May 20, 2020 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 Submittal of "Technical Specifications Regulatory Conformance and Development," TR-1116-52011, Revision 4

REFERENCE: Letter from NuScale Power to NRC, "NuScale Power, LLC Submittal of 'Technical Specifications Regulatory Conformance and Development,' TR-1116-52011, Revision 3," dated November 15, 2019 (ML19319C787)

NuScale Power, LLC (NuScale) hereby submits Revision 4 of "Technical Specifications Regulatory Conformance and Development" (TR-1116-52011).

The enclosure to this letter contains the nonproprietary report entitled "Technical Specifications Regulatory Conformance and Development."

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Rebecca Norris at 541-602-1260 or at RNorris@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC
Prosanta Chowdhury, NRC
Michael Dudek, NRC
Michael Snodderly, NRC
Christiana Liu, NRC
Christopher Brown, NRC
Marieliz Johnson, NRC
Getachew Tesfaye, NRC

Enclosure: "Technical Specifications Regulatory Conformance and Development," TR-1116-52011, Revision 4

Enclosure:

"Technical Specifications Regulatory Conformance and Development," TR-1116-52011, Revision 4

Licensing Technical Report

Technical Specifications

Regulatory Conformance and Development

May 2020

Revision 4

Docket: 52-048

NuScale Nonproprietary

NuScale Power, LLC

1100 NE Circle Blvd., Suite 200

Corvallis, Oregon 97330

www.nuscalepower.com

Licensing Technical Report

Department of Energy Acknowledgement and Disclaimer

This material is based upon work supported by the Department of Energy under Award Number DE-NE0008928.

Licensing Technical Report

CONTENTS

TABLES

FIGURES

Abstract

This report describes the development process of the NuScale Power Plant technical specifications (TS) to conform with regulatory requirements and expectations regarding scope, content, and format. This report also provides the basis for including the requirements chosen for the NuScale TS.

Revisions 1, 2, and 3 were previously prepared to clarify the development process and changes made to the technical specifications resulting from NRC requests for additional information. Additional changes are reflected that occurred as the result of changes to the FSAR and plant design that are now reflected in the technical specifications.

1.0 Introduction

1.1 Purpose

The purpose of this report is to describe the development process of the NuScale Power Plant technical specifications (TS) to conform with the applicable regulatory requirements and expectations regarding scope, content, and format. Revision 2 of this report includes consideration of changes made after revision 1 of this report and reflects changes in contents of the technical specifications through July 15, 2019. This report also provides the basis for including the specifications chosen for the NuScale TS. The report focuses on three aspects:

- Conformance with 10 CFR 50.36 (Reference 6.2.1) and 10 CFR 50.36a (Reference 6.2.2), including considerations in 58 FR 39132, *Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors* (Reference 6.2.11).
- Conformance with regulatory expectations as expressed by industry standards and precedent as defined in NRC-published standard technical specifications (STS), approved generic technical specifications (GTS), and subsequent changes as delineated in industry / NRC STS travelers.
- Conformance with the technical specification format and content guidance established by TSTF-GG-05-01, Revision 1, *Writer's Guide for Plant Specific Improved Technical Specifications,* August 2010 (Reference 6.2.4).

The report also describes additional changes to the technical specifications and bases that resulted from

- NRC requests for additional information (RAI),
- Industry/NRC traveler status changes since December 2016, and
- design or regulatory changes implemented in the FSAR that result in changes that need to be reflected in the technical specifications.

1.2 Scope

This report addresses the development of the GTS applicable to an individual NuScale module. The NuScale GTS are drafted in the context of the design certification application for a 12-module NuScale facility, however the content is applicable to an individual module.

1.3 Abbreviations

Table 1-1 Acronyms

Term	**Definition**
AP1000	Westinghouse AP1000 (as described in 10 CFR 52, Appendix D)
B&W	Babcock and Wilcox
BWR	boiling water reactor
CE	Combustion Engineering
CFDS	containment flooding and drain system
CHF	critical heat flux
CNV	containment vessel
COL	combined license
COLR	core operating limits report
CRA	control rod assembly
CVCS	chemical and volume control system
DCA	Design Certification Application
DHRS	decay heat removal system
ECCS	emergency core cooling system
ESBWR	GE Hitachi Economic Simplified Boiling Water Reactor (as described in 10 CFR 52, Appendix E)
ESF	engineered safety feature
ESFAS	engineered safety features actuation system
FSAR	Final Safety Analysis Report
GDC	General Design Criterion
GE	General Electric
GTS	generic technical specifications
HELB	high-energy line break
HFP	hot full power
HZP	hot zero power
ISTS	Improved Standard Technical Specifications
LCO	limiting condition of operation
LOCA	loss-of-coolant accident
LTOP	low temperature overpressure protection
MPS	module protection system
MSIV	main steam isolation valve (typically includes associated drain and bypass valves)
NPM	NuScale Power Module
ODCM	Offsite Dose Calculation Manual
PTLR	Pressure Temperature Limits Report
PWR	pressurized water reactor

RAI	request for additional information (from NRC)
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RTP	rated thermal power
RTS	reactor trip system
SDM	shutdown margin
SG	steam generator
SL	safety limit
SP	Setpoint Program
SR	surveillance requirement
SSC	structures, systems, and components
SSI	Secondary System Isolation
STS	standard technical specifications
TS	technical specifications
TSTF	Technical Specification Task Force
UHS	ultimate heat sink
W	Westinghouse

Table 1-2 Definitions

Term	**Definition**
Decay heat removal system (DHRS) actuation	DHRS 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 (SGs) in accordance with the descriptions provided in the Design Certification Application (DCA). This is accomplished by a combination of the module protection system DHRS actuation signal and the secondary system isolation (SSI) signal.
Emergency core cooling system (ECCS) actuation	ECCS actuation describes the signal which permits the ECCS valves (reactor vent valves and reactor recirculation valves) to open. The valves may not immediately open in response to actuation depending on the function of the pressure interlock feature that compares reactor coolant pressure with the pressure in the containment in accordance with the descriptions provided in the DCA.
k_{eff}	effective neutron multiplication factor, k_{eff}=1 is a critical configuration

2.0 Background

Historically, the NRC approved TS evolved from plant-specific custom TS, which evolved into the 'old' standard TS, and more recently to the existing improved standard TS.

The current standard technical specifications (STS) are published by the NRC as NUREGs 1430 – 1434 to address Combustion Engineering (CE), Westinghouse (W), Babcock and Wilcox (B&W), and General Electric (GE) reactor designs (References 6.2.5 through 6.2.9). Additionally, the NRC issued NUREG-2194 as the STS for the AP1000 certified design (Reference 6.2.10). The format and content of the TS has evolved as facility operations and designs were refined and operating experience was gained.

The NRC initially developed the STS based on criteria in an interim NRC policy published in 1987 (Reference 6.2.3) that was later refined and published as 58 FR 39132, *Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors,* on July 22, 1993 (Reference 6.2.11). The policy resulted in changes to ensure availability of safety systems and functions assumed or that mitigate design basis accidents, while minimizing technical specification content that reduced the focus of the plant operating staff on safety.

The transition to the published STS was preceded by the development and issuance of a report (commonly referred to as the Split Report) by the NRC on May 9, 1988 (Reference 6.2.12) that applied the NRC policy and defined the appropriate content for TS. The Split Report considered nuclear steam supply system vendor and owners group submittals that applied the interim policy inclusion criteria to the vendor-specific old standard TS. The Split Report explicitly identified TS content that should be included in the new STS and identified TS content that could be relocated to other licensee-controlled documents. Being based on previously existing, old STS, the contents of the new STS that were retained evolved from the content and consideration of the historical TS.

The NuScale Power Plant is a unique integral pressurized water reactor design for which existing STS are not applicable and for which representative TS have not been previously issued. The plant design, safety functions, structures, systems, and components (SSC), and behavior are significantly different from those in previously licensed designs and facilities. Integrated and simplified behavior reduces the scope of safety systems and combines safety functions into a smaller set of SSC whose operability are assumed as initial conditions or credited to respond in the safety analyses.

Because of this, NuScale does not have a historical basis of existing TS, nor design-specific operating experience to inform the content of the TS. Rather, the NuScale TS are developed directly from the design and planned operations, and are informed by industry operating experience and TS content for which similar or parallel functions and features exist in other designs. These factors are then compared with the criteria for inclusion as TS.

This report describes the consideration of the NuScale design and operations, and applies the TS inclusion criteria in 10 CFR 50.36 and 10 CFR 50.36a, consistent with

- the final NRC policy,
- considerations detailed in the Split Report,
- the Writer's Guide for Plant-Specific Improved Technical Specifications,
- the content of the current versions of the STS, and
- the refinements to the STS developed by the industry / NRC travelers through June 30, 2019 to the extent they are available and applicable.

This report also describes the changes made to the technical specifications as a result of NuScale responses to RAIs. In some cases, NuScale included additional content that does not strictly align with the above requirements or industry precedent regarding technical specification content.

2.1 Approach

The determinations required to define the content of the TS are primarily based on the requirements of 10 CFR 50.36, 10 CFR 50.36a, and the discussion in the associated NRC policy and statements of consideration.

Chapters 1, 2, 4, and 5 of the STS generally provide parallels that are applied to define corresponding NuScale GTS content. This has the advantage of generally aligning the NuScale GTS with the NRC requirements and expectations in these areas, and addressing the requirements of 10 CFR 50.36a.

Chapter 3 of the TS presented a significant set of issues related to application of the criteria for inclusion. To perform the review and identify appropriate limiting condition of operation (LCO) contents, a TS structure that generally parallels the contents in NUREG-1431 and the other pressurized water reactor (PWR) designs was adopted for the proposed NuScale TS, albeit with some significant changes. The existing organization and groupings of requirements in the STS have been refined since their initial issuance, and have been demonstrated to provide clear information to the operating staff. This organization also permits some level of comparison of the NuScale TS to existing STS 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.*

Section 3 of this report describes the assessment of each TS chapter and its incorporation into the NuScale GTS.

2.2 Regulatory Requirements

10 CFR 52.47 Paragraph (11) requires applicants for standard design certifications to provide proposed TS prepared in accordance with the requirements of 10 CFR 50.36 and 10 CFR 50.36a.

10 CFR 50.36 describes requirement for and the content to be included in the TS.

10 CFR 50.36a requires applicants for a design certification to include technical specifications that address applicable provisions of 10 CFR 20.1301, and procedures related to the control of effluents and radioactive waste systems.

3.0 Content of NuScale Generic Technical Specifications

The NuScale design includes up to 12 NuScale Power Modules (NPMs) in a single large reactor pool as described in the Final Safety Analysis Report (FSAR). Each NPM is an independently operated, nominal 160 MWt pressurized water nuclear reactor that supplies steam to an associated turbine generator and condenser to produce electrical power. The reactor pool provides an essential shared function as the ultimate heat sink (UHS) for the passive core cooling systems of each of the 12 individual NPMs.

Significantly, the plant and NPM design precludes the need for electrical power to safely shut down or remove residual heat from the NPMs if a design basis loss of power occurs.

The 12 NPMs share a single main control room that incorporates individual digital instrumentation and control systems for each NPM.

A major difference from existing plant designs is that individual NPMs are disconnected from their operating systems, instrumentation, and controls and moved to a separate area of the Reactor Building for refueling operations and maintenance.

More detailed information regarding the design and operation of the NuScale plant is provided in the DCA.

Based on the unique nature of the NuScale design, the proposed NuScale GTS are structured similar to the legacy large light water reactors and the AP1000 STS, but with significant differences. Table 3-1 provides a comparison of the structure of the legacy large light water reactor STS and AP1000 STS with the NuScale GTS at the chapter and section levels.

Table 3-1 Comparison of standard technical specifications and the proposed NuScale generic technical specifications

<table>
<tr><th rowspan="2">Chapter/ Section</th><th colspan="4">Rev. 4 STS NUREG-</th><th rowspan="2">AP1000 STS NUREG-2194</th><th rowspan="2">NuScale GTS</th></tr>
<tr><th>1430</th><th>1431</th><th>1432</th><th>1433 and 1434</th></tr>
<tr><td>1.0</td><td colspan="6">USE AND APPLICATION</td></tr>
<tr><td>2.0</td><td colspan="6">SAFETY LIMITS (SLs)</td></tr>
<tr><td>2.1</td><td colspan="6">SLs</td></tr>
<tr><td>2.2</td><td colspan="6">SAFETY LIMIT VIOLATIONS</td></tr>
<tr><td>3.0</td><td colspan="6">LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY
SURVEILLANCE REQUIREMENT (SR) APPLICABILITY</td></tr>
<tr><td>3.1</td><td colspan="6">REACTIVITY CONTROL SYSTEMS</td></tr>
<tr><td>3.2</td><td colspan="6">POWER DISTRIBUTION LIMITS</td></tr>
<tr><td>3.3</td><td colspan="6">INSTRUMENTATION</td></tr>
<tr><td>3.4</td><td colspan="6">REACTOR COOLANT SYSTEM (RCS)</td></tr>
<tr><td>3.5</td><td colspan="3">EMERGENCY CORE COOLING SYSTEM (ECCS)</td><td>ECCS AND REACTOR CORE ISOLATION COOLING SYSTEM (RCIC)</td><td colspan="2">PASSIVE CORE COOLING SYSTEMS</td></tr>
<tr><td>3.6</td><td colspan="6">CONTAINMENT SYSTEMS</td></tr>
<tr><td>3.7</td><td colspan="6">PLANT SYSTEMS</td></tr>
<tr><td>3.8</td><td colspan="5">ELECTRICAL POWER SYSTEMS</td><td>REFUELING OPERATIONS</td></tr>
<tr><td>3.9</td><td colspan="5">REFUELING OPERATIONS</td><td>Relocated to 3.8</td></tr>
<tr><td>3.10</td><td colspan="3">(not used)</td><td>SPECIAL OPERATIONS</td><td colspan="2">(not used)</td></tr>
<tr><td>4.0</td><td colspan="6">DESIGN FEATURES</td></tr>
<tr><td>4.1</td><td colspan="6">Site Location</td></tr>
<tr><td>4.2</td><td colspan="6">Reactor Core</td></tr>
<tr><td>4.3</td><td colspan="6">Fuel Storage</td></tr>
<tr><td>5.0</td><td colspan="6">ADMINISTRATIVE CONTROLS</td></tr>
<tr><td>5.1</td><td colspan="6">Responsibility</td></tr>
<tr><td>5.2</td><td colspan="6">Organization</td></tr>
<tr><td>5.3</td><td colspan="6">Facility Staff Qualifications</td></tr>
<tr><td>5.4</td><td colspan="6">Procedures</td></tr>
<tr><td>5.5</td><td colspan="6">Programs and Manuals</td></tr>
<tr><td>5.6</td><td colspan="6">Reporting Requirements</td></tr>
<tr><td>5.7</td><td colspan="6">High Radiation Area</td></tr>
</table>

The NuScale GTS adopted the structure and relevant content from Chapters 1, 2, 4, and 5 in recognition of the commonality of this content. Changes proposed to the “Use and Application,” and “Applicability” sections were made to reflect the contents of the NuScale GTS. Where the STS Chapter 1 describes a situation that is not consistent with the NuScale plant design and specifications, the NuScale GTS are modified to reflect the NuScale use or excluded, as appropriate.

NuScale GTS Sections 3.1 through 3.8 were drafted based largely on the NUREG-2194 and NUREG-1431 specifications; however other STS and the Economic Simplified Boiling Water Reactor GTS (Reference 6.2.13) were also used during development. Individual specifications that do not have a parallel SSC or function in the NuScale Power Plant design were omitted. Similarly, NuScale Power Plant SSC or functions that initially screened to meet one or more of the criteria in 10 CFR 50.36(c)(ii), but that are not represented by similar SSC or functions in the NUREGs, were prepared and included. The integrated PWR nature of the NuScale Power Plant, SSC, and control systems resulted in consolidation of some specifications that were formerly defined individually.

The 12-NPM NuScale design and operating paradigm resulted in changes to conform with the STS-like document construction, specifically those related to the relocation of individual NPMs during refueling.

Sections 3.1 through 3.5 of this report outline the NuScale GTS chapter-level structure and discuss the associated content.

3.1 Chapter 1, Use and Application

With the exception of the MODE definitions and certain definitions related to the design of individual SSC, the NuScale GTS generally adopt the structure and relevant content from Chapters 1 of the STS. A significant change was the replacement of definitions for response times due to variation from the industry standard meaning and significant differences in the way safety system response times are measured. The digital safety system response time is continuously monitored while in service and evaluated for consideration by conservative calculations as described in TR-1015-18653-P-A, “Design of the Highly Integrated Protection System Platform." Sensor response times are measured from the sensor through input to the digital systems. Actuation response times are measured from the output of the digital system until the associated system or component has performed its safety function. Overall function response times are evaluated for acceptability based on the sum of the sensor, digital, and actuation response times.

Changes proposed to the “Use and Application” sections are limited to those necessary to remain consistent with the balance of the NuScale GTS. For example Section 1.3 refers to a credited pump in the STS, however the NuScale design does not credit any pumps and the example was re-drafted to use a credited valve to illustrate the condition.

MODES

The MODE definitions applicable to PWRs were determined to be inconsistent with NuScale design and operation. Individual NPMs use a comparatively small reactor that depends on natural circulation for flow in the reactor, NPMs are passively cooled in postulated accident conditions, and the design includes relocation of NPMs to perform refueling operations. Therefore, NuScale developed a new MODE structure that more appropriately addresses the NuScale operations paradigm. The NuScale GTS MODES are described in Table 3-2.

Table 3-2 NuScale technical specification MODES

MODE	TITLE	REACTIVITY CONDITION (k_{eff})	INDICATED REACTOR COOLANT TEMPERATURES (°F)
1	Operations	≥ 0.99	All ≥ 420
2	Hot Shutdown	< 0.99	Any ≥ 420
3	Safe Shutdown[a]	< 0.99	All < 420
4	Transition[b][c]	< 0.95	N/A
5	Refueling[d]	N/A	N/A

(a) Any CRA capable of withdrawal, any CVCS or CFDS connection to the module not isolated.
(b) All CRAs incapable of withdrawal, CVCS and CFDS connections to the module isolated, and all reactor vent valves (RVV) de-energized.
(c) All reactor vessel flange bolts fully tensioned.
(d) One or more reactor vessel flange bolts less than fully tensioned.

Descriptions and Rationale for NuScale MODE Structure

MODE 1 – Operations

This MODE replaces both MODE 1 Power Operation and MODE 2 Startup used in pressurized water reactor STS. This MODE is defined by the reactivity condition (k_{eff}) being greater than or equal to 0.99, i.e., approaching criticality or in critical operation. The NuScale design uses an external heat source to raise temperatures above the minimum required for criticality. Once critical, the design initially functions somewhat like a boiling water reactor design, using nuclear heat to increase temperatures during power ascension from initial criticality to full power temperatures at approximately 15 percent of rated thermal power (RTP). The minimum temperature for criticality is included with a requirement that all indicated reactor coolant temperatures be ≥ 420 degrees F. The temperature requirements are specified in this manner because dependence on natural circulation makes identification of highest and lowest coolant temperatures difficult during low-flow conditions. This definition ensures consistency with design and accident analyses.

The NuScale reactor operates at comparatively lower power levels and power densities, yet with a fuel design similar to a large PWR. Individual specifications that require a power restriction are explicitly described in their Applicability rather than by using a distinct Startup MODE as found in the PWR STS. Therefore, there is no need for a distinct Startup MODE.

MODE 2 – Hot Shutdown

MODE 2 is defined by reactivity condition (k_{eff}) less than 0.99 and any indicated reactor coolant temperature at or above 420 degrees F. This MODE corresponds to the transition from safe shutdown toward critical operations, and to conditions immediately after a reactor trip and during intial cooldown. While in this MODE during startup, the NPM has been reconfigured from passive cooling to permit heatup toward operations. The containment vessel (CNV) is drained, and the ECCS and decay heat removal system (DHRS) are aligned to operate if needed.

MODE 3 – Safe Shutdown

This MODE is defined by reactivity condition (k_{eff}) less than 0.99 and all indicated reactor coolant temperatures less than 420 degrees F. In this MODE, the NPM is in a safe, stable state. Passive decay heat removal is by conduction and convective heat transfer through the CNV walls to the reactor pool. The containment is or will be flooded to a level that ensures passive cooling of the reactor. During unit startup, re-alignment and configuration changes to increase temperatures and enter Hot Shutdown occur in MODE 3.

Graphically, MODES 1, 2, and 3 interact as shown below.

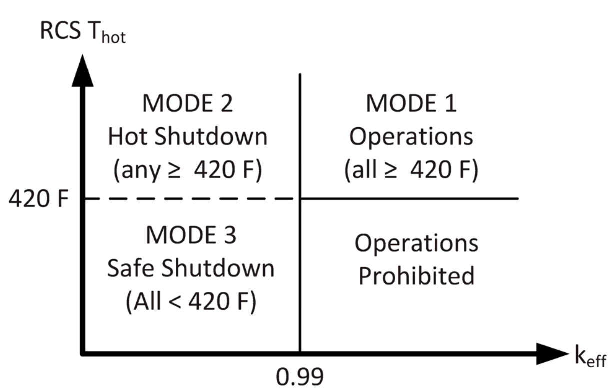

Figure 3-1 MODES Temperature and Reactivity Conditions

MODE 4 – Transition

The relocation of an NPM from its operating position to the refueling area results in definition of a MODE that addresses this transitional configuration. The module remains partially submerged in the reactor pool throughout movement in this MODE. The MODE governs the period from isolation and disconnection of reactivity controls in the operating position, until the first reactor vessel flange bolt is detensioned in preparation for further disassembly and fuel movement. This MODE includes relocation of the NPM from the operating position in the reactor pool to the containment vessel flange tool and the reactor vessel flange tool for reactor vessel disassembly. The MODE also governs the process from re-tensioning the reactor vessel bolts through reassembly of the CNV and return to the operating position.

Entry into MODE 4 from MODE 3 requires verification of reactivity condition (k_{eff}) less than 0.95 and compliance with footnote (b) of Table 3-2, which prevents reactivity changes during movement and ensures low temperature overpressure events cannot occur. Overpressure protection in MODE 4 is provided by disconnected, de-energized and therefore open, ECCS reactor vessel vent valves.

The subcriticality and safety of the NPM in this MODE is ensured by the inability to alter the reactivity condition of the core, prevention of overpressure conditions, and the passive cooling of the NPM.

MODE 5 – Refueling

This MODE is similar to Refueling in the STS and addresses the conditions during which the reactor pressure vessel (RPV) is not intact, including during movement of the reactor fuel in and around the reactor core in the refueling tool. The upper portion of the CNV and upper portion of the reactor vessel are removed and typically located away from the reactor core during this MODE. The reactor core and lower reactor vessel remain fully submerged in the reactor pool during operations in this MODE. Decay heat removal and core reactivity are ensured by submersion in the reactor pool in this MODE.

3.2 Chapter 2, Safety Limits

The structure and content of Chapter 2 of the NuScale GTS closely align with Chapter 2 content of existing reactors. Reactor core safety limits are established to protect the integrity of the reactor coolant system (RCS) and the reactor fuel cladding, which are the two principal physical barriers that guard against the uncontrolled release of radioactivity.

The combination of THERMAL POWER, Reactor Coolant System (RCS) hot temperature (T_{hot}), pressurizer pressure specified in the Core Operating Limits Report (COLR), and safety limits (SLs) on the critical heat flux (CHF) ratio and peak fuel centerline temperature prevent overheating of the fuel and clad, as well as possible cladding perforation that would result in the release of fission products to the reactor coolant. These variables were chosen to best address the conditions and variables in the NuScale design that uses natural circulation to maintain flow through the reactor core.

The SL on RCS pressure protects the integrity of the RCS from overpressurization that could result in a breach of the reactor coolant pressure boundary (RCPB). If such a breach occured in conjunction with a fuel cladding failure, fission products could enter the containment, raising concerns about radioactive releases.

3.3 Chapter 3, Limiting Conditions for Operation and Surveillance Requirements

This chapter of the NuScale GTS reflects and addresses the NuScale design and operating paradigm. The integrated, passive, modular NuScale design is reflected in the chapter. As stated in 10 CFR 50.36(c)(2)(i) the limiting conditions for operation (LCO) are the lowest functional capability or performance levels of equipment required for safe operation of the facility. Additionally 10 CFR 50.36(c)(2)(ii) provides four criteria to be applied to specify LCO.

The criteria of 10 CFR 50.36(c)(2)(i) and 10 CFR 50.36(c)(2)(ii) were applied as follows.

<u>Criterion 1</u>

Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Consistent with PWR designs, instrumentation is provided to detect significant abnormal leakage from the RCPB. Specification 3.4.7 specifies operability and surveillance requirements for instrumentation to detect leakage and provide indication in the control room of a significant abnormal degradation of the RCPB.

<u>Criterion 2</u>

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.

As a new design, the NuScale LCO were developed applying the criteria of 10 CFR 50.36(c)(2)(i) and 10 CFR 50.36(c)(2)(ii), consistent with the standard technical specifications and existing plant-specific technical specifications. Safety analyses are performed as described in Chapter 15 and elsewhere in the FSAR.

In general, normal operating conditions are described in relevant portions of the FSAR analyses or system operational descriptions. LCO are not appropriate for many of these conditions as they do not satisfy the criteria of 10 CFR 50.36(c)(2)(i) in that they are not the lowest functional capability or performance level of equipment required for safe operation. Consistent with the Commission final policy on technical specification improvements (58 FR 39132) control over those normal operating conditions are established by the facility operating license by reference to the FSAR. 10 CFR 50.71 maintains the description of the facility consistent with the FSAR. Any proposed changes thereto are subject to 10 CFR 50.59 review and require NRC approval before implementation of the change if an unreviewed safety question is identified.

In other cases, the operating limit is an initial condition of a design basis accident or transient analysis that satisfies the criteria of both 10 CFR 50.36(c)(2)(i) and 10 CFR 50.36(c)(2)(ii). This results in the need for an LCO to ensure the initial condition is met.

As an example, the pressurizer pressure normal operating condition range is established at about 1850 psia, with a normal range from 1780 to 1920 psia (values are illustrative only – not for use.) Operating a NuScale plant with the pressurizer pressure slightly above or below this normal operating range does not exceed a 'lowest functional capability or performance level required for safe operation.' Operation outside the normal operating range would be inconsistent with the description in the FSAR. Operating outside these limits would require corrective actions to be taken. If the normal operating range was to be changed then a 10 CFR 50.59 evaluation and modification of the FSAR for continued operation outside these previously described limits would be required. Based on this, the normal operating range of pressurizer pressure does not require an LCO, although operation within that range should generally be assumed to exist (with appropriate conservative bias) at the beginning of a transient analysis.

Pressurizer pressure limits do define the 'lowest functional capability or performance level required for safe operation' at the limits assumed in the safety analyses for actuations as described in LCO 3.3.1, Module Protection System, and analyses as set by the Core Operating Limits Report (COLR) and referenced by LCO 3.4.1. Safety analyses and the topical reports used to generate the COLR limits may, however are not expected to coincide with the normal operating range.

The unique NuScale design provides extensive conservatism in many parameters which provide flexibility in choosing both normal operating ranges and analysis limits that are demonstrated to be the 'lowest functional capability or performance level required for safe operation.'

The FSAR describes normal operating limits and operating limits that are credited as initial conditions or operating restrictions. The tables below identify the parameters and the limits that satisfy 10 CFR 50.36(c)(2)(i) and 10 CFR 50.36(c)(2)(ii). Where identified, normal operating limits are identified by reference to the FSAR location if they are different from the LCO limits that define the 'lowest functional capability or performance level required for safe operation.

Table 3-3 Parameters and operating restrictions that require an LCO

	10 CFR 50.36(c)(2)(i) and 10 CFR 50.36(c)(2)(ii)	**Normal Operating Range FSAR Reference**
Parameters and operating restrictions		
Design core power	Yes	(Operating License)
Minimum RCS temperature for criticality	Yes	Same as LCO
RCS pressure for criticality	Yes	4.4
RCS temperatures	Yes	5.1.4
Containment pressure	Yes	6.2.1, 6.2.2
RCS flow	Yes	5.1.4, 4.4
Shutdown margin (SDM)	Yes	4.3.2
Enthalpy rise hot channel factor ($F_{\Delta H}$)	Yes	Same as LCO
Axial offset (AO)	Yes	Same as LCO
CRA positions	Yes	4.3.2
RCS specific activity	Yes	11.1
Primary-to-secondary leakage	Yes	Same as LCO
Containment leak rate	Yes	Same as LCO
Decay time before fuel handling	Yes	Same as LCO
Design features		
Fuel design		
• enrichment	No	4.2.2
• cladding	No	4.2.2
• geometry	No	4.2.2
Reactor pool		
• Depth	Yes	9.2.5
• Temperature	Yes	9.2.5
• Boron concentration	Yes	9.2.5, 9.1.3

The variables listed above are mapped to the GTS in Table A-1.

Some of the LCO limits are established by reference to the COLR; additional parameters in Table 3-4 below are managed and established in the COLR required by TS 5.6.3.

Table 3-4 Core design limits that are initial conditions used in design basis event evaluation specified in the Core Operating Limits Report

Maximum assembly radial peaking
Peak rod exposure
Maximum hot zero power (HZP) critical boron concentration
Maximum hot full power (HFP) critical boron concentration
Minimum refueling boron concentration
Most positive moderator temperature coefficient (MTC) at power ≤ 25% RTP
Most positive MTC at power >25% RTP
Most negative MTC at HZP
Most negative MTC at HFP
Minimum HFP SDM (critical k_{eff})
Minimum HZP SDM (critical k_{eff})
Cold shutdown criterion
Maximum fuel pellet enrichment (wt. %)
CRA position uncertainty
Boric Acid supply boron concentration
Maximum CVCS makeup pump water flow path flowrate

Criterion 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.

The FSAR describes SSC and functions that operate or actuate to mitigate design basis accident or transients in the safety analyses of the NuScale design.

The module protection system (MPS) and neutron monitoring system provide instrumentation and actuation functions credited to detect and mitigate design basis events in the analyses. Table 3-5 lists the signals of the MPS that are used in the design basis analyses.

Table 3-5 Module protection system signals used in the Chapter 15 analyses

Signal	**Basis and Protected Limit or Parameter**
High power	Exceeding CHF limits for reactivity and overcooling events
Startup and intermediate range log power rate	Exceeding CHF and energy deposition limits during startup power excursions
High power rate	Exceeding CHF limits for reactivity and over-cooling events
High startup range countrate	Exceeding CHF and energy deposition limits during rapid startup power excursions
High subcritical multiplication	Detect and mitigate inadvertent subcritical boron dilutions in operating MODES 2 and 3
High RCS hot temperature	Exceeding CHF limits for reactivity and heatup events
High containment pressure	Mitigate RCS or secondary leaks above the allowable limits to protect RCS inventory and ECCS function during these events
High pressurizer pressure	Protect against exceeding RPV pressure limits for reactivity and heatup events
High pressurizer level	Mitigate CVCS malfunctions to protect against overfilling the pressurizer
Low pressurizer pressure	Detect and mitigate primary high-energy line break (HELB) outside the CNV and protect RCS subcooled margin against instability events
Low-low pressurizer pressure	Detect and mitigate primary HELB outside the CNV and protect RCS subcooled margin against instability events
Low pressurizer level	Mitigate primary HELB outside the CNV and CVCS malfunctions to protect the pressurizer heaters from uncovering and overheating
Low-low pressurizer level	Mitigate loss-of-coolant accidents (LOCAs) to protect RCS inventory and ECCS functionality during LOCA and primary HELB outside CNV events
Low steam pressure	Mitigate secondary HELB outside the CNV to protect SG inventory and DHRS functionality

Signal	Basis and Protected Limit or Parameter
Low-low steam pressure	Mitigate secondary HELB outside the CNV to protect SG inventory and DHRS functionality
High steam pressure	Mitigate loss of steam demand to protect primary and secondary pressure limits during heatup events
High steam superheat	Mitigate SG boil-off to protect DHRS functionality during at-power and post-trip conditions
Low steam superheat	Mitigate SG overfilling to protect DHRS functionality during at- power and post-trip conditions
Low RCS flow	Ensure boron dilution cannot be performed at low RCS flowrates because the loop transit time is too long to be able to detect the reactivity change in the core within sufficient time to mitigate the event
Low-low RCS flow	Ensure flow remains measureable and positive during low power startup conditions
High CNV water level	Protect water level above the core in LOCA events
Low RCS Pressure	Actuate ECCS during small loss of coolant events to prevent boron distribution gradients in the RCS
Low AC voltage	Ensure appropriate load shedding occurs in the highly reliable DC power system in the event of extended loss of AC power to the battery chargers
High under-the-bioshield temperature	Detect high energy leaks or breaks at the top of the NPM under-the-bioshield to reduce the consequences of HELB on the safety-related equipment located on top of the NPM

The NuScale engineered safety feature (ESF) systems consist primarily of the CNV, the ECCS, and the DHRS. Additionally, the reactor pool provides the UHS, and aspects of other design features and safety functions are credited in the accident and transient analyses. Table 3-6 identifies the SSC, features, and safety functions that satisfy Criterion 3 as they may actuate to mitigate a design basis accident or transient.

Table 3-6 NuScale structures, systems, and components credited to actuate or function in design basis accident and transient analyses

Nuclear fuel, fuel cladding, and fuel assemblies
Control rod drives and assemblies
MPS, including • neutron monitoring system • reactor trip system (RTS) • engineered safety features actuation system (ESFAS)
Reactor vessel
Reactor safety valves
Emergency core cooling valves
Steam generators
DHRS heat exchanger, piping, and valves including main steam and feedwater isolation
CNV and containment isolation valves
CVCS demineralized water isolation valves
Reactor pool

The variables listed above are mapped to the GTS in Table A-1.

Criterion 4

A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

The NuScale Power Plant is designed to a fundamentally different safety paradigm and has no design-specific operating experience. Probabilistic risk assessment results show that while core damage frequencies (CDFs) for the previously operating nuclear power plants typically fall between 10^{-5} to 10^{-6} per reactor year, the NuScale CDF is approximately 10^{-8} per reactor year.

The NRC guidance on risk-informed applications was developed in the context of risk results calculated for large PWRs. Regulatory Guide 1.174 (Reference 6.2.14) describes changes in CDF that the NRC has considered acceptable when making permanent changes to a plant's licensing basis. This provides a relative scale for assessing whether the function of an SSC is significant to the public health and safety, i.e., in the region approaching or greater than 10^{-6} per reactor year.

NuScale used risk information in many decision-making processes employed to develop a safe, economical, and efficient design. These include risk-informed SSC categorization, risk-informed inservice inspections and inservice testing, design reliability assurance program, as well as assessing specific design issues as risk-significant or not risk-significant.
Based on the combination of orders-of-magnitude difference between the NuScale CDF and the NRC-accepted CDF, and programs used to ensure that the CDF remains at extremely low levels, no non-safety related SSC or functions approached the 10^{-6} per reactor year criteria for inclusion in the TS in accordance with Criterion 4.

Based on criterion 4 operating experience, LCOs are proposed for:

- manual actuation functions of the MPS
- the remote shutdown station
- nonsafety-related main steam line and feedwater isolation functions
- refueling neutron monitoring instrumentation

The MPS allows manual actuation of protective functions, however this capability is not credited in the design; it is included based on industry inclusion as a Criterion 4 condition. The remote shutdown station does not have an active function in the NuScale design other than monitoring conditions; it is included based on industry inclusion as a Criterion 4 condition. Similarly, the refueling neutron monitoring instrumentation does not provide an actuation; however, it does provide information about the reactor core condition during conduct of refueling activities and is included based on Criterion 4.

The backup nonsafety-related main steam line and feedwater line isolation valves and their automatic actuation were identified for inclusion by application of the design reliability assurance program and are also included in the TS in accordance with Criterion 4.

The functions listed above are mapped to the GTS in Table A-1.

<u>Other Limiting Conditions of Operation</u>

Other requirements or limitations in the Limiting Conditions of Operation that are included but that do not satisfy any of the criteria in 10 CFR 50.36(c)(2).

After receiving input from the staff, NuScale set RCS specific activity limits in LCO 3.4.8 at a level consistent with the dose rates that may exist in adjoining areas based on calculated design basis source terms. This limit is not consistent with 10 CFR 50.36, however is conservative compared with RCS specific activity limits established at traditional plants based on safety analysis assumptions, and therefore commonly established in accordance with criterion 2 above.

Similarly, after staff provided input to NuScale to establish a limit on in-containment secondary side piping leakage. The LCO was included because if a seismic event occurs when the in-containment secondary leakage is greater than the LCO limit, a main steam or feedwater pipe break could occur. This could result in an adverse interaction between the affected in-containment secondary system piping and other safety related equipment located inside the containment.

3.4 Chapter 4, Design Features

The structure, content, and level of detail of Chapter 4 of the NuScale GTS directly align with Chapter 4 content typical of PWRs. A description of the site location, the site and exclusion area boundaries, and the low population zone around the reactor are to be provided by combined license (COL) applicants, consistent with the descriptions provided in the FSAR.

A description of the reactor core is provided, including the number of fuel assemblies and the materials used in their construction. A description of the CRA makeup and arrangement is provided.

Additionally, a description of the storage of new and irradiated fuel assemblies, including measures to prevent inadvertent criticality, limit exposures associated with storage, and the overall capacity of the storage area is provided.

This content directly aligns with the requirements of 10 CFR 50.36(c)(4). Other features of the facility that could have a significant effect on safety are described in Chapter 2 or 3 of the TS.

3.5 Chapter 5, Administrative Controls

The structure and content of Chapter 5 of the NuScale GTS closely aligns with Chapter 5 content typical of STS for PWRs. Deviations occur primarily in Section 5.5, Programs and Manuals, because the NuScale design does not include the features that require the identified program or manual. A comparison of the contents of Section 5.5 and explanation of omissions are provided in Table 3-7 below.

Table 3-7 Comparison of NuScale generic technical specifications with NUREG-1431 Section 5.5 contents

NuScale GTS Section 5.5	NUREG-1431 Section 5.5
5.5.1 Offsite Dose Calculation Manual (ODCM)	5.5.1 Offsite Dose Calculation Manual (ODCM)
The NuScale design does not include safety-related RCS flow loops outside the containment. See AP1000 SER Chapter 20, TMI Item III.D.1	5.5.2 Primary Coolant Sources Outside Containment
The NuScale design does not include a post-accident sampling system that contributes significantly to plant safety and accident recovery. See travelers 366, 413, 442, and NUREG-2194.	5.5.3 [Post Accident Sampling]
5.5.2 Radioactive Effluent Controls Program	5.5.4 Radioactive Effluent Controls Program
5.5.3 Component Cyclic or Transient Limit	5.5.5 Component Cyclic or Transient Limit
The NuScale design does not include a pre-stressed concrete containment.	5.5.6 [Pre-Stressed Concrete Containment Tendon Surveillance Program]
The NuScale design does not include reactor coolant pumps.	5.5.7 Reactor Coolant Pump Flywheel Inspection Program
Relocated consistent with the incorporation of traveler 545.	5.5.8 Inservice Testing Program
5.5.4 Steam Generator (SG) Program	5.5.9 Steam Generator (SG) Program
5.5.5 Secondary Water Chemistry Program	5.5.10 Secondary Water Chemistry Program
The NuScale design does not credit ventilation filtration; therefore, the GTS do not include ventilation filtration.	5.5.11 Ventilation Filter Testing Program
5.5.6 Explosive Gas and Storage Tank Radioactivity Monitoring Program	5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program
The NuScale design does not credit emergency diesel generators.	5.5.13 Diesel Fuel Oil Testing Program
5.5.7 Technical Specifications (TS) Bases Control Program	5.5.14 Technical Specifications (TS) Bases Control Program
5.5.8 Safety Function Determination Program (SFDP)	5.5.15 Safety Function Determination Program (SFDP)
5.5.9 Containment Leakage Rate Testing Program	5.5.16 Containment Leakage Rate Testing Program
The NuScale design does not include safety-related batteries.	5.5.17 Battery Monitoring and Maintenance Program
The NuScale design does not include specifications that result in a need for a system level OPERABILITY program.	5.5.18 Control Room Envelope (CRE) Habitability Program

NuScale GTS Section 5.5	NUREG-1431 Section 5.5
5.5.10 Setpoint Program (SP)	5.5.19 [Setpoint Control Program]
5.5.11 Surveillance Frequency Control Program	5.5.20 [Surveillance Frequency Control Program]
5.5.12 Spent Fuel Storage Rack Neutron Absorber Monitoring Program	5.5.21 Spent Fuel Storage Rack Neutron Absorber Monitoring Program

Programs were omitted either because the associated SSC do not exist as credited features in the NuScale design, or the program was removed or relocated in accordance with associated industry / NRC STS travelers.

3.6 Mapping of 10 CFR 50.36 Selected Limits to Proposed Technical Specifications

The parameters, SSC, and functions identified in Sections 3.1 through 3.5 above are correlated with the proposed TS in Appendix A.

4.0 Comparison with Standard Technical Specifications

The NuScale power plant design is different from previously licensed nuclear power plants. Plant operations are also different from previously operating nuclear power plants. Experience and lessons learned from the improved technical specifications were extensively considered during development of the proposed GTS. However, in many safety-related functional areas the NuScale GTS do not directly correspond with similarly named systems or components. As described in Section 2.0 above, the GTS were developed without direct reliance on previously existing generic technical specifications. Development required consideration of the applicable regulations, accompanying Commission policies and statements of consideration, evolution of the STS, and industry experience including the industry/NRC STS change travelers.

Consideration of the contents of the STS and travelers does not imply direct correspondence or functional equivalent unless described as such. The NuScale design is not addressed in the traveler process, so none of the travelers are explicitly or directly applicable to the NuScale GTS. Rather the intent of the traveler was considered based on available information related to the changes made or proposed to the STS. The following sections provide additional details regarding the extent of consideration and where appropriate, the adoption of similar controls and limitations in the proposed NuScale GTS.

4.1 Specification Comparisons

As described in Section 3, the NuScale GTS were developed to be consistent with the NuScale-specific safety analyses and the design-specific probabilistic risk analyses. Additionally, the STS were used as a basis for the content and format of the proposed GTS.

The STS are published by the NRC as six NUREGs tailored to various reactor designs. The versions of the STS that were considered during GTS development included:

NUREG	Design	Current Revision	Manuscript Completion Date	Publication Date
NUREG-1430	Babcock and Wilcox	4	October 2011	April 2012
NUREG-1431	Westinghouse	4	October 2011	April 2012
NUREG-1432	Combustion Engineering	4	October 2011	April 2012
NUREG-1433	General Electric BWR4	4	October 2011	April 2012
NUREG-1434	General Electric BWR6	4	October 2011	April 2012
NUREG-2194	Westinghouse AP1000	0	December 2015	April 2016

Additionally, the NRC has issued the design certification for the ESBWR which included GTS. The specifications and bases provided in the GTS for the ESBWR (Reference 6.2.13) were also consulted as a basis of comparison during preparation of some parts of the NuScale proposed TS.

The focus of the development and comparisons were generally against the Westinghouse, CE, and AP1000 STS. Appendix B provides a comparison of the contents of those STS with the contents of the proposed NuScale GTS.

4.2 Industry/NRC STS Traveler Consideration

The previously existing nuclear electric power generation industry works with the NRC staff in the STS maintenance and change process through the Technical Specification Task Force (TSTF), a joint activity of the PWR, boiling water reactor (BWR), and AP1000 Owners Groups. The TSTF coordinates with the NRC to implement the change process (referred to as the traveler process) that maintains the STS and has periodically resulted in publication of revisions to the STS. Although no travelers were developed with direct consideration of the NuScale design or proposed GTS, information regarding travelers available to NuScale through the June 30, 2018 was considered during preparation of the NuScale GTS.

Appendix C of this report describes the consideration and extent of incorporation of traveler contents and intentions. The scope of the Appendix is generally limited to travelers issued since the last published revision of the NUREG STS listed in Section 4.1 above.

5.0 Conformance with Standard Technical Specification Writer's Guide

The guidance provided in the Writer's Guide (Reference 6.2.4) for Plant Specific Improved Technical Specifications, TSTF-GG-05-01, Revision 1, August 2010 was used to prepare the proposed NuScale GTS. Portions of the guide are specific to BWRs or PWRs and in some cases conformance was not possible or was otherwise inappropriate.

Chapter 1 and specifications LCO 3.0 and SR 3.0 of the proposed specifications are minimally modified to conform to the facility design and specifications. This results in formatting generally consistent with the Writer's Guide. Additional emphasis was placed on conformance with the expectations described in section 4 of the guide related to content of each portion of the TS and Bases.

6.0 References

6.1 Source Documents

6.1.1 10 CFR 20, Standards for Protection Against Radiation.

6.1.2 10 CFR 50, Domestic Licensing of Production and Utilization Facilities.

6.1.3 10 CFR 52, Licenses, Certifications, and Approvals for Nuclear Power Plants.

6.2 Referenced Documents

6.2.1 U.S. Code of Federal Regulations, "Technical Specifications," Section 50.36, Part 50, Chapter I, Title 10, "Energy," (10 CFR 50.36).

6.2.2 U.S. Code of Federal Regulations, "Technical Specifications on Effluents from Nuclear Power Reactors," Section 50.36a, Part 50, Chapter I, Title 10, "Energy," (10 CFR 50.36a).

6.2.3 U.S. Nuclear Regulatory Commission, "Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," Federal Register, Vol. 52 FR 3788, February 6, 1987.

6.2.4 Technical Specification Task Force, "Writer's Guide for Plant-Specific Improved Technical Specifications," TSTF-GG-05-01, Revision 1, Rockville, MD, August 2010.

6.2.5 U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Babcock and Wilcox Plants," NUREG-1430, Revision 4.0, Volumes 1 and 2, April 2012.

6.2.6 U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Westinghouse Plants," NUREG-1431, Revision 4.0, Volumes 1 and 2, April 2012.

6.2.7 U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Combustion Engineering Plants," NUREG-1432, Revision 4.0, Volumes 1 and 2, April 2012.

6.2.8 U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, General Electric BWR/4 Plants," NUREG-1433, Revision 4.0, Volumes 1 and 2, April 2012.

6.2.9 U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, General Electric BWR/6 Plants," NUREG-1434, Revision 4.0, Volumes 1 and 2, April 2012.

6.2.10 U.S. Nuclear Regulatory Commission, "Standard Technical Specifications, Westinghouse Advanced Passive 1000 (AP1000) Plants," NUREG-2194, Volumes 1 and 2, April 2016.

6.2.11 U.S. Nuclear Regulatory Commission, "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors," Federal Register, Vol. 58 FR 39132, July 22, 1993.

6.2.12 Murley, T.E., U.S. Nuclear Regulatory Commission, letter to Walter S. Wilgus, B&W Owners Group, May 9, 1988, Agencywide Document Access and Management System (ADAMS) Accession No. ML11264A057.

6.2.13 GE-Hitachi Nuclear Energy, ESBWR Design Control Document, Tier 2 and Generic Technical Specifications, Chapter 16, Technical Specifications, and Chapter 16B, Bases, Revision 10, April 2014.

6.2.14 Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 2, May 2011.

7.0 Appendices

Appendix A	Criteria for Inclusion of Technical Specifications
Appendix B	Summary Comparison of Standard Technical Specifications with NuScale Generic Technical Specifications Contents
Appendix C	Technical Specification Task Force Traveler Evaluations

Appendix A Criteria for Inclusion of Technical Specifications

The accident analyses provided in the DCA with emphasis on Final Safety Analysis Report (FSAR) Chapter 15 and supporting calculations were reviewed to identify SSC and parameters that satisfy regulatory requirements and are discussed in Sections 3.1 through 3.5. This appendix correlates the identified parameters, SSC, and functions with the proposed TS. The results provided in tables below.

Table A-1 Technical specifications inclusion criteria

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
Reactor core safety limits • critical heat flux ratio • peak fuel centerline temperature	(c)(1)(i)(A)	2.1.1 Reactor Core SLs	Consistent with and similar to large plant safety limits with variables selected based on consideration of natural circulation flow through the reactor core.
Safety limit on RCS pressure	(c)(1)(i)(A)	2.1.2 RCS Pressure SL	Consistent with and similar to large PWR SLs.
Safety limit violations	(c)(1)(i)(A)	2.2 Safety Limit Violations	Consistent with and similar to large PWR SLs.
Design core power	(c)(2)(ii)(B)	Facility Operating License	This is an assumed input parameter of safety analyses calculations. It is maintained by manual and automatic controls in accordance with operating procedures.
Reactor coolant system (RCS) temperatures	(c)(2)(ii)(B)	1.1 Definition of MODES as provided in Table 1.1-1 3.4.1 RCS Pressure and Temperature Critical Heat Flux (CHF) Limits 3.4.2 RCS Minimum Temperature for Criticality 3.4.3 RCS Pressure and Temperature (P/T) Limits 5.6.4 Pressure Temperature Limits Report (PTLR)	These are assumed input parameters of safety analyses calculations. They are maintained by manual and automatic controls in accordance with operating procedures.
Pressurizer pressure	(c)(2)(ii)(B)	3.4.1 RCS Pressure and Temperature Critical Heat Flux (CHF) Limits 3.4.3 RCS Pressure and Temperature (P/T) Limits 5.6.4 Pressure Temperature Limits Report (PTLR)	This is an assumed input parameters of safety analyses calculations. It is maintained by manual and automatic controls in accordance with operating procedures.
Pressurizer level	(c)(2)(ii)(B)	3.3.1 Module Protection System Instrumentation 5.5.10 Setpoint Program (SP)	Operation within conservative normal operating limits is ensured by manual and automatic controls in accordance with operating procedures. This is also an actuation parameter credited in safety analyses. Protective actuation setpoints are established in accordance with the Setpoint Program such that protective features will initiate at established limits.
Containment pressure	(c)(2)(ii)(A) (c)(2)(ii)(B)	3.3.1 Module Protection System Instrumentation 5.5.10 Setpoint Program (SP) 3.4.7 RCS Leakage Detection	Operation within conservative normal operating limits is ensured by manual and automatic controls in accordance with operating procedures. Operation within limits is also ensured by the OPERABILITY requirements of LCO 3.4.7, RCS Leakage Detection Instrumentation. This is also an actuation parameter credited in safety analyses. Protective actuation setpoints are established in accordance with the Setpoint Program such that protective features will initiate at established limits.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
Steam pressure	(c)(2)(ii)(B)	3.3.1 Module Protection System Instrumentation 5.5.10 Setpoint Program (SP)	This is an actuation parameter credited in safety analyses. Operation within the assumed limits is ensured by manual and automatic controls in accordance with operating procedures. Protective actuation setpoints are established in accordance with the Setpoint Program such that protective features will initiate at established limits.
Feedwater temperature	(c)(2)(ii)(B)	3.3.1 Module Protection System Instrumentation 5.5.10 Setpoint Program (SP)	This is an actuation parameter credited in safety analyses. Operation within the assumed limits is ensured by manual and automatic controls in accordance with operating procedures. Protective actuation setpoints are established in accordance with the Setpoint Program such that protective features will initiate at established limits.
RCS flow	(c)(2)(ii)(B)	Natural phenomenon 3.3.1 Module Protection System Instrumentation 5.5.10 Setpoint Program (SP)	This is an actuation parameter credited in safety analyses. It is ensured by physical phenomena caused by density differences in the natural circulation in the RCS. Protective actuation setpoints are established in accordance with the Setpoint Program such that protective features will initiate at established limits.
Shutdown margin	(c)(2)(ii)(B)	3.1.1 SHUTDOWN MARGIN 3.1.8 PHYSICS TESTS Exceptions	This is an assumed input parameter of the safety analyses calculations. It is maintained by manual and automatic controls in accordance with operating procedures.
Control rod assembly positions	(c)(2)(ii)(B)	3.1.4 Rod Group Alignment Limits 3.1.5 Shutdown Group Insertion Limits 3.1.6 Regulating Group Insertion Limits 3.1.7 Rod Position Indication	These are assumed input parameters of the safety analyses calculations. It is maintained by manual and automatic controls in accordance with operating procedures. LCO 3.1.7 supports the parameter determination by ensuring OPERABLE indication is used to verify compliance with those LCO.
RCS specific activity	(c)(2)(ii)(B) Input from NRC staff other than criteria of 10 CFR 50.36.	3.4.8 RCS Specific Activity	This is an assumed input parameter of certain safety analyses calculations. However, the limits were restricted further and established in response to NRC staff direction. The RCS specific activity limits in LCO 3.4.8 are set at a level consistent with the dose rates that may exist in adjoining areas based on calculated design basis source terms. These limits are not consistent with the safety analyses; however, they are conservative compared with RCS specific activity limits established at traditional plants based on safety analysis assumptions. RCS Specific Activity is monitored and maintained in accordance with plant procedures.
Primary to secondary leakage	(c)(2)(ii)(B)	3.4.5 RCS Operational LEAKAGE	This is an assumed input parameter of certain safety analyses calculations. It is monitored and maintained in accordance with plant procedures.
Containment leak rate	(c)(2)(ii)(B)	3.6.1 Containment – Operating 3.6.2 Containment Isolation Valves 5.5.9 Containment Leakage Rate Testing Program	This is an assumed input parameter of certain safety analyses calculations. It is monitored and maintained in accordance with plant programs and procedures.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
In-containment secondary piping leakage	Input from NRC staff other than criteria of 10 CFR 50.36	3.7.3 In-Containment Secondary Piping Leakage	This establishes a limit implementing controls and actions to be taken if secondary piping leakage exceeds those used to satisfy the leak-before-break criteria as applied to piping inside the containment.
Decay time before fuel handling	(c)(2)(ii)(B)	3.8.2 Decay Time	This is an assumed input parameter of certain safety analyses calculations. It is met in accordance with plant procedures.
Fuel design • enrichment • cladding • geometry	(c)(2)(ii)(B)	2.1.1 Reactor Core SLs 3.1.2 Core Reactivity 3.1.8 PHYSICS TESTS Exceptions 4.2 Reactor Core	These are assumed input parameters of certain safety analyses calculations. It includes design features of the fuel and assemblies, and it is maintained in accordance with plant procedures.
Core design • reactivity • temperature coefficients of reactivity • core power distribution • fuel burnup	(c)(2)(ii)(B)	2.1.1 Reactor Core SLs 3.1.1 SHUTDOWN MARGIN (SDM) 3.1.2 Core Reactivity 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 3.1.7 Rod Position Indication 3.1.8 PHYSICS TESTS Exceptions 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 3.4.2 RCS Minimum Temperature for Criticality 4.2 Reactor Core 5.6.3 Core Operating Limits Report (COLR)	These are assumed input parameters of safety analyses calculations. It includes design features of the fuel and assemblies, and it are maintained in accordance with plant design, programs, and procedures.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
Fuel storage and handling system • geometry • location	(c)(2)(ii)(B)	3.5.3 Ultimate Heat Sink 4.3 Fuel Storage 5.5.12 Spent Fuel Storage Rack Neutron Absorber Monitoring Program	These are assumed input parameters of safety analyses calculations. They consist of design features of the fuel storage location and racks.
Reactor pool • depth • temperature • boron concentration	(c)(2)(ii)(B)	3.5.3 Ultimate Heat Sink	These are assumed input parameters of safety analyses calculations. They are maintained in accordance with plant procedures.
High range linear power	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
High power rate	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
High source range and intermediate range log power rate	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
High source range count rate	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
High source range log power rate	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
High pressurizer pressure	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
Low pressurizer pressure	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
Low low pressurizer pressure	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
High pressurizer level	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
Low pressurizer level	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
Low low pressurizer level	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
High RCS narrow range T_{hot} temperature	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
Low RCS flow	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
Low low RCS flow	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
High main steam pressure	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
Low main steam pressure	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
Low low main steam pressure	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
High steam superheat	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program..
Low steam superheat	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
High narrow range containment pressure	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
High containment water level	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
Low RCS Pressure – ECCS	(c)(2)(ii)(C)	3.3.1 Module Protection System 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
High RCS pressure – LTOP	(c)(1)(ii)(A) (c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 3.4.10 Low Temperature Overpressure Protection (LTOP) Valves 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits and actuating the associated engineered safety features. The limit(s) constitute a limiting safety setpoint as described in 5.5.10, the Setpoint Program. The limit(s) are established in accordance with the Setpoint Program.
Low AC voltage to ELVS battery chargers	(c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) are established in accordance with the Setpoint Program.
High under the bioshield temperature	(c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This function is credited with detecting when the parameter is out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The limit(s) are established in accordance with the Setpoint Program.
Manual actuation functions	(c)(2)(ii)(D)	3.3.4 Manual Actuation Functions	While not credited in the design bases analyses, the availability of the capability to manually actuate safety functions is considered a function which operating experience has shown to be significant to public health and safety.
Remote shutdown station	(c)(2)(ii)(D)	3.3.5 Remote Shutdown Station	The remote shutdown station is not credited in the design bases analyses. Remote shutdown station availability to monitor plant conditions if an event occurs that requires evacuation of the main control room is considered a function that operating experience has shown to be significant to public health and safety.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
Control rod drives and assemblies	(c)(2)(ii)(C)	3.1.1 SHUTDOWN MARGIN 3.1.5 Shutdown Group Insertion Limits 3.1.6 Regulating Group Insertion Limits 3.1.7 Rod Position Indication 3.1.8 PHYSICS TESTS Exceptions 4.2 Reactor Core 5.6.3 Core Operating Limits Report (COLR)	Control rod assembly (CRA) insertion is credited in response to an actuation of the Reactor Trip System. The ability to trip the reactor and insert appropriate negative reactivity by insertion of the CRAs in a timely manner is established by the combination of total rod worth and reactivity insertion rate.
Module protection system	(c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 5.5.10 Setpoint Program (SP)	This system is credited with detecting when parameters are out of limits, initiating a reactor trip, and actuating the associated engineered safety features. The actuation setpoints are established in accordance with the Setpoint Program.
Neutron monitoring system	(c)(2)(ii)(C) (c)(2)(ii)(D)	3.3.1 Module Protection System Instrumentation 3.8.1 Nuclear Instrumentation	This system is used to monitor neutron flux and provide a signal to the MPS that can be compared to suitable setpoints to determine the need to initiate a reactor trip or actuate engineered safety features. Additionally, the system provides indication of the neutron flux at the refueling tool to provide an indication of the subcritical multiplication rate during the movement of fuel in the area of the reactor core during refueling activities.
Reactor trip system	(c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.2 Reactor Trip System (RTS) Logic and Actuation	This system initiates a reactor trip in response to signals from the MPS, causing the control rod assemblies to be inserted into the reactor core. This is accomplished by interrupting the electrical power that holds the control rod assemblies out of the core.
Engineered safety features actuation system	(c)(2)(ii)(C)	3.3.1 Module Protection System Instrumentation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation	This system initiates engineered safety features actuations depending on the combination of signals that are determined not to be within limits. The ESFAS actuation logic utilizes signals from the MPS to initiate ESF actuations.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
Reactor vessel and safety valves	(c)(2)(ii)(A) (c)(2)(ii)(C)	2.1.2 RCS Pressure SL 3.3.1 Module Protection System Instrumentation 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation 3.4.1 RCS Pressure, Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits 3.4.3 RCS Pressure and Temperature (P/T) Limits 3.4.4 Reactor Safety Valves (RSVs) 3.5.1 Emergency Core Cooling System (ECCS) 5.6.4 Pressure Temperature Limits Report (PTLR)	The reactor pressure boundary formed by the reactor vessel is one of the principal physical barriers that prevent the release of radioactive materials to the environment. The ability of the reactor vessel to perform this function is controlled by a combination of the specifications listed. In addition to the pressurizer safety valves providing overpressure protection for the reactor vessel during operations, the MPS, ESFAS, and ECCS valves provide low temperature overpressure protection to ensure vessel integrity.
Emergency core cooling valves	(c)(2)(ii)(C)	3.5.1 Emergency Core Cooling System (ECCS) 3.5.3 Ultimate Heat Sink	The ECCS provides a passive means of depressurizing and cooling the RCS if a LOCA occurs inside the containment by forming a closed cooling loop between the RCS and the containment vessel walls which are directly cooled by the Ultimate Heat Sink.
Steam generators	(c)(2)(ii)(C)	3.4.9 Steam Generator (SG) Tube Integrity 5.5.4 Steam Generator (SG) Program 5.5.5 Secondary Water Chemistry Program 5.6.5 Steam Generator Tube Inspection Report	The steam generators form a portion of the reactor coolant pressure boundary. They also serve as the heat exchangers used to cool and remove decay heat from the reactor during normal shutdowns and design basis events other than postulated LOCAs.
Decay heat removal system heat exchanger, piping, and valves	(c)(2)(ii)(C)	3.5.2 Decay Heat Removal System (DHRS) 3.5.3 Ultimate Heat Sink	The DHRS, in combination with the steam generators serve as the passive, safety related means to cool and transfer decay heat from the reactor to the reactor pool by natural circulation during design basis events other than postulated LOCAs.
Main steam and feedwater isolation valves	(c)(2)(ii)(C)	3.7.1 Main Steam Isolation Valves (MSIVs) 3.7.2 Feedwater Isolation	The Main Steam and Feedwater Isolation functions provide multiple functions including limiting releases from postulated SG tube failures, and forming portions of boundary of the DHRS naturally circulating cooling loop.
Backup main steam isolation and feedwater regulating valves	(c)(2)(ii)(D)	3.7.1 Main Steam Isolation Valves 3.7.2 Feedwater Isolation	The Backup Main Steam Isolation Valves and the Feedwater Regulating Valves functions provide non-safety related backup isolation capability that forms portions of boundary of the DHRS naturally circulating cooling loop if needed. The availability of this backup capability is considered a function which industry operating experience and probabilistic risk assessment has shown to be potentially significant to public health and safety.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
Containment vessel and containment isolation valves	(c)(2)(ii)(C)	3.4.6 Chemical and Volume Control System (CVCS) Isolation Valves 3.6.1 Containment – Operating 3.6.2 Containment Isolation Valves 5.5.9 Containment Leakage Rate Testing Program	Provides the containment fission product boundary.
CVCS demineralized water isolation valves Boric acid supply boron concentration RCS makeup flowrate	(c)(2)(ii)(C)	3.1.9 Boron Dilution Control 5.6.3 Core Operating Limits Report (COLR)	Limits the potential for inadvertent reduction of boron concentration by ensuring the OPERABILITY of the automatic isolation valves, ensuring that boric acid sources are within limits, and the flowrate to the RCS is consistent with the safety analyses and the limits in COLR.
Reactor pool	(c)(2)(ii)(C)	3.5.3 Ultimate Heat Sink	The reactor pool serves as the Ultimate Heat Sink and supports numerous safety functions including serving as the heat sink for the ECCS and DHRS, and mitigating postulated fuel handling events. The reactor pool level also provides buoyance supporting a fraction of the weight of a module being supported by the reactor building crane during transition.
Site description • location • site boundaries • exclusion area boundaries • low population zone	(c)(4)	4.1 Site Location	This specification will be prepared by the COL applicant consistent with the description in the FSAR.
Reactor core • number of fuel assemblies • materials used in their construction • control rod assemblies makeup • control rod assemblies arrangement.	(c)(4)	4.2 Reactor Core	A description similar to those provided for PWRs is included to address the requirements of 10 CFR 50.36.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
Storage of new and irradiated fuel assemblies • measures to prevent inadvertent criticality • limit exposures associated with storage • the overall capacity of the storage area.	(c)(4)	4.3 Fuel Storage	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36.
Responsibility	(c)(5)	5.1 Responsibility	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36.
Organization	(c)(5)	5.2 Organization	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36. Modified to reflect NuScale-specific staffing levels.
Facility staff qualifications	(c)(5)	5.3 Facility Staff Qualifications	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36.
Procedures	(c)(5) 10CFR50.36a	5.4 Procedures	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36 and 10 CFR 50.36a.
Offsite dose calculation manual (ODCM)	10CFR50.36a	5.5.1 Offsite Dose Calculation Manual (ODCM)	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36a.
Radioactive effluent controls program	10CFR50.36a	5.5.2 Radioactive Effluent Control Program	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36a.
Component cyclic or transient limit	(c)(5)	5.5.3 Component Cyclic or Transient Limit	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36. Details of the program are modified to reflect the NuScale design.
Steam generator (SG) program	(c)(5)	5.5.4 Steam Generator (SG) Program	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36. Details of the program are modified to reflect the NuScale design.
Secondary water chemistry program	(c)(5)	5.5.5 Secondary Water Chemistry Program	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36. Details of the program are modified to reflect the NuScale design.
Explosive gas and storage tank radioactivity monitoring program	(c)(5)	5.5.6 Explosive Gas and Storage Tank Radioactivity Monitoring Program	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36.
Technical specifications (TS) bases control program	(c)(5)	5.5.7 Technical Specification (TS) Bases Control Program	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36.

Parameter, SSC, or Function	Relevant 10 CFR 50.36 Criteria or Reference	Associated NuScale Technical Specifications	Discussion
Safety function determination program (SFDP)	(c)(5)	5.5.8 Safety Function Determination Program (SFDP)	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36.
Containment leakage rate testing program	(c)(5)	5.5.9 Containment Leakage Rate Testing Program	A description similar to those provided for PWRs is included to address the requirements of 10 CFR 50.36. Details of the Program are modified to reflect the NuScale design.
Setpoint program (SP)	(c)(5)	5.5.10 Setpoint Program (SP)	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36.
Surveillance frequency control program	(c)(5)	5.5.11 Surveillance Frequency Control Program	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36.
Spent fuel storage rack neutron absorber monitoring program	(c)(5)	5.5.12 Spent Fuel Storage Rack Neutron Absorber Monitoring Program	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36.
Annual radiological environmental operating report	10CFR50.36a	5.6.1 Annual Radiological Environmental Operating Report	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36a.
Radioactive effluent release report	10CFR50.36a	5.6.2 Radioactive Effluent Release Report	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36a.
Core operating limits report (COLR)	(c)(5)	5.6.3 Core Operating Limits Report (COLR)	A description similar to those provided for PWRs is included to address the requirements of 10 CFR 50.36. Details of the report are modified to reflect the NuScale design and analysis methodologies.
Reactor coolant system (RCS) pressure and temperature limits report (PTLR)	(c)(5)	5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	A description similar to those provided for PWRs is included to address the requirements of 10 CFR 50.36. Details of the report are modified to reflect the NuScale design.
Steam generator tube inspection report	(c)(5)	5.6.5 Steam Generator Tube Inspection Report	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36.
High radiation area	(c)(5)	5.7 High Radiation Area	A description similar to that provided for PWRs is included to address the requirements of 10 CFR 50.36 and provide the allowance.

Appendix B Summary Comparison of Standard Technical Specifications with NuScale Generic Technical Specifications Contents

The focus of the development and comparisons of the NuScale GTS were generally focused on the Westinghouse, CE, and AP1000 STS as published in NUREG-1431, Rev. 4, NUREG-1432, Rev. 4, and NUREG-2194, Rev. 0. The 'digital' specifications of NUREG-1432 were used in this comparison.

Table C-1 provides a summary comparison of the contents of those STS with the contents of the proposed NuScale GTS.

Table B-1 Comparison of standard technical specifications with NuScale generic technical specifications

Specification Number					
NuScale	**NUREG-2194 (AP1000)**	**NUREG-1431 (W)**	**NUREG-1432 (Digital) (CE)**	**Specification Title**	**Discussion**
1.0				**USE AND APPLICATION**	
1.1	1.1	1.1	1.1	Definitions	NuScale specifications generally consistent with ISTS, MODE and other definition changed to align with NuScale design and operations. Minor changes to 1.2, 1.3, and 1.4 to reflect contents of NuScale specifications.
1.2	1.2	1.2	1.2	Logical Connectors	
1.3	1.3	1.3	1.3	Completion Times	
1.4	1.4	1.4	1.4	Frequency	
2.0				**SAFETY LIMITS (SLs)**	
2.1	2.1	2.1	2.1	SLs	Required by 10 CFR 50.36, NuScale specifications generally consistent with ISTS, modified to reflect NuScale design.
2.2	2.2	2.2	2.2	SL Violations	
3.0				**APPLICABILITY**	
LCO 3.0	LCO 3.0	LCO 3.0	LCO 3.0	LIMITING CONDITION FOR OPERATION APPLICABILITY	NuScale specifications generally consistent with ISTS, aligned to reflect NuScale design and specifications.
SR 3.0	SR 3.0	SR 3.0	SR 3.0	SURVEILLANCE REQUIREMENT APPLICABILITY	

Specification Number					
NuScale	NUREG-2194 (AP1000)	NUREG-1431 (W)	NUREG-1432 (Digital) (CE)	Specification Title	Discussion
3.1				**REACTIVITY CONTROL SYSTEMS**	
3.1.1	3.1.1	3.1.1	3.1.1	SHUTDOWN MARGIN (SDM)	NuScale specifications generally consistent with ISTS.
3.1.2	3.1.2	3.1.2		Core Reactivity	NuScale specifications generally consistent with ISTS.
			3.1.2	Reactivity Balance	
3.1.3	3.1.3	3.1.3	3.1.3	Moderator Temperature Coefficient (MTC)	NuScale specifications generally consistent with ISTS.
3.1.4	3.1.4	3.1.4		Rod Group Alignment Limits	NuScale specifications generally consistent with ISTS, modified to reflect NuScale design.
			3.1.4	Control Element Alignment Limits	
3.1.5				Shutdown Group Insertion Limits	NuScale specifications generally consistent with ISTS, modified to reflect NuScale design.
	3.1.5	3.1.5		Shutdown Bank Insertion Limits	
			3.1.5	Shutdown Control Element Assembly Insertion Limits	
3.1.6				Regulating Group Insertion Limits	NuScale specifications generally consistent with ISTS, modified to reflect NuScale design.
	3.1.6	3.1.6		Control Bank Insertion Limits	
			3.1.6	Regulating Control Element Assembly Insertion Limits	
			3.1.7	Part Length Control Element Assembly Insertion Limits	
3.1.7	3.1.7	3.1.7		Rod Position Indication	NuScale specifications generally consistent with ISTS, modified to reflect NuScale design.
3.1.8				PHYSICS TEST Exceptions	NuScale specifications generally consistent with ISTS, modified to reflect NuScale design and testing.
	3.1.8	3.1.8		PHYSICS TESTS Exceptions - MODE 2	
				Special Test Exceptions - SHUTDOWN MARGIN	
			3.1.9	Special Test Exceptions - MODES 1 and 2	
3.1.9				Boron Dilution Control	NuScale specifications generally consistent with AP1000 with changes to reflect NuScale design and operations.
	3.1.9			Chemical and Volume Control System (CVCS) Demineralized Water Isolation Valves and Makeup Line Isolation Valves	

Specification Number					
NuScale	NUREG-2194 (AP1000)	NUREG-1431 (W)	NUREG-1432 (Digital) (CE)	Specification Title	Discussion
3.2				POWER DISTRIBUTION LIMITS	
3.2.1				Enthalpy Rise Hot Channel Factor	NuScale specifications consistent with NuScale design.
	3.2.2	3.2.2		Nuclear Enthalpy Rise Hot Channel Factor	
			3.2.2	Total Planar Radial Peaking Factor	
			3.2.1	Linear Heat Rate	
3.2.2				AXIAL OFFSET	NuScale specifications consistent with NuScale design.
	3.2.3	3.2.3		AXIAL FLUX DIFFERENCE	
			3.2.5	AXIAL SHAPE INDEX	
	3.2.1	3.2.1		Heat Flux Hot Channel Factor	Not applicable to NuScale analysis methodology and design.
	3.2.4	3.2.4		QUADRANT POWER TILT RATIO	
			3.2.3	AZIMUTHAL POWER TILT	
			3.2.4	Departure from Nuclear Boiling Ratio	
	3.2.5			On-Line Power Distribution Monitoring System (OPDMS) - Monitored Parameters	
3.3				INSTRUMENTATION	
3.3.1				Module Protection System (MPS) Instrumentation	Integrated NuScale instrumentation and actuation system design addressed through revised NuScale-specific specification construction consistent with ISTS Writer's Guide.
3.3.2				Reactor Trip System (RTS) Logic and Actuation	
3.3.3				Engineered Safety Features Actuation System (ESFAS) Logic and Actuation	
3.3.4				Manual Actuation Functions	
	3.3.1	3.3.1		Reactor Trip System Instrumentation	Structure of existing PWR Instrumentation was not used due to NuScale design differences. Integrated NuScale instrumentation and actuation system design addressed through revised NuScale-specific specification construction consistent with ISTS Writer's Guide.
			3.3.1	Reactor Protection System Instrumentation - Operating	
			3.3.2	Reactor Protective System Instrumentation - Shutdown	
	3.3.2			Reactor Trip System (RTS) Source Range Instrumentation	
	3.3.3			Reactor Trip System (RTS) Intermediate Range Instrumentation	
			3.3.3	Control Element Assembly Calculators	

Specification Number					
NuScale	**NUREG-2194 (AP1000)**	**NUREG-1431 (W)**	**NUREG-1432 (Digital) (CE)**	**Specification Title**	**Discussion**
			3.3.4	Reactor Protective System Logic and Trip Initiation	Structure of existing PWR Instrumentation was not used due to NuScale design differences. Integrated NuScale instrumentation and actuation system design addressed through revised NuScale-specific specification construction consistent with ISTS Writer's Guide.
	3.3.6			Reactor Trip System (RTS) Automatic Trip Logic	
	3.3.7			Reactor Trip System (RTS) Trip Actuation Devices	
	3.3.5			Reactor Trip System (RTS) Manual Actuation	
	3.3.8	3.3.2	3.3.5	Engineered Safety Feature Actuation System (ESFAS) Instrumentation	
	3.3.4			Reactor Trip System (RTS) Engineered Safety Feature Actuation Systems (ESFAS) Instrumentation	
			3.3.6	Engineered Safety Features Actuation System Logic and Manual Trip	
	3.3.15			Engineered Safety Feature Actuation System (ESFAS) Actuation Logic - Operating	
	3.3.16			Engineered Safety Feature Actuation System (ESFAS) Actuation Logic - Shutdown	
	3.3.9			Engineered Safety Feature Actuation System (ESFAS) Manual Initiation	
	3.3.13			Engineered Safety Feature Actuation System (ESFAS) Control Room Air Supply Radiation Instrumentation	Not applicable to NuScale design. Control room HVAC is not a credited feature.
			3.3.9	Control Room Isolation Signal	
		3.3.7		Control Room Emergency Filtration System (CREFS) Actuation Instrumentation	
		3.3.8		Fuel Building Air Cleanup System (FBACS) Actuation Instrumentation	Not applicable to NuScale design. No corresponding credited features.
			3.3.10	Fuel Handling Isolation Signal	
		3.3.5		Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation	Not applicable to NuScale design. No corresponding credited features.
			3.3.7	Diesel Generator - Loss of Voltage Start	
	3.3.11			Engineered Safety Feature Actuation System (ESFAS) Startup Feedwater Flow Instrumentation	Not applicable to NuScale design. No corresponding credited features.

Specification Number					
NuScale	**NUREG-2194 (AP1000)**	**NUREG-1431 (W)**	**NUREG-1432 (Digital) (CE)**	**Specification Title**	**Discussion**
	3.3.12			Engineered Safety Feature Actuation System (ESFAS) Reactor Trip Initiation	Not applicable to NuScale design. No corresponding credited features.
		3.3.6		Containment Purge and Exhaust Isolation Instrumentation	Not applicable to NuScale design. No corresponding credited features.
			3.3.8	Containment Purge Isolation Signal	Not applicable to NuScale design. No corresponding credited features.
(3.3.1 and 3.3.3)	3.3.10			Engineered Safety Feature Actuation System (ESFAS) Reactor Coolant System (RCS) Hot Leg Level Instrumentation	Integrated NuScale instrumentation and actuation system design addressed through revised NuScale-specific specification construction consistent with ISTS Writer's Guide. Corresponding function is implemented as part of MPS and ESFAS specifications.
(3.5.3)	3.3.14			Engineered Safety Feature Actuation System (ESFAS) Spent Fuel Pool Level Instrumentation	Reactor pool level is controlled by Specification 3.5.3. No credited actuations included in the design.
3.3.5	3.3.18	3.3.4	3.3.12	Remote Shutdown System	NuScale specifications generally consistent with ISTS, modified to reflect NuScale design. No credited safety-related controls located at the RSS in the NuScale design.
	3.3.17	3.3.3	3.3.11	Post Accident Monitoring (PAM) Instrumentation	Not applicable to NuScale design. No Type-A PAM variables identified.
(3.3.1 and 3.3.3)		3.3.9		Boron Dilution Protection System (BDPS)	Integrated NuScale instrumentation and actuation system design addressed through revised NuScale-specific specification construction consistent with ISTS Writer's Guide. Implemented as part of MPS and ESFAS specifications.
(3.3.1 and 3.3.2)			3.3.13	Logarithmic Power Monitoring Channels	Integrated NuScale instrumentation and actuation system design addressed through revised NuScale-specific specification construction consistent with ISTS Writer's Guide. Implemented as part of MPS and RTS specifications.
	3.3.19			Diverse Actuation System (DAS) Manual Controls	Not applicable to NuScale design. MPS and RTS design incorporates features that obviate need for DAS.

Specification Number					
NuScale	**NUREG-2194 (AP1000)**	**NUREG-1431 (W)**	**NUREG-1432 (Digital) (CE)**	**Specification Title**	**Discussion**
3.4				**REACTOR COOLANT SYSTEM (RCS)**	
3.4.1	3.4.1	3.4.1	3.4.1	RCS Pressure, Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits	NuScale specifications generally consistent with ISTS.
3.4.2	3.4.2	3.4.2	3.4.2	RCS Minimum Temperature for Criticality	NuScale specifications generally consistent with ISTS.
3.4.3	3.4.3	3.4.3	3.4.3	RCS Pressure and Temperature Limits	NuScale specifications generally consistent with ISTS.
	3.4.4			RCS Loops	Not applicable to NuScale design. No corresponding credited features.
		3.4.4	3.4.4	RCS Loops - MODES 1 and 2	
		3.4.5	3.4.5	RCS Loops - MODE 3	
		3.4.6	3.4.6	RCS Loops - MODE 4	
		3.4.7	3.4.7	RCS Loops - MODE 5, Loops Filled	
		3.4.8	3.4.8	RCS Loops - MODE 5, Loops Not Filled	
		3.4.18		RCS Isolated Loop Startup	
(3.3.1)	3.4.5	3.4.9	3.4.9	Pressurizer	NuScale pressurizer parameters are limited by Specification 3.3.1 and automatic actuations if required. Pressurizer heaters are not credited with a safety related function in the NuScale design.
3.4.4	3.4.6	3.4.10	3.4.10	Reactor Safety Valves	NuScale specifications generally consistent with ISTS.
3.4.5	3.4.7	3.4.13	3.4.13	RCS Operational LEAKAGE	NuScale specifications generally consistent with ISTS.
	3.4.11			Automatic Depressurization System (ADS) - Operating	Not applicable to NuScale design. No corresponding credited features.
	3.4.12			Automatic Depressurization System (ADS) - Shutdown, RCS Intact	
	3.4.13			Automatic Depressurization System (ADS) - Shutdown, RCS Open	
		3.4.11	3.4.11	Pressurizer Power Operated Relief Valves	Not applicable to NuScale design. No corresponding credited features.
(3.3.1 and 3.3.3)	3.4.8			Minimum RCS Flow	Not applicable to NuScale design. Credited flow limits established by MPS and ESFAS specifications.
3.4.6				CVCS Isolation Valves	NuScale specifications created consistent with ISTS Writer's Guide.
3.4.7	3.4.9	3.4.15	3.4.15	RCS Leakage Detection Instrumentation	NuScale specifications generally consistent with ISTS.
3.4.8	3.4.10	3.4.16	3.4.16	RCS Specific Activity	NuScale specifications generally consistent with ISTS, modified to reflect NuScale design and limits added to provide additional conservatism.

Specification Number					
NuScale	**NUREG-2194 (AP1000)**	**NUREG-1431 (W)**	**NUREG-1432 (Digital) (CE)**	**Specification Title**	**Discussion**
	3.4.15	3.4.14	3.4.14	RCS Pressure Isolation Valve Leakage	NuScale does not include valves that would require this specification.
	3.4.16			Reactor Vessel Head Vent (RVHV)	Not applicable to NuScale design. No corresponding credited features.
3.4.9	3.4.17	3.4.20	3.4.18	Steam Generator (SG) Tube Integrity	NuScale specifications generally consistent with ISTS.
3.4.10 (3.3.1 and 3.3.3)				Low Temperature Overpressure Protection (LTOP) Valves	NuScale implements LTOP using the MPS and ESFAS specifications actuating the ECCS reactor vent valves.
	3.4.14	3.4.12	3.4.12	Low Temperature Overpressure Protection (LTOP)	
		3.4.17		RCS Loop Isolation Valves	Not applicable to NuScale design. No corresponding credited features.
			3.4.17	Special Test Exceptions - RCS Loops	Not applicable to NuScale design. No corresponding credited features.
		3.4.19		RCS Loops - Test Exceptions	

Specification Number					
NuScale	**NUREG-2194 (AP1000)**	**NUREG-1431 (W)**	**NUREG-1432 (Digital) (CE)**	**Specification Title**	**Discussion**
3.5				**PASSIVE CORE COOLING SYSTEMS (PCCS)** **EMERGENCY CORE COOLING SYSTEM (ECCS)**	**NUREG section name consistent with AP1000 adopted by NuScale**
3.5.1				Emergency Core Cooling System (ECCS)	NuScale-specific design consisting of five valves resulted in NuScale-specific specification constructed consistent with ISTS Writer's Guide. Decay heat removal via convection and conduction through the CNV to the reactor pool.
	3.5.1	3.5.1	3.5.1	Accumulators/Safety Injection Tanks	
		3.5.2	3.5.3	ECCS - Operating	
		3.5.3	3.5.3	ECCS - Shutdown	
3.5.2				Decay Heat Removal System	NuScale-specific design of DHRS resulted in NuScale-specific specification constructed consistent with ISTS Writer's Guide. System path for decay heat removal from reactor core via RCS and the SGs to transfer heat to reactor pool.
3.5.3				Ultimate Heat Sink	NuScale-specific shared and integrated pool design functions as UHS, spent fuel storage pool, core inventory during refueling, and other credited features.
	3.5.2			Core Makeup Tanks (CMTs) - Operating	Not applicable to NuScale design. No corresponding credited features.
	3.5.3			Core Makeup Tanks (CMTs) - Shutdown, RCS Intact	Not applicable to NuScale design. No corresponding credited features.
		3.5.4	3.5.4	Refueling Water (Storage) Tank	Not applicable to NuScale design.
(3.6.1 and 3.5.2)	3.5.4			Passive Residual Heat Removal Heat Exchanger (PRHR HX) - Operating	DHRS heat exchangers and the CNV provide heat transfer path for passive heat removal.
(3.6.1 and 3.5.2)	3.5.5			Passive Residual Heat Removal Heat Exchanger (PRHR HX) - Shutdown, Reactor Coolant System (RCS) Intact	
Passively inherent in the NuScale design	3.5.6			In-containment Refueling Water Storage Tank (IRWST) - Operating	RCS inventory provides adequate inventory during operations and transition, with reactor pool providing inventory during refueling operations.
	3.5.7			In-containment Refueling Water Storage Tank (IRWST) - Shutdown, MODE 5	
	3.5.8			In-containment Refueling Water Storage Tank (IRWST) - Shutdown, MODE 6	
		3.5.5		Seal Injection Flow	Not applicable to NuScale design. No corresponding credited features.
			3.5.5	Trisodium Phosphate	
		3.6.7	3.6.7	Spray Additive System	

Specification Number					
NuScale	**NUREG-2194 (AP1000)**	**NUREG-1431 (W)**	**NUREG-1432 (Digital) (CE)**	**Specification Title**	**Discussion**
	3.6.8			pH Adjustment	Not applicable to NuScale design. No corresponding credited features.
		3.6.11	3.6.10	Iodine Cleanup System (ICS)	
		3.5.6		Boron Injection Tank (BIT)	
3.6				**CONTAINMENT SYSTEMS**	
3.6.1	3.6.1	3.6.1	3.6.1	Containment	NuScale specifications generally consistent with ISTS.
	3.6.2	3.6.2	3.6.2	Containment Air Locks	Not applicable to NuScale design. No corresponding credited features.
3.6.2	3.6.3	3.6.3	3.6.3	Containment Isolation Valves	NuScale specifications generally consistent with ISTS.
	3.6.4	3.6.4	3.6.4	Containment Pressure	NuScale containment pressure is monitored and automatically actuates appropriate safety functions if out of limits.
	3.6.5	3.6.5	3.6.5	Containment Air Temperature	NuScale containment air temperature is established by conditions in the containment and not controllable.
(3.5.3)	3.6.6			Passive Containment Cooling System	Not applicable to NuScale design. Containment cooling accomplished by contact heat transfer from containment shell into reactor pool addressed in TS 3.5.3.
		3.6.6	3.6.6	Containment Spray and Cooling Systems	
		3.6.6		Quench Spray (QS) System	
		3.6.6		Recirculation Spray (RS) System	
		3.6.14		Air Return System (ARS)	
		3.6.15		Ice Bed (Ice Condenser)	
		3.6.16		Ice Condenser Doors (Ice Condenser)	
		3.6.17		Divider Barrier Integrity (Ice Condenser)	
		3.6.18		Containment Recirculation Drains (Ice Condenser)	
	3.6.9	3.6.12	3.6.12	Vacuum Relief Valves	
		3.6.10		Hydrogen Ignition System (HIS)	
		3.6.9	3.6.9	Hydrogen Mixing System (HMS)	
				Containment Oxygen Concentration	
		3.6.8	3.6.11	Shield Building	Not applicable to NuScale design. No corresponding credited features.
		3.6.13	3.6.8	Shield Building (Exhaust) Air Cleanup System	

Specification Number					
NuScale	**NUREG-2194 (AP1000)**	**NUREG-1431 (W)**	**NUREG-1432 (Digital) (CE)**	**Specification Title**	**Discussion**
3.6.2	3.6.7			Containment Penetrations	NuScale specification generally consistent with ISTS.
		3.7		**PLANT SYSTEMS**	
3.7.1		3.7.2	3.7.2	Main Steam Isolation Valves (MSIVs)	NuScale specifications generally consistent with ISTS, modified to reflect NuScale design.
3.7.2	3.7.3	3.7.3	3.7.3	Main Feedwater Isolation and Control Valves	
(3.5.2)	3.7.1	3.7.1	3.7.1	Main Steam Safety Valves (MSSVs)	Not applicable to NuScale design. DHRS in TS 3.5.2 provides secondary system decay heat removal.
	3.7.2			Main Steam Line Flow Path Isolation Valves	Not applicable to NuScale design. Manages via specific activity limit and the use of DHR to remove heat to the UHS if needed.
	3.7.8			Main Steam Line Leakage	Not applicable to NuScale design. No corresponding credited features.
3.7.3				In-Containment Secondary Piping Leakage	NuScale-specific LCO added to address the leak-before-break criteria as applied to secondary piping inside the containment.
	3.7.10			Steam Generator (SG) Isolation Valves	Not applicable to NuScale design. No corresponding credited features.
(3.4.8)	3.7.4	3.7.18	3.7.19	Secondary Specific Activity	Not applicable to NuScale design. Once-through SG inventory is accounted for or otherwise addressed consistent with limits established on primary activity in TS 3.4.8.
(3.5.2)		3.7.4	3.7.4	Atmospheric Dump Valves (ADVs)	Not applicable to NuScale design. DHRS in TS 3.5.2 provides secondary system decay heat removal.
		3.7.5	3.7.5	Auxiliary Feedwater (AFW) System	
		3.7.6	3.7.6	Condensate Storage Tank (CST)	
	3.7.6			Main Control Room Habitability System (VES)	Not applicable to NuScale design. No corresponding credited features.
		3.7.10	3.7.11	Control Room Emergency Filtration (Air Cleanup) System	
		3.7.11	3.7.12	Control Room Emergency Air Temperature Control System (CREATCS)	
	3.7.7			Startup Feedwater Isolation and Control Valves	Not applicable to NuScale design. No corresponding credited features.
(3.5.3)		3.7.7	3.7.7	Component Cooling Water (CCW) System	Not applicable to NuScale design. Ultimate Heat Sink TS 3.5.3 establishes temperature limits and inventory of the reactor pool, which removes decay heat and acts as the UHS.
			3.7.10	Essential Chilled Water	
		3.7.8	3.7.8	Service Water System (SWS)	
		3.7.9	3.7.9	Ultimate Heat Sink (UHS)	

Specification Number					
NuScale	**NUREG-2194 (AP1000)**	**NUREG-1431 (W)**	**NUREG-1432 (Digital) (CE)**	**Specification Title**	**Discussion**
		3.7.12	3.7.13	Emergency Core Cooling System (ECCS) Pump Room Exhaust Air Cleanup System (PREACS)	Not applicable to NuScale design. No corresponding credited features.
		3.7.13	3.7.14	Fuel Building Air Cleanup System (FBACS)	
		3.7.14	3.7.15	Penetration Room Exhaust Air Cleanup System (PREACS)	
(3.5.3)	3.7.11	3.7.16	3.7.17	Spent Fuel Pool Boron Concentration	Ultimate Heat Sink TS 3.5.3 establishes limits on boron concentration in the refueling area as well as the rest of the pool.
	3.7.9			Spent Fuel Pool Makeup Water Sources	
	3.7.5	3.7.15	3.7.16	Fuel Storage Pool Water Level	
	3.7.12	3.7.17	3.7.18	Spent Fuel Pool Storage	
N/A		**3.8**		**ELECTRICAL POWER SYSTEMS**	
	3.8.1	3.8.4	3.8.4	DC Sources - Operating	Not applicable to NuScale design. No corresponding credited features.
	3.8.2	3.8.5	3.8.5	DC Sources - Shutdown	
		3.8.1	3.8.1	AC Sources - Operating	
		3.8.2	3.8.2	AC Sources - Shutdown	
	3.8.3	3.8.7	3.8.7	Inverters - Operating	
	3.8.4	3.8.8	3.8.8	Inverters - Shutdown	
	3.8.5	3.8.9	3.8.9	Distribution Systems - Operating	
	3.8.6	3.8.10	3.8.10	Distribution Systems - Shutdown	
	3.8.7	3.8.6	3.8.6	Battery Parameters	
		3.8.3	3.8.3	Diesel Fuel Oil, Lube Oil, and Starting Air	

Specification Number					
NuScale	**NUREG-2194 (AP1000)**	**NUREG-1431 (W)**	**NUREG-1432 (Digital) (CE)**	**Specification Title**	**Discussion**
3.8		**3.9**		**REFUELING OPERATIONS**	
(3.5.3)	3.9.1	3.9.1	3.9.1	Boron Concentration	Ultimate Heat Sink TS 3.5.3 establishes limits on boron concentration in the refueling area as well as the rest of the pool.
	3.9.2			Unborated Water Source Flow Paths	
		3.9.2		Unborated Water Source Isolation Valves	
3.8.1	3.9.3	3.9.3	3.9.2	Nuclear Instrumentation	NuScale specification generally consistent with ISTS, modified to reflect NuScale design.
				Refueling Equipment Interlocks	
3.8.2	3.9.5			Decay Time	NuScale specification generally consistent with ISTS.
	3.9.4	3.9.7	3.9.6	Refueling (Cavity) Water Level	
		3.9.4	3.9.3	Containment Penetrations	
(3.5.3)			3.9.4	Shutdown Cooling and Coolant Circulation - High Water Level	Ultimate Heat Sink TS 3.5.3 establishes limits on water level in the refueling area as well as the rest of the pool.
		3.9.5		Residual Heat Removal (RHR) and Coolant Circulation - High Water Level	
		3.9.6		Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level	
			3.9.5	Shutdown Cooling and Coolant Circulation - Low Water Level	
	4.0			**DESIGN FEATURES**	
4.1	4.1	4.1	4.1	Site Location	NuScale conforms – COL Information.
4.2	4.2	4.2	4.2	Reactor Core	NuScale conforms.
4.3	4.3	4.3	4.3	Fuel Storage	NuScale conforms.

Specification Number					
NuScale	NUREG-2194 (AP1000)	NUREG-1431 (W)	NUREG-1432 (Digital) (CE)	Specification Title	Discussion
5.0				**ADMINISTRATIVE CONTROLS**	
5.1	5.1	5.1	5.1	Responsibility	NuScale conforms.
5.2	5.2	5.2	5.2	Organization	NuScale conforms.
5.3	5.3	5.3	5.3	Facility Staff Qualifications	NuScale conforms.
5.4	5.4	5.4	5.4	Procedures	NuScale conforms.
5.5	5.5	5.5	5.5	Programs and Manuals	NuScale conforms, see Table 3-7 for individual program and manual applicability.
5.6	5.6	5.6	5.6	Reporting Requirements	NuScale conforms.
5.7	5.7	5.7	5.7	High Radiation Area	NuScale conforms.

Appendix C Industry / NRC STS Traveler Consideration

The NuScale power plant design is different from previously licensed nuclear power plants. Plant operations are also different from previously operating nuclear power plants. Experience and lessons learned from the improved technical specifications were extensively considered during development of the proposed GTS.

Consideration of the contents of travelers does not imply direct correspondence or functional equivalent unless described as such. The NuScale design is not addressed in the traveler process, so none of the travelers are explicitly applicable to the NuScale GTS. Rather the intent of the traveler was considered based on available information related to the changes made or proposed to the STS. The term 'implemented' as used below indicates the traveler changes were made to the extent practicable and appropriate for the NuScale design.

The table provides details of the extent of consideration of features from the listed STS travelers that correspond with specifications included in the proposed NuScale GTS.

The travelers that are were considered are those that were issued as new or revised since the earliest manuscript date of the NUREG STS, October 2011, and by comparison of the traveler content with the contents of the STS with the changes identified in the TSTF.

Table C-1 Industry / NRC STS traveler evaluation

Traveler No. Revision	Subject	Typically Affected STS Specifications	NuScale Consideration	NuScale Specifications Affected	Discussion
426-A Revision 5	Revise or Add Actions to Preclude Entry into LCO 3.0.3 - RITSTF Initiatives 6b & 6c	Numerous CE PWR specifications	Not incorporated. The topical report does not apply to NuScale.	None	The proposed NuScale TS including operational paradigm is significantly different from that addressed in the traveler. The TS have been written to minimize the potential for conditions leading to explicit or default entry into LCO 3.0.3
432-A Revision 1	Change in Technical Specifications End States (WCAP-16294)	Numerous W PWR specifications	Not incorporated. The topical report does not apply to NuScale.	None	The proposed NuScale TS including operational paradigm is significantly different from that addressed in the traveler.
454 Revision 3	Staggered Integrated ESFAS Testing (WCAP-15830)	CE PWR ESFAS and ESF surveillance tests	Withdrawn - not incorporated. The topical report does not apply to NuScale.	None	The proposed NuScale design uses ESFAS and ESF systems that are not similar to those used in Combustion Engineering PWRs. The proposed NuScale TS are based on that design.
490 Revision 1	Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec	PWR 1.1, Definitions 3.4 specifications on RCS Specific Activity	Addressed. The proposed NuScale TS generally implement the traveler changes modified to reflect the NuScale specific limits.	TS 3.4.8, RCS Specific Activity	The NRC staff requested use of RCS specific activity limits more conservative than the traveler and 10 CFR 50.36. See Table A-1 entry for LCO 3.4.8.
493-A Revision 4	Clarify Application of Setpoint Methodology for LSSS Functions	BWR and PWR 3.3 instrumentation specifications	Addressed. The proposed NuScale TS implement Option B of the traveler through inclusion of an SP in Section 5.5.	3.3, Instrumentation 5.5, Programs	The Setpoint Program description is patterned after the NUREG-2194, Rev 0, W-AP1000-STS, Section 5.5.14 requirements that are more appropriate for a new reactor design generic TS and the 10 CFR 52 licensing process.
494-T Revision 2	Correct Bases Discussion of Figure B 3.0-1	PWR and BWR Bases for LCO 3.0.6	Not incorporated.	None	NuScale has not incorporated the expanded explanation provided by the traveler, consistent with NUREG-2194, Rev. 0 and the ESBWR GTS that did not incorporate the traveler.
501-A Revision 1	Relocate Stored Fuel Oil and Lube Oil Volume Values to Licensee Control	BWR and PWR specifications 3.8.3, Diesel Fuel Oil, Lube Oil, and Starting Air	Not incorporated.	None	The NuScale design does not require or include safety-related onsite diesel generators. Therefore, no corresponding specification is proposed, and the traveler is not applicable.

Traveler No. Revision	Subject	Typically Affected STS Specifications	NuScale Consideration	NuScale Specifications Affected	Discussion
502-T Revision 1	Correct Containment Isolation Valve Bases Regarding Closed Systems	BWR and PWR Bases for specifications on containment isolation valves	Addressed.	3.6.2, Containment Isolation Valves	The proposed NuScale Bases incorporate the corrected wording in the Bases.
504-T Revision 0	Revised the MSIV and MFIV Specifications to Provide Actions for Actuator Trains	PWR 3.7.1, Main Steam Isolation Valves and 3.7.2, Main Feedwater Isolation Valves	Not incorporated.	None	The NuScale MSIV and feedwater isolation valve design do not incorporate dual actuators such that the traveler changes should be incorporated.
505-A Revision 1	Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b	Numerous specifications in NUREGs	Not incorporated.	None	NuScale has chosen not to incorporate this traveler into the proposed GTS.
513-A Revision 3	Revise PWR Operability Requirements and Actions for RCS Leakage Instrumentation	PWR TS related to RCS Leakage Detection Instrumentation	Addressed. The contents of the traveler were considered during construction of proposed NuScale TS 3.4.7, RCS Leakage Detection Instrumentation.	3.4.7, RCS Leakage Detection Instrumentation	The NuScale leakage detection methods are significantly different from those used in PWRs. However, the content of the traveler was used to inform the construction of the proposed NuScale specification.
514-A Revision 3	Revise BWR Operability Requirements and Actions for RCS Leakage Instrumentation	NUREG-1433 and 1434 GE PWR TS related to RCS Leakage Detection Instrumentation	Not applicable.	None	NuScale leakage detection instrumentation and methods are not similar to those used in GE BWRs.
515 Revision 0	Revise Post-Accident Monitoring Instrumentation based on Reg. Guide 1.97, Rev. 4 and NEDO-33349	NUREG-1433 and 1434 GE PWR TS 3.3.3, Post Accident Monitoring (PAM) Instrumentation	Not applicable.	None	Withdrawn by TSTF. Also, the NuScale design does not include any PAM instrumentation that meets the threshold for inclusion in the TS.
520-T Revision 0	Correct Specification 3.1.4, Required Action A.1 Bases	CE PWR NUREG-1432 TS 3.1.4, CEA Alignment Bases	Not applicable.	None	The proposed NuScale TS Bases do not include the conflicting statements.
521 Revision 0	Exclusion of Time Constants from Channel Operational Tests in Specifications 3.3.1 and 3.3.2	Westinghouse PWR NUREG-1431 TS 3.3.1 and 3.3.2	Not applicable.	None	The NuScale protective instrumentation does not include functions similar to the Westinghouse PWR design that this traveler is applicable to.
522-A Revision 0	Revise Ventilation System Surveillance Requirements to Operate for 10 hours per Month	BWR and PWR ventilation system SRs	Not applicable.	None	The NuScale design does not include credited ventilation systems and no TS are proposed.

Traveler No. Revision	Subject	Typically Affected STS Specifications	NuScale Consideration	NuScale Specifications Affected	Discussion
523 Revision 2	Generic Letter 2008-01, Managing Gas Accumulation	All BWR and PWR credited systems that have potential to be adversely impacted by gas accumulation	Addressed. The NuScale DHRS was conservatively determined to have the potential for accumulation of non-condensible gases. Instrumentation is provided to permit monitoring of the volume where gases could accumulate, and safety analyses are performed assuming the presence of gases in the volume above the instrumentation.	3.5.2, Decay Heat Removal System	NuScale design incorporates design features to detect postulated accumulation of non-condensible gases and safety analyses are conservatively performed assuming gases are present in the quantity that could exist before indication of their presence.
524-T Revision 0	Clarify the Application of SR 3.0.2 to SR 3.1.3.2, MTC	NUREG-1431 Westinghouse PWR Bases for SR 3.1.3.2, MTC	Not applicable.	None	The NuScale MTC specification SR does not include Notes that correspond directly with those in the NUREG-1431 TS and the NuScale Bases are consistent with the proposed specifications.
525 Revision 0	Post Accident Monitoring instrumentation Requirements (WCAP-15981-NP-A)	NUREG-1431 Westinghouse PWR TS 3.3.3, Post Accident Monitoring (PAM) Instrumentation	Not applicable. Traveler is specific to PAM instrumentation selection for Westinghouse designs.	None	The NuScale design does not include any PAM instrumentation that meets the threshold for inclusion in the TS.
526-T Revision 0	Clarify LCO 3.8.1 SR Notes Regarding Momentary Transients Outside the Load Band	AC Sources – Operating for BWRs and PWRs	Not applicable.	None	The NuScale design does not depend on emergency AC power sources and there are no corresponding requirements in the proposed NuScale TS.
527-T Revision 0	Incorporate Model Application Commitments as Reviewer's Notes	Numerous various BWR and PWR	Not applicable. Traveler describes the use of Reviewer’s Notes in the Bases of the published STS.	None	The traveler describes the management and identification of commitments into travelers and Bases. The proposed NuScale TS are based on the licensing basis provided in the DCA.
528-T Revision 0	Bracket Accident Analysis Discussion in LCO 3.4.4	Bases for LCO 3.4.4, RCS Loops – MODES 1 and 2, of B&W plants	Not applicable.	None	The NuScale plant does not include ‘loops’ or associated TS. The proposed NuScale Bases reflect the safety analyses applicable to the design and the use of brackets for non-COLA items is contrary to DC/COL-ISG-8.
529-A Revision 4	Clarify Use and Application Rules	1.3, Completion Times, and 3.0 LCO Applicability for BWRs and PWRs	Addressed. The proposed NuScale SR 3.0.3 incorporates the content of the Traveler.	1.3, Completion Times 3.0, LCO Applicability	
530 Revision 0	Clarify SR 3.0.3 to be Consistent with Generic Letter 87-09	SR 3.0.3 SR Applicability for BWRs and PWRs	Addressed. The proposed NuScale SR 3.0.3 incorporates the content of the Traveler	TS SR 3.0.3, SR Applicability	

Traveler No. Revision	Subject	Typically Affected STS Specifications	NuScale Consideration	NuScale Specifications Affected	Discussion
531 Revision 0	Revision of Specification 3.8.1, Required Actions B.3.1 and B.3.2	AC Sources – Operating for BWRs and PWRs	Not applicable.	None	The NuScale design does not depend on emergency AC power sources and there are no corresponding requirements in the proposed NuScale TS.
532-T Revision 0	Eliminate Incorrect Reference to Appendix R in the Remote Shutdown System Bases	Bases of NUREG-1432 CE PWR Remote Shutdown System specification	Not applicable.	None	The incorrect reference in the CE Bases is not included in the NuScale Bases for the RSS, TS 3.3.5.
533-T Revision 0	Remove COLR and PTLR Revision and Date Relocation Provisions Added by travelers 363, 408, and 419	COLR and PTLR administrative specifications of BWRs and PWRs	The NuScale administrative specifications that describe the COLR and PTLR will include the number, title, date, and NRC-approved document describing the methodology. The bracketed reviewer's note was revised as requested by the staff in the response to RAI 16-43.	5.6, Reporting Requirements	The proposed NuScale specifications were drafted generally consistent with the intent of the traveler.
534 Revision 0	Clarify Application of Pressure Boundary Leakage Definition	BWR and PWR operational leakage specifications	Addressed. Incorporated into the Bases of 3.4.5, RCS Operational Leakage	3.4.5, RCS Operational LEAKAGE	Considered and incorporated consistent with the NuScale design.
535 Revision 0	Revise Shutdown Margin Definition to Address Advanced Fuel Designs	BWR SDM Definition	Not applicable.	None	The NuScale definition of shutdown margin is consistent with usage of the term in the design and operation of NuScale facilities.
536 Revision 0	Resolve CE Digital TS Inconsistencies Regarding CPCs and CEACs	CE PWR instrumentation and control specifications	Addressed. The NuScale digital control system does not include CE CPC or CEACs, however the underlying purpose of the traveler was considered in the development of the Actions and Surveillance Requirements applicable to the corresponding NuScale specifications.	3.3.1, Module Protection System 3.3.2, Reactor Trip System Logic and Actuation 3.3.3, Engineered Safety Features Actuation System Logic and Actuation 3.3.4, Manual Actuation Functions	The NuScale TS considered the reason for and changes proposed to the STS by the traveler. The specification Actions and Surveillance Requirements do not include conditions unrelated to system Operability.
537 Revision 0	Increase CIV Completion Time; Update of traveler 373	CE PWR containment isolation valve specifications	Not applicable. Traveler is based on a risk-informed technical basis applicable to CE designed plants.	None	The NuScale design is not consistent with the CE design and the technical basis for the traveler is not applicable to the NuScale design.
538 Revision 0	Add Actions to Preclude Entry into LCO 3.0.3 - RITSTF Initiatives 6b & 6c	B&W PWR containment spray and cooling systems, and emergency ventilation systems	Not applicable.	None	The NuScale design does not include a containment spray system or emergency ventilation systems. Containment cooling is a passive function utilizing heat transfer through the containment vessel walls to the reactor pool. There are no credited safety-related ventilation systems in the design that need TS.

Traveler No. Revision	Subject	Typically Affected STS Specifications	NuScale Consideration	NuScale Specifications Affected	Discussion
539-T Revision 0	Correction of Post-Accident Monitoring Instrumentation Bases	Post-Accident Monitoring Instrumentation specifications in 3.3 of BWRs and PWRs	Not applicable. The proposed NuScale TS do not include any PAM requirements.	None	The NuScale design does not include any variables that result in inclusion of a PAM technical specification.
540 Revision 1	Provide Completion Times in Lieu of Immediate Shutdown	BWR specifications for secondary containment and control room ventilation and filtration systems.	Not applicable. The proposed NuScale design does not credit safety-related SSC that perform a function similar to those addressed in the traveler.	None	The NuScale design does not incorporate a containment gas treatment system similar to that used by the secondary containment design of BWRs. Nor does the NuScale design credit the control room ventilation systems with performing a function that is required to be performed in response to a DBE.
541 Revision 1	Add Exceptions to Surveillance Requirements When the Safety Function is Being Performed	Various specifications of BWRs and PWRs	Although not directly applicable, the intent of the traveler was adopted in the NuScale GTS. NuScale safety-related reactor trip system and engineered safety features components are credited with a single safety-related position, each of which is achieved by the component being de-energized.	3.1.9, Boron Dilution Control 3.3.1, MPS Instrumentation 3.3.2, Reactor Trip System Logic and Actuation 3.3.3, Engineered Safety Feature Actuation System Logic and Actuation 3.4.6, Chemical and Volume Control System Isolation Valves 3.4.10, Low Temperature Overpressure Protection Valves 3.5.2, Decay Heat Removal System 3.6.2, Containment Isolation Valves 3.7.1, Main Steam Isolation Valves 3.7.2, Feedwater Isolation	The implementation of this traveler is under additional review and consideration as requested by the NRC staff at the time this technical report was developed. See RAI 16-28.
542 Revision 1	Reactor Pressure Vessel Water Inventory Control	BWR Instrumentation	Not applicable. The NuScale design and operating paradigm does not include operations at reduced inventories or water levels.	None	The NuScale design and operations, including refueling activities, will not result in a potential for water inventory in the reactor vessel to be reduced to the level of the fuel. All refueling operations are conducted with the reactor vessel and fuel remaining submerged in the reactor pool.

Traveler No. Revision	Subject	Typically Affected STS Specifications	NuScale Consideration	NuScale Specifications Affected	Discussion
543-T Revision 0	Clarify Verification of Time Constants	Westinghouse PWR Instrumentation	Not applicable.	None	The NuScale protective instrumentation does not include functions similar to the Westinghouse PWR design that this traveler is applicable to.
545-A Revision 3	TS Inservice Testing Program Removal & Clarify SR Usage Rule Application to Section 5.5 Testing	PWR and BWR Surveillance Requirements and 5.5 Programs	Addressed. The changes described in the traveler were implemented in appropriate locations throughout the proposed NuScale GTS.	1.1, Definitions 3.4.4, Reactor Safety Valves 3.4.6, CVCS Isolation Valves 3.4.10, Low Temperature Overpressure Protection (LTOP) Valves 3.5.1, Emergency Core Cooling System 3.5.2, Decay Heat Removal System 3.6.2, Containment Isolation Valves 3.7.1, MSIVs 3.7.2, Feedwater Isolation	The program was incorporated into the Definition section. Surveillance requirements applicable to similar components associated with functions or SSCs in the GTS were revised to be consistent with the traveler. Consistent with the traveler, the IST program description is not provided in 5.5 Programs.
546 Revision 0	Revise APRM Channel Adjustment Surveillance Requirement	BWR 3.3.1, RPS Instrumentation	Addressed. The NuScale design does not incorporate APRMs, however the excore neutron monitoring system that provides a similar function incudes requirements for calibration by comparison with a heat balance. The limits on acceptable deviation between the neutron flux monitor indication and the value measured by heat balance distinguishes between conservative and non-conservative differences, and establishes a limit and required actions to make adjustments if the difference is not in the conservative direction.	3.3.1, Module Protection System Instrumentation	The allowances provided by the traveler are incorporated in the proposed NuScale GTS.
547-A Revision 1	Clarification of Rod Position Requirements	Westinghouse 3.1, Reactivity specifications related to rod position requirements.	Not incorporated. The additional operational flexibility provided by this traveler is not required due to the core design and margins inherent in the operating limits established by the COLR.	3.1, Reactivity Control	The NuScale core design is significantly different from that of large PWRs. The traveler was not incorporated because the proposed changes are not necessary.
548-T Revision 0	Safety Function Determination Program Changes for Consistency	Westinghouse and GE BWR4 5.5 Program descriptions	Addressed. The NuScale SFDP description provided in 5.5.8 is consistent with the intended content as previously described in NUREGs 1430, 1432, 2194, etc.	5.5.8, Safety Function Determination Program (SFDP)	Addressed by specification that is consistent with the appropriate STS.
549-T Revision 0	Correct Quadrant Power Tilt Ratio Required Action Bases	Westinghouse 3.2.4 Bases	Not Applicable. The NuScale design does not include monitoring of a QPTR or QPTR-like variable. The traveler is specific to an inappropriate wording that existed in the Bases of NUREG-1431, Rev. 1.	None	

Traveler No. Revision	Subject	Typically Affected STS Specifications	NuScale Consideration	NuScale Specifications Affected	Discussion
550-T Revision 1	Correct Misleading Bases Statements in Systems not Required to be Operable in Shutdown Modes	PWR and BWR Bases for systems that perform a support function for other systems required to be operable when the facility is shutdown. Specifically cooling water systems.	Not applicable. The NuScale design uses a large reactor pool as the UHS during operational modes and during transition and refueling operations. Specification 3.5.3 is 'at all times' and the Bases reflect this. There are no other corresponding systems in the NuScale design that are required to be operable during operations, but that provide support functions during shutdown conditions.	None	
551 Revision 3	Revise Secondary Containment Surveillance Requirements	BWR Bases of 3.6.4, Secondary Containment	Not applicable. The NuScale design does not include or credit a secondary containment or similar functional boundary and does not include a corresponding specification.	None	
553 Revision 1	Add Action for Two Inoperable CREATCS Trains	PWR 3.7.11, Control Room Emergency Air Temperature Control systems	Withdrawn – not addressed. The NuScale design does not credit a CREATCS or similar functional boundary and does not include a corresponding specification.	None	
555-T Revision 0	Clarify the Nuclear Instrumentation Bases Regarding the Detection of an Improperly Loaded Fuel Assembly	PWR Bases for 3.9 Nuclear Instrumentation specifications	Addressed. The NuScale design includes neutron flux instrumentation at the refueling tool that corresponds to and performs a function similar to that of the source range neutron monitors used at PWRs.	3.8.1, Nuclear Instrumentation	Bases for TS 3.8.1 do not include a description of an ability to detect an improperly loaded fuel assembly.
556-T Revision 1	Modify TS 3.8.1 and TS 3.8.2 Bases to Address an Open Phase Condition	PWR and BWR 3.8, AC Sources	Not applicable. NuScale design does not credit electrical power and therefore does not include corresponding TS.	None	
557-T Revision 1	Addition of Spent Fuel Rack Neutron Absorber Monitoring Program	BWR and PWR 5.5, Programs	Incorporated. NuScale design includes spent fuel racks that use neutron absorber material.	5.5.12, Spent Fuel Storage Rack Neutron Absorber Monitoring Program	Incorporated as TS 5.5.12.
558-T Revision 0	Clarify SR Bases added by TSTF-523	PWR and BWR specifications related to ECCS, decay heat removal, RHR, SDC and Containment Spray systems.	Not applicable. No corresponding SSC or function in the NuScale design. The NuScale DHRS was conservatively determined to have potential for accumulation of non-condensible gases. Instrumentation is provided to permit monitoring of the volume where gases could accumulate and safety analyses are performed assuming the presence of gases in the volume above the instrumentation.	3.5.2, Decay Heat Removal System	NuScale design incorporates features to detect postulated accumulation of non-condensible gases and safety analyses are conservatively performed assuming gases are present in the quantity that could exist before indication of their presence.
559-T Revision 0	Revise Bases to Reflect Revised SL Pressure Limit	BWR Bases for 2.1.1, 3.3.1, and 3.3.6	Not applicable. Traveler resolves an issue specific to the GE design that does not correspond to a NuScale SSC or function.	None	
560-T Revision 0 7/20/16	Addition of Surveillance Requirements Note for LCO 3.7.7 (BWR/4) and LCO 3.7.6 (BWR/6)	BWR 3.7.7 or 3.7.6, Turbine Bypass System	Not applicable. No corresponding SSC or function in the NuScale design that is credited or otherwise would result in inclusion in the TS.	None	

Traveler No. Revision	Subject	Typically Affected STS Specifications	NuScale Consideration	NuScale Specifications Affected	Discussion
561-T Revision 0	Bracket LCO 3.5.1 LCO Note in the ISTS	BWR 3.5.1, ECCS	Not applicable. The NuScale design does not require the allowance provided to some BWRs by this traveler.	None	
562-T Revision 0	Bases Clarification for TS 3.8.1, Required Actions B.3.1 and B.3.2	PWR and BWR 3.8.1, AC Sources Operating	Not applicable. NuScale design does not credit electrical power and therefore does not include corresponding TS.	None	
563 Revision 0	Revise Instrument Testing Definitions to Incorporate the Surveillance Frequency Control Program	1.1, Definitions	Incorporated. NuScale has adopted changes to the definitions to permit application of the SFCP to testing in a manner similar to that described in the traveler.	1.1, Definitions	Definitions for ACTUATION LOGIC TEST, CHANNEL CALIBRATION, and CHANNEL OPERATIONAL TEST (COT) were modified from previous STS definitions.
564 Revision 1	Safety Limit MCPR	BWR 2.1.1, Safety Limits	Not applicable. The traveler is related to calculating the MCPR limit at BWRs. The NuScale design uses a design-specific methodology for calculating core parameters and limits.	None	
565 Revision 1	Revise the LCO 3.0.2 and LCO 3.0.3 Bases	LCO 3.0.2 Bases and LCO 3.0.3 Bases	Addressed as described in response to RAI 16-9S1.	LCO 3.0.2 and LCO 3.0.3 Bases	See RAI response 16-9S1.
566 Revision 0	Revise Actions for Inoperable RHR Shutdown Cooling Subsystems	BWR RHR Shutdown Cooling System	Not Applicable - limited to BWR plants. The NuScale passive shutdown cooling design does not include component or configuration issues similar to those addressed in this traveler.	None	
567-A Revision 1	Add Containment Sump TS to Address GSI-191 Issues	PWR ECCS LCOs and new LCO for Containment Sump operability	Not Applicable. The NuScale containment and recirculation occurs directly from the reactor pressure vessel to the containment volume via the ECCS valves. There are no equivalent components to those addressed in the traveler.	None	
568 Revision 0	Clarify Applicability of BWR/4 TS 3.6.2.5 and TS 3.6.2.3	BWR Drywell to Suppression Chamber pressure and Primary Oxygen Concentration LCO	Not Applicable. The NuScale design does not include a BWR-like drywell and suppression containment design. The NuScale containment operates at a very low pressure to support RCS leakage detection OPERABILITY as required by LCO 3.4.5.	None	
569 Revision 0	Revise Response Time Testing Definition	1.1 Definitions 3.3 LCO that surveil RTS and ESFAS Response Times	Not Applicable. The NuScale digital safety system implemented in LCOs 3.3.1, 3.3.2, and 3.3.2 is significantly different from existing PWR systems and response time testing is conducted in a different manner.	None	The NuScale protection system is different from existing plant designs. This results in the need for different testing boundaries and approaches. DCA Revision 2 incorporates the revised methods and approach to response time testing.
570-T Revision 0	Revise Turbine Stop Valve and Turbine Control Valve Closure RPS Function Bases	BWR RPS SR Bases	Not Applicable. NuScale design does not include or credit a reactor trip that is initiated by a turbine trip and does not include corresponding TS.	None	