September 16, 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
3. NuScale Power, LLC Supplemental Response to NRC "Request for Additional Information No. 205 (eRAI No. 9044)," dated July 31, 2019

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) supplemental response to the referenced NRC Request for Additional Information (RAI). This supplemental response supersedes the July 31, 2019 response (reference 3).

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

- 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 Elizabeth English at 541-452-7333 or at eenglish@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Getachew Tesfaye, NRC, OWFN-8H12
Michael Dudek, NRC, OWFN-8H12

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

NUSCALE
Power for all humankind

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 response to RAI 9044 (Question 09.03.02-3) originally provided in letter RAIO-1017-56948, dated October 31, 2017, and supplemented in letter RAIO-0719-66421, dated July 31, 2019 (ADAMS Accession Nos. ML17304B483 and ML19212A689 respectively). The supplement provided in letter RAIO-0719-66421, dated July 31, 2019, was 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) were discussed. The Post-Accident Sampling exemption was transmitted by letter LO-0119-64386, dated January 31, 2019 (ADAMS Accession No. ML19031C975).

This supplemental response supersedes the July 31, 2019 response and is provided as a result of discussions with the NRC during a phone call on August 28, 2019. NuScale proposed to delete a sentence from DCA section 9.3.2.2.3 that specifies the design pressure of the containment evacuation system (CES) and containment flood and drain system (CFDS) piping downstream of the containment isolation valves (CIVs) and passive containment isolation barrier flange connections used in post-accident combustible gas monitoring (i.e., post-accident hydrogen and oxygen monitoring of containment gas). NuScale also proposed to make a tie in section 9.3.2.2.3 to DCA table 3.2-1. NuScale agreed to leave the subject sentence in DCA section 9.3.2.2.3 and to make the tie in section 9.3.2.2.3 to DCA table 3.2-1. NuScale also informed the NRC staff that the design pressure of the CFDS piping and components downstream of the CIVs and passive containment isolation barrier flange connections used in post-accident combustible gas monitoring needed to be changed from 150 psig to 250 psig. This supplemental response reflects that the design pressure and temperature of the CFDS piping has changed from 150 psig and 300 degrees F to 250 psig and 550 degrees F.

The supplemental response provided in letter RAIO-0719-66421, dated July 31, 2019, included FSAR changes made in conformance with the PAS exemption and it clarified the actions required for post-accident hydrogen and oxygen monitoring. The changes included 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 Sections 9.3.2.1, 9.3.2.3, and 12.4.1.8.

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 CES and CFDS 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 is removed automatically 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 and CFDS piping and components downstream of the CIVs and passive containment isolation barrier flange connection that support post-accident combustible gas monitoring, are 250 psig and 550 degrees F.

In severe accidents with core damage, the post-accident combustible gas monitoring may require operator action outside the MCR to override the CIS and open the CES CIVs from the CIV hydraulic control skids. 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 monitoring loop where the containment gas can be routed from the CES to the 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 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 return line connection to the CFDS before the respective containment isolation valve.

3. The design pressure and temperature of the CES and CFDS piping and components that support the post-accident combustible gas monitoring are 250 psig and 550 degrees F. The post-accident containment gas monitoring loop will not be put into operation if the containment conditions exceed CES or CFDS design limits.

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 pressures and temperatures 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 severe accident 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. 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 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 and transmitted previously as described in the response above and new changes to FSAR Sections 9.3.2.1, 9.3.2.2.3, 9.3.2.3, 12.4.1.8 and Table 1.9-5 are shown in the markup provided in this response.

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 09.03.02-3S2, 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

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)(iii)	Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts (I.D.1)	Conforms	None.	18.7
50.34(f)(2)(iv)	Provide a plant safety parameter display console (I.D.2)	Conforms	The NuScale safety display and indication system is integrated into the control room human-system interface design rather than having a separate console.	7.1 7.2.13 18.7.2
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	The NuScale design does not contain vital areas, as defined by NUREG-0737, Item II.B.2, other than the areas for initiating combustible gas monitoring, main control room and technical support center. Protection of necessary equipment from radiation is reasonably assured through demonstrating equipment survivability.	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	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.	9.3.2 11.5 12.4

Consistent with GDC 5, SSC shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Consistent with GDC 13, instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary (RCPB), and the containment and its associated systems. Sampling of the reactor coolant and other process systems enables the PSS to provide information on variables that can affect the fission process, the integrity of the reactor core, and the RCPB during normal modes of operation.

Consistent with GDC 14, the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The PSS supports ensuring the integrity of the RCPB by sampling reactor coolant to ensure that water chemistry parameters are within predetermined values to preclude affecting the RCPB.

Consistent with GDC 26, the PSS is used to verify the boron concentration necessary for the control of core reactivity changes by sampling reactor coolant and the contents of the boric acid storage tanks of the boron addition system (BAS).

General Design Criteria 41 is not applicable to the PSS. The containment design does not use a containment spray system or a containment atmosphere cleanup system to mitigate the consequences of postulated accidents. Therefore, sampling of the chemical additive tank is not applicable to the design.

Consistent with GDC 60, the nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. The PSS supports the capability to control the release of radioactive materials to the environment.

Consistent with GDC 63, appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and to initiate appropriate safety actions. The PSS supports detecting conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems.

RAI 09.03.02-3S2

Consistent with GDC 64, 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. The PSS supports the capability to monitor the post-accident containment atmosphere, and the capability to sample and analyze for radioactivity that may be released during normal operations, and anticipated operational occurrences~~spaces containing components for recirculation of loss-of-coolant accident fluids, and effluent discharge paths for radioactivity~~.

Consistent with 10 CFR 50.34(f)(2)(xvii)(c) and 10 CFR 50.44(c)(4) the PSS design provides equipment capable of continuous monitoring of hydrogen and oxygen concentration in the containment atmosphere. The equipment used for monitoring hydrogen is reliable and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design basis accident for accident management and provides indication in the MCR.

RAI 01.05-40

Consistent with 10 CFR 50.34(f)(2)(xxvi), the PSS design contains provisions for leakage detection, and to control leakage to levels as low as practical~~control and detection~~, to minimize exposures to workers and the public and to maintain control and use of the system during an accident (Item III.D.1.1 in NUREG-0737).

RAI 01.05-40

COL Item 9.3-1: A COL applicant that references the NuScale Power Plant design certification will submit a leakage control program for systems outside containment that contain (or might contain) accident source term radioactive materials following an accident (including systems and components used in post-accident hydrogen and oxygen monitoring of the containment atmosphere). The leakage control program will include~~, including~~ an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems to as low as practical.

Consistent with 10 CFR 20.1101(b), the PSS design supports keeping radiation exposures as low as reasonably achievable (ALARA). Consistent with 10 CFR 20.1406, the PSS design supports minimization of contamination of the facility and the environment, minimizing generation of radioactive waste, and facilitating eventual plant decommissioning.

9.3.2.2 System Description

9.3.2.2.1 General Description

The PSS is designed to collect representative liquid and gaseous samples from various plant systems using the following sampling system features:

- the primary sampling system
- the containment sampling system (CSS)
- the secondary sampling system (SSS)
- local grab sample provisions

The PSS is operable during normal operations, including at power, shutdown, and startup. The system has the ability to obtain samples at the normal system operating temperatures and pressures from various locations. These samples can 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. 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.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-3S2, 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 CNV, post-accident hydrogen and oxygen monitoring with PSS requires opening the CES and CFDS CIVs. If post-accident hydrogen and oxygen monitoring must be performed while containment isolation conditions exist, overriding the containment isolation signal (CIS) is required via operator action outside the MCR.

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 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.

RAI 09.03.02-3S1

Containment Gas Post-Accident Monitoring

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-3S2, 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 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. Initiation of hydrogen and oxygen monitoring is consistent with the survivability of the associated equipment, as described in Section 19.2.3.

RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-3S2, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

Post-accident hydrogen and oxygen monitoring of containment gas can be initiated when plant conditions are amenable to opening the CES and CFDS CIVs, and do not exceed design limitations of the associated CES and CFDS piping and components. The design pressure and temperature of the CES and CFDS piping and components that are part of the combustible gas (i.e., hydrogen and oxygen) monitoring path are 250 psig and 550 degrees F. The component pressure boundaries of the CES, PSS and CFDS that are part of the combustible gas monitoring path are designed to withstand combustion events, as described in FSAR Table 3.2-1.~~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 09.03.02-3S2, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

~~The plant responses in accident conditions show the containment pressure is reduced to 150 psia in approximately two hours after event initiation. 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 hydrogen and oxygen monitoring, the CES and CFDS CIVs are opened to establish the 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 that is equipped with online hydrogen and oxygen monitoring equipment. The gas is then returned to the CNV via the containment sampling system effluent discharge line connected to the CFDS return line to CNV as shown on Figure 9.3.6-2. Returning the 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 PSS design limits the potential reactor coolant loss from the rupture of a sample line. A failure of a sample line would result in a loss of flow to either a continuous analyzer or a grab sample panel which would be observed by plant personnel. In addition, a break in a sample line would result in activity release that might actuate the fixed area radiation monitors located in the containment sampling system equipment area and the primary sampling system equipment area, as described in Table 12.3-10. The three PSS sample points to the CVCS are each provided with two fail-closed isolation valves. These isolation valves are downstream of the environmentally qualified CIVs associated with the CVCS discharge line and are also downstream of the CVCS module isolation valves as shown on Figure 9.3.4-1. The PSS line sizes range from 3/4" to 3/8" which further restricts the break flow of a sample line outside containment.

The PSS design satisfies GDC 63 by allowing the detection of conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems. The PSS includes sampling capability of the spent fuel pool and reactor pool water via local sample points in the pool cooling and cleanup system. The PSS also includes sampling capability via local sample points for the radioactive waste management systems. This enables analyses to be performed to detect conditions in the fuel storage and radioactive waste systems that could result in excessive radiation levels and excessive personnel exposure.

RAI 09.03.02-3S1, RAI 09.03.02-3S2

The PSS design satisfies GDC 64 as it provides the capability to monitor the post-accident containment atmosphere, and to sample and analyze for radioactivity that may be released during normal operations~~,~~ and anticipated operational occurrences~~, and postulated 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

The PSS design satisfies 10 CFR 50.34(f)(2)(xvii)(c) by providing capability to monitor hydrogen and oxygen concentration in containment atmosphere during operation and 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.

RAI 01.05-40, 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 levels as low as practical 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.

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

RAI 09.03.02-3S2

There are no vital areas, as defined by NUREG-0737, Item II.B.2, other than the areas for initiating combustible gas monitoring (described in Section 9.3.2.2.3), the main control room, and the technical support center, which are in compliance with 10 CFR 50.34(f)(2)(vii). 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.

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.