RAIO-0919-66890

September 05, 2019

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 9690 (eRAI No. 9690) on the NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 3

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9690 (eRAI No. 9690)," dated June 27, 2019
2. NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 3, dated April 2019
3. NuScale Power, LLC Response to NRC "Request for Additional Information No. 9690 (eRAI No.9690)," dated July 31, 2019

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

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 9690:

- 01.05-40

The responses to RAI Questions 01.05-39, 01.05-41 and 01.05-42 were previously provided in Reference 3. This completes all responses to eRAI 9690.

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
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9690

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9690

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

eRAI No.: 9690
Date of RAI Issue: 06/27/2019

NRC Question No.: 01.05-40

Regulatory Basis:

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are completed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses; and 10 CFR Part 50, Appendix A, GDC 19, 10 CFR 50.34(f)(2)(vii) and 10 CFR 50.34(f)(2)(xxviii) for control room radiological habitability. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) Section 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

Background:

On January 31, 2019, NuScale submitted a request for exemption from the 10 CFR 50.34(f)(2)(viii) requirements related to post-accident sampling. In the technical basis for the exemption request, the applicant describes that the capability to continuously monitor hydrogen and oxygen concentration in the containment atmosphere to meet the requirements of 10 CFR 50.44 is accomplished using the process sampling system (PSS) in-line monitors during accident conditions, including beyond-design-basis events with core damage. The path that the highly radioactive post-accident containment atmosphere would take to achieve continuous monitoring of hydrogen and oxygen concentration is outside of the containment, and the PSS is not related to safety. NuScale topical report TR- 0915-17565, Revision 3, "Accident Source Term Methodology," was submitted on April 21, 2019. This topical report describes the accident source term and radiological consequence analysis methodology for the core damage event (CDE), which is used to show compliance with the regulatory requirements described above.

The description of the CDE radiological consequence analysis in Section 4.2.5 of the topical report does not include discussion of the potential releases from the post-accident combustible gas monitoring pathway outside containment.

RG 1.183, Appendix A provides guidance on modeling of potential pathways to the environment for core damage accidents in the radiological consequence analyses which show compliance with the regulations stated above. Guidance on the modeling of ESF system leakage and containment purging in RG 1.183, although not directly describing the NuScale design post-accident combustible gas monitoring capability, provides indication that potential release pathways to the environment for the accident should be included in the analysis.

Issue:

Additional information is needed to describe the modeling of potential releases to the environment through the systems used in post-accident monitoring of hydrogen and oxygen concentration in the containment atmosphere to demonstrate compliance with the regulatory requirements described above.

Request:

Please describe the methods, models, and assumptions used for calculating the contribution to the dose from a potential release to the environment through leakage from the systems used in post-accident monitoring of the hydrogen and oxygen concentration in the containment atmosphere for the CDE. Additionally, please update the topical report to provide this description and make any necessary conforming changes to the FSAR.

NuScale Response:

The hydrogen and oxygen monitoring capability is provided in the NuScale design as part of the process sampling system (PSS), via the containment evacuation system (CES) and the core flood and drain system (CFDS). Unlike the engineered safety feature (ESF) systems, for which leakage is addressed by Regulatory Guide 1.183, the hydrogen and oxygen monitoring capability is not relied upon for mitigating any design basis event and is not credited in any design basis event analysis. Combustible gas monitoring is provided pursuant to 10 CFR 50.44(c)(4), which requires combustible gas monitoring capability only for beyond-design basis events (reference 68 FR 54126).

Regulatory Guide 1.183, Appendix A, Section 7 states, in part, “If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated.” As with severe accident containment purge capability, the NuScale hydrogen and oxygen monitoring capability is only provided for the purpose of severe accident management, and is not credited in any design basis analysis. Therefore, as with the radiological consequences of a containment purge for combustible gas control (a large release), evaluation of the radiological consequences of a potential combustible gas monitoring leak is unnecessary (a small release by comparison).

Notwithstanding the Regulatory Guide 1.183 position on severe accident management doses, NuScale recognizes that the Core Damage Event (CDE) analyzed in FSAR section 15.10 is a beyond-design-basis accident for which NuScale has elected to analyze dose consequences in comparison to offsite and control room dose limits. NRC Staff have stated this analysis implies that all potential dose contributors, including those from the combustible gas monitoring function required by 10 CFR 50.44 for severe accidents, must be accounted for. However, NuScale notes that the CDE is analogous to the “design basis loss of coolant accident” (DBLOCA) radiological consequence analysis prescribed by SRP section 15.6.5, which also assumes a significant core damage event, and in turn would seem to necessitate the use of combustible gas monitoring pursuant to 10 CFR 50.44. Also, all plants are required to have a capability for post-accident sampling following an accident. However, SRP 15.6.5 and Regulatory Guides 1.183 and 1.195 address neither combustible gas monitoring leakage nor post-accident sampling releases and leakage.

Three Mile Island Action Plan Item III.D.1.1.1, applicable to NuScale by way of 10 CFR 50.34(f)(2)(xxvi), addresses potential leakage from systems such as the combustible gas monitoring equipment that may contain accident source term following a core damage accident. Licensees are required to have a leakage control program including actions to minimize leakage from such systems, where the “goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.” NUREG-0737 indicates that implementation of the leakage control program provides reasonable assurance that doses from such systems is acceptable, without specifically calculating the dose contribution from hypothetical leakage. For example, Item II.B.2 provides that, in calculating the dose to operators in conducting post-accident sampling, “radiation from leakage of systems located outside of containment need not be considered” because “leakage measurement and reduction is treated under Item III.D.l.l.” Similarly, Item III.D.3.4 prescribes that the control room dose analysis should be for containment leakage and ESF system leakage outside containment. In sum, NUREG-0737 identifies potential leakage

outside containment as an issue to address by reducing leakage under Item III.D.1.1, but omits this known, potential leakage source as a source term to be evaluated for doses to operators under Items II.B.2 and III.D.3.4.

Therefore, NuScale's position is that implementation of a leakage control program pursuant to 10 CFR 50.34(f)(2)(xxvi) and NUREG-0737 item III.D.1.1 demonstrates compliance with the regulatory requirements Staff identified, without modeling the dose consequences of hypothetical leakage from combustible gas monitoring. NUREG-0737 item III.D.1.1 requires that a licensee "reduce leakage to as-low-as-practical levels."

Precedent supports NuScale's position. For example, the ESBWR design includes a Containment Monitoring System (CMS) that is partially routed outside of containment (ESBWR DCD, Chapter 7, Figure 7.5-1, Rev. 10; ML14100A523) and is used to monitor the hydrogen and oxygen gas concentrations in the drywell and wetwell during post-accident conditions, however there is no leakage assumed from the CMS in the accident dose evaluations (ESBWR DCD, Chapter 15, Rev. 10; ML14100A547), and the leakage control program specifies that leakage from that and other systems containing accident source term outside containment be "as low as practicable" (ESBWR DCD, Chapter 16, Generic Technical Specification 5.5.2). The APR1400 also includes a hydrogen monitor outside containment as part of its CMS (APR1400 DCD, Section 6.2.5.2.2), which is likewise included in scope of the leakage control program without quantifying an acceptance criterion (APR1400 DCD, Chapter 16, Generic Technical Specification 5.5.2). The ABWR includes 13 systems outside containment that may contain source term following an accident, including post-accident sampling, process sampling, containment atmospheric monitoring, fission product monitoring (ABWR DCD, Section 1A.2.34, Rev. 4). Such systems are subject to the leakage control program but leakage from them does not appear to be calculated in accident dose evaluations. The ABWR SER concludes that a requirement for COL applicant's procedures to "reduce detected leakages to lowest practical levels" satisfied TMI item III.D.1.1 (NUREG-1503, Section 20.5.38).

In the NuScale design, the combustible gas monitoring loop is a gaseous process stream, and the system is used during normal operations. Substantial system leakage during normal operation would be evident because it would inhibit the ability to maintain a vacuum inside containment. Also, the combustible gas monitoring loop is included in the leakage control program pursuant to COL Item 9.3-1 in order to "minimize potential exposures to workers and the public, and provide reasonable assurance that excessive leakage will not prevent the system's use in an emergency" (10 CFR 50.34(f)(2)(xxvi)). Consistent with precedent and NUREG-0737 item III.D.1.1, COL Item 9.3-1 will be updated to explicitly identify "as low as practical" as the acceptance criterion for the leakage control program, and to identify as within

program scope the systems and components used in post-accident hydrogen and oxygen monitoring of the containment atmosphere. A COL licensee's procedures will implement the leakage reduction program consistent with this requirement.

In a recent public meeting, NRC Staff indicated that the ESBWR precedent is not applicable to the NuScale design because the ESBWR has CMS isolation valves inside containment and can isolate the CMS from the control room. NuScale does not believe these differences are relevant. Leakage up to and including that from the NuScale CES and CFDS containment isolation valves is included in the allowable and measured containment leakage. Isolation of the ESBWR CMS would appear to render combustible gas monitoring non-functional in contravention of 10 CFR 50.44, as NRC Staff earlier stated would be the case for NuScale, and the location of isolation is not relevant to offsite and control room dose (local operator dose is addressed in RAI 9682, Q12.03-66).

Therefore, NuScale has not performed a radiological consequence analysis of a potential leak from the beyond design basis post-accident hydrogen and oxygen monitoring process.

Impact on DCA:

FSAR Section 9.3.2 and Table 1.8-2 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 01-61, RAI 01.05-40, 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~~site 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~~of these potential hazards, 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

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 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.	9.3
COL Item 9.3-2:	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 used.	10.2

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 be in the form of continuously analyzed samples or grab samples. The PSS obtains samples that are representative of the process or vessel under evaluation. For sampling of process streams, sample points are located in a turbulent flow zone which minimizes particulate dropout and re-entrainment in sample piping. For

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

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.

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.

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