RAIO-0618-60376

June 11, 2018

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 Response to NRC Request for Additional Information No. 414 (eRAI No. 9430) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 414 (eRAI No. 9430)," dated April 11, 2018

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

The Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 9430:

- 13.05.02.01-7
- 13.05.02.01-8
- 13.05.02.01-9
- 13.05.02.01-10
- 13.05.02.01-11
- 13.05.02.01-12
- 13.05.02.01-13
- 13.05.02.01-14
- 13.05.02.01-15
- 13.05.02.01-16

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 414 (eRAI No. 9430). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

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

If you have any questions on this response, please contact Steven Mirsky at 240-833-3001 or at smirsky@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Prosanta Chowdhury NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9430, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9430, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0618-60377

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9430, proprietary

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
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Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9430, nonproprietary

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

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

eRAI No.: 9430
Date of RAI Issue: 04/11/2018

NRC Question No.: 13.05.02.01-7

REGULATORY BASIS REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47(a)(8) requires an applicant for a design certification to provide an FSAR (Final Safety Analysis Report) which includes the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), with certain exceptions. Section 10 CFR 50.34(f)(2)(ii) requires an applicant to "Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, ... “

TMI Action Plan Item I.C.1, a Post-TMI requirement approved by the Commission for implementation, requires the preparation of emergency procedure technical guidelines for development of the Emergency Operating Procedures (EOPs). Preparation of the technical guidelines is conducted in accordance with NUREG-0737, “Clarification of TMI Action Plan Requirements,” and NUREG-0737, Supplement 1, “Requirements for Emergency Response Capability,” which also specify submittal of the technical guidelines to the NRC for review and approval.

Meeting the requirements of TMI Action Plan Item I.C.1 as prescribed in NUREG-0737, Section I.C.1, and Supplement 1 to NUREG-0737, Section 7, is acceptance criteria in SRP 13.5.2.1, “Operating and Emergency Operating Procedures.” Design-specific Generic Technical Guidelines (GTGs), otherwise referred to as the Emergency Operating Guidelines (EOGs), will be used by COL applicants to develop their Plant-Specific Technical Guidelines (P-STGs), from which their EOPs will be developed, and are the responsibility of the DC applicant.

By letter dated November 30, 2017 (ADAMS Accession No. ML17334B822) NuScale submitted technical report TR-1117-57216, “NuScale Generic Technical Guidelines,” for docketing.

ISSUE

The Reactivity Safety Function flowchart in Section 5.2 of the NuScale GTGs depicts the logic and specifies the operator actions necessary to assess and maintain the Reactivity Safety

Function. Chapter 7, Section 7.1.1.2.2, “Post-Accident Monitoring,” “Type B Variables,” Page 7.1-8, states:

> “*Type B variables are those that provide primary information to the control room operators to assess the plant’s critical safety functions that have been defined for the NuScale power plant. These are accomplishing or maintaining the following three critical safety functions:*
>
> - *Reactivity control*
> - *Remove fuel assembly heat*
> - *Containment*
>
> *NuScale has selected these three critical safety functions based on the plant design.”*

Chapter 7, Section 7.1.1.2.2, “Post-Accident Monitoring,” “Type B Variables,” “Reactivity Control Safety Function Variables,” Page 7.1-8, states:

> “*The Type B variables that provide direct indication and are used to assess the process of accomplishing or maintaining reactivity control are neutron flux and* ***core inlet and exit temperature****.”*

Chapter 7, Table 7.1-7, “Summary of Type A, B, C, D, and E Variables,” Page 7.1-73, classifies Core Inlet and Exit Temperatures as Type B variables. Core Inlet and Exit Temperatures are not evaluated on the Reactivity Safety Function flowchart. Given that Core Inlet and Exit Temperatures are both identified by Chapter 7 as Type B variables *that provide direct indication and are used to assess the process of accomplishing or maintaining reactivity control,* NRC staff is questioning why these two variables are not utilized as decision variables on the Reactivity Safety Function flowchart to determine whether the “Reactivity Control” Critical Safety Function (CSF) can be met.

INFORMATION NEEDED

NRC staff requests that NuScale: (1) explain why, on the basis of Chapter 7 information provided above, Core Inlet and Exit Temperatures (Post-Accident Monitoring Type B variables) have been excluded from the Reactivity Safety Function flowchart as a means to assess and maintain the “Reactivity Control” CSF, and (2) make any necessary changes to technical report TR-1117-57216 and/or Chapter 7 of the DCD, to ensure the completeness and accuracy of the NuScale GTGs (flowcharts and basis) and Chapter 7 DCD Tier 2 information.

NuScale Response:

The primary purpose of the post-accident monitoring (PAM) instrumentation is to display unit

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variables that provide information required by the main control room operators during accident conditions. The PAM instrumentation variables and their type classification are based on their accident management function as identified in abnormal and emergency procedures and emergency procedure guidelines. Since the abnormal and emergency operating procedures and guidelines were not developed prior to design certification application submittal, the approach to identify PAM variables included:

- review of systems with radiation monitoring equipment and potential effluent release paths
- operator actions assumed in the probabilistic risk assessment (PRA)
- operator actions identified during human factors engineering (HFE) task analysis
- industry operating experience
- engineering judgment
- the identification process and its results reviewed by a multidisciplinary group

Reg Guide 1.97 Revision 4 Position #4 provides a regulatory position for the development of PAM variables: "[T]he scope of instruments that could potentially be selected for accident monitoring (based on selected criteria) should initially be as encompassing as possible. Then, in the process of selecting the actual list of variables to be monitored, licensees could screen out instruments associated with contingency actions that take place beyond the plant's licensing basis."

Core Inlet and Exit Temperatures were selected as PAM variables using the evaluation criteria listed above. These variables were selected prior to the creation of the Generic Technical Guidelines (GTGs), which form the basis for the Emergency Operating Procedures (EOPs), and were specifically selected in accordance with RG 1.97 by compiling a conservative list of variables that would then be revised as guidance and procedures were developed.

Until the GTGs are validated during the conduct of the integrated system validation (ISV) testing, Chapter 7 describes the current approved design configuration. Following ISV testing and the subsequent evaluation, changes to FSAR Chapter 7 and/or the NuScale GTGs may result.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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

eRAI No.: 9430
Date of RAI Issue: 04/11/2018

NRC Question No.: 13.05.02.01-8

REGULATORY BASIS REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47(a)(8) requires an applicant for a design certification to provide an FSAR (Final Safety Analysis Report) which includes the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), with certain exceptions. Section 10 CFR 50.34(f)(2)(ii) requires an applicant to "Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, ... “

TMI Action Plan Item I.C.1, a Post-TMI requirement approved by the Commission for implementation, requires the preparation of emergency procedure technical guidelines for development of the Emergency Operating Procedures (EOPs). Preparation of the technical guidelines is conducted in accordance with NUREG-0737, “Clarification of TMI Action Plan Requirements,” and NUREG-0737, Supplement 1, “Requirements for Emergency Response Capability,” which also specify submittal of the technical guidelines to the NRC for review and approval.

Meeting the requirements of TMI Action Plan Item I.C.1 as prescribed in NUREG-0737, Section I.C.1, and Supplement 1 to NUREG-0737, Section 7, is acceptance criteria in SRP 13.5.2.1, “Operating and Emergency Operating Procedures.” Design-specific Generic Technical Guidelines (GTGs), otherwise referred to as the Emergency Operating Guidelines (EOGs), will be used by COL applicants to develop their Plant-Specific Technical Guidelines (P-STGs), from which their EOPs will be developed, and are the responsibility of the DC applicant.

By letter dated November 30, 2017 (ADAMS Accession No. ML17334B822) NuScale submitted technical report TR-1117-57216, “NuScale Generic Technical Guidelines,” for docketing.

ISSUE

The Core Heat Removal (CHR) Safety Function flowchart in Section 5.3 of the NuScale GTGs depicts the logic and specifies the operator actions necessary to assess and maintain the CHR

Safety Function. Chapter 7, Section 7.1.1.2.2, “Post-Accident Monitoring,” “Type B Variables,” Page 7.1-8, states:

> “*Type B variables are those that provide primary information to the control room operators to assess the plant’s critical safety functions that have been defined for the NuScale power plant. These are accomplishing or maintaining the following three critical safety functions:*
>
> - *Reactivity control*
> - *Remove fuel assembly heat*
> - *Containment*
>
> *NuScale has selected these three critical safety functions based on the plant design.”*

Chapter 7, Section 7.1.1.2.2, “Post-Accident Monitoring,” “Type B Variables,” "Remove Fuel Assembly Heat Critical Safety Function Variables,” Pages 7.1-8/9, states:

> *“The Type B variables selected that provide direct indication and verification and used to assess the process of accomplishing or maintaining the combined remove fuel assembly heat and RCS integrity critical safety functions are core exit temperature, RPV riser water level, wide range RCS pressure, containment water level,* ***degrees of subcooling****, and wide range RCS hot temperature.”*

Chapter 7, Table 7.1-7, “Summary of Type A, B, C, D, and E Variables,” Page 7.1-73, classifies Degrees of Subcooling as a Type B variable. Degrees of Subcooling is not evaluated on the Core Heat Removal Safety Function flowchart. Given that Degrees of Subcooling is identified by Chapter 7 as a Type B variable *that provides direct indication and verification and is used to assess the process of accomplishing or maintaining the combined remove fuel assembly heat and RCS integrity critical safety functions,* NRC staff is questioning why this variable is not utilized as a decision variable on the CHR Safety Function flowchart to determine whether the “Remove Fuel Assembly Heat” Critical Safety Function (CSF) can be met.

INFORMATION NEEDED

NRC staff requests that NuScale: (1) explain why, on the basis of Chapter 7 information provided above, Degrees of Subcooling (a Post-Accident Monitoring Type B variable) has been excluded from the CHR Safety Function flowchart as a means to assess and maintain the “Remove Fuel Assembly Heat” CSF, and (2) make any necessary changes to technical report TR-1117-57216 and/or Chapter 7 of the DCD, to ensure the completeness and accuracy of the NuScale GTGs (flowcharts and basis) and Chapter 7 DCD Tier 2 information.

NuScale Response:

The primary purpose of the post-accident monitoring (PAM) instrumentation is to display unit variables that provide information required by the main control room operators during accident conditions. The PAM instrumentation variables and their type classification are based on their accident management function as identified in abnormal and emergency procedures and emergency procedure guidelines. Since the abnormal and emergency operating procedures and guidelines were not developed prior to design certification application submittal, the approach to identify PAM variables included:

- review of systems with radiation monitoring equipment and potential effluent release paths
- operator actions assumed in the probabilistic risk assessment (PRA)
- operator actions identified during human factors engineering (HFE) task analysis
- industry operating experience
- engineering judgment
- the identification process and its results reviewed by a multidisciplinary group

Reg Guide 1.97 Revision 4 Position #4 provides a regulatory position for the development of PAM variables. ".. the scope of instruments that could potentially be selected for accident monitoring (based on selected criteria) should initially be as encompassing as possible. Then, in the process of selecting the actual list of variables to be monitored, licensees could screen out instruments associated with contingency actions that take place beyond the plant's licensing basis."

Reactor coolant system (RCS) subcooling was selected as a PAM variable using the evaluation criteria listed above. This variable was selected prior to the creation of the Generic Technical Guidelines (GTGs), which form the basis for the Emergency Operating Procedures, and was specifically selected in accordance with RG 1.97 by compiling a conservative list of variables that would then be pared down as guidance and procedures were developed. Although RCS subcooling can be used to determine if adequate core heat removal is occurring, it was not deemed to be required once emergency procedure guidance was developed based on the NuScale design.

Until the GTGs are validated during the conduct of the integrated system validation (ISV) testing, Chapter 7 describes the current approved design configuration. Following ISV testing and the subsequent evaluation, changes to FSAR Chapter 7 and/or the NuScale GTGs may result.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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

eRAI No.: 9430
Date of RAI Issue: 04/11/2018

NRC Question No.: 13.05.02.01-9

REGULATORY BASIS REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47(a)(8) requires an applicant for a design certification to provide an FSAR (Final Safety Analysis Report) which includes the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), with certain exceptions. Section 10 CFR 50.34(f)(2)(ii) requires an applicant to "Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, ... "

TMI Action Plan Item I.C.1, a Post-TMI requirement approved by the Commission for implementation, requires the preparation of emergency procedure technical guidelines for development of the Emergency Operating Procedures (EOPs). Preparation of the technical guidelines is conducted in accordance with NUREG-0737, "Clarification of TMI Action Plan Requirements," and NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," which also specify submittal of the technical guidelines to the NRC for review and approval.

Meeting the requirements of TMI Action Plan Item I.C.1 as prescribed in NUREG-0737, Section I.C.1, and Supplement 1 to NUREG-0737, Section 7, is acceptance criteria in SRP 13.5.2.1, "Operating and Emergency Operating Procedures." Design-specific Generic Technical Guidelines (GTGs), otherwise referred to as the Emergency Operating Guidelines (EOGs), will be used by COL applicants to develop their Plant-Specific Technical Guidelines (P-STGs), from which their EOPs will be developed, and are the responsibility of the DC applicant.

By letter dated November 30, 2017 (ADAMS Accession No. ML17334B822) NuScale submitted technical report TR-1117-57216, "NuScale Generic Technical Guidelines," for docketing.

ISSUE

The Containment Integrity (CI) Safety Function flowchart in Section 5.1 of the NuScale GTGs depicts the logic and specifies the operator actions necessary to assess and maintain the CI

Safety Function. Chapter 7, Section 7.1.1.2.2, “Post-Accident Monitoring,” “Type B Variables,” Page 7.1-8, states:

> “*Type B variables are those that provide primary information to the control room operators to assess the plant’s critical safety functions that have been defined for the NuScale power plant. These are accomplishing or maintaining the following three critical safety functions:*
>
> - *Reactivity control*
> - *Remove fuel assembly heat*
> - *Containment*
>
> *NuScale has selected these three critical safety functions based on the plant design.”*

Chapter 7, Section 7.1.1.2.2, “Post-Accident Monitoring,” “Type B Variables,” "Maintain Containment Integrity Critical Safety Function Variables,” Page 7.1-9, states:

> *“Maintain Containment Integrity is both a critical safety function and a fission product barrier (Containment) which serves as the primary means to control radioactive effluent releases. The same variables that are used to provide direct indication and support the containment integrity critical safety function are: wide range containment pressure, containment isolation valve position,* ***containment water level****, and* ***inside bioshield area radiation monitor****.”*

Chapter 7, Table 7.1-7, “Summary of Type A, B, C, D, and E Variables,” Page 7.1-73, classifies Inside Bioshield Area Radiation Monitor and Containment Water Level as Type B variables. Inside Bioshield Area Radiation Monitor and Containment Water Level are not evaluated on the CI Safety Function flowchart and neither PAM variable is specified in Table 10- 2, “List of General Technical Guidelines Decision Variables,” on Page 80 of TR-1117-5726. Given that Inside Bioshield Area Radiation Monitor and Containment Water Level are both identified by Chapter 7 as Type B variables *that are used to provide direct indication and support the containment integrity critical safety function,* NRC staff is questioning why these two variables are not utilized as decision variables on the CI Safety Function flowchart to determine whether the “Containment” Critical Safety Function (CSF) can be met.

INFORMATION NEEDED

NRC staff requests that NuScale: (1) explain why, on the basis of Chapter 7 information provided above, Inside Bioshield Area Radiation Monitor and Containment Water Level (Post-Accident Monitoring Type B variables) have been excluded from the CI Safety Function flowchart as a means to assess and maintain the “Containment” CSF, and (2) make any necessary changes to technical report TR-1117-57216 and/or Chapter 7 of the DCD, to ensure the completeness and accuracy of the NuScale GTGs (flowcharts and basis) and Chapter 7 DCD Tier 2 information.

NuScale Response:

The primary purpose of the post-accident monitoring (PAM) instrumentation is to display unit variables that provide information required by the main control room operators during accident conditions. The PAM instrumentation variables and their type classification are based on their accident management function as identified in abnormal and emergency procedures and emergency procedure guidelines. Since the abnormal and emergency operating procedures and guidelines were not developed prior to design certification application submittal, the approach to identify PAM variables included:

- review of systems with radiation monitoring equipment and potential effluent release paths
- operator actions assumed in the probabilistic risk assessment (PRA)
- operator actions identified during human factors engineering (HFE) task analysis
- industry operating experience
- engineering judgment
- the identification process and its results reviewed by a multidisciplinary group

Reg Guide 1.97 Revision 4 Position #4 provides a regulatory position on the development of PAM variables: "[T]he scope of instruments that could potentially be selected for accident monitoring (based on selected criteria) should initially be as encompassing as possible. Then, in the process of selecting the actual list of variables to be monitored, licensees could screen out instruments associated with contingency actions that take place beyond the plant's licensing basis."

The radiation monitor inside the bioshield remains an indicator of failed fuel. There are no actions within the Generic Technical Guidelines (GTGs) that operators would or would not take based on bioshield radiation monitor readings. Indications of failed fuel would be used by the site to determine what long term repair and recovery actions would be taken.

During development of the GTGs, containment water level was found to be a poor indicator of containment integrity during transients due to how water volume changes with temperature. It was not included as part of the GTGs due to this limitation. The GTGs will be validated during the conduct of the integrated system validation (ISV) testing. Following ISV testing and subsequent evaluation, changes to FSAR Chapter 7 and/or the NuScale GTGs may result.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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

eRAI No.: 9430
Date of RAI Issue: 04/11/2018

NRC Question No.: 13.05.02.01-10

REGULATORY BASIS REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47(a)(8) requires an applicant for a design certification to provide an FSAR (Final Safety Analysis Report) which includes the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), with certain exceptions. Section 10 CFR 50.34(f)(2)(ii) requires an applicant to "Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, ... “

TMI Action Plan Item I.C.1, a Post-TMI requirement approved by the Commission for implementation, requires the preparation of emergency procedure technical guidelines for development of the Emergency Operating Procedures (EOPs). Preparation of the technical guidelines is conducted in accordance with NUREG-0737, “Clarification of TMI Action Plan Requirements,” and NUREG-0737, Supplement 1, “Requirements for Emergency Response Capability,” which also specify submittal of the technical guidelines to the NRC for review and approval.

Meeting the requirements of TMI Action Plan Item I.C.1 as prescribed in NUREG-0737, Section I.C.1, and Supplement 1 to NUREG-0737, Section 7, is acceptance criteria in SRP 13.5.2.1, “Operating and Emergency Operating Procedures.” Design-specific Generic Technical Guidelines (GTGs), otherwise referred to as the Emergency Operating Guidelines (EOGs), will be used by COL applicants to develop their Plant-Specific Technical Guidelines (P-STGs), from which their EOPs will be developed, and are the responsibility of the DC applicant.

By letter dated November 30, 2017 (ADAMS Accession No. ML17334B822) NuScale submitted technical report TR-1117-57216, “NuScale Generic Technical Guidelines,” for docketing.

ISSUE

The following disparities/inconsistencies were identified with respect to the Post-Accident Monitoring (PAM) instrumentation variable information presented in Chapter 7 of the DCD and

the NuScale GTG technical report, TR-1117-57216:

Containment Isolation Valve (CIV) Position Indication:

1. Chapter 7, Table 7.1-7, “Summary of Type A, B, C, D, and E Variables,” Page 7.1- 73, classifies all CIV Position Indication as PAM variable Types B, C, D.
2. {{

}}2(a),(c)

3. {{

}}2(a),(c)

NRC staff is questioning why the PAM variable designation for Main Steam and Feedwater CIV Position Indication would be different from that of other CIVs.

INFORMATION NEEDED

NRC staff requests that NuScale: (1) explain the PAM variable designation for Main Steam and Feedwater CIV Position Indication, and (2) make any necessary changes to technical report TR-1117-57216, to ensure the completeness and accuracy of the NuScale GTGs.

NuScale Response:

Main steam and feedwater isolation valve position indications are post-accident monitoring (PAM) Type B, C, and D variables. TR-1117-57216 has been updated to reflect this correction.

Impact on DCA:

Technical Report TR-1117-57216, NuScale Generic Technical Guidelines, has been revised as described in the response above and as shown in the markup provided with the response to question 13.05.02.01-15.

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

eRAI No.: 9430
Date of RAI Issue: 04/11/2018

NRC Question No.: 13.05.02.01-11

REGULATORY BASIS REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47(a)(8) requires an applicant for a design certification to provide an FSAR (Final Safety Analysis Report) which includes the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), with certain exceptions. Section 10 CFR 50.34(f)(2)(ii) requires an applicant to "Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, ... "

TMI Action Plan Item I.C.1, a Post-TMI requirement approved by the Commission for implementation, requires the preparation of emergency procedure technical guidelines for development of the Emergency Operating Procedures (EOPs). Preparation of the technical guidelines is conducted in accordance with NUREG-0737, "Clarification of TMI Action Plan Requirements," and NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," which also specify submittal of the technical guidelines to the NRC for review and approval.

Meeting the requirements of TMI Action Plan Item I.C.1 as prescribed in NUREG-0737, Section I.C.1, and Supplement 1 to NUREG-0737, Section 7, is acceptance criteria in SRP 13.5.2.1, "Operating and Emergency Operating Procedures." Design-specific Generic Technical Guidelines (GTGs), otherwise referred to as the Emergency Operating Guidelines (EOGs), will be used by COL applicants to develop their Plant-Specific Technical Guidelines (P-STGs), from which their EOPs will be developed, and are the responsibility of the DC applicant.

By letter dated November 30, 2017 (ADAMS Accession No. ML17334B822) NuScale submitted technical report TR-1117-57216, "NuScale Generic Technical Guidelines," for docketing.

ISSUE

The following disparities/inconsistencies were identified with respect to the Post-Accident Monitoring (PAM) instrumentation variable information presented in Chapter 7 of the DCD and

the NuScale GTG technical report, TR-1117-57216:

RCS T_{HOT}:

1. Chapter 7, Table 7.1-4, “Engineered Safety Feature Actuation System Functions,” Pages 7.1-65 to 7.1-68, specifies Narrow Range **(NR) RCS T_{HOT}** as an actuation parameter for the following ESF Functions:
   - Decay Heat Removal System (DHRS)
   - Demineralized Water System Isolation (DWSI)
2. Chapter 7, Table 7.1-7, “Summary of Type A, B, C, D, and E Variables,” Page 7.1- 73, specifies Wide Range **(WR) RCS T_{HOT}** only. NR RCS T_{HOT} is **not included** in the Table. Table 7.1-7 specifies PAM variable Type B for WR RCS T_{HOT}.
3. Chapter 7, Section 7.1.1.2.2, “Post-Accident Monitoring,” “Type B Variables,” Remove Fuel Assembly Heat Critical Safety Function Variables,” Pages 7.1-8/9, specifies **WR RCS T_{HOT}** in the last line of the following paragraph:

*“The Type B variables selected that provide direct indication and verification and used to assess the process of accomplishing or maintaining the combined remove fuel assembly heat and RCS integrity critical safety functions are core exit temperature, RPV riser water level, wide range RCS pressure, containment water level, degrees of subcooling, and **wide range RCS hot temperature**.”*

4. {{

}}2(a),(c)

5. {{

}}2(a),(c)

INFORMATION NEEDED

NRC staff requests that NuScale: (1) explain the disparities/inconsistencies associated with the specification of RCS T_{HOT} instrumentation ranges identified in the above information, and (2) make any necessary changes to technical report TR-1117-57216 and/or Chapter 7 of the DCD,

to ensure the completeness and accuracy of the NuScale GTGs and Chapter 7 DCD Tier 2 information.

NuScale Response:

(1) Nuscale has determined that Wide Range (WR) T_{hot} will be used anywhere T_{hot} is required. WR T_{hot} has the accuracy necessary and is environmentally qualified and therefore is the best choice to determine temperature in the post accident environment.

(2) TR-1117-57216 has been updated to reflect this change. FSAR Chapter 7 does not require any changes as a result of this RAI response.

Impact on DCA:

Technical Report TR-1117-57216, NuScale Generic Technical Guidelines, has been revised as described in the response above and as shown in the markup provided with the response to question 13.05.02.01-15.

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

eRAI No.: 9430
Date of RAI Issue: 04/11/2018

NRC Question No.: 13.05.02.01-12

REGULATORY BASIS REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47(a)(8) requires an applicant for a design certification to provide an FSAR (Final Safety Analysis Report) which includes the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), with certain exceptions. Section 10 CFR 50.34(f)(2)(ii) requires an applicant to "Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, ... “

TMI Action Plan Item I.C.1, a Post-TMI requirement approved by the Commission for implementation, requires the preparation of emergency procedure technical guidelines for development of the Emergency Operating Procedures (EOPs). Preparation of the technical guidelines is conducted in accordance with NUREG-0737, “Clarification of TMI Action Plan Requirements,” and NUREG-0737, Supplement 1, “Requirements for Emergency Response Capability,” which also specify submittal of the technical guidelines to the NRC for review and approval.

Meeting the requirements of TMI Action Plan Item I.C.1 as prescribed in NUREG-0737, Section I.C.1, and Supplement 1 to NUREG-0737, Section 7, is acceptance criteria in SRP 13.5.2.1, “Operating and Emergency Operating Procedures.” Design-specific Generic Technical Guidelines (GTGs), otherwise referred to as the Emergency Operating Guidelines (EOGs), will be used by COL applicants to develop their Plant-Specific Technical Guidelines (P-STGs), from which their EOPs will be developed, and are the responsibility of the DC applicant.

By letter dated November 30, 2017 (ADAMS Accession No. ML17334B822) NuScale submitted technical report TR-1117-57216, “NuScale Generic Technical Guidelines,” for docketing.

ISSUE

The following disparities/inconsistencies were identified with respect to the Post-Accident Monitoring (PAM) instrumentation variable information presented in Chapter 7 of the DCD and

the NuScale GTG technical report, TR-1117-57216:

Containment Pressure:

1. Chapter 7, Table 7.1-4, “Engineered Safety Feature Actuation System Functions,” Pages 7.1-65 to 7.1-68, specifies Narrow Range **(NR) Containment Pressure** as an actuation parameter for the following ESF Functions:
   - Decay Heat Removal System (DHRS)
   - Containment System Isolation (CSI) Signal
   - Chemical and Volume Control System Isolation (CVCSI)
   - Demineralized Water System Isolation (DWSI)
2. Chapter 7, Table 7.1-7, “Summary of Type A, B, C, D, and E Variables,” Page 7.1- 73, specifies Wide Range **(WR) Containment Pressure** only. NR Containment Pressure is **not included** in the Table. Table 7.1-7 specifies PAM variable Types B, C, D for WR Containment Pressure.
3. Chapter 7, Section 7.1.1.2.2, “Post-Accident Monitoring,” “Type B Variables,” Maintain Containment Integrity Critical Safety Function Variables,” Pages 7.1-9, specifies **WR Containment Pressure** in the following paragraph:

*“Maintain Containment Integrity is both a critical safety function and a fission product barrier (Containment) which serves as the primary means to control radioactive effluent releases. The same variables that are used to provide direct indication and support the containment integrity critical safety function are: **wide range containment pressure**, containment isolation valve position, containment water level, and inside bioshield area radiation monitor.”*

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INFORMATION NEEDED

NRC staff requests that NuScale: (1) explain the disparities/inconsistencies associated with the specification of Containment Pressure instrumentation ranges identified in the above information, (2) explain the PAM variable designation of “N/A” for the Safety Function flowchart decision points identified in Items 4 and 5 above, and (3) make any necessary changes to technical report TR-1117-57216 and/or Chapter 7 of the DCD, to ensure the completeness and accuracy of the NuScale GTGs and Chapter 7 DCD Tier 2 information.

NuScale Response:

The Generic Technical Guidelines (GTGs) were developed after submittal of the design certification application. The analysis work supporting the GTGs identified that the wide range containment pressure indication did not provide the accuracy needed to support operator recognition of the smallest analyzed containment leak size of 20 gpm. Narrow range containment pressure was added as a post-accident monitoring (PAM) Type B variable to provide the needed diagnostic accuracy.

The GTGs have been revised to reflect containment narrow range pressure as a PAM Type B variable. FSAR Chapter 7 Table 7.1-7 has been revised to include containment narrow range pressure as a PAM Type B variable and Section 7.1.1.2.2 has been updated to reflect the use of narrow range containment pressure. Conforming changes have been made to FSAR Tier 2 Table 3.11-1 and Tier 1 Table 2.5-5.

Impact on DCA:

FSAR Tier 2 Section 7.1.1.2.2, FSAR Tier 2 Table 7.1-7, Table 3.11-1, and Tier 1 Table 2.5-5 and related Technical Report TR-1117-57216, NuScale Generic Technical Guidelines, have been revised as described in the response above and as shown in the markup provided with this response.

RAI 11.05-2

Table 7.1-7: Summary of Type A, B, C, D, and E Variables

Variable	Range	System	Type A	Type B	Type C	Type D	Type E
Neutron Flux (Note 1)	0-200% RTP	MPS		X		X	
Core Exit Temperatures	0-2300°F	MPS		X	X	X	
Core Inlet Temperatures	0-2300°F	MPS		X			
Wide Range RCS Pressure	0-2500 psia	MPS		X	X	X	
Degrees of Subcooling	N/A-calculated variable	MPS		X			
Wide Range RCS T_{HOT}	40-700°F	MPS		X			
RPV Riser Level	Top of upper core plate to top of RPV Riser	MPS		X	X	X	
Narrow Range Containment Pressure	0-20 psia	MPS		X			
Wide Range Containment Pressure	0-1200 psia	MPS		X	X	X	
Containment Water Level	ECCS RRVs to Top of Containment	MPS		X	X	X	
Containment Isolation Valve Positions	Closed	MPS		X	X	X	
Inside Bioshield Area Radiation Monitor	Note 3	MPS		X	X		
ECCS Valve Position	Open/Closed	MPS				X	
Reactor Pool Temperature (Operating Bay)	40-220°F	MPS				X	
Spent Fuel Pool Water Level	Top of spent fuel racks to top of pool	PPS				X	
DHRS Valve Position	Open	MPS				X	
Secondary MSIV Position	Closed	MPS				X	
Secondary MSIV Bypass Valve Position	Closed	MPS				X	
FWRV Position	Closed	MPS				X	
Main Steam Temperature (DHRS Inlet Temperature)	100-700°F	MPS				X	
Main Steam Pressure (DHRS inlet pressure)	0-1200 psia	MPS				X	
DHRS Outlet Temperature	40-440°F	MPS				X	
DHRS Outlet Pressure	0-1200 psia	MPS				X	
RCS Flow	0-120% flow	MPS				X	
Reactor Trip Breaker Position Feedback	Open	MPS				X	
Pressurizer Heater Trip Breaker Position Feedback	Open	MPS				X	
Demineralized Water Supply Isolation Valve Position	Closed	MPS				X	
Under-the-Bioshield Temperature	40-700°F	MPS				X	
EDSS-MS and EDSS-C Bus Voltage	0-150 Vdc	MPS/PPS				X	

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}}2(a),(c)

4.7 Setpoint Selection

The listed setpoints have been derived from safety analysis (Table 7.1-4 of Reference 7.2.1), calculations, or best estimate. These ~~final~~ setpoints may be adjusted to reflect design changes~~deviate from those listed here~~ due to final selection of instrumentation, accuracy, and allowing appropriate time for the operator to respond. The values have been included within these guidelines to provide a reference and it is anticipated that the basis for the setpoints will remain constant.

Instrumentation requirements have been provided based on NuScale requirements, regulatory requirements, or vendor recommendations, but must be refined once the actual instrumentation is selected or purchased. Emergency procedures developed from these GTGs will need to reference the plant specific equipment values, ranges, and accuracies.

4.8 Implementation Strategy

Upon meeting an entry condition as listed in section 4.5, all of the the safety function and defense-in-depth flow charts for the affected unit are entered concurrently. By design, the various flow chart analysis is completed by the human-system interface automatically more than once per second. This analysis consists of evaluating system parameters against the flow chart decision setpoints and resulting in an end point. An end point can be either the green safety function met indication and no procedure to be performed, OR a red/yellow condition in which there are applicable procedure steps to be performed. When the procedure steps have been successful, the flow chart is re-evaluated to determine the appropriate end point.

Safety functions always take priority over defense-in-depth functions. The safety functions are arranged in order of importance: containment integrity, reactivity, and core heat removal. If two or more red paths exist on a single unit, then the crew starts with containment integrity then reactivity and core heat removal as actions are completed. If an operator is performing steps in a higher priority function but is waiting for an action to complete, then actions may be taken on lower level safety functions or defense-in-depth

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5.1.3 Containment Isolation Signal Actuation

This subfunction is intended to verify that the automatic protective action of containment isolation is complete. Containment isolation is prioritized higher than CVC isolation since the CVC valves are a subset of the containment isolation valves and would be verified

along with all other containment isolation valves if the conditions for a containment isolation are present.

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5.1.4 Chemical and Volume Control System Isolation Actuation

CVCS Isolation is prioritized after CIS actuation since these valves represent a subset of all containment isolation valves. If a CIS actuation signal is generated, these valves are checked by CI-4 and CI-5. Only if the specific conditions requiring a less comprehensive CVCS isolation is required would the valves be assess by CV-3 and CV-4.

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5.2.3 Reactivity Verification

Reactivity verification is the lowest priority subfunction. The reactivity safety function is not jeopardized during ATWS and return to criticality events as the reactor fuel does not become damaged in these events. For this reason the reactivity safety function remains yellow if control rods are mechanically stuck or reactor power exhibits anomalous behavior. Failure of reactivity control safety systems to actuate are addressed in either the RTS actuation or dilultion isolation subfunctions.

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5.3 Core Heat Removal Safety Function

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Technical basis: This evaluation point determines if the ECCS should have actuated on the low AC voltage timer. There are three timers per MPS division that have actuation logic of ‘two-out-of-three’ sensing loss of voltage to the 480 VAC diesel backed motor control centers.

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5.3.65.3.5 Pressurizer Heater Trip Actuation

If the pressurizer heaters were to become uncovered and remain energized, it is possible that the heaters could challenge the RCS pressure vessel as the heat from the heaters is transferred directly to the metal of the RCS. The heater trip is last of the heat removal subfunctions since it is designed to prevent a loss of coolant event during normal cooldown and shrink of the RCS volume. If a loss of coolant event were to occur, the event would be mitigated by ECCS which is evaluated as a higher subfunction.

This subfunction is prioritized last within the core heat removal safety function since it is intended to protect the RCS as a pressure boundary during cooldown events. If an actual loss of coolant accident occurs, the RCS pressure boundary is intentionally lost when ECC actuates. A DHRS actuation also results in a PZR heater trip actuation, so this subfunction is a direct support function to the DHRS subfunction.

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Table 10-2 List of general technical guidelines decision variables

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RAI 13.05.02.01-12

Table 2.5-5: Safety Display and Indication System Accident Monitoring Variables

Variable	Type B	Type C
Source range count rate	X	
Intermediate range log power	X	
Power range linear power	X	
Neutron monitoring system-flood	X	
Core exit temperature	X	X
Core inlet temperature	X	
Wide range RCS pressure	X	X
Degrees of subcooling (calculated from WR RCS T_{hot} and WR RCS pressure)	X	
Wide range RCS hot temperature	X	
RPV riser level	X	X
Wide range containment pressure	X	X
Containment water level	X	X
CIV positions	X	X
Inside bioshield area radiation monitor	X	X
Narrow range containment pressure	X	

RAI 03.11-18, RAI 06.02.04-2, RAI 08.01-1S1, RAI 13.05.02.01-12

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments

Description[6] ~~(6)~~	Location[1] ~~(1)~~	EQ Environment	Qualification Program	PAM[2] ~~(2)~~	EQ Category[3] ~~(3)~~	Operating Time
Nuclear Power Module	-					
Containment System (CNT-A013)	-					
I&C Division I Electrical Penetration Assembly (EPA)	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
I&C Division II Electrical Penetration Assembly (EPA)	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
PZR Heater Power #1 Nozzle Electrical Penetration Assembly (EPA)	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	N/A	A	Extended Term (<= 720 hr)
PZR Heater Power #2 Nozzle Electrical Penetration Assembly (EPA)	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	N/A	A	Extended Term (<= 720 hr)
I&C Channel A Electrical Penetration Assembly (EPA)	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
I&C Channel B Electrical Penetration Assembly (EPA)	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
I&C Channel C Electrical Penetration Assembly (EPA)	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
I&C Channel D Electrical Penetration Assembly (EPA)	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	C	A	Extended PAM (100 days)
CRD Power Nozzle Electrical Penetration Assembly (EPA)	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	N/A	A	Extended Term (<= 720 hr)
RPI Group #1 Electrical Penetration Assembly (EPA)	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	N/A	A	Extended Term (<= 720 hr)
RPI Group #2 Electrical Penetration Assembly (EPA)	EQ Zone F EQ Zone G	Harsh	Electrical Mechanical	N/A	A	Extended Term (<= 720 hr)
MS #1 CIV (MSIV #1)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
MS #2 CIV (MSIV #2)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)

Table 3.11-1: List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments (Continued)

Description(6) ~~(6)~~	Location(1) ~~(1)~~	EQ Environment	Qualification Program	PAM(2) ~~(2)~~	EQ Category(3) ~~(3)~~	Operating Time
MS line #1 Bypass Valve (MSIV Bypass #1)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
MS line #2 Bypass Valve (MSIV Bypass #2)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
FW #1 CIV (FWIV #1)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
FW #2 CIV (FWIV #2)	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
FW line #1 Check Valve	EQ Zone G	Harsh	Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
FW line #2 Check Valve	EQ Zone G	Harsh	Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
CVC Discharge CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
CVC Injection CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
CVC PZR Spray CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
RPV High Point Degas CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
RCCW Supply CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
RCCW Return CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
CE CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
CFDS CIV	EQ Zone G	Harsh	Electrical Mechanical	N/A	A B	Short Term (<= 1 hr) Extended Term (<= 720 hr)
Hydraulic Skid Note 5	EQ Zone M EQ Zone N	Harsh	Electrical Mechanical	N/A	A	Short Term (<= 1 hr)
Containment Pressure Transducer (Narrow Range)	EQ Zone E EQ Zone F	Harsh	Electrical	~~N/A~~B	A	~~Short Term (<= 1 hr)~~Extended Term (<= 720 hr)
Containment Pressure Transducer (Wide Range)	EQ Zone E EQ Zone F	Harsh	Electrical	C	A	Extended PAM (100 days)

protect the fuel clad fission product barrier. This is automatically actuated when there is an existing loss of RCS integrity as indicated by low reactor pressure vessel (RPV) riser water level or high containment water level.

The Type B variables identified in Table 7.1-7 are those necessary to implement the plant abnormal operating procedures, emergency operating procedures and functional restoration procedures, and to maintain the plant critical safety functions described below.

Reactivity Control Safety Function Variables:

The Type B variables that provide direct indication and are used to assess the process of accomplishing or maintaining reactivity control are neutron flux and core inlet and exit temperature.

Remove Fuel Assembly Heat Critical Safety Function Variables

The Type B variables selected that provide direct indication and verification and used to assess the process of accomplishing or maintaining the combined remove fuel assembly heat and RCS integrity critical safety functions are core exit temperature, RPV riser water level, wide range RCS pressure, containment water level, degrees of subcooling, and wide-range RCS hot temperature.

Maintain Containment Integrity Critical Safety Function Variables

RAI 13.05.02.01-12

Maintain Containment Integrity is both a critical safety function and a fission product barrier (Containment) which serves as the primary means to control radioactive effluent releases. The same variables that are used to provide direct indication and support the containment integrity critical safety function are: narrow range containment pressure, wide range containment pressure, containment isolation valve position, containment water level, and inside bioshield area radiation monitor.

Type C Variables

Type C variables are those variables that provide primary information to the control room operators to indicate the potential for breach or the actual breach of fission product barriers: fuel cladding, reactor coolant system, and containment pressure boundary.

The selection of these variables represents a minimum set of plant variables that provide the most direct indication of the integrity of the three fission product barriers and provide the capability for monitoring beyond the design limits (extended range).

Fuel Cladding Fission Product Barrier Variables

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

eRAI No.: 9430
Date of RAI Issue: 04/11/2018

NRC Question No.: 13.05.02.01-13

REGULATORY BASIS REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47(a)(8) requires an applicant for a design certification to provide an FSAR (Final Safety Analysis Report) which includes the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), with certain exceptions. Section 10 CFR 50.34(f)(2)(ii) requires an applicant to "Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, ... “

TMI Action Plan Item I.C.1, a Post-TMI requirement approved by the Commission for implementation, requires the preparation of emergency procedure technical guidelines for development of the Emergency Operating Procedures (EOPs). Preparation of the technical guidelines is conducted in accordance with NUREG-0737, “Clarification of TMI Action Plan Requirements,” and NUREG-0737, Supplement 1, “Requirements for Emergency Response Capability,” which also specify submittal of the technical guidelines to the NRC for review and approval.

Meeting the requirements of TMI Action Plan Item I.C.1 as prescribed in NUREG-0737, Section I.C.1, and Supplement 1 to NUREG-0737, Section 7, is acceptance criteria in SRP 13.5.2.1, “Operating and Emergency Operating Procedures.” Design-specific Generic Technical Guidelines (GTGs), otherwise referred to as the Emergency Operating Guidelines (EOGs), will be used by COL applicants to develop their Plant-Specific Technical Guidelines (P-STGs), from which their EOPs will be developed, and are the responsibility of the DC applicant.

By letter dated November 30, 2017 (ADAMS Accession No. ML17334B822) NuScale submitted technical report TR-1117-57216, “NuScale Generic Technical Guidelines,” for docketing.

ISSUE

The following disparities/inconsistencies were identified with respect to the Post-Accident Monitoring (PAM) instrumentation variable information presented in Chapter 7 of the DCD and

the NuScale GTG technical report, TR-1117-57216:

Pressurizer Level:

1. Chapter 7, Table 7.1-7, “Summary of Type A, B, C, D, and E Variables,” Page 7.1- 73, **does not** specify Pressurizer Level.
2. {{

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3. {{

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INFORMATION NEEDED

NRC staff requests that NuScale: (1) explain the disparities/inconsistencies associated with the specification of Pressurizer Level and the associated PAM variable classifications identified in the above information, and (2) make any necessary changes to technical report TR-1117-57216 and/or Chapter 7 of the DCD, to ensure the completeness and accuracy of the NuScale GTGs and Chapter 7 DCD Tier 2 information.

NuScale Response:

Pressurizer level and reactor pressure vessel riser level are both detected by the same instrument, just a different portion of the range. Pressurizer level is only used in the the generic technical guidelines (GTGs) to perform automatic action verifications and therefore should not be classified as a post-accident monitoring (PAM) variable.

TR-1117-57216 has been revised to reflect that pressurizer level is not a PAM variable. No change is required to FSAR Chapter 7.

Impact on DCA:

Technical Report TR-1117-57216, NuScale Generic Technical Guidelines, has been revised as

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described in the response above and as shown in the markup provided with the response to question 13.05.02.01-15.

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

eRAI No.: 9430
Date of RAI Issue: 04/11/2018

NRC Question No.: 13.05.02.01-14

REGULATORY BASIS REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47(a)(8) requires an applicant for a design certification to provide an FSAR (Final Safety Analysis Report) which includes the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), with certain exceptions. Section 10 CFR 50.34(f)(2)(ii) requires an applicant to "Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, ... “

TMI Action Plan Item I.C.1, a Post-TMI requirement approved by the Commission for implementation, requires the preparation of emergency procedure technical guidelines for development of the Emergency Operating Procedures (EOPs). Preparation of the technical guidelines is conducted in accordance with NUREG-0737, “Clarification of TMI Action Plan Requirements,” and NUREG-0737, Supplement 1, “Requirements for Emergency Response Capability,” which also specify submittal of the technical guidelines to the NRC for review and approval.

Meeting the requirements of TMI Action Plan Item I.C.1 as prescribed in NUREG-0737, Section I.C.1, and Supplement 1 to NUREG-0737, Section 7, is acceptance criteria in SRP 13.5.2.1, “Operating and Emergency Operating Procedures.” Design-specific Generic Technical Guidelines (GTGs), otherwise referred to as the Emergency Operating Guidelines (EOGs), will be used by COL applicants to develop their Plant-Specific Technical Guidelines (P-STGs), from which their EOPs will be developed, and are the responsibility of the DC applicant.

By letter dated November 30, 2017 (ADAMS Accession No. ML17334B822) NuScale submitted technical report TR-1117-57216, “NuScale Generic Technical Guidelines,” for docketing.

ISSUE

The following disparities/inconsistencies were identified with respect to the Post-Accident Monitoring (PAM) instrumentation variable information presented in Chapter 7 of the DCD and

the NuScale GTG technical report, TR-1117-57216:

Neutron Flux:

1. Chapter 7, Table 7.1-7, “Summary of Type A, B, C, D, and E Variables,” Page 7.1- 73, classifies Neutron Flux as PAM variable Types B and D.
2. {{

}}[2(a),(c)]
3. NuScale DCD Tier 1, Table 2.5-5, “Safety Display and Indication System Accident Monitoring Variables,” classifies Source, Intermediate, and Power Range indication as PAM variable Type B only (Note: DCD Tier 1, Table 2.5-5, variable classifications are limited to Types B and C).

INFORMATION NEEDED

NRC staff requests that NuScale: (1) explain the disparities/inconsistencies associated with classification of the PAM variable Type for Neutron Flux identified in the above information, and (2) make any necessary changes to technical report TR-1117-57216 and/or Chapter 7 of the DCD, to ensure the completeness and accuracy of the NuScale GTGs and Chapter 7 DCD Tier 2 information.

NuScale Response:

The Neutron Flux variable is only a PAM Type B and D variable. TR-1117-57216 has been updated to reflect this correction. No changes are required to FSAR Chapter 7 or Tier 1 Table 2.5-5.

Impact on DCA:

Technical Report TR-1117-57216, NuScale Generic Technical Guidelines, has been revised as described in the response above and as shown in the markup provided with the response to question 13.05.02.01-15.

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

eRAI No.: 9430
Date of RAI Issue: 04/11/2018

NRC Question No.: 13.05.02.01-15

REGULATORY BASIS REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47(a)(8) requires an applicant for a design certification to provide an FSAR (Final Safety Analysis Report) which includes the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), with certain exceptions. Section 10 CFR 50.34(f)(2)(ii) requires an applicant to "Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, ... “

TMI Action Plan Item I.C.1, a Post-TMI requirement approved by the Commission for implementation, requires the preparation of emergency procedure technical guidelines for development of the Emergency Operating Procedures (EOPs). Preparation of the technical guidelines is conducted in accordance with NUREG-0737, “Clarification of TMI Action Plan Requirements,” and NUREG-0737, Supplement 1, “Requirements for Emergency Response Capability,” which also specify submittal of the technical guidelines to the NRC for review and approval.

Meeting the requirements of TMI Action Plan Item I.C.1 as prescribed in NUREG-0737, Section I.C.1, and Supplement 1 to NUREG-0737, Section 7, is acceptance criteria in SRP 13.5.2.1, “Operating and Emergency Operating Procedures.” Design-specific Generic Technical Guidelines (GTGs), otherwise referred to as the Emergency Operating Guidelines (EOGs), will be used by COL applicants to develop their Plant-Specific Technical Guidelines (P-STGs), from which their EOPs will be developed, and are the responsibility of the DC applicant.

By letter dated November 30, 2017 (ADAMS Accession No. ML17334B822) NuScale submitted technical report TR-1117-57216, “NuScale Generic Technical Guidelines,” for docketing.

ISSUE

The following disparities/inconsistencies were identified with respect to the Post-Accident Monitoring (PAM) instrumentation variable information presented in Chapter 7 of the DCD and

the NuScale GTG technical report, TR-1117-57216:

Internal MPS Timer:

1. Chapter 7, Table 7.1-7, “Summary of Type A, B, C, D, and E Variables,” Page 7.1- 73, **does not** specify the Low Voltage AC Electrical Distribution System (ELVS) 24- Hour Internal MPS Timer.
2. {{

}}2(a),(c)

INFORMATION NEEDED

NRC staff requests that NuScale: (1) explain the disparities/inconsistencies associated with specification of the Internal MPS Timer and the associated PAM variable classification identified in the above information, and (2) make any necessary changes to technical report TR-1117-57216 and/or Chapter 7 of the DCD, to ensure the completeness and accuracy of the NuScale GTGs and Chapter 7 DCD Tier 2 information.

NuScale Response:

The critical safety function status trees in the NuScale Generic Technical Guidelines (GTGs) include direction to both assess critical safety function status and ensure automatic actions related to the critical safety function have occurred. This is done because the automated actions are integral to the logic for controlling the safety function response. The operators verify automation occurs correctly by monitoring the post-accident monitoring (PAM) variable response. Reactor pressure vessel level and containment vessel level are the PAM variables used to assess the critical safety function response.

Variables used to initiate automated actions are not always PAM variables as in this case. Consistent with industry practice, the initiation variables for automation are not listed as PAM variables unless they are a direct indicator of the safety function status.

The internal Module Protection System (MPS) timer will continue to be displayed to preserve the logic of the status tree, but the GTGs have been revised so that the timer is not categorized as a PAM variable. Table 10-2 did not need to be changed since it lists variables supporting operator decisions (in this case the success of automation) rather than PAM variables. The change will make the GTGs consistent with FSAR Chapter 7.

Impact on DCA:

Technical Report TR-1117-57216, NuScale Generic Technical Guidelines, has been revised as described in the response above and as shown in the markup provided in this response.

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}}2(a),(c)

4.7 Setpoint Selection

The listed setpoints have been derived from safety analysis (Table 7.1-4 of Reference 7.2.1), calculations, or best estimate. These ~~final~~ setpoints may be adjusted to reflect design changes~~deviate from those listed here~~ due to final selection of instrumentation, accuracy, and allowing appropriate time for the operator to respond. The values have been included within these guidelines to provide a reference and it is anticipated that the basis for the setpoints will remain constant.

Instrumentation requirements have been provided based on NuScale requirements, regulatory requirements, or vendor recommendations, but must be refined once the actual instrumentation is selected or purchased. Emergency procedures developed from these GTGs will need to reference the plant specific equipment values, ranges, and accuracies.

4.8 Implementation Strategy

Upon meeting an entry condition as listed in section 4.5, all of the the safety function and defense-in-depth flow charts for the affected unit are entered concurrently. By design, the various flow chart analysis is completed by the human-system interface automatically more than once per second. This analysis consists of evaluating system parameters against the flow chart decision setpoints and resulting in an end point. An end point can be either the green safety function met indication and no procedure to be performed, OR a red/yellow condition in which there are applicable procedure steps to be performed. When the procedure steps have been successful, the flow chart is re-evaluated to determine the appropriate end point.

Safety functions always take priority over defense-in-depth functions. The safety functions are arranged in order of importance: containment integrity, reactivity, and core heat removal. If two or more red paths exist on a single unit, then the crew starts with containment integrity then reactivity and core heat removal as actions are completed. If an operator is performing steps in a higher priority function but is waiting for an action to complete, then actions may be taken on lower level safety functions or defense-in-depth

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5.1.3 Containment Isolation Signal Actuation

This subfunction is intended to verify that the automatic protective action of containment isolation is complete. Containment isolation is prioritized higher than CVC isolation since the CVC valves are a subset of the containment isolation valves and would be verified

along with all other containment isolation valves if the conditions for a containment isolation are present.

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5.1.4 Chemical and Volume Control System Isolation Actuation

CVCS Isolation is prioritized after CIS actuation since these valves represent a subset of all containment isolation valves. If a CIS actuation signal is generated, these valves are checked by CI-4 and CI-5. Only if the specific conditions requiring a less comprehensive CVCS isolation is required would the valves be assess by CV-3 and CV-4.

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5.2.3 Reactivity Verification

Reactivity verification is the lowest priority subfunction. The reactivity safety function is not jeopardized during ATWS and return to criticality events as the reactor fuel does not become damaged in these events. For this reason the reactivity safety function remains yellow if control rods are mechanically stuck or reactor power exhibits anomalous behavior. Failure of reactivity control safety systems to actuate are addressed in either the RTS actuation or dilultion isolation subfunctions.

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5.3 Core Heat Removal Safety Function

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Technical basis: This evaluation point determines if the ECCS should have actuated on the low AC voltage timer. There are three timers per MPS division that have actuation logic of ‘two-out-of-three’ sensing loss of voltage to the 480 VAC diesel backed motor control centers.

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5.3.65.3.5 Pressurizer Heater Trip Actuation

If the pressurizer heaters were to become uncovered and remain energized, it is possible that the heaters could challenge the RCS pressure vessel as the heat from the heaters is transferred directly to the metal of the RCS. The heater trip is last of the heat removal subfunctions since it is designed to prevent a loss of coolant event during normal cooldown and shrink of the RCS volume. If a loss of coolant event were to occur, the event would be mitigated by ECCS which is evaluated as a higher subfunction.

This subfunction is prioritized last within the core heat removal safety function since it is intended to protect the RCS as a pressure boundary during cooldown events. If an actual loss of coolant accident occurs, the RCS pressure boundary is intentionally lost when ECC actuates. A DHRS actuation also results in a PZR heater trip actuation, so this subfunction is a direct support function to the DHRS subfunction.

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Table 10-2 List of general technical guidelines decision variables

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

eRAI No.: 9430
Date of RAI Issue: 04/11/2018

NRC Question No.: 13.05.02.01-16

REGULATORY BASIS REQUIREMENTS

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 52.47(a)(8) requires an applicant for a design certification to provide an FSAR (Final Safety Analysis Report) which includes the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), with certain exceptions. Section 10 CFR 50.34(f)(2)(ii) requires an applicant to "Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, ... “

TMI Action Plan Item I.C.1, a Post-TMI requirement approved by the Commission for implementation, requires the preparation of emergency procedure technical guidelines for development of the Emergency Operating Procedures (EOPs). Preparation of the technical guidelines is conducted in accordance with NUREG-0737, “Clarification of TMI Action Plan Requirements,” and NUREG-0737, Supplement 1, “Requirements for Emergency Response Capability,” which also specify submittal of the technical guidelines to the NRC for review and approval.

Meeting the requirements of TMI Action Plan Item I.C.1 as prescribed in NUREG-0737, Section I.C.1, and Supplement 1 to NUREG-0737, Section 7, is acceptance criteria in SRP 13.5.2.1, “Operating and Emergency Operating Procedures.” Design-specific Generic Technical Guidelines (GTGs), otherwise referred to as the Emergency Operating Guidelines (EOGs), will be used by COL applicants to develop their Plant-Specific Technical Guidelines (P-STGs), from which their EOPs will be developed, and are the responsibility of the DC applicant.

By letter dated November 30, 2017 (ADAMS Accession No. ML17334B822) NuScale submitted technical report TR-1117-57216, “NuScale Generic Technical Guidelines,” for docketing.

ISSUE

The Containment Integrity Safety Function flowchart in Section 5.1 of the NuScale GTGs depicts the logic and specifies the operator actions necessary to assess and maintain the

Containment Integrity (CI) Safety Function. The CI Safety Function Bases discussion for decision point {{

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In addition, the Setpoint Basis paragraph on Page 29 discusses sensor capabilities for measuring hydrogen concentration, but does not provide a comparable discussion with respect to sensor capabilities for measuring oxygen concentration.

INFORMATION NEEDED

NRC staff requests that NuScale: (1) explain omission of the "Containment H_2" parameter from Table 10-2 of the GTGs, (2) explain the discrepancy noted in the Setpoint Basis paragraph associated with sensor capabilities for measuring oxygen concentration, and (3) make any necessary changes to technical report TR-1117-57216 to ensure the completeness and accuracy of the NuScale GTGs.

NuScale Response:

The containment vessel (CNV) in the NuScale design is maintained at a vacuum and, as such, is inert during normal operation. This is a paradigm shift in the monitoring requirements for a typical pressurized water reactor (PWR) containment that contains air. It is expected that some hydrogen generation will occur in accident scenarios when the reactor coolant system (RCS) coolant is transferred to containment.

Oxygen concentration is expected to be negligible in all conditions and so it is a key parameter to determine if a combustible atmosphere exists. Only Oxygen concentration is required to be monitored as it should never be in concentrations above three percent with the assumption that hydrogen is present. This is why only Oxygen is listed in Table 10-2 of the generic technical guidelines (GTGs).

The monitor that we are specifying measures both Oxygen and Hydrogen concentration. Since the monitor will be dual purpose, it is prudent that operators check both indications, even though Oxygen is the prime element of interest. The setpoint basis only includes the information known to date on the expected monitor capabilities. Future revisions to the GTGs will provide updated information as the specific instrumentation is selected.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Enclosure 3:

Affidavit of Zackary W. Rad, AF-0618-60377

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

1. I am the Director, Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.
2. I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:
    a. The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.
    b. The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.
    c. Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.
    d. The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.
    e. The information requested to be withheld consists of patentable ideas.
3. Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the method by which NuScale develops its generic technical guidelines.

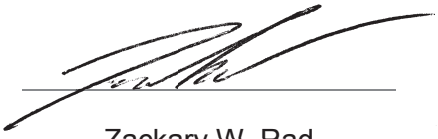
    NuScale has performed significant research and evaluation to develop a basis for this method and has invested significant resources, including the expenditure of a considerable sum of money.

    The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

    If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

4. The information sought to be withheld is in the enclosed response to NRC Request for Additional Information No. 414, eRAI 9430. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.
5. The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).
6. Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:
    a. The information sought to be withheld is owned and has been held in confidence by NuScale.
    b. The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.
    c. The information is being transmitted to and received by the NRC in confidence.
    d. No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.
    e. Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 11, 2018.



Zackary W. Rad