RAIO-0618-60408

June 12, 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. 472 (eRAI No. 9445) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 472 (eRAI No. 9445)," dated May 10, 2018

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

The Enclosure to this letter contains NuScale's response to the following RAI Questions from NRC eRAI No. 9445:

- 16-42
- 16-43
- 16-44

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 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
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9445

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9445

NuScale Power

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

eRAI No.: 9445
Date of RAI Issue: 05/10/2018

NRC Question No.: 16-42

10 CFR 50.36(c)(2)(ii)(B) requires that a technical specification limiting condition for operation (LCO) be established for a “process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.” FSAR, Tier 2, Section 4.3.2.2.1 states that a limit on the heat flux hot channel factor (F_Q) is used to ensure that none of the fuel design criteria are exceeded. However, the currently proposed NuScale generic technical specifications (GTS) do not include an LCO for F_Q. The NRC staff relies upon such an LCO to establish a finding that each NuScale Power MODULE will be operated within the bounds of the safety analyses. Accordingly, NRC staff requests that NuScale either (1) update GTS to include an LCO for F_Q, or (2) update TR-1116-52011, "Technical Specifications Regulatory Conformance and Development," to provide justification for not including F_Q as an LCO in the NuScale GTS.

NuScale Response:

The heat flux hot channel factor (F_Q) is used in the NuScale design to calculate the peak linear heat generation rate to ensure that the specified acceptable fuel design limit for fuel centerline melting is not exceeded. The NuScale design is characterized by a relatively low linear heat rate (kW/ft) compared to the PWR operating fleet and has substantial margin to fuel centerline melting at normal power levels. F_Q is not used as an initial condition for any transient or design basis accident, including loss of coolant accident. As a result, a Limiting Condition for Operation for F_Q is not needed in the NuScale design. FSAR Sections 4.3 and 4.4 are modified to clarify this point.

Impact on DCA:

FSAR Sections 4.3 and 4.4 have been revised as described in the response above and as shown in the markup provided in this response.

discussion of power distribution uncertainty, including application and a means for updating the uncertainty values.

4.3.2.2.1 Definitions

Maximum $F_{\Delta H}$

The maximum enthalpy rise hot channel factor, $F_{\Delta H}$, is defined as the ratio of the maximum integrated fuel rod power to the average fuel rod power. The limit on $F_{\Delta H}$ is established to ensure that the fuel design criteria are not exceeded and the accident analysis assumptions remain valid. This limit ensures that the design basis value for the CHF ratio is met for normal operation, anticipated operational occurrences, and infrequent events. The $F_{\Delta H}$ limit is representative of the coolant flow channel with the maximum enthalpy rise. This channel has the highest power input to the coolant and therefore the highest probability for CHF.

The NuScale design limit for $F_{\Delta H}$ is 1.50 and is based on the safety analysis.

Maximum F_Q

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The heat flux hot channel factor (or total peaking factor), F_Q, is the ratio of maximum local heat flux on the surface of a fuel rod to the average fuel rod heat flux for the entire core. The maximum F_Q value is used to calculate the peak linear heat generation rate (LHGR). The ~~limit on~~maximum value of F_Q is used to ensure ~~none of~~ the ~~fuel design criteria are~~specified acceptable fuel design limit for fuel centerline melting is not exceeded.

Axial Peaking Factor F_z

The axial peaking factor, F_z, is the maximum relative power at any axial point in a fuel rod, divided by the average power of the fuel rod.

Engineering Hot Channel Factor, F_E

The engineering heat flux hot channel factor, F_E, accounts for manufacturing tolerances on such parameters as enrichment, pellet density, and pellet diameter.

Measurement Uncertainty Factor, F_M

The measurement uncertainty factor, F_M, accounts for the measurement error associated with power distribution predictions. F_M is accounted for in the nuclear reliability factor (NRF) determined for F_Q. The NRF is discussed in more detail in Section 4.3.2.2.7 and in Reference 4.3-1.

The range of applicability of the NSP2 CHF correlation is:

Pressure, psia	300 to 2300
RAI 04.04-6, RAI 04.04-9 Local Mass Flux, $10^6 lb_m/hr\text{-}ft^2$	0.110 to 0.700
RAI 04.04-6, RAI 04.04-9 Local equilibrium quality, %	≤ 90.0
Inlet equilibrium quality, %	≤ 0

The range of applicability for the NSP4 correlation is:

Pressure, psia	500 to 2300
Mass flux, 10^6 lbm/hr-ft^2	0.116 to 0.635
Local equilibrium quality, %	≤ 95
Inlet equilibrium quality, %	≤ 0

The transient response of the reactor system is dependent on the initial power distribution. Limits provided by the core system and the protection system ensure that the design meets CHF design bases for AOOs. The core operating limits report (COLR) specifies the cycle-specific, power-peaking limits that maintain the core power distribution within prescribed limits during power operation. These power-peaking limits are expressed as limits on total heat flux (F_Q), enthalpy rise ($F_{\Delta H}$), and axial peak (F_Z). These power peaking factors are functions of burnup and power level. Section 4.3 provides additional discussion about the development and use of these limits.

- Enthalpy rise hot channel factor ($F_{\Delta H}$) is the ratio of the power in the hot rod divided by the power in the average rod. This all rods out (ARO) limit ensures that the design basis value for the CHFR is met for normal operation, operational transients, and IEs.

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- The heat flux hot channel factor (or total peaking factor), F_Q, is the ratio of maximum local heat flux on the surface of a fuel rod to the average fuel rod heat flux. The maximum F_Q value is used to calculate the peak linear heat generation rate. The ~~limit on~~maximum F_Q value is ~~established~~utilized to ensure the ~~none of the fuel design criteria are~~specified acceptable fuel design limit for fuel centerline melting is not exceeded ~~and the assumptions made in the accident analysis remain valid~~.
- Axial Peaking Factor (F_Z), is defined as the maximum relative power at any axial point in a fuel rod divided by the average power of the fuel rod. F_Z can be defined for a rod, an assembly, or the entire core.

The Module Protection System (MPS) automatically initiates and controls the protective actions necessary to mitigate the effects of the design basis events (DBEs) identified in Table 7.1-1. The MPS reactor trip functions are listed in Table 7.1-3, including the associated parameter and analytical limits.

The core design and thermal limits are developed such that the thermal margin criteria are not exceeded for normal operation and AOOs. Specifically, there is a 95-percent

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Response to Request for Additional Information

eRAI No.: 9445
Date of RAI Issue: 05/10/2018

NRC Question No.: 16-43

10 CFR 50.36(c)(2)(ii)(B) requires that a technical specification limiting condition for operation (LCO) be established for a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." NuScale generic technical specifications (GTS) contain several LCO subsections that reference limits specified in the CORE OPERATING LIMITS REPORT (COLR), which is a defined term in GTS Section 1.1, "Definitions," and which is specified in GTS Subsection 5.6.3.

By letter dated July 13, 2017 (ML17194B384), NuScale provided a response to RAI 8772, Question 4.3-1 that stated the methodologies and/or values that set the limits in GTS are contained in several locations in DCD Tier 2 (the FSAR). However, there is no direct connection provided in GTS Subsection 5.6.3 to these methodologies and/or values. The NRC staff relies upon the LCOs to establish a finding that the plant will be operated within the bounds of the safety analyses. However, the staff cannot make this finding because GTS Subsection 5.6.3 has insufficient information for the staff to determine which specific methodology is used to set each limit specified by the GTS LCO subsections listed in paragraph a of GTS Subsection 5.6.3.

Accordingly, the NRC staff requests that NuScale revise the DCA in accordance with one of the following options:

1. Revise GTS Subsection 5.6.3, paragraph b, by replacing the proposed bracketed general reference to staff approved topical reports with a list of documents (including FSAR subsections) describing the NRC staff approved analytical methods used to establish the core operating limits; each document should state its revision number and the date of the revision.

   The GTS LCO subsections, which are listed in GTS Subsection 5.6.3, paragraph a, specify these limits in accordance with Criterion 2 of 10 CFR 50.36(c)(2)(ii) by referencing the COLR. The listing of each document should therefore list all LCO subsections that specify limits derived using the analytical method described in the document.

   This option requires NRC staff prior review and approval of all analytical methods used by

NuScale to establish the core operating limits, as described in documents submitted as part of the NuScale design certification application (DCA). The NRC staff would need to confirm that the limits derived using the approved analytical methods are bounded by the accident analyses, and therefore, conclude that the NuScale design provides adequate protection.

To reduce the administrative burden on a COL applicant, designate these documents as a COL action item by bracketing each listed method document (or FSAR reference). In addition, the staff understands that a specific document addressing steps of a core reload analysis is not included in the DCA. Accordingly, the document list should also include a bracketed placeholder for the "reload analytical method."

In order to make clear the meaning of the brackets in this document list, please include a Reviewer's Note in Subsection 5.6.3 for use by a COL applicant to guide its completion of the COL action item. For example (ignore any double line spacing):

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   [------------------------------REVIEWER'S NOTE------------------------------------

   The COL applicant shall confirm the validity of each listed document and the listed Specifications for the associated core operating limits, or state the valid NRC approved analytical method document and list of associated Specifications.

   The COL applicant shall state the valid core reload analysis methodology document and list of associated Specifications.

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   1. [Analytical method document A, Revision number, date.

      (Methodology for Specifications 3.1.3 – Moderator Temperature Coefficient, 3.1.4 – Rod Group Alignment Limits, 3.1.5 - Shutdown Group Insertion Limits, 3.1.6 - Regulating Group Insertion Limits, and 3.1.9 – Boron Dilution Control.)]

   2. [Analytical method document B, Revision number, date.

      (Methodology for Specifications 3.1.1 – SDM, 3.2.1 - Enthalpy Rise Hot Channel Factor, 3.2.2 – AXIAL OFFSET, 3.4.1 - RCS Pressure, Temperature, and Flow Resistance Critical Heat Flux Limits, and 3.5.3 - Ultimate Heat Sink.)]

   3. [Core Reload Analysis Methodology document, Revision #, date. (Methodology for Specifications...]

With this option, a COL applicant or COL holder would be able to change the limits in the COLR in accordance with the specified approved analytical methods without prior NRC approval.

2. Remove GTS Subsection 5.6.3 and revise each GTS LCO subsection that references the COLR by replacing each reference to the COLR with the numerical values specified for each limit, including any figures for limits presented as curves, such as control rod assembly (CRA) group power dependent insertion limits (PDILs), Moderator Temperature Coefficient (MTC), and Axial Offset (AO). This option requires stating core operating limit values and curves in the GTS LCO subsections that now reference a COLR, since GTS Section 5.6 would no longer specify a COLR. This option also requires NRC staff prior review and approval of all specified core operating limit values and curves. Conforming changes to the Bases of the affected GTS LCO subsections would likely be needed. Note that the NRC staff would review the analytical methods used by NuScale to derive the proposed core operating limit values and curves to the degree needed to confirm that the limits are bounded by the accident analyses, and therefore, that the NuScale design provides adequate protection.

   With this option, a change in the core operating limits would require a COL applicant or COL holder to seek prior NRC approval by utilizing the applicable regulatory process for changing the GTS and Bases.

NuScale Response:

The requested changes to Technical Specification 5.6.3.b have been incorporated as shown on the attached pages.

5.6 Reporting Requirements

5.6.3 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

~~2.1.1, "Reactor Core Safety Limits";~~

3.1.1, "SHUTDOWN MARGIN (SDM)";

3.1.3, "Moderator Temperature Coefficient (MTC)";

3.1.4, "Rod Group Alignment Limits";

3.1.5, "Shutdown Group Insertion Limits";

3.1.6, "Regulating Group Insertion Limits";

3.1.8, "PHYSICS TESTS Exceptions";

3.1.9, "Boron Dilution Control";

3.2.1, "Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)";

3.2.2, "AXIAL OFFSET (AO)";

3.4.1, "RCS Pressure, Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits"; and

3.5.3, "Ultimate Heat Sink".

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[--------------------------------REVIEWER'S NOTE--------------------------------
The COL applicant shall confirm the validity of each listed document and the listed Specifications for the associated core operating limits, or state the valid NRC approved analytical method document and list of associated Specifications.

The COL applicant shall state the valid core reload analysis methodology document and list of associated Specifications.
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5.6 Reporting Requirements

5.6.3 Core Operating Limits Report (COLR) (continued)

1. [NuScale Standard Design Certification Analysis (DCA), Part 2, Tier 2, NuScale Final Safety Analysis Report (FSAR), Section 4.3, “Nuclear Design,” Revision 1, March 2018, and TR-0516-49416, “Non-Loss-of-Coolant Accident Analysis Methodology,” Revision 0, May 2016 (NuScale Proprietary).

   (Methodology for Specifications 3.1.1 – SHUTDOWN MARGIN (SDM), 3.1.3 – Moderator Temperature Coefficient, 3.1.4 – Rod Group Alignment Limits, 3.1.5 – Shutdown Group Insertion Limits, 3.1.6 - Regulating Group Insertion Limits, and 3.1.8 - PHYSICS TESTS Exceptions.)] ~~[List of NRC approved Topical Reports that are used to determine the core operating limits listed in 5.6.3.a above.]~~

2. [NuScale DCA, Part 2, Tier 2, NuScale FSAR, Section 9.3.4, “Chemical and Volume Control System,“ Revision 1, March 2018, and TR-0516-49416, “Non-Loss-of-Coolant Accident Analysis Methodology,” Revision 0, May 2016 (NuScale Proprietary).

   (Methodology for Specification 3.1.9 – Boron Dilution Control.)]

3. [NuScale DCA, Part 2, Tier 2, NuScale FSAR, Sections 4.3, “Nuclear Design,” and 4.4, “Thermal and Hydraulic Design,“ Revision 1, March 2018; TR-0516-49416, “Non-Loss-of-Coolant Accident Analysis Methodology,” Revision 0, May 2016 (NuScale Proprietary); and TR-0915-17564, “Subchannel Analysis Methodology,” Revision 1, September 2015 (NuScale Proprietary).

   (Methodology for Specifications 3.2.1 – Enthalpy Rise Hot Channel Factor ($F\Delta H$), and 3.2.2 – AXIAL OFFSET (AO).)]

4. [NuScale DCA, Part 2, Tier 2, NuScale FSAR, Section 4.4, “Thermal and Hydraulic Design,“ Revision 1, March 2018 and TR-0516-49416, “Non-Loss-of-Coolant Accident Analysis Methodology,” Revision 0, May 2016 (NuScale Proprietary).

   (Methodology for Specification 3.4.1 – RCS Pressure, Temperature, and Flow Resistance CHF Limits.)]

5. [NuScale DCA, Part 2, Tier 2, NuScale FSAR, Section 4.3, “Nuclear Design,” Revision 1, March 2018.

   (Methodology for Specifications 3.5.3 – Ultimate Heat Sink, and 3.8.1 – Nuclear Instrumentation.)]

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

eRAI No.: 9445
Date of RAI Issue: 05/10/2018

NRC Question No.: 16-44

10 CFR 50.36(c)(1)(i)(A) requires that technical specifications include safety limits for nuclear reactors that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity.

In a letter dated December 8, 2017 (ML17342B343), the applicant updated DCA Part 2, FSAR Chapters 1, 2, and 15, and Part 4, Technical Specifications (TS), with conforming changes to reflect Revision 1 of Licensing Topical Report, NuScale Power Critical Heat Flux Correlations

TR-0116-21012, which adopts a new critical heat flux (CHF) correlation, NSP4. The letter stated in part:

Note that the Technical Specification Safety Limits affected by the implementation of the NSP4 correlation were also modified to relocate the critical heat flux correlation values from the Safety Limit to the Core Operating Limits Report (COLR). The requirement for and contents of the COLR are described in Technical Specification 5.6.3. This relocation is consistent with similar approved Technical Specification changes implemented at the Farley nuclear plant (ML013400451).

The staff reviewed the safety evaluation for Amendment No. 151 to Facility Operating License No. NPF-2 and Amendment No. 143 to Facility Operating License No. NPF-8, for the Joseph M. Farley Nuclear Plant, Units 1 and 2, respectively, which were issued on December 4, 2001. The amendments included a change to plant-specific TS Subsection 2.1.1, "Reactor Core SLs," that moved the curves depicting departure from nucleate boiling (DNB) criterion correlation limits to the Core Operating Limits Report (COLR), a report specified by plant-specific TS Subsection 5.6.5. This relocation was based on Generic Letter (GL) 88-16, "Removal of Cycle-Specific Parameter Limits from TS," dated October 4, 1988 (ML031200485). The Farley amendments were also based on a topical report WCAP-14483-A, approved January 19, 1999 (ML020430092). The NRC staff safety evaluation for WCAP-14483 states, "Safety limits, however, may not be placed in the COLR."

Accordingly, the staff does not accept the proposed relocation of reactor core SL critical heat flux correlation values from GTS SL 2.1.1.1 to the COLR. The applicant is requested to restore these SL values to SL 2.1.1.1 and make conforming changes to the associated Bases in Subsection B 2.1.1 and the list of specifications, which reference the COLR, in Subsection

5.6.3, paragraph a.

NuScale Response:

The requested safety limits have been inserted into Chapter 2 of the Technical Specifications as shown in the attached pages.

Impact on DCA:

The Technical Specifications have been revised as described in the response above and as shown in the markup provided in this response.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODE 1 the critical heat flux ratio shall be maintained at or above the following correlation safety limits:~~≥ the 95/95 critical heat flux ratio criterion for the critical heat flux correlation(s) specified in Section 5.6.3.~~

Correlation	Safety Limit
NSP2	[1.17]
Hench-Levy	[1.06]
NSP4	[1.21]

2.1.1.2 In MODE 1 the peak fuel centerline temperature shall be maintained ≤ 4901 - (1.37E-3 x Burnup, MWD/MTU) °F.

2.1.2 RCS Pressure SL

In MODES 1, 2, and 3 pressurizer pressure shall be maintained ≤ 2285 psia.

2.2 Safety Limit Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 2 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1, restore compliance and be in MODE 2 within 1 hour.

2.2.2.2 In MODE 2 or 3, restore compliance within 5 minutes.

5.6 Reporting Requirements

5.6.3 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

~~2.1.1, "Reactor Core Safety Limits";~~

3.1.1, “SHUTDOWN MARGIN (SDM)”;

3.1.3, “Moderator Temperature Coefficient (MTC)”;

3.1.4, “Rod Group Alignment Limits”;

3.1.5, “Shutdown Group Insertion Limits”;

3.1.6, “Regulating Group Insertion Limits”;

3.1.8, “PHYSICS TESTS Exceptions”;

3.1.9, “Boron Dilution Control”;

3.2.1, “Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)”;

3.2.2, “AXIAL OFFSET (AO)”;

3.4.1, “RCS Pressure, Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits”; and

3.5.3, “Ultimate Heat Sink”.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

[---------------------------------REVIEWER'S NOTE---------------------------------
The COL applicant shall confirm the validity of each listed document and the listed Specifications for the associated core operating limits, or state the valid NRC approved analytical method document and list of associated Specifications.

The COL applicant shall state the valid core reload analysis methodology document and list of associated Specifications.
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