RAIO-0719-66421

July 31, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 205 (eRAI No. 9044) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 205 (eRAI No. 9044)," dated September 01, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 205 (eRAI No.9044)," dated October 31, 2017

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

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Questions from NRC eRAI No. 9044:

- 09.03.02-2
- 09.03.02-3

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
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9044

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9044

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

eRAI No.: 9044
Date of RAI Issue: 09/01/2017

NRC Question No.: 09.03.02-2

Regulatory Requirements and Guidance:

10 CFR Part 50, Appendix A, General Design Criterion 64, requires that means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

10 CFR 50.34(f)(2)(viii) requires that applicants provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. In addition, NUREG-0737 recommends prompt sampling under accident conditions.

DSRS Section 9.3.2 states that the primary review organization and the organization responsible for the review of radiation protection verify that provisions are made for purging sampling lines and for reducing plateout in sample lines (e.g., with heat tracing).

Key Issue: There is not enough information how heat tracing will be implemented in the design.

DCD Section 9.3.2.2.3, under “Containment Gas Post-Accident Monitoring Sampling,” the applicant indicates that the PSS piping is heat traced to prevent the build-up of condensate within the containment gas monitoring lines and analyzer to ensure monitoring capability under accident conditions.

Requested Additional Information:

It is unclear how heat tracing will be implemented in the design, e.g., what power supply will be used to heat trace the sample lines. If the heat tracing is not supplied from a reliable power supply and the power supply is non-functional following a design basis accident, it is unclear how it can be ensured that reliable samples will be able to be taken. Please provide additional information regarding which power supply will be used to perform heat tracing and how it can be assured that heat tracing can be performed following a design basis accident.

NuScale Response:

NuScale is supplementing its original response to RAI 9044 (Question 09.03.02-2) provided in letter RAIO-1017-56948, dated October 31, 2017. This supplemental information involves the source of power for the process sampling system (PSS) piping heat tracing. The remaining information provided in the original response remains valid.

The original RAI 9044 Question 09.03.02-2 response indicated that the PSS piping heat tracing power is provided by the normal DC power system. Upon further engineering review, the containment sampling system skid, which also includes sample line heat trace, will be powered from the low voltage AC electrical distribution system (ELVS). The containment sampling system is connected to the ELVS motor control centers which are backed up by the backup diesel generator (part of the backup power supply system) in loss of normal AC power source event.

The original RAI 9044 Question 09.03.02-2 response stated, “As described in FSAR Section 9.3.2, it is expected that plant conditions would be amenable to perform containment gas sampling at approximately 24 hours after an event initiation, assuming power is available.” As part of the RAI 9044 Question 09.03.02-3 response, Section 9.3.2 has been revised to delete mention of post-accident containment gas sampling.

FSAR Section 9.3.2.2.3 has been revised to reflect that the PSS containment sampling system, including sample line heat tracing, is powered from ELVS.

Impact on DCA:

FSAR Section 9.3.2.2.3 has been revised as described in the response above and as shown in the markup provided with the response to question 09.03.02-3.

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

eRAI No.: 9044
Date of RAI Issue: 09/01/2017

NRC Question No.: 09.03.02-3

Regulatory Requirements:

10 CFR Part 50, Appendix A, General Design Criterion 64, requires that "means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents."

10 CFR 50.34(f)(2)(vii) requires the performance of radiation shielding design reviews to ensure the design permits adequate access to important areas and provides for protection of safety equipment from radiation, following an accident. DSRS Section 12.3-12.4, references this requirement and the associated NUREG-0737, Section II.B.2, which provides additional guidance on meeting this requirement.

10 CFR 50.34(f)(2)(viii) requires that applicants provide a "capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities." In addition, NUREG-0737 recommends prompt sampling under accident conditions.

Key Issues: The application does not have sufficient detail and clarity to determine if and how gaseous samples will be obtained post-accident and that applicable requirements will be met.

DCD Section 9.3.2 indicates that post-accident sampling of containment gas is possible in the NuScale design and "would be used for post-accident sampling only if the information sought is

essential and cannot be determined or estimated by other means". However, the DCD is unclear and inconsistent regarding how post accident gaseous samples of the containment atmosphere will be obtained.

In DCD Section 9.3.2.2.3, under "Off-Normal Operations", it states, "The CES (containment evacuation system) is a low pressure system not designed for full containment design pressure and has not been provided with override capability. Accident simulations project that in approximately 24 hours following a containment isolation initiation, RCS temperatures will fall below 200 degrees Fahrenheit, permitting the opening of the containment evacuation system CIVs to support sampling at that time, if necessary."

While in Section 9.3.2.2.3, under "Containment Gas Post-Accident Monitoring and Sampling" it states, "Plant conditions amenable to plant sampling exist within 2 hours of the most limiting design basis event, and will require override of the CNV (containment vessel) containment isolation valves for the CES and CFDS (containment flooding and drain system)." Furthermore, it also states, "the CNV isolation valves for CES and CFDS are opened to establish the monitoring and sampling flow paths. A manual logic override is required to open the CNV isolation valves if RCS temperature is greater than 200 degrees F and containment parameters are greater than the containment isolation setpoints."

Requested Additional information:

Based on the above information and apparent inconsistencies, please address the following.

1. It is unclear to staff at what time after an accident and under what conditions, containment gaseous samples are capable of being taken. It is also unclear if the isolation valves for the CES are provided with override capability or not. Please provide this information and update the DCD as appropriate to correct any inconsistencies.

2. It is unclear which valves are required to be opened to take gaseous samples (only the CES or both the CES and CFDS?). Please clarify which valves need to be opened. If both CES and CFDS valves need to be opened to take gaseous samples, please clarify why the isolation valves for the CFDS (which goes to a part of the containment vessel that is expected to be submerged following an accident), needs to be opened to obtain a gas sample. Update the DCD as appropriate.

3. Likewise, it is unclear if the systems are appropriately designed to handle the temperatures and pressures that will be present. DCD Section 9.3.6, "Containment Evacuation System and Containment Flooding and Drain System," does not specify the design limitations of the system. It is not clear if any relief valves are provided and at what pressure such relief valves would actuate (a significant release into the Reactor Building could occur, even if the piping were still intact, if a relief valve lifted, or a seal was damaged by heat). Please clarify the design limitations of the CES and CFDS systems and if the CES and CFDS systems downstream of the containment isolation valves are capable of withstanding the temperatures and pressures present 2 hours after an accident or if approximately 24 hours and less than 200 degrees Fahrenheit is required to open these valves. Update the DCD as appropriate.

4. It is unclear if appropriate equipment and power will be available to manually override and open valves to take samples during accident condition. Please describe the process and equipment that will be needed to re-open these valves and if this equipment is ensured to be operational following a design basis accident. Update the DCD as appropriate. Is this equipment operable from the Main Control Room, or is operator action in the field required? Is AC electrical power required to open these valves? How is it ensured that the required equipment can be appropriately operated following a design basis accident?

NuScale Response:

NuScale is supplementing its original response to RAI 9044 (Question 09.03.02-3) provided in letter RAIO-1017-56948, dated October 31, 2017. This supplement supersedes the original response in its entirety and is provided as a result of discussions with the NRC during a phone call on May 8, 2019, in which the Accident Source Term White Paper and the Post-Accident Sampling (PAS) exemption from 10 CFR 50.34(f)(2)(viii) was discussed. The Post-Accident Sampling exemption was transmitted by letter LO-0119-64386, dated January 31, 2019.

This supplemental response includes FSAR changes made in conformance with the PAS exemption and it clarifies the actions required for post-accident hydrogen and oxygen monitoring. The changes include the deletion of COL Item 9.3-2 from FSAR Table 1.8-2 and Section 9.3.2.2.3. Conforming changes have also been made to the 50.34(f)(2)(vii) and 50.34(f)(2)(viii) rows of FSAR Table 1.9-5 and to Section 12.4.1.8. COL Item 9.3-2 is discussed in the NRC Staff's Safety Evaluation with Open Items for the NuScale Chapter 9, "Auxiliary Systems," dated February 28, 2019 (ADAMS No. ML18201A382 and ML18201A383). Specifically, COL

Item 9.3-2 is mentioned in the discussion for Open Items RAI 9044, Questions 09.03.02-5 and 09.03.02-6, Open Item RAI 9044, Question 09.03.02-8, and in Table 9.3.2-1 of the NRC Staff's Safety Evaluation.

The NRC Staff's Safety Evaluation indicates that RAI 9044, Questions 09.03.02-2, -3, -4, -5, -6, and -8 are being held open pending the evaluation of the PAS exemption and revised accident source term methodology. This supplemental response to RAI 9044 Question 09.03.02-3 may be considered by the Staff in the review of Safety Evaluation Open Items RAI 9044, Questions 09.03.02-2, -3, -4, -5, -6, and -8.

1. The NuScale plant does not require post-accident grab sample of containment gas. However, hydrogen and oxygen monitoring capability of post-accident containment gas is provided. To perform hydrogen and oxygen monitoring, opening of the containment evacuation system (CES) and containment flood and drain system (CFDS) containment isolation valves (CIVs) is required to send containment gas to the hydrogen and oxygen monitor located outside of containment and return the gas back to the containment.

The containment isolation signal (CIS) actuates on high narrow range containment pressure (narrow range containment pressure > 9.5 psia) or low pressurizer level (level < 20%). The CIS automatically is removed when the CIS input parameters (high narrow range containment pressure and low pressurizer level) are clear or the reactor coolant system (RCS) temperature is less than 200 degrees F. When the RCS temperature is below 350 degrees F, the narrow range containment pressure input is no longer used for CIS (i.e., automatic operating bypass of CIS). If the pressurizer level remains below 20% due to a decrease in reactor coolant inventory, the RCS temperature must be less than 200 degrees F for automatic operating bypass of CIS.

With CIS actuated, the CES CIVs cannot be opened using normal controls available in the main control room (MCR). The CIS manual override switches are provided for CFDS CIVs in the MCR, but not provided for the CES CIVs. To override the CIS for CES CIVs, operator actions are required to open the CES CIVs from the CIV hydraulic control skids located outside the MCR.

Additionally, containment gas hydrogen and oxygen monitoring will be performed when plant conditions do not exceed design limitations of the CES and CFDS piping. The design pressure and temperature of the CES piping downstream of the CIVs are 250 psig and 650 degrees F. The design pressure and temperature of the CFDS piping downstream of the CIVs are 150 psig and 300 degrees F.

While the plant responses in accident conditions show the containment pressure is reduced to 150 psia in approximately two (2) hours after event initiation, the CIS might not be automatically bypassed by this time to enable opening of the CES and CFDS CIVs using normal controls available inside the MCR. In severe accidents with core damage, the CIS signal may not clear by this time and post-accident hydrogen/oxygen monitoring of containment gas will require operator action outside the MCR to override the CIS and open the CES CIVs from the CIV hydraulic control skids. Plant responses in design basis events shows the RCS cooling down to less than 200 degrees F within 24 hours after event initiation in most cases. For the most limiting design basis event, it can take more than 24 hours after an event initiation for the RCS temperature to be less than 200 degrees F. It is expected that the containment gas post-accident hydrogen and oxygen monitoring can be performed 24 hours after event initiation. FSAR Section 9.3.2.2.3 has been revised to clarify post-accident containment gas hydrogen and oxygen monitoring capability.

2. Post-accident hydrogen and oxygen monitoring of containment gas requires aligning PSS, CES, and CFDS to create a closed sample loop where the containment gas sample can be routed from the CES to PSS containment sampling system, and can be returned to the CNV via the CFDS process line. While the CFDS piping in the CNV is expected to be partially submerged following an accident, the CFDS provides the optimal return path for the gas discharged from the containment sampling system sample pump to the CNV. Therefore, opening of CFDS CIVs is required for returning the gas back to the containment. Returning the containment gas sample back to the CNV limits potential radioactive release outside of the containment. FSAR Figure 9.3.6-2 has been revised to show the PSS sample return line connection to the CFDS before the respective module isolation valve.

3. The design pressure and temperature of the CES piping downstream of the CIVs are 250 psig and 650 degrees F. The design pressure and temperature of the CFDS piping downstream of the CIVs are 150 psig and 300 degrees F. The post-accident containment gas sample loop will not be put into operation if the containment pressure exceeds CES and CFDS design limits.

While plant responses in accident conditions show containment pressure is reduced to 150 psia in approximately two (2) hours following an event initiation, the CIS will not be cleared at 2 hours if the PZR level is below 20%. It is expected that post-accident containment gas hydrogen and oxygen monitoring can be performed 24 hours after event initiation even if the CIS is still active. However, overriding the CIS will require operator actions outside the MCR to open the CIVs from the CIV hydraulic control skids.

FSAR Section 9.3.2.2.3 has been revised to clarify the plant conditions that are amenable to perform post-accident containment gas hydrogen and oxygen monitoring. The revision also includes discussion of the design limitations of the CES and CFDS and how the CES and CFDS components downstream of the CIVs are capable of withstanding the temperatures and pressures expected during post-accident hydrogen and oxygen monitoring.

4. The CVCS, CES, and CFDS CIVs are the primary system containment isolation valves (PSCIVs) as discussed in FSAR Section 6.2.4. The PSCIV design features ensure that the PSCIVs can be re-opened following a design basis event to support post-accident hydrogen and oxygen monitoring of containment gas. Two different hydraulic control skids are located on different levels of the reactor building are provided to satisfy the single-failure separation requirement. The low voltage AC electrical distribution system (ELVS) supplies power to the hydraulic pump drivers on the hydraulic skids. The ELVS loads are powered by the backup power supply system (BPSS) in a loss of normal AC power source event. The hydraulic control skids are also designed with a set of accumulators to support a limited number of reopenings of the CIVs after a design basis event without reliance on AC power.

FSAR Sections 9.3.2.2.3 and 9.3.2.5 and FSAR Figure 9.3.6-2 have been revised to clarify the post-accident containment gas hydrogen and oxygen monitoring process and equipment required to support monitoring activity. The changes to FSAR Section 9.3.2.5 and FSAR Figure 9.3.6-2 referred to in this RAI were included with the response to RAI 9044 (Question 09.03.02-8) in letter RAIO-1017-56948, dated October 31, 2017.

Impact on DCA:

FSAR Sections 9.3.2.2.3, 9.3.4.1, 9.3.4.3, 9.3.6.2.3, 11.5.2, and 12.4.1.8, FSAR Tables 1.8-2, 1.9-3, 1.9-5, 1.9-8, 3.2-1, 9.3.2-1, 9.3.2-2, and 14.2-53 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 03.05.01.03-1, RAI 03.05.01.04-1, RAI 03.05.02-2, RAI 03.05.03-4, RAI 03.06.02-6, RAI 03.06.02-15, RAI 03.06.03-11, RAI 03.07.01-2, RAI 03.07.01-3, RAI 03.07.02-4S3, RAI 03.07.02-6S1, RAI 03.07.02-6S2, RAI 03.07.02-8, RAI 03.07.02-12, RAI 03.07.02-15S5, RAI 03.07.02-16S1, RAI 03.07.02-23S1, RAI 03.07.02-26, RAI 03.08.04-1S1, RAI 03.08.04-3S2, RAI 03.08.04-23S1, RAI 03.08.04-23S2, RAI 03.08.04-23S3, 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 03.11-19S2, 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 05.04.02.01-19, RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-19, RAI 06.02.06-22, RAI 06.02.06-23, RAI 06.04-1, RAI 09.01.01-20, RAI 09.01.01-20S1, RAI 09.01.02-4, RAI 09.01.05-3, RAI 09.01.05-6, RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, 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 11.01-2S2, RAI 12.03-55S1, RAI 12.03-63, 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 16-65, 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, except Section 2.4.8 and Section 2.4.10.	2.4

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

Item No.	Description of COL Information Item	Section
COL Item 9.1-9:	A COL applicant that references the NuScale Power Plant design certification will provide a neutron absorber material qualification report which demonstrates that the neutron absorber material can meet the neutron attenuation and environmental compatibility design functions described in Technical Report TR-0816-49833. The COL applicant will establish procedures to evaluate the neutron attenuation uncertainty associated with the material lot variability and will establish procedures to inspect the as-manufactured material for contamination and manufacturing defects.	9.1
COL Item 9.2-1:	A COL applicant that references the NuScale Power Plant design certification will select the appropriate chemicals for the reactor component cooling water system based on site-specific water quality and materials requirements.	9.2
COL Item 9.2-2:	A COL applicant that references the NuScale Power Plant design certification will describe the source and pre-treatment methods of potable water for the site, including the use of associated pumps and storage tanks.	9.2
COL Item 9.2-3:	A COL applicant that references the NuScale Power Plant design certification will describe the method for sanitary waste storage and disposal, including associated treatment facilities.	9.2
COL Item 9.2-4:	A COL applicant that references the NuScale Power Plant design certification will provide details on the prevention of long-term corrosion and organic fouling in the site cooling water system.	9.2
COL Item 9.2-5:	A COL applicant that references the NuScale Power Plant design certification will identify the site-specific water source and provide a water treatment system that is capable of producing water that meets the plant water chemistry requirements.	9.2
COL Item 9.3-1:	A COL applicant that references the NuScale Power Plant design certification will submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems.	9.3
COL Item 9.3-2:	~~A COL applicant that references the NuScale Power Plant design certification will develop the post-accident sampling contingency plans for using the process sampling system and the containment evacuation system off-line radiation monitor to obtain reactor coolant and containment atmosphere samples. The contingency plan will describe the process for collecting representative samples and disposing radioactive samples. A COL applicant will identify temporary equipment (e.g., temporary shielding, sample transport cask, etc.) required to support post-accident sampling.~~Not used.	9.3
COL Item 9.4-1:	A COL applicant that references the NuScale Power Plant design certification will specify a periodic testing and inspection program for the normal control room heating ventilation and air conditioning system.	9.4
COL Item 9.4-2:	A COL applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the Reactor Building heating ventilation and air conditioning system in accordance with Regulatory Guide 1.140.	9.4
COL Item 9.4-3:	A COL applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the Radioactive Waste Building heating ventilation and air conditioning system.	9.4
COL Item 9.4-4:	A COL applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the Turbine Building heating ventilation and air conditioning system.	9.4
COL Item 9.5-1:	A COL applicant that references the NuScale Power Plant design certification will provide a description of the offsite communication system, how that system interfaces with the onsite communications system, as well as how continuous communications capability is maintained to ensure effective command and control with onsite and offsite resources during both normal and emergency situations.	9.5
COL Item 9.5-2:	A COL applicant that references the NuScale Power Plant design certification will determine the location for the security power equipment within a vital area in accordance with 10 CFR 73.55(e)(9)(vi)(B).	9.5
COL Item 10.2-1:	Not used.	10.2
COL Item 10.2-2:	Not used.	10.2
COL Item 10.2-3:	Not ~~U~~used.	10.2

RAI 03.06.02-6, RAI 03.08.04-10, RAI 05.03.01-3, RAI 06.02.04-8, RAI 08.01-1, RAI 08.01-1S1, RAI 08.02-4, RAI 08.02-6, RAI 08.03.02-1, RAI 09.02.06-1, RAI 09.03.02-3S1, RAI 10.02-3, RAI 10.02.03-1, RAI 10.02.03-2, RAI 10.03.06-4, RAI 10.04.07-1, RAI 14.03.12-2, RAI 14.03.12-3

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 1.0, Rev 2: Introduction and Interfaces	II.1	No Specific Acceptance Criteria	-	No Specific Acceptance Criteria.	Not Applicable
SRP 1.0, Rev 2: Introduction and Interfaces	II.2	SRP Acceptance Criteria Associated with Each Referenced SRP section	Conforms	None.	Ch 1
SRP 1.0, Rev 2: Introduction and Interfaces	II.3	Performance of New Safety Features and Design Qualification Testing Requirements	Conforms	None.	Ch 1
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	II.1	Specific SRP Acceptance Criteria Contained in Related SRP Chapter 2 or Other Referenced SRP sections	Conforms	This acceptance criterion is a pointer to other SRP sections.	2.0
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	II.2	COL Application Referencing an Early Site Permit	Not Applicable	This acceptance criterion is applicable only to COL applicants that do not reference the DCA.	Not Applicable
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	II.3	COL Application Referencing a Certified Design	Not Applicable	This acceptance criterion is for COL applicants to meet the design parameters established in the DCA.	Not Applicable
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	II.4	COL Application Referencing an Early Site Permit and a Certified Design	Not Applicable	This acceptance criterion is for COL applicants to meet the design parameters established in the DCA.	Not Applicable
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	II.5	COL Application Referencing Neither an Early Site Permit Nor a Certified Design	Not Applicable	This acceptance criterion is applicable only to COL applicants that do not reference the DCA.	Not Applicable
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	App A	Table 1: Examples of Site Characteristics and Site Parameters	Partially Conforms	NuScale provides design parameters where applicable.	Table 2.0-1
SRP 2.0, (March 2007): Site Characteristics and Site Parameters	App A	Table 2: Examples of Site-Related Design Parameters and Design Characteristics	Partially Conforms	NuScale provides design parameters where applicable.	Table 2.0-1
SRP 2.1.1, Rev 3: Site Location and Description	All	Specification of Location and Site Area Map	Not Applicable	Site-specific.	Not Applicable

Table 1.9-3: Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS) (Continued)

SRP or DSRS Section, Rev: Title	AC	AC Title/Description	Conformance Status	Comments	Section
SRP 13.3, Rev 3: Emergency Planning	II.23	ITAAC Associated with Combined License Application	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.24	Generic Emergency Planning ITAAC	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.25	Design and Implementation of Emergency Response Facilities	Partially Conforms	The NuScale design includes a technical support center. The operational support center and the emergency operations facility are the responsibility of the COL applicant that references the NuScale certified design.	13.3
SRP 13.3, Rev 3: Emergency Planning	II.26	Safety Parameter Display System	Conforms	Safety parameter displays are provided in the technical support center. The emergency operations facility is the responsibility of the COL applicant that references the NuScale design certification.	13.3
SRP 13.3, Rev 3: Emergency Planning	II.27	Reactor Coolant System and Containment Sampling	Departure~~Partially Conforms~~	The NuScale design supports an exemption from 10 CFR 50.34(f)(2)(viii).~~Programmatic aspects of post-accident sampling are the responsibility of the COL applicant.~~	9.3.2
SRP 13.3, Rev 3: Emergency Planning	II.28	Containment Monitoring and Continuous Sampling from Potential Accident Release Points	Partially Conforms	Programmatic aspects of containment ~~sampling~~and effluent monitoring are the responsibility of the COL applicant.	9.3.2, 11.5
SRP 13.3, Rev 3: Emergency Planning	II.29	NRC Notifications and Communications	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.30	Generic Communications and Commission Orders Pertaining to Emergency Planning	Not Applicable	COL applicant.	Not Applicable
SRP 13.3, Rev 3: Emergency Planning	II.31	Operational Programs	Not Applicable	COL applicant.	Not Applicable
SRP 13.4, Rev 3: Operational Programs	Not Applicable	Various (Including Attachment, Sample FSAR Table 13.4-x)	Not Applicable	There are no specific requirements for this SRP section.	Not Applicable
SRP 13.5.1.1, Rev 1: Administrative Procedures - General	All	Various	Not Applicable	COL applicant.	Not Applicable

RAI 03.09.06-11S1, RAI 06.02.04-4S1, RAI 06.02.04-4S2, RAI 06.02.04-7S1, RAI 06.02.04-9, RAI 06.02.04-9S1, RAI 08.01-1, RAI 08.02-4, RAI 08.02-6, RAI 08.03.02-1, RAI 09.02.06-1, RAI 09.03.02-2S1, RAI 09.03.02-3S1, RAI 12.03-31, RAI 12.03-32, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35, RAI 12.03-36, RAI 12.03-37, RAI 12.03-39, RAI 12.03-40, RAI 12.03-64

Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)

Item	Regulation Description / Title	Conformance Status	Comments	Section
50.34(f)(1)(i)	Perform a plant/site-specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant (II.B.8)	Partially Conforms	Design certification will address reliability of core and containment heat removal systems, with an update required by COL applicant to reflect site-specific conditions.	19.0 19.1 19.2
50.34(f)(1)(ii)	Perform an evaluation of the proposed auxiliary feedwater system (II.E.1.1)	Not Applicable	This rule requires an evaluation of proposed PWR auxiliary feedwater (AFW) systems. The NuScale plant design does have an AFW system like a typical LWR. Neither the literal language nor the intent of this rule applies to the NuScale design.	Not Applicable
50.34(f)(1)(iii)	Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA (II.K.2.16 and II.K.3.25)	Not Applicable	The NuScale reactor design differs from large PWRs because the NuScale design does not require or include reactor coolant pumps. Rather, the NuScale design uses passive natural circulation of the primary coolant, eliminating the need for reactor coolant pumps.	Not Applicable
50.34(f)(1)(iv)	Perform an analysis of the probability of a small-break LOCA caused by a stuck-open power-operated relief valve (PORV) (II.K.3.2)	Not Applicable	This guidance is applicable only to PWRs that are designed with power-operated pressurizer relief valves. The NuScale design does not use power-operated relief valves.	Not Applicable
50.34(f)(1)(v)	Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection and reactor core isolation cooling system initiation levels (II.K.3.13)	Not Applicable	This requirement applies only to BWRs.	Not Applicable
50.34(f)(1)(vi)	Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves (II.K.3.16)	Not Applicable	This requirement applies only to BWRs. Regardless, the issue contemplated by this requirement was related to power-operated relief valves. The NuScale design does not use power-operated relief valves.	Not Applicable
50.34(f)(1)(vii)	Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system design modifications that would eliminate the need for manual activation (II.K.3.18)	Not Applicable	This requirement applies only to BWRs.	Not Applicable

Table 1.9-5: Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933) (Continued)

Item	Regulation Description / Title	Conformance Status	Comments	Section
50.34(f)(2)(v)	Provide for automatic indication of the bypassed and operable status of safety systems (I.D.3)	Conforms	None.	7.1 7.2.4 7.2.13
50.34(f)(2)(vi)	Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)	Departure	The venting of noncondensible gases is unnecessary to ensure long term core cooling capability.	5.4.4
50.34(f)(2)(vii)	Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access (II.B.2)	Conforms	~~None.~~The NuScale design does not contain vital areas, as defined by NUREG-0737, Item II.B.2, other than the main control room and technical support center. Protection of necessary equipment from radiation is reasonably assured through demonstrating equipment survivability.	~~12.2~~ ~~12.3.1~~12.4 ~~12.4~~19.2
50.34(f)(2)(viii)	Provide capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials (II.B.3)	Departure~~Partially Conforms~~	The NuScale design does not rely on primary coolant or containment samples to assess the extent of potential core damage. The NuScale design relies upon radiation monitors under the bioshield and core exit temperature indications for this assessment. The NuScale design supports an exemption from 10 CFR 50.34(f)(2)(viii) design criterion for obtaining and analyzing post-accident samples of the reactor coolant system and containment without exceeding prescribed radiation dose limits.~~As described by SRP 9.3.2, I.6, and RG 1.206, C.I.9.3.2, a post-accident sampling system is not required provided that the guidance provided in SRP 9.3.2 for utilizing the normal process sampling system (post-accident) has been satisfied.~~	9.3.2 11.5 12.4
50.34(f)(2)(ix)	Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction (II.B.8)	Not Applicable	Pursuant to 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), Paragraph (f)(2)(ix) is excluded from the information required to be included in an application for a design certification.	Not Applicable

RAI 02.03.01-4, RAI 09.03.02-3S1, RAI 19-36

Table 1.9-8: Conformance with SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs"

Issue	Description	Conformance Status	Comments	Section
I.A	Use of a Physically-Based Source Term: Incorporation of engineering judgment and a more realistic source term in design that deviates from the siting requirements in 10 CFR 100.	Conforms	None.	15.0.3 15.10
I.B	Anticipated Transient without SCRAM (ATWS): Position on the current practices and design features to achieve a high degree of protection against an ATWS.	Partially Conforms	The NuScale design relies on diversity within the module protection system (MPS) to reduce the risk associated with ATWS events.	15.8
I.C	Mid-Loop Operation: Position on design features necessary to ensure a high degree of reliability of RHR systems in PWR.	Not Applicable	Design does not use external loops and no drain down condition for refueling.	Not Applicable
I.D	Station Blackout (SBO): Position on methods to mitigate the effects of a loss of all AC power.	Not Applicable	The relevance of the SECY-90-016 SBO issue to passive ALWR designs was deferred to and addressed in Section F of SECY-94-084 and SECY-95-132. The NuScale design conforms to the passive plant guidance these documents.	Not Applicable
I.E	Fire Protection: Positions on design configuration and features the fire protection system and other management schemes to ensure safe shutdown of the reactor.	Conforms	None.	Appendix 9A
I.F	Intersystem LOCA: Position on acceptable design practices and preventative measures to minimize the probability of an ISLOCA.	Conforms	None.	9.3.4 19.2.2
I.G	Hydrogen Control: Position on acceptable requirements to measure and mitigate the effects of hydrogen produced due to a water reaction with zirconium fuel cladding.	Partially Conforms		6.2.5
I.H	Core Debris Coolability: Acceptability criteria for cooling area and quenching ability regarding corium interaction with concrete.	Conforms	None.	19.2
I.I	High-Pressure Core Melt Ejection: Position on acceptable design features to prevent the event of a high-pressure core melt ejection.	Conforms	None.	19.2.3
I.J	Containment Performance: Position on acceptable conditional containment failure probabilities or other analyses to ensure a high degree of protection from the containment.	Conforms	None.	19.1 19.2
I.K	Dedicated Containment Vent Penetration: Position for a dedicated vent penetration to preclude containment failure resulting from a containment over-pressurization event.	Conforms	None.	19.2.4

Table 1.9-8: Conformance with SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs" (Continued)

Issue	Description	Conformance Status	Comments	Section
II.I	Post-Accident Sampling System: Position on the required capability to analyze dissolved hydrogen, oxygen, and chloride in accordance with applicable regulations.	Departure~~Conforms~~	The NuScale design supports an exemption from 10 CFR 50.34(f)(2)(viii).~~As described by SRP 9.3.2, I.6, and RG 1.206, C.I.9.3.2, a post-accident sampling system is not required provided that the guidance provided in SRP 9.3.2 for utilizing the normal process sampling system (post-accident) has been satisfied.~~	9.3.2
II.J	Level of Detail: Position on a design certification submittal with depth of detail similar to that in an FSAR.	Conforms	None.	All FSAR Sections
II.K	Prototyping: No guidance provided; information only	Conforms	None.	1.5
II.L	ITAAC: Position on providing ITAAC to demonstrate that a nuclear power plant referencing a certified design is built and operates consistent with the design certification.	Conforms	None.	14.3
II.M	Reliability Assurance Program: Position on providing a description of purpose, scope, objectives, and implementation of a design reliability assurance program.	Conforms	None.	17.4
II.N	Site-Specific PRAs and Analyses of External Events: Position on the inclusion of external event analysis beyond the design basis that needs to be addressed as part of the plant PRA during the design certification review.	Conforms	None.	19.1
II.O	Severe Accident Mitigation Design Alternatives (SAMDAs): Position on the consideration of SAMDA as part of the final design approval/design certification of an advanced reactor.	Conforms	None.	19.2.6
II.P	Generic Rulemaking Related to Design Certification: No guidance provided; information only.	Not Applicable	Information Only.	Not Applicable
II.Q	Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems: Position on the use of defense-in-depth and diversity of instrumentation and control systems as part of the final design approval/design certification of an advanced reactor.	Conforms	None.	7.1.5
II.R	Multiple SG Tube Failures: Position on requiring that analysis of multiple SG Tube Failures of 2 to 5 SG tubes be included in the application for design certification of passive ALWRs.	Conforms	None.	15.6 19.1
II.S	PRA Beyond Design Certification: Position on requiring conversion of the design certification PRA into a plant-specific PRA	Conforms	None.	19.1

RAI 03.02.01-2, RAI 03.02.01-3, RAI 03.02.02-2, RAI 03.02.02-6, RAI 03.08.02-14, RAI 03.08.04-1S1, RAI 03.09.02-64, RAI 05.04.02.01-6, RAI 06.02.04-2, RAI 09.01.03-1, RAI 09.02.02-1, RAI 09.02.04-1, RAI 09.02.04-1S1, RAI 09.02.05-1, RAI 09.02.06-1, RAI 09.02.07-4, RAI 09.02.07-5, RAI 09.02.09-2, RAI 09.03.02-3S1, RAI 09.03.04-5, RAI 09.04.02-1, RAI 09.04.02-1S1, RAI 10.04.07-2, RAI 11.02-1, RAI 12.02-32, RAI 15-17, RAI 15-17S1, RAI 19-14

Table 3.2-1: Classification of Structures, Systems, and Components

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
CNTS, Containment System							
All components (except as listed below)	RXB	A1	N/A	Q	None	B	I
• CVC Injection Check Valve • CVC Discharge Excess Flow Check Valve • CVC PZR Spray Check Valve	RXB	B2	None	AQ-S	None	C	I
• CVC Injection & Discharge Nozzles • CVC PZR Spray Nozzle • CVC PZR Spray CIV • CVC RPV High Point Degasification Nozzle • CVC RPV High Point Degasification CIV • RVV & RRV Trip/Reset # 1 & 2 Nozzles • RVV Trip 1 & 2/Reset #3 Nozzles • CVC Injection & Discharge CIVs	RXB	A1	N/A	Q	None	A	I
• NPM Lifting Lugs • Top Support Structure • Top Support Structure Diagonal Lifting Braces	RXB	B1	None	AQ-S	• ~~ANSI/ANS 57.1-1992~~ • ANSI N14.6~~ASME NOG-1~~ • NUREG-0612~~0554~~	N/A	I
• CNV Fasteners • Hydraulic skid • CNV Seismic Shear Lug • CNV CRDM Support Frame • Containment Pressure Transducer (Narrow Range) • Containment Water Level Sensors (Radar Transceiver) • SG 1 & 2 Steam Temperature Sensors (RTD)	RXB	A1	N/A	Q	None	N/A	I
CNTS CFDS Piping in containment	RXB	B2	None	AQ-S	None	B	II
Piping from (CES, CFDS, FWS, MSS, and RCCWS) CIVs to disconnect flange (outside containment)	RXB	B2	None	AQ-S	None	D	I
CVCS Piping from CIVs to disconnect flange (outside containment)	RXB	B2	None	AQ-S	None	C	I
CIV Close and Open Position Sensors: • CES, Inboard and Outboard • CFDS, Inboard and Outboard • CVCS, Inboard and Outboard PZR Spray Line • CVCS, Inboard and Outboard RCS Discharge • CVCS, Inboard and Outboard RCS Injection • CVCS, Inboard and Outboard RPV High-Point Degasification • ~~FWS, Supply to SGs and DHR HXs FWIV~~ • RCCWS, Inboard and Outboard Return and Supply • SGS, Steam Supply CIV/MSIVs and CIV/MSIV Bypasses	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
CIV Close and Open Position Indication • FWS, Supply to SGs and DHR HXs FWIV	RXB	A1	None	Q	None	N/A	I
Containment Pressure Transducer (Wide Range)	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
• Containment Air Temperature (RTDs) • FW Temperature Transducers	RXB	B2	None	AQ-S	None	N/A	II
SGS, Steam Generator System							
• SG tubes • ~~Feedwater plenums~~Integral steam plenums • ~~Steam plenums~~Feedwater plenums	RXB	A1	N/A	Q	None	A	I
• SG tube supports • Upper and lower SG supports	RXB	A1	N/A	Q	None	N/A	I

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. RG 1.29 or RG 1.143) (Note 5)
Radiation Monitor	RXB	B2	None	AQ	• ANSI N42.18-2004 • ANSI/HPS N13.1-2011 • Table 1 of SRP 11.5 • Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	N/A	III
Sample Vessel Radiation Transmitter	RXB	B2	None	AQ	• ANSI N42.18-2004 • Table 1 of SRP 11.5	N/A	III
Gas Discharge Radiation Transmitter	RXB	B2	None	AQ	• ANSI/HPS N13.1-2011 • Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	N/A	III
• PSS Sample Panel Inlet and Outlet Isolation Valves • Vacuum Pump Bypass Valve	RXB	B2	None	AQ	Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	D	III
• Charcoal Pre-Filter • Charcoal Filter • Discharge Filter	RXB	B2	None	AQ	• RG 1.140	D	III
• Containment Service Air Pressure Valve • Sample Vessel Drain Sampler	RXB	B2	None	None	None	D	III
CFDS, Containment Flooding And Drain System							
All components (except as listed below)	RXB	B2	None	None	None	D	III
CFD Module Post Accident Monitoring~~Sampling~~ Return Valves	RXB	B2	None	AQ	Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	D	III
Radiation Transmitter	RXB	B2	None	AQ	ANSI N42.18-2004	N/A	III
RCCWS, Reactor Component Cooling Water System							
All components (except as listed below)	RXB	B2	None	None	None	D	III
Radioactivity Transmitters for: • RCCW CE Vacuum Pumps and Condensers • RCCW CVC NRHXs and PSS Coolers • RCCW PSS Cooling Water TCU	RXB	B2	None	AQ	ANSI N42.18-2004	N/A	III
RCCWS instrumentation	RXB	B2	None	None	None	N/A	III
PSS, Process Sampling System							
All components (except as listed below)	RXB, TGB	B2	None	None	None	N/A	III
Reactor coolant discharge sample line isolation valve	RXB	B2	None	AQ	ANSI N13.1	D	III
Primary sampling system analysis panel	RXB	B2	None	AQ	ANSI N13.1	N/A	III
• Containment evacuation system sample line isolation valve	RXB	B2	None	AQ	• ANSI N13.1 • Pressure boundary components of any monitoring path outside of containment shall be designed to withstand combustion events corresponding to the capability of containment.	D	III

RAI 09.03.02-3S1

Containment ~~Monitoring and~~ Sampling System

RAI 09.03.02-3S1

During normal operation, the containment vessel is monitored for hydrogen and oxygen gas concentration using the containment sampling system. From a sample point downstream of the discharge of the CES vacuum pumps condenser for each CNV, the CSS sample pump draws the sample gas through in-line hydrogen and oxygen monitors and returns the sample gas to the CES. The in-line hydrogen and oxygen monitors provide continuous gas concentration indication to the MCR. Grab sampling capability is provided from the off-line CES radiation monitor as described in Section 11.5. The analysis of grab samples provides an independent indication of process hydrogen and oxygen content to validate in-line monitor indication and serves as a redundant means to determine process gas concentration in the event that in-line hydrogen and oxygen monitoring is unavailable. The CSS also provides post-accident hydrogen and oxygen monitoring of containment atmosphere following significant beyond design basis accidents.

RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

~~For post-accident conditions, the gas monitors used during normal plant conditions provide hydrogen and oxygen indication. To establish sample flow, the CIVs for CES and CFDS are opened, and the CSS sample pump aligned to take a suction from the CES vacuum pump bypass line. The sample gas is returned to the CNV using the associated containment flooding and drain system (CFDS) piping and valves, downstream of the CFDS return-side NPM isolation valves as shown on Figure 9.3.6-2.~~

RAI 09.03.02-3S1

~~For both the normal and post-accident hydrogen and oxygen monitoring sampling alignments there are no shared gas monitoring components between NPMs, a CSS is provided for each module.~~

RAI 09.03.02-3S1, RAI 12.03-67

Table 9.3.2-2 summarizes the CSS sample point and the analysis capability of the CSS.~~installed continuous sample panel.~~ A diagram of the CSS is provided in Figure 9.3.2-1.

Secondary Sampling System

The SSS provides a means for monitoring and collecting fluid samples in the steam cycle systems. The systems serviced by the SSS are the condensate and feedwater system, the main steam system (MSS), and the auxiliary boiler system (ABS). Emphasis is placed on continuous monitoring of the secondary system condensate pump discharge, condensate polisher effluents, feedwater, and main steam. The SSS also includes grab sample capability for diagnostic sampling.

addition, grab sampling capability is provided from the off-line CES radiation monitor as described in Section 11.5. Normal sample points of the CSS are provided in Table 9.3.2-2.

For sampling at power, the SSS collects samples from the ABS, condensate and feedwater system, and the MSS. Emphasis is placed on continuous monitoring of the secondary system hotwells, condensate pump discharge, condensate polisher effluents, feedwater, and main steam. The SSS also includes grab sample capability for diagnostic sampling. Normal operation sample points of the SSS are provided in Table 9.3.2-3.

Local sample points are provided for systems not being serviced by the primary sampling system, the CSS, or the SSS. These local sample points for normal operation sampling are provided in Table 9.3.2-4. The frequency for sample collection and required analyses for these local process sample points are addressed in the primary, secondary, and ancillary chemistry program and procedures.

Off-Normal Operations

RAI 09.03.02-2S1, RAI 09.03.02-3S1

The NuScale design supports an exemption from 10 CFR 50.34(f)(2)(viii) that requires capability for obtaining and analyzing post-accident samples of reactor coolant and containment atmosphere. However, post-accident sampling is an option that operators may implement to obtain additional information for accident recovery. The PSS design includes capabilities to collect post-accident samples of reactor coolant. Additionally, the PSS design includes capability to monitor hydrogen and oxygen in containment atmosphere following significant beyond design-basis accident for combustible gas control and accident management in compliance with 10 CFR 50.44(c)(4). Off-normal operations of the PSS, therefore, are to support post-accident hydrogen and oxygen monitoring of containment atmosphere, and post-accident sampling of reactor coolant.~~A dedicated post-accident sampling system is not provided; therefore, off-normal operation of the PSS involves sampling under post-accident conditions. It is expected that the PSS would be used for post-accident sampling only if the information sought is essential and cannot be determined or estimated by other means. The primary means to detect and monitor fuel damage uses core exit temperature indication and radiation monitors located under the NPM bio-shields. Therefore, post-accident sampling is not used as a primary means for identifying fuel damage. However, post-accident sampling may be used to provide additional assessment of the extent of fuel damage.~~

RAI 09.03.02-2S1, RAI 09.03.02-3S1

~~The configuration of the PSS supports development of contingency plans that would permit post-accident sampling of the reactor coolant and containment atmosphere using available CVCS and CES sample paths and equipment. Prior to using PSS for post-accident sampling, contingency plans will establish expected dose rates under the prevailing conditions and sampling is delayed or temporary~~

RAI 09.03.02-2S1, RAI 09.03.02-3S1

The NuScale design supports an exemption from 10 CFR 50.34(f)(2)(viii) that requires capability for obtaining and analyzing post-accident samples of reactor coolant and containment atmosphere. The PSS design includes capability to monitor hydrogen and oxygen in containment atmosphere following significant beyond design-basis accident for combustible gas control and accident management in compliance with 10 CFR 50.44(c)(4). Off-normal operations of the PSS, therefore, are to support post-accident hydrogen and oxygen monitoring of containment atmosphere.~~A dedicated post-accident sampling system is not provided; therefore, off-normal operation of the PSS involves sampling under post-accident conditions. It is expected that the PSS would be used for post-accident sampling only if the information sought is essential and cannot be determined or estimated by other means. The primary means to detect and monitor fuel damage uses core exit temperature indication and radiation monitors located under the NPM bio-shields. Therefore, post-accident sampling is not used as a primary means for identifying fuel damage. However, post-accident sampling may be used to provide additional assessment of the extent of fuel damage.~~

RAI 09.03.02-2S1, RAI 09.03.02-3S1

~~The configuration of the PSS supports development of contingency plans that would permit post-accident sampling of the reactor coolant and containment atmosphere using available CVCS and CES sample paths and equipment. Prior to using PSS for post-accident sampling, contingency plans will establish expected dose rates under the prevailing conditions and sampling is delayed or temporary shielding is provided as necessary to protect personnel and accommodate sampling and purge water disposal.~~

RAI 09.03.02-2S1, RAI 09.03.02-3S1

~~Due to the unique design of the NPM and emergency core cooling system, sampling of containment liquid in a post-accident scenario is not practical or beneficial. During emergency core cooling system operation, the lower containment fills with condensing steam that has been released from the reactor vent valves. When this condensate reaches a sufficient level above the reactor recirculation valves, it returns to the reactor vessel. Sampling and analysis of this condensate before it returns and mixes with the coolant remaining in the reactor vessel would not provide useful information and therefore sampling of containment liquid is not performed.~~

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

Since the PSS connects outside of ~~the CIVs associated with the CVCS and CES~~CNV, post-accident ~~sampling~~ hydrogen and oxygen monitoring with PSS requires opening the~~se~~ CES and CFDS CIVs. ~~It is expected that post-accident sampling will be performed when plant conditions are amenable to opening CIVs without overriding the containment isolation signal (CIS) to minimize risks to plant equipment and personnel.~~ If post-accident ~~sampling~~hydrogen and oxygen monitoring must be performed while containment isolation conditions exist, overriding the containment isolation signal (CIS) is required.

~~shielding is provided as necessary to protect personnel and accommodate sampling and purge water disposal.~~

RAI 09.03.02-2S1, RAI 09.03.02-3S1

~~Due to the unique design of the NPM and emergency core cooling system, sampling of containment liquid in a post-accident scenario is not practical or beneficial. During emergency core cooling system operation, the lower containment fills with condensing steam that has been released from the reactor vent valves. When this condensate reaches a sufficient level above the reactor recirculation valves, it returns to the reactor vessel. Sampling and analysis of this condensate before it returns and mixes with the coolant remaining in the reactor vessel would not provide useful information and therefore sampling of containment liquid is not performed.~~

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

Since the PSS connects outside of ~~the CIVs associated with the CVCS and CES~~CNV, post-accident sampling and hydrogen and oxygen monitoring with PSS requires opening the~~se~~ CVCS, CES, and CFDS CIVs. It is expected that post-accident sampling will be performed when plant conditions are amenable to opening CIVs without overriding the containment isolation signal (CIS) to minimize risks to plant equipment and personnel. If post-accident sampling must be performed while containment isolation conditions exist, overriding the containment isolation signal (CIS) is required.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

The CIV hydraulic actuator design and control as described in Section 6.2.4.2.2 are utilized in opening the CIV. Design features of the CIV hydraulic actuators ensure that the valves can be re-opened following the design basis event. The hydraulic cylinder on the actuator applies force to open the CIV. The hydraulic cylinders are pressurized by the hydraulic skid. The hydraulic pump drivers on the CIV hydraulic skids is powered by the ELVS, which has a backup power source if normal AC power source is not available. The hydraulic control skid is also designed to support a limited number of CIV re-openings without reliance on AC power after a design basis event. ~~However, the post-accident sampling will be performed when there is power available to the CIV hydraulic skids such that the valves can be reopened using normal control provided in the MCR.~~

Reactor Coolant Post-Accident Sampling

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

Post-accident water level in the reactor vessel for some events may be below the level of the normal CVCS discharge line nozzle on the reactor vessel. Under these conditions, RCS sampling would be delayed unless restoration of water level above the CVCS discharge line reactor vessel nozzle is directed by emergency operating procedures or severe accident mitigating guidelines.

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 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

The CIV hydraulic actuator design and control as described in Section 6.2.4.2.2 are utilized in opening the CIV. Design features of the CIV hydraulic actuators and hydraulic control skids ensure that the valves can be re-opened following the design basis event. The hydraulic cylinder on the actuator applies force to open the CIV. The hydraulic cylinders are pressurized by the hydraulic control skid. The hydraulic pump drivers on the CIV hydraulic control skids ~~is~~are powered by the ELVS, which has a backup power source if normal AC power source is not available. The hydraulic control skids are also designed with a set of accumulators to support a limited number of reopenings of the CIVs after a design basis event without reliance on AC power.~~The hydraulic control skid is also designed to support a limited number of CIV re-openings without reliance on AC power after a design basis event. However, the post-accident sampling will be performed when there is power available to the CIV hydraulic skids such that the valves can be reopened using normal control provided in the MCR.~~

RAI 09.03.02-3S1

~~Reactor Coolant Post-Accident Sampling~~

~~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 09.03.02-3S1

~~Post-accident water level in the reactor vessel for some events may be below the level of the normal CVCS discharge line nozzle on the reactor vessel. Under these conditions, RCS sampling would be delayed unless restoration of water level above the CVCS discharge line reactor vessel nozzle is directed by emergency operating procedures or severe accident mitigating guidelines.~~

~~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 09.03.02-3S1

~~At temperatures below 200 degrees Fahrenheit with RCS pressure insufficient to withdraw a sample, nitrogen can be injected into RCS to raise pressure and draw samples out of the reactor vessel. Nitrogen is provided via the nitrogen distribution system connection to the RPV high point degasification line as shown on Figure 9.3.4-1.~~

~~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 09.03.02-3S1

~~When adequate RCS level and pressure have been established, the CVCS discharge line CIVs are opened to collect the reactor coolant post-accident sample via the CVCS sample line flow path to the primary sampling system sample panel. Collection of a pressurized reactor coolant sample is not required. The purged sample may be disposed to the radioactive waste drain and subsequently sent to the liquid radioactive waste system (LRWS) or collected in a temporary disposal tank if radiation levels are expected to be too high for disposal to the LRWS. Use of temporary shielding and portable air filtration equipment may be required to support reactor coolant post-accident sample collection. If necessary, the sample vessel is put inside a radiation-shielded cask and the cask is loaded to a transport cart for movement to the counting room and hot lab. Provisions for portable shielding may be required to protect personnel performing sample transport.~~

At temperatures below 200 degrees Fahrenheit with RCS pressure insufficient to withdraw a sample, nitrogen can be injected into RCS to raise pressure and draw samples out of the reactor vessel. Nitrogen is provided via the nitrogen distribution system connection to the RPV high point degasification line as shown on Figure 9.3.4-1.

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

When adequate RCS level and pressure have been established, the CVCS discharge line CIVs are opened to collect the reactor coolant post-accident sample via the CVCS sample line flow path to the primary sampling system sample panel. Collection of a pressurized reactor coolant sample is not required. The purged sample may be disposed to the radioactive waste drain and subsequently sent to the liquid radioactive waste system (LRWS) or collected in a temporary disposal tank if radiation levels are expected to be too high for disposal to the LRWS. Use of temporary shielding and portable air filtration equipment may be required to support reactor coolant post-accident sample collection. If necessary, the sample vessel is put inside a radiation-shielded cask and the cask is loaded to a transport cart for movement to the counting room and hot lab. Provisions for portable shielding may be required to protect personnel performing sample transport.

Containment Gas Post-Accident Monitoring and Sampling

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

The PSS design has capabilities for ~~post-accident containment gas sampling to allow~~ monitoring of hydrogen and oxygen inside containment post-accident for combustible gas control. CNV structural integrity is not challenged by combustion events propagated by combustible gas concentrations generated within the first 72 hours of any design basis or beyond design basis event, and no mitigating actions are required during this period. As a result, monitoring of hydrogen and oxygen concentrations in the CNV to inform mitigating actions is not required prior to 72 hours after initiation of an event.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

Post-accident ~~sampling and~~hydrogen and oxygen monitoring of containment gas can be initiated when plant conditions are amenable to opening the CES and CFDS CIVs ~~without overriding CIS~~, and do not exceed design limitations of the CES and CFDS piping. The design pressure of the CES piping downstream of the CIVs are 250 psig. The design pressure the CFDS piping downstream of the CIVs are 150 psig.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

~~While t~~The plant responses in accident conditions show the containment pressure is reduced to 150 psia in approximately two hours after event initiation.~~, the CIS is not cleared by this time to open the CES and CFDS CIVs for sampling. The RCS is expected to cool down to less than 200 degrees Fahrenheit in approximately 24 hours after event initiation such that there is automatic operating bypass of CIS. Therefore, the containment gas post-accident sampling can be performed without overriding the CIS~~ For severe accidents with core damage, the CIS signal may not

clear by this time and opening the CIVs to support hydrogen and oxygen monitoring activity would require overriding the CIS via operator action outside the MCR. It is expected that the CES and CFDS CIVs can be opened and hydrogen and oxygen monitoring can be performed in approximately 24 hours after event initiation.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

To initiate post-accident ~~containment gas sampling~~hydrogen and oxygen monitoring, the CES and CFDS CIVs are opened ~~using normal means in the MCR~~ to establish the ~~monitoring and sampling~~containment gas flow paths to the hydrogen and oxygen monitor located outside the containment and return the gas back to the containment after monitoring. The containment gas released from the CNV is routed from the CES to the containment sampling system ~~for~~which is equipped with online hydrogen and oxygen monitoring equipment. The ~~sampled~~ gas is then returned to the CNV via the containment sampling system~~PSS~~ effluent discharge line connected to the CFDS return line to CNV as shown on Figure 9.3.6-2. Returning the ~~sampled~~ gas back to the CNV eliminates releasing effluent to the environment.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

The CES piping is sloped to allow condensed liquid to drain back the CNV and the CES sample line piping is heat traced to prevent the build-up of condensate within the containment gas monitoring lines and analyzer to ensure monitoring capability under accident conditions. The heat trace of CES sample line is powered by the ~~EDNS~~ELVS, which has a backup power source if the normal sources of AC electrical power are unavailable.

RAI 09.03.02-2S1, RAI 09.03.02-3S1

~~Grab~~Post-accident containment gas grab sampling capability is provided from the off-line CES radiation monitor described in Section 11.5. The analysis of grab samples provides information to allow the licensee to understand the magnitude of any remaining threat that the accident may pose after plant conditions stabilize post-accident. Grab sample analysis also provides independent indication of process hydrogen and oxygen content to validate in-line monitor indication and serves as a redundant means to determine process gas concentration in the event that ~~in-line~~ hydrogen and oxygen monitoring equipment is unavailable. If necessary, the sample vessel is placed inside a radiation-shielded cask and the cask loaded to a transport cart for movement to the counting room and hot lab. Provisions for portable shielding may also be required to protect personnel performing sampling and sample transport.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

COL Item 9.3-2: ~~A COL applicant that references the NuScale Power Plant design certification will develop the post-accident sampling contingency plans for using the process sampling system and the containment evacuation system off-line radiation monitor to obtain reactor coolant and containment atmosphere samples. The contingency plan will describe the process for collecting representative samples and disposing radioactive samples. A COL applicant will identify temporary~~

RAI 09.03.02-3S1

Containment Gas Post-Accident Monitoring ~~and Sampling~~

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

The PSS design has capabilities for ~~post-accident containment gas sampling to allow~~ monitoring of hydrogen and oxygen inside containment post-accident for combustible gas control. CNV structural integrity is not challenged by combustion events propagated by combustible gas concentrations generated within the first 72 hours of any design basis or beyond design basis event, and no mitigating actions are required during this period. As a result, monitoring of hydrogen and oxygen concentrations in the CNV to inform mitigating actions is not required prior to 72 hours after initiation of an event.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

Post-accident ~~sampling and~~hydrogen and oxygen monitoring of containment gas can be initiated when plant conditions are amenable to opening the CES and CFDS CIVs ~~without overriding CIS~~, and do not exceed design limitations of the CES and CFDS piping. The design pressure of the CES piping downstream of the CIVs are 250 psig. The design pressure the CFDS piping downstream of the CIVs are 150 psig.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

~~While t~~The plant responses in accident conditions show the containment pressure is reduced to 150 psia in approximately two hours after event initiation.~~, the CIS is not cleared by this time to open the CES and CFDS CIVs for sampling. The RCS is expected to cool down to less than 200 degrees Fahrenheit in approximately 24 hours after event initiation such that there is automatic operating bypass of CIS. Therefore, the containment gas post-accident sampling can be performed without overriding the CIS~~ For severe accidents with core damage, the CIS signal may not clear by this time and opening the CIVs to support hydrogen and oxygen monitoring activity would require overriding the CIS via operator action outside the MCR. It is expected that the CES and CFDS CIVs can be opened and hydrogen and oxygen monitoring can be performed in approximately 24 hours after event initiation.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

To initiate post-accident ~~containment gas sampling~~hydrogen and oxygen monitoring, the CES and CFDS CIVs are opened ~~using normal means in the MCR~~ to establish the ~~monitoring and sampling~~containment gas flow paths to the hydrogen and oxygen monitor located outside the containment and return the gas back to the containment after monitoring. The containment gas released from the CNV is routed from the CES to the containment sampling system ~~for~~that is equipped with online hydrogen and oxygen monitoring equipment. The ~~sampled~~ gas is then returned to the CNV via the containment sampling system~~PSS~~ effluent discharge line connected to the CFDS return line to CNV as shown on Figure 9.3.6-2. Returning the ~~sampled~~ gas back to the CNV eliminates releasing effluent to the environment.

~~equipment (e.g., temporary shielding, sample transport cask, etc.) required to support post-accident sampling.~~Not used.

9.3.2.3 Safety Evaluation

The PSS has no safety-related or risk-significant functions and is not required to prevent or mitigate the consequences of a design basis accident, to shut down the reactor and maintain safe shutdown conditions, or to maintain the integrity of the RCPB.

Consistent with GDC 1, process sampling system SSC are designed, fabricated, erected, and tested to appropriate quality standards such that their failure does not impact the function of safety-related or risk-significant systems. The quality group classification of the PSS sample line isolation valves and piping which directly interface with the system being sampled is equivalent to the quality classification of the system to which each sampling line and component is connected. The highest quality group classification of PSS components is Quality Group D conforming to RG 1.26. PSS piping conforms to American Society of Mechanical Engineers (ASME) B31.1 (Reference 9.3.2-2).

General Design Criteria 2 was considered in the design of the PSS. The primary sampling system and the CSS components are located inside the Seismic Category I Reactor Building (RXB). No portions of the PSS or components are safety-related or required to perform a safety-related or risk-significant function. The PSS does not connect to any Seismic Category I piping. The PSS connects to the CVCS and CES on the portions of these systems that are not designed to Seismic Category I requirements. The PSS piping and components downstream of the sample line isolation valve are designed to Seismic Category III, but are upgraded to Seismic Category II (see Section 3.2.1.2) if the routing of PSS piping or the location of components is determined to result in a condition where failure of process sampling system SSC could adversely impact Seismic Category I SSC.

The PSS does not employ sample lines which penetrate the CNV and the reactor pressure vessel; therefore, there is no containment isolation function associated with the system. There is no physical interaction of process sampling system SSC with safety-related SSC. Process sampling system failure does not adversely affect the integrity of safety-related systems.

General Design Criteria 4 was considered in the design of the PSS. The PSS and its components are designed to accommodate the effects of the environmental conditions associated with normal operation and shutdown. Even though the PSS has no safety-related or risk-significant functions, the primary sampling system and the CSS portions of the PSS are expected to operate after design basis accidents; therefore, these PSS subsystems are also designed to withstand postulated accident environmental conditions to ensure that the post-accident sampling function of the system can be performed.

General Design Criteria 5 was considered in the design of the PSS. The sharing of the PSS components between power modules does not affect the ability of the SSC to perform required safety functions, including in the unlikely event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

The CES piping is sloped to allow condensed liquid to drain back the CNV and the CES sample line piping is heat traced to prevent the build-up of condensate within the containment gas monitoring lines and analyzer to ensure monitoring capability under accident conditions. The heat trace of CES sample line is powered by the ~~EDNS~~ELVS, which has a backup power source if the normal sources of AC electrical power are unavailable.

~~RAI 09.03.02-2S1, RAI 09.03.02-3S1~~RAI 09.03.02-3S1

~~Grab sampling capability is provided from the off-line CES radiation monitor described in Section 11.5. The analysis of grab samples provides information to allow the licensee to understand the magnitude of any remaining threat that the accident may pose after plant conditions stabilize post-accident. Grab sample analysis also provides independent indication of process hydrogen and oxygen content to validate in-line monitor indication and serves as a redundant means to determine process gas concentration in the event that in-line hydrogen and oxygen monitoring is unavailable. If necessary, the sample vessel is placed inside a radiation-shielded cask and the cask loaded to a transport cart for movement to the counting room and hot lab. Provisions for portable shielding may also be required to protect personnel performing sampling and sample transport.~~

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

COL Item 9.3-2: ~~A COL applicant that references the NuScale Power Plant design certification will develop the post-accident sampling contingency plans for using the process sampling system and the containment evacuation system off-line radiation monitor to obtain reactor coolant and containment atmosphere samples. The contingency plan will describe the process for collecting representative samples and disposing radioactive samples. A COL applicant will identify temporary equipment (e.g., temporary shielding, sample transport cask, etc.) required to support post-accident sampling.~~Not used.

9.3.2.3 Safety Evaluation

The PSS has no safety-related or risk-significant functions and is not required to prevent or mitigate the consequences of a design basis accident, to shut down the reactor and maintain safe shutdown conditions, or to maintain the integrity of the RCPB.

Consistent with GDC 1, process sampling system SSC are designed, fabricated, erected, and tested to appropriate quality standards such that their failure does not impact the function of safety-related or risk-significant systems. The quality group classification of the PSS sample line isolation valves and piping which directly interface with the system being sampled is equivalent to the quality classification of the system to which each sampling line and component is connected. The highest quality group classification of PSS components is Quality Group D conforming to RG 1.26. PSS piping conforms to American Society of Mechanical Engineers (ASME) B31.1 (Reference 9.3.2-2).

General Design Criteria 2 was considered in the design of the PSS. The primary sampling system and the CSS components are located inside the Seismic Category I

Reactor Building (RXB). No portions of the PSS or components are safety-related or required to perform a safety-related or risk-significant function. The PSS does not connect to any Seismic Category I piping. The PSS connects to the CVCS and CES on the portions of these systems that are not designed to Seismic Category I requirements. The PSS piping and components downstream of the sample line isolation valve are designed to Seismic Category III, but are upgraded to Seismic Category II (see Section 3.2.1.2) if the routing of PSS piping or the location of components is determined to result in a condition where failure of process sampling system SSC could adversely impact Seismic Category I SSC.

The PSS does not employ sample lines which penetrate the CNV and the reactor pressure vessel; therefore, there is no containment isolation function associated with the system. There is no physical interaction of process sampling system SSC with safety-related SSC. Process sampling system failure does not adversely affect the integrity of safety-related systems.

RAI 09.03.02-3S1

General Design Criteria 4 was considered in the design of the PSS. The PSS and its components are designed to accommodate the effects of the environmental conditions associated with normal operation and shutdown. Even though the PSS has no safety-related or risk-significant functions, the primary sampling system and the CSS portions of the PSS are expected to operate after design basis accidents; therefore, these PSS subsystems are also designed to withstand postulated accident environmental conditions to ensure that the post-accident ~~sampling~~monitoring function of the system can be performed.

General Design Criteria 5 was considered in the design of the PSS. The sharing of the PSS components between power modules does not affect the ability of the SSC to perform required safety functions, including in the unlikely event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

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

Sharing of the PSS components across NPMs has been minimized to the extent practical. The shared components are downstream of CIVs to ensure that SSC ability to perform safety functions is not impaired. Shared functions are described in Section 9.3.2.2.

The PSS design satisfies GDC 13 in that sampling of reactor coolant enables the PSS to provide information on variables that can affect the fission process, the integrity of the reactor core, and the RCPB during all normal modes of operation. The PSS is relied upon to collect water and gaseous samples from the RCS and associated auxiliary systems during normal modes of operation.

The PSS design satisfies GDC 14 as it relates to ensuring integrity of the RCPB by sampling reactor coolant for chemicals that can affect the RCPB. Sampling and analysis of reactor coolant samples verify that key chemistry parameters, such as chloride, hydrogen, and oxygen concentrations, are within prescribed limits and that impurities are properly controlled, providing assurance that the many mechanisms for corrosive attack are mitigated and will not adversely affect the RCPB.

RAI 09.03.02-3S1

The PSS design satisfies GDC 64 as it provides the capability to sample and analyze for radioactivity that may be released during normal operations, anticipated operational occurrences, and postulated accidents.~~The PSS can provide information to indicate the potential for breaching a fission product barrier (i.e., fuel cladding, primary coolant pressure boundary, and containment) and information that indicates that a fission product barrier has been breached.~~

RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

The PSS design satisfies 10 CFR 50.34(f)(2)(xvii)(c) by providing capability to monitor hydrogen and oxygen concentration in containment atmosphere~~containment sampling including oxygen and hydrogen analyzers. These monitors are nonsafety-related instruments that continuously monitor oxygen and hydrogen concentrations in containment~~ during operation and ~~are capable of monitoring~~ during beyond design-basis conditions. The monitor is a nonsafety-related instrument that sends output signal to the MCS to provide readout in the main control room.~~The hydrogen analyzer output signal is sent to the MCS, which provides readout in the main control room.~~

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

The PSS design satisfies 10 CFR 50.34(f)(2)(xxvi) (Item III.D.1.1 in NUREG-0737), as it relates to including provisions for leakage control and detection to prevent unnecessarily high exposures to workers and the public and to maintain control and use of the system post-accident. The PSS design includes provisions for leakage control and detection. Flow and pressure instrumentation on the sample lines can provide indication of potential leaks. Radiation monitoring capabilities are provided for detecting excessive radiation level resulting from system leakage. The sample line can be isolated upon detection of high radiation by the CVCS or CES process radiation monitor located upstream of the sample line as shown in Figure 9.3.4-1 and Figure 9.3.6-1 respectively. Excessive radiation level detected by the fixed area radiation monitor located in the primary sampling system or the containment sampling system equipment areas described in Table 12.3-10 can also provide indication of system leakage that warrants system isolation for leakage control.

The PSS design satisfies the requirements of 10 CFR 50.44(c)(4), as the equipment design attributes conform to RG 1.7 regulatory position C.2. It provides the ability to monitor containment hydrogen and oxygen using an in-line monitor for both normal and accident conditions. In addition grab sampling provisions are provided on the CES off-line radiation monitor to validate in-line monitor indication and serve as a redundant means to determine process gas concentration in the event that in-line hydrogen and oxygen monitoring is unavailable.

The PSS design features and configuration support ALARA program goals and objectives with regard to minimizing dose and contamination. The PSS design ensures that the ALARA requirements of 10 CFR 20.1101 and contamination minimization requirements of 10 CFR 20.1406 are addressed.

~~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 09.03.02-3S1

~~Post-accident sampling should be performed when plant conditions are amenable to sampling and CIVs can be opened without bypassing the CIS. However, if post-accident sampling must be commenced prior to clearing CIS, the CIS can be overridden to operate the certain CIVs. The CIS manual override switches (one per division) are provided in the MCR to bypass the CIS and open the CVCS and CFDS CIVs. The CES CIVs are not provided with CIS manual override switches in the MCR. Operator actions outside of the MCR is required to bypass CIS and open CES CIVs.~~

9.3.2.6 References

9.3.2-1 American National Standards Institute/Health Physics Society, "Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities," ANSI/HPS N13.1-2011, Washington, DC.

9.3.2-2 American Society of Mechanical Engineers, Power Piping - ASME Code for Pressure Piping B31, ASME B31.1, New York, NY.

RAI 09.03.04-13, RAI 09.03.02-3S1

Table 9.3.2-1: Primary Sampling System Normal ~~and Post-Accident~~ Sample Points

Sample Points	System	Process Fluid Type	Sampling Methods	Analysis[1]
Chemical and volume control system (CVCS) suction line from RCS, upstream of CVCS purification equipment	CVCS	liquid	continuous semi-continuous[2] grab~~[3]~~	dissolved hydrogen, dissolved oxygen, conductivity chloride, fluoride, sulfate lithium
CVCS sample point downstream of purification equipment	CVCS	liquid	grab	
CVCS injection line to RCS	CVCS	liquid	grab	

Notes:

1. Specific analyses, limits, and monitoring frequencies will be specified in plant procedures.
2. Semi-continuous (i.e., intermittent) analyses are performed by the applicable ion chromatography analysis unit provided in the hot lab.
3. ~~Normal and post-accident sampling.~~

RAI 09.03.02-3S1

Table 9.3.2-2: Containment Sampling System Normal ~~and Post-Accident~~ Sample Point

Sample Point	System	Process Fluid Type	Sampling Methods	Analysis(1)
Containment evacuation system (CES) gas discharge (downstream of the CES condenser outlet)	CES	gas	continuous	hydrogen and oxygen

Note:
1. Specific analyses, limits, and monitoring frequencies will be specified in plant procedures.

9.3.4 Chemical and Volume Control System

A chemical and volume control system (CVCS) is provided for each NuScale Power Module (NPM). The system purifies reactor coolant, manages chemistry of the coolant (including boron concentration), provides reactor coolant inventory makeup and letdown, and supplies spray flow to the pressurizer to reduce the reactor coolant system (RCS) pressure. A CVCS line from the reactor pressure vessel (RPV), separate from the primary CVCS circulation flow path, is provided for the removal of non-condensable gases which collect in the pressurizer vapor space. This line also permits supplying nitrogen to the RPV when utilizing a nitrogen bubble in the pressurizer vapor space during module startup.

The CVCS is used in combination with the module heatup system (MHS) during startup to raise reactor coolant temperature and to generate natural circulation flow in the RCS before nuclear heat addition. The MHS may also be used during shutdown to maintain RCS flow if decay heat is insufficient. Two MHS subsystems are provided for the plant with each subsystem serving up to six NPMs, one at a time. Figure 9.3.4-1 provides a simplified diagram of the CVCS during normal operation.

The boron addition system (BAS) is a shared system for up to 12 NPMs. The system prepares, stores, and transfers borated water for use by the CVCS or by the spent fuel pool cooling system (SFPCS) for adding boron to the spent fuel pool as needed. Figure 9.3.4-2 provides a system flow diagram of the BAS.

9.3.4.1 Design Bases

This section identifies the required or credited functions, the regulatory requirements that govern the performance of those functions, and the controlling parameters and associated values that ensure that the functions are fulfilled for the CVCS, MHS and BAS. Together, this information represents the design bases, defined in 10 CFR 50.2, as required by 10 CFR 52.47(a) and (a)(3)(ii).

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As described in Section 9.3.4.3, the CVCS, MHS, and BAS are not safety-related systems. However the CVCS is equipped with two automatic, safety-related, fail-closed, demineralized water isolation valves to ensure CVCS operation does not inadvertently cause a dilution of the reactor coolant system (RCS) boron concentration. Other important functions of the CVCS include providing the means for controlling RCS chemistry, monitoring for RCS leakage, and providing flow paths to the process sampling system (PSS) for normal~~ and post-accident~~ sampling.

Consistent with General Design Criteria (GDC) 1, CVCS, MHS and BAS structures, systems, and components (SSC) are designed, fabricated, erected, and tested to appropriate quality standards such that their failure does not impact the function of other safety-related or risk significant systems. The safety-related CVCS demineralized water isolation valves and associated piping are Quality Group C per Regulatory Guide (RG) 1.26. The CVCS piping and components outboard of the containment isolation valves (CIVs) up to the next valve that is normally closed or capable of automatic closure are also Quality Group C. Other SSC are Quality Group D. The classifications of CVCS, MHS, and BAS components are discussed further in Section 3.2.

RCS piping. Shielding for the CVCS ion exchanger vessels and reactor coolant filters is provided by concrete cubicles. Primary coolant piping in CVCS equipment rooms is shielded to minimize surveillance and maintenance dose rates. The CVCS has several features to reduce radiation exposure to ALARA levels.

- Steel alloys with low cobalt content are specified for materials of construction for components containing reactor coolant to minimize the generation of Cobalt-60.
- CVCS equipment drains and pressure relief valves are routed to the RWDS.
- Area radiation monitors are provided for rooms containing CVCS equipment which are radiological sources (NRHX, RHX, recirculation pumps, ion exchangers, filters and resin traps).
- Sluicing of ion exchanger resin is done remotely (outside of the cubicles), to remove resins and flush the vessels.
- Control panels and valve stations for CVCS equipment are provided with permanent shielding to limit worker exposure and meet site ALARA goals.
- Manual valves on pipes filled with reactor coolant have valve operators extending outside of shielded barriers to minimize dose for manual actions.
- Components that are not radiological sources (hydrogen bottles, makeup pumps, and chemical addition tanks) which require periodic access are separated from radiological sources.

The CVCS is designed to maintain occupational radiation exposure to ALARA levels as discussed in Section 12.1 and Section 12.3.

The BAS does not normally contain radioactive material and check valves are provided to minimize the potential for cross contamination from connected systems. BAS discharge paths are to systems that are compatible with receiving contaminated water. The BAS is supplied by the DWS via a direct connection to the BAS batch tank and an indirect connection to the BAS storage tank that can be used to supply demineralized water for recirculation and flushing through any of the tanks and piping, and then discharged to the LWRS.

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With respect to 10 CFR 50.34(f)(2)(xxvi) (Item III.D.1.1 in NUREG-0737), the CVCS is designed to be as leak free as practical. The system is in continuous use during normal operation and is provided with leakage detection instrumentation. During accident conditions, the CVCS is isolated from the RCS by the CIVs and is not needed to circulate primary coolant outside of containment. In addition, there are no safety systems that circulate reactor coolant outside of containment. ~~However, in order to support post-accident sampling by the PSS as described in Section 9.3.2, the CVCS is capable of being unisolated from the RCS, when conditions permit, to establish the sample flow path from the CVCS discharge piping upstream of the RHX.~~

RAI 19-36, RAI 19-36S1

Consistent with SECY-93-087, Item I.F, the design of the CVCS reduces the possibility of a LOCA outside containment. As shown in FSAR Figure 6.2-4, the CVCS is the only system with connections to the RCS and piping that runs outside containment.

9.3.6 Containment Evacuation System and Containment Flooding and Drain System

The containment evacuation system (CES) and the containment flooding and drain system (CFDS) are used to transfer liquids and gases between the containment vessel (CNV) free volume and other plant systems.

The functions of the CES include:

- establishing and maintaining a vacuum in the CNV during NPM operation by removing non-condensable gases from the CNV, which reduces convective heat transfer from the reactor vessel to the reactor pool.
- measuring CNV pressure during NPM operation via pressure sensors on the CES vacuum pump suction line to monitor leakage into the CNV from all sources.
- monitoring radioactivity levels in the non-condensable gas removed from the CNV and, depending on the radioactivity level in the gas, either filtering and discharging the gas through the reactor building ventilation system (RBVS) plant exhaust stack or transferring the gas to the gaseous radioactive waste system (GRWS).

RAI 09.03.02-3S1

- support the ~~plant~~process sampling system (PSS), as described in Section 9.3.2 providing a suction and return path for continuous monitoring of hydrogen and oxygen concentration in the containment atmosphere during normal operations.
- support post-accident monitoring, providing a suction path for post-accident monitoring of hydrogen and oxygen concentration in the containment atmosphere as described in Section 9.3.2. The return path for monitoring of hydrogen and oxygen concentration in the containment atmosphere is the CFDS.

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- ~~support post-accident sampling, providing grab sample capability of the containment atmosphere using the CES particulate, iodine, and gaseous radiation monitor, as described in Section 9.3.2.~~
- vaporizing and removing water from the CNV during NPM startup following refueling, condensing the water vapor, and discharging the water to the radioactive waste drain system (RWDS).
- removing water vapor from the CNV during NPM operation and providing a method to condense, collect, and sample the water removed from the CNV prior to the water being discharged to the RWDS.
- quantifying the amount of water vapor removed from the CNV during NPM operation to monitor leakage into the CNV from all sources and to allow leak-before-break (LBB) methodology to be applied to leakage from feedwater and main steam piping in the CNV.
- removing non-condensable gases from the reactor coolant system (RCS), prior to CFDS pump-down of the CNV.

RAI 19-23S2

- providing a path for pressurization of the CNV in support of refueling and maintenance operations.

The CFDS is not required to operate during or after any design basis accident. As a defense-in-depth measure, the CFDS may be used to inject water from the reactor pool into a NPM containment in a beyond design basis event. This function is not safety-related.

When not in operation, the CFDS pump suction lines are normally maintained filled and vented to facilitate readiness for emergency containment flooding operations. Before initiating emergency containment flooding, CNV pressure must be low enough to allow opening the CFDS containment isolation valve and water injection into the CNV using a CFDS pump. Emergency flooding is initiated from the MCR using the MCS to open the containment isolation valves. The CFDS is pre-aligned for containment flooding operation, allowing flooding to begin by starting a CFDS pump using the PCS. When the water in containment reaches the desired level, operators shut off the operating CFDS pump and close the respective CFDS containment isolation valves.

RAI 09.03.02-3S1

Post-accident Containment Atmosphere Monitoring ~~and Sampling~~

RAI 09.03.02-3S1

The CES and CFDS support post-accident containment atmosphere monitoring. The CES provides a suction path for post-accident monitoring of hydrogen and oxygen concentration in the containment atmosphere by the PSS as described in Section 9.3.2. The return path for the PSS hydrogen and oxygen monitor is supported by the CFDS, returning the discharge of the PSS sample pump to the CNV.

RAI 09.03.02-3S1

~~The CES supports post-accident sampling. The CES provides grab sample capability of the containment atmosphere using the CES particulate, iodine, and gaseous radiation monitor, as described in Section 9.3.2 and 11.5.~~

9.3.6.3 Safety Evaluation

The CES and the CFDS are nonsafety-related systems that have no safety-related SSC required to operate during or after any design basis accident. Additionally, the CES and CFDS are not required for an NPM to reach safe shutdown mode and temperature as defined in the technical specifications regardless of the status of the other NPMs.

General Design Criterion 2 was considered in the design of the CES and CFDS.

The RXB provides protection from external natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami and seiches. The SSC that could adversely affect Seismic Category I components are appropriately supported to prevent damage due to a safe shutdown earthquake. The classification of the piping and components for the CES and CFDS conforms to the requirements of Regulatory Guide (RG) 1.29, Revision 5, with the exception as noted in Section 3.2.1 for classifying SSC as Seismic Category II. In both the CES and CFDS, the piping from the NPM disconnect flange to the pipe gallery wall is Seismic Category II because the piping does not perform a Seismic Category I

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

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

RAI 09.03.02-3S1

~~Reactor coolant system and containment sampling capability during post-accident conditions is addressed in Section 9.3.2.~~

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

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

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

Occupational doses are estimated for a single NPM refueling outage and for an entire year, assuming six NPM refueling outages. Table 12.4-7 provides dose estimates for the various refueling activities.

12.4.1.7 Overall Plant Doses

The estimated annual personnel doses associated with the activities discussed above are summarized in Table 12.4-1.

Occupational personnel dose estimates are calculated assuming a 12-NPM site and 24-month fuel cycle for NPM operation, which equates to six refueling outages per year.

12.4.1.8 Post-Accident Actions

There are no credited post-accident operator actions outside of the main control room for design basis events, as described in Chapter 15. The operator dose assessments for the main control room and the technical support center are provided in Section 15.0.3.

RAI 09.03.02-2S1, RAI 09.03.02-3S1, RAI 12.03-1, RAI 12.03-31, RAI 12.03-32, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35, RAI 12.03-36, RAI 12.03-37, RAI 12.03-39, RAI 12.03-40

~~As described in Section 9.3.2, the process sampling system may be used as part of a contingency action to obtain post-accident samples, which would potentially expose the operator to post-accident radiation sources. The primary means to detect and monitor fuel damage uses core exit temperature indication and radiation monitors located under the NPM bioshield. The post-accident primary coolant sample is collected via the normal CVCS sample line flow path to the primary system sample panel located in the CVCS gallery. To perform primary liquid sampling, operators would enter the RXB at the 100' elevation, descend to the 50' elevation using a stairwell, and traverse to the sample panels in the CVCS gallery and the counting room and hot lab on elevation 50' of the RXB. If the background dose rate in the counting room is too high, operators would use the counting room in the Annex Building. These areas are depicted in Figure 12.3-4a through Figure 12.3-4d. These post-accident radiation zone maps represent a composite of maximum dose rates developed by using the highest dose rate in a particular Reactor Building area resulting from design basis accidents occurring on the module with the highest calculated dose rate. Therefore, the radiation zones depicted in Figure 12.3-4a through Figure 12.3-4d will not occur simultaneously. For post-accident sampling operator activities, the limiting design basis accident is a break of a small line carrying primary coolant, with a coincident iodine spike. This also assumes that the sampling system has not been rendered inoperable due to the accident. For example, a small line break on a CVCS line upstream of the sample line tap causes the sampling system to be non-functional. Consistent with 10 CFR 50.34(f)(2)(vii), post-accident radiological conditions were evaluated and determined that primary coolant sampling activities expose operators to dose rates up to 70 mrem/hr at the sample panel, with much of the collection activities resulting in dose rates less than 13 mrem/hr. Post-accident doses in the counting room and hot lab were determined to be less than 2.5 mrem/hr. To perform containment gas sampling, operators would perform the necessary functions from the main control room.~~

RAI 12.03-1, RAI 12.03-31, RAI 12.03-32, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35, RAI 12.03-36, RAI 12.03-37, RAI 12.03-39, RAI 12.03-40

~~The operator's exposure to airborne activity is considered as part of the dose evaluation, but is not expected to result in significant doses. This is due to the barrier created by the RXB walls around the reactor pool, which include airtight seals for doors and other penetrations, between the reactor pool area and other areas in the RXB. This barrier minimizes the migration of airborne contamination from the airspace above the pool to other areas of the RXB. The RXB HVAC system uses smoke dampers to minimize the leakage between the pool area and other portions of the building. In addition, the generation of airborne contamination is minimized in the CVCS gallery area through the control of the temperature and pressure of the primary coolant during sampling. Also, if radiological conditions warrant, respiratory protection equipment can be provided to post-accident sampling personnel.~~

RAI 09.03.02-2S1, RAI 09.03.02-3S1, RAI 12.03-1, RAI 12.03-31, RAI 12.03-32, RAI 12.03-33, RAI 12.03-34, RAI 12.03-35, RAI 12.03-36, RAI 12.03-37, RAI 12.03-39, RAI 12.03-40

~~The decision to perform post-accident sampling will be determined and initiated by the site staff considering the expected radiological conditions and radiological dose to operating personnel when making the decision to access these areas to perform post-accident sampling. A summary of the assumed parameters for the post-accident sampling operator dose evaluation are provided in Table 12.4-8. Post-accident sampling performed consistent with approved procedures prevents radiation exposures to individuals from exceeding 5 rem to the whole body or 50 rem to the extremities, consistent with 10 CFR 50.34(f)(2)(viii).~~

12.4.1.9 Construction Activities

For the construction of an additional NuScale Power Plant adjacent to an existing NuScale Power Plant, the estimated annual radiation exposure to a construction worker is estimated based upon a construction staffing plan over the estimated construction period. It is estimated that the annual dose for a construction worker is 1.64 mrem/year.

COL Item 12.4-1: A COL applicant that references the NuScale Power Plant design certification will estimate doses to construction personnel from a co-located existing operating nuclear power plant that is not a NuScale Power Plant.

RAI 02.03.01-2, RAI 02.03.05-1

12.4.2 Radiation Exposure at the Restricted Area Boundary

RAI 02.03.01-2, RAI 02.03.05-1

The direct radiation to the restricted area boundary from on-site sources, such as buildings, is negligible.

RAI 09.03.02-3S1

Table 14.2-53: Process Sampling System Test # 53

Preoperational test is required to be performed for each NPM.		
The PSS is described in Section 9.3.2 and the functions verified by this test are:		
System Function	**System Function Categorization**	**Function Verified by Test #**
1. The PSS supports the RCS during normal operations by providing sampling and analysis of reactor coolant discharge (letdown) liquid.	nonsafety-related	Test #53-1
2. The PSS supports the CVCS by providing sampling of reactor coolant at process points in the CVCS.	nonsafety-related	Test #53-1
~~3. The PSS supports the RCS during accident conditions by providing post-accident grab sample of the reactor coolant.~~	~~nonsafety-related~~	~~Test #53-1~~
~~4~~3. The PSS supports the CNTS during normal operations by providing sampling of containment gas and analysis of hydrogen and oxygen concentration in containment.	nonsafety-related	Test #53-2
~~5~~4. The PSS supports the condensate and FWS by providing sampling and analysis of condensate and feedwater.	nonsafety-related	Test #53-3
~~6~~5. The PSS supports the MSS by providing sampling and analysis of main steam.	nonsafety-related	Test #53-3
~~7~~6. The PSS system supports the ABS by providing sampling and analysis of the auxiliary boiler steam and feedwater.	nonsafety-related	Test #53-3
~~8~~7. PSS supports the CNTS during accident condition by providing~~-sampling of~~ containment ~~gas~~atmosphere monitoring and analysis of hydrogen and oxygen concentration~~ in containment~~ to respond to emergencies.	nonsafety-related	Test #53-2
Prerequisites Verify an instrument calibration has been completed, with approved records and within all calibration due dates, for all instruments required to perform this test.		