RAIO-0818-61468

August 21, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 356 (eRAI No. 9253) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 356 (eRAI No. 9253)," dated January 29, 2018

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 9253:

- 11.01-2

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

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

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9253

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9253

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9253
Date of RAI Issue: 01/29/2018

NRC Question No.: 11.01-2

Regulatory Requirements/Guidance:

10 CFR 52.47(a)(5) requires applicants to identify the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radiation exposures set forth in part 20 of this chapter. 10 CFR Part 20, Appendix I to 10 CFR Part 50, and 40 CFR Part 190 specify the annual dose limits to workers and members of the public, and the As Low As is Reasonably Achievable numerical objectives in the design of radwaste systems for controlling and limiting liquid and gaseous effluent releases.

The Design Specific Review Standard (DSRS) Acceptance Criteria section for NuScale, DSRS Section 11.1, “Coolant Source Terms,” states all normal operation, anticipated operation occurrence, and design basis source terms, etc., will be considered. In addition, DSRS Section 11.1, “Coolant Source Terms,” states when the applicant’s calculation technique or any source term parameters differ from that given in NUREG-0017 or ANSI/ANS 18.1-1999, these differences should be described with sufficient detail; and the basis of the alternate method and model parameters should be provided to allow the staff to evaluate this approach. DSRS Sections 11.2, “Liquid Waste Management System,” and 11.3, “Gaseous Waste Management System,” describe that the calculated annual total quantity of radioactive materials released from each reactor will not result in exceeding the annual exposure pathway doses from liquid and gaseous effluents in Appendix I to 10 CFR Part 50; annual dose limits in 10 CFR 20.1301; and annual liquid and gaseous effluent concentration limits in Table 2, Columns 1 and 2 of Appendix B to 10 CFR Part 20. Further, DSRS Section 12.2, “Radiation Sources,” states that applications should contain the methods, models, and assumptions used as the bases for all sources described.

NRC guidance in Branch Technical Position (BTP) 11-6, “Postulated Radioactive Releases due to Liquid- Containing Tank Failures,” identifies radionuclides (both parent and progeny) such as H-3, C-14, Sr-90, Tc-99, I-129, and Cs-137, etc., (and additionally for pressurized water power reactors, Br-84, Rb-88, Y- 91m, Te-129, Te-131, and Ce-143), that should be included, at a minimum, in an environmental transport analysis for assessment of an accidental liquid release due to their long half-lives and mobility. Conformance with BTP 11-6 is considered, in part, as an

acceptable method for demonstrating compliance with the applicable regulations in 10 CFR Part 20, Appendix I to 10 CFR Part 50, and 40 CFR Part 190.

Key Issue 1: Realistic and design basis source terms in the NuScale design exclude Tc-99, a long-lived and environmentally mobile radionuclide produced in the fuel, which can escape as a fission product into the reactor coolant system for release into the environment. Tc-99 should be included in source terms, environmental transport analyses, and dose assessments, or its exclusion justified. (Question related to NRC eRAI 8750 and NuScale RAIO-0817-55643.)

Design Certification Application Content:

Based on information obtained during the NRC's regulatory audit of the NuScale Design Control Document (DCD) Rev. 0, Chapter 11, "Radioactive Waste Management," and Chapter 12, "Radiation Protection," and Environmental Qualification aspects of the NuScale design, the staff observed that Tc-99 activity in the fuel had been calculated (Tc-99 activity appears to be about three orders of magnitude greater than the I-129 activity in the fuel), but this information was excluded from DCD Chapter 11, Table 11.1-1: Maximum Core Isotopic Inventory and in NuScale Technical Report (TR) Effluent Release (GALE Replacement) Methodology and Results, TR-1116-25065-NP Rev. 0, Appendix A Summary Tables. Furthermore, this information was also not carried forward and tracked throughout the source term tables in DCD Chapter 11 for consideration in the calculation analyses and assessment of effluent releases and public doses. In addition, DCD Chapter 12, Table 12.2-10: "Reactor Pool Cooling, Spent Fuel Pool Cooling, Pool Cleanup and Pool Surge Control System Component Source Terms - Radionuclide Content," excludes Tc-99 activity in the liquid release source term that would be used by the COL applicant in its postulated accidental liquid-containing tank failure analysis and dose assessment to a member of the public for completion of COL Item 11.2-3.

<u>Question 1</u>

Therefore, the staff requests that NuScale:
a. Include Tc-99 activities in the fuel, primary and secondary coolant for realistic and design basis source terms in DCD Chapter 11 Tables 11.1-1, 11.1-4, 11.1-5, 11.1-6, and 11.1-7, and TR-1116-25065-NP, or justify its exclusion;
b. Include Tc-99 activity in the postulated accidental liquid release source term for the pool surge control system in DCD Chapter 12 Table 12.2-10, or justify its exclusion; and
c. Based on the above, provide a DCD and TR-1116-25065-NP markup to include these changes.

Key Issue 2: The proposed alternate methodology used to calculate the tritium (H-3) production rate in the reactor coolant system (RCS) is non-conservative and underestimates H-3 concentrations in liquid and gaseous effluent releases during normal operations because it does not account for the buildup of H- 3 due to recycling of previously used RCS. In the NuScale design, H-3 is the largest calculated water activation production reaction due to more water in the RCS per megawatt generated, higher capacity factor, and higher starting lithium

concentration, which results in more H-3 production than a standard light water pressurized reactor. An increase in H-3 concentration results in a proportional increase to offsite public dose. (Question related to NRC eRAI 9270.)

Design Certification Application Content:

Based on information obtained during the NRC's regulatory audit of the NuScale Design Control Document (DCD) Rev. 0, Chapter 11, "Radioactive Waste Management" and Chapter 12, "Radiation Protection," and Environmental Qualification aspects of the NuScale design, the staff observed that the H- 3 concentration in the RCS, which moves readily throughout the plant system pathways for eventual release in liquid and gaseous effluents into the environment, assumes no recycling of the RCS; that is, all make up water supplied to the RCS during the operating cycle is assumed to contain zero radioactivity including H-3.

NuScale TR-1116-52065 Section 4.1.1, "Water Activation Products," describes cumulative water injection (i.e. RCS makeup water) and bleed out of the RCS over the two year operating cycle, Figure 4-2, "Tritium reactor coolant system balance" shows a peak H-3 activity of 74 Ci, and Figure 4-3, "Tritium concentration and time weighted average" shows a H-3 concentration of 0.97 uCi/ml. NuScale TR-1116-52065 Figure 4-3, "Tritium concentration and time weighted average" shows the "Tritium Concentration in RCS - No Recycle (uCi/g)" versus "Time (years)" for the operating cycle.

DCD Section 9.3.4.2.1, "General Description," states that recycled, degassed reactor coolant from the Liquid Radioactive Waste System (LRWS) can also be added back to the Chemical and Volume Control System (CVCS) by a supply line upstream of the makeup pumps. DCD Chapter 9 Figure 9.3.4-1: Chemical and Volume Control System Diagram shows a return from the LRWS. DCD Section 11.2 describes that liquids may be recycled for use in the RCS. DCD Chapter 11, Tables 11.1-4: Primary Coolant Design Basis Source Term and 11.1-6: Primary Coolant Realistic Source Term provide H-3 concentrations of 9.7000E-01 uCi/g in the RCS. DCD Chapter 11 Tables 11.2-5: Estimated Annual Releases to Liquid Radioactive Waste System Discharge Header and 11.3-5: Gaseous Estimated Discharge for Normal Effluents show total plant release rates of 1.55E+03 Ci/y for liquid effluent releases and 7.37E+02 Ci/y for gaseous effluent releases, respectively.

Question 2

Therefore, the staff requests that NuScale:

a. Provide the method, model, and assumptions used to calculate the H-3 production rate in the RCS to account for H-3 buildup due to recycling of previously used RCS, or justify the current H-3 activity, concentration, and liquid and gaseous effluent release rates are conservative and bounding.

b. Based on the above, provide a DCD and TR-1116-25065-NP markup to include these changes.

NuScale Response:

NuScale Response (1):

Consistent with NRC Branch Technical Position (BTP) 11-6, the NuScale source terms in FSAR Chapters 11 and 12, and in TR-1116-52065, have been revised to include the radioisotope Tc-99. The revised FSAR Table 11.1-1, which now includes Tc-99, is attached to this response. The revised FSAR Tables 11.1-4, 11.1-5, 11.1-6 and 11.1-7 are included with the response to RAI 9270 (Q12.02-20). The revised technical report (TR-1116-52065, Rev. 1) is included with the response to RAI 9161 (Q11.01-1).

Because the durable and passive metal lined pool surge control system (PSCS) storage tank dike is an approved mitigative design feature, according to BTP 11-6 NuScale does not perform an evaluation of a postulated accidental liquid release from the PSCS storage tank. NuScale deleted the associated COL Item (COL Item 11.2-3) and revised FSAR Section 11.2.3.2 and Table 1.8-2. NuScale has included Tc-99 in the source terms for tanks containing radioactive liquids, including the PSCS storage tank, which is included in the response to RAI 12.02-20 (9270).

The following FSAR Tables have been updated to include Tc-99, as follows:

- FSAR Table 11.1-1 Maximum Core Isotopic Inventory - attached to this RAI response.
- FSAR Tables 11.1-4, 11.1-5, 11.1-6 & 11.1-7 for the primary and secondary coolant activities are shown in the NuScale response to RAI 9270 (12.02-20).
- FSAR Table 11.2-5 Estimated Annual Releases to LRWS Discharge Header - RAI 9264 (12.02-4)
- FSAR Table 11.2-8 Liquid Release Concentration Compared to Part 20 limits - RAI 9264 (12.02-4)
- FSAR Table 11.3-5 Gaseous Estimated Discharge for Normal Effluents - RAI 9264 (12.02-4)
- FSAR Table 12.2-7 CVCS Source Terms - RAI 9257 (12.02-14)
- FSAR Table 12.2-10 RPC, SFPC, PCU & PSCS Component Source Terms is shown in the NuScale response to RAI 9270 (12.02-20)
- FSAR Table 12.2-13a & 13b LRWS Component Source Terms - RAI 9256 (12.02-10)
- FSAR Table 12.2-16 GRWS Component Source Terms - RAI 9161
- FSAR Table 12.2-19 SRWS Component Source Terms - RAI 9264 (12.02-4)
- The revised FSAR Table 12.2-33 RXB Airborne Concentrations is shown in the NuScale response to RAI 12.02-20 - RAI 9270 (12.02-20)
- Tech Report TR-1116-52065, Rev. 1 (RAI 9161).

NuScale Response (2):

NuScale has evaluated the tritium concentration in various process streams based on three recycling modes: 1) no recycling of the primary coolant; 2) recycling of primary coolant to the reactor pool; and 3) recycling of primary coolant back to the chemical and volume control system (CVCS) makeup. The first mode (no recycling) maximizes the tritium concentration in the normal liquid discharge effluent stream; therefore, this tritium concentration is used for liquid effluent in FSAR Section 11.2 and TR-1116-52065. The second mode (recycling to the reactor pool) maximizes the tritium concentration in the reactor pool, in the airborne concentration in the airspace above the reactor pool, and in the normal gaseous effluent, and thus is used for the reactor pool water source term in FSAR Section 12.2 and normal gaseous effluent in FSAR Section 11.3 and TR-1116-52065. The third mode (recycling back to CVCS makeup) maximizes the tritium concentration in the primary coolant and is used in determining the airborne concentration due to primary coolant leakage, as shown in FSAR Table 12.2-33.

Impact on DCA:

FSAR Section 11.2.3.2, Table 1.8-2, and Table 11.1-1 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 01-61, RAI 02.04.13-1, RAI 03.04.01-4, RAI 03.04.02-1, RAI 03.04.02-2, RAI 03.04.02-3, RAI 03.05.01.04-1, RAI 03.05.02-2, RAI 03.06.02-15, RAI 03.06.03-11, RAI 03.07.01-2, RAI 03.07.01-3, RAI 03.07.02-8, RAI 03.07.02-12, RAI 03.08.04-23S1, RAI 03.08.04-23S2, RAI 03.08.05-14S1, RAI 03.09.02-15, RAI 03.09.02-48, RAI 03.09.02-67, RAI 03.09.02-69, RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-6, RAI 03.09.06-16, RAI 03.09.06-16S1, RAI 03.09.06-27, RAI 03.11-8, RAI 03.11-14, RAI 03.11-14S1, RAI 03.11-18, RAI 03.13-3, RAI 04.02-1S2, RAI 05.02.03-19, RAI 05.02.05-8, RAI 05.04.02.01-13, RAI 05.04.02.01-14, RAI 06.02.06-22, RAI 06.02.06-23, RAI 06.04-1, RAI 09.01.02-4, RAI 09.01.05-3, RAI 09.01.05-6, RAI 09.03.02-3, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8, RAI 10.02-1, RAI 10.02-2, RAI 10.02-3, RAI 10.02.03-1, RAI 10.02.03-2, RAI 10.03.06-1, RAI 10.03.06-5, RAI 10.04.06-1, RAI 10.04.06-2, RAI 10.04.06-3, RAI 10.04.10-2, RAI 11.01-2, RAI 12.03-55S1, RAI 13.01.01-1, RAI 13.01.01-1S1, RAI 13.02.02-1, RAI 13.03-4, RAI 13.05.02.01-2, RAI 13.05.02.01-2S1, RAI 13.05.02.01-3, RAI 13.05.02.01-3S1, RAI 13.05.02.01-4, RAI 13.05.02.01-4S1, RAI 14.02-7, RAI 19-31, RAI 19-31S1, RAI 19-38, RAI 20.01-13

Table 1.8-2: Combined License Information Items

Item No.	Description of COL Information Item	Section
COL Item 1.1-1:	A COL applicant that references the NuScale Power Plant design certification will identify the site-specific plant location.	1.1
COL Item 1.1-2:	A COL applicant that references the NuScale Power Plant design certification will provide the schedules for completion of construction and commercial operation of each power module.	1.1
COL Item 1.4-1:	A COL applicant that references the NuScale Power Plant design certification will identify the prime agents or contractors for the construction and operation of the nuclear power plant.	1.4
COL Item 1.7-1:	A COL applicant that references the NuScale Power Plant design certification will provide site-specific diagrams and legends, as applicable.	1.7
COL Item 1.7-2:	A COL applicant that references the NuScale Power Plant design certification will list additional site-specific piping and instrumentation diagrams and legends as applicable.	1.7
COL Item 1.8-1:	A COL applicant that references the NuScale Power Plant design certification will provide a list of departures from the certified design.	1.8
COL Item 1.9-1:	A COL applicant that references the NuScale Power Plant design certification will review and address the conformance with regulatory criteria in effect six months before the docket date of the COL application for the site-specific portions and operational aspects of the facility design.	1.9
COL Item 1.10-1:	A COL applicant that references the NuScale Power Plant design certification will evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk significant structures, systems, and components of existing operating unit(s) and newly constructed operating unit(s) at the co-located site per 10 CFR 52.79(a)(31). The evaluation will include identification of management and administrative controls necessary to eliminate or mitigate the consequences of potential hazards and demonstration that the limiting conditions for operation of an operating unit would not be exceeded. This COL item is not applicable for construction activities (build-out of the facility) at an individual NuScale Power Plant with operating NuScale Power Modules.	1.10
COL Item 2.0-1:	A COL applicant that references the NuScale Power Plant design certification will demonstrate that site-specific characteristics are bounded by the design parameters specified in Table 2.0-1. If site-specific values are not bounded by the values in Table 2.0-1, the COL applicant will demonstrate the acceptability of the site-specific values in the appropriate sections of its combined license application.	2.0
COL Item 2.1-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site geographic and demographic characteristics.	2.1
COL Item 2.2-1:	A COL applicant that references the NuScale Power Plant design certification will describe nearby industrial, transportation, and military facilities. The COL applicant will demonstrate that the design is acceptable for each potential accident, or provide site-specific design alternatives.	2.2
COL Item 2.3-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific meteorological characteristics for Section 2.3.1 through Section 2.3.5, as applicable.	2.3
COL Item 2.4-1:	A COL applicant that references the NuScale Power Plant design certification will investigate and describe the site-specific hydrologic characteristics for Section 2.4.1 through Section 2.4.14, ~~as applicable~~except Section 2.4.8 and Section 2.4.10.	2.4
COL Item 2.5-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific geology, seismology, and geotechnical characteristics for Section 2.5.1 through Section 2.5.5, below.	2.5

Table 1.8-2: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 11.2-1:	A COL applicant that references the NuScale Power Plant design certification will ensure mobile equipment used and connected to plant systems is in accordance with ANSI/ANS-40.37, Regulatory Guide (RG) 1.143, 10 CFR 20.1406, NRC IE Bulletin 80-10 and 10 CFR 50.34a.	11.2
COL Item 11.2-2:	A COL applicant that references the NuScale Power Plant design certification will calculate doses to members of the public using the site-specific parameters, compare those liquid effluent doses to the numerical design objectives of 10 CFR 50, Appendix I, and comply with the requirements of 10 CFR 20.1302 and 40 CFR 190.	11.2
COL Item 11.2-3:	~~A COL applicant that references the NuScale Power Plant design certification will perform a site-specific evaluation of the consequences of an accidental release of radioactive liquid from the pool surge control system storage tank in accordance with NRC Branch Technical Position 11-6.~~Not used.	11.2
COL Item 11.2-4:	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific evaluation using the site-specific dilution flow.	11.2
COL Item 11.2-5:	A COL applicant that references the NuScale Power Plant design certification will perform a cost-benefit analysis as required by 10 CFR 50.34a and 10 CFR 50, Appendix I, to demonstrate conformance with regulatory requirements. This cost-benefit analysis is to be performed using the guidance of Regulatory Guide 1.110.	11.2
COL Item 11.3-1:	A COL applicant that references the NuScale Power Plant design certification will perform a site-specific cost-benefit analysis.	11.3
COL Item 11.3-2:	A COL applicant that references the NuScale Power Plant design certification will calculate doses to members of the public using the site-specific parameters, compare those gaseous effluent doses to the numerical design objectives of 10 CFR 50, Appendix I, and comply with the requirements of 10 CFR 20.1302 and 40 CFR 190.	11.3
COL Item 11.3-3:	A COL applicant that references the NuScale Power Plant design certification will perform an analysis in accordance with Branch Technical Position 11-5 using the site-specific parameters.	11.3
COL Item 11.4-1:	A COL applicant that references the NuScale Power Plant design certification will describe mobile equipment used and connected to plant systems in accordance with ANSI/ANS 40.37, Regulatory Guide 1.143, 10 CFR 20.1406, NRC IE Bulletin 80-10, and 10 CFR 50.34a.	11.4
COL Item 11.4-2:	A COL applicant that references the NuScale Power Plant design certification will develop a site-specific process control program following the guidance of Nuclear Energy Institute (NEI) 07-10A (Reference 11.4-3).	11.4
COL Item 11.5-1:	A COL applicant that references the NuScale Power Plant design certification will describe site-specific process and effluent monitoring and sampling system components and address the guidance provided in ANSI N13.1-2011, ANSI N42.18-2004 and Regulatory Guides 1.21, 1.33 and 4.15.	11.5
COL Item 11.5-2:	A COL applicant that references the NuScale Power design certification will develop an offsite dose calculation manual (ODCM) that contains a description of the methodology and parameters used for calculation of offsite doses for gaseous and liquid effluents, using the guidance of Nuclear Energy Institute (NEI) 07-09A (Reference 11.5-8).	11.5
COL Item 11.5-3:	A COL applicant that references the NuScale Power design certification will develop a radiological environmental monitoring program (REMP), consistent with the guidance in NUREG-1301 and NUREG-0133, that considers local land use census data for the identification of potential radiation pathways radioactive materials present in liquid and gaseous effluents, and direct external radiation from systems, structures, and components.	11.5
COL Item 12.1-1:	A COL applicant that references the NuScale Power Plant design certification will describe the operational program to maintain exposures to ionizing radiation as far below the dose limits as practical, as low as reasonably achievable (ALARA).	12.1
COL Item 12.2-1:	A COL applicant that references the NuScale Power Plant design certification will describe additional site-specific contained radiation sources that exceed 100 millicuries (including sources for instrumentation and radiography) not identified in Section 12.2.1.	12.2
COL Item 12.3-1:	A COL applicant that references the NuScale Power Plant design certification will develop the administrative controls regarding access to high radiation areas per the guidance of Regulatory Guide 8.38.	12.3

RAI 11.01-2

Table 11.1-1: Maximum Core Isotopic Inventory

Nuclide	Core Inventory (Ci)	Nuclide	Core Inventory (Ci)
Noble Gases		Other Fission Products	
Kr83m	7.289E+05	Y92	8.103E+06
Kr85m	1.698E+06	Y93	8.584E+06
Kr85	1.339E+05	Zr95	8.547E+06
Kr87	3.478E+06	Zr97	8.288E+06
Kr88	4.662E+06	Nb95	8.510E+06
Kr89	5.994E+06	Mo99	8.658E+06
Xe131m	5.994E+04	Mo101	7.881E+06
Xe133m	2.871E+05	Tc99m	7.622E+06
Xe133	9.509E+06	Ru103	8.843E+06
Xe135m	2.098E+06	Ru105	7.215E+06
Xe135	4.958E+06	Ru106	5.698E+06
Xe137	8.547E+06	Rh103m	8.769E+06
Xe138	8.621E+06	Rh105	6.771E+06
Halogens		Rh106	6.142E+06
Br82	2.568E+04	Ag110m	6.475E+04
Br83	7.289E+05	Sb124	1.302E+04
Br84	1.310E+06	Sb125	1.143E+05
Br85	1.691E+06	Sb127	5.328E+05
I129	5.365E-01	Sb129	1.532E+06
I130	2.683E+05	Te125m	2.697E+04
I131	4.662E+06	Te127m	8.658E+04
I132	6.660E+06	Te127	5.254E+05
I133	9.546E+06	Te129m	2.498E+05
I134	1.092E+07	Te129	1.462E+06
I135	8.991E+06	Te131m	9.916E+05
Rubidium, Cesium		Te131	3.922E+06
Rb86m	2.013E+03	Te132	6.438E+06
Rb86	1.595E+04	Te133m	4.625E+06
Rb88	4.699E+06	Te134	9.546E+06
Rb89	6.253E+06	Ba137m	1.558E+06
Cs132	3.230E+02	Ba139	8.806E+05
Cs134	2.671E+06	Ba140	8.510E+06
Cs135m	3.193E+04	La140	8.547E+06
Cs136	5.883E+05	La141	8.066E+06
Cs137	1.635E+06	La142	7.955E+06
Cs138	9.213E+06	Ce141	7.955E+06
Other Fission Products		Ce143	8.103E+06
P32	7.622E+02	Ce144	6.586E+06
Co57	5.254E+00	Pr143	7.844E+06
Ni63	4.144E+01	Pr144	6.623E+06
Sr89	5.957E+06	Np239	1.288E+08
Sr90	1.129E+06	Tc99	2.105E+02
Sr91	7.770E+06		
Sr92	7.992E+06		
Y90	1.158E+06		
Y91m	4.588E+06		
Y91	7.363E+06		

The activities indicated as secondary coolant in Table 11.2-3 are the radionuclide activities listed in Table 11.2-8.

The detergent waste collection tank uses the source term from NUREG-0017, Table 2-27, multiplied by 12 reactors. No credit is taken for the detergent waste drain filter.

The effluents from the three LRWS waste processing pathways are discharged through a single discharge header to the utility water system (UWS) discharge basin, where it is diluted, monitored and released. The assumed dilution factor is described in Section 11.2.3.3 and is listed in Table 11.2-4. An additional adjustment is added to the total non-tritium liquid release to account for unidentified AOOs. This adjustment is a reactor thermal power scaled value from NUREG-0017, Section 2.2.23. This value is also provided in Table 11.2-4. The total resultant liquid release concentrations are provided in Table 11.2-11, and demonstrate compliance with 10 CFR 20 Appendix B, Table 2.

The maximum individual doses are calculated using the LADTAP II Code, using the input parameters listed in Table 11.2-7. The resultant doses are presented in Table 11.2-8 and demonstrate compliance with the limits of 10 CFR 50 Appendix I.

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

11.2.3.2 Compliance with Branch Technical Position 11-6

RAI 02.04.13-1

The only outdoor tank expected to contain radioactive liquids is the PSCS storage tank, described in FSAR Section 9.1.3.2.4. The PSCS storage tank catch basin has sufficient volume to store the contents of the PSCS storage tank plus the contents of related piping. The radionuclide inventory of the PSCS storage tank is provided in Table 12.2-10. The PSCS storage tank is the limiting, source term liquid tank for use in a postulated radioactive liquid-containing tank failure analysis.

RAI 11.01-2~~RAI 02.04.13-1~~

~~An analysis of an accidental release of radioactive liquid effluents in groundwater and surface water is site-specific and addressed by COL Item 2.4-1 and COL Item 11.2-3.~~No accidental release to the environment of radioactive liquid from a PSCS storage tank failure is assumed because the PSCS storage tank is designed with a passive and durable mitigative design feature (a metal lined concrete dike).

RAI 11.01-2~~RAI 02.04.13-1~~

COL Item 11.2-3: ~~A COL applicant that references the NuScale Power Plant design certification will perform a site-specific evaluation of the consequences of an accidental release of radioactive liquid from the pool surge control system storage tank in accordance with NRC Branch Technical Position 11-6.~~ Not used.