November 13, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of "Effluent Release (GALE Replacement) Methodology and Results," TR-1116-52065, Revision 1

REFERENCES: Letter from NuScale Power, LLC to Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of Technical Reports Supporting the NuScale Design Certification Application (NRC Project No. 0769), dated December 30, 2016 (ML17005A112)

NuScale Power, LLC (NuScale) hereby submits Revision 1 of the "Effluent Release (GALE Replacement) Methodology and Results," (TR-1116-52065). The purpose of this submittal is to request that the NRC review and approve this revised report, which has been updated as a result of oral and written comments received during the NRC's Phase 1 audit and written Requests for Additional Information.

Enclosure 1 contains the proprietary version of the report entitled "Effluent Release (GALE Replacement) Methodology and Results." NuScale requests that the proprietary version 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 of the report entitled "Effluent Release (GALE Replacement) Methodology and Results."

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

If you have any questions, please contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

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Enclosure 1: "Effluent Release (GALE Replacement) Methodology and Results,"
TR-1116-52065-P, Revision 1, proprietary version

Enclosure 2: "Effluent Release (GALE Replacement) Methodology and Results,"
TR-1116-52065-NP, Revision 1, nonproprietary version

Enclosure 3: Affidavit of Thomas A. Bergman, AF-1018-62369

Enclosure 1:

"Effluent Release (GALE Replacement) Methodology and Results," TR-1116-52065-P, Revision 1, Proprietary version

Enclosure 2:

"Effluent Release (GALE Replacement) Methodology and Results," TR-1116-52065-P, Revision 1, Nonproprietary version

Licensing Technical Report

Effluent Release

(GALE Replacement)

Methodology and Results

November 2018

Revision 1

Docket: 52-048

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Abstract

This technical report describes the methodology used to calculate normal operation, including anticipated operational occurrences, annual radioactive gaseous and liquid effluents to the environment from an operating NuScale Power Plant. The application of this methodology demonstrates compliance with regulatory requirements for normal radioactive effluents. No exemptions from existing regulations related to radioactive effluents are requested. 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 plant design. The NuScale small modular reactor design is significantly smaller (a single NuScale Power Module (NPM) provides approximately five percent of the electrical output of a large PWR), 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 NuScale design is expandable with multiple NPMs within the overall plant envelope. While the majority of individual NuScale Power Plant system designs are similar to traditional PWRs, a few systems vary from the large PWRs, such as the use of integral helical coil steam generators.

The primary and secondary coolant isotopic distribution is in Table A-3. The total effluents are calculated to be 975 Ci of gaseous effluent and 1,114 Ci of liquid effluent, with tritium being the largest contributor to both. The isotopic distribution totals can be found in Table A-4.

Executive Summary

The NuScale Power Plant 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 of that time and does not appropriately address the NuScale Power Plant design. The NuScale Power Plant

- is significantly smaller - a single NuScale Power Module (NPM) provides approximately five percent of the electrical output.

- relies upon a significantly different passive design based on conduction, convection, gravity and natural circulation.

- is expandable with multiple 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 GALE code that are not appropriate for the NuScale Power Plant 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 NuScale Power Plant containing 12 NPMs. This report also includes specific in-plant source terms and results of effluent releases. The application of this methodology is used to demonstrate compliance with regulatory requirements, including a combination of site boundary isotopic concentrations and off-site dose consequence limits. No exemptions to effluent related regulations are requested.

The NuScale 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 is accomplished using conservative, yet realistically generated source terms, by evaluating radionuclide transport throughout reactor and other radioactive plant systems, and by evaluating effluent releases. Appropriate primary and secondary coolant concentrations of fission products, activated corrosion and wear products, and water activation products are calculated for the NuScale Power Plant. Source terms also include water activation products that are produced in the large reactor pool, which is a unique NuScale design feature.

One important input parameter in this methodology is the assumed fuel failure fraction. Industry operating experience over the past 25 years shows long-term and continuing reductions in fuel failures. In U.S. PWRs, the annual fuel failure fraction has been decreasing and 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), which is used for this analysis. Over 90 percent of U.S. nuclear power plants now experience no fuel failures. The NuScale design includes various design features that further mitigate fuel failure mechanisms. These design features are expected to further improve fuel performance. Based on the continued industry trend in fuel performance, a realistic, yet conservative, fuel failure fraction value is used in the calculation of fission product related source term effluents.

1.0   Introduction

1.1   Purpose

The purpose of this report is to describe the methodology used to calculate the NuScale Power Plant 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 NuScale Power Plant, assuming the combined effect of 12 operating NuScale Power Modules (NPMs) and AOOs. The report discusses the differences and similarities between the NUREG-0017 methodology and assumptions and the NuScale 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, which are explained and quantified. This report also discusses the similarities and differences in the NuScale design compared to existing pressurized water reactor (PWR) designs as they relate to effluent releases.

The scope does not include the calculation of site boundary radionuclide concentrations or doses to the public that result from the effluents. The report also does not include a discussion of the methodology used for the determination of personnel protection design features of the NuScale Power Plant. 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 Design Certification Application.

1.3   Abbreviations

Table 1-1   Abbreviations

Term	Definition
AOO	anticipated operational occurrence
BONAMI	Bondarenko AMPX Interpolator (code)
CES	containment evacuation system
CENTRM	continuous energy transport module (code)
CNV	containment vessel
CVCS	chemical and volume control system
DF	decontamination factor
EPRI	Electric Power Research Institute
Eq.	Equation
FSAR	Facility Safety Analysis Report
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
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
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
PCI	pellet-cladding interface
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)
SCC	stress corrosion-cracking
SCFM	standard cubic feet per minute
SG	steam generator

Term	Definition
TGB	Turbine Generator Building
TRITON	Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion (code)

2.0   Background

There are many similarities between the NuScale Power Plant design and currently operating PWRs with regard to radioactive effluents. The NuScale design houses 12 NPMs in a Reactor Building (RXB). Airborne releases in the RXB and in the Radioactive Waste Building (RWB) are gathered and processed by heating, ventilation and air conditioning (HVAC) systems 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 12 NPMs. Each NPM supplies steam to a dedicated turbine located in one of two Turbine Generator Buildings (TGBs). Each TGB contains up to six turbine-generators. Gaseous releases from the main condensers are removed by the condenser air ejector systems and are monitored and released via the TGBs to the environment. As a consequence, effluent release locations are essentially the same as for large PWRs.

There are important differences in the NuScale Power Plant design that influence effluent releases. The NuScale reactor 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 due to proximity of the steam generator 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 steel containment vessel (CNV), which is evacuated to a low pressure under normal operations. There are up to 12 NPMs per plant located in a large, common below grade reactor pool. The RXB encloses the NPMs and reactor pool. Refueling operations are performed underwater in the refueling and spent fuel areas of the large 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 large 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), was developed in the 1980s for evaluation of the large PWRs of that time and does not appropriately address the NuScale plant design. The NUREG-0017 methodology was developed using empirical data from existing large reactors and is still the current NRC-endorsed effluent release code. The NuScale Power Plant design

- is significantly smaller - a single NPM provides approximately five percent of the electrical output of current large PWRs.

- relies upon a significantly different passive design based on conduction, convection, gravity and natural circulation.

- is expandable up to 12 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 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. While the new ANSI/ANS 18.1-1999 standard was endorsed for use in applications to the NRC, newer versions of the GALE code have not been. Significant differences in NuScale plant 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 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 NuScale Power Plant 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 were 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 NuScale Power Plant design would result 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, along with one more parameter, steam flow, that represents the range of applicability for the secondary coolant system.

Table 2-1   GALE applicability range

Parameter	Units	GALE Applicability Range	NuScale Value
Thermal power	MW$_{th}$	3000 – 3800	160
Primary coolant mass	lb	500,000 – 600,000	103,000
Primary system letdown flow	lb/hr	32,000 – 42,000	10,920 nominal (20,160 maximum)
Shim bleed flow	lb/hr	250 – 1,000	21
Letdown cation demineralizer flow	lb/hr	≤ 7,500	0
Steam flow	lb/hr	13,000,000 – 17,000,000	530,000

The NuScale design is outside the range of these parameters, indicating that the GALE code is not appropriate for analysis of NuScale coolant activity concentrations or effluents. Values from NUREG-0017 are used, where appropriate, and where not, are explained in this report with justification to why each provides 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 NuScale methodology for effluents is separated into three major phases:

- production (water activation, 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 that realistic yet conservative source terms are generated for further evaluation.

As mentioned in Section 2.1, the GALE code includes some hard-coded parameters that do not reflect the NuScale design, such as the capacity factor. NuScale utilizes a higher, more conservative, and more appropriate capacity factor of 95 percent. The radionuclide list in GALE is also hard-coded, omitting a variety of nuclides, including environmentally

mobile nuclides such as I-129 and Tc-99. NuScale uses a more comprehensive list of isotopics that are carried forward throughout the evaluation of effluents. The list of isotopics is based on 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 (DCD) applications for the AP-1000 (Reference 7.2.20), U.S. EPR (Reference 7.2.21), US-APWR (Reference 7.2.22), and APR1400 (Reference 7.2.5). This comprehensive list of isotopes can be found in Table A-1 of Appendix A.

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 a change is justified, the NuScale 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 NuScale radioactive waste systems uses similar processes and methods to reduce radioactive effluent releases as are 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, are released to the environs as normal operations effluents. Figure 2-1 shows the general locations of effluent releases. Liquid effluents are consolidated in the liquid radioactive waste system (LRWS) and discharged in a controlled fashion while being mixed with the utility water system as a dilution source. Airborne releases from the RXB and RWB are combined to be released through one plant exhaust stack. Airborne releases from the TGB, constituting a small fraction of total effluents, are released directly and monitored through the secondary systems.

Figure 2-1   NuScale plant layout with release points identified

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 Facility Safety Analysis Report (FSAR) are used to demonstrate compliance with 10 CFR 20 Appendix B, as well as 10 CFR 20.1301-20.1302, "Radiation Dose Limits for Members of the Public" (Reference 7.2.27) 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 7.2.28).

The NuScale effluent release methodology presented in this report establishes a basis for compliance with applicable regulations, with no exemptions requested. Governing regulations and guidance include the following:

- 10 CFR 20 (radiation protection)
    - 10 CFR 20, Subpart D (public dose limits)
    - 10 CFR 20, Appendix B (effluent concentration limits)
- 10 CFR 50 (domestic licensing)
    - Appendix A (general design criteria)

- – Appendix I (public dose limits)
- 10 CFR 51 (environmental protection regulations)
- 10 CFR 52 (design certifications)
- 40 CFR 190 (environmental radiation protection standards)

Additional guidance considered is provided in the following:

- Standard Review Plans
  - – 11.1, 11.2, 11.3, 11.4, and 11.5
- NuScale DSRS
  - – 11.1, 11.2, 11.3, 11.4, 11.5, and 11.6
- Interim Staff Guidance
  - – DC/COL-ISG-5 (calculation of routine releases)
- NUREGs
  - – NUREG-0017 (calculation of releases PWRGALE)
- Regulatory Guides
  - – RG 1.109 (compliance with Appendix I)
  - – RG 1.112 (calculation of gaseous and liquid effluents)
  - – RG 1.206 (combined license applications)
- Industry standards
  - – ANSI/ANS 18.1-1999 (normal operation source terms)

3.0   Source Term Production

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

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

Each of these categories is discussed in detail below.

3.1   Water Activation Products

The NuScale 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 calculated to be insignificant. Each CNV is submerged within 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. These activation products are evaluated using a first-principle physics model as shown in Equation (Eq.) 3-1

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

where,

RR_x = number of reactions of type "x",

Φ_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 and are not used in downstream calculations.

3.1.1 Tritium

Tritium is usually one of the major effluent release contributors for PWRs. Tritium is primarily produced in the primary coolant by fission neutron capture resulting in several different reactions. Of those, the majority is produced by activation of soluble boron (Reference 7.2.23). In addition to the borated primary coolant, tritium production is also evaluated in the secondary coolant and borated reactor pool. Tritium production reactions are listed below in Eq. 3-2 through Eq. 3-9.

$$^{10}_{5}\text{B} + {}^{1}_{0}\text{n}_\text{f} \rightarrow 2 {}^{4}_{2}\alpha + {}^{3}_{1}\text{H} \qquad \text{Eq. 3-2}$$

$$^{10}_{5}\text{B} + {}^{1}_{0}\text{n}_\text{f} \rightarrow {}^{8}_{4}\text{Be} + {}^{3}_{1}\text{H} \qquad \text{Eq. 3-3}$$

$$^{10}_{5}\text{B} + {}^{1}_{0}\text{n}_\text{th} \rightarrow {}^{4}_{2}\alpha + {}^{7}_{3}\text{Li} \qquad \text{Eq. 3-4}$$

$$^{7}_{3}\text{Li} + {}^{1}_{0}\text{n}_\text{f} \rightarrow {}^{1}_{0}\text{n} + {}^{4}_{2}\alpha + {}^{3}_{1}\text{H} \qquad \text{Eq. 3-5}$$

$$^{7}_{3}\text{Li} + {}^{1}_{0}\text{n}_\text{f} \rightarrow {}^{5}_{2}\text{He} + {}^{3}_{1}\text{H} \qquad \text{Eq. 3-6}$$

$$^{11}_{5}\text{B} + {}^{1}_{0}\text{n}_\text{f} \rightarrow {}^{9}_{4}\text{Be} + {}^{3}_{1}\text{H} \qquad \text{Eq. 3-7}$$

$$^{6}_{3}\text{Li} + {}^{1}_{0}\text{n}_\text{th} \rightarrow {}^{4}_{2}\alpha + {}^{3}_{1}\text{H} \qquad \text{Eq. 3-8}$$

$$^{2}_{1}\text{H} + {}^{1}_{0}\text{n}_\text{th} \rightarrow {}^{0}_{0}\gamma + {}^{3}_{1}\text{H} \qquad \text{Eq. 3-9}$$

The largest boron letdown curve calculated for any planned cycle is assumed in this calculation to conservatively estimate the amount of tritium generated in the core. Tritium production based on all of the mechanisms in Eq. 3-2 through Eq. 3-9 above is calculated and the production over a two-year operating cycle is shown below in Figure 3-1. Buildup

of deuterium was also investigated and determined to be a negligible contribution to the overall tritium production or concentrations.
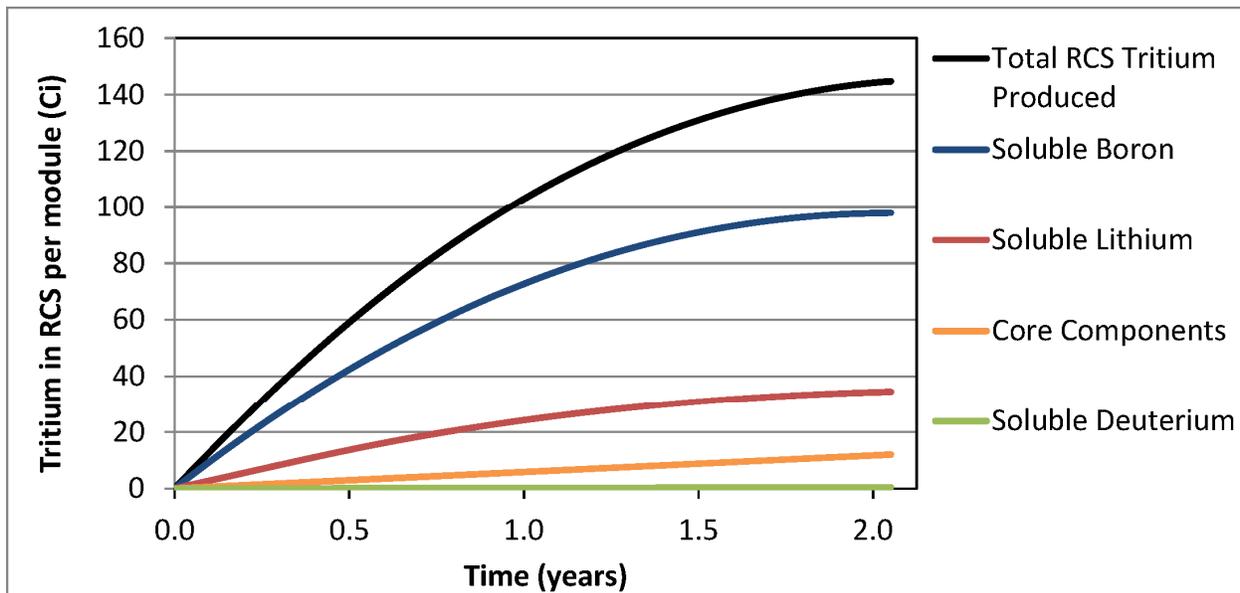
Figure 3-1   Time dependent NuScale isotopic tritium production breakdown in primary coolant

The NuScale calculated tritium production from soluble species (boron, lithium, and deuterium) is 65 Ci/yr per NPM in the primary coolant, which is 10 percent more than the Electric Power Research Institute (EPRI) value of 59 Ci/yr per NPM (Reference 7.2.23). The NuScale design also includes more water in the coolant per megawatt generated than a standard PWR. Combined with its higher capacity factor, the NuScale design has a substantial neutron flux for longer in a larger relative amount of coolant than a typical PWR, which 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 NuScale Power Plant. 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.
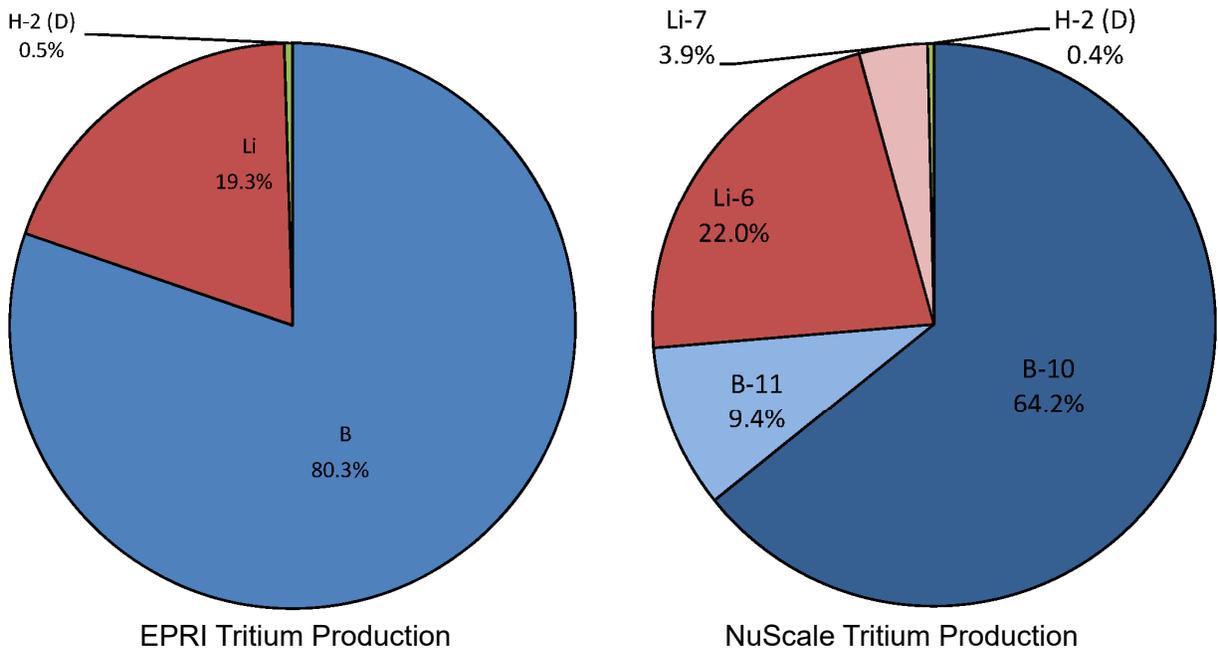
Figure 3-2   Total NuScale isotopic tritium production breakdown in primary coolant

Tritium is also produced by ternary fission of U-235. Only a small fraction of the total tritium produced in the fuel is diffused through the cladding into the coolant. The EPRI tritium management model (Reference 7.2.23, Table 7-2 on page 7-4) provides primary coolant tritium production values from fission. Scaling the tritium production rate for NuScale's power output provides an estimate of 5.8 Ci/yr per NPM coming from within the core components, such as fuel pins. Due to the low neutron flux, the direct tritium production through activation in the reactor pool and secondary coolant is negligible. Therefore, the total tritium production is 71 (65 + 6) Ci/yr per NPM compared to the EPRI value of 65 (59 + 6) Ci/yr per reactor prediction.

Tritium is a mobile radionuclide because it is chemically the same as protium (hydrogen with an atomic weight of 1) and bonds with water, typically as HTO. It cannot be removed from the water by filtering, so it has a DF of 1 for all cleanup systems. Tritium emits a beta particle with a half-life of 12.32 years. Therefore, it decays very little before being released. Once the tritium source term is generated, tritium is transported throughout the plant systems, until being released through both liquid and gaseous pathways. The total release rate of tritium is assumed to be 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 160 MW$_{th}$ reactor, that would equal 64 Ci/yr per module. A comparison of the NUREG-0017, EPRI, and NuScale values is shown below in Figure 3-3.
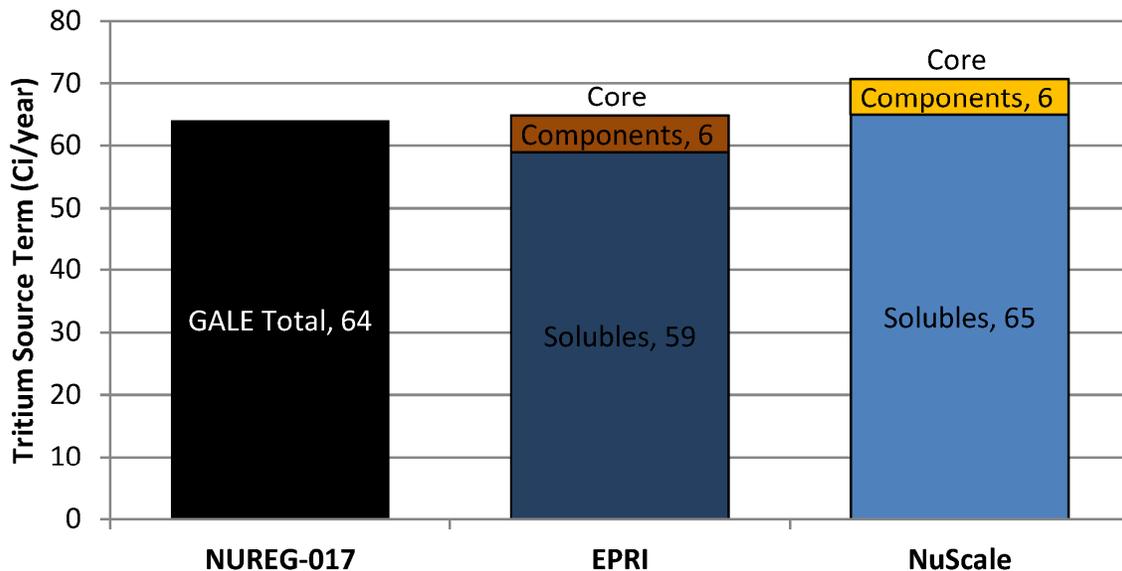
Figure 3-3   Comparison of GALE, EPRI, and NuScale yearly tritium production

3.1.2   Carbon-14

Carbon-14 (or radiocarbon) is primarily produced in reactor coolant during power operation. Carbon-14 can be produced in the primary coolant, secondary coolant, and reactor pool, taking several possible chemical forms. The chemistry of carbon-14 is complex and only two production reactions involving isotopes dissolved in water are significant in LWRs and for NuScale. These two reactions are listed below in Eq. 3-10 and Eq. 3-11:

$$\isotope[17][8]{O} + \isotope[1][0]{n}_{th} \rightarrow \isotope[4][2]{\alpha} + \isotope[14][6]{C} \qquad \text{Eq. 3-10}$$

$$\isotope[14][7]{N} + \isotope[1][0]{n}_{th} \rightarrow \isotope[1][1]{p} + \isotope[14][6]{C} \qquad \text{Eq. 3-11}$$

Nitrogen can be found both as 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 potential production of carbon-14 from the two reactions is calculated in all three water sources and found to be 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 is likely to be found in multiple chemical forms having different properties (affecting removal DFs, partition factors, etc.), 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 NuScale's much lower power production, smaller core volume, and smaller active fuel region, there is substantially less carbon-14 produced, and NuScale should be 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 1.0 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 be activated to form radioactive nitrogen (N-16). Nitrogen-16 is produced by neutron activation of oxygen by the reaction in Eq. 3-12:

$$^{16}_{8}O + \, ^{1}_{0}n_f \rightarrow \, ^{1}_{1}p + \, ^{16}_{7}N \qquad\qquad \text{Eq. 3-12}$$

The radioactive nitrogen (N-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 (6.13 MeV 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, N-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, Section 1.5.2.12.2 states that N-16 is not considered in the GALE code as an effluent. Transit times are longer in the NPM than traditional large PWRs due to the slower primary flow of natural circulation. The total reactor coolant system (RCS) loop transit time is approximately 69 seconds, which is almost ten half-lives of N-16, thus preventing buildup in the core. The N-16 concentration at various locations (e.g., at the bottom of the helical coil SG) within the RCS loop is calculated and discussed in Section 4.1.1.

3.1.4   Argon-41

Argon-41 is produced by neutron activation of argon-40, which is naturally found in air. The amount of argon in air is 0.934 percent and the production of Ar-41 is shown below in Eq. 3-13:

$$^{40}_{18}Ar + ^{1}_{0}n_{th} \rightarrow ^{0}_{0}\gamma + ^{41}_{18}Ar$$

Eq. 3-13

Radioactive argon-41 is an inert gas that is transformed into a stable isotope of potassium (K-41) through a relatively complex set of decay emissions (see Figure 3-4). Argon-41 decay primarily produces both a 1.2 MeV beta particle and a 1.3 MeV gamma ray, as shown in Figure 3-4 with a half-life of approximately 110 minutes.
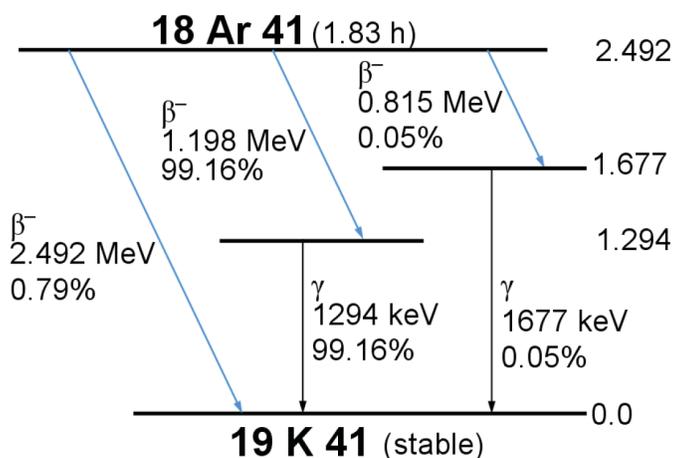
Figure 3-4.   Decay of argon-41 to potassium 41

In existing large PWRs, production of argon-41 has been dominated by the activation of natural argon-40 in the air in containment that surrounds the reactor vessel. Before operating the plant, primary and secondary coolant streams are purged of air, making production of argon in the coolant streams negligible in large PWRs. In the NuScale design, there is very little air surrounding the reactor vessel because the reactor vessel is surrounded by the steel CNV being maintained at a low pressure (less than 1 psia) during power operation. Argon-40 is calculated as being 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 has dissolved 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 NuScale'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 NuScale is well below the

argon-41 average releases for large PWRs. The NuScale calculated argon-41 production in the pool is 0.12 Ci/yr per NPM, a fraction of the average release from a large PWR.

Argon-40 can also be added to the primary coolant as a tracer for leaks through the helical coil SG into the secondary. If this is done, it is added to achieve a desired argon-41 activity concentration in the primary of 0.1 µCi/ml (Reference 7.2.40). Argon-40 addition is assumed in this analysis.

3.2   Corrosion and Wear Activation Products (CRUD)

3.2.1   Mechanism Overview

CRUD is formed 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 are activated by neutron interactions. When these alloys are exposed to the primary reactor coolant 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 are released into water as soluble metal ions (Reference 7.2.38). This way 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 very low; however, mass transfer still exists between the metal alloy and the primary coolant (Reference 7.2.39).

CRUD can manifest itself in a solid phase, either as metal oxide films or as micrometer-sized particles of metal oxide (Reference 7.2.38). It can also exist as hydrolyzed species of metal oxides in the aqueous phase. Species resulting from metallic corrosion are introduced into the coolant, where they are transported through convection onto other surfaces (Reference 7.2.39), including the surface of the fuel and of in-core structure materials. Thus, they are transformed into radioactive nuclides in the neutron flux, meaning that they become activated. 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 are released from fuel surface deposits by erosion and spalling caused by hydraulic shear forces or dissolution. Some activated products are released from in-core materials by dissolution and wear. They are then transported by water to all parts of the primary system, where they can become deposited 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 has prohibited first-principle physics models of CRUD.

3.2.2   Modeling CRUD

Models have been developed for the estimation of radioactivity buildup and corrosion product transport in LWRs. These include empirical and semi-empirical models containing coefficients that must be derived from experimental data or plant design data. Some   examples include: Japanese ACE, Korean CRUDTRAN, Czech DISER, Bulgarian MIGA, and French PACTOLE (Reference 7.2.39). All of these were developed using empirical data from the specific reactors whose behavior they were designed to

model. For this reason, they would not be applicable for reactors with different designs and geometries. In particular, the NPM has some characteristics that make it fundamentally different from other PWRs, and so none of the available models can accurately describe NPM behavior with regard to the activated corrosion products transport and deposition. Therefore, differences in the NPM design and the existing fleet preclude the use of any of these reactor-specific models.

Because there are no models available for the generation and transportation of corrosion and wear activation products, conservative empirical data is used. 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, those values are calculated directly, rather than calculating a production rate. This standard was specifically developed for the purposes of calculating, through adjustment factors, radionuclide concentrations in support of the design and licensing process. The data contained in ANS 18.1 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 to use in calculating anticipated corrosion and wear activation products in the primary coolant for the NuScale Power Plant design. The calculated CRUD source term numbers in the primary are shown below in Table 3-1.

Table 3-1   CRUD isotopic primary concentrations

Isotope	Primary Coolant Concentration (µCi/g)
Na24	9.1E-03
Cr51	5.2E-04
Mn54	2.7E-04
Fe55	2.0E-04
Fe59	5.0E-05
Co58	7.7E-04
Co60	8.8E-05
Ni-63	4.4E-05
Zn65	8.5E-05
Zr-95	6.5E-05
Ag-110m	2.2E-04
W187	4.6E-04

CRUD production has decreased over time as a result of incorporating lessons learned from the industry. The NuScale design follows modern guidelines for the reduction of CRUD and employs design features that minimize CRUD production. The reactor is designed to use the lowest possible cobalt and nickel materials appropriate for design conditions, along with lessons learned about reactor coolant system (RCS) chemistry control (e.g., highest pH). As a result, the values derived from the ANS standard are conservative for the NuScale plant.

Additionally, the RPV and the CNV are either made out of or are coated on both sides with stainless steel which is designed to survive the life of the plant in the borated water chemistry. As a result, minimal corrosion activation products are expected on the vessels themselves.

3.3   Fission Products

The spent fuel isotopic distribution and magnitude are developed using the industry standard, Standardized Computer Analyses for Licensing Evaluation (SCALE) computer code. To ensure conservative results, the NuScale methodology assumes a maximum peak burnup of 60 GWd/MtU for all fuel rods in the core. The fuel isotopics per assembly at that burnup are listed 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 NuScale methodology, the SCALE 6.1 modular code package, developed by Oak Ridge National Laboratory, is used for developing reactor core and primary coolant fission product source terms. Specifically, the Transport Rigor Implemented with Time-dependent Operation for Neutronic (TRITON) depletion 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, are used to generate radiation source terms for the NuScale fuel assemblies and various waste streams (Reference 7.2.25).

This industry standard commercial off-the-shelf software is used without modification by NuScale and has been extensively used in the evaluation of operating large LWRs. The SCALE code package is used in accordance with NuScale's Software Configuration Management Plan. 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 can be used to generate problem-dependent and burnup-dependent cross-sections as well as perform 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 is used to generate

burnup-dependent cross-sections for NuScale fuel assemblies for subsequent use in the ORIGEN-ARP depletion module.

The T-DEPL (time-depletion) sequence of the TRITON control module is called in order to generate problem-dependent (i.e., NuScale assembly-specific) and burnup-dependent cross-sections. The Continuous Energy Transport Module (CENTRM)-based option of the T-DEPL sequence is used, in which microscopic cross-sections are processed by the Bondarenko AMPX Interpolator (BONAMI) for the unresolved resonance energy range. Cross-sections from the continuous-energy library are processed by CENTRM for the resolved resonance energy range. CENTRM 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-sections 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 are also used to properly format the cross-section libraries at different stages of the processing.

A two-dimensional, discrete ordinates transport calculation is performed with the New Extended Step Characteristic-based Weighting Transport (NEWT) code module. The results of the transport calculation are post-processed by NEWT 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 complete depletion sequence is modeled by TRITON 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. Each problem-dependent and burnup-dependent nuclear data library is saved 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

ORIGEN-ARP 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. ORIGEN-S nuclear data libraries containing these cross-sections are prepared by the ORIGEN-ARP module 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 as compared to repeated use of TRITON. Many variations in fuel assembly irradiation history

can be modeled. For depletion calculations involving NuScale fuel assemblies, the ORIGEN-S nuclear data libraries are generated by the TRITON sequence, as described in the previous section.

The ORIGEN-S module of SCALE 6.1 is used to calculate the time-dependent isotopic concentrations of materials in a NuScale 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 so the proper nuclear data library is used 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 (see Section 4.8).

4.1   Primary Coolant Water System

The source term inputs to the primary coolant are discussed in Section 3.0. 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 NuScale's design facilitates recycling of primary water, the tritium concentration in process streams is calculated based on three recycling modes: 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 letdown tritium concentration from no recycling is used for the liquid effluent calculation.

The production rate (above in Figure 3-1) along with the cumulative water injection and bleed out of the primary coolant (below in Figure 4-1) can be used to develop a time-dependent balance of how much tritium is in the coolant versus how much has been bled out of the coolant (below in Figure 4-2). The letdown removal is based on the removal of primary coolant to control boron levels in the reactor and subsequently, reactivity control in the core. Primary coolant is let down from the reactor to the liquid radioactive waste system via the CVCS.
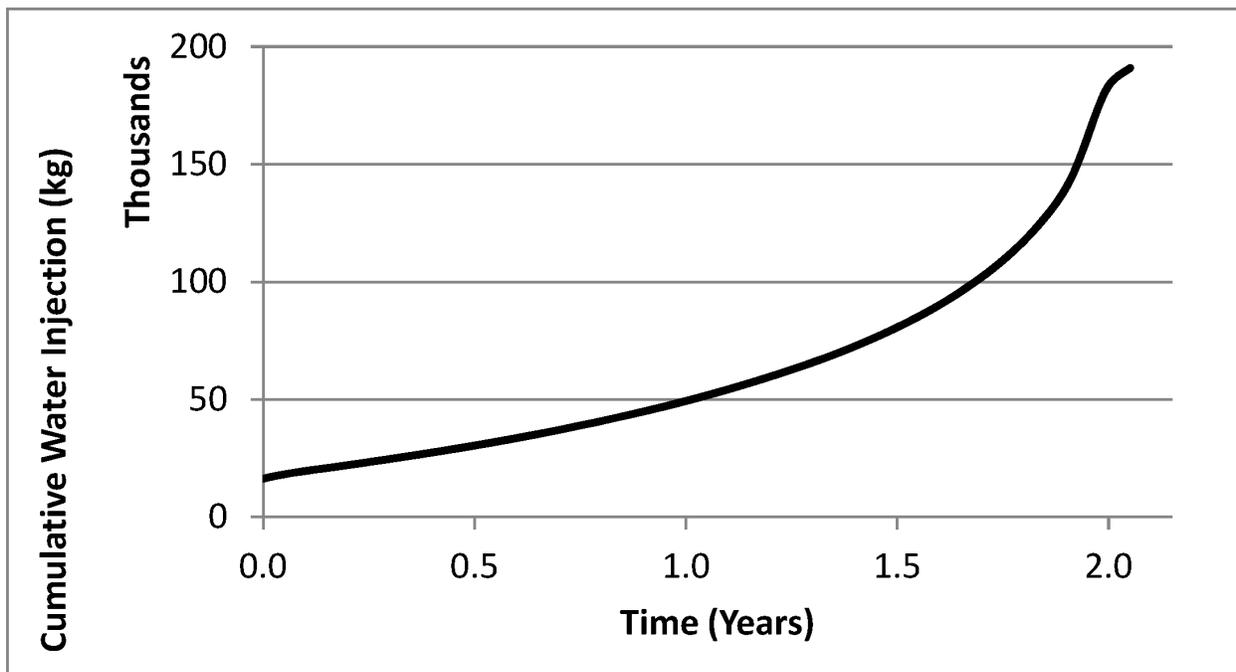
Figure 4-1   Water injection and bleed in the primary coolant
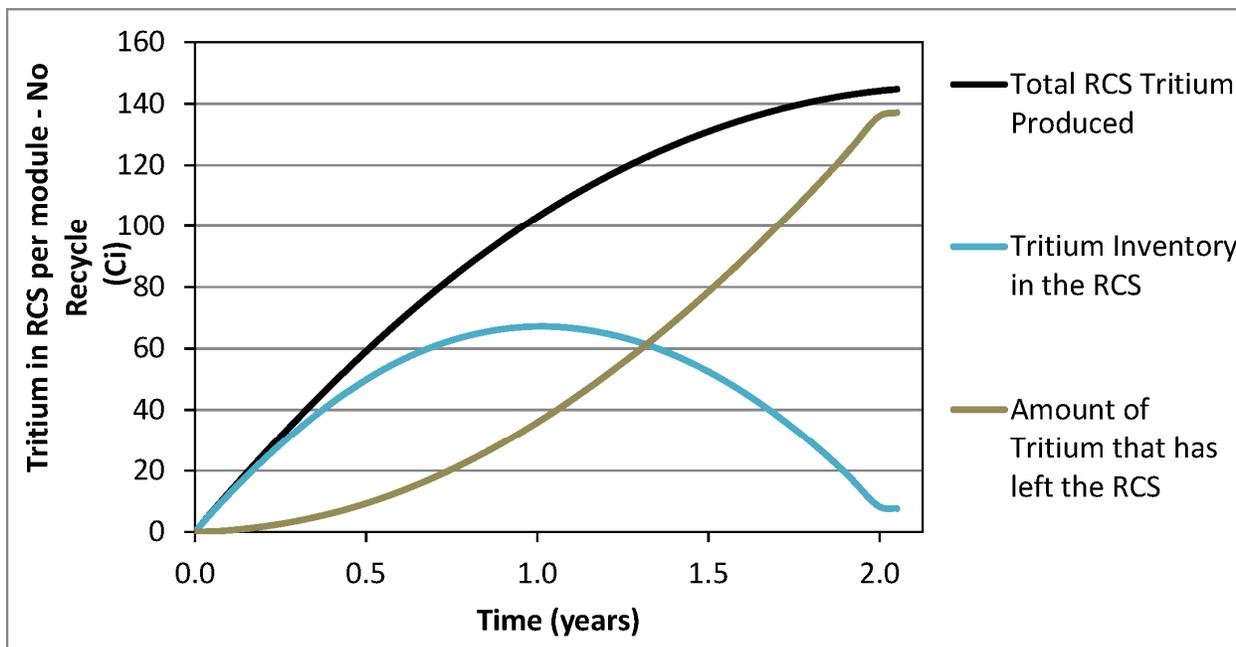
Figure 4-2   Tritium reactor coolant system balance

The tritium inventory curve can then be turned into a concentration and the time weighted average taken 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 0.96 µCi/ml. In addition, the average concentration of primary coolant being let down from the RCS is 0.72 µCi/ml. Note: these tritium concentrations are based on mode 1, no recycling of primary coolant.

Carbon-14 is not listed in NUREG-0017 as being in the primary or secondary coolant, although it is listed as being 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 NuScale core, the coolant flow rate is slow enough that N-16 has a substantial amount of decay during its transit time through the primary system. The best estimate, full power, total RCS transit time is approximately 69 seconds, almost ten half-lives of N-16. Therefore, by the time the N-16 transits to the integral helical coil SG in 16 seconds, its concentration is 15 µCi/g, which is much smaller than the NUREG-0017 value. Further, the N-16 concentration at the CVCS inlet with a transit time of 38 seconds, is an order of magnitude smaller than the helical coil SG at 1.8 µCi/g.

Due to the low concentration of N-16 at the helical coil SG, it is likely to be below the minimum detectable limit in the secondary in the case of a helical coil SG leak. Therefore, argon-40 could be added to the primary coolant 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 7.2.40).

4.1.2 CRUD

As discussed in Section 3.2, CRUD is calculated as primary coolant concentrations.

4.1.3 Fission Products

Fission product leakage into the primary coolant from the previously calculated fuel inventory is determined using 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 (References 7.2.20, 7.2.21, 7.2.22), as shown below in Table 4-1. These values are also conservative for the NuScale design because escape rate coefficients are a function of linear heat generation rate (LHGR) and NuScale has a lower LHGR value than larger PWRs.

Table 4-1   Fuel isotopic escape coefficients

Isotope	Value (s⁻¹)
Kr	6.5E-8
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

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.

NuScale's primary water volume-to-fuel ratio is much higher than a typical large PWR. Even if a proportional source term were assumed, this would result in a lower

concentration in the primary water due to greater dilution in the larger reactor coolant system 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, two for Cs and Rb, and 50 for other isotopes, per section 2.2.18.1 of NUREG-0017. There is no specific degasification of the primary coolant; therefore, noble gas removal through the pressurizer is neglected.

Although the concentration of individual isotopes in the primary coolant varies considerably over the operating cycle, the maximum calculated equilibrium activity is conservatively assumed to be present for the entire operating cycle, with the exception of some of the water activation products, which are treated separately. To calculate the activity of the isotopes, Eq. 4-1 is used:

$$A_{cp} = \frac{A_{sp}}{(\lambda_p + \lambda_L + \lambda_U)}$$

Eq. 4-1

where,

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

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

λ_p = decay constant of the parent nuclide,

λ_L = letdown removal coefficient through LRWS degasifiers, and

λ_U = removal coefficient for purification.

To calculate the activity of the ingrowth of daughter product isotopes, Eq. 4-2 is used:

$$A_{cd} = \frac{A_{cp}\lambda_d f_p}{(\lambda_d + \lambda_L + \lambda_U)}$$

Eq. 4-2

where,

λ_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-3 in Appendix A.

4.2   Secondary Coolant Water System

The concentration of radionuclides in the secondary system is determined by direct neutron activation of water products in the secondary coolant using reaction rate calculations and primary-to-secondary leakage. EPRI (Reference 7.2.24) has evaluated primary-to-secondary leakage in the industry and has developed SG management guidelines, which NuScale follows. As operational experience with the NuScale helical coil SGs is accumulated, modifications to EPRI guidelines may occur to optimize the mitigation of potential leakage. The direct activation of the secondary water impurities was calculated as being 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 total secondary coolant mass is conservatively underestimated to be 5.6E4 lbm by summing water mass values from the various main components of the secondary system (including both helical coil SGs and other components), and neglecting the mass of the fluids in the turbine and condenser. 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 Cs and Rb, 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 leakage terms from the secondary system are scaled from values provided in NUREG-0017 based on the low power level of each NPM (160 MW$_{th}$) compared to a traditional large PWR with an assumed power level of 3400 MW$_{th}$. Power scaling is appropriate because system capabilities are scaled to the size of the reactor. Main steam production is approximately proportional to core thermal power. Also, component sizing (e.g., pipe diameter) is related 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 (160 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	3.5
CVCS to RXB leak rate (lb/day/NPM)	160	7.5
TGB floor drains (gal/day/NPM)	7200	600
Secondary coolant sampling system drains (gal/day/NPM)	1400	120
Steam leak rate in TGB (lb/hr/NPM)	1700	80

The concentration of most of the radionuclides in the secondary coolant is found by a means similar to that of the primary coolant, as 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 Eq. 4-3:

$$P_S = A_P * L_{PS}$$

Eq. 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 leads to an equilibrium activity in the secondary coolant that is similar to Eq. 3-12. The equation that models the secondary activity is shown below in Eq. 4-4:

$$A_S = \frac{C_P * L_{PS}}{\lambda_d + \lambda_U}$$

Eq. 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,

λ_d = decay constant for the radionuclide, and

λ_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). 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 then dividing by the sum of the secondary flow rate and primary-to-secondary leak rate. This is given as

$$C_{Secondary} = C_{Primary} * \left(\frac{L_{PS}}{L_{PS} + \dot{m}_{secondary}} \right)$$ Eq. 4-5

Tritium, as an isotope of hydrogen, is chemically identical to hydrogen. This prevents typical methods of cleanup from working on tritium, which has 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 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 leak rate of coolant out of the secondary system. The secondary coolant concentration is

$$A_{Secondary} = C_{Primary} * \left(\frac{L_{PS}}{\lambda_d + \lambda_L} \right)$$ Eq. 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,

λ_d = decay constant for tritium, and

λ_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. NuScale calculates a tritium activity concentration in the secondary coolant of 1.8E-03 µCi/ml.

A comprehensive list of radionuclide activity concentrations in the secondary coolant is in Table A-3 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. This is consistent with current designs of large PWRs.

Leakage from the CVCS that goes to drain collections is assumed to be leaked before the demineralizers. The activity of the exiting water through letdown is determined following 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 Cs and Rb, and 50 for all others. The activity of the coolant after passing through the demineralizers is calculated with Eq. 4-7:

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

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 NuScale Power Plant 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 primary coolant is cleaned up by the CVCS. The cleanup time period is determined by assuming the primary coolant is sufficiently cleaned after a chemically-induced CRUD burst and an iodine spike to meet two dose rate targets. The first target is to maintain the accessible areas above and around the pool

under a dose rate of 2.5 mRem/hour. The second target is to maintain the doses one meter above the pool below 5 mRem/hour per EPRI guidelines (Reference 7.2.31). When the NPM is disassembled for refueling, the cleaned primary coolant is released into the refueling area of the pool.

Direct neutron activation of surrounding reactor pool water products from 12 operating NPMs has been calculated and determined to be negligible compared to 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 several orders of magnitude lower than that of the active core. Additionally, it drops off very quickly because the pool is borated to 1800 ppm boron.

Inadvertent impurities introduced into the pool were evaluated for activation potential. Resin backwash and breakthrough, lubricating oils, and hydraulic fluids have the potential to be introduced into the pool in small quantities. They are hydrocarbon chemicals that would not introduce any new radioisotopes into the effluent stream. The postulated impurities will 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 would be negligible neutron flux available for activation. These small quantities would be diluted throughout a very large pool water mass making their concentrations negligible to radioisotope production.

It is assumed that the activity released from a disassembled NPM in the refueling area of the pool will instantly mix homogenously throughout the entire pool volume (reactor pool and spent fuel pool). This is conservative for effluent release in that 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 will be released near the bottom of the refueling area of the pool and mix 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 would have been 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 is designed to reduce 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, and is governed by Eq. 4-8:

$$N(t) \ = N_o * \exp\left(-\left(\lambda + \frac{FR*\epsilon}{M}\right) * t\right)$$

Eq. 4-8

where,

N   = concentration of the given radionuclide,

λ   = decay constant for the given radionuclide,

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

ϵ   = efficiency of the cleanup system, between 0 (no effect) and 1 (perfectly efficient),

M   = mass of water, and

t   = time.

The exception to this treatment of radionuclides is tritium, which is not easily removed 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 Eq. 4-9:

$$Tritium\ Pool\ Inventory(\infty) = \frac{Production\ rate\ \left(\frac{Ci}{year}\right)}{\left(\lambda + \left(\frac{evaporation\ rate\left(\frac{g}{day}\right)*365.25\frac{days}{year}}{pool\ mass\ (g)}\right)\right)}$$ Eq. 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 tritium concentration in the pool from recycling primary water to the pool is used for the gaseous effluent calculation.

4.5   Airborne Activity

The main source of airborne activity in the NuScale Power Plant is evaporation from the RXB reactor pool and spent fuel pool. 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. NuScale evaluates a design-specific AOO of an inadvertent emergency core cooling system actuation, which results in pressurizing the CNV. 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 are used for primary coolant leaks, pool evaporation and secondary coolant leaks. Pool evaporation partition coefficient for iodine is 2000 based on a pool temperature of 120 degree-F and a pH of 5 (Reference 7.2.6). These values are steam/water partition factors designated for U-tube SGs and are used for pool evaporation for conservatism. These values are conservative because more radionuclides will 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 are contributed 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

This system is included in the gaseous radioactive waste system (GRWS) discussed in Section 4.6. Potential leakage from this system may result in airborne contamination. This system is evaluated at locations where the potential for airborne radioactivity exists.

4.5.2　Steam Generator Blowdown System

The NuScale helical coil SG is an integral, once-through helical-coil design. Since the secondary coolant circulates on the inside of the tubes, the NuScale 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 80 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 are pumped through water separator tanks and vented 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 NuScale Power Plant 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 NuScale, the CNV air is managed by the containment evacuation system (CES). The CES maintains the CNV under evacuated conditions. The CES normally vents to the Reactor Building HVAC system (RBVS), which is filtered with both a HEPA and charcoal filter. If the CES radiation monitors detect high radiation, the exhaust flow is redirected to the GRWS for processing. RPV leakage (0.47lbm/hr/NPM) into the CNV is removed by CES and routed via the GRWS decay beds for normal effluent. This method is based on the low volumetric flow rate of gases leaving the CNV via the CES vacuum pump, will provide sensitive radiation monitoring detection. Additionally 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/reactor leak rate of primary coolant into the auxiliary building. Assuming a NuScale plant has twelve times the primary leak rates of a larger PWR is overly conservative and unrealistic. NuScale modules are much smaller and have less inventory. The NuScale methodology linearly scales the 160 lbm/day/reactor leak rate value by thermal power to 7.5 lbm/day per NPM, for a total plant leakage of 90 lbm/day. The total plant leakage of 90 lbm/day is used to form the basis for the effluent airborne inventory in the RXB from primary leaks from the CVS. The NuScale RXB functions similar to the auxiliary building of a large PWR, in terms of release pathways from the CVS. The radionuclide activity removal mechanism is HEPA filters in the RBVS for particulate capture. The filtering efficiency is 99 percent per Section 2.2.11.1 of

NUREG-0017. Upon a high radiation signal in the RXB, the ventilation flow is also routed through charcoal filters before it is released. Charcoal filtration is not credited in the normal operation effluent calculations.

4.5.6 Steam Leakage from Secondary System

Steam leakage from the secondary system is assumed to occur in the TGB at the rate of 80 lbm/hour/NPM, for a total plant leak rate of 960 lbm/hour. This is linearly scaled by reactor thermal power to the 1700 lb/hr/reactor leak rate per unit 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 is mixed with the pool water, as previously described in Section 4.4. The time-weighted average pool source term over a year is assumed to be evaporating into the RXB airspace, which then goes through the RBVS and out the plant exhaust stack. The calculated total reactor pool evaporation rate is 2,100 lbm/hour.

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 is assumed to leak 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 assumed to be a steady state gas leak into the region below the bioshield for 30 hours, the period of time it takes the NPM to depressurize following an accident, based on containment transient thermal-hydraulic calculations. This leakage is quantified using the same method as the primary coolant leaks. This release is calculated to be 100 mCi of fission product gases into the RXB airspace.

4.6 Gaseous Radioactive Waste System

The GRWS is shared with all 12 NPMs in a single plant. 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 has passed through a gas cooler and a moisture separator. It is assumed that the guard bed does not collect or delay any radionuclides, so the input goes directly into the decay beds.

4.6.2   Activity Input to the Decay Beds

The main function of the charcoal decay beds is to 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 four decay beds in the GRWS, an A and B train. Typically, one train is operated at any given time. Each decay bed has a charcoal mass of 1150 pounds. The absorption coefficients and delay times for each bed are listed below in Table 4-3.

Table 4-3   Charcoal decay bed information

Element	Absorption Coefficient (cm^3/g)	Holdup Time (days/bed)
Argon	8.9	0.072
Krypton	60	0.48
Xenon	1400	11

Radionuclides present in the gaseous stream that are collected 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.

Since the charcoal filters are capable of collecting at least 90 percent of iodine species from the gaseous stream, it is assumed that 90 percent of the chemically similar bromine species are also collected.

Halogens produced as the result of parent-to-daughter decay chains are handled as follows. One-half of the halogen production is assumed to be 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 is collected 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 5 percent is released to the next bed.

For noble gas daughter products, the total daily production rate is accounted for and treated as 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

The LRWS is shared with all 12 NPMs in a single plant. It 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.

Decay of radionuclides, including development of daughter products, is calculated in radioactive waste process streams, 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 paths are shown below in Table 4-4.

Table 4-4   Processing paths for liquid radioactive waste

LCW Liquid Processing Path	HCW Liquid Processing Path
Granulated activated charcoal filter	Granulated activated charcoal filter
Tubular ultrafiltration skid	Tubular ultrafiltration skid
Reverse osmosis skid	Reverse osmosis skid
Cation demineralizer	
Anion demineralizer	
Mixed bed demineralizer	
Cesium demineralizer	
Antimony demineralizer	

Table 1-4 of NUREG-0017 provides DFs for common treatment systems for PWR liquid waste. The subset that are applicable to the NuScale Power Plant design have been applied without modification to their respective components in the LRWS. These DFs are reproduced in Table 4-5.

Table 4-5   Decontamination factors from NUREG-0017
(Reference 7.2.1)

Treatment System		Decontamination Factor		
Demineralizer		**Anion**	**Cs, Rb**	**Other Nuclides**
Mixed bed	Primary coolant letdown (CVCS)	100	2	50
	Radwaste (H+OH-)	100 (10)	2(10)	100(10)
Cation bed (any system)		1(1)	10(10)	10(10)
Anion bed (any system)		100 (10)	1(1)	1(1)
Reverse osmosis		10 (liquid wastes – all nuclides)		
Carbon bed for gaseous radioactive waste treatment		90% for iodines		
Evaporators (radwaste)		1000 for all except iodine, 100 for iodine		

The granular activated charcoal beds in the LRWS are not credited for liquid radioactive waste treatment of effluent source terms.

For the tubular ultrafiltration skids, the average DF value of 2.5 from IAEA Tech Doc 1336 (Reference 7.2.33) is applied for treatment of liquid effluent source terms.

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

Table 4-6   Expected liquid waste inputs

LRWS Input Source	Expected Input Rate	Expected Activity
LCW collection tank		
RXB and RWB equipment drains	2.9E+04 gpy 80 gpd	0.001 PCA
Pool leak detection	2.6E+05 gpy 700 gpd	Pool source term

LRWS Input Source	Expected Input Rate	Expected Activity
Other equipment drains	1.1E+04 gpy 29 gpd	0.093 PCA
Normal letdown (12 operating units)	2.7E+05 gpy 730 gpd	CVCS outlet
Letdown from cold shutdown to normal operating temperature (9 times per year)	3.5E+04 gpy	CVCS outlet
Letdown from hot standby to normal operating temperature (2 times per year)	2.8E+03 gpy	CVCS outlet
Degasification prior to shutdown (12 times per year)	2.0E+02 gpy	1.0 PCA
Fresh resin rinse mid-cycle (1 time per year)	1.8E+03 gpy	CVCS outlet
LCW Total	**6.0E+05 gpy**	
HCW collection tank		
RXB and RWB floor drains (via oil separator)	7.3E+04 gpy 200 gpd	0.1 PCA
RXB reactor component cooling water drain tank (via oil separator)	3.6E+01 gpy	0.001 PCA
Annex Building hot machine shop, decontamination room sump (via oil separator)	9.0E+04 gpy	0.01 PCA
RXB chemical drain tank (hot lab sink) (via oil separator)	8.8E+03 gpy 24 gpd	0.05 PCA
RXB chemical drain tank (CES sample tank & floor drains) (via oil separator)	4.4E+04 gpy 120 gpd	CES liquid
Pump seal leaks (via oil separator)	1.1E+04 gpy 30 gpd	0.1 PCA
Valve packing leaks (via oil separator)	6.6E+03 gpy 18 gpd	0.1 PCA
Groundwater and condensation (via oil separator)	2.5E+05 gpy 680 gpd	0.001 PCA
Equipment area decontamination (outside hot machine shop) (via oil separator)	1.5E+04 gpy 40 gpd	0.01 PCA
CVCS demineralizer sluice water (19 events per year)	2.9E+03 gpy	CVCS outlet
Pool cleanup system demineralizer sluice water (1.2 events per year)	3.6E+03 gpy	Pool source term

LRWS Input Source	Expected Input Rate	Expected Activity
LRWS demineralizer sluice water (except mixed bed; 4 events per year)	1.0E+03 gpy	CVCS outlet
LRWS mixed bed demineralizer sluice water (1 event per year)	4.5E+02 gpy	CVCS outlet
Granulated activated charcoal filter sluice water (0.2 events per year)	7.5E+01 gpy	CVCS outlet
Spent resin storage tank transfer water	3.4E+03 gpy	CVCS outlet
Phase separator tank transfer water	1.4E+03 gpy	CVCS outlet
Pool surge control storage tank dike water	9.1E+03 gpy	Pool source term
Miscellaneous clean-in-place water	2.0E+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
Pool boron adjustment	1.6E+04 gpy	Pool source term
HCW Total	**6.2E+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 contained within a system boundary, whereas HCW has come 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 total fill volume of the tanks is limited to 12,800 gallons to prevent spilling and sloshing of liquid. This methodology uses the 12,800 gallon volume as the batch volume to be transferred to the liquid radioactive waste processing skids for treatment. Once a tank has been filled, the contents are sent through the processing equipment. The radionuclide content is summed 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

- Annex Building decon room sump

- RWB floor drain sump

- RXB chemical drain tank

The oil separators process these liquids prior to 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 has been processed 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 has been processed through the HCW processing skids. To determine the radionuclide content, it is assumed that the sample tank is filled with HCW liquid that has been treated by the HCW reverse osmosis and tubular ultrafiltration skids.

4.8 Plant Effluent Release

Effluent releases from the NuScale Power Plant are determined by summing individual liquid and gaseous releases. Liquid and gaseous effluents are tracked and tabulated by isotope. Once the radionuclides have left the plant, the analysis of site boundary concentrations and doses are treated 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 NuScale Power Plant:

- 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 TGBs the gland seal steam condenser and system leaks are combined 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. The added decay times allow for a reduction in total activity coming from the plant, as described in Section 4.6.

As described in Section 4.5.7, 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 the peak activity concentrations are used for RXB ventilation design purposes, the gaseous effluent from reactor pool evaporation is determined based on a time-weighted annual average reactor pool water source term, pool water evaporation rate, airspace ventilation rate, and ventilation system filter efficiencies. To estimate the annual off-site dose from pool evaporation, an average airborne concentration is calculated using the Bevelacqua equation (Reference 7.2.41):

$$A(\infty) = P/K \qquad \qquad \text{Eq. 4-10}$$

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:

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

where,

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

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

The evaporated pool water is released to the environment via the RXB ventilation system at a constant rate equal to the pool room exhaust flow rate. Another contribution of airborne activity to the RXB ventilation system is primary system coolant leaks into the RXB originating from the CVCS, as described in Section 4.5.5. The airborne radionuclides captured by the building ventilation are monitored and released from the plant exhaust stack, after being filtered by HEPA filters.

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. Additional sources of gaseous effluent from the TGB include secondary coolant steam leaks and the condenser air removal systems, which are direct (unfiltered) ground releases. This is described in Section 4.5. The total gaseous effluent release from the plant is presented in Table A-4 of Appendix A.

4.8.2 Liquid Effluent Release

Liquid radioactive waste is collected and the HCW and LCW are sent 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 liquids are processed in the LRWS, sampled, and discharged through a common release point through the utility water system. The LRWS input volumes and processing parameters are described in Section 4.7.

An additional 0.090 Ci per year release to the cumulative non-tritium liquid effluent releases is added as an adjustment factor to account for AOOs. This value is linearly scaled with reactor thermal power (160 MWth * 12 vs. 3400 MWth) from the 0.16 Ci per year value from NUREG-0017.

The total liquid effluent release from the plant is presented in Table A-4 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. NuScale employs a first-principles calculation to determine 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 NuScale-assumed fuel failure fraction is used in the evaluation of fuel isotopics, with radionuclide release, buildup and removal, equilibrium concentrations in the primary coolant, and forms the basis for determining liquid and gaseous contributions.

The GALE code does not accurately represent the NuScale Power Plant design. Also, there is no NuScale operating history. Based on the similarities between the NuScale core and fuel design compared to existing PWRs, NuScale used industry operating experience. NuScale uses the same 17 x 17 PWR fuel assemblies, shorter in length, with AREVA M5™ cladding and low enriched U-235 uranium dioxide pellets in helium-backfilled and pressurized fuel rods. The applicability of the fuel analysis methods to the NuScale design is demonstrated in the NuScale technical report, Applicability of AREVA Fuel Methodology for the NuScale Design, TR-0116-20825-P Revision 1 (Reference 7.2.4). The selection of a fuel failure fraction was based on PWR fuel failure mechanisms and long term PWR fuel performance observed in the operating fleet. For each failure mechanism, an evaluation of how the NuScale design mitigates these fuel failure mechanisms is described.

5.1   Pressurized Water Reactor Fuel Failure Mechanisms

The PWR fuel failure mechanisms of grid fretting, debris, manufacturing defects, pellet-cladding interaction (PCI), stress corrosion-cracking (SCC), and cladding corrosion, have been studied over the last few decades. Both analytical and experimental data have been studied to better understand the underlying causes for these failure mechanisms. There have historically been "unknown" causes of fuel failures as well. Those failures classified as unknown have decreased over time and fuel failures have generally been attributed to known mechanisms. The relative average fraction of fuel failures for existing large PWRs for each known mechanism (1987–2010) is shown in Figure 5-1. The relative contribution of each of the fuel failure mechanisms discussed above is listed in Table A-5 of Appendix A. NuScale design features are expected to further reduce overall fuel failure fractions, resulting in changes to their relative contribution.
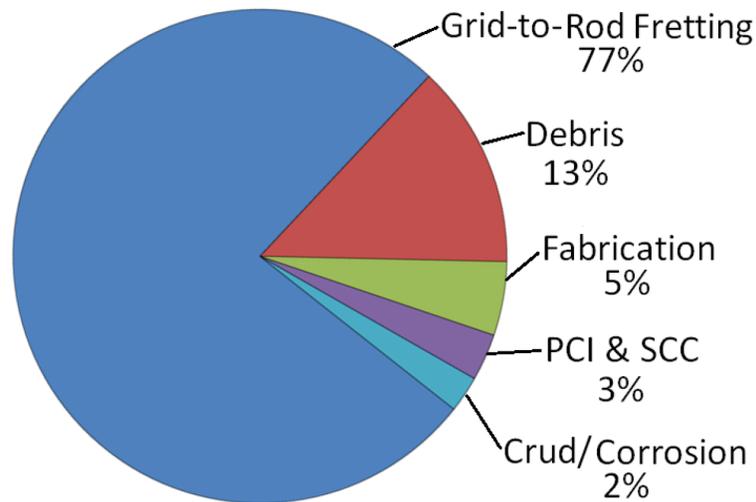
(References in Table A-4 of Appendix A)

Figure 5-1   Average known fuel failure mechanisms for zirconium alloy clad U.S. pressurized water reactors

5.1.1   Grid-to-Rod Fretting

Fretting of PWR rods typically occurs in the lower part of the fuel assembly where there are high cross-flows due to flow redistribution after passing through the bottom nozzle. These high cross-flow velocities are the main driver for fretting wear. The primary factors affecting the propensity for fretting to occur are rod and spacer materials, rod and spacer contact geometry and force, and cross-flow velocity.

The NuScale NuFuel-HTP2 fuel design incorporates a bottom HMP spacer grid fabricated out of precipitation hardened Inconel 718 alloy for increased strength and improved resistance to irradiation induced relaxation over the life of the assembly. The HMP spacer cell design incorporates eight lines of contact with the fuel rod providing an increased bearing surface that results in lower contact stresses. The combination of lowered contact stresses, lower relaxation, and eight lines of contact providing improved positional control reduces the potential for fretting wear of the fuel rods.

The remaining four HTP spacer grids made from Zr-4 are similar to the HMP. They provide the same eight lines of contact and reduced contact stresses that reduce the potential for fretting wear.

In the NuScale design, lower coolant flow rates (average coolant velocity of 2.7 ft/s as compared to 15.8 ft/s for the AP1000) due to the use of natural circulation, help mitigate this mechanism. Lower flow rates generate corresponding lower cross flows resulting in a lower potential for fretting.

5.1.2 Debris

The presence of debris in the RCS results in the potential for reactor coolant flow to lift this material into the core region. Debris can lodge in the interstitial spaces between fuel rods, and between fuel rods and spacers within the fuel assembly. Trapped debris driven by coolant flow turbulence can cause wear on the fuel rod potentially resulting in failure. To prevent debris related failures, the reactor pool cleanup system and operational foreign material exclusion practices reduce the potential for inclusion of debris into the reactor vessel during refueling. During operation, the low natural circulation primary flow rates in the NuScale design (average coolant velocity of 2.7 ft/s as compared to the AP1000 value of 15.8 ft/s) results in lifting less and smaller debris. This lowers the potential for fuel failures from debris fretting.

5.1.3 Fabrication

The NuScale design is based on a standard design AREVA 17 x 17 fuel assembly, which is approximately half the length of current large PWRs. The NuScale fuel uses the same fabrication techniques, quality assurance, and testing as the fuel assemblies fabricated by AREVA and irradiated in large PWRs. The resulting effects of fabrication-related fuel failures should be similar to the currently operating PWR fleet.

5.1.4 Pellet-Cladding-Interaction and Stress Corrosion-Cracking

Pellet-cladding interaction is a fuel failure mechanism driven by stresses resulting from mechanical contact of fuel pellets with the cladding in an aggressive chemical environment.

Stress corrosion-cracking is a mechanism by which fission product interaction with susceptible cladding material under tensile stress results in crack formation. This cracking can lead to cladding perforation by crack growth through the wall during a power ramp.

With burnup, pellet cracking and swelling induced by fission gas production, along with irradiation-induced cladding creep down, causes hard contact between the pellet and the cladding inner surface. A local increase in power over a short time period causes differential thermal expansion between the pellet and the cladding, and increased rod internal gas pressure, which results in cladding dimensional changes and additional stresses. The combination of the increased stress on the clad and SCC can result in fuel failure.

NuScale limits the potential PCI stress by adhering to conservative maneuvering rates when increasing reactor power. In addition, NuScale's lower linear heat generation rate and core average heat flux (NuScale fuel heat flux of 0.02 MBTU/hr-ft^2 as compared to the AP1000 value of 0.2 MBTU/hr-ft^2) reduces the fuel temperature, and correspondingly, the probability of SCC.

5.1.5 Cladding Corrosion

Cladding corrosion is caused by oxidation of the zirconium on the waterside of the cladding. There is a positive correlation between corrosion and increasing cladding and

coolant temperature. In addition, reactor data for PWR fuel rods shows accelerated corrosion at higher total fuel burnup (References 7.2.25 and 7.2.26).

For NuScale, the potential for cladding corrosion is reduced relative to the PWR operating fleet as a result of improved cladding materials (AREVA's M5™ cladding in the NuFuel HTP2™ fuel), water chemistry controls (EPRI guidelines in Reference 7.2.31), and lower end of life burnup than existing large PWRs.

5.2   US Pressurized Water Reactor Fuel Failure History

In the early 1970s, U.S. commercial PWR measured fuel failure fractions were on the order of 0.1 to 1.0 percent (1,000 to 10,000 rods/million) due to mechanisms that were not well understood, including issues such as pellet densification, SCC, and clad collapse. These phenomena were combined with manufacturing defects resulting in many failures. By the late 1970s, the number of failures dropped as clad collapse and hydride failure mechanisms were better understood, and appropriate fuel design, operational, and manufacturing changes were instituted (Reference 7.2.12).

Design and operational changes such as improved manufacturing quality, higher fuel rod internal helium pressurization, and better primary coolant system water chemistry were implemented to further reduce the effects of various failure mechanisms. (Reference 7.2.13)

Tables 2-9 and 2-10 of NUREG-0017 present primary coolant radionuclide concentration data from PWRs during the time period from 1971 to 1981. This corresponds to the early operation of nuclear power plants in which many fuel failure mechanisms were not well understood and resulted in more significant fuel failure fractions than have occurred in the last two decades.

Higher fuel duty in the 1980s in terms of fuel surface heat flux, linear heat generation rates and greater burnups introduced other fuel failure mechanisms. Improvements in manufacturing, new cladding materials, lower fuel assembly nozzle filters, and primary coolant system chemistry have been used to minimize these failure mechanisms. Identified causes of fuel failures in the 21st century have been categorized as grid fretting, debris, fabrication defects, CRUD, cladding corrosion, PCI, and SCC (References 7.2.14, 7.2.15, 7.2.16).

A literature search of fuel failure related data was conducted for US PWRs with zirconium-clad fuel (References 7.2.8, 7.2.9, 7.2.10, 7.2.12, 7.2.13, 7.2.17, and 7.2.18). NuScale analyzed all of the data that was found. While the literature search qualitatively informed this methodology and showed a clear trend of reductions in fuel failure over time, the definitive numerical basis for fuel failure rates comes from EPRI's "PWR Fuel Rod Failure Rate Analysis" Report (Reference 7.2.32). Data from this report is presented in Table A-6. A summary of the data from this report representing the time period 2000–2016 highlighting the highest and lowest values of failed fuel fraction is presented in Table 5-1.

Table 5-1   Fuel failure values

(Reference 7.2.32)

{{

}}[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)], which was in {{   }}[2(a),(c)] (Reference 7.2.10). The highest data point is 66 rods per million (0.0066 percent), which was in {{   }}[2(a),(c)] (Reference 7.2.13). For comparison, NUREG-0017, which PWRGALE-86 was based on, was written 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. PWRGALE-09 was benchmarked against operational reactor data from 2005 to 2010 (Reference 7.2.19). The average fuel failure fraction from that time period was {{   }}[2(a),(c)]. Figure 5-2 shows that the industry trend of reducing fuel failures has continued in recent years. The percentage of U.S. LWRs with no fuel failures has improved to 95 percent as of 2014.
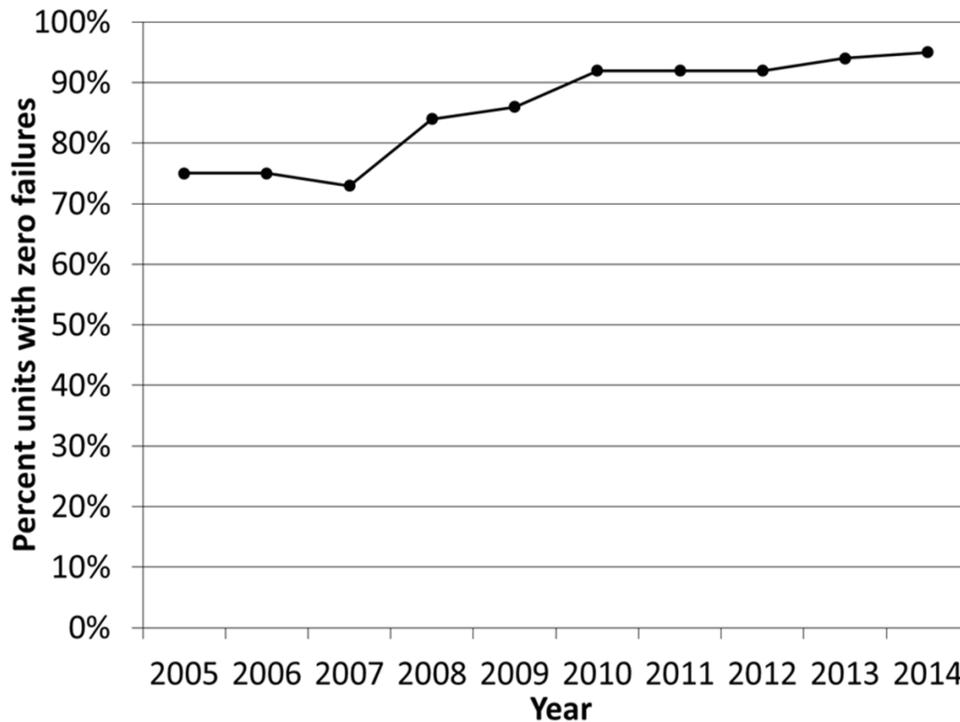
(Reference 7.2.29)

Figure 5-2   Percentage of U.S. power reactors with zero fuel defects

The NRC 2010 annual report on the Radioactive Effluents from Nuclear Power Plants (Reference 7.2.30) shows the importance of fuel failures to effluent releases. Both liquid and gaseous effluents have decreased orders of magnitude since the mid-1970s and for both, the report states, "One of the primary contributors to the reduction in … effluents is improved fuel integrity." Plots from this report showing this positive trend are shown below in Figure 5-3 and Figure 5-4.
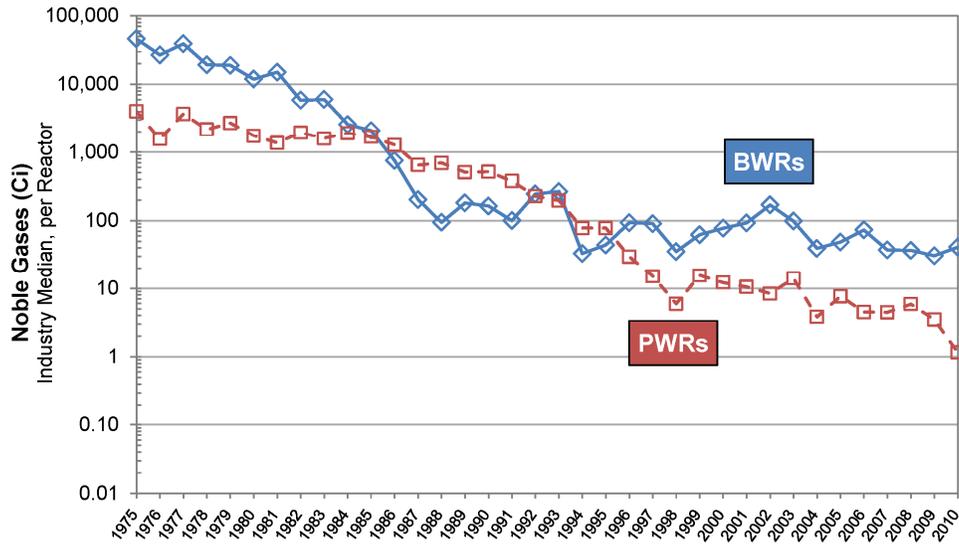
(Reference 7.2.30)

Figure 5-3   Gaseous effluent release data for U.S. pressurized water reactors and boiling water reactors, 1975 through 2010
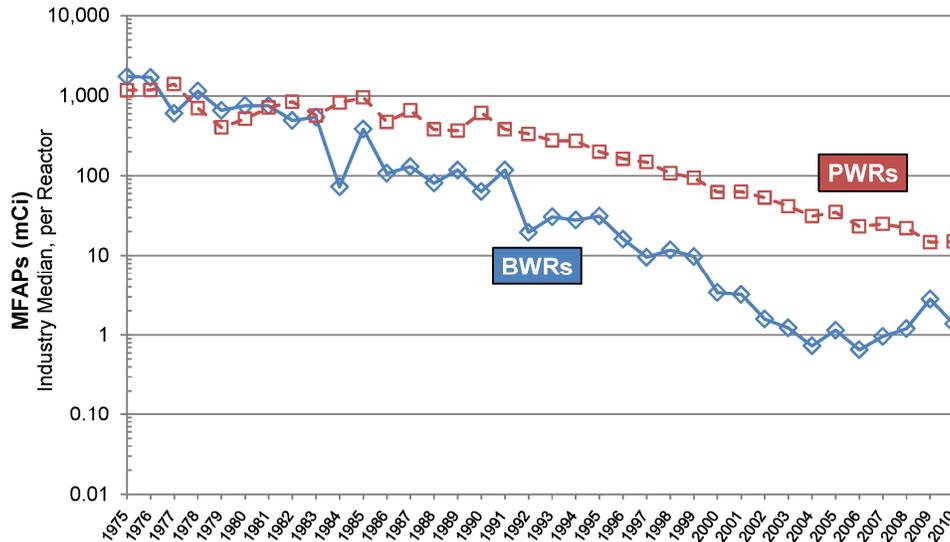
(Reference 7.2.30)

Figure 5-4   Liquid effluent release data for U.S. pressurized water reactors and boiling water reactors, 1975 through 2010

5.3   Fuel Failure Fraction

One input into effluent determination is an assumed fuel failure fraction, which replaces GALE empirical data. The long-term industry trend on improved fuel performance is well defined and highlights the continuing improvement. NuScale design features further mitigate fuel failure mechanisms and should continue the trend in fuel performance improvement.

There are four conclusions regarding fuel failure and effluent determination for this methodology:

1. 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 {{
    }}[2(a),(c)] of 66 rods per million (0.0066 percent).

2. More than 90 percent of U.S. nuclear power plants now experience no fuel failures.

3. The NuScale design includes features that further mitigate fuel failure mechanisms.

4. NuScale uses a realistic and conservative fuel failure fraction based on industry performance of 0.0066 percent (66 rods per million) for fission product-related effluent releases. A different, larger value is used for design basis shielding analysis. Design basis shielding analysis is not in the scope of this technical report.

6.0   Summary and Conclusion

The NuScale Power Plant 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 NuScale design, the GALE code is not representative of the NuScale design and does not accurately estimate NuScale effluent releases. This results in the need for NuScale to develop an alternate "GALE replacement" methodology. The NuScale effluent release methodology described in this report is based on compliance with applicable regulations, with no exemptions needed.

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

Table 6-1   Primary contributors and methodology employed for effluents

Primary Contributors	NuScale Methodology
Water activation products • H-3 (tritium) • C-14 (radiocarbon) • N-16 • Ar-41	• Calculations based on first-principles physics (Including primary coolant, secondary coolant, and the reactor pool.)
Activated corrosion and wear products (CRUD)	• Recent large PWR operating data • Lessons learned
Fission products (failed-fuel related)	• Calculations based on first-principles physics • Recent large PWR operating data

The primary and secondary coolant isotopic distribution is in Table A-3. The total effluents are calculated to be 975 Ci of gaseous effluent and 1,114 Ci of liquid effluent, with tritium being the largest contributor to both. The isotopic distribution totals are found in Table A-4.

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   TR-0116-20825, Applicability of AREVA Fuel Methodology for the NuScale Design, Revision 1.

7.2.5   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

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

7.2.7   Reference not used.

7.2.8   Crawford, D., "LWR Fuel Performance (with Emphasis on BWR Fuel)," Global Nuclear Fuel (GNF), June 3, 2009.

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

7.2.10 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.

7.2.11 American Nuclear Society, "Kurt Edsinger: EPRI and the zero fuel failure program," Nuclear News, pp. 40-43, December 2010.

7.2.12 Garzarolli, F., R. von Jan, and H. Stehle, "The Main Causes of Fuel Element Failure in Water-Cooled Power Reactors," Atomic Energy Review (AER), 1979.

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

7.2.14 International Atomic Energy Agency, "Results of the IAEA Study of Fuel Failures in Water Cooled Reactors in 2006-2010," Presentation at Technical Working Group on Fuel Performance and Technology (TWGFPT) Meeting, April 24, 2012.

7.2.15 Nuclear Energy Institute, "Fuel Failure Phaseout," September 1, 2008.

7.2.16 Deshon, J., "Fuel Reliability Trends Strongly Moving in Right Direction," Electric Power Research Institute, Palo Alto, CA, 2013.

7.2.17 International Atomic Energy Agency, "Fuel Failure in Normal Operation of Water Reactors: Experience, Mechanisms and Management," IAEA-TECDOC-709, June 1993.

7.2.18 U.S. Nuclear Regulatory Commission, "Nuclear Fuel Performance," Office of Nuclear Regulatory Research and Office of Nuclear Reactor Regulation Presentation, February 24, 2005, Agencywide Document Access and Management System (ADAMS) Accession No. ML050560020.

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

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

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

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

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

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

7.2.25 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.

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

7.2.27 *U.S. Code of Federal Regulations*, "Standards for Protection against Radiation," Part 20, Chapter 1, Title 10, "Energy," (10 CFR 20).

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

7.2.29 American Nuclear Society, "2014 Performance Indicators issued for U.S. Power Reactors," Nuclear News, June 2015.

7.2.30 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

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

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

7.2.33 International Atomic Energy Agency, "Combined methods for liquid radioactive waste treatment: Final report of a co-ordinated research project 1997-2001," IAEA-TECDOC-1336, Vienna, Austria, February 2003.

7.2.34 Reference not used.

7.2.35 Reference not used.

7.2.36 Reference not used.

7.2.37 Reference not used.

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

7.2.39 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.

7.2.40 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.

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

Appendix A.   Summary Tables

Table A-1   NuScale source term isotopes list and source documents

NuScale Isotope List	AP1000[1], U.S. EPR[2], US-APWR[3], APR1400[4]	ANSI/ANS-18.1-1999[5]	NUREG-0017 PWRGALE-09[6]
Fission Products			
Noble Gases	*Noble Gases*	*Class 1*	*Noble Gases*
Kr-83m	X		
Kr-85m	X	X	X
Kr-85	X	X	X
Kr-87	X	X	X
Kr-88	X	X	X
Kr-89	X		
Xe-131m	X	X	X
Xe-133m	X	X	X
Xe-133	X	X	X
Xe-135m	X	X	X
Xe-135	X	X	X
Xe-137	X	X	X
Xe-138	X	X	X
Halogens	*Halogens*	*Class 2*	*Iodine*
Br-82	X		
Br-83	X		
Br-84	X	X	
Br-85	X		
I-129	X		
I-130	X		
I-131	X	X	X
I-132	X	X	X

[1] AP1000 DCD (Reference 7.2.20)
[2] US EPR DCD (Reference 7.2.21)
[3] US-APWR DCD (Reference 7.2.22)
[4] APR1400 DCD (Reference 7.2.5)
[5] ANSI/ANS-18.1-1999 (Reference 7.2.2)
[6] PWRGALE-09 (Reference 7.2.19)

NuScale Isotope List	AP1000[1], U.S. EPR[2], US-APWR[3], APR1400[4]	ANSI/ANS-18.1-1999[5]	NUREG-0017 PWRGALE-09[6]
I-133	X	X	X
I-134	X	X	
I-135	X	X	X
Rubidium, Cesium	*Rubidium, Cesium*	*Class 3*	*Fission Products*
Rb-86m	X		
Rb-86	X		
Rb-88	X	X	
Rb-89	X		
Cs-132	X		
Cs-134	X	X	X
Cs-135m	X		
Cs-136	X	X	X
Cs-137	X	X	X
Cs-138	X		
Other FPs	*Miscellaneous*	*Class 6*	*Fission Products*
P-32	X		X
Co-57	X		X
Sr-89	X	X	X
Sr-90	X	X	X
Sr-91	X	X	
Sr-92	X		
Y-90	X		
Y-91m	X	X	
Y-91	X	X	X
Y-92	X		
Y-93	X	X	
Zr-97	X		
Nb-95	X	X	X
Mo-99	X	X	X
Mo-101	X		
Tc-99m	X	X	
Tc-99	X		
Ru-103	X	X	X

NuScale Isotope List	AP1000[1], U.S. EPR[2], US-APWR[3], APR1400[4]	ANSI/ANS- 18.1-1999[5]	NUREG-0017 PWRGALE-09[6]
Ru-105	X		
Ru-106	X	X	X
Rh-103m	X		X
Rh-105	X		
Rh-106	X		X
Ag-110	X	X	X
Sb-124	X		X
Sb-125	X		X
Sb-127	X		
Sb-129	X		
Te-125m	X		
Te-127m	X		
Te-127	X		
Te-129m	X	X	
Te-129	X	X	
Te-131m	X	X	
Te-131	X	X	
Te-132	X	X	
Te-133m	X		
Te-134	X		
Ba-137m	X	X	X
Ba-139	X		
Ba-140	X	X	X
La-140	X	X	X
La-141	X		
La-142	X		
Ce-141	X	X	X
Ce-143	X	X	
Ce-144	X	X	X
Pr-143	X		
Pr-144	X		
Np-239		X	

NuScale Isotope List	AP1000[1], U.S. EPR[2], US-APWR[3], APR1400[4]	ANSI/ANS-18.1-1999[5]	NUREG-0017 PWRGALE-09[6]
Corrosion Activation Products - CRUD	Corrosion Activation Products	Class 6	Corrosion Activation Products
Na-24	X	X	X
Cr-51	X	X	X
Mn-54	X	X	X
Fe-55	X	X	X
Fe-59	X	X	X
Co-58	X	X	X
Co-60	X	X	X
Ni-63	X		X
Zn-65	X	X	X
Zr-95	X	X	X
Ag-110m	X	X	X
W-187	X	X	
Water Activation Products	H-3, C-14, N-16, Ar-41	Classes 4-5	H-3, C-14, Ar-41
H-3	X	X	X
C-14	X		X
N-16	X	X	
Ar-41	X		X

Table A-2   Maximum fuel isotopics per assembly (Ci)

Radionuclide	Assembly Activity (Ci)
Noble Gases	
Kr83m	1.3E+04
Kr85m	2.6E+04
Kr85	3.6E+03
Kr87	5.0E+04
Kr88	6.6E+04
Kr89	8.1E+04
Xe131m	1.6E+03
Xe133m	7.8E+03
Xe133	2.4E+05
Xe135m	5.7E+04
Xe135	9.6E+04
Xe137	2.1E+05
Xe138	2.0E+05
Halogens	
Br82	6.9E+02
Br83	1.3E+04
Br84	2.1E+04
Br85	2.6E+04
I129	1.4E-02
I130	7.2E+03
I131	1.3E+05
I132	1.8E+05
I133	2.4E+05
I134	2.7E+05
I135	2.3E+05
Rubidium, Cesium	
Rb86m	5.4E+01
Rb86	4.3E+02
Rb88	6.7E+04
Rb89	8.8E+04
Cs132	8.7E+00
Cs134	7.2E+04
Cs135m	8.6E+02

Radionuclide	Assembly Activity (Ci)
Cs136	1.6E+04
Cs137	4.4E+04
Cs138	2.2E+05
Other FPs	
P32	1.9E+01
Co57	1.4E-01
Sr89	9.1E+04
Sr90	3.1E+04
Sr91	1.2E+05
Sr92	1.3E+05
Y90	3.1E+04
Y91m	7.0E+04
Y91	1.2E+05
Y92	1.3E+05
Y93	1.6E+05
Zr97	1.9E+05
Nb95	1.8E+05
Mo99	2.2E+05
Mo101	2.1E+05
Tc99m	1.9E+05
Tc99	5.7E+00
Ru103	2.4E+05
Ru105	2.0E+05
Ru106	1.5E+05
Rh103m	2.4E+05
Rh105	1.8E+05
Rh106	1.7E+05
Ag110	6.1E+04
Sb124	3.5E+02
Sb125	3.1E+03
Sb127	1.4E+04
Sb129	4.1E+04
Te125m	7.3E+02
Te127m	2.3E+03
Te127	1.4E+04
Te129m	6.7E+03

Radionuclide	Assembly Activity (Ci)
Te129	4.0E+04
Te131m	2.7E+04
Te131	1.1E+05
Te132	1.7E+05
Te133m	1.1E+05
Te134	2.0E+05
Ba137m	4.2E+04
Ba139	2.1E+05
Ba140	2.0E+05
La140	2.1E+05
La141	1.9E+05
La142	1.8E+05
Ce141	1.9E+05
Ce143	1.7E+05
Ce144	1.6E+05
Pr143	1.7E+05
Pr144	1.6E+05
Np239	3.4E+06

Table A-3   Primary and secondary coolant radionuclide activity concentrations

Radionuclide	Primary Activity (µCi/g)	Secondary Activity (µCi/g)
Noble Gases		
Kr83m	4.9E-04	1.4E-10
Kr85m	2.1E-03	5.7E-10
Kr85	1.8E-01	5.0E-08
Kr87	1.1E-03	3.1E-10
Kr88	3.3E-03	9.1E-10
Kr89	7.5E-05	2.1E-11
Xe131m	7.4E-03	2.1E-09
Xe133m	7.2E-03	2.0E-09
Xe133	5.3E-01	1.5E-07
Xe135m	7.0E-04	1.9E-10
Xe135	1.8E-02	5.1E-09
Xe137	2.4E-04	6.7E-11
Xe138	8.3E-04	2.3E-10
Halogens		
Br82	1.4E-05	3.8E-12
Br83	7.8E-05	2.1E-11
Br84	3.6E-05	8.9E-12
Br85	4.4E-06	4.9E-13
I129	3.4E-10	9.4E-17
I130	1.1E-04	3.0E-11
I131	2.8E-03	7.9E-10
I132	1.3E-03	3.5E-10
I133	4.3E-03	1.2E-09
I134	7.6E-04	2.0E-10
I135	2.7E-03	7.4E-10
Rubidium, Cesium		
Rb86m	3.2E-09	1.7E-16
Rb86	1.9E-05	5.9E-12
Rb88	3.3E-03	7.9E-10
Rb89	1.5E-04	3.5E-11
Cs132	3.7E-07	1.1E-13
Cs134	3.3E-03	1.0E-09
Cs135m	2.5E-06	7.1E-13

Radionuclide	Primary Activity (µCi/g)	Secondary Activity (µCi/g)
Cs136	7.0E-04	2.2E-10
Cs137	2.0E-03	6.2E-10
Cs138	1.2E-03	3.2E-10
Other FPs		
P32	5.5E-11	1.5E-17
Co57	4.1E-13	1.1E-19
Sr89	2.5E-06	6.8E-13
Sr90	5.5E-07	1.5E-13
Sr91	1.3E-06	3.5E-13
Sr92	6.8E-07	1.9E-13
Y90	1.3E-07	3.7E-14
Y91m	6.8E-07	1.8E-13
Y91	3.6E-07	9.9E-14
Y92	5.8E-07	1.6E-13
Y93	2.7E-07	7.5E-14
Zr97	4.0E-07	1.1E-13
Nb95	1.0E-06	2.9E-13
Mo99	7.2E-04	2.0E-10
Mo101	2.7E-05	5.8E-12
Tc99m	6.6E-04	1.8E-10
Tc99	2.1E-08	5.7E-15
Ru103	6.9E-07	1.9E-13
Ru105	2.3E-07	6.2E-14
Ru106	4.5E-07	1.2E-13
Rh103m	6.8E-07	1.8E-13
Rh105	4.8E-07	1.3E-13
Rh106	4.5E-07	1.3E-14
Ag110	3.2E-06	7.6E-14
Sb124	1.0E-09	2.8E-16
Sb125	8.9E-09	2.5E-15
Sb127	3.9E-08	1.1E-14
Sb129	4.8E-08	1.3E-14
Te125m	1.3E-06	3.7E-13
Te127m	4.2E-06	1.2E-12
Te127	1.7E-05	4.6E-12
Te129m	1.2E-05	3.4E-12

Radionuclide	Primary Activity (µCi/g)	Secondary Activity (µCi/g)
Te129	1.7E-05	4.5E-12
Te131m	4.0E-05	1.1E-11
Te131	1.9E-05	4.6E-12
Te132	2.9E-04	8.0E-11
Te133m	2.5E-05	6.4E-12
Te134	3.5E-05	8.8E-12
Ba137m	1.9E-03	2.0E-10
Ba139	6.5E-07	1.7E-13
Ba140	3.5E-06	9.9E-13
La140	1.0E-06	2.9E-13
La141	2.0E-07	5.5E-14
La142	9.6E-08	2.6E-14
Ce141	5.5E-07	1.5E-13
Ce143	4.1E-07	1.2E-13
Ce144	4.6E-07	1.3E-13
Pr143	4.9E-07	1.4E-13
Pr144	4.5E-07	1.0E-13
Np239	8.7E-06	2.4E-12
Corrosion Activation Products - CRUD		
Na24	9.1E-03	2.5E-09
Cr51	5.2E-04	1.4E-10
Mn54	2.7E-04	7.5E-11
Fe55	2.0E-04	5.6E-11
Fe59	5.0E-05	1.4E-11
Co58	7.7E-04	2.1E-10
Co60	8.8E-05	2.5E-11
Ni63	4.4E-05	1.2E-11
Zn65	8.5E-05	2.4E-11
Zr95	6.5E-05	1.8E-11
Ag110m	2.2E-04	6.1E-11
W187	4.7E-04	1.3E-10
Water Activation Products		
H3	9.6E-01	1.8E-03
C14	2.2E-04	6.2E-11

Radionuclide	Primary Activity (µCi/g)	Secondary Activity (µCi/g)
N16[7]	1.5E+1	4.1E-06
Ar41	1.4E-01	3.8E-08

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

Table A-4   Gaseous and liquid yearly effluent release values for a NuScale Power Plant (with 12 operating modules)

Radionuclide	Gaseous Effluent (Ci/yr)			Liquid Effluent (Ci/Yr)
	Plant Exhaust Stack Releases	Turbine Generator Building Releases	Total	
Noble Gases				
Kr83m	7.4E-03	3.4E-03	1.1E-02	0.0E+00
Kr85m	3.3E-02	1.4E-02	4.8E-02	0.0E+00
Kr85	2.1E+02	1.3E+00	2.2E+02	0.0E+00
Kr87	1.7E-02	7.9E-03	2.5E-02	0.0E+00
Kr88	5.0E-02	2.3E-02	7.3E-02	0.0E+00
Kr89	1.1E-03	5.3E-04	1.7E-03	0.0E+00
Xe131m	1.5E+00	5.2E-02	1.6E+00	0.0E+00
Xe133m	1.4E+00	5.0E-02	1.4E+00	0.0E+00
Xe133	2.8E+01	3.7E+00	3.2E+01	0.0E+00
Xe135m	3.7E-01	4.9E-03	3.8E-01	0.0E+00
Xe135	4.8E-01	1.3E-01	6.1E-01	0.0E+00
Xe137	3.6E-03	1.7E-03	5.3E-03	0.0E+00
Xe138	1.2E-02	5.8E-03	1.8E-02	0.0E+00
Halogens				
Br82	8.3E-07	1.8E-08	8.5E-07	2.4E-06
Br83	4.7E-06	1.0E-07	4.8E-06	9.6E-21
Br84	2.2E-06	4.2E-08	2.2E-06	0.0E+00
Br85	2.7E-07	2.3E-09	2.7E-07	0.0E+00
I129	2.1E-11	4.5E-13	2.1E-11	7.5E-10
I130	6.7E-06	1.5E-07	6.8E-06	1.7E-07
I131	6.3E-04	3.8E-06	6.4E-04	4.3E-03
I132	7.9E-05	1.7E-06	8.1E-05	2.0E-04
I133	3.1E-04	5.7E-06	3.2E-04	1.3E-04
I134	4.6E-05	9.3E-07	4.7E-05	0.0E+00
I135	1.6E-04	3.5E-06	1.7E-04	6.9E-09
Rubidium, Cesium				
Rb86m	9.7E-13	6.6E-13	1.6E-12	0.0E+00
Rb86	1.2E-08	2.3E-08	3.5E-08	7.9E-05
Rb88	9.8E-07	3.0E-06	4.0E-06	0.0E+00

Radionuclide	Gaseous Effluent (Ci/yr)			Liquid Effluent (Ci/Yr)
	Plant Exhaust Stack Releases	Turbine Generator Building Releases	Total	
Rb89	4.5E-08	1.3E-07	1.8E-07	0.0E+00
Cs132	2.2E-10	4.3E-10	6.5E-10	1.0E-06
Cs134	2.2E-06	3.9E-06	6.1E-06	1.7E-02
Cs135m	7.5E-10	2.7E-09	3.5E-09	0.0E+00
Cs136	4.4E-07	8.2E-07	1.3E-06	2.7E-03
Cs137	1.3E-06	2.4E-06	3.7E-06	1.0E-02
Cs138	3.6E-07	1.2E-06	1.6E-06	0.0E+00
Other Fission Products				
P32	2.6E-14	5.9E-14	8.5E-14	9.3E-11
Co57	2.0E-16	4.4E-16	6.4E-16	8.9E-13
Sr89	1.2E-09	2.6E-09	3.8E-09	5.2E-06
Sr90	2.7E-10	5.9E-10	8.6E-10	1.2E-06
Sr91	3.9E-10	1.3E-09	1.7E-09	2.5E-10
Sr92	2.0E-10	7.1E-10	9.1E-10	3.2E-21
Y90	9.7E-11	1.4E-10	2.4E-10	9.9E-07
Y91m	2.1E-10	6.7E-10	8.8E-10	1.6E-10
Y91	1.8E-10	3.8E-10	5.5E-10	7.4E-07
Y92	1.7E-10	6.0E-10	7.8E-10	5.5E-17
Y93	8.4E-11	2.9E-10	3.7E-10	8.8E-11
Zr97	1.3E-10	4.2E-10	5.6E-10	4.1E-09
Nb95	4.5E-07	1.1E-09	4.5E-07	3.0E-05
Mo99	3.0E-07	7.6E-07	1.1E-06	4.0E-04
Mo101	8.1E-09	2.2E-08	3.0E-08	0.0E+00
Tc99m	2.8E-07	7.0E-07	9.8E-07	3.9E-04
Tc99	1.0E-11	2.2E-11	3.2E-11	4.5E-08
Ru103	3.3E-10	7.3E-10	1.1E-09	1.4E-06
Ru105	6.8E-11	2.4E-10	3.0E-10	8.1E-16
Ru106	2.2E-10	4.7E-10	6.9E-10	9.7E-07
Rh103m	3.3E-10	6.7E-10	1.0E-09	1.4E-06
Rh105	1.8E-10	5.1E-10	7.0E-10	8.9E-08
Rh106	2.2E-10	4.8E-11	2.7E-10	9.7E-07
Ag110	5.7E-07	2.9E-10	5.7E-07	4.3E-09
Sb124	5.0E-13	1.1E-12	1.6E-12	3.0E-10

Radionuclide	Gaseous Effluent (Ci/yr)			Liquid Effluent (Ci/Yr)
	Plant Exhaust Stack Releases	Turbine Generator Building Releases	Total	
Sb125	4.4E-12	9.5E-12	1.4E-11	2.8E-09
Sb127	1.7E-11	4.1E-11	5.8E-11	4.5E-09
Sb129	1.4E-11	5.0E-11	6.4E-11	2.0E-17
Te125m	6.4E-10	1.4E-09	2.0E-09	2.7E-06
Te127m	2.1E-09	4.5E-09	6.6E-09	9.0E-06
Te127	5.9E-09	1.8E-08	2.4E-08	8.8E-06
Te129m	5.9E-09	1.3E-08	1.9E-08	2.4E-05
Te129	6.5E-09	1.7E-08	2.4E-08	1.5E-05
Te131m	1.5E-08	4.2E-08	5.7E-08	4.4E-06
Te131	6.4E-09	1.8E-08	2.4E-08	9.8E-07
Te132	1.2E-07	3.1E-07	4.3E-07	2.0E-04
Te133m	7.4E-09	2.4E-08	3.2E-08	0.0E+00
Te134	1.0E-08	3.4E-08	4.4E-08	0.0E+00
Ba137m	1.3E-06	7.5E-07	2.0E-06	9.7E-03
Ba139	1.9E-10	6.6E-10	8.5E-10	0.0E+00
Ba140	1.7E-09	3.8E-09	5.5E-09	5.8E-06
La140	7.5E-10	1.1E-09	1.8E-09	6.0E-06
La141	6.0E-11	2.1E-10	2.7E-10	5.0E-17
La142	2.9E-11	9.8E-11	1.3E-10	0.0E+00
Ce141	2.7E-10	5.8E-10	8.5E-10	1.1E-06
Ce143	1.5E-10	4.4E-10	5.9E-10	6.0E-08
Ce144	2.3E-10	4.9E-10	7.1E-10	1.0E-06
Pr143	2.4E-10	5.2E-10	7.5E-10	8.8E-07
Pr144	2.2E-10	3.9E-10	6.1E-10	9.9E-07
Np239	3.6E-09	9.2E-09	1.3E-08	3.9E-06
Corrosion Activation Products - CRUD				
Na24	2.9E-06	9.7E-06	1.3E-05	4.9E-05
Cr51	9.6E-05	5.5E-07	9.7E-05	8.3E-06
Mn54	5.2E-05	2.8E-07	5.2E-05	1.2E-05
Fe55	3.9E-05	2.1E-07	3.9E-05	9.5E-04
Fe59	9.5E-06	5.3E-08	9.5E-06	2.2E-04
Co58	1.5E-03	8.2E-07	1.5E-03	1.5E-03
Co60	1.7E-05	9.4E-08	1.7E-05	6.3E-05

Radionuclide	Gaseous Effluent (Ci/yr)			Liquid Effluent (Ci/Yr)
	Plant Exhaust Stack Releases	Turbine Generator Building Releases	Total	
Ni63	8.6E-06	4.7E-08	8.6E-06	2.1E-04
Zn65	1.6E-05	9.1E-08	1.7E-05	4.0E-04
Zr95	1.2E-05	6.9E-08	1.2E-05	2.9E-04
Ag110m	4.2E-05	2.3E-07	4.2E-05	3.1E-07
W187	2.4E-05	4.9E-07	2.5E-05	3.1E-05
Water Activation Products				
H3	7.0E+02	6.9E+00	7.1E+02	1.1E+03
C14	2.7E-01	2.4E-07	2.7E-01	3.7E-01
N16	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Ar41	1.4E+01	9.7E-01	1.5E+01	0.0E+00

Table A-5   Fuel failure mechanism distribution

Date	Fretting	Unknown	Debris	Fabrication	PCI	CRUD	Source	Ref.
2000-2006	73.5%	18.4%	3.7%	2.2%	0.4%	1.8%	GNF 2009, pg. 13	7.2.8
2007	73.7%	26.3%	0.0%	0.0%	0.0%	0.0%	GNF 2009, pg. 13	7.2.8
1987–1990	8.6%	51.8%	28.8%	10.8%	0.0%	0.0%	IAEA 2010, pg. 38	7.2.9
1991–1994	22.7%	49.0%	24.8%	3.6%	0.0%	0.0%	IAEA 2010, pg. 38	7.2.9
1995–1998	53.8%	26.9%	10.7%	7.0%	0.0%	1.6%	IAEA 2010, pg. 38	7.2.9
1999–2002	75.0%	14.6%	6.1%	2.9%	0.0%	1.3%	IAEA 2010, pg. 38	7.2.9
2003–2006	52.1%	33.2%	9.3%	4.8%	0.6%	0.0%	IAEA 2010, pg. 38	7.2.9
1990–2010	66.7%	9.2%	11.3%	3.0%	5.9%	3.6%	DOE 2012, pg. 19	7.2.10
2000–2007	79.9%	7.5%	5.5%	2.8%	2.2%	1.8%	NN 2010, pg. 42	7.2.11
2008–2010	77.1%	11.8%	4.2%	5.6%	1.4%	0.0%	NN 2010, pg. 42	7.2.11
Normalized Averages	76.4%	-	13.3%	4.8%	3.1%	2.4%	-	-

Table A-6   Fuel failure data for U.S. pressurized water reactors with zirconium-alloy cladding
(Reference 7.2.32)

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Enclosure 3:

Affidavit of Thomas A. Bergman, AF-1018-62369

NuScale Power, LLC

AFFIDAVIT of Thomas A. Bergman

I, Thomas A. Bergman, state as follows:

(1)   I am the Vice President 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 technical report reveals distinguishing aspects about the method by which NuScale develops its normal failed fuel fraction.

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 technical report entitled "Effluent Release (GALE Replacement) Methodology and Results." 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.

(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 November 13, 2018.

_____
Thomas A. Bergman