermic reactions, the upstream systems (LRWS, pool cooling and cleanup system, CVCS) that transfer resins to the SRST or phase separator tank (PST) do not use chemicals (e.g., nitrates, nitrites) that can generate exothermic reactions with resins.

The main source of oily waste is expected to come from floor drains. Operators direct the oil to the SRWS from the LRWS oil separators and manually collect it in drums. The drums of contaminated oil are sent to an offsite treatment facility.

#### 11.4.1.3 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 low. Mixed waste can only be disposed of in a permitted mixed waste disposal facility. Operators collect mixed waste near the source and transfer in drums to a permitted facility.

#### 11.4.1.4 Packaging, Storage, and Shipping

The Process Control Program (PCP) classifies waste as Class A, Class B, Class C, or greater than Class C in accordance with 10 CFR 61.55 and 10 CFR 61.56. Table 11.4-2 and Table 11.4-3 provide the expected annual volumes of solid waste and shipment offsite estimates. 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 RWB provides space for both Class A and Class B/C waste storage. Solid waste is typically stored below grade on the lower level. There is a storage area on the upper level for Class A waste. At the expected waste generation rates, there is storage capacity for at least 30 days.

The design and construction of SRWS components are in compliance with 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. Table 11.4-1 provides design parameters for each of the major components.

#### 11.4.1.4.1 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.

#### 11.4.1.4.2 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 high-efficiency particulate air filtration routed to the Radioactive Waste Building HVAC system (RWBVS) to control airborne contamination. Liquid removed by the dewatering system is routed to the LRWS low-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, ANSI/ANS-55.1-1992 (Reference 11.4-1), and ANSI/ANS-40.37-2009 (Reference 11.4-2).

### 11.4.1.5 Effluent Controls

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.

### 11.4.1.6 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 Section 11.2 and Section 11.3.

#### 11.4.1.7 Mobile or Temporary Equipment

The design of SRWS includes modular equipment (e.g., spent resin dewatering system) and options for additional mobile equipment (e.g., shredders, laundry unit). The purpose of modular and mobile equipment is to provide ease of equipment replacement due to either advances in treatment technologies or equipment problems.

### 11.4.2 Radioactive Effluent Releases

The SRWS sends liquid and gaseous effluents to the LRWS and RWBVS, respectively. As a result, other than solid waste shipments offsite, the SRWS does not 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 Section 11.2 and Section 11.3.

The SRWS design complies with the requirements of 10 CFR 20.1406. Section 12.3 discusses the SRWS design features to prevent the spread of contamination, facilitate decommissioning, and reduce the generation of radioactive waste.

The PCP follows the guidance of Nuclear Energy Institute 07-10A (Reference 11.4-3). The PCP describes the administrative and operational controls used for the solidification of liquid or WSW and the dewatering of WSW.

### 11.4.3 Malfunction Analysis

To demonstrate the design's resistance to failures, a malfunction analysis is performed. Table 11.4-4 summarizes this malfunction analysis.

### 11.4.4 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.

### 11.4.5 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, Revision 0, 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 BVPC Section VIII | Vertical Conical | 10,000 gal | 175 | 155 | Austenitic Stainless Steel | 12.2-18 |
| SRST transfer pump (2) | RW-IIc | API-685 | Sealless, centrifugal | 75 gpm & 200 gpm | 238 | 180 | Austenitic Stainless Steel | - |
| Phase separator tank (2) | RW-IIa | ASME BVPC Section VIII | Vertical Conical | 12,500 gal | 175 | 155 | Austenitic Stainless Steel | 12.2-18 |
| PST transfer pump (2) | RW-IIc | API-685 | Sealless, centrifugal | 75 gpm | 238 | 180 | Stainless Steel | - |
| Dewatering skid (1) | RW-IIc | ANS-55.1 | - | 35 gpm | 238 | 180 | Stainless Steel | - |
| Compactor (1) | - | ANS-55.1 | - | 40 $ft^3$ | 238 | 130 | 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 | Filters | 194 | B-25 box | 90 | 3 |
| Class A | PPE/rags | 2500 | B-25 box | 90 | 28 |
| Class A | Tools | 9 | drum | 7.4 | 2 |
| Total Class A | | 2700 | | | |
| Class B/C | Failed equipment | 14 | drum | 7.4 | 2 |

## 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 off) |
|---|---|---|---|---|---|
| Class B/C | Resin | 320 | HIC | 120 | 3 |
| Class B/C | Cartridge Filters | 26 | HIC | 120 | 1 |
| Class B/C | Membrane Filters | 4 | Drum | 7.4 | 1 |
| Total Class B/C | | 350 | | | |
| Class A | Resin/Activated Charcoal | 940 | HIC | 120 | 9 |
| Class A | Filters | 4 | Drum | 7.4 | 1 |
| Class A | self-contained filter | 24 | IP-1 | 24 | 1 |
| Total Class A | | 970 | | | |

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

| Equipment Item | Malfunction | Results (Consequences) | Mitigating or Alternate Actions |
|---|---|---|---|
| Spent Resin Storage Tank | Tank failure | There are two Spent Resin Storage Tanks to collect Class B/C spent resins from the PCW and the CVC ion exchangers. The PCW/CVC demineralizers would not be able to send spent resins to the Spent Resin Storage Tanks for processing. | If one of the tanks fail, the other tank can receive the PCW and CVC resins. Alternate action includes direct sluicing to the Dewatering Skid. |
| Spent Resin Transfer Pump | Pump failure | There are two Spent Resin Transfer Pumps with one pump dedicated to each Spent Resin Storage Tank. The consequence would be that liquid waste could not be transferred to the LRW collection tanks and the pumps would not be able to transfer spent resin to the Spent Resin Storage Tanks or HICs. | The tank transfer line is cross-connected to the pump suction, allowing the process to continue using the redundant pump when one pump fails[1]. |
| Phase Separator Tank | Tank failure | There are two Phase Separator Tanks to collect Class A spent media. The LRW processing equipment or CVC auxiliary ion exchangers would not be able to transfer spent media to the phase separator tanks or the HIC. | If one tank fails, the other tank would continue to receive spent media. Alternate action includes direct sluicing to the Dewatering Skid. |
| Phase Separator Transfer Pump | Pump failure | There are two pumps, with one pump dedicated to each tank. The consequence would be that liquid waste could not be transferred to the LRW collection tanks and the pumps would not be able to be used to transfer spent resin to the Phase Separator Tank or HICs. | The separator transfer line is cross connected to the pump suction. The other pump allows processing to continue when one pump fails[1]. |
| Dewatering Skid | Skid component failure | The major components for the Dewatering Skid are not equipped with standby units. Excess liquid waste from the HIC cannot be extracted using the dewatering pump. If the level control valve on the dewatering skid fails, the HIC may be overflowed. | No impact on collection of WSW. The Phase Separator Tanks or the Spent Resin Storage Tanks can store the waste until dewatering skid components are repaired or replaced. The HICs are equipped with a camera on the fill head to monitor the HIC level. The video monitors external leaks associated with the HIC. |
| High Integrity Container | Container is dropped during transportation | HICs are transported between the fill station, storage area, and truck bay area. Dropping a HIC can cause local contamination. The floor drains in the area collect the liquid; however, the solid portion needs to be removed by a shop vacuum. | The grapple assembly has limit switches to ensure all the legs are engaged prior to lifting the HIC. In addition, the crane has its own safety brake system to ensure the HIC is not dropped during the power failure. |
| Spent Resin Storage Tanks and Phase Separator Tanks | Service Air Supply | The tanks cannot be pressurized to perform pneumatic sluicing.<br>The Spent Resin Storage Tanks and Phase Separator Tanks would not be able to send spent resins to the HICs for shipping offsite. | If one service air compressor fails, the backup service air compressor is used to pressurize the tanks to complete the sluicing[2]. If service air fails during the resin transfer, the lines are flushed to the HIC and system is restored to standby position. |

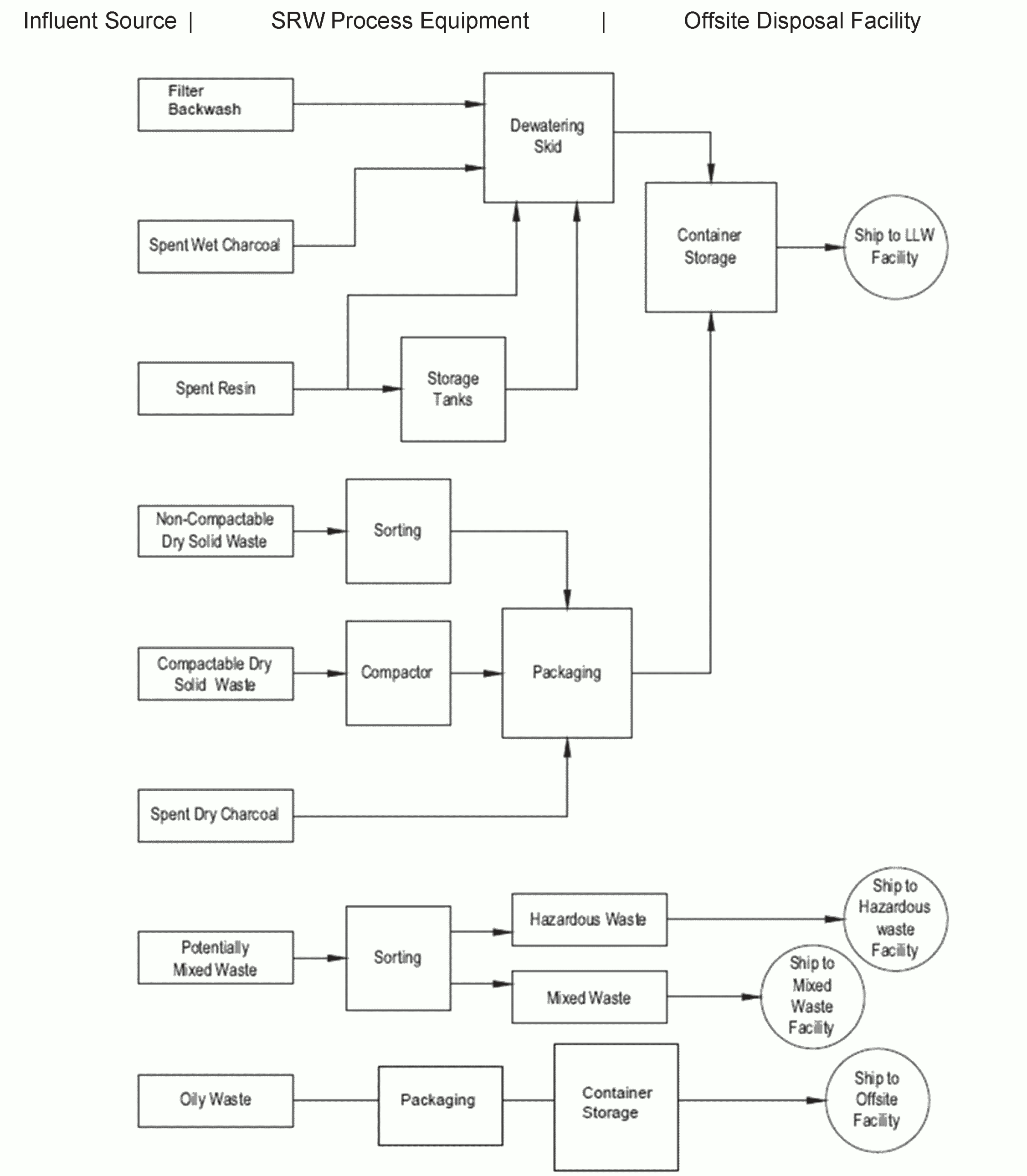
**Table 11.4-4: Solid Radioactive Waste System Equipment Malfunction Analysis**

| Equipment Item | Malfunction | Results (Consequences) | Mitigating or Alternate Actions |
|---|---|---|---|
| Dewatering Skid hoses | Hose ruptures or flange failure | The dewatering room is contaminated, and resin slurry enters the drainage system. | The video in the room shows the hoses connected to the HIC and dewatering operation. Operation is stopped manually, and affected hose is replaced after decontaminating the area. |

Notes:

1. Pumps are provided with drain connections to the RWD to prevent the spread of contamination from leaks or from repairs.
2. Service air compressors are part of the service air system (SAS). Section 9.3 contains information on the SAS.

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




Notes:

1) The Drum Dryer skid is part of the LRW. Filled, 55-gallon drums are transferred to the SRW for storage ("Container Storage") and eventual disposal.

2) Oily waste is part of the LRWS. Oily waste influent from the LRWS is packaged in 55-gallon drums in the RWB and then shipped offsite.

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

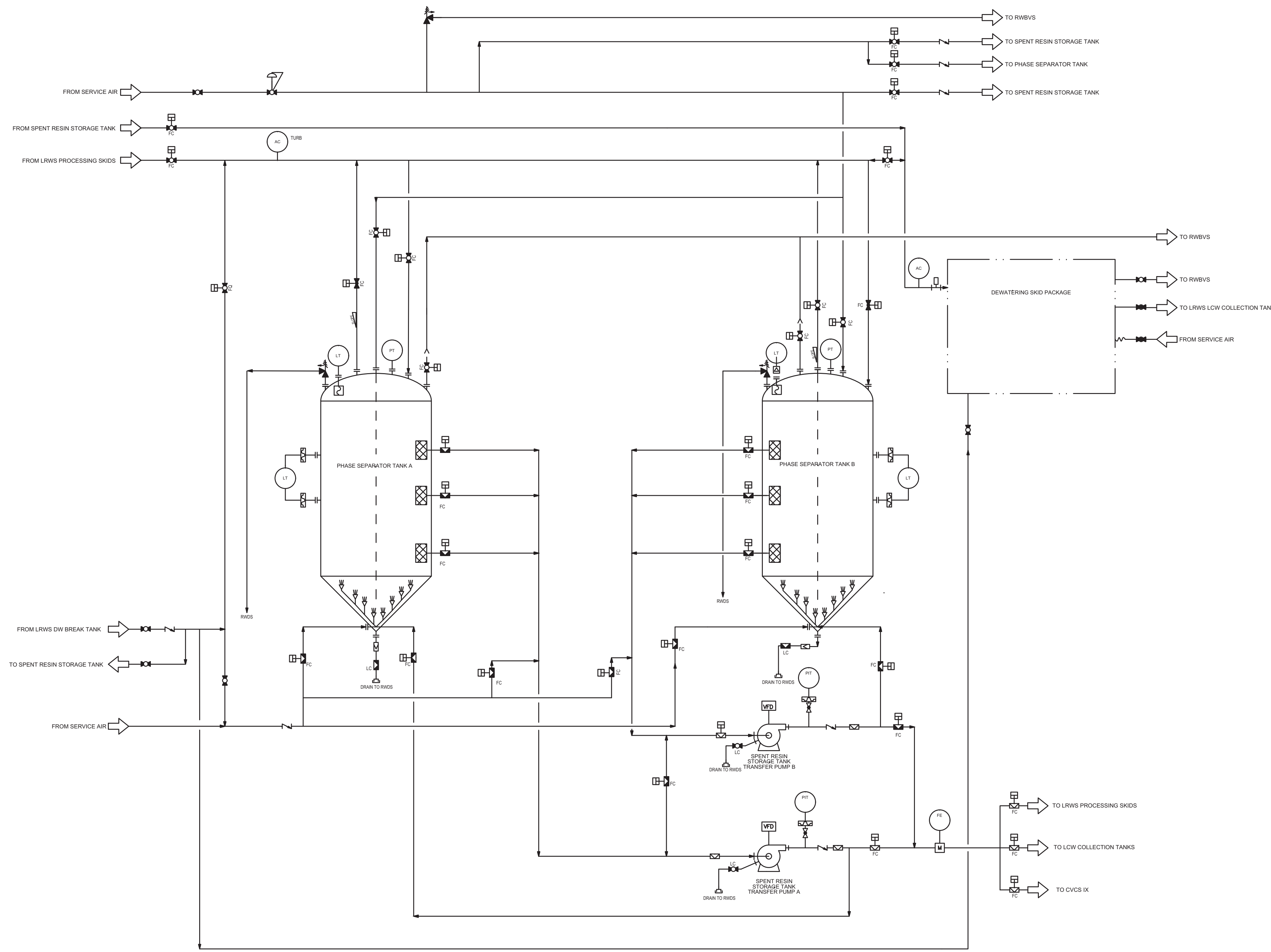
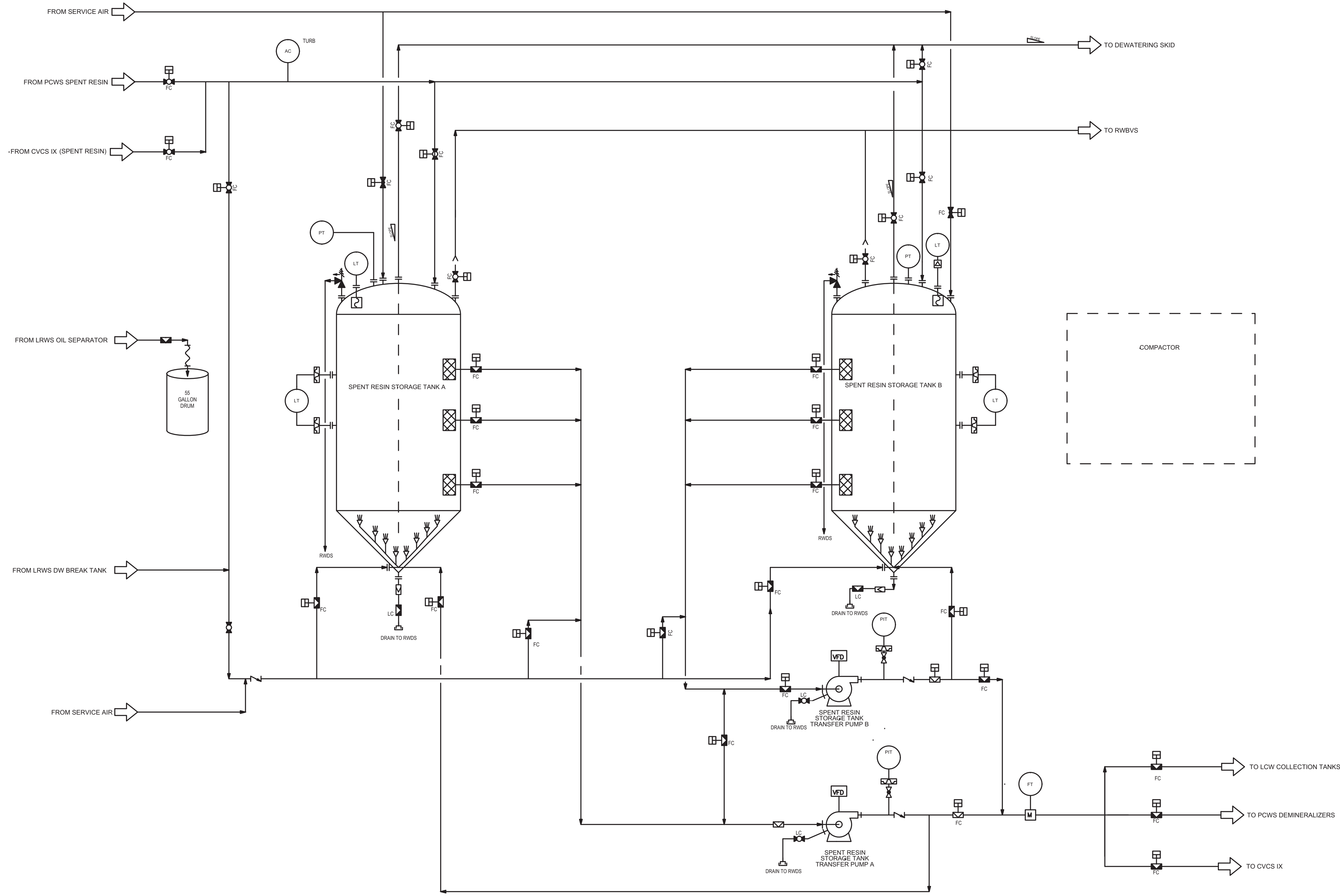


**Figure 11.4-2b: Solid Radioactive Waste System Diagram**



## 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.

### 11.5.1 System Description

Effluent Radiation Monitoring is provided for:

- Air cooled condenser system (ACCS) (Section 10.4.1)
- Liquid radioactive waste system (LRWS) (Section 11.2)
- Pool cooling and cleanup system (PCWS) (Section 9.1.3)
- Reactor Building HVAC system (RBVS) (Section 9.4.2)
- Site cooling water system (SCWS) (Section 9.2.7)
- Utility water system (UWS) (Section 9.2.9)

Process Radiation Monitoring is provided for:

- Auxiliary boiler system (ABS) (Section 10.4.7)
- Balance-of-plant drain system (BPDS) (Section 9.3.3)
- Chemical and volume control system (CVCS) (Section 9.3.4)
- Condensate polisher resin regeneration system (CPS) (Section 10.4.5)
- Containment evacuation system (CES) (Section 9.3.6)
- Containment flooding and drain system (CFDS) (Section 9.3.7)
- Normal control room HVAC system (CRVS) (Section 9.4.1)
- Demineralized water system (DWS) (Section 9.2.3)
- Gaseous radioactive waste system (GRWS) (Section 11.3)
- Main steam system (MSS) (Section 10.3)
- Reactor component cooling water system (RCCWS) (Section 9.2.2)
- Radioactive Waste Building HVAC system (RWBVS) (Section 9.4.3)
- Radioactive waste drain system (RWDS) (Section 9.3.3)
- Turbine generator system (TGS) (Section 10.2)

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

- Detector information including number, type, location, and measurement 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.
- Effluent and process monitoring off-normal radiation conditions are described in Table 11.5-4.
- Figure 11.5-1a and Figure 11.5-1b present 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.

Monitoring and operator response for effluent and process radiation monitors is performed in accordance with site procedures. Controls ensure that gaseous effluent content meet the objectives of 10 CFR 50 Appendix I and 10 CFR 20 before being released into the environment, and ensures compliance with GDC 60, 63, and 64.

Setpoints for radiation alarms (Section 11.5.1.2) and automated function initiation (Section 11.5.1.3) are based on ensuring that the limitations of 10 CFR 20 and 10 CFR 50 are met for plant conditions. Additionally, the alarms and isolations ensure compliance with GDC 60, 61, 63 and 64, and the applicable 10 CFR 20 and 10 CFR 50 requirements and limitations.

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.

Stack flow measurement capability supports the consideration of atmospheric dispersion ($\chi$/Q) and deposition (D/Q) factors when developing alarm setpoints.

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

#### 11.5.1.1 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.

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

#### 11.5.1.2 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 Offsite Dose Calculation Manual (ODCM), with consideration given to site-specific liquid effluent dilution factors and gaseous effluent atmospheric dispersion conditions.

#### 11.5.1.3 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.

#### 11.5.1.4 Offsite Dose Calculation Manual and Radiological Environmental Monitoring Program

The 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.

The ODCM and Radiological Environmental Monitoring Program are developed and implemented in accordance with the recommendations and guidance of NEI 07-09A (Reference 11.5-2).

#### 11.5.1.5 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, 1.45, and 1.97.

### 11.5.2 References

11.5-1 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.

11.5-2 Nuclear Energy Institute, "Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description," NEI 07-09A, Revision 0, March 2009.

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

| System | Quantity | Type | Service | Isotopes | Measurement Range | Location/Function | Safety-related | Media | Instrument type |
|---|---|---|---|---|---|---|---|---|---|
| ABS | 1 | γ | ABS Skid Vent ATM | Cs-137 | 3E-10 to 1E-6 µCi/cc | ABS Skid Vent to Atmosphere | No | Gas | Off-line (PING) |
| | | γ | | I-131 | 3E-10 to 5E-8 µCi/cc | | | | |
| | | β | | Kr-85<br>Xe-133 | 3E-7 to 1E-2 µCi/cc | | | | |
| | | γ | | Ar-41 | 1E-7 to 1E-1 µCi/cc | | | | |
| ABS | 1 | γ | ABS/Superheater skid BPDS Outlet | Ar-41 | 1E-7 to 1E-1 µCi/cc | ABS/Superheater Skid BPDS Outlet | No | Gas | Adjacent-to-line |
| ABS | 1 | γ | ABS Skid to AB Superheater Skid Vent | Cs-137 | 3E-10 to 1E-6 µCi/cc | ABS/Superheater Skid BPDS Vent | No | Gas | Off-line (PING) |
| | | γ | | I-131 | 3E-10 to 5E-8 µCi/cc | | | | |
| | | β | | Kr-85<br>Xe-133 | 3E-7 to 1E-2 µCi/cc | | | | |
| | | γ | | Ar-41 | 1E-7 to 1E-1 µCi/cc | | | | |
| ABS | 1 | γ | Auxiliary Steam to BPD | Ar-41 | 1E-7 to 1E-1 µCi/cc | Auxiliary Steam to BPDS | No | Gas | Adjacent-to-line |
| ABS | 1 | γ | TGB Auxiliary Steam Header | Ar-41 | 1E-7 to 1E-1 µCi/cc | TGB Auxiliary Steam Header | No | Gas | Adjacent-to-line |
| ACCS | 6 | γ | SJAE Gaseous Effluent | Ar-41 | 1E-6 to 1E-1 µCi/cc | SJAE Vent Air Evacuation Line | No | Gas | Adjacent-to-line |
| ACCS | 1 | γ | LRVP Gaseous Effluent | Ar-41 | 1E-6 to 1E-1 µCi/cc | LRVP Vent Air Evacuation Line | No | Gas | Adjacent-to-line |
| ACCS | 1 | β | CARS Common Vent Air Evacuation Line | Cs-137 | 3E-10 to 1E-6 µCi/cc | CARS Common Vent Air Evacuation Line | No | Gas | Off-line (PING) |
| | | γ | | I-131 | 3E-10 to 5E-8 µCi/cc | | | | |
| | | β | | Kr-85<br>Xe-133 | 3E-7 to 1E-2 µCi/cc | | | | |
| | | γ | | Ar-41 | 1E-6 to 1E-1 µCi/cc | | | | |
| BPDS | 1 | γ | Turbine Building Floor Drains Radiation | Cs-137 | 1E-7 to 1E-2 µCi/ml | Turbine Building Floor Drains Radiation | No | Liquid | In-line |
| BPDS | 1 | γ | Condensate Regeneration Skid | Cs-137 | 1E-7 to 1E-2 µCi/ml | Condensate Regeneration Skid | No | Liquid | In-line |
| BPDS | 1 | γ | Aux Boiler Blowdown | Cs-137 | 1E-7 to 1E-2 µCi/ml | Aux Boiler Blowdown | No | Liquid | In-line |
| CES | 6 | γ | Sample Tank Liquid Radiation | Cs-137 | 1E-7 to 1E-1 µCi/ml | Sample Tank Liquid Radiation | No | Liquid | Adjacent-to-line |

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

| System | Quantity | Type | Service | Isotopes | Measurement Range | Location/Function | Safety-related | Media | Instrument type |
|---|---|---|---|---|---|---|---|---|---|
| CES | 6 | γ<br>γ<br>β | Gaseous Discharge Particulate, Noble Gas, Iodine Radiation | Cs-137<br>I-131<br>Kr-85 Xe-133 | 3E-10 to 1E-6 μCi /cc<br>3E-10 to 5E-8 μCi /cc<br>3E-7 to 3E-2 μCi /cc | Gaseous Discharge Particulate, Noble Gas, Iodine Radiation | No | Gas | Off-line (PING) |
| CES | 6 | γ | Gaseous Discharge Ar-41 Radiation | Ar-41 | 1E-7 to 1E-1 μCi /cc | Gaseous Discharge Ar-41 Radiation | No | Gas | Off-line |
| CFDS | 1 | β | CFDS Drain Separator Gas Discharge | Kr-85 Xe-133 | 3E-7 to 1E-2 μCi /cc | CFDS radiation flow to HEPA filter | No | Gas | Off-line |
| CPS | 1 | γ | Resin Transfer Radioisotope concentration | Cs-137 | 1E-7 to 1E-2 μCi /ml | Turbine Generator Building (TGB)/Condenser Polisher Resin Regen Skid Inlet | No | Liquid | Adjacent-to-line |
| CPS | 1 | γ | Regeneration Sump Radioisotope Concentration | Cs-137 | 1E-7 to 1E-2 μCi /ml | TGB/Regeneration Sump | No | Liquid | Adjacent-to-line |
| CRVS | 2 | γ | CRVS Outside Air Intake Radiation Monitor | Kr-85 Xe-133 | 1E-5 to 1E+1 Rad/hr | Outside Air Intake | No | Air | Off-line |
| CRVS | 2 | γ<br>γ<br>β | CRE Supply Air Radiation | Cs-137<br>I-131<br>Kr-85 Xe-133 | 3E-10 to 1E-4 μCi /cc<br>1E-9 to 1E+2 μCi /cc<br>4E-5 to 1E+4 μCi /cc | CRE Supply Duct | No | Gas | Off-line (PING) |
| CVCS | 6 | γ | RPV Supply to regenerative heat exchanger (RHX) Cooling Inlet Radiation | Cs-137 | 1E-7 to 1E-2 μCi /ml | RPV Supply to RHX Cooling Inlet Radiation | No | Liquid | Adjacent-to-line |
| DWS | 1 | γ | DW North Reactor Building (RXB) Contaminated Header Radiation | Cs-137 | 1E-7 to 1E-2 μCi /ml | DW North RXB Contaminated Header Radiation | No | Liquid | Off-line |
| DWS | 1 | γ | DW South RXB Contaminated Header Radiation | Cs-137 | 1E-7 to 1E-2 μCi /ml | DW South RXB Contaminated Header Radiation | No | Liquid | Off-line |
| GRWS | 2 | β | Charcoal Decay Bed Skid A/B Outlet Radiation | Kr-85 Xe-133 | 3.0E-7 to 1.0E-2 μCi/ml | GRW Charcoal Decay Bed Skid A/B Outlet Radiation | No | Gas | Off-line |

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

| System | Quantity | Type | Service | Isotopes | Measurement Range | Location/Function | Safety-related | Media | Instrument type |
|---|---|---|---|---|---|---|---|---|---|
| GRWS | 1 | β | GRW Outlet Radiation | Kr-85<br>Xe-133 | 3.0E-7 to 1.0E-2 µCi/ml | GRW Outlet Radiation | No | Gas | Off-line |
| LRWS | 2 | γ | LRWS LCW Processing to UW radiation | Cs-137 | 1.0E-7 to 1.0E-2 µCi/ml | LWR discharge path | No | Liquid | Adjacent to-line |
| MSS | 6 | γ | SG1 Main Steam Line Radiation | Ar-41 | 1.0E-7 to 1.0E-1 µCi/cc | Main Steam Line<br>2 per steam header in the RXB | No | Gas | Adjacent to-line |
| MSS | 6 | γ | SG2 Main Steam Line Radiation | Ar-41 | 1.0E-7 to 1.0E-1 µCi/cc | Main Steam Line<br>2 per steam header in the RXB | No | Gas | Adjacent to-line |
| PCWS | 1 | β | Surge Control Storage Tank Vent | Kr-85<br>Xe-133 | 3.0E-7 to 1.0E-2 µCi/ml | Monitor the PCWS surge control storage tank vent line on PCWS | No | Gas | Off-line |
| RBVS | 1 | γ | Spent Fuel Pool (SFP) Exhaust Filter Upstream Radiation | Cs-137 | 3E-10 to 1E-6 µCi/cc | SFP Exhaust Filter Upstream | No | Gas | Off-line (PING) |
| | | γ | | I-131 | 3E-10 to 5E-8 µCi/cc | | | | |
| | | β | | Kr-85<br>Xe-133 | 3E-7 to 1E-2 µCi/cc | | | | |
| RBVS | 1 | γ | Reactor Pool and Gallery Area Exhaust | Cs-137 | 3E-10 to 1E-6 µCi/cc | Reactor Pool and Gallery Area Exhaust | No | Gas | Off-line (PI) |
| | | γ | | I-131 | 3E-10 to 5E-8 µCi/cc | | | | |
| RBVS | 1 | γ | RBVS Exhaust Stack | Cs-137 | 1E-7 to 1E-2 µCi/cc | RBVS Exhaust Stack | No | Gas | Off-line (PING) |
| | | γ | | I-131 | 3E-10 to 1E-6 µCi/cc | | | | |
| | | β | | Kr-85<br>Xe-133 | 3E-7 to 1E+4 µCi/cc | | | | |
| RBVS | 2 | γ | RBVS South and North Module Battery Rooms air-handling unit (AHU) Radiation | Cs-137 | 3E-10 to 1E-6 µCi/cc | RBVS North and South Module Battery Rooms AHU Radiation | No | Gas | Off-line (PING) |
| | | γ | | I-131 | 3E-10 to 5E-8 µCi/cc | | | | |
| | | β | | Kr-85<br>Xe-133 | 3E-7 to 1E-2 µCi/cc | | | | |
| RBVS | 1 | γ | SFP Exhaust Filter Upstream Radiation | Cs-137 | 3E-10 to 1E-6 µCi/cc | SFP Exhaust Filter Upstream | No | Gas | Off-line (PING) |
| | | γ | | I-131 | 3E-10 to 5E-8 µCi/cc | | | | |
| | | β | | Kr-85<br>Xe-133 | 3E-7 to 1E-2 µCi/cc | | | | |

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

| System | Quantity | Type | Service | Isotopes | Measurement Range | Location/Function | Safety-related | Media | Instrument type |
|---|---|---|---|---|---|---|---|---|---|
| RCCWS | 6 | γ | RCCW CVCS Non-regenerative heat exchanger Outlet Radiation | CS-137 | 1E-7 to 1E-2 μCi/ml | Located in the RCCW-downstream of loads that have potential for a radioactive release to alert the control room when there is a leak in the RCCWS | No | Liquid | Adjacent-to-line |
| RCCWS | 6 | γ | RCCW CES Condenser and Vacuum Pumps Outlet Radiation | CS-137 | 1E-7 to 1E-2 μCi/ml | Located in the RCCWS downstream of loads that have potential for a radioactive release to alert the control room when there is a leak in the RCCWS | No | Liquid | Adjacent-to-line |
| RCCWS | 1 | γ | RCCW PSS Primary Sample Chiller Outlet Radiation | CS-137 | 1E-7 to 1E-2 μCi/ml | Located in the RCCWS downstream of all cooled components | No | Liquid | Adjacent-to-line |
| RWBVS | 1 | γ | RWBV Exhaust Radiation Monitoring Skid | CS-137 | 3E-10 to 1E-6 μCi/cc | Located in the RWBV main exhaust duct before connecting to the RBVS exhaust duct | No | Gas | Off-line (PI) |
| | | γ | | I-131 | 3E-10 to 5E-8 μCi/cc | | | | |
| RWDS | 1 | γ | RXB RCCWS Drain Tank Radiation | Cs-137 | 1E-7 to 1E-2 μCi /ml | RCCWS drain tank | No | Liquid | Off-line |
| SCWS | 3 | γ | PCWS heat exchanger A/B/C Outlet Radiation | Cs-137 | 1E-7 to 1E-2 μCi /ml | Downstream of RXB Heat Exchangers/ PCWS outlet radiation | No | Liquid | In-line |
| SCWS | 1 | γ | CT Blowdown Radiation | C-137 | 1E-7 to 1E-2 μCi /ml | Cooling Tower Blowdown Line | No | Liquid | Off-line with Sampling |
| SCWS | 2 | γ | RCCW Heat Exchanger A/B Outlet Radiation | Cs-137 | 1E-7 to 1E-2 μCi /ml | Downstream of RXB Heat Exchangers/ RCCW outlet radiation | No | Liquid | In-line |
| TGS | 6 | γ | Gland Steam Outlet | Cs-137 | 3E-10 to 1E-6 μCi/cc | Turbine generator skid common exhaust vent point particulates | No | Gas | Off-line (PING) |
| | | γ | | I-131 | 3E-10 to 5E-8 μCi/cc | | | | |
| | | β | | Kr-85<br>Xe-133 | 3E-7 to 1E-2 μCi/cc | | | | |
| | | γ | | Ar-41 | 1E-6 to 1E-1 μCi/cc | | | | |

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

| System | Quantity | Type | Service | Isotopes | Measurement Range | Location/Function | Safety-related | Media | Instrument type |
|---|---|---|---|---|---|---|---|---|---|
| UWS | 1 | γ | Letdown Line Radiation | Cs-137 | 1E-7 to 1E-2 μCi/ml | UWS effluent path | No | Liquid | Off-line |

General Notes:
(a) - Main control room monitoring available
(b) - Waste management control room monitoring available
(c) - Local Monitoring Available
(d) - Designed to meet ANSI/HPS N13.1-2011 (Reference 11.5-1)
PING - Particulate, Iodine, Noble Gas
PI - Particulate, Iodine

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

| No. | Gaseous Process or Waste System | Sample Provisions[(a)] | |
|---|---|---|---|
| | | Process | Effluent |
| 1 | Auxiliary Boiler System | I, S&A | |
| 2 | Air Cooled Condensing System | I, S&A | S&A |
| 3 | Containment Evacuation System | I, S&A | |
| 4 | Control Room HVAC System | I | |
| 5 | Containment Flooding Drain System | S&A | |
| 6 | Gaseous Radioactive Waste System | I | |
| 7 | Main Steam System | S&A | |
| 8 | Pool Cooling & Clean Up System | S&A | S&A, H3 |
| 9 | Radioactive Waste Building HVAC System | I | - |
| 10 | Reactor Building HVAC System | I | NG, H3 |
| 11 | Reactor Building HVAC System (Spent Fuel Area) | I | |
| 12 | Turbine Generator System | I, 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 11.5-3: Provisions for Sampling Liquid Process and Effluent Streams

| No. | Liquid Process or Waste System | Sample Provisions[a] | |
|---|---|---|---|
| | | Process | Effluent |
| 1 | Balance of Plant Drain System | S&A, H3 | - |
| 2 | Containment Evacuation System[c] | S&A | - |
| 3 | Condensate Polisher Resin Regeneration System | S&A | - |
| 4 | Chemical and Volume Control System | S&A, H3 | - |
| 5 | Demineralized Water System | S&A, H3 | - |
| 6 | Liquid Radioactive Waste System[b] | S&A | S&A, H3 |
| 8 | Reactor Component Coolant Water System | S&A | |
| 9 | Radioactive Waste Drain System[b] | S&A | - |
| 10 | Site Cooling Water System | S&A | S&A, H3 |
| 11 | Utility Water System | S&A | S&A,H3 |

(a) - Sample point is available to obtain grab samples for laboratory analyses.
(b) - The provisions for sampling potentially contaminated system 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 CFR20.1406.
(c) - An installed mechanical liquid grab sampler located downstream of the CES sample vessel allows for the 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 process radiation level. 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 with 10 CFR 20.1406. Compliance with RG 1.45 requirements and the capabilities of the CES sample vessel are discussed in Section 5.2.5.
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 Monitoring Off Normal Radiation Conditions

| System | Condition | System Response |
|---|---|---|
| ABS | Radiation Detected | If high radiation is detected in the auxiliary boiler system skid vents, skid drains, or steam header drains, then the auxiliary boiler superheater skid outlet valve closes, auxiliary boiler skid to superheater skid valve closes, the module-specific main steam to auxiliary boiler header valves close, and the MCR receives an alarm. |
| BPDS | High Radiation | Upon alarm, the wastewater collection tank pumps are shut down, the discharge isolation valves directing the disposition of the water are both closed, and manual intervention is initiated. The chemical waste collection tank is also monitored for radiation and upon radiation detection, the two affected chemical waste collection tank pumps are disabled and the two affected discharge isolation valves are closed. The radiation monitor alarms in the main control room for action and the RWBS control for information and an operator is dispatched to assess the situation. |
| CES | High Radiation | Upon detection, the Purge Gas Supply to the vacuum pumps are shut off and that discharge path is switched from the RBVS to the GRWS valve. The SA Connection valve receives a close signal. |
| CPS | Radiation Detection | Spent resin being sent to the condensate polisher resin regeneration skid and the regeneration sump are monitored for radiation. If radiation is detected, a local alarm and an alarm in the MCR alerts operations staff. |
| CRVS | High Radiation levels continue to degrade<br><br>or<br><br>Radiation monitor power loss | 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 HEPA and charcoal filters, to filter outside air and minimize radiation exposure to personnel with the CRB.<br><br>If power is not available to either CRVS AHU or to any of the four EDS-C battery chargers (after a 10-minute time delay), or if levels of radiation greater than 10 times background in the CRE supply air duct or if toxic gas is detected in the CRE supply air duct, the PPS automatically isolates the CRE from the adjacent areas by closing the redundant CRE isolation dampers). The time delay is to allow operators time to restore power and start the stand-by AHU. The operating supply AHU and associated components, the general exhaust fan, and the battery exhaust fan are also turned off and the CRH is automatically initiated. The CRH provides a supply of breathable air for the CRE occupants and maintains the CRE at a positive pressure with respect to the surrounding areas. The heat sink capacity of surrounding structures of the CRE helps maintain the temperature in the CRE within acceptable tolerances. |

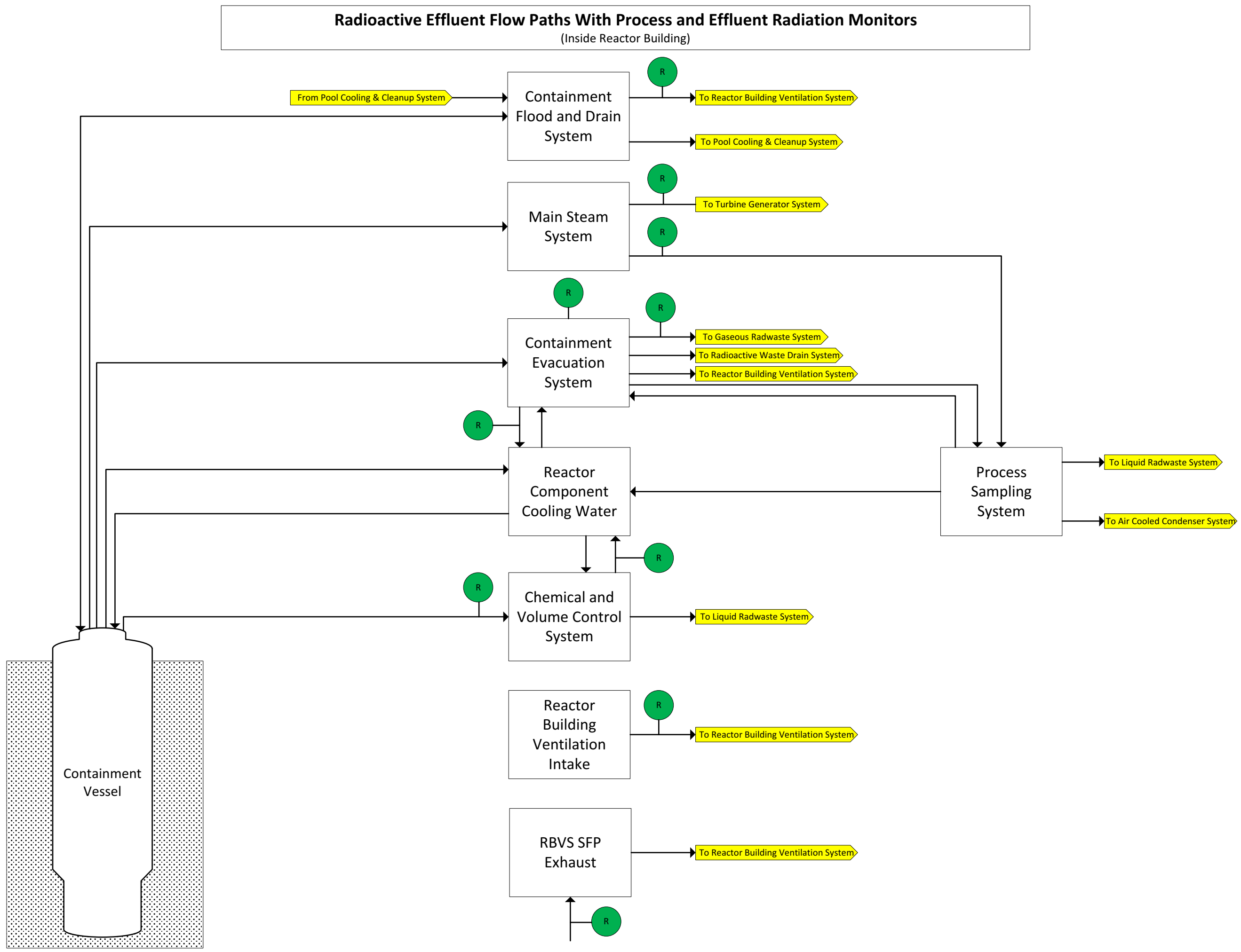
## Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions (Continued)

| System | Condition | System Response |
|---|---|---|
| CVCS | High Radiation<br><br>or<br><br>Radiation monitor power loss | Process sample line isolation |
| DWS | Radiation Alarm | PCS alarms and automatically close the associated upstream on-off valve. |
| GRWS | High radiation level in a decay bed skid outlet | Close the inlet valve and outlet valve from the affected decay bed skid as well as the outlet valves to RWBVS. |
| GRWS | High radiation level in the connection line to the RWBVS | Close the outlet valves to the RWBVS to stop the system flow. |
| GRWS | High radiation level in charcoal bed cubicle | Close the inlet valves to the GRWS to stop the system flow. Open the nitrogen purge valve. |
| LRWS | High Radiation on Sample Tank | Pump to Collection Tank for reprocessing (manual operator action) |
| LRWS | High Radiation on single Point LRW discharge | Table 11.2-2 |
| MSS | High Radiation | If a high radiation condition is detected on the main steam line radiation monitors, an alarm in the MCR cues the operators to take actions to mitigate the event per applicable operating procedures. If required, the main steam lines can be manually isolated from the MCR. Additionally, the MSS drain pots automatically isolate during high radiation for both normal operation or when a MPS isolation signal is present in order to ensure that the MSS does not contribute to unmonitored release of high radioactivity to the environment in the event of an abnormal tube leak. High radiation detection provides an alarm in the control room. If the drain pots and/or isolation valves to the ACC CCT are open, they close. Operator action from the MCR can isolate the MS lines, if required. |
| PCWS | High Radiation | Alarm in MCR to initiate appropriate safety actions. |
| RBVS | High Radiation | Alarm in MCR but no automatic actions. Operating staff takes actions to determine the source of the contamination and isolate it. |
| RBVS (SFP Area) | High Radiation | SFP exhaust is diverted through both the HEPA filters and the charcoal absorbers. Isolation dampers of the RXB general exhaust fans reduce speed in response to the damper closures to maintain the design exhaust header setpoint. |
| RWBVS | High Radiation | Upon high limit detection of radiation in the RWBVS exhaust effluent to the RBVS system, action is taken by plant operators to locate the source of contamination, however RWBVS continues to operate. |
| RWDS | High Radiation | PCS alarms and interlock closes the valve back to the RCCW expansion tank. |

**Table 11.5-4: Effluent and Process Monitoring Off Normal Radiation Conditions (Continued)**

| System | Condition | System Response |
|---|---|---|
| SCWS | Radiation Detection | Upon alarm, operators are alerted to abnormal condition, prompting them to investigate and isolate leaks or terminate other conditions that contribute to the off-normal conditions, through valve closures. |
| UWS | High Radiation | Alarm in the MCR and locally |

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



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

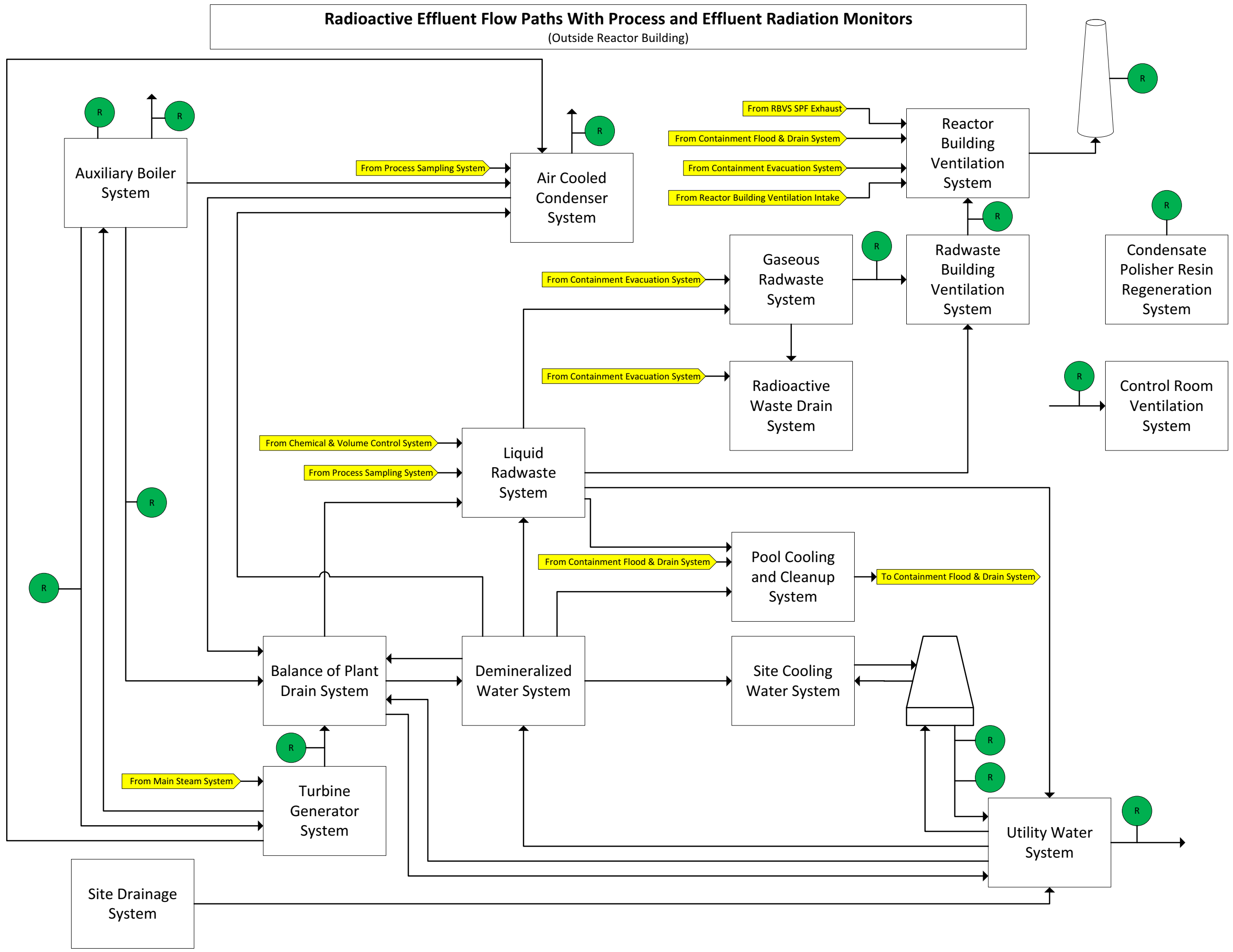


**Figure 11.5-2: Process and Effluent Radiation Monitoring System Instrumentation and Control Configuration**

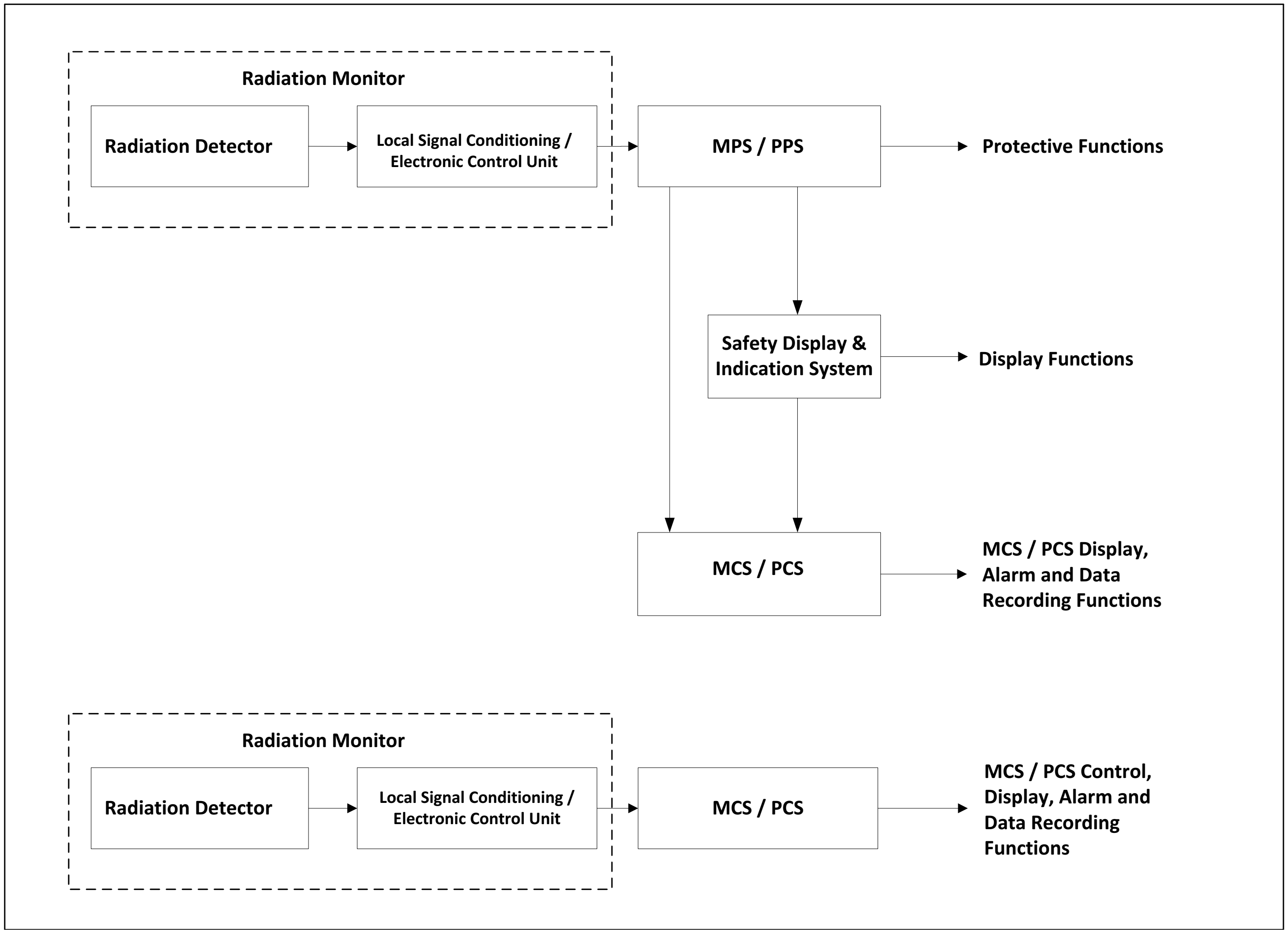


Figure 11.5-3: Off-Line Radiation Monitor

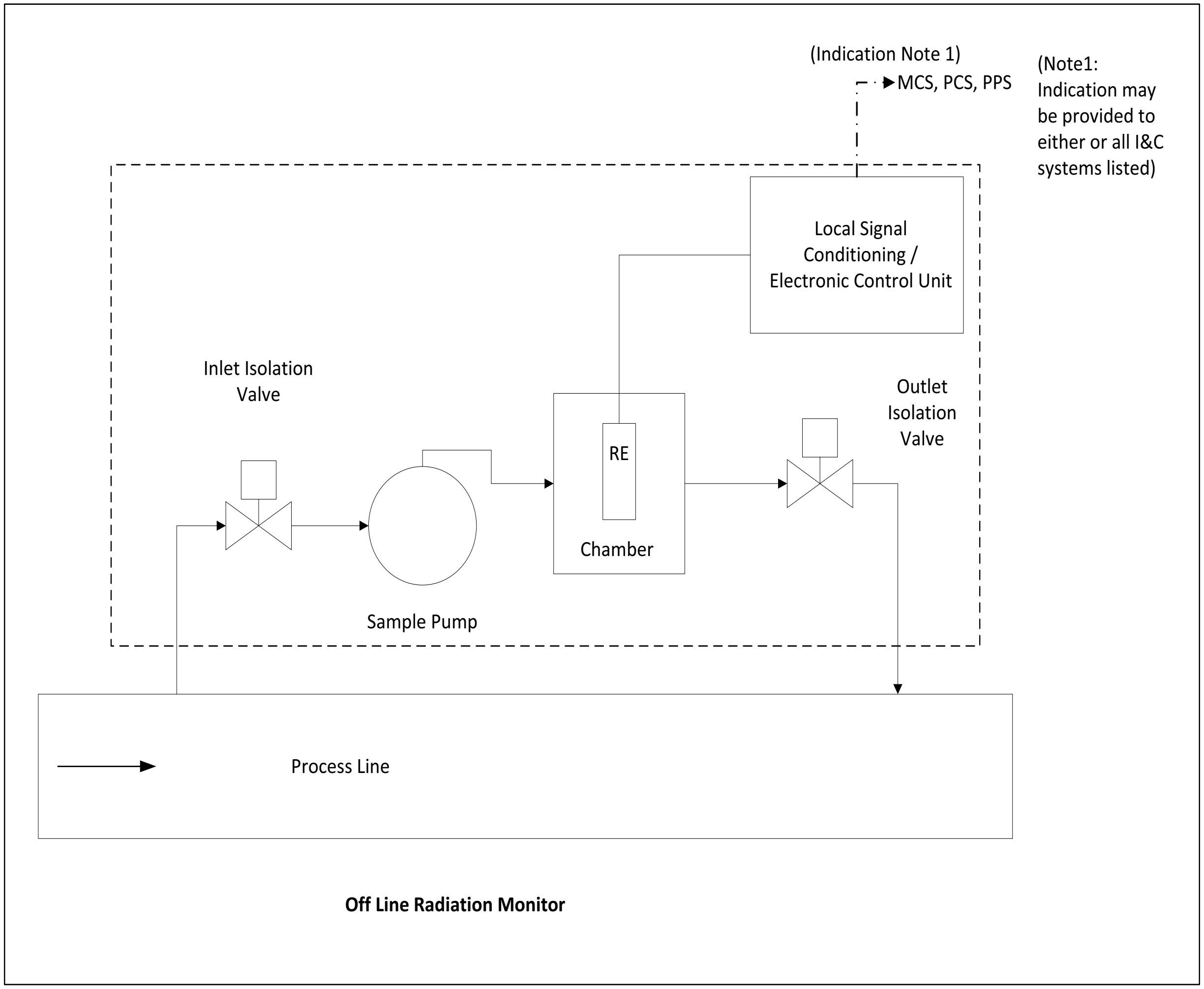


**Figure 11.5-4: Adjacent-to-Line Radiation Monitor**

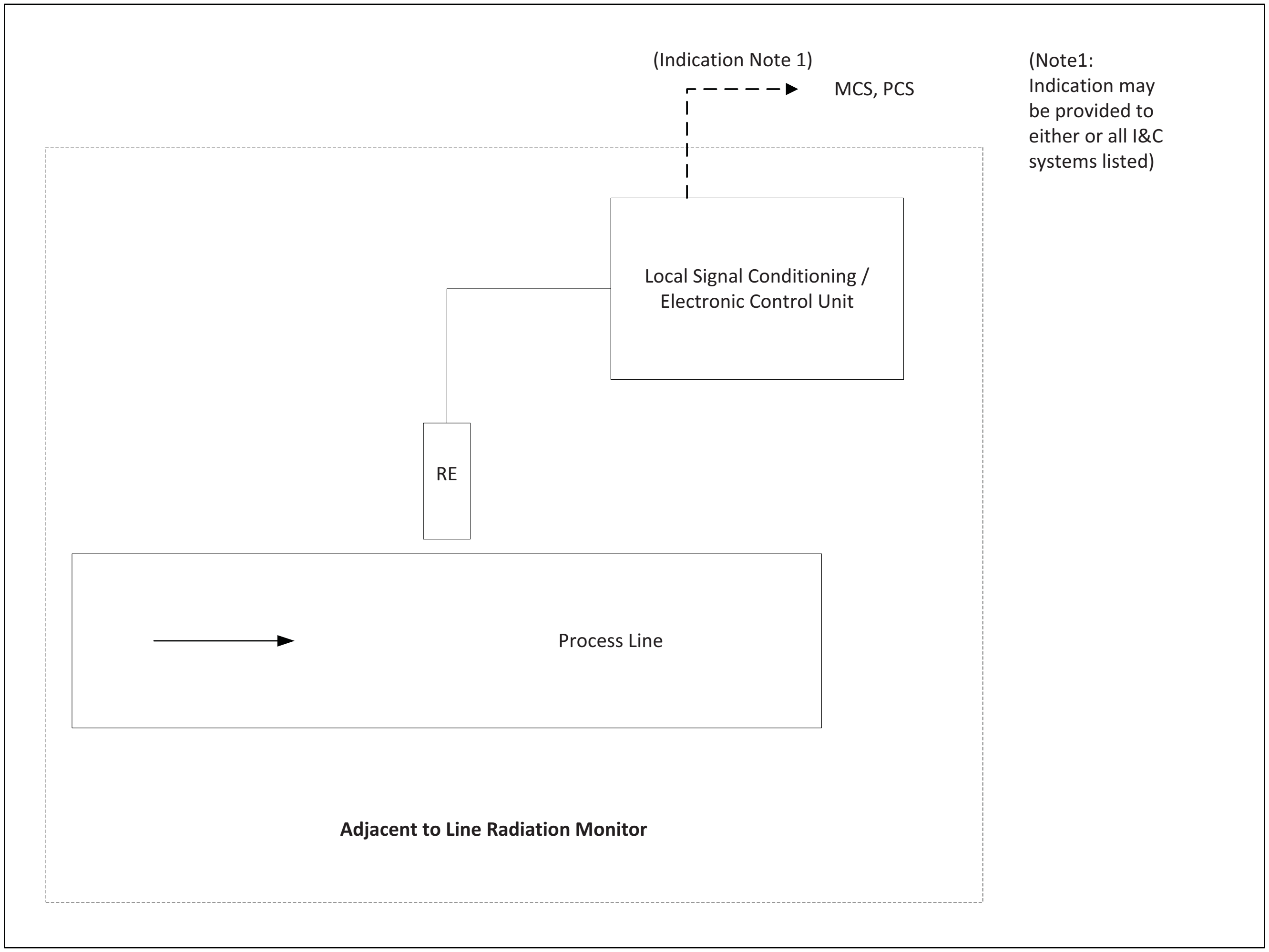


**Figure 11.5-5: In-Line Radiation Monitor**

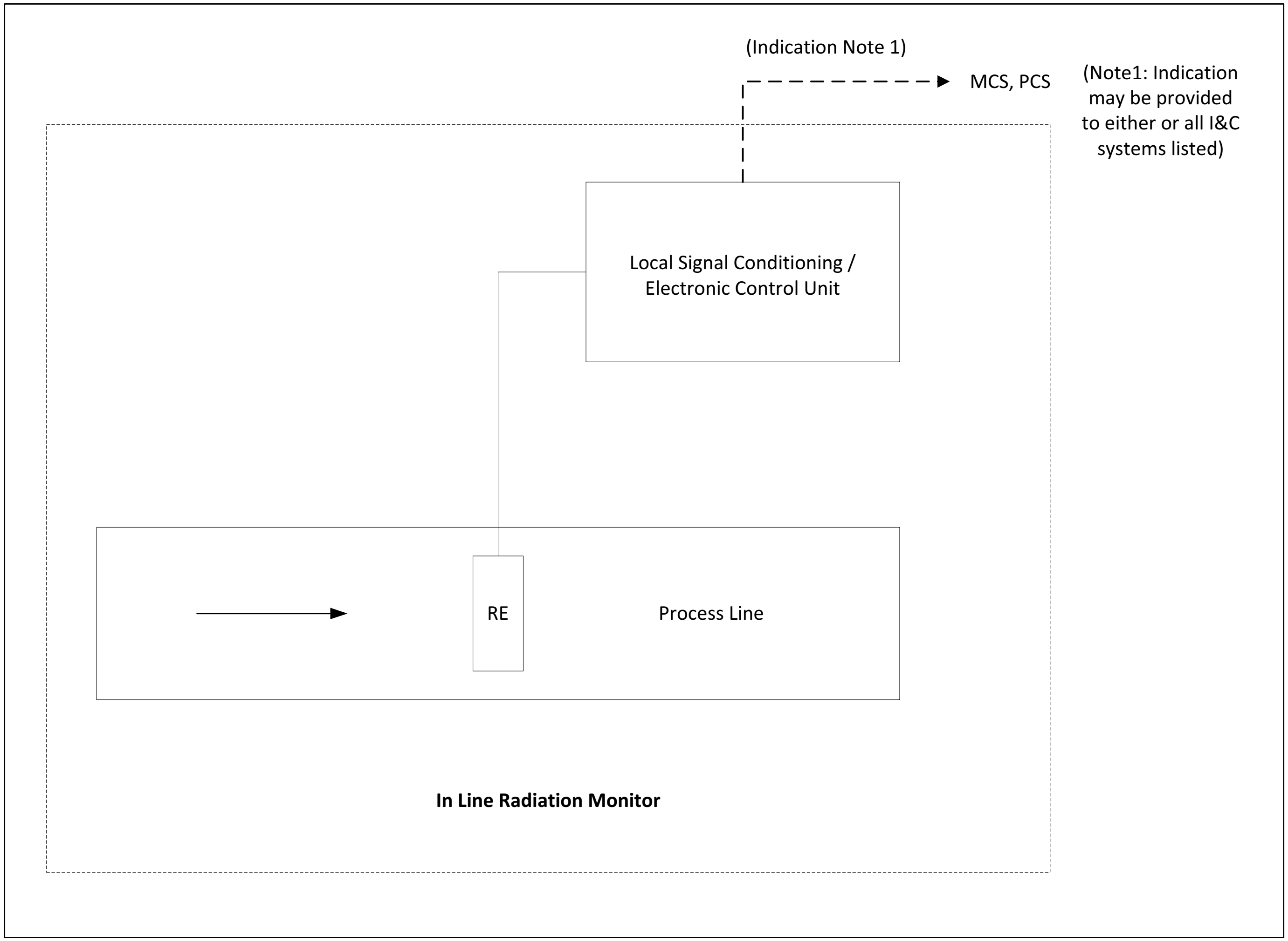
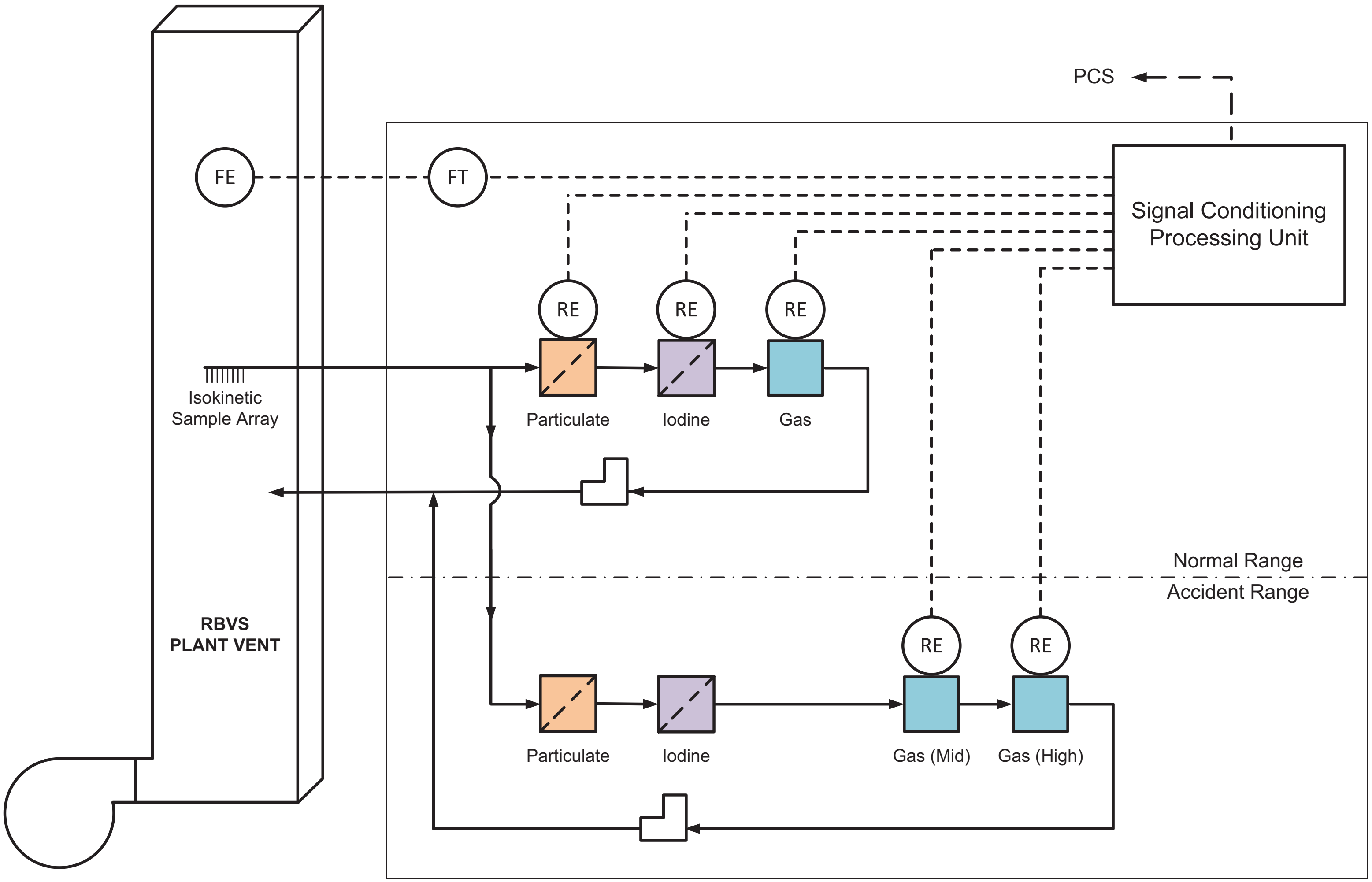


**Figure 11.5-6: Reactor Building HVAC System Plant Exhaust Stack Effluent Radiation Monitor**



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

Section 11.5 discusses effluent and process radiation monitors. 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.

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

Section 12.3.4 discusses area radiation and airborne contamination monitors. 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.

Section 7.2.13 describes effluent and area RM that provide input to the emergency response data system and the electronic data communication interface. Section 11.5 and Section 12.3.4 address effluent and area radiological monitoring parameters and equipment.

Applicable portions of the quality assurance program described in Section 17.5 administer the programs and procedures for the control of measuring and test equipment. Section 17.6 describes the program for monitoring the effectiveness of maintenance.

# Section B

The table below provides the NuScale responses to each of the Nuclear Regulatory Commission readiness assessment observations on draft Chapter 11, “Radioactive Waste Management” of the Standard Design Approval Application.

| Section | Observation | Response |
|---|---|---|
| 11.4 | Is storage of packaged radioactive waste in the RWB in the above grade portions of the building? If not, where will packaged radioactive waste be stored? | Section 11.4.1.4 states:<br>“The RWB provides space for both Class A and Class B/C waste storage. Solid waste is typically stored below grade on the lower level. There is a storage area on the upper level for Class A waste. At the expected waste generation rates, there is storage capacity for at least 30 days.” |
| 11.4 | The draft SDAA removed some design feature discussions related to Branch Technical Position 11-3. Examples are: (1) Components and piping that contain radioactive slurries should have flushing connections and piping runs that minimize the number of bends and traps that may retain radioactivity and lead to increased ambient external radiation exposure rates; and (2) Discussions related to the venting of storage tanks. Will this information be in Chapter 12? | Section 11.4.1.2 contains a discussion about how Branch Technical Position 11-3 is met. Section 11.4.1.4.1 discusses piping and valve design features. Additionally, Section 12.3.1.1.3 discusses piping ALARA design features and Section 12.3.1.1.1 discusses tank venting and overflow design features. |
| 11.4 | Section on component descriptions was removed from the draft SDAA. Table 11.4-1 and draft SDAA 11.4 figures inform the reader of the tanks and pumps part of the solid radioactive waste system, but not having descriptions of each component causes draft SDAA Section 11.4 not to address some of the as low as reasonably achievable (ALARA) design features involved with the design of the cubicles the tanks are in, features of pumps as they relate to the movement of resin between tanks and storage containers, as well as any accompanying assumptions that would be used for a dose analysis such as decay times prior to packaging. | Figures 11.4-2a and Figure 11.4-2b contain component details. Dose analysis specific to decay of waste in shipment containers is in the scope of 12.2 which defines the applicable source used for shielding analysis of the SRWS. SRWS shielded cubicles are discussed in 12.3.2.4.4. |
| 11.4 | Not many details are provided on the dewatering system. The document stated that it is a modular system, but details on the purpose of this system should be stated consistently as information stated in the DCA. Will it be a part of the SDAA design? | Section 11.4.1.4.2 contains a discussion about the dewatering system. |
| 11.4 | Instrumentation details are removed from draft SDAA Section 11.4. There are Section 11.5 pointers back to this Section 11.4 but no information on the monitoring equipment is provided. Will this information be provided in Section 11.4 in the SDAA? | There are no pointers from Section 11.5 to 11.4. Process monitoring is discussed in Section 11.5. Area and airborne monitoring is discussed in 12.3. |
| 11.5 | DCA Section 11.5 provided: (1) information on the sources of radioactivity being monitored, (2) locations of monitors, and (3) purposes of monitors. In draft SDAA Section 11.5 these details are either removed or moved to another section that are not available at this point of the readiness assessment. Draft SDAA Section 11.5 figures and tables are not | Table 11.5-1 in Section 11.5 provides information on the sources of radioactivity being monitored, locations of monitors, and the purposes of the monitors. Figure 11.5-1a through Figure 11.5-6 are now complete. The system |

| Section | Observation | Response |
|---|---|---|
| | available for review currently, although information on ranges, locations and function were previously found in Tables 11.5 for the DCA. | descriptions were moved to their respective sections within the SDAA. |
| 11.2 | Information related to the components details that make up the Liquid Radioactive Waste System (LRWS) are not present. Details related to what could be understood as ALARA design features are no longer discussed. Examples of this would be information on having sufficient volumes in tank cubicle rooms to prevent unmitigated release, tank venting when overfilling the tank, valves and pipes with backflow protections, pumps being in shielded cubicles and being well ventilated. | The description of the major components are located in Section 11.2.2, Table 11.2-1, and in Figure 11.2-1a through Figure 11.2-1j. Section 11.2.1 contains a discussion about ALARA features. Additional information about tanks, tank venting, piping, backflow protection, and shielding is available in Section 12.3. |
| 11.2 | Discussion of the LRWS in Section 11.2 is not as detailed as what was provided in the DCA and not as detailed as the discussion of the Gaseous Radioactive Waste System. Details that were provided in DCA Section 11.2 on the cleanup functions of the Low Conductivity Waste and High Conductivity Waste processing skids are not in the draft SDAA. Limited details are provided on how NuScale plans to process liquid waste (for example - will NuScale use Cation, Anion, Mixed Bed, and Cesium demineralizers?). Only brief mentions of the skid-based processing equipment are made in draft SDAA Section 11.2.2. More information is needed on how NuScale plans to process liquid waste. | Section 11.2.2 describes the liquid radioactive waste system (LRWS) and its major components. Figures 11.2-1a through Figure 11.2-1j provide process flows for the LRWS. Section 11.2 defines the performance requirements for the LRWS process skid. |
| 11.2, 11.3,11.4 | Based on the information currently provided, one cannot determine what is above and below grade for the RG 1.143 assessments. The purpose of this would be to identify the stated boundaries of each system and where they are in the RWB. This comment is applicable to draft SDAA Sections 11.2, 11.3, and 11.4. | Sections 11.2.2.6, 11.3.2.1, 11.3.2.3 and 11.4.1.4 provide a description of the design compliance with RG 1.143. Tables 11.2-1, 11.3-2 11.4-1 provide component classifications. |
| 11.2 | Not clear about the actual changes to the radwaste processing equipment. Cannot tell if the SDAA 11.2 changes have an impact on the liquid effluent wastes that will be released. Process flow figures were removed from SDAA Section 11.2. Will system diagrams be in the official submittal? | Section 11.2.2 describes the liquid radioactive waste system (LRWS) and its major components. Figures 11.2-1a through Figure 11.2-1j provide process flows for the LRWS. Section 11.2 defines the performance requirements for the LRWS process skid. Technical Report TR-123242 describes the radioactive effluent release methodology, inputs, and results. |

# Section C

| TR Number | TR Title |
|---|---|
| TR-123242-NP Revision 0 | Effluent Release (Gale Replacement) Methodology and Results |

Licensing Technical Report

# Effluent Release (GALE Replacement) Methodology and Results

December 2022

Revision 0

Docket: 52-050

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Licensing Technical Report

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This material is based upon work supported by the Department of Energy under Award Number DE-NE0008928.

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## Abstract

This technical report describes the methodology used to calculate normal operation, including anticipated operational occurrences (AOOs), annual radioactive gaseous and liquid effluents to the environment from an operating NuScale US460 Power Plant. The application of this methodology demonstrates compliance with regulatory requirements for normal radioactive effluents. There are no exemptions from existing regulations related to radioactive effluents. Regulatory requirements for effluents consist of a combination of annual release quantities, site boundary concentrations, and doses to members of the public. The methodology presented in this report uses first principles-based calculations where appropriate, combined with recent nuclear industry experience where applicable, and lessons learned where available, to determine NuScale-appropriate primary and secondary coolant concentrations of fission products, along with activated corrosion and wear products and coolant water activation products. These in-plant source terms form the basis for the evaluation of effluents.

The development of an alternate effluent release methodology is necessary because the existing PWRGALE code was developed in the 1980s for evaluation of the traditional large pressurized water reactors (PWRs) of that time and does not appropriately address unique characteristics of the NuScale Power Plant (NPP). The small modular reactor design is smaller, relies upon a significantly different passive design based on the natural processes of conduction, convection, gravity, and natural circulation to ensure safe shutdown, and the design is expandable with up to six NuScale Power Modules (NPMs) within the overall plant envelope. While the majority of individual NPM system designs are similar to traditional PWRs, a few systems vary from the large PWRs.

The primary and secondary coolant isotopic distribution is in Table A-2. The total calculated effluents are 850 Ci of gaseous effluent and 1,200 Ci of liquid effluent, with tritium being the largest contributor to both. The isotopic distribution totals are in Table A-3.

## Executive Summary

The NuScale Power Plant (NPP) design is similar to large pressurized water reactors (PWRs) in the existing fleet with regard to normal radioactive effluent release calculations. The development of an alternate methodology is necessary because the existing PWRGALE code was developed in the 1980s for evaluation of the large PWRs and does not appropriately address the US460 design. The US460

- is smaller. A single NuScale Power Module (NPM) provides approximately eight percent of the electrical output of the example PWR in NUREG-0017 (Reference 7.2.1). Table 2-1 provides a comparison between the NPM and the example PWR in NUREG-0017.
- relies upon a different passive design based on conduction, convection, gravity, and natural circulation.
- is expandable with up to six NPMs within the overall plant envelope.

While the majority of individual plant system designs are similar to traditional PWRs, a few systems vary from larger PWRs, such as the use of integral helical coil steam generators (SGs). In addition, there are some hard-coded parameters in the PWRGALE code that are not appropriate for the NPP design.

This technical report describes the methodology used to calculate normal operation, including anticipated operational occurrences (AOOs), radioactive annual gaseous and liquid effluents to the environment from an operating NPP containing up to six NPMs. This report also includes specific in-plant source terms and results of effluent releases. The application of this methodology demonstrates compliance with regulatory requirements, including a combination of site boundary isotopic concentrations and off-site dose consequence limits.

The methodology is realistic, yet conservative, using first principles-based calculations where appropriate, combined with recent nuclear industry experience where applicable, and lessons learned where available. Calculation of effluents uses conservative yet realistically generated source terms by evaluating radionuclide transport throughout reactor and other radioactive plant systems and by evaluating effluent releases. This technical report documents the appropriate primary and secondary coolant concentrations of fission products, activated corrosion and wear products, and water activation products for the NPP. Source terms also include water activation products produced in the reactor pool, which is a unique design feature.

One important input parameter in this methodology is the assumed fuel failure fraction. Industry operating experience over the past 30 years shows long-term and continuing reductions in fuel failures. Because the annual fuel failure fraction in U.S. PWRs continues to decrease over time with the most recent data {{ }}$^{2(a),(c)}$, showing a minimum value of {{ }}$^{2(a),(c)}$ and a maximum value of 66 rods per million (0.0066 percent), this analysis uses the maximum value of 66 rods per million. More than 90 percent of current U.S. nuclear power plants experience no fuel failures. The NPP design includes various design features that further mitigate fuel failure mechanisms. These design features further improve fuel performance. Based on the continued industry trend in fuel performance, the calculation of fission product related source term effluents uses a realistic yet conservative fuel failure fraction value.

# 1.0 Introduction

## 1.1 Purpose

This report describes the methodology used to calculate the NuScale Power Plant (NPP) gaseous and liquid effluents to the environment during normal operations, including anticipated operational occurrences (AOOs). This report describes a conservative NuScale design-specific, alternative method to NUREG-0017 (Reference 7.2.1).

## 1.2 Scope

The scope of this report includes the methodology and results of calculating normal gaseous and liquid effluent releases to the environment associated with a single US460, assuming the combined effect of up to six operating NuScale Power Modules (NPMs), considering AOOs. The report discusses the differences and similarities between the NUREG-0017 methodology and assumptions and the methodology. This report includes specific in-plant source terms and applies to all radioactive plant systems. Releases from these systems through intended (e.g., letdown or discharge) or unintended (e.g., leakage) events may result in an off-site release of radioisotopes; this report explains and quantifies these releases. This report also discusses the similarities and differences in the NPP compared to existing pressurized water reactor (PWR) designs as they relate to effluent releases.

This report does not include the calculation of site boundary radionuclide concentrations or doses to the public that result from the effluents. This report also does not include a discussion of the methodology used for the determination of personnel protection design features of the NPP. The methodology to characterize design basis events is out of scope for this technical report. This information and the supporting calculations are addressed in the NuScale Standard Design Approval Application.

## 1.3 Abbreviations

**Table 1-1 Abbreviations**

| Term | Definition |
|---|---|
| AOO | anticipated operational occurrence |
| ANS | American Nuclear Society |
| ANSI | American National Standards Institute |
| BONAMI | Bondarenko AMPX Interpolator (code) |
| CES | containment evacuation system |
| CENTRM | Continuous Energy Transport Module (code) |
| CNV | containment vessel |
| CRUD | corrosion and wear activation products |
| CVCS | chemical and volume control system |
| DF | decontamination factor |
| EPRI | Electric Power Research Institute |
| GALE | Gaseous and Liquid Effluents (NRC code implementing the methodology of NUREG-0017) |
| gpd | gallons per day |
| gpy | gallons per year |
| GRWS | gaseous radioactive waste system |
| HCW | high-conductivity waste |
| HEPA | high efficiency particulate air filter |
| HVAC | heating, ventilation and air conditioning |
| IAEA | International Atomic Energy Agency |
| LCW | low-conductivity waste |
| LRWS | liquid radioactive waste system |
| LWR | light water reactor |
| NEWT | New Extended Step Characteristic-based Weighting Transport (code) |
| NPM | NuScale Power Module |
| NPP | Nuscale Power Plant |
| NRC | U.S. Nuclear Regulatory Commission |
| OPUS | ORIGEN-S Post-Processing Utility for SCALE (code) |
| ORIGEN | Oak Ridge Isotope Generation (code) |
| ORIGEN-ARP | Oak Ridge Isotope Generation-Automatic Rapid Processing |
| ORIGEN-S | ORIGEN-SCALE code |
| PCA | primary coolant activity |
| PNNL | Pacific Northwest Nuclear Laboratory |
| PWR | pressurized water reactor |
| RBVS | Reactor Building HVAC system |
| RCS | reactor coolant system |
| RPV | reactor pressure vessel |
| RWB | Radioactive Waste Building |
| RXB | Reactor Building |
| SCALE | Standardized Computer Analyses for Licensing Evaluation (modular code) |
| SG | steam generator |
| TGB | Turbine Generator Building |
| TRITON | Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion (code) |

## 2.0 Background

The NPP design houses up to six NPMs in a Reactor Building (RXB). The heating ventilation and air conditioning (HVAC) systems gather and process airborne releases in the RXB and in the Radioactive Waste Building (RWB) before being released as effluents. The processing provided by the HVAC systems includes high-efficiency particulate air (HEPA) filters for particulates and charcoal filters for iodine removal associated with spent fuel pool releases. The RXB includes a separate, dedicated chemical and volume control system (CVCS) for each NPM for cleanup of primary coolant. There is also a common RWB located adjacent to the RXB that manages and processes radioactive waste for up to six NPMs. The condenser air ejector systems remove gaseous releases from the main condensers and also monitor and release gaseous releases from the main condensers via the TGB to the environment.

There are important differences in the NuScale Power Plant US460 design that influence effluent releases. The design is an integral PWR that includes the reactor core, pressurizer, and two helical coil steam generators (SGs), which leads to the potential of direct activation of the secondary coolant because of proximity of the SG to the reactor core. The primary coolant flow is solely natural circulation; a lower primary flow rate results in increased reactor coolant loop transit time and additional decay of activation products before they reach the secondary coolant. Also, each NPM consists of a reactor pressure vessel (RPV) surrounded by a high-pressure containment vessel (CNV), which is evacuated to a low pressure under normal operations. There are up to six NPMs per plant located in a large, common, below grade reactor pool. The RXB encloses the NPMs and reactor pool. Performance of refueling operations is underwater in the refueling and spent fuel areas of the common reactor pool. During this time, the primary coolant water within the NPMs (after being cleaned up post shutdown by the CVCS) mixes with water in the reactor pool.

## 2.1 GALE Code Applicability

The development of an alternate methodology is necessary because NUREG-0017, the existing PWRGALE-86 code (Reference 7.2.1), is from the 1980s for evaluation of the large PWRs of that time and does not appropriately address the NPP design. The NUREG-0017 methodology used empirical data from existing large reactors and is still the current U.S. Nuclear Regulatory Commission (NRC)-endorsed effluent release code. The US460 design

- is smaller. A single NPM provides approximately 8 percent of the electrical output of the example PWR in NUREG-017 (Reference 7.2.1). Table 2-1 provides a comparison between the NPM and the example PWR in NUREG-0017.
- relies upon a different passive design based on conduction, convection, gravity, and natural circulation.
- contains up to six NPMs within the common reactor pool, RXB envelope, and radioactive waste management system.

In an update to GALE in 2008, PWRGALE-08 incorporated equations and quantities from the American National Standards Institute/American Nuclear Society (ANSI/ANS) 18.1-1999 standard, “Radioactive Source Term for Normal Operation of Light Water Reactors” (Reference 7.2.2). The ANSI/ANS standard developed the calculation of radioactivity in the principal fluid streams of a light water reactor (LWR) based on historical data from the existing U.S. PWR fleet. Significant differences in NPP system parameters compared to a large PWR make direct scaling of most of this industry data an unsuitable extrapolation.

Another update to GALE in 2009, PWRGALE-09, incorporated a number of changes. The capacity factor was increased from 80 to 90 percent, although it was recognized in a 2012 Pacific Northwest National Laboratory (PNNL) report (Section 3.1.7, Reference 7.2.3) that this would still be too low for integral PWRs. The change in capacity factor, along with other hard-coded parameters in the GALE code, are not representative of the NPP design and cannot be changed as inputs. They could potentially be changed in the source code and recompiled, but recompiling would not address other applicability issues. Water activation product release rates decreased. As noted in a 2012 PNNL report (Section 3.1, Reference 7.2.3), “NRC staff expressed concern that there were certain limits of applicability on the parameters built into the GALE code.”

The PNNL report noted that there are five parameters that have narrow ranges of applicability to the empirical data. An attempt to adjust these parameters to better reflect the NPP design results in primary coolant concentrations outside the basis of the GALE code. These five parameter applicability ranges are also in Table 2-5 of NUREG-0017 Revision 1, along with one more parameter (steam flow) that represents the range of applicability for the secondary coolant system.

The latest update to the GALE code, GALE-PWR 3.2, was released in 2020. The GALE-PWR 3.2 Code comprehensively verified the applicability of the PWR-GALE Code from 1986, and added a graphical user interface. All of the inapplicability discussions above also apply to the GALE-PWR 3.2 Code as shown in Table 2-1.

**Table 2-1 (GALE) Applicability Range**

| Parameter | Units | GALE Applicability Range | NuScale Value |
|---|---|---|---|
| Thermal power | $MW_{th}$ | 3000 - 3800 | 250 |
| Primary coolant mass | lb | 500,000 - 600,000 | 100,000 |
| Primary system letdown flow | lb/hr | 32,000 - 42,000 | 10,800 nominal (20,160 maximum) |
| Shim bleed flow | lb/hr | 250 - 1,000 | 31 |
| Letdown cation demineralizer flow | lb/hr | ≤ 7,500 | 0 |
| Steam flow | lb/hr | 13,000,000 - 17,000,000 | 650,000 |

The NPP design is outside the range of these parameters, indicating that the GALE code is not appropriate for analysis of NPM coolant activity concentrations or effluents. This report uses values from NUREG-0017, where appropriate, and explains with justification why using alternatives to values that are not appropriate yields an acceptable level of safety.

## 2.2 Theory

Being unique and first-of-a-kind, NuScale does not rely on empirical effluent release data as the PWRGALE code does. The methodology separates effluents into three major phases:

- production (water activation, corrosion and wear activation products [CRUD], and fission products)
- transport (including removal mechanisms)
- release (liquid and airborne)

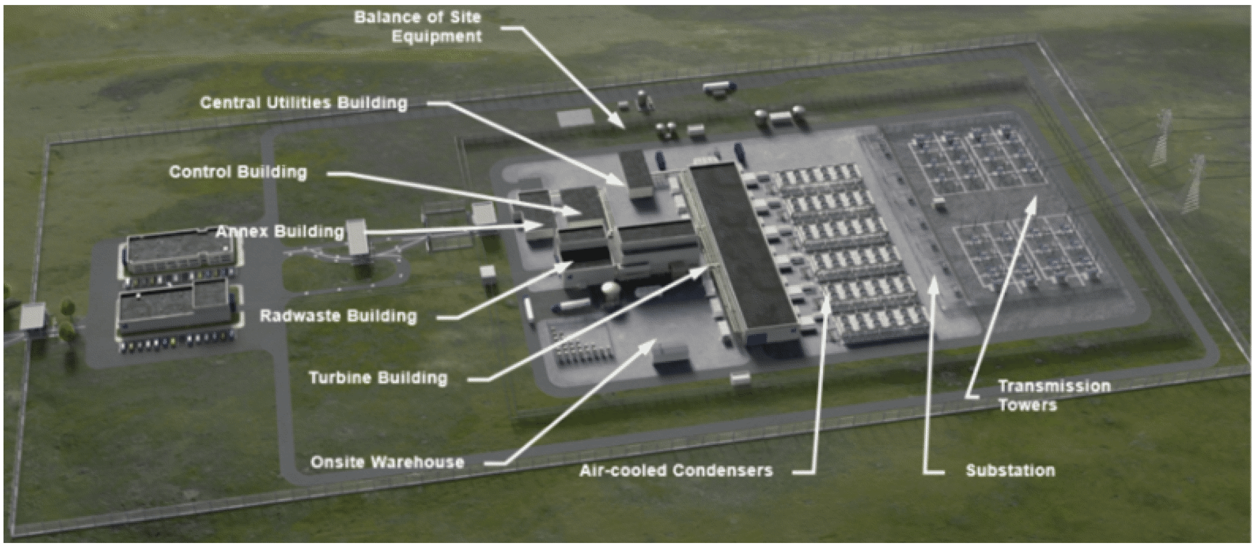
Production of radioactive isotopes (water activation, CRUD, and fission products) uses first-principles-based calculations where appropriate, combined with recent nuclear industry experience where applicable, and lessons learned where available, as appropriate in the development of source terms (Section 3.0). This process ensures the realistic yet conservative generation of source terms for further evaluation.

As mentioned in Section 2.1, the GALE code includes some hard-coded parameters that do not reflect the NPP design, such as the capacity factor. The method utilizes a higher, more conservative, and more appropriate capacity factor of 95 percent. The hard-coded radionuclide list in GALE omits a variety of nuclides, including environmentally mobile nuclides such as I-129 and Tc-99. The method uses a more comprehensive list of isotopics that carry forward throughout the evaluation of effluents. The isotopes reported in GALE (Reference 7.2.1) and ANSI/ANS-18.1-1999 (Reference 7.2.2), as well as the isotopes listed in the Design Control Document applications for the AP-1000 (Reference 7.2.9), U.S. EPR (Reference 7.2.10), US-APWR (Reference 7.2.11), and APR1400 (Reference 7.2.4) are the basis for the list of isotopics.

Calculations of radionuclide transport throughout the plant use guidance from NUREG-0017, especially with regard to the removal mechanisms appropriate to the system process and type of hardware. Unless there is a justified change, the methodology uses the assumed process parameters found in NUREG-0017 such as ion exchanger decontamination factors (DFs) in liquid process applications, and HVAC, HEPA, and charcoal iodine filtration efficiencies for particulates and iodines in airborne process applications. Although outside the scope of this technical report, the radioactive waste systems reduce radioactive effluent releases using similar processes and methods to those currently used at large PWRs, including filtration, resin absorption, liquid dilution, decay, and controlled liquid and gaseous releases.

The last phase of effluent evaluation is the release of radioactive materials from the plant site. The conservatively developed isotope activity levels, processed and reduced in quantity as appropriate, release to the environs as normal operations effluents. Figure 2-1 shows the general locations of effluent releases. Liquid effluents consolidate in the liquid radioactive waste system (LRWS) and discharge in a controlled fashion while being mixed with the utility water system as a dilution source. Airborne releases from the RXB and RWB combine to be released through one plant exhaust stack. Airborne releases from the TGB, constituting a small fraction of total effluents, release directly and undergo monitoring through the secondary systems.

**Figure 2-1 NuScale US460 Plant Layout**




## 2.3 Regulatory Requirements

Application of the methodology presented in this report provides a basis to ensure compliance with regulatory requirements. While site boundary concentrations and off-site dose calculations are outside the scope of this report, the radioactive effluent results presented in the Final Safety Analysis Report demonstrate compliance with 10 CFR 20 Appendix B, as well as with 10 CFR 20.1301-20.1302, “Radiation Dose Limits for Members of the Public,” (Reference 7.2.15) through site-specific, off-site dose calculations. In addition, effluent calculations demonstrate compliance with 10 CFR 50, Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low As Is Reasonably Achievable,’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents” (Reference 7.2.16).

## 3.0 Source Term Production

Production of source terms is the initial phase in determining plant radioactive effluents. There are three categories of radioactive isotopes generated as a result of reactor operations:

- water activation products (in waterborne elements).
- CRUD (corrosion and wear activation products).
- fission products (isotopes created in the fuel that migrate into the primary coolant).

## 3.1 Water Activation Products

The NPM containment vessel (CNV) is evacuated to a very low vacuum pressure (i.e., less than 1 psia) during operation (very little air surrounding the reactor vessel); therefore, air activation inside the CNV is insignificant. The CNV is partially immersed in the reactor pool, and there are several neutron activation reactions that can occur with stable isotopes in the primary coolant, secondary coolant, or reactor pool. These reactions produce activation products that can be a source of radioactive effluents. The evaluation of these activation products uses a first-principle physics model as shown in Equation 3-1:

$$RRx = \sum_{g=1}^{G} \Phi_g \sigma_{x,g} N = \sum_{g=1}^{G} \Phi_g \Sigma_{x.g} \qquad \text{Equation 3-1}$$

where,

$RRx$ = number of reactions of type "x,"

$\Phi_g$ = neutron flux in energy group "g,"

$G$ = maximum energy group,

$\sigma_{x,g}$ = microscopic cross-section for reaction "x" in energy group "g,"

$N$ = number density of target atoms, and

$\Sigma_{x.g} \equiv \sigma_{x,g} \cdot N$ = macroscopic cross-section for reaction x in energy group "g."

To provide some conservatism, this methodology assumes no depletion of target isotopes in the primary coolant. Benchmarks to industry data in these calculated production values shown below are only for information and comparison purposes; downstream calculations do not use them.

### 3.1.1 Tritium

Tritium is usually one of the major effluent release contributors for PWRs. Production of tritium in the primary coolant by fission neutron capture results in several different reactions. Of those, activation of soluble boron (Reference 7.2.12) produces the majority. In addition to the borated primary coolant, the analysis also evaluates the secondary coolant and borated reactor pool. Tritium production reactions are in Equation 3-2 through Equation 3-9.

$$ {}^{10}_{5}B + {}^{1}_{0}n_{f} \rightarrow 2\,{}^{4}_{2}\alpha + {}^{3}_{1}H $$ Equation 3-2

$$ {}^{10}_{5}B + {}^{1}_{0}n_{f} \rightarrow {}^{8}_{4}Be + {}^{3}_{1}H $$ Equation 3-3

$$ {}^{10}_{5}B + {}^{1}_{0}n_{th} \rightarrow {}^{4}_{2}\alpha + {}^{7}_{3}Li $$ Equation 3-4

$$ {}^{7}_{3}Li + {}^{1}_{0}n_{f} \rightarrow {}^{1}_{0}n + {}^{4}_{2}\alpha + {}^{3}_{1}H $$ Equation 3-5

$$ {}^{7}_{3}Li + {}^{1}_{0}n_{f} \rightarrow {}^{5}_{2}He + {}^{3}_{1}H $$ Equation 3-6

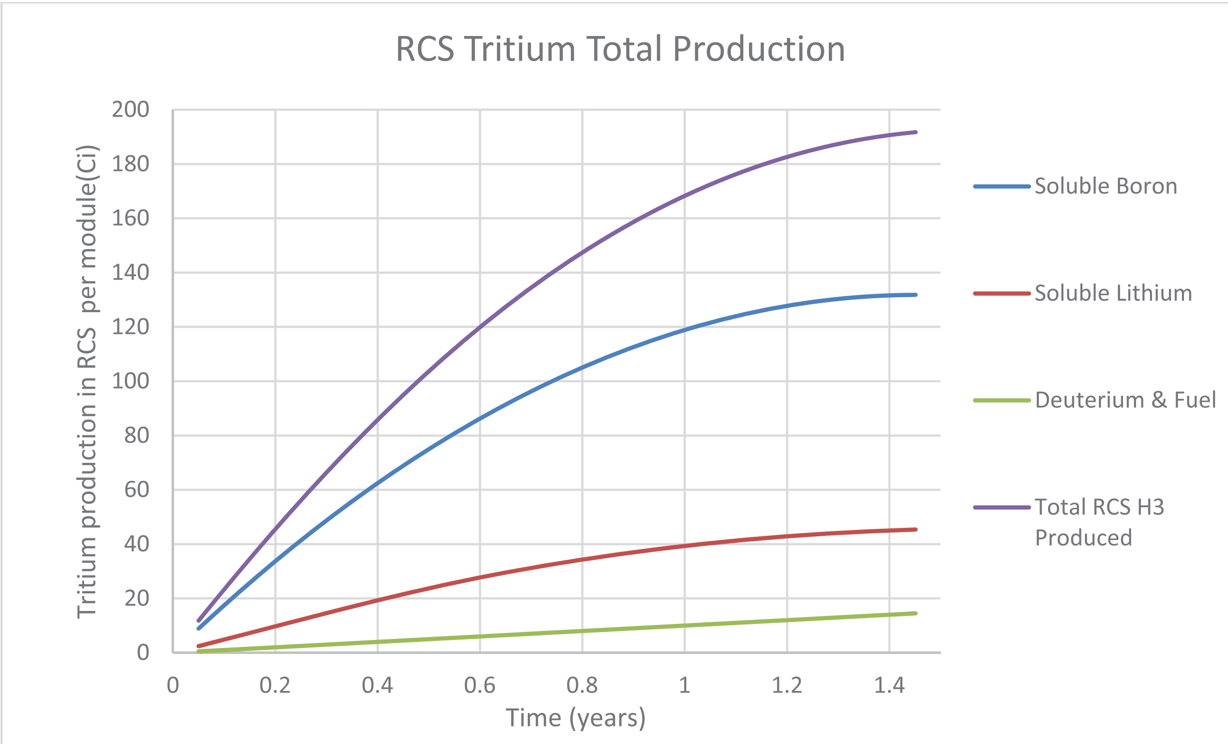
$$ {}^{11}_{5}B + {}^{1}_{0}n_{f} \rightarrow {}^{9}_{4}Be + {}^{3}_{1}H $$ Equation 3-7

$$ {}^{6}_{3}Li + {}^{1}_{0}n_{th} \rightarrow {}^{4}_{2}\alpha + {}^{3}_{1}H $$ Equation 3-8

$$ {}^{2}_{1}H + {}^{1}_{0}n_{th} \rightarrow {}^{0}_{0}\gamma + {}^{3}_{1}H $$ Equation 3-9

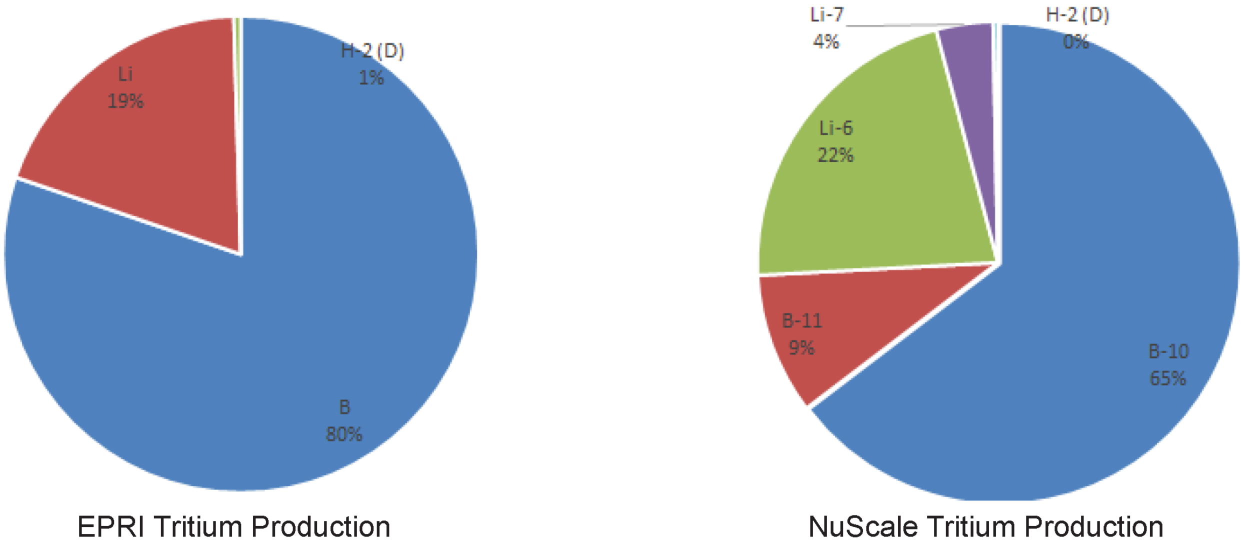
This calculation assumes the largest boron letdown curve calculated for any planned cycle to conservatively estimate the amount of tritium generated in the core. The analysis calculates tritium production based on all of the mechanisms in Equation 3-2 through Equation 3-9, and the production over an 18-month operating cycle is in Figure 3-1. There is an investigation of the buildup of deuterium, which offers a negligible contribution to the overall tritium production or concentrations.

**Figure 3-1 Time Dependent NuScale Isotopic Tritium Production Breakdown in Primary Coolant**




The calculated tritium production from soluble species (boron, lithium, and deuterium) is 120 Ci/yr per NPM in the primary coolant, which is higher than the Electric Power Research Institute (EPRI) example plant value of 78 Ci/yr per NPM (Reference 7.2.12). The design also includes more water in the coolant per megawatt generated than a standard PWR. Combined with its higher capacity factor, the design has a substantial neutron flux for a longer period of time, in a larger relative amount of coolant, than a typical PWR. This results in more tritium production reactions with the coolant soluble species. Figure 3-2 shows a comparison between the relative contribution of the production from soluble species and the calculated values for a NPP. The relative difference is due to starting with a higher lithium concentration than in a typical PWR, to maximize the pH for minimization of CRUD production.

**Figure 3-2 Total NuScale Isotopic Tritium Production Breakdown in Primary Coolant**

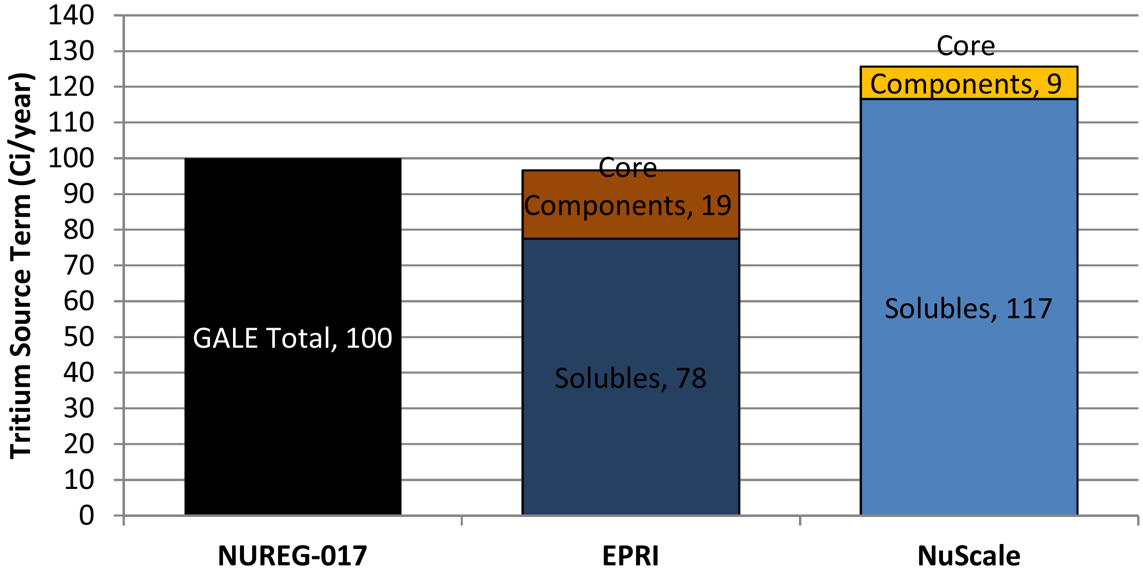



Ternary fission of uranium-235 also produces tritium. Only a small fraction of the total tritium produced in the fuel diffuses through the cladding into the coolant. The EPRI tritium management model (Reference 7.2.12, Table 7-2) provides primary coolant tritium production values from fission. Reference 7.2.12 also provides tritium generation values for an example plant. Scaling the tritium production rate for the NPP power output provides an estimate of 9.1 Ci/yr per NPM coming from within the core components, such as fuel pins. Because of the low neutron flux, there is negligible direct tritium production through activation in the reactor pool and secondary coolant. Therefore, the total tritium production is 126 (117 + 9) Ci/yr per NPM compared to the EPRI example plant value of 97 (78 + 19) Ci/yr per reactor prediction.

Tritium is a mobile radionuclide because it is chemically the same as protium (hydrogen with an atomic weight of one) and bonds with water, typically as $H_2O$. Filtering does not remove it from the water, so it has a DF of one for cleanup systems. Tritium emits a beta particle with a half-life of 12.32 years. Therefore, it decays very little before release. Once the analysis generates a tritium source term, tritium transports throughout the plant systems until being released through both liquid and gaseous pathways. The total assumed release rate of tritium is approximately equal to its production rate.

Section 2.2.17.1 of NUREG-0017 lists a total value for tritium effluent release rates of $0.4 Ci/yr/MW_{th}$. For a NuScale 250 $MW_{th}$ reactor, that equals 100 Ci/yr per module. A comparison of the NUREG-0017, EPRI, and NuScale values is in Figure 3-3.

**Figure 3-3 Comparison of GALE, Electric Power Research Institute, and NuScale Yearly Tritium Production**




### 3.1.2 Carbon-14

The reactor coolant during power operation produces carbon-14 (or radiocarbon). The primary coolant, secondary coolant, and reactor pool produces carbon-14, taking several possible chemical forms. The chemistry of carbon-14 is complex, and there are only two significant production reactions involving isotopes dissolved in water in LWRs (including the NPP). These two reactions are in Equation 3-10 and Equation 3-11.

$$ {}^{17}_{8}\mathrm{O} + {}^{1}_{0}\mathrm{n}_{\mathrm{th}} \rightarrow {}^{4}_{2}\alpha + {}^{14}_{6}\mathrm{C} \qquad \text{Equation 3-10} $$

$$ {}^{14}_{7}\mathrm{N} + {}^{1}_{0}\mathrm{n}_{\mathrm{th}} \rightarrow {}^{1}_{1}\mathrm{p} + {}^{14}_{6}\mathrm{C} \qquad \text{Equation 3-11} $$

Nitrogen is both an impurity in the fuel or other core materials and dissolved in water as a gas or as a chemical compound (e.g., ammonia or hydrazine). The calculated potential production of carbon-14 from the two reactions in all three water sources is negligible in the pool and secondary coolant system due to the small neutron fluxes.

Carbon-14 is pervasive in PWR systems, similar to tritium, and any location or system that contains tritium likely also contains carbon-14. Carbon-14 beta decays with a half-life of 5,700 years, making decay negligible. Carbon-14 has multiple chemical forms with different properties (affecting removal DFs, partition factors, and others), such that carbon-14 is typically a component of both liquid and gaseous effluents.

Section 2.2.25 of NUREG-0017 lists values of carbon-14 effluent release rates that vary between 0.58 Ci/yr and 46 Ci/yr with an average of 7.3 Ci/yr. With the NPP's much lower power production, smaller core volume, and smaller active fuel region, it produces less carbon-14, and the NPP is well below the carbon-14 average releases for large PWRs. Based on first-principle physics, the calculated carbon-14 production in the primary coolant is 1.2 Ci/yr per module, a fraction of the total yearly average effluent release of radionuclides.

### 3.1.3 Nitrogen-16

Oxygen-16 (99.76 percent of naturally occurring oxygen) in water can activate to form radioactive nitrogen (nitrogen-16). Nitrogen-16 is produced by neutron activation of oxygen via Equation 3-12.

$$ {}^{16}_{8}\mathrm{O} + {}^{1}_{0}\mathrm{n}_{f} \rightarrow {}^{1}_{1}\mathrm{p} + {}^{16}_{7}\mathrm{N} \qquad \text{Equation 3-12} $$

The nitrogen-16 atoms combine with oxygen and hydrogen in the coolant to form ions or compounds such as NO, $NO_2$, $NO_3$, $N_2$, and $NH_4$. Nitrogen-16 has a high formation rate and a short half-life of 7.13 seconds. Nitrogen-16 emits high-energy gamma rays (6.13MeV and 7.12 MeV).

Nitrogen-16 activity is high in the primary coolant in and near the active core; however, due to its short half-life, longer transit times through various plant systems, and off-site receptors, nitrogen-16 is not a significant contributor to radiation exposure beyond the primary coolant system and is, therefore, not a significant contributor to effluents. That is why NUREG-0017 Revision 1, Section 1.5.2.12.2 states that the GALE code does not consider nitrogen-16 as an effluent. Transit times are longer in the NPM than traditional large PWRs because of the slower natural circulation primary flow. The total reactor coolant system (RCS) loop transit time is approximately 46 seconds, which is more than six half-lives of nitrogen-16, which prevents buildup in the core. Section 4.1.1 calculates and discusses the nitrogen-16 concentration at various locations (e.g., at the bottom of the helical coil SG) within the RCS loop.

### 3.1.4 Argon-41

Neutron activation of argon-40 produces argon-41. Argon-40 is naturally found in air. The amount of argon in air is 0.934 percent (Reference 7.2.13 Table E-1), and Equation 3-13 shows the production of argon-41.

$$ {}^{40}_{18}\mathrm{Ar} + {}^{1}_{0}\mathrm{n}_{th} \rightarrow {}^{0}_{0}\gamma + {}^{41}_{18}\mathrm{Ar} \qquad \text{Equation 3-13} $$

Radioactive argon-41 is an inert gas that transforms into a stable isotope of potassium (potassium-41) through a relatively complex set of decay emissions. Argon-41 decay primarily produces both a 1.2 MeV beta particle and a 1.3 MeV gamma ray with a half-life of approximately 110 minutes (Reference 7.2.13 Table 5-1, B-1, and E-2).

In existing large PWRs, the activation of natural argon-40 in the air within the containment that surrounds the reactor vessel dominates production of argon-41. Primary and secondary coolant streams purge air before operating the plant, making production of argon in the coolant streams negligible in large PWRs. In the NPP design, there is little air surrounding the reactor vessel because the steel CNV, maintained at a low pressure (less than 1 psia) during power operation, surrounds the reactor vessel. Argon-40 is negligible inside containment. As a result, the main contributor of argon-41 effluent release from production is activation of argon-40 contained in air that dissolves in the water of the reactor pool surrounding the NPMs.

Section 2.2.26 of NUREG-0017 lists values of argon-41 effluent release rates that vary between 0.02 Ci/yr and 208 Ci/yr with an average of 34 Ci/yr. With the NPP's lower power production, smaller fluxes, and the NPM submerged in water instead of air, there is substantially less argon-41 produced outside of the NPM, and the NPP is well below the argon-41 average releases for large PWRs.

The primary coolant can contain argon-40 as a tracer for leaks through the helical coil SG into the secondary side. If the tracing is performed, it achieves a desired argon-41 activity concentration in the primary coolant of 0.1 μCi/ml (Reference 7.2.23). This analysis assumes argon-40 addition.

## 3.2 Corrosion and Wear Activation Products

### 3.2.1 Mechanism Overview

The activated corrosion and wear products form as a result of oxidation and wear of the materials of construction in the primary reactor coolant circuit that come in contact with the reactor coolant and activate by neutron interactions. When exposure of these alloys to the primary reactor coolant occurs at high temperature, oxygen diffuses into the base metal at the wetted surface and converts the elements in the alloy from the metallic state to an oxide state. In the process, divalent metal ions release into water as soluble metal ions (Reference 7.2.21). Thus, a protective layer of corrosion products forms on the surface of an alloy, which separates it from the coolant. The ion conductivity of this layer is low; however, mass transfer still exists between the metal alloy and the primary coolant (Reference 7.2.22).

The activated corrosion and wear products can manifest itself in a solid phase, either as metal oxide films or as micrometer-sized particles of metal oxide (Reference 7.2.21). It can also exist as hydrolyzed species of metal oxides in the aqueous phase. Introduced species resulting from metallic corrosion are in the coolant, where they transport through convection onto other surfaces (Reference 7.2.22), including the surface of the fuel and the surface of in-core structure materials. Thus, they transform into radioactive nuclides in the neutron flux, meaning that they activate. Neutron activation is possible when metal oxide species travel in the reactor core region or when they deposit on in-core surfaces.

The activated corrosion products release from fuel surface deposits by erosion and spalling caused by hydraulic shear forces or dissolution. Some activated products release from in-core materials by dissolution and wear. They then transport by water to all parts of the primary system, where they can deposit on surfaces by the following mechanisms: turbulent diffusion, Brownian diffusion, inertial impaction, sedimentation, and thermophoresis. The production, transportation, solubility, and deposition have many complicated mechanisms. These include pH, temperature, materials of construction, flow rates and regimes, surface conditions, and chemistry. This complexity prohibits first-principle physics models of CRUD.

### 3.2.2 Modeling Corrosion and Wear Activation Products

There are developed models for the estimation of radioactivity buildup and corrosion product transport in LWRs. These include empirical and semi-empirical models containing coefficients that derive from experimental data or plant design data. Some examples include: Japanese ACE, Korean CRUDTRAN, Czech DISER, Bulgarian MIGA, and French PACTOLE (Reference 7.2.22). These models use empirical data from the specific reactors whose behavior they model. For this reason, they are not applicable for reactors with different designs and geometries. In particular, the NPM has some characteristics that make it fundamentally different from other PWRs; therefore, none of the available models accurately describes the NPM behavior with regard to the activated corrosion products' transport and deposition. Differences in the NPM design and the existing fleet preclude the use of these reactor-specific models.

Because there are no models available for the generation and transportation of corrosion and wear activation products, the model uses conservative empirical data. The ANSI/ANS-18.1-1999 standard provides a basis for determining the concentrations of radionuclides in the primary and secondary coolant of a nuclear power plant. Therefore, the standard calculates those values directly rather than calculating a production rate. This standard is for the purposes of calculating, through adjustment factors, radionuclide concentrations in support of the design and licensing process. The data contained in ANS18.1 is based on actual historical large PWR plant measurements, from a time when CRUD production was much higher in the industry. As such, it is a suitable and conservative standard for calculating anticipated corrosion and wear activation products in the primary coolant for the design. The calculated CRUD source term numbers in the primary are in Table 3-1.

**Table 3-1 CRUD Isotopic Primary Concentrations**

| Isotope | Primary Coolant Concentration (μCi/g) |
|---|---|
| Na24 | 1.4E-02 |
| Cr51 | 7.7E-04 |
| Mn54 | 4.0E-04 |
| Fe55 | 3.0E-04 |
| Fe59 | 7.5E-05 |
| Co58 | 1.1E-03 |
| Co60 | 1.3E-04 |
| Ni-63 | 6.6E-05 |
| Zn65 | 1.3E-04 |
| Zr-95 | 9.7E-05 |
| Ag-110m | 3.2E-04 |
| W187 | 7.0E-04 |

Incorporating lessons learned from the industry decreased CRUD production over time. The design follows modern guidelines for the reduction of CRUD and employs design features that minimize CRUD production. The reactor uses the lowest possible cobalt and nickel materials appropriate for design conditions, along with lessons learned about RCS chemistry control (e.g., highest pH). As a result, values derived from the ANS standard for the NPP are conservative.

Additionally, the RPV and the CNV are stainless steel, which is designed to survive the life of the plant in the borated water chemistry. As a result, the vessels should have minimal corrosion activation products.

## 3.3 Fission Products

The industry standard, Standardized Computer Analyses for Licensing Evaluation (SCALE), computer code develops spent fuel isotopic distribution and magnitude. To ensure conservative results, the methodology assumes a maximum peak burnup of 62 GWd/MtU for all fuel rods in the core. The fuel isotopic inventory per assembly at that burnup is in Table A-2 of Appendix A.

### 3.3.1 Software Use and Qualification

To further support the use of a first principles approach in the methodology, the SCALE 6.1 modular code package, developed by Oak Ridge National Laboratory, develops reactor core and primary coolant fission product source terms. Specifically, the Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion (TRITON) and Oak Ridge Isotope Generation - Automatic Rapid Processing (ORIGEN-ARP) analysis sequences of the SCALE 6.1 modular code package, and ORIGEN-SCALE code (ORIGEN-S), run as a standalone module, generate radiation source terms for the fuel assemblies and various waste streams (Reference 7.2.14).

This industry standard commercial off-the-shelf software is used without modification by the methodology. This software has been extensively used in the evaluation of operating large LWRs. NuScale's Software Configuration Management Plan directs the use of the SCALE code package. The SCALE code is in compliance with ASME NQA-1 2008/2009A through the NuScale commercial grade dedication process.

### 3.3.2 TRITON Code Sequence

The TRITON sequence of the SCALE code package is a multipurpose control module for nuclide transport and depletion, including sensitivity and uncertainty analysis. TRITON generates problem-dependent and burnup-dependent cross-sections. It performs multi-group transport calculations in one-dimensional, two-dimensional, or three-dimensional geometries. The ability of TRITON to model complex fuel assembly designs improves transport modeling accuracy in problems that have a spatial dependence on the neutron flux. In this case, TRITON generates burnup-dependent cross-sections for fuel assemblies for subsequent use in the ORIGEN-ARP depletion module.

The T-DEPL (time-depletion) sequence of the TRITON control module generates problem-dependent (i.e., assembly-specific) and burnup-dependent cross-sections. The model uses the Continuous Energy Transport Module (CENTRM)-based option of the T-DEPL sequence in which the Bondarenko AMPX Interpolator (BONAMI) processes microscopic cross-sections for the unresolved resonance energy range. The CENTRM code processes cross-sections from the continuous-energy library for the resolved resonance energy range. The CENTRM code uses a one-dimensional discrete ordinates calculation to generate point-wise fluxes, properly taking into account overlapping resonances from different isotopes. The multi-group cross-section module creates a problem-dependent multi-group library for the resolved resonance energy range using the weighting spectrum from CENTRM and combines it with the multi-group library processed by BONAMI. The Code to Read and Write Data for Discretized (CRAWDAD) solution and WORKER modules properly format the cross-section libraries at different stages of the processing.

The New Extended Step Characteristic-based Weighting Transport (NEWT) code module performs a two-dimensional, discrete ordinates transport calculation. The NEWT code post-processes the results of the transport calculation to generate region-averaged multi-group cross-sections and fluxes for each depletion material. The COUPLE module essentially couples NEWT and ORIGEN-S by collapsing the multi-group cross-sections into a one-group cross-section library for each depletion material using the fluxes from NEWT. The COUPLE module then combines the one-group cross-section library with decay data and energy-dependent fission product yields to produce a binary-formatted ORIGEN-S nuclear data library. Finally, ORIGEN-S depletes each material using the normalized material power and the problem- and burnup-dependent nuclear data library. Decay intervals between depletion steps are also modeled by ORIGEN-S. The TRITON code models the complete depletion sequence by repeating the cross-section processing, transport calculations, depletion, and decay calculations for a user-specified series of depletion and decay intervals, using a predictor-corrector algorithm. The TRITON code saves each problem-dependent and burnup-dependent nuclear data library for future use with ORIGEN-ARP. After the final depletion step, TRITON can call the ORIGEN-S post-processing utility for SCALE (OPUS) module to post-process the ORIGEN-S time-dependent isotopic concentrations, producing an ASCII-formatted file of isotopic concentrations or source spectra for further analysis or plotting.

### 3.3.3 ORIGEN (ORIGEN-ARP and ORIGEN-S) Code Sequences

The ORIGEN-ARP code is a SCALE depletion analysis sequence used to perform point-depletion and decay calculations with the ORIGEN-S module using problem-dependent and burnup-dependent cross-sections. The ORIGEN-ARP module prepares ORIGEN-S nuclear data libraries containing these cross-sections using interpolation in enrichment and burnup between pre-generated nuclear data libraries containing cross-section data that span the desired range of fuel properties and operating conditions. The ORIGEN-ARP sequence produces calculations with accuracy comparable to that of the TRITON sequence with a great savings in problem setup and computational time compared to repeated use of TRITON. There are many possible modeling variations in fuel assembly irradiation history. For depletion calculations involving fuel assemblies, the TRITON sequence generates ORIGEN-S nuclear data libraries, as described.

The ORIGEN-S module of SCALE 6.1 calculates the time-dependent isotopic concentrations of materials in a fuel assembly by modeling the fission, transmutation, and radioactive decay of fuel isotopes, fission products, and activation products in the assembly. The ORIGEN-ARP module sets up the input data for ORIGEN-S, ensuring use of the proper nuclear data library for each depletion or decay interval of the fuel assembly irradiation history.

# 4.0 Radionuclide Transport, Removal Mechanisms, and Release

Transportation of radionuclides within the plant throughout the various systems, and selective removal of isotopes based on processing capabilities, is the second phase in determining plant radioactive effluents. Release of processed radionuclides into the environs through either liquid or gaseous effluent pathways, is the third phase (Section 4.8).

## 4.1 Primary Coolant Water System

Section 3.0 discusses the source term inputs to the primary coolant. The three inputs to the primary coolant are direct neutron activation in the water, CRUD, and fission products that leak and diffuse from failed or damaged fuel.

### 4.1.1 Water Activation Products

Because tritium cannot be removed from the primary coolant water, it does not reach an equilibrium value over a cycle during operation. Because the design facilitates recycling of primary water, three recycling modes calculate the tritium concentration in process streams: 1) no recycling of the primary coolant; 2) recycling of the primary coolant to the reactor pool; and 3) recycling of the primary coolant back to the CVCS as makeup. The first mode (no recycling) maximizes the tritium concentration in the liquid discharge effluent stream. Therefore, the liquid effluent calculation uses the letdown tritium concentration from no recycling.

The production rate (Figure 3-1), along with the cumulative water injection and bleed out of the primary coolant (Figure 4-1), develops a time-dependent balance of how much tritium is in the coolant versus how much has bled out of the coolant (Figure 4-2). The removal of primary coolant to control boron levels in the reactor and, subsequently, reactivity control in the core, forms the basis for letdown removal. Primary coolant is let down from the reactor to the LRWS via the CVCS.

**Figure 4-1 Water Injection and Bleed in the Primary Coolant**

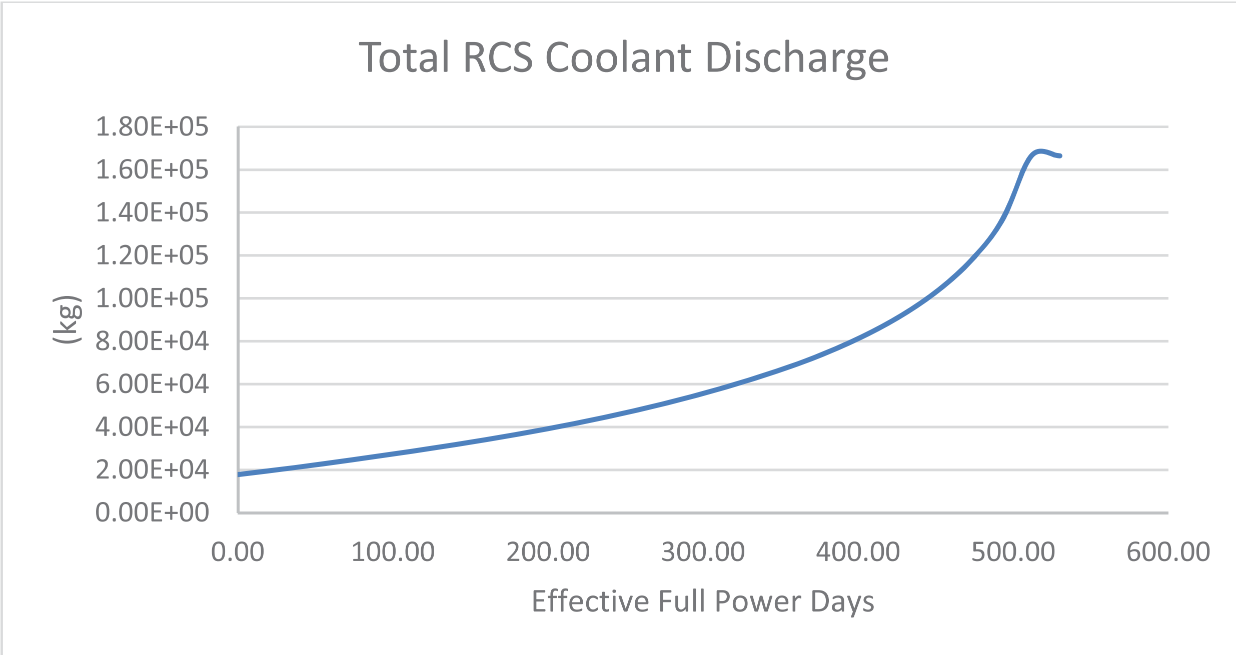
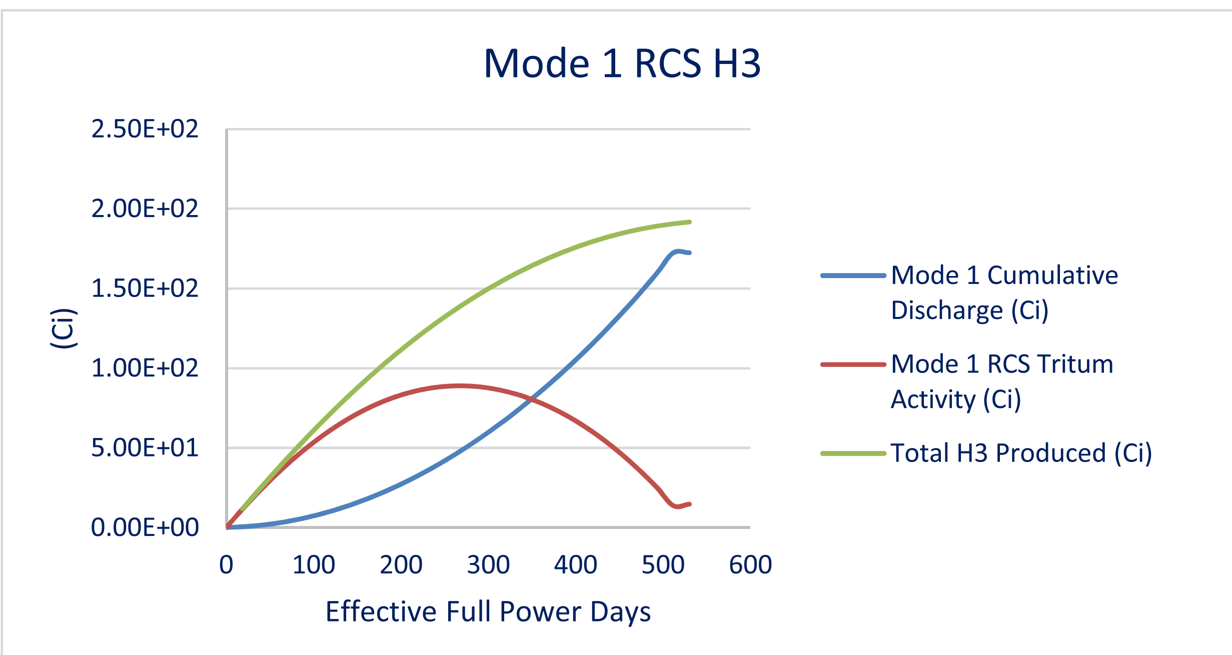



**Figure 4-2 Tritium Reactor Coolant System Balance**



The tritium inventory curve transforms into a concentration and the time weighted average to determine the average tritium concentration in the primary coolant. For comparison, Section 2.2.17.1 of NUREG-0017 lists an average tritium primary coolant concentration in PWRs of 1.0 µCi/ml. For normal operations with primary letdown, NuScale calculates an average concentration of 1.3 µCi/ml. In addition, the average concentration of primary coolant being let down from the RCS is 1.0 µCi/ml. Mode 1 (no recycling of primary coolant) forms the basis of these tritium concentrations.

NUREG-0017 does not list carbon-14 in the primary or secondary coolant, although it is included as a small contributor to the effluent.

Table 2-2 of NUREG-0017 states that there is a nitrogen-16 primary concentration of 40 µCi/ml at the SG on the primary loop, where N-16 could leak into the secondary coolant.

With natural circulation in the NPM core, the coolant flow rate is slow enough that nitrogen-16 has a substantial amount of decay during its transit time through the primary system. Using the maximum full power coolant flow rate of 821 kg/s, the minimum, full power, total RCS transit time is approximately 46 seconds, more than six half-lives of nitrogen-16. Therefore, by the time the nitrogen-16 transits to the integral helical coil SG in 11 seconds, its concentration is 41 µCi/g, which is close to the NUREG-0017 value. Further, the nitrogen-16 concentration at the CVCS inlet with a transit time of 26 seconds is much smaller than the helical coil SG at 10 µCi/g.

The primary coolant can contain argon-40 for use as a tracer for SG leaks. This analysis assumes argon-40 addition to reach target argon-41 levels in the primary coolant of 0.1 µCi/ml (Reference 7.2.23).

### 4.1.2 CRUD

As discussed in Section 3.2, CRUD is evaluated in terms of primary coolant concentrations.

### 4.1.3 Fission Products

Fission product leakage into the primary coolant from the previously calculated fuel inventory uses a realistic yet conservative fuel failure fraction of 66 rods per million (discussed in Section 5.0) along with typical industry fission product isotopic escape coefficients (Reference 7.2.9, Reference 7.2.10, Reference 7.2.11), as shown in Table 4-1. These values are also conservative for the NPP design because escape rate coefficients are a function of linear heat generation rate and the NPM has a lower linear heat generation rate than larger PWRs.

**Table 4-1 Fuel Isotopic Escape Coefficients**

| Isotope | Value ($s^{-1}$) |
|---|---|
| Kr | 6.5E-8 |

**Table 4-1 Fuel Isotopic Escape Coefficients (Continued)**

| Isotope | Value ($s^{-1}$) |
|---|---|
| Xe | 6.5E-8 |
| Br | 1.3E-8 |
| Rb | 1.3E-8 |
| I | 1.3E-8 |
| Cs | 1.3E-8 |
| Mo | 2.0E-9 |
| Tc | 2.0E-9 |
| Ag | 2.0E-9 |
| Te | 1.0E-9 |
| Sr | 1.0E-11 |
| Ba | 1.0E-11 |
| Y | 1.6E-12 |
| Zr | 1.6E-12 |
| Nb | 1.6E-12 |
| Ru | 1.6E-12 |
| Rh | 1.6E-12 |
| La | 1.6E-12 |
| Ce | 1.6E-12 |
| Pr | 1.6E-12 |
| Np | 1.6E-12 |
| Sb | 1.6E-12 |
| P | 1.6E-12 |

Table A-1 lists the maximum fuel inventory per assembly.

### 4.1.4 Primary Coolant Activity Concentrations

The primary coolant activity also includes the build-in of radioactive daughter products from the decay process. The equilibrium concentration of radionuclides in the primary coolant assumes a homogenized mixture of radionuclides throughout the entire water volume with the exception of nitrogen-16, as previously described.

The NPP's primary water volume-to-fuel ratio is much higher than a typical large PWR. Even assuming a proportional source term, this results in a lower concentration in the primary water due to greater dilution in the larger RCS volume.

The removal mechanisms of most of the radionuclides from the primary system are radioactive decay, purification (CVCS demineralizers) and letdown to the LRWS. The DFs for the mixed-bed demineralizers are 100 for halogens, 2 for cesium and rubidium, and 50 for other isotopes, per Section 2.2.18.1 of NUREG-0017. There is no specific degasification of the primary coolant, thus neglecting noble gas removal through the pressurizer is appropriate.

Although the concentration of individual isotopes in the primary coolant varies considerably over the operating cycle, the conservative assumption is that maximum calculated equilibrium activity is present for the entire operating cycle, with the exception of some of the water activation products, which are treated separately. Equation 4-1 calculates the activity of the isotopes:

$$A_{cp} = \frac{A_{sp}}{(\lambda_p + \lambda_L + \lambda_U)} \qquad \text{Equation 4-1}$$

where,

$A_{cp}$ = activity of parent isotopes in the primary coolant,

$A_{sp}$ = activity generation rate of the source term parent isotopes,

$\lambda_p$ = decay constant of the parent nuclide,

$\lambda_L$ = letdown removal coefficient through LRWS degasifiers, and

$\lambda_U$ = removal coefficient for purification.

Equation 4-2 calculates the activity of the ingrowth of daughter product isotopes:

$$A_{cd} = \frac{A_{cp}\lambda_d f_p}{(\lambda_d + \lambda_L + \lambda_U)} \qquad \text{Equation 4-2}$$

where,

$\lambda_d$ = decay constant of the daughter nuclide, and

$f_p$ = branching fraction for the parent nuclide(s) that decay to the daughter isotope.

The list of radionuclide activity concentrations in the primary coolant is in Table A-2 in Appendix A.

## 4.2 Secondary Coolant Water System

Primary-to-secondary leakage determines the concentration of radionuclides in the secondary system. The Electric Power Research Institute (Reference 7.2.13) evaluated primary-to-secondary leakage in the industry and developed SG management guidelines, which NuScale follows. As operational experience with the NuScale helical coil SGs accumulates, modifications to EPRI guidelines may occur to optimize the mitigation of potential leakage. The direct activation of the secondary water impurities is negligible due to the small flux at the bottom of the helical coil SGs, which is closest to the active core.

The flux at the bottom of the helical coil SGs is several orders of magnitude less than the average active core flux.

The methodology underestimates the total secondary coolant mass as 5.0E4 lbm for the effluent models. The NPP value for secondary coolant mass equates to the sum of water mass estimates for the various main components of the secondary system (including both helical coil SGs and other components), and neglects the contribution of the fluids in the turbine and condenser as well as condensate polishing systems, structures, and components. This smaller mass is conservative because it overestimates the radionuclide concentrations.

One secondary side removal mechanism is cleanup through the demineralizers that have DFs of 100 for halogens, 10 for cesium and rubidium, and 100 for other isotopes, per Section 2.2.18.1 of NUREG-0017. Other secondary side removal mechanisms are liquid and gaseous leakage to the TGB (assumed to be upstream of the condensate polishers for conservatism), condensate air removal, and the turbine gland seal steam. The scaled leakage terms from the secondary system are from values provided in NUREG-0017 based on the low power level of each NPM (250 $MW_{th}$) compared to a traditional large PWR with an assumed power level of 3400 $MW_{th}$. Power scaling is appropriate because system capabilities scale to the size of the reactor. Main steam production is approximately proportional to core thermal power. Also, component sizing (e.g., pipe diameter) relates to core thermal power. This approach results in larger, more conservative values for the secondary coolant radionuclide concentrations.

The secondary coolant sampling system drain rate, TGB floor drain rate, and steam leakage rate to the TGB are NUREG-0017 values linearly scaled to the power output of a NuScale core (250 $MW_{th}$) from the nominal power output of a standard PWR (3400 $MW_{th}$), as shown in Table 4-2.

**Table 4-2 NUREG-0017 and Corresponding NuScale Parameters**

| Parameter | NUREG-0017 | NuScale Module |
|---|---|---|
| Primary-to-secondary leak rate (lb/day/NPM) | 75 | 5.5 |
| CVCS to RXB leak rate (lb/day/NPM) | 160 | 11.8 |
| TGB floor drains (gal/day/NPM) | 7200 | 9.9 |
| Secondary coolant sampling system drains (gal/day/NPM) | 1400 | 1.9 |
| Steam leak rate in TGB (lb/hr/NPM) | 1700 | 125 |

The means to determine concentration of most of the radionuclides in the secondary coolant is similar to that of the primary coolant, because it shares the same basic governing equation. The main difference is that the production term for the secondary coolant is just the leakage of radionuclides from the primary into the secondary, given by Equation 4-3:

$$P_s = A_p \times L_{PS} \qquad \text{Equation 4-3}$$

where,

$P_s$ = production rate in the secondary coolant,

$A_p$ = equilibrium activity of a radionuclide in the primary coolant, and

$L_{PS}$ = leak rate of coolant from the primary to the secondary.

This calculation leads to an equilibrium activity in the secondary coolant that is similar to Equation 4-1. The equation that models the secondary activity is in Equation 4-4:

$$A_s = \frac{C_p \times L_{PS}}{\lambda_d + \lambda_U} \qquad \text{Equation 4-4}$$

where,

$A_s$ = equilibrium activity in the secondary coolant,

$C_p$ = equilibrium concentration in the primary coolant,

$L_{PS}$ = leak rate from the primary to the secondary,

$\lambda_d$ = decay constant for the radionuclide, and

$\lambda_U$ = cleanup constant for the radionuclide.

The concentration of radionuclides in the secondary coolant is the calculated secondary activity divided by the total mass of secondary coolant.

Because noble gases are not chemically reactive, cleanup systems do not generally remove noble gases from the coolant. Noble gases leave the secondary coolant quickly through gaseous removal mechanisms (primarily the condenser air removal system). Multiplying the concentration of the noble gas in the primary coolant by the primary-to-secondary leak rate and then dividing by the sum of the secondary flow rate and primary-to-secondary leak rate calculates the concentration of noble gases in the secondary coolant using Equation 4-5.

$$C_{Secondary} = C_{Primary} \times \left( \frac{L_{PS}}{L_{PS} + \dot{m}_{secondary}} \right) \qquad \text{Equation 4-5}$$

Tritium, as an isotope of hydrogen, is chemically identical to hydrogen, preventing typical methods of cleanup from working on tritium, resulting in two important consequences.

The first is that without cleanup or any other removal mechanism, the secondary coolant concentration of tritium reaches the same value as the primary coolant concentration. This modeling approach is not a reasonable approximation due to removal of tritium through leakage and decay. The second consequence is that tritium does not buildup in the cleanup systems. Therefore, tritium does not impact any shielding calculations for these systems because it is a weak beta emitter. The calculations in this document account for the eventual effluent release of tritium by considering the leakage rate of coolant out of the secondary system. The secondary coolant concentration is in Equation 4-6:

$$A_{Secondary} = C_{Primary} \times \left(\frac{L_{PS}}{\lambda_d + \lambda_L}\right) \qquad \text{Equation 4-6}$$

where,

$A_{Secondary}$ = activity of tritium in the secondary,

$C_{Primary}$ = concentration in the primary,

$L_{PS}$ = leak rate from primary to secondary,

$\lambda_d$ = decay constant for tritium, and

$\lambda_L$ = leakage removal constant.

The total tritium concentration is the total tritium activity divided by the total mass of secondary coolant. For comparison, Table 2-3 of NUREG-0017 lists a tritium secondary coolant concentration of 1.0E-03 µCi/ml. The calculation determined a tritium activity concentration in the secondary coolant of 2.5E-03 µCi/ml. This tritium concentration relates to Mode 2, recycle of RCS to the pool.

A comprehensive list of radionuclide activity concentrations in the secondary coolant is in Table A-2 in Appendix A.

## 4.3 Chemical and Volume Control System

The radionuclide concentrations at the inlet to the CVCS are from the primary coolant system letdown at primary coolant concentrations. Demineralizers remove radionuclides in the coolant by an ion-exchange mechanism. Parameters that impact the removal of activity include the concentration of the isotope entering the demineralizer and the removal efficiency for each isotope, which is consistent with current designs of large PWRs.

Leakage from the CVCS that goes to drain collections assumes leakage before the demineralizers. The activity of the exiting water through letdown follows the guidance and DF values found in NUREG-0017 for process components such as isotope-specific DFs for demineralizers. The DFs for the CVCS mixed-bed demineralizers are 100 for halogens, 2 for cesium and rubidium, and 50 for all others. The activity of the coolant after passing through the demineralizers is in Equation 4-7:

$$C_{out} = \frac{C_{in}}{D_f} \qquad \text{Equation 4-7}$$

where,

$C_{out}$ = Concentration levels on the outlet (µCi/g),

$C_{in}$ = Concentration levels on the inlet (µCi/g), and

$D_f$ = Decontamination factor for an isotope i in particulate filter or demineralizer.

Consistent with NUREG-0017, no credit is taken for CVCS filters.

## 4.4 Reactor Pool and Spent Fuel Pool

The activity of the reactor pool (including the refueling area of the common reactor pool) and the connected spent fuel pool in the NPP is dependent on the primary coolant activity within an NPM at the time of module disassembly for refueling. When an NPM is shut down after an operating cycle, the CVCS cleans up the primary coolant. The cleanup time period assumes sufficient cleaning of the primary coolant after a chemically-induced CRUD burst and an iodine spike to meet two dose rate targets. The first target maintains the accessible areas above and around the pool under a dose rate of 2.5 mRem/hour. The second target maintains the doses one meter above the pool below 5 mRem/hour per EPRI guidelines (Reference 7.2.19). When NPM disassembly occurs for refueling, the cleaned primary coolant releases into the refueling area of the pool.

Direct neutron activation of surrounding reactor pool water products from operating NPMs is negligible compared to the contribution from the primary coolant during refueling, due to the small flux in the pool. At the outside of the CNV, the largest neutron flux is at the core centerline and is several orders of magnitude lower than that of the active core. Additionally, it drops off very quickly because the pool is borated to 2000 ppm.

An evaluation for potential activation considers inadvertent impurities introduced into the pool. Resin backwash and breakthrough, lubricating oils, and hydraulic fluids have the potential for introduction into the pool in small quantities. They are hydrocarbon chemicals that do not introduce any new radioisotopes into the effluent stream. The postulated impurities either float on the top of the pool or sink to the bottom. In either case, they would not be close to the active core except for a very brief transit period while sinking. Therefore, there is negligible neutron flux available for activation. These small quantities dilute throughout a very large pool water mass, making their concentrations negligible to radioisotope production.

The activity released from a disassembled NPM in the refueling area of the pool is assumed to instantly mix homogenously throughout the entire pool volume (reactor pool and spent fuel pool). This modeling approach is conservative for effluent release because it does not take into account pool water cleanup during the time it takes the released activity to mix throughout the pool. During an event, the activity releases near the bottom of the refueling area of the pool and mixes both vertically and horizontally. By the time the released activity diffuses to the top of the pool, where it can become airborne (becoming an effluent source), there is some pool cleanup system removal as well as some decay. The concentration of the pool reaches a peak concentration for a short period before removal by radioactive decay, pool cleanup, and evaporation reduces the pool activity.

The pool purification system reduces the activity of the pool water to pre-refueling conditions, so that subsequent reloads do not result in a continuous buildup of radionuclides in the pool over time. The concentration is governed by Equation 4-8:

$$N(t) = N_o \times exp\left(-\left(\lambda + \frac{FR \times \varepsilon}{M}\right) \times t\right) \qquad \text{Equation 4-8}$$

where,

$N$ = concentration of the given radionuclide,

$\lambda$ = decay constant for the given radionuclide,

$FR$ = flow rate of the water through the cleanup system,

$\varepsilon$ = efficiency of the cleanup system, between zero (no effect) and one (perfectly efficient),

$M$ = mass of water, and

$t$ = time.

The exception to this treatment of radionuclides is tritium, which is difficult to remove from the water through cleanup. Tritium continues to build up to an equilibrium concentration in the pool due to losses from evaporation and decay, and is governed by Equation 4-9.

$$Tritium\ Pool\ Inventory(\infty) = \frac{Production\ rate\left(\frac{Ci}{year}\right)}{\left(\lambda + \left(\frac{evaporation\ rate\left(\frac{g}{day}\right) \times 365.25\left(\frac{days}{year}\right)}{pool\ mass(g)}\right)\right)} \quad \text{Equation 4-9}$$

The second mode of recycling primary water directly to the pool maximizes the tritium concentration in the pool, which also maximizes the tritium in the gaseous effluent stream, due to pool evaporation. Therefore, the gaseous effluent calculation uses the tritium concentration in the pool from recycling primary water to the pool.

## 4.5 Airborne Activity

Evaporation from the RXB reactor pool and spent fuel pool is the main source of airborne activity in the NPP. NUREG-0017 identifies numerous locations and sources of airborne radioactive material in a PWR as the main contributors of the gaseous effluent releases from normal operation and AOOs. The methodology evaluates a design-specific AOO of an inadvertent emergency core cooling system actuation that results in pressurizing the CNV. Primary coolant leaks, pool evaporation, and secondary coolant leaks use partition factors of 1 for gases and tritium, 0.01 for halogens, and 0.005 for other nuclides taken from NUREG-0017, page 2-10, Table 2-6. The pool evaporation partition coefficient for iodine is 2000 based on a pool temperature of 120 degrees F and a pH of 5 (Reference 7.2.5). For conservatism, these values are steam/water partition factors designated for U-tube SGs and used for pool evaporation. These values are conservative because more radionuclides become airborne from pressurized steam than from pool evaporation due to the excess energy acting as a driving force of both the pressure and the energy from the higher temperatures. Primary coolant leaks into the RXB contribute to airborne activity using a 40 percent flash fraction, where 60 percent of the leak remains in liquid form and 40 percent leaves as steam per Table 2-26 of NUREG-0017.

### 4.5.1 Waste Gas Processing System

Section 4.6 discusses the gaseous radioactive waste system (GRWS), which includes the waste gas processing system. Potential leakage from this system may result in airborne contamination. The system evaluation occurs at locations where the potential for airborne radioactivity exists.

### 4.5.2 Steam Generator Blowdown System

The NPM helical coil SG is an integral, once-through, helical coil design. Because the secondary coolant circulates on the inside of the SG tubes, the NPM helical coil SG does not have the capability to blowdown, and therefore does not have a blowdown system.

### 4.5.3 Condenser Air Ejector Exhaust

Each NPM has a dedicated secondary system with independent condenser air ejector systems. The condenser air ejector system's exhaust is a source of noble gases as well as halogens at an average release rate of 125 Ci/yr/NPM per µCi/g of secondary coolant. This value is linearly scaled by reactor thermal power from 1700 (Ci/y releases per µCi/g of primary coolant) from Table 2-22 of NUREG-0017 (Reference 7.2.1). The condenser air removal system maintains a vacuum on the condenser to remove gases. Removed gases pump through water separator tanks and vent to the atmosphere. This report determines the annual release rate for halogens and noble gases based on primary-to-secondary coolant system leak rates as well as leak rates out of the condenser air removal system. The condenser air removal system and gland seal steam system exhausts have direct, unfiltered pathways out of the TGB to the atmosphere.

### 4.5.4 Containment Purge Exhaust

The NPP design uses a steel CNV surrounding the RPV. Section 2.2.6 of NUREG-0017 attributes three percent of the primary coolant inventory of noble gases as leakage to containment every day. For the NPP, the containment evacuation system (CES) manages the CNV air and maintains the CNV under evacuated conditions. The CES normally vents to the Reactor Building HVAC system (RBVS). If the CES radiation monitors detect high radiation, the exhaust flow redirects to the GRWS for processing. The CES removes RPV leakage (0.47lbm/hr/NPM) into the CNV and routes RPV leakage via the GRWS decay beds for normal effluent. This method uses the low volumetric flow rate of gases leaving the CNV via the CES vacuum pump. Therefore, there is a need for sensitive radiation monitoring detection. Additionally, due to the benefit of the integral and natural circulation features of the NPM, there is less opportunity for gas leaks from the RCS.

### 4.5.5 Ventilation Exhaust Air from the Radioactive Waste Building and the Reactor Building

Sources of airborne radionuclides include primary leakage from the CVCS. Section 2.2.6 of NUREG-0017 attributes 160 lbm/day/NPM leak rate of primary coolant into the Auxiliary Building. Assuming a NPP has six times the primary leak rates of a larger PWR is overly conservative and unrealistic. The NPMs are much smaller and have less inventory. The methodology linearly scales the 160 lbm/day/reactor leak rate value by thermal power to 11.8 lbm/day per NPM, for a total plant leakage of 70.6 lbm/day. The total plant leakage of 70.6 lbm/day forms the basis for the effluent airborne inventory in the RXB from primary leaks from the CVCS. The NPM RXB functions similar to the Auxiliary Building of a large PWR, in terms of release pathways from the CVCS. Upon a high radiation signal in the RXB, the ventilation flow routes through HEPA and charcoal filters before release. The normal operation effluent calculations do not credit both HEPA and charcoal filtration.

### 4.5.6 Steam Leakage from Secondary System

Assumed steam leakage from the secondary system occurs in the TGB at the rate of 125 lbm/hour/NPM, for a total plant leak rate of 750 lbm/hour. This leak rate is linearly scaled by reactor thermal power to the 1700 lb/hr/reactor leak rate per RXM from NUREG-0017.

### 4.5.7 Reactor Pool Evaporation

In the RXB, evaporation from the reactor pool has the capability to release radioactive contaminants into the RXB airspace, which are then available for release to the environment. The pool source term rises during refueling events because the cleaned post-CRUD-burst primary coolant mixes with the pool water, as described in Section 4.4. The time-weighted average assumed pool source term over a year evaporates into the RXB airspace, which then goes through the RBVS and out the plant exhaust stack. The calculated total reactor pool evaporation rate is 1300 lbm/hour, with the drydock included. The total evaporation rate includes evaporation from the drydock for conservativism in gaseous effluent releases. The total pool water volume calculation does not include the drydock water volume for conservativism in the pool radionuclide concentrations.

### 4.5.8 Inadvertent Emergency Core Cooling System Actuation Anticipated Operational Occurrence

An AOO that is NuScale-specific is a single inadvertent emergency core cooling system actuation that floods the CNV with primary water, resulting in pressurization of the CNV. The CNV leaks an assumed 0.2 weight percent per day into the pool or the airspace under the bioshield. For the purpose of evaluating the effluent consequence of these AOOs, the CNV leakage is an assumed steady state gas leak into the region below the bioshield for 30 hours. Thirty hours is the period of time it takes the NPM to depressurize following an accident, based on containment transient thermal-hydraulic calculations. This leakage quantification uses the same method as the primary coolant leaks. This release is calculated to be 140 mCi into the RXB airspace.

## 4.6 Gaseous Radioactive Waste System

Up to six NPMs in a single NPP share the GRWS. The GRWS processes gaseous waste from degasification of the primary system letdown and the CES upon actuation of a high radiation signal through decay beds before discharge through the filtered plant exhaust stack.

### 4.6.1 Activity Input to the Guard Bed

The guard bed is the first charcoal bed to receive gaseous input from the LRWS degasifiers and the CES after the gas passes through a gas cooler and a moisture separator. For effluent release, the guard bed does not collect or delay any radionuclides, so the assumed input goes directly into the decay beds.

### 4.6.2 Activity Input to the Decay Beds

The charcoal decay beds delay noble gases from being released long enough to decay, thus reducing the amount released as gaseous effluent from the plant. There are two trains of decay beds in the GRWS, an A and B train. Each decay bed train has a charcoal mass of 4600 pounds. The adsorption coefficients and delay times for each bed are in Table 4-3.

**Table 4-3 Charcoal Decay Bed Information**

| Element | Adsorption Coefficient ($cm^3/g$) | Holdup Time (days/train) |
|---|---|---|
| Argon | 8.9 | 0.44 |
| Krypton | 60 | 2.9 |
| Xenon | 1400 | 69 |

Radionuclides present in the gaseous stream that collect in the beds decay over time. In some cases, these radionuclides decay to daughter products that are also radioactive. The calculation of daughter products is taken into account for the beds and evaluates parent radionuclides that buildup up to an equilibrium activity.

Because the charcoal filters collect at least 90 percent of iodine species from the gaseous stream, an assumed 90 percent of the chemically similar bromine species also collect.

Parent-to-daughter decay chains produce halogens. One-half of the halogen production is volatile in a gaseous form. Fifty percent of the daughter halogens produced in the bed are non-gaseous and stay in the bed. The volatile fraction of halogen production collects at a 90 percent efficiency by the charcoal bed, resulting in a 45 percent (0.5*0.9=0.45) retention. A total of 95 percent of the daughter halogen production is retained in the bed and five percent releases to the next bed.

Accounting and treatment of the total daily production rate of noble gas daughter products is an additional incoming activity (i.e., it is added to the system with the input source streams from the LRWS and CES).

## 4.7 Liquid Radioactive Waste System

Up to six NPMs in a single plant share the LRWS. The LRWS processes liquid waste from primary system letdown and other sources such as RXB floor drains, hot machine shop waste, spent resins, and other contaminated inputs resulting from plant operations.

Radioactive waste process streams calculate decay of radionuclides, including development of daughter products, taking into account the time for fluid collection and processing operations to complete.

### 4.7.1 Overall Liquid Radioactive Waste System Flow and Parameters

The processing equipment for the LRWS is site-specific. As such, there is a simplified LRWS processing skid model. For the presented method and conclusion to be applicable, the specifications shown in Table 4-4 and Table 4-5 must be maintained by site-specific LWRS process equipment.

**Table 4-4 Processing Paths for Liquid Radioactive Waste**

| **Low-Conductivity Waste (LCW) Liquid Processing Path** | | **High-Conductivity Waste (HCW) Liquid Processing Path** |
|---|---|---|
| **Combined component as modeled** | **Representative components in planned LCW skid** | Granulated activated charcoal filter (2x) |
| LCW filters, IX, accumulators | Pre-conditioning filter (2x) | LCW liquid processing path as needed |
| | Solids collection filter | NA |
| | Accumulator vessels (3x) | NA |
| | Ion exchanger vessels (5x) | NA |
| Reverse Osmosis (RO) Skid | Reverse osmosis skid | NA |
| Polishers downstream of RO | Polishing IX vessels (4x) | NA |

Table 1-4 of NUREG-0017 provides DFs for common treatment systems for PWR liquid waste. The removal DFs applied to the systems listed above are in Table 4-5.

**Table 4-5 Decontamination Factors Used in Liquid Radioactive Waste System Processing for Effluent Release**
(Reference 7.2.1)

| **Treatment System** | **Decontamination Factor** | | |
|---|---|---|---|
| | **Anion** | **Cs, Rb** | **Other Nuclides** |
| LCW filters, IX, accumulators | 100 | 17 | 100 |
| LCW Reverse osmosis | 10 (liquid wastes - all nuclides) | | |
| LCW polishers downstream of RO | 10 | 10 | 10 |
| HCW granulated activated carbon filter | 0 | 0 | 0 |
| Carbon bed for gaseous radioactive waste treatment | 90% for iodines | | |
| Evaporators (radwaste) | 1000 for all except iodine, 100 for iodine | | |

Liquid radioactive waste treatment of effluent source terms do not credit the granular activated charcoal beds in the LRWS.

The expected liquid waste inputs are shown in Table 4-6.

**Table 4-6 Expected Liquid Waste Inputs**

| **LRWS Input Source** | **Expected Input Rate** | **Expected Activity** |
|---|---|---|
| **LCW collection tank** | | |

**Table 4-6 Expected Liquid Waste Inputs**

| LRWS Input Source | Expected Input Rate | Expected Activity |
|---|---|---|
| RXB and RWB equipment drains | 2.9E+04 gpy<br>80 gpd | 0.001 primary coolant activity (PCA) |
| Other equipment drains | 1.1E+04 gpy<br>30 gpd | 0.093 PCA |
| Normal letdown (6 operating modules) | 1.9E+05 gpy<br>520 gpd | CVCS outlet |
| Additional CVCS letdown streams | 3.8E+04 gpy<br>100 gpd | CVCS outlet |
| Degasification before shutdown (6 times per year) | 3.0E+03 gpy | PCA through evaporator, modeling PZR venting |
| **LCW Total** | **2.7E+05 gpy** | |
| **HCW collection tank** | | |
| RXB and RWB floor drains<br>(via oil separator) | 7.3E+04 gpy<br>200 gpd | 0.1 PCA |
| RXB reactor component cooling water drain tank<br>(via oil separator) | 3.6E+01 gpy | 0.001 PCA |
| Hot machine shop, decontamination room sump<br>(via oil separator) | 9.0E+04 gpy | 0.01 PCA |
| RXB chemical drain tank (hot lab sink)<br>(via oil separator) | 4.4E+03 gpy<br>12 gpd | 0.05 PCA |
| RXB chemical drain tank (CES sample tank & floor drains)<br>(via oil separator) | 2.2E+04 gpy<br>60 gpd | CES liquid |
| Pump seal leaks<br>(via oil separator) | 8.1E+03 gpy<br>22 gpd | 0.1 PCA |
| Valve packing leaks<br>(via oil separator) | 4.8E+03 gpy<br>13 gpd | 0.1 PCA |
| Groundwater and condensation<br>(via oil separator) | 2.5E+05 gpy<br>680 gpd | 0.001 PCA |
| Equipment area decontamination (outside hot machine shop)<br>(via oil separator) | 1.5E+04 gpy<br>40 gpd | 0.01 PCA |
| Pool inlets to HCW | 2.9E+05 gpy | Pool source term |
| CVCS outlet sources into HCW | 3.5E+04 gpy | CVCS outlet |
| 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 |
| TGB floor drains | 2.2E+04 gpy | Secondary coolant |
| **HCW Total** | **8.6E+05 gpy** | |

### 4.7.2 Activity Input to Liquid Radioactive Waste Collection Tanks

The LRWS collection tanks are two 16,000 gallon HCW tanks and two 16,000 gallon LCW tanks. The difference between HCW and LCW streams is that LCW is within a system boundary, whereas HCW comes through the floor or equipment drain system.

In addition to a radiological component, the HCW may contain non-radiological contaminants such as dirt and oil.

Although the volume of the tanks is 16,000 gallons, the limit on total fill volume of the tanks is 14,400 gallons to prevent spilling and sloshing of liquid. This methodology uses the 14,400 gallon volume as the batch volume transferred to the liquid radioactive waste processing skids for treatment. Once a tank fills, the contents go through the processing equipment. The radionuclide content sums up from all incoming streams.

### 4.7.3 Activity Input to the Oil Separators

The oil separators receive input from the following sources:

- RXB floor drain sump
- RXB reactor component cooling water drain tank
- Hot machine shop decontamination room sump
- RWB floor drain sump
- RXB chemical drain tank

The oil separators process these liquids before entry to the HCW collection tanks.

### 4.7.4 Low-Conductivity Waste Sample Tanks

The LCW sample tanks receive treated low-conductivity liquid radioactive waste after it processes through the LCW processing skids.

### 4.7.5 High-Conductivity Waste Sample Tanks

The HCW sample tanks receive treated high conductivity liquid radioactive waste after it processes through the HCW processing skids. To determine the radionuclide content, the sample tank fills with HCW liquid that processes through the HCW granulated activated charcoal filter and optional LCW processing skid.

## 4.8 Plant Effluent Release

Effluent releases from the NPP consider the sum of individual liquid and gaseous releases. Liquid and gaseous effluents tracking and tabulation are by isotope. Once the radionuclides have left the plant, the analysis of site boundary concentrations and doses are the same as if the effluents were derived from GALE.

### 4.8.1 Gaseous Effluent Release

During normal operations, gaseous effluent releases come from the GRWS through the gaseous charcoal decay beds and from building exhausts (both processed and direct). The sum of these gaseous effluent release pathways constitutes the total annual gaseous effluent release from the plant. The following is a list of the modeled gaseous effluent pathways from a NPP:

- GRWS
  - Degasifier letdown
  - RPV leakage via CES
- RBVS
  - Pool evaporation
  - Containment vessel leakage AOO
  - Primary system leaks
- TGB
  - Condenser air removal system
  - System steam leaks, including from the gland seal steam condenser

In the TGB, the gland seal steam condenser and system leaks combine together into a single leakage term. The GRWS normally receives fission product gases from the primary coolant letdown (degasification) and processes them through decay beds before releasing them to the environment through the plant exhaust stack. Section 4.6 describes the added decay times allow for a reduction in total activity coming from the plant.

Section 4.5.7 describes how the concentration of radionuclides in the reactor pool water spikes during refueling events and then decreases as the water is cleaned up before the next refueling event. As a result, the airborne concentration in the airspace above the reactor pool water exhibits a similar behavior. While RXB ventilation design is based on the peak activity concentrations, the gaseous effluent from reactor pool evaporation is based on a time-weighted annual average reactor pool water source term, pool water evaporation rate, airspace ventilation rate, and ventilation system filter efficiencies. An average airborne concentration estimates the annual off-site dose from pool evaporation in the following Equation 4-10:

$$A(\infty) = P/K \quad \text{Equation 4-10}$$

where,

$A(\infty)$ = activity in the system at equilibrium (μCi),

$P$ = production term by which activity is added to the system (μCi/hr), and

$K$ = total removal rate of activity from the system (1/hr).

Then, the total airborne activity is divided by the volume of the airspace.

$$C_{RXB\ Air} = A(\infty)/V_{air} \qquad \text{Equation 4-11}$$

where,

$C_{RXB\ Air}$ = airborne equilibrium concentration (μCi/ml), and

$V_{air}$ = volume of the airspace (ml).

The evaporated pool water releases to the environment via the RXB ventilation system at a constant rate equal to the pool room exhaust flow rate. Section 4.5.5 describes another contribution of airborne activity to the RXB ventilation system: primary system coolant leaks into the RXB originating from the CVCS. The plant exhaust stack monitors and releases airborne radionuclides captured by the building ventilation.

To account for a design-specific AOO, NuScale includes the gaseous effluent from an inadvertent emergency core cooling system actuation, as described in Section 4.5.8. Section 4.5 describes additional sources of gaseous effluent from the TGB, including secondary coolant steam leaks and the condenser air removal systems, which are direct (unfiltered) ground releases. The total gaseous effluent release from the plant is in Table A-3 of Appendix A.

### 4.8.2 Liquid Effluent Release

Liquid radioactive waste collects, and the HCW and LCW go to collection tanks in the RWB for processing. The collection tanks collect plant waste from normal reactor letdown, drains, resin backwash, and other contaminated liquids. The LRWS processes, samples, and discharges the liquids through a common release point through the utility water system. Section 4.7 describes the LRWS input volumes and processing parameters.

An adjustment factor to account for AOOs adds an additional 0.071 Ci per year release to the cumulative non-tritium liquid effluent releases. This value scales linearly with reactor thermal power (250 $MW_{th}$ * 6 vs. 3400 $MW_{th}$) from the 0.16 Ci per year value from NUREG-0017.

The total liquid effluent release from the plant is in Table A-3 of Appendix A.

# 5.0 Fuel Failure Fraction

The GALE code is based on empirical (operating) data. Therefore, NUREG-0017 does not specify a fuel failure fraction. The methodology employs a first-principles calculation that determines fission-product related contributions to effluents by assuming a realistic and conservative fuel failure fraction. The industry-reported fuel failure fraction is an equivalent release value that represents the effects from several failure mechanisms. The evaluation of fuel isotopic inventory uses a NuScale-assumed fuel failure fraction, with radionuclide release, buildup and removal, equilibrium concentrations in the primary coolant, and forms the basis for determining liquid and gaseous contributions.

Without operating history, the methodology uses available industry operating experience based on the similarities between the NuScale core and fuel design compared to existing PWRs. The design uses the same 17 x 17 PWR fuel assemblies, shorter in length, with Framatome M5™ cladding and low enriched uranium-235 uranium dioxide pellets in helium-backfilled and pressurized fuel rods. The selection of a fuel failure fraction is based on recent PWR fuel performance observed in the operating fleet for similar type fuel.

## 5.1 US Pressurized Water Reactor Fuel Failure History

The numerical basis for fuel failure rates comes from EPRI's "PWR Fuel Rod Failure Rate Analysis" report (Reference 7.2.20). Table A-4 presents data from the Fuel Reliability Database report. Table 5-1 presents a summary of the data from this report representing the years 2007-2016 highlighting the highest and lowest values of failed fuel fraction. This data represents fuel failures determined with the 'outage method' as described in the International Atomic Energy Agency (IAEA) Review of Fuel Failures in Water Cooled Reactors, IAEA Nuclear Energy Series No. NF-T-2.1 (Reference 7.2.24), as the number of failed pins during an operating cycle divided by the total number of pins with a refueling outage that year. This data includes only operating PWRs within the U.S. with zirconium clad fuel.

**Table 5-1 Fuel Failure Values**
(Reference 7.2.20)

{{

| | | | |
|---|---|---|---|
| | | | |
| | | | |

}}[2(a),(c)]

Table 5-1 shows that the lowest data point in the most recent ten years of U.S. PWR data is {{ }}[2(a),(c)] reported in {{ }}[2(a),(c)] (Reference 7.2.6). The highest data point is 66 rods per million (0.0066 percent) reported in {{ }}[2(a),(c)] (Reference 7.2.7). For comparison, NUREG-0017 was published in

1985 and based on data from the 1970s (Reference 7.2.1).The ANSI/ANS ANS-18.1-1999 standard for primary and secondary coolant concentrations was published in 1999 based on industry data of that time. The PWRGALE-09 code was benchmarked against operational reactor data from 2005 to 2010 (Reference 7.2.8). The average fuel failure fraction from that time period was {{ }}2(a),(c).

Table A-4 lists the fuel failure data for U.S. PWRs with zirconium-alloy cladding for the years 2002 to 2016.

## 5.2 Fuel Failure Fraction Conclusions

The NPP fuel design is based on a standard design Framatome 17 x 17 fuel assembly, which is approximately half the length of current large PWRs. The fuel uses the same fabrication techniques, quality assurance, and testing as the fuel assemblies fabricated by Framatome and irradiated in large PWRs. Thus the utilization of fuel performance from the operating PWR fleet is applicable for this purpose.

The long-term industry trend on improved fuel performance is well defined and highlights the continuing improvement. Therefore, to replace the GALE empirical data, the methodology uses a realistic and conservative fuel failure fraction based on industry performance of 0.0066percent (66 rods per million) for fission product-related effluent releases. The following supports this value.

- In U.S. PWRs, the fuel failure fraction has decreased and continues to decrease over time, with the most recent data {{ }}2(a),(c) and a maximum value of the most recent ten years of available data {{ }}2(a),(c) of 66 rods per million (0.0066 percent).
- The method uses a realistic and conservative fuel failure fraction based on industry performance of 0.0066 percent with conservative escape rate coefficients for the purpose of evaluating fission product-related effluent releases.
- The realistic and conservative fuel failure fraction is initiated at the beginning of cycle and is sustained for the duration of the fuel cycle.

## 6.0 Summary and Conclusion

The NPP design is similar to large PWRs in the existing fleet with regard to effluent releases (production, process, and release). Due to differences associated with a smaller, passive NPP design, the GALE code is not representative of the NPP design and does not accurately estimate NPP effluent releases. This results in the need to develop an alternate “GALE replacement” methodology. The effluent release methodology described in this report is based on compliance with applicable regulations, with no exemptions needed.

The methodology is realistic and conservative, using first-principles-based calculations where appropriate, combined with recent nuclear industry experience and lessons learned. Table 6-1 presents a summary of the effluent release methodology. Liquid and gaseous effluents use realistic and conservative source terms. The calculation of effluents includes the design-specific treatment of liquid and gaseous radioactive source terms such as filtration, resin absorption, holdup, dilution, and decay.

**Table 6-1 Primary Contributors and Methodology Employed for Effluents**

<table>
<tr><th>Primary Contributors</th><th>NuScale Methodology</th></tr>
<tr><td>Water activation products</td><td rowspan="5">Calculations based on first-principles physics</td></tr>
<tr><td>• H-3 (tritium)</td></tr>
<tr><td>• C-14 (radiocarbon)</td></tr>
<tr><td>• N-16</td></tr>
<tr><td>• Ar-41</td></tr>
<tr><td>Activated corrosion and wear products (CRUD)</td><td>• Recent large PWR operating data<br>• Lessons learned</td></tr>
<tr><td>Fission products (failed-fuel related)</td><td>• Calculations based on first-principles physics<br>• Recent large PWR operating data</td></tr>
</table>

The primary and secondary coolant isotopic distribution is in Table A-3. The total effluents are 850 Ci of gaseous effluent and 1,200 Ci of liquid effluent, with tritium being the largest contributor to both. The gaseous effluent is evaluated in Mode 2 (RCS recycled to pool), and the liquid effluent is evaluated in Mode 1 (no recycling of RCS), which maximizes the respective effluent releases.

## 7.0 References

### 7.1 Source Documents

7.1.1 American Society of Mechanical Engineers, Quality Assurance Program Requirements for Nuclear Facility Applications (QA), ASME NQA-1-2008, ASME NQA-1a-2009 Addenda, as endorsed by Regulatory Guide 1.28, Rev. 4, New York, NY.

7.1.2 U.S. Code of Federal Regulations, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Facilities,” Appendix B, Part 50, Chapter 1, Title 10, “Energy,” (10 CFR 50 Appendix B).

### 7.2 Referenced Documents

7.2.1 U.S. Nuclear Regulatory Commission, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRGALE Code),” NUREG-0017, Rev. 1, April 1985.

7.2.2 American National Standard Institute/American Nuclear Society, “Radioactive Source Term for Normal Operation of Light Water Reactors,” ANSI/ANS-18.1-1999, LaGrange Park, IL.

7.2.3 Pacific Northwest National Laboratory, “Applicability of GALE-86 Codes to Integral Pressurized Water Reactor Designs,” PNNL-21386, May 2012.

7.2.4 Korea Electric Power Corporation (KEPKO) and Korea Hydro & Nuclear Power Co., Ltd. (KHNP) , APR1400 Design Control Document Revision 0,” December 2014, NRC Agencywide Document Access and Management System (ADAMS) Accession No. ML15006A059

7.2.5 Electric Power Research Institute "Nuclear Power Plant Related Iodine Partition Coefficients," EPRI-NP-1271, December 1979

7.2.6 Idaho National Laboratory, Bragg-Sitton, S., “Light Water Reactor Sustainability Program, Advanced LWR Nuclear Fuel Cladding System Development: Technical Program Plan,” INL/MIS-12-25696, Rev. 1, December 2012.

7.2.7 Electric Power Research Institute, “The Path to Zero Defects: EPRI Fuel Reliability Guidelines,” EPRI, Palo Alto, CA, 2008.

7.2.8 Geelhood, K.J., “Benchmarking of GALE-09 Release Predictions Using Site Specific Data from 2005 to 2010,” PNNL-22076, November 2012.

7.2.9 Westinghouse, “AP1000 Design Control Document Revision 19,” June 2011, NRC Agencywide Document Access and Management System (ADAMS) Accession No. ML11171A500.

7.2.10 AREVA NP Inc., NRC Agencywide Document Access and Management System (ADAMS) Accession No. ML13220A883.

7.2.11 Mitsubishi Heavy Industries, LTD., “US-APWR Design Control Document Revision 4,” September 2013, NRC Agencywide Document Access and Management System (ADAMS) Accession No. ML13262A304.

7.2.12 Electric Power Research Institute, “EPRI Tritium Management Model Project Summary Report,” (EPRI #1009903), Palo Alto, CA, November 2005.

7.2.13 Electric Power Research Institute, “Steam Generator Management Program: PWR Primary-to Secondary Leak Guidelines,” (EPRI #1022832), Rev. 4, Palo Alto, CA, September 2011.

7.2.14 Oak Ridge National Laboratory, “SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design,” ORNL/TM-2005/39, Version 6.1, Oak Ridge, Tennessee, June 2011.

7.2.15 U.S. Code of Federal Regulations, “Standards for Protection against Radiation,” Part 20, Chapter 1, Title 10, “Energy,” (10 CFR 20).

7.2.16 U.S. Code of Federal Regulations, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Chapter 1, Title 10, “Energy,” (10 CFR 50).

7.2.17 American Nuclear Society, “2014 Performance Indicators issued for U.S. Power Reactors,” Nuclear News, June 2015.

7.2.18 U.S. Nuclear Regulatory Commission, “Radioactive Effluents from Nuclear Power Plants-Annual Report 2010 - Final Report,” NUREG/CR-2907, Vol. 16, May 2018. NRC Agencywide Document Access and Management System (ADAMS) Accession No. ML18151A529

7.2.19 Electric Power Research Institute, “EPRI Pressurized Water Reactor Primary Water Chemistry Guidelines,” 3002000505, Vol. 1, Rev. 7, Palo Alto, CA, 2014.

7.2.20 Electric Power Research Institute, “PWR Fuel Rod Failure Rate Analysis,” FRP_2018_1 Final Letter Report, Palo Alto, CA, February 2018.

7.2.21 Castelli, R. A., Nuclear Corrosion Modelling, The Nature of CRUD, Elsevier, Oxford, 2010.

7.2.22 International Atomic Energy Agency, “Modelling of Transport of Radioactive Substances in the Primary Circuit of Water-Cooled Reactors,” IAEA-TECDOC-1672, Vienna, Austria, March 2012.

7.2.23 Electric Power Research Institute, “Steam Generator Management Program: PWR Primary-to-Secondary Leak Guidelines Revision 4," EPRI Technical Report 1022832, Palo Alto, Ca, September 2011.

7.2.24 International Atomic Energy Agency, “Review of Fuel Failures in Water Cooled Reactors,” IAEA Nuclear Energy Series No. NF-T-2.1, June 2010.

# Appendix A Summary Tables

## Table A-1 Maximum Fuel Inventory per Assembly (Ci)

| Radionuclide | Assembly Activity (Ci) |
|---|---|
| Noble Gases | |
| Kr83m | 2.0E+04 |
| Kr85m | 4.0E+04 |
| Kr85 | 4.1E+03 |
| Kr87 | 7.7E+04 |
| Kr88 | 1.0E+05 |
| Kr89 | 1.2E+05 |
| Xe131m | 2.8E+03 |
| Xe133m | 1.2E+04 |
| Xe133 | 3.8E+05 |
| Xe135m | 8.8E+04 |
| Xe135 | 1.1E+05 |
| Xe137 | 3.4E+05 |
| Xe138 | 3.1E+05 |
| Halogens | |
| Br82 | 1.1E+03 |
| Br83 | 1.9E+04 |
| Br84 | 3.2E+04 |
| Br85 | 4.0E+04 |
| I129 | 1.5E-02 |
| I130 | 1.1E+04 |
| I131 | 1.9E+05 |
| I132 | 2.8E+05 |
| I133 | 3.8E+05 |
| I134 | 4.2E+05 |
| I135 | 3.6E+05 |
| Rubidium, Cesium | |
| Rb86m | 8.4E+01 |
| Rb86 | 6.7E+02 |
| Rb88 | 1.0E+05 |
| Rb89 | 1.3E+05 |
| Cs132 | 1.4E+01 |
| Cs134 | 9.3E+04 |
| Cs135m | 1.2E+03 |
| Cs136 | 2.1E+04 |
| Cs137 | 4.8E+04 |
| Cs138 | 3.4E+05 |
| Other FPs | |
| P32 | 2.9E+01 |
| Co57 | 2.2E-01 |
| Sr89 | 1.4E+05 |
| Sr90 | 3.3E+04 |
| Sr91 | 1.8E+05 |
| Sr92 | 2.0E+05 |

## Table A-1 Maximum Fuel Inventory per Assembly (Ci) (Continued)

| Radionuclide | Assembly Activity (Ci) |
|---|---|
| Y90 | 3.4E+04 |
| Y91m | 1.1E+05 |
| Y91 | 1.9E+05 |
| Y92 | 2.1E+05 |
| Y93 | 2.4E+05 |
| Zr97 | 3.0E+05 |
| Nb95 | 2.9E+05 |
| Mo99 | 3.4E+05 |
| Mo101 | 3.3E+05 |
| Tc99m | 3.0E+05 |
| Tc99 | 5.9E+00 |
| Ru103 | 3.7E+05 |
| Ru105 | 3.0E+05 |
| Ru106 | 2.3E+05 |
| Rh103m | 3.7E+05 |
| Rh105 | 2.8E+05 |
| Rh106 | 2.6E+05 |
| Ag110 | 9.7E+04 |
| Sb124 | 5.6E+02 |
| Sb125 | 4.1E+03 |
| Sb127 | 2.2E+04 |
| Sb129 | 6.4E+04 |
| Te125m | 9.6E+02 |
| Te127m | 3.6E+03 |
| Te127 | 2.2E+04 |
| Te129m | 1.1E+04 |
| Te129 | 6.2E+04 |
| Te131m | 4.2E+04 |
| Te131 | 1.6E+05 |
| Te132 | 2.7E+05 |
| Te133m | 1.7E+05 |
| Te134 | 3.2E+05 |
| Ba137m | 4.5E+04 |
| Ba139 | 3.3E+05 |
| Ba140 | 3.1E+05 |
| La140 | 3.3E+05 |
| La141 | 2.9E+05 |
| La142 | 2.8E+05 |
| Ce141 | 3.0E+05 |
| Ce143 | 2.7E+05 |
| Ce144 | 2.5E+05 |
| Pr143 | 2.6E+05 |
| Pr144 | 2.5E+05 |
| Np239 | 5.2E+06 |

**Table A-2 Primary and Secondary Coolant Radionuclide Activity Concentrations (Mode 1)**

| Radionuclide | Primary Activity (µCi/g) | Secondary Activity (µCi/g) |
|---|---|---|
| Noble Gases | | |
| Kr83m | 7.7E-04 | 2.7E-10 |
| Kr85m | 3.2E-03 | 1.1E-09 |
| Kr85 | 1.6E-01 | 5.7E-08 |
| Kr87 | 1.8E-03 | 6.2E-10 |
| Kr88 | 5.1E-03 | 1.8E-09 |
| Kr89 | 1.2E-04 | 4.2E-11 |
| Xe131m | 1.3E-02 | 4.5E-09 |
| Xe133m | 1.1E-02 | 4.0E-09 |
| Xe133 | 8.3E-01 | 2.9E-07 |
| Xe135m | 1.1E-03 | 3.9E-10 |
| Xe135 | 2.3E-02 | 8.1E-09 |
| Xe137 | 3.9E-04 | 1.4E-10 |
| Xe138 | 1.3E-03 | 4.7E-10 |
| Halogens | | |
| Br82 | 2.1E-05 | 7.6E-12 |
| Br83 | 1.2E-04 | 4.3E-11 |
| Br84 | 5.7E-05 | 1.8E-11 |
| Br85 | 6.9E-06 | 1.2E-12 |
| I129 | 3.5E-10 | 1.3E-16 |
| I130 | 1.7E-04 | 6.2E-11 |
| I131 | 4.4E-03 | 1.6E-09 |
| I132 | 2.1E-03 | 7.2E-10 |
| I133 | 6.7E-03 | 2.4E-09 |
| I134 | 1.2E-03 | 4.1E-10 |
| I135 | 4.3E-03 | 1.5E-09 |
| Rubidium, Cesium | | |
| Rb86m | 5.2E-09 | 4.5E-16 |
| Rb86 | 3.0E-05 | 1.2E-11 |
| Rb88 | 5.1E-03 | 1.7E-09 |
| Rb89 | 2.4E-04 | 7.5E-11 |
| Cs132 | 6.0E-07 | 2.3E-13 |
| Cs134 | 4.3E-03 | 1.7E-09 |
| Cs135m | 3.6E-06 | 1.3E-12 |
| Cs136 | 9.4E-04 | 3.7E-10 |
| Cs137 | 2.2E-03 | 8.7E-10 |
| Cs138 | 1.9E-03 | 6.8E-10 |
| Other FPs | | |
| P32 | 8.4E-11 | 3.0E-17 |
| Co57 | 6.4E-13 | 2.3E-19 |
| Sr89 | 3.8E-06 | 1.4E-12 |
| Sr90 | 5.9E-07 | 2.1E-13 |
| Sr91 | 2.0E-06 | 7.1E-13 |
| Sr92 | 1.1E-06 | 3.7E-13 |

## Table A-2 Primary and Secondary Coolant Radionuclide Activity Concentrations (Mode 1) (Continued)

| Radionuclide | Primary Activity (µCi/g) | Secondary Activity (µCi/g) |
|---|---|---|
| Y90 | 1.4E-07 | 5.2E-14 |
| Y91m | 1.1E-06 | 3.6E-13 |
| Y91 | 5.6E-07 | 2.0E-13 |
| Y92 | 9.0E-07 | 3.2E-13 |
| Y93 | 4.3E-07 | 1.5E-13 |
| Zr97 | 6.3E-07 | 2.2E-13 |
| Nb95 | 1.6E-06 | 5.6E-13 |
| Mo99 | 1.1E-03 | 4.0E-10 |
| Mo101 | 4.3E-05 | 1.3E-11 |
| Tc99m | 1.0E-03 | 3.7E-10 |
| Tc99 | 2.1E-08 | 7.6E-15 |
| Ru103 | 1.1E-06 | 3.9E-13 |
| Ru105 | 3.6E-07 | 1.3E-13 |
| Ru106 | 6.8E-07 | 2.4E-13 |
| Rh103m | 1.1E-06 | 3.6E-13 |
| Rh105 | 7.3E-07 | 2.6E-13 |
| Rh106 | 6.8E-07 | 3.2E-14 |
| Ag110 | 4.8E-06 | 1.9E-13 |
| Sb124 | 1.6E-09 | 5.8E-16 |
| Sb125 | 1.2E-08 | 4.3E-15 |
| Sb127 | 6.1E-08 | 2.2E-14 |
| Sb129 | 7.6E-08 | 2.7E-14 |
| Te125m | 1.7E-06 | 6.2E-13 |
| Te127m | 6.6E-06 | 2.4E-12 |
| Te127 | 2.6E-05 | 9.4E-12 |
| Te129m | 1.9E-05 | 6.8E-12 |
| Te129 | 2.7E-05 | 9.3E-12 |
| Te131m | 6.2E-05 | 2.2E-11 |
| Te131 | 3.1E-05 | 9.8E-12 |
| Te132 | 4.6E-04 | 1.6E-10 |
| Te133m | 3.9E-05 | 1.3E-11 |
| Te134 | 5.6E-05 | 1.8E-11 |
| Ba137m | 2.1E-03 | 3.3E-10 |
| Ba139 | 1.0E-06 | 3.6E-13 |
| Ba140 | 5.6E-06 | 2.0E-12 |
| La140 | 1.6E-06 | 5.8E-13 |
| La141 | 3.2E-07 | 1.1E-13 |
| La142 | 1.5E-07 | 5.3E-14 |
| Ce141 | 8.6E-07 | 3.1E-13 |
| Ce143 | 6.5E-07 | 2.3E-13 |
| Ce144 | 7.3E-07 | 2.6E-13 |
| Pr143 | 7.6E-07 | 2.7E-13 |
| Pr144 | 7.2E-07 | 2.2E-13 |
| Np239 | 1.4E-05 | 4.9E-12 |

### Table A-2 Primary and Secondary Coolant Radionuclide Activity Concentrations (Mode 1) (Continued)

| Radionuclide | Primary Activity (µCi/g) | Secondary Activity (µCi/g) |
|---|---|---|
| Corrosion Activation Products - CRUD | | |
| Na24 | 1.4E-02 | 4.9E-09 |
| Cr51 | 7.7E-04 | 2.8E-10 |
| Mn54 | 4.0E-04 | 1.4E-10 |
| Fe55 | 3.0E-04 | 1.1E-10 |
| Fe59 | 7.5E-05 | 2.7E-11 |
| Co58 | 1.1E-03 | 4.1E-10 |
| Co60 | 1.3E-04 | 4.7E-11 |
| Ni63 | 6.6E-05 | 2.3E-11 |
| Zn65 | 1.3E-04 | 4.5E-11 |
| Zr95 | 9.7E-05 | 3.5E-11 |
| Ag110m | 3.2E-04 | 1.2E-10 |
| W187 | 7.0E-04 | 2.5E-10 |
| Water Activation Products | | |
| H3 | 1.3E+00 | 2.4E-03 |
| C14 | 2.6E-04 | 9.4E-11 |
| N16[1] | 4.1E+01 | 1.5E-05 |
| Ar41 | 1.4E-01 | 5.0E-08 |

[1] N16 concentration values represented are for the top (entrance) of the SG region. N16 values vary throughout the NPM primary coolant volume due to decay during transit from low primary coolant flow rate.

### Table A-3 Gaseous and Liquid Yearly Effluent Release Values for a NuScale Power Plant (with Six Operating Modules)

| Radionuclide | Gaseous Effluent (Ci/yr) | | | Liquid Effluent (Ci/Yr)[1] |
|---|---|---|---|---|
| | Plant Exhaust Stack Releases | Turbine Generator Building Releases | Total | |
| Noble Gases | | | | |
| Kr83m | 9.1E-03 | 4.2E-03 | 1.3E-02 | - |
| Kr85m | 3.8E-02 | 1.8E-02 | 5.6E-02 | - |
| Kr85 | 1.4E+02 | 8.9E-01 | 1.5E+02 | - |
| Kr87 | 2.1E-02 | 9.7E-03 | 3.1E-02 | - |
| Kr88 | 6.1E-02 | 2.8E-02 | 8.9E-02 | - |
| Kr89 | 1.4E-03 | 6.4E-04 | 2.0E-03 | - |
| Xe131m | 7.0E-01 | 6.9E-02 | 7.7E-01 | - |
| Xe133m | 5.6E-01 | 6.3E-02 | 6.2E-01 | - |
| Xe133 | 1.6E+01 | 4.6E+00 | 2.0E+01 | - |
| Xe135m | 6.7E-02 | 6.1E-03 | 7.3E-02 | - |
| Xe135 | 3.0E-01 | 1.3E-01 | 4.3E-01 | - |
| Xe137 | 4.6E-03 | 2.1E-03 | 6.7E-03 | - |
| Xe138 | 1.6E-02 | 7.2E-03 | 2.3E-02 | - |
| Halogens | | | | |
| Br82 | 1.0E-06 | 2.8E-08 | 1.0E-06 | 2.3E-09 |

**Table A-3 Gaseous and Liquid Yearly Effluent Release Values for a NuScale Power Plant (with Six Operating Modules) (Continued)**

| Radionuclide | Gaseous Effluent (Ci/yr) | | | Liquid Effluent (Ci/Yr)[1] |
|---|---|---|---|---|
| | Plant Exhaust Stack Releases | Turbine Generator Building Releases | Total | |
| Br83 | 5.8E-06 | 1.6E-07 | 5.9E-06 | - |
| Br84 | 2.7E-06 | 6.9E-08 | 2.8E-06 | - |
| Br85 | 3.2E-07 | 4.3E-09 | 3.3E-07 | - |
| I129 | 1.7E-11 | 4.7E-13 | 1.7E-11 | 1.9E-12 |
| I130 | 8.2E-06 | 2.3E-07 | 8.5E-06 | 2.1E-11 |
| I131 | 5.3E-04 | 5.9E-06 | 5.4E-04 | 1.7E-05 |
| I132 | 9.8E-05 | 2.7E-06 | 1.0E-04 | 3.8E-07 |
| I133 | 3.5E-04 | 9.0E-06 | 3.5E-04 | 5.8E-08 |
| I134 | 5.7E-05 | 1.5E-06 | 5.9E-05 | - |
| I135 | 2.0E-04 | 5.6E-06 | 2.1E-04 | 5.3E-14 |
| Rubidium, Cesium | | | | |
| Rb86m | 1.2E-10 | 1.3E-12 | 1.2E-10 | - |
| Rb86 | 1.5E-06 | 3.5E-08 | 1.6E-06 | 5.0E-06 |
| Rb88 | 1.2E-04 | 5.0E-06 | 1.3E-04 | - |
| Rb89 | 5.5E-06 | 2.2E-07 | 5.7E-06 | - |
| Cs132 | 2.8E-08 | 7.0E-10 | 2.9E-08 | 3.3E-08 |
| Cs134 | 2.3E-04 | 5.0E-06 | 2.3E-04 | 1.4E-03 |
| Cs135m | 8.4E-08 | 3.9E-09 | 8.8E-08 | - |
| Cs136 | 4.7E-05 | 1.1E-06 | 4.8E-05 | 1.2E-04 |
| Cs137 | 1.2E-04 | 2.6E-06 | 1.2E-04 | 7.2E-04 |
| Cs138 | 4.5E-05 | 2.0E-06 | 4.7E-05 | - |
| Other Fission Products | | | | |
| P32 | 2.7E-12 | 8.9E-14 | 2.7E-12 | 3.4E-13 |
| Co57 | 2.1E-14 | 6.8E-16 | 2.1E-14 | 4.5E-15 |
| Sr89 | 1.2E-07 | 4.1E-09 | 1.3E-07 | 3.6E-08 |
| Sr90 | 1.9E-08 | 6.3E-10 | 2.0E-08 | 4.4E-09 |
| Sr91 | 4.7E-08 | 2.1E-09 | 4.9E-08 | 1.3E-14 |
| Sr92 | 2.5E-08 | 1.1E-09 | 2.6E-08 | - |
| Y90 | 6.6E-09 | 1.5E-10 | 6.7E-09 | 4.1E-09 |
| Y91m | 2.5E-08 | 1.1E-09 | 2.6E-08 | 8.1E-15 |
| Y91 | 1.8E-08 | 5.9E-10 | 1.9E-08 | 3.5E-09 |
| Y92 | 2.1E-08 | 9.5E-10 | 2.2E-08 | - |
| Y93 | 1.0E-08 | 4.5E-10 | 1.1E-08 | 5.6E-15 |
| Zr97 | 1.5E-08 | 6.7E-10 | 1.6E-08 | 1.2E-12 |
| Nb95 | 3.9E-05 | 1.7E-09 | 3.9E-05 | 7.7E-07 |
| Mo99 | 3.2E-05 | 1.2E-06 | 3.4E-05 | 6.8E-07 |
| Mo101 | 1.0E-06 | 3.8E-08 | 1.1E-06 | - |
| Tc99m | 3.0E-05 | 1.1E-06 | 3.1E-05 | 6.6E-07 |
| Tc99 | 7.0E-10 | 2.3E-11 | 7.2E-10 | 1.6E-10 |
| Ru103 | 3.5E-08 | 1.2E-09 | 3.6E-08 | 6.3E-09 |
| Ru105 | 8.4E-09 | 3.8E-10 | 8.8E-09 | - |
| Ru106 | 2.2E-08 | 7.2E-10 | 2.3E-08 | 4.9E-09 |
| Rh103m | 3.4E-08 | 1.1E-09 | 3.6E-08 | 6.3E-09 |

## Table A-3 Gaseous and Liquid Yearly Effluent Release Values for a NuScale Power Plant (with Six Operating Modules) (Continued)

| Radionuclide | Gaseous Effluent (Ci/yr) | | | Liquid Effluent (Ci/Yr)[1] |
|---|---|---|---|---|
| | Plant Exhaust Stack Releases | Turbine Generator Building Releases | Total | |
| Rh105 | 2.0E-08 | 7.8E-10 | 2.0E-08 | 8.5E-11 |
| Rh106 | 2.2E-08 | 9.7E-11 | 2.2E-08 | 4.9E-09 |
| Ag110 | 4.0E-05 | 5.7E-10 | 4.0E-05 | 1.9E-07 |
| Sb124 | 5.2E-11 | 1.7E-12 | 5.4E-11 | 1.0E-11 |
| Sb125 | 3.9E-10 | 1.3E-11 | 4.1E-10 | 8.8E-11 |
| Sb127 | 1.8E-09 | 6.5E-11 | 1.9E-09 | 6.7E-11 |
| Sb129 | 1.8E-09 | 7.9E-11 | 1.8E-09 | - |
| Te125m | 5.6E-08 | 1.9E-09 | 5.8E-08 | 1.1E-08 |
| Te127m | 2.2E-07 | 7.1E-09 | 2.2E-07 | 4.5E-08 |
| Te127 | 6.8E-07 | 2.8E-08 | 7.1E-07 | 4.4E-08 |
| Te129m | 6.1E-07 | 2.0E-08 | 6.3E-07 | 1.1E-07 |
| Te129 | 7.4E-07 | 2.8E-08 | 7.7E-07 | 6.8E-08 |
| Te131m | 1.6E-06 | 6.6E-08 | 1.7E-06 | 3.5E-09 |
| Te131 | 7.6E-07 | 2.9E-08 | 7.9E-07 | 7.9E-10 |
| Te132 | 1.3E-05 | 4.8E-07 | 1.4E-05 | 3.7E-07 |
| Te133m | 9.2E-07 | 3.9E-08 | 9.5E-07 | - |
| Te134 | 1.3E-06 | 5.5E-08 | 1.4E-06 | - |
| Ba137m | 1.1E-04 | 9.7E-07 | 1.1E-04 | 6.8E-04 |
| Ba139 | 2.4E-08 | 1.1E-09 | 2.5E-08 | - |
| Ba140 | 1.8E-07 | 5.9E-09 | 1.8E-07 | 2.1E-08 |
| La140 | 7.3E-08 | 1.7E-09 | 7.5E-08 | 2.3E-08 |
| La141 | 7.5E-09 | 3.4E-10 | 7.8E-09 | - |
| La142 | 3.6E-09 | 1.6E-10 | 3.7E-09 | - |
| Ce141 | 2.8E-08 | 9.1E-10 | 2.9E-08 | 4.8E-09 |
| Ce143 | 1.7E-08 | 6.9E-10 | 1.8E-08 | 5.4E-11 |
| Ce144 | 2.4E-08 | 7.7E-10 | 2.4E-08 | 5.2E-09 |
| Pr143 | 2.4E-08 | 8.1E-10 | 2.5E-08 | 3.2E-09 |
| Pr144 | 2.3E-08 | 6.5E-10 | 2.4E-08 | 5.1E-09 |
| Np239 | 3.8E-07 | 1.5E-08 | 4.0E-07 | 5.8E-09 |
| Corrosion Activation Products - CRUD | | | | |
| Na24 | 3.3E-04 | 1.5E-05 | 3.5E-04 | 1.0E-08 |
| Cr51 | 6.7E-03 | 8.2E-07 | 6.7E-03 | 2.7E-05 |
| Mn54 | 3.6E-03 | 4.2E-07 | 3.6E-03 | 1.7E-05 |
| Fe55 | 2.7E-03 | 3.2E-07 | 2.7E-03 | 1.3E-05 |
| Fe59 | 6.6E-04 | 8.0E-08 | 6.6E-04 | 2.9E-06 |
| Co58 | 1.0E-01 | 1.2E-06 | 1.0E-01 | 4.0E-04 |
| Co60 | 1.2E-03 | 1.4E-07 | 1.2E-03 | 5.9E-06 |
| Ni63 | 6.0E-04 | 7.0E-08 | 6.0E-04 | 3.0E-06 |
| Zn65 | 1.2E-03 | 1.3E-07 | 1.2E-03 | 5.5E-06 |
| Zr95 | 8.7E-04 | 1.0E-07 | 8.7E-04 | 3.9E-06 |
| Ag110m | 2.9E-03 | 3.4E-07 | 2.9E-03 | 1.4E-05 |
| W187 | 1.3E-03 | 7.4E-07 | 1.3E-03 | 3.6E-08 |
| Water Activation Products | | | | |

## Table A-3 Gaseous and Liquid Yearly Effluent Release Values for a NuScale Power Plant (with Six Operating Modules) (Continued)

| Radionuclide | Gaseous Effluent (Ci/yr) | | | Liquid Effluent (Ci/Yr)[1] |
|---|---|---|---|---|
| | Plant Exhaust Stack Releases | Turbine Generator Building Releases | Total | |
| H3 | 6.7E+02 | 7.4E+00 | 6.8E+02 | 1.2E+03 |
| C14 | 2.4E-01 | 2.8E-07 | 2.4E-01 | 2.8E-01 |
| N16 | - | - | - | - |
| Ar41 | 4.0E+00 | 7.7E-01 | 4.8E+00 | - |

1. The total does not include an adjustment factor to account for AOOs that adds an additional 0.071 Ci per year release to the cumulative non-tritium liquid effluent releases.

## Table A-4 Fuel Failure Data for U.S. Pressurized Water Reactors with Zirconium-Alloy Cladding

(Reference 7.2.20)

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**Enclosure 3:**

Affidavit of Carrie Fosaaen, AF-132449

**NuScale Power, LLC**

AFFIDAVIT of Carrie Fosaaen

I, Carrie Fosaaen, state as follows:

(1) I am the Senior Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the method by which NuScale develops its Radioactive Waste Management.

NuScale has performed significant research and evaluation to develop a basis for this method and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed report entitled [Final Safety Analysis Report, Chapter 11, "Radioactive Waste Management," Revision 0. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document. [Delete this line if presentation]

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon

the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 12/30/2022.

Carrie Fosaaen