Response to NuScale Technical Report Audit Question

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Question:

In TR-101310-NP, US460 Standard Design Approval Technical Specifications Development, Revision 0, Section 3.0, “Changes to the Content of Standard NuScale Technical Specifications,” what does the following phrase in the section title mean: “Standard NuScale Technical Specifications”?

Response:

NuScale revises Technical Report TR-101310-NP, US460 Standard Design Approval Technical Specifications Development, Section 3.0 title to “Changes to the Content of NuScale US460 Standard Design Approval Technical Specifications.” This change aligns the Section 3.0 title with the title of TR-101310-NP.

Markups of the affected changes, as described in the response, are provided below:

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1.3 Abbreviations

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Table 1-1 Acronyms

Term	Definition
ADAMS	(NRC) Agencywide Documents Access and Management System
BWR	boiling water reactor
CFR	Code of Federal Regulations
COLR	core operating limits report
CRA	control rod assembly
CRDS	control rod drive system
CVCS	chemical and volume control system
DCA	Design Certification Application for the NuScale US600, 12-module plant design
DHRS	decay heat removal system
ECCS	emergency core cooling system
ESB	ECCS supplementary boron
FSAR	Final Safety Analysis Report
GTS	generic technical specifications
IAB	inadvertent actuation block
ISTS	improved standard technical specifications
LCO	limiting condition for~~of~~ operation
LTOP	low temperature overpressure protection
LWR	light water reactor
MCR	main control room
PLI	pressurized line isolation
PWR	pressurized water reactor
RCS	reactor coolant system
RRV	reactor recirculation valve
RTP	rated thermal power
RVV	reactor vent valve
SDA	Standard Design Approval
SR	surveillance requirement
STS	standard technical specifications
TS	technical specifications
T-traveler	technical traveler
UHS	ultimate heat sink

Table 1-2 Definitions

Term	Definition
Decay heat removal system (DHRS) actuation	Decay heat removal system actuation means actuation of the DHRS and includes isolation of the steam and feedwater flow paths outside of the decay heat removal interfaces with the steam generators in accordance with the descriptions provided in the US460 SDA application. This is accomplished by a combination of the module protection system DHRS actuation signal and the secondary system isolation signal.

2.0 Background

2.1 Approach

The determinations required to define the content of the TS are primarily based on the requirements of 10 CFR 50.36 (Reference 5.1.4), 10 CFR 50.36a (Reference 5.1.5), and the discussion in the associated NRC policy (Reference 5.1.6) as described in TR-1116-52011, Technical Specifications Regulatory Conformance and Development (Reference 5.1.3).

Chapters 1, 2, 4, and 5 of the NuScale US600 DCA technical specifications (Reference 5.1.1) and the NuScale US460 SDA TS are generally aligned with the corresponding sections of the legacy plant GTS. This has the advantage of generally aligning the NuScale TS with the NRC requirements and expectations in these areas, and addressing the requirements of 10 CFR 50.36a. It also assists future internal and external communications and interpretations by generally conforming with the expectations and knowledge experience base of plant, industry, and regulatory staff.

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Chapter 3 of the NuScale US600 DCA technical specifications presented a significant set of issues related to application of the criteria for inclusion. To perform the review and identify appropriate limiting condition ~~of~~for operation (LCO) contents, a TS structure that generally parallels the contents in NUREG-1431 and the other pressurized water reactor (PWR) designs is adopted for the proposed NuScale TS, albeit with some significant changes as described in TR-1116-52011 (Reference 5.1.3).

The US460 SDA technical specifications utilize the organization and groupings of LCO requirements adopted in the NuScale US600 DCA TS that have shown to provide clear information to the operating staff. This organization also permits some level of comparison of the NuScale US460 SDA technical specifications to existing large light water reactor (LWR) standard technical specifications when appropriate.

Inclusion of individual Chapter 3 specifications in the NuScale GTS is based on application of the four criteria in 10 CFR 50.36(c)(2)(ii):

1. *Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.*
2. *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.*
3. *A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.*
4. *A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.*

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3.0 Changes to the Content of ~~Standard~~ NuScale US460 Standard Design Approval Technical Specifications

This discussion describes the changes from the NuScale DCA technical specifications that are incorporated into the NuScale US460 SDA technical specifications. The reason for the changes is described in general terms, and includes removals, relocations, and new requirements. Details of design changes and safety analyses are described in the relevant and referenced US460 FSAR sections.

3.1 Modifications to Chapter 1, Use and Application

3.1.1 1.1 Definitions

LEAKAGE

The definition of Pressure Boundary LEAKAGE is modified consistent with the applicable portions of the LEAKAGE definition provided in NUREG-1431, Revision 5 as appropriate for the NuScale design. The change includes modifications to the punctuation used in this definition, consistent with the Writer's Guide for Plant-Specific Improved Technical Specifications, TSTF-GG-05-01, Reference 5.1.7. This change also affects LCO 3.4.5, reactor coolant system (RCS) Operational LEAKAGE and associated Bases and additional details regarding this change are provided in the description of changes to LCO 3.4.5.

MODE

The MODE definition used in the TS is changed to better align with the plant response behavior. Specifically, the upper temperature limit on MODE 3, Safe Shutdown, is removed and the operational region expanded to include temperatures above the minimum temperature for criticality. This is accomplished by including 'and' and 'or' requirements so that above the minimum temperature for criticality the plant is in MODE 3 if it is PASSIVELY COOLED, and in MODE 2 if it is not PASSIVELY COOLED. Below the minimum temperature for criticality, the plant is in MODE 3. Figure 3-1 illustrates this.

This MODE definition clarifies that the plant is in a passively safe configuration once PASSIVE COOLING is established, regardless of the reactor coolant temperature relative to the minimum temperature for criticality.

3.2 Modifications to Chapter 2, Safety Limits

The reactor core critical heat flux correlations and limits, and the RCS pressure safety limits are revised to reflect the increased reactor power and changes to the plant design as described in the FSAR. Surveillance Requirement 3.4.4.1 also modified to reflect new limits.

3.3 Changes to Chapter 3, Limiting Conditions for Operation and Surveillance Requirements

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3.3.1 Modification of Limiting Condition ~~of~~for Operation 3.0.3

The legacy nuclear plant owners have proposed changes to the time provided to initiate a shutdown when LCO 3.0.3 applies. The changes are described in a proposed NRC/industry traveler that is applicable to legacy plant STS. NuScale monitored these efforts in public meetings and believes that a corresponding change is appropriate for incorporation into the NuScale specifications.

Similarly, the Bases for LCO 3.0.3 are being revised to align to the appropriate extent with the proposed change to the legacy plant STS.

3.3.2 Addition of Surveillance Requirement 3.1.9.5 - Verification of Isolation of Module Heatup System from Other Modules

This surveillance requirement is added to verify inter-module alignment exists to prevent interactions that could affect boration of the RCS.

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3.3.3 Modification of Limiting Condition ~~of~~for Operation 3.2.1, Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$), and 3.2.2, Axial Offset

This change expanded Applicability of LCO 3.2.1 and 3.2.2 to require the LCOs limits be met at or above 20 percent RTP, rather than 25 percent RTP, expanding the applicability of the requirements to a larger region of the operating power levels.

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3.3.4 Changes to Limiting Condition ~~of~~for Operation 3.3.1, Module Protection System

Modifications are made to actuation logic to align with safety analyses and design changes consistent with the increased reactor rated thermal power, safety analyses, refinements in operational intentions, and lessons learned since the submittal of the DCA technical specifications. Changes include editorial renumbering and arrangement of functions.

Modification of Emergency Core Cooling System Actuation Signals

The design of the ECCS and the associated actuation signals have been changed. These changes to the ECCS components are described in FSAR Chapter 7, Section a single CRA. This change is necessary to allow energization of a portion of the control rod drive system (CRDS) in MODE 3 when preparing for module disassembly. This capability is required to verify the CRA is disconnected from its extension rod. The CRDS design and administrative control ensure that no more than one CRA may be manipulated, and the definition of, and limits on SHUTDOWN MARGIN specified in LCO 3.1.1 ensure the plant remains safely shutdown. A description of the functional design of the CRDS is provided in FSAR Section 4.6. This change results in changes to the Bases discussions for the associated Functions.

Footnotes limiting the Applicability of the Demineralized Water System Isolation are added to only require OPERABILITY when RCS temperature is above the T-3 interlock. The RCS T-3 bypass is active when RCS temperature is less than approximately 340 degrees F. This change also resulted in addition of a footnote to Table 3.3.3-1, Engineered Safety Features Actuation System Logic and Actuation Functions.

Some other footnotes required modification because of the combination of allowances described above.

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3.3.5 Addition of Limiting Condition ~~of~~for Operation 3.3.3 Condition for Pressurizer Line Isolation Inoperability

The description of Condition F in LCO 3.3.3 is modified to address circumstances when both divisions of the pressurizer line isolation (PLI) function are inoperable. The PLI signal closes a subset of the chemical and volume control system (CVCS) isolation valves so the revised Condition and existing Required Actions ensure the safety function is met. The revised Required Action requires closure of affected valves. If the PLI actuation logic is inoperable then the pressurizer spray and high point vent lines must be isolated. If the CVCS actuation logic is inoperable then all four CVCS lines must be isolated.

3.3.6 Addition of Surveillance Requirement 3.3.3.3, Monitoring Emergency Core Cooling System Actuation Time Delay

The delayed ECCS actuation signal added to LCO 3.3.1 is implemented by time delays established in the module protection system logic. Surveillance Requirement 3.3.3.3 is added to ensure the time delays is within limits. The time delay limits are core cycle-specific depending on fuel makeup so the time delay limits are specified in the core operating limits report (COLR). The baseline delay is approximately 8 hours. The surveillance frequency is in accordance with the Surveillance Frequency Control Program, with an initial interval of 24 months.

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3.3.7 Removal of Limiting Condition ~~of~~for Operation 3.3.5, Remote Shutdown Station

The DCA technical specifications LCO 3.3.5 is removed from the US460 SDA technical specifications. The Design Certification Application LCO was included in accordance with Criterion 4 of 10 CFR 50.36(c)(2)(ii); however further consideration during the development of the US460 SDA design resulted in concluding that it is no longer appropriate for inclusion. This conclusion is based on the system design and details provided in US460 FSAR Chapters 7 and 18.

Chapter 7 of the FSAR describes the capability of the plant design to respond in the event of a fire in the main control room (MCR). As described there, in the event of a fire in the MCR the operators trip the reactors, initiate decay heat removal and initiate containment isolation before evacuating the MCR. These actions result in passive cooling that achieves and maintains the modules in a safe shutdown condition.

Operators can also place the reactors in safe shutdown from outside the MCR in the module protection system equipment rooms within the Reactor Building.

The operators then use alternate operator workstations to monitor plant conditions. Following shutdown and initiation of passive cooling, the design does not rely on operator action, instrumentation, or controls outside of the MCR to maintain a safe stable shutdown condition.

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3.3.8 Modification of Limiting Condition ~~of~~for Operation 3.4.2, Minimum Temperature for Criticality

The minimum temperature for criticality is reduced to 345 degrees F to improve the ability of the reactor to startup in a timely manner after an outage by using nuclear heat to increase temperatures to the normal operating range. Safety analyses include consideration of this new limit.

3.3.9 Modification of Surveillance Requirement 3.4.4.1 - Reactor Safety Valve Setpoints

Reactor safety valve setpoints are changed to reflect increase reactor power, reactor vessel design pressure, and associated safety analyses.

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3.3.10 Editorial Clarification of Condition D of Limiting Condition ~~of~~for Operation 3.4.3, Reactor Coolant System Pressure / Temperature Limits

Condition D is clarified to specifically address initiation of containment flooding and to modify the Required Action to immediately initiate action to be in MODE 2. This change more accurately reflects plant operations and more closely aligns with similar Required Actions used in similar circumstances requiring immediate actions be taken.

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3.3.11 Modification of Limiting Condition ~~of~~for Operation 3.4.5, Reactor Coolant System Operational Leakage

The definition of LEAKAGE and LCO 3.4.5 are revised based on changes to large legacy PWR standard technical specifications issued as Rev 5; however modified to reflect the design and operation of a NuScale plant. The change clarifies the requirements for pressure boundary leakage such as could be postulated to exist on RCS piping outside the containment before the outermost containment isolation valves. The FSAR Section 5.2, Integrity of Reactor Coolant Boundary describes these lines up to the outermost containment isolation valves as part of the reactor coolant pressure boundary.

The change adds a new Condition requiring action to isolate the affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and deactivated automatic valve, blind flange, or check valve within four hours. Subsequent Conditions and Required Actions are renumbered.

The Bases for LCO 3.4.5 are also revised based on changes to large legacy PWR standard technical specifications with changes to reflect the NuScale design and operations.

Maintaining alignment with large LWR technical specifications to the extent appropriate for the design promotes understanding and interpretation of TS requirements during internal and external communications between plant and regulatory staff.

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3.3.12 Modification of Limiting Condition ~~of~~for Operation 3.4.8, Reactor Coolant System Specific Activity

Limits on I-131 and Xe-133 are modified to reflect safety analyses for the new reactor design, including the increased reactor power level compared to the US600 design. The US460 FSAR Section 11.1.3 describes the realistic source term used to develop the source terms used as described in FSAR Section 15.0.3 to describe consequence analyses of design basis events, including the concentration limits in US460 Standard Design Approval LCO 3.4.8. The FSAR Table 11.1-2 describes parameters used to calculate coolant source terms, including the reactor core thermal power.

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3.3.13 Modification of Limiting Condition ~~of~~for Operation 3.4.10, Low Temperature Overpressure Protection Valves

This LCO is modified to reflect the revised ECCS design that uses two RVVs. One RVV is adequate to provide low temperature overpressure protection as described in the FSAR.

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3.3.14 Modification of Limiting Condition ~~of~~for Operation 3.5.1, Emergency Core Cooling System

This LCO is modified to reflect the revised ECCS design that only uses two RVVs, and removes the inadvertent actuation block function from the RVVs. Surveillance Requirement 3.5.1.3 is also modified to reflect the removal of the inadvertent actuation block function from the RVVs.

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3.3.15 Modification of Limiting Condition ~~of~~for Operation 3.5.3, Ultimate Heat Sink

The ultimate heat sink (UHS) is redesigned consistent with the other changes to the plant design and analyses, primarily the increased RTP and a reduction in the number of reactors in the design to a maximum of six modules. The redesign resulted in reanalysis and redefinition of the UHS and caused changes in the credited functions of the UHS in LCO 3.5.3.

The UHS water level requirements are specified to a new band defined by upper and lower limits that improve containment heat removal behavior. The redesigned UHS and its functions are described in FSAR Section 9.2.5. The new limits are consistent with the safety analyses in the FSAR that credit the UHS function. Similarly, the maximum bulk average pool temperature is increased to align with the safety analyses assumptions. The structure of the Actions in LCO 3.5.3 are changed to reflect the removal of distinct limits that the DCA credited for separate safety functions. This change removed the need for Condition B of the DCA technical specifications, which is now addressed by Condition A. Completion Times remain consistent with the credited functions of the UHS. Subsequent Conditions are renumbered. Corresponding changes are made to the Bases.

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3.3.16 Addition of Limiting Condition ~~of~~for Operation 3.5.4, Emergency Core Cooling System Supplemental Boron

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The US460 design adds a passive system that provides soluble boron in dissolvers mounted inside the containment. The dissolvers provide a reservoir of boron that mixes with condensate from the upper inner surfaces of the containment vessel when the ECCS is actuated. Limiting Condition ~~of~~for Operation 3.5.4 is added to ensure that the quantity of boron available for dissolution when the ECCS actuates conforms to the assumptions in the safety analyses. The boron ensures the reactor remains subcritical after certain events in combination with limiting conditions, and subsequent cooldown of the reactor system. The quantity of boron required is specified in the COLR. The ESB satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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3.3.17 Addition of Limiting Condition ~~of~~for Operation 3.6.3, Containment Closure

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Limiting Condition ~~of~~for Operation 3.6.3 is added to ensure that module inventory is preserved during movement of the module between the operating location and the containment closure tool. The LCO requires a module that is in MODE 4, with the upper module assembly seated on the lower containment vessel flange, be maintained closed. The LCO and allowances are patterned on portions of NUREG-1431, LCO 3.9.4 with extensive modifications to align with the NuScale application. Maintaining containment closure ensures that the decay heat removal mechanism required to assure core cooling is maintained during periods when the module is isolated from other systems such as CVCS, or when the containment is disassembled from the UHS via the de-energized ECCS valves. Limiting Condition ~~of~~for Operation 3.6.3 satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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3.3.18 Removal of Limiting Condition ~~of~~for Operation 3.7.3, In-Containment Secondary Piping Leakage

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Limiting Condition ~~of~~for Operation 3.7.3 is deleted as no longer necessary because the break exclusion design criteria is applied to the secondary system piping within the containment. The DCA design for secondary system piping met the leak-before-break design criteria of General Design Criteria 4.

US460 Standard Design Approval FSAR Section 3.6 describes the application of design measures to prevent or mitigate postulated dynamic effects associated with postulated rupture of US460 piping. The US460 SDA design of secondary piping inside the containment meets the criteria for exclusion from postulated breaks and cracks provided in NRC Branch Technical Position (BTP) 3-4. Based on this change the US600 Design Certification Application LCO is no longer needed because the piping is excluded from consideration of postulated breaks and cracks.

3.3.19 Other Bases Changes

In addition to the specific changes described above, Applicable Safety Analyses sections are modified to reflect changes to the safety analyses, primarily as a result of the increased reactor power. Other changes are made in response to operational analysis feedback to clarify and ease understanding of the requirements.

3.4 Chapter 4, Design Features

Section 4.3 Fuel Storage

The fuel storage design description is modified to reflect changes to the design and analyses. Key variables are bracketed to allow replacement with actual plant-specific values when design details are finalized by a future applicant that references the NuScale power plant US460 standard design. NuScale is monitoring industry efforts to relocate fuel storage detailed requirements to a COLR-like document and anticipates adopting this practice when the concept matures.

3.5 Chapter 5, Administrative Controls

Section 5.2.2, Facility Staff

This section is modified to reflect approved topical report TR-0420-69456, “NuScale Control Room Staffing Plan,” TR-0420-69456, Revision 1-A.