16 TECHNICAL SPECIFICATIONS

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review of Chapter 16, "Technical Specifications," of the NuScale Power, LLC (hereinafter referred to as the applicant), US460 Standard Design Approval Application (SDAA), Part 2, "Final Safety Analysis Report (FSAR)," Revision 2 (Agencywide Documents Access and Management System Accession No. ML25099A237) and the referenced proposed generic technical specifications (GTS) and associated GTS bases (Bases) in SDAA Part 4, "US460 Generic Technical Specifications," (ML25099A279 and ML25099A280). Together, these portions of the SDAA constitute the information related to technical specifications (TS) in the NuScale SDAA.

The staff's regulatory findings documented in this SER are based on Revision 2 of the SDAA, dated April 9, 2025 (ML25099A237). The precise parameter values, as reviewed by the staff in this safety evaluation, are provided by the applicant in the SDAA using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this safety evaluation to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate and is presented first, followed by the applicant-provided parameter value in English units within parentheses. If only one value appears in either SI or English units, it is directly quoted from the SDAA and not converted.

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.36(a)(1) requires that each license authorizing operation of a utilization facility (i.e., an operating license (OL) or a combined license (COL)) issued by the Commission contain TS that set forth the safety limits (SLs), limiting safety system settings (LSSSs), limiting conditions for operation (LCOs), and other limitations on facility operation that are necessary for adequate protection of public health and safety. In addition, 10 CFR 50.36(a)(1) requires that each application for an OL or a COL include a "summary statement of the bases or reasons for such TS."

The regulations in 10 CFR Part 52, Subpart E, "Standard Design Approvals," do not require an applicant to propose TS as part of an SDAA. For this design, however, the applicant elected to propose GTS and Bases in its SDAA as it had done in the NuScale US600 design certification application (DCA) in accordance with 10 CFR 50.36, 10 CFR 50.36a, and 10 CFR 52.47(a)(11). This is consistent with the Commission's expectation that an SDAA includes proposed GTS.[1] In light of this expectation, the staff reviewed NuScale's proposed GTS and Bases included in the SDAA for the US460 design by applying the same standards as the staff would have applied to a DCA. This SER Chapter presents the results of the staff's review of the proposed GTS and Bases.

In 10 CFR 50.36(a)(2), the NRC requires that each applicant for a design certification include in its application proposed GTS in accordance with the requirements of this section

[1] When making changes to 10 CFR Part 52 in 2007, the Commission explained the role of proposed TS in a Standard Design Approval (SDA) application and in a final design approval (FDA). The Commission discussed, among other things, SDAs under Subpart E to 10 CFR Part 52, 10 CFR §§ 52.131 through 52.147 and stated that it decided that the contents of applications for design approvals should contain essentially the same technical information required of design certification applications (Licenses, Certifications, and Approvals for Nuclear Power Plants, 72 Fed. Reg. at 49,352, 49,390-49,391 (August 28, 2007) (https://www.federalregister.gov/d/07-3861/p-366). In a separate discussion concerning TS, the Commission discussed which applications for various approvals under 10 CFR Part 52 require applicants to submit TS (72 Fed. Reg. at 49,398) (https://www.federalregister.gov/d/07-3861/p-421).

(i.e., 10 CFR 50.36, “Technical Specifications”) for the portion of the plant that is within the scope of the DCA but does not explicitly require including GTS Bases. Because the staff needs to find that the rationale for each GTS requirement is consistent with the proposed design, as described in DCA Part 2, it is customary for a design certification applicant to include a summary statement of the bases or reasons for the proposed GTS (i.e., Bases) in DCA Part 4, using the formatting conventions and applicable contents of the standard TS (STS) Bases.

In 10 CFR 50.36a(a)(1), the NRC requires, among other things, that each applicant for a design certification include TS that require that “[o]perating procedures developed pursuant to [Section] 50.34a(c) for the control of effluents be established and followed and that the radioactive waste system, pursuant to [Section] 50.34a, be maintained and used.”

The regulations in 10 CFR 52.47(a)(11) and 10 CFR 52.79(a)(30) state that a design certification applicant and a COL applicant, respectively, are to propose TS prepared in accordance with 10 CFR 50.36 and 10 CFR 50.36a, “Technical specifications on effluents from nuclear power reactors.” COL applicants that reference a certified design or a standard design approval are to propose plant-specific TS and Bases, which would include the GTS and Bases approved during the design certification or, if applicable, reviewed as part of the standard design approval review. The COL applicant may propose deviations from the approved GTS or Bases before issuance of the COL by requesting an exemption from the associated appendix to 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” that codifies the certified design. If referencing a standard design approval, however, the requirement for an exemption for such a deviation would not apply. A holder of a COL may propose changes to the plant-specific TS in accordance with 10 CFR 50.90, “Application for amendment of license, construction permit, or early site permit.”

16.1 Introduction

The staff review of the GTS and Bases is for completeness and correctness in regard to NRC requirements and conformance with applicable guidance, and for consistency with related portions of NuScale SDAA Part 2. In SDAA Part 2, Chapter 16, and in SDAA Part 4, the applicant proposed GTS and Bases in accordance with 10 CFR 50.36, 10 CFR 50.36a, and 10 CFR 52.47(a)(11). The GTS are derived from the analyses and evaluations in NuScale SDAA Part 2.

16.2 Summary of Application

SDAA Part 2 contains the NuScale US460 design FSAR. FSAR Chapter 16 addresses the topics listed below related to the proposed GTS and Bases.

GTS Content: Most GTS requirements for this application are written to provide operating limitations on an individual NuScale Power Module (NPM or module) or unit. Operability requirements on some systems (e.g., the reactor pool) and limits on the values of monitored variables (e.g., the reactor pool water level, temperature, and boron concentration) apply to multiple NPMs. The limitations on such systems and variables are applied individually and concurrently to the operation of each applicable NPM. In FSAR Section 16.1.1, “Introduction to Technical Specifications,” the applicant stated the following:

> The [GTS] content differs from the [Improved Standard Technical Specifications (ISTS)] and the previous NuScale US600 technical specifications as necessary to reflect technical differences between large light water reactor (LWR) and the

NuScale US600 [certified] design and the NuScale Power Plant US460 standard design.

The GTS and Bases are formatted consistent with the [Improved] STS, such as NUREG-1431, “Standard Technical Specifications—Westinghouse Plants,” Revision 5, issued September 2021 (W-STS); NUREG-1432, “Standard Technical Specifications—Combustion Engineering Plants,” Revision 5, issued September 2021 (CE-STS); and NUREG-2194, “Standard Technical Specifications for Westinghouse Advanced Passive 1000 Plants,” Revision 1, issued January 2024 (W-AP1000-STS). They are also written consistent with the pressurized-water reactor (PWR) and boiling-water reactor (BWR) owner groups’ Technical Specification Task Force (TSTF) guidelines in TSTF-GG-05-01, Revision 1, “Writer’s Guide for Plant-Specific Improved Technical Specifications,” issued August 2010 (writer’s guide) (ML12046A089).

<u>Selection Criteria for LCOs</u>: Technical Report (TR)-1116-52011-NP, Revision 4, “Technical Specifications Regulatory Conformance and Development,” dated May 20, 2020 (Regulatory Conformance and Development Report) (ML20141L804), which the applicant submitted as part of the US600 DCA, documents the application of the LCO selection criteria of 10 CFR 50.36(c)(2)(ii) to the NuScale US600 design and safety analyses of design-basis accidents (DBAs), anticipated operational occurrences (AOOs), and transients. The Regulatory Conformance and Development Report provides the basis for including the LCOs chosen for the US600 DC GTS and not including LCOs for systems typically addressed by an LCO in the STS NUREGs. TR-101310-NP, Revision 1, “US460 Standard Design Approval Technical Specifications Development,” dated February 2025 (ML25059A411), incorporates by reference the Regulatory Conformance and Development Report and supplements it by identifying differences with the US600 DC GTS and additional approved changes to the STS NUREGs, which the applicant considered in the development of the US460 SDAA GTS.

<u>Completion Times and Surveillance Frequencies</u>: The GTS required action completion times are proposed consistent with those completion times provided in the STS for similar conditions in which the associated LCO is not met. Likewise, the applicant indicated that GTS surveillance requirement (SR) performance frequencies (test intervals) are proposed consistent with the frequencies of similar SRs in the STS. The applicant stated the initial surveillance frequencies and the basis for each frequency in FSAR Table 16.1-1, “Surveillance Frequency Control Program Base Frequencies.” The plant-specific TS issued with a COL or OL referencing the NuScale US460 standard design will include GTS Subsection 5.5.11, “Surveillance Frequency Control Program.” The COL or OL holder will incorporate the information in FSAR Table 16.1-1 in the documentation specified by plant-specific TS Subsection 5.5.11 for implementing the surveillance frequency control program (SFCP).

<u>Consideration of TSTF Traveler Changes to STS</u>: Section 4.2 of the NuScale US600 DCA Regulatory Conformance and Development Report states that available information regarding travelers through June 30, 2018, was considered by NuScale during preparation of the US600 GTS. This traveler information was supplemented by technical report TR-101310-NP in Section 4.0.

SDAA Part 2 and Part 4: The applicant provided proposed GTS and Bases for the NuScale US460 standard design in FSAR Chapter 16, and SDAA Part 4, summarized here, in part, as follows.

The applicant provided the proposed GTS and Bases in SDAA Part 4 for the staff’s review and approval in accordance with 10 CFR 50.36 and 10 CFR 50.36a. In its SDAA, the applicant stated that it had largely developed the GTS and Bases using the W-STS, CE-STS, and

W-AP1000-STS. In support of SDAA Part 4, FSAR Chapter 16 references the DCA Regulatory Conformance and Development Report.

Inspection, Test, Analysis and Acceptance Criteria (ITAAC): There are no ITAAC for this area of review.

Technical Reports: As noted above, the applicant submitted a TS development report as part of the US460 SDAA that supplemented a TS development report prepared as part of the US600 DCA.

Combined License Information: FSAR Chapter 16 lists COL Item 16.1-1, which will account for all instances of bracketed site-specific information in the GTS and Bases. It also lists COL Item 16.1-2, which will require preparation and maintenance of an owner-controlled-requirements manual that includes owner-controlled limits and requirements described in the Bases of the plant-specific TS or as otherwise specified in the FSAR; and COL Item 16.1-3 on the Bases discussion about using assumed conservative time interval values (allocations) for sensor response times to satisfy the SR to verify the channel response time is within limits for module protection system (MPS) instrumentation channels.

16.3 Regulatory Basis

The "Design-Specific Review Standard [DSRS] for the NuScale SMR [Small Modular Reactor] Design," Chapter 16.0, "Technical Specifications," (ML15355A312), which was derived from NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Chapter 16, "Technical Specifications," Revision 3, issued March 2010 (ML100351425), describes the relevant requirements of the NRC's regulations for this area of review, the associated acceptance criteria, and the review interfaces with other DSRS and SRP sections.

Section 182a. of the Atomic Energy Act of 1954, as amended, requires applicants for licenses to operate nuclear power plants to state the following:

> ...such technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization…of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.

In 10 CFR 50.36, the NRC established its regulatory requirements related to the content of TS. In doing so, the NRC placed emphasis on those matters related to the prevention of accidents and the mitigation of accident consequences. As recorded in the Statements of Consideration, "Technical Specifications for Facility Licenses; Safety Analysis Reports" (Volume 33 of the *Federal Register* (FR), page 18610 (33 FR 18610); December 17, 1968), the NRC noted that applicants were expected to incorporate into their TS "...those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity." Accordingly, 10 CFR 50.36(c) requires that TS for utilization facilities contain (1) SLs and LSSSs, (2) LCOs, (3) SRs, (4) design features, and (5) administrative controls.

In 10 CFR 50.36(c)(2)(ii), the NRC requires that an LCO be established in TS for each item meeting one or more of the following four criteria (referred to as LCO selection criteria):

(A) *Criterion 1.* Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

(B) *Criterion 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.

(C) *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.

(D) *Criterion 4.* A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

In accordance with 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 17, "Electric power systems"; GDC 21, "Protection system reliability and testability"; GDC 34, "Residual heat removal"; GDC 35, "Emergency core cooling"; GDC 38, "Containment heat removal"; GDC 41, "Containment atmosphere cleanup"; and GDC 44, "Cooling water," those structures, systems, and components (SSCs) important to safety need to have sufficient independence, redundancy, and testability to perform their safety functions.

In 10 CFR 50.36a, the NRC requires that TS contain procedures for control of radioactive effluents.

In 10 CFR 52.47(a)(11), the NRC requires that a design certification applicant propose TS prepared in accordance with 10 CFR 50.36 and 10 CFR 50.36a. The applicant elected to propose TS in its SDAA for the NuScale US460 to establish approved GTS and Bases for referencing by an applicant in a COL application in developing proposed plant-specific TS.

For the reasons discussed in detail below, the STS documents noted above include the acceptance criteria adequate to meet the above requirements. The STS for PWR designs currently in operation or under construction in the United States appear in four NRC documents: NUREG-1430, "Standard Technical Specifications—Babcock and Wilcox Plants," Revision 5, issued September 2021; NUREG-1431; NUREG-1432; and NUREG-2194. For each document, volume 1 contains the Specifications, and volume 2 contains the associated Bases. The STS include Bases for SLs, LSSSs, LCOs, and associated requirements for applicability, actions, and surveillances. For the reasons discussed below, guidance documents applicable to the NuScale proposed GTS and Bases mostly include portions of the model STS in NUREG-1431 (W-STS), NUREG-1432 (CE-STS), and NUREG-2194 (W-AP1000-STS).

The STS reflect the detailed effort used to apply the criteria discussed in the Interim Policy Statement on "Technical Specification Improvements for Nuclear Power Reactors" (52 FR 3788; February 6, 1987) to generic system functions, which were published in a "Split Report" and issued to the nuclear steam supply system vendor owners' groups in May 1988 (see SER Section 16.4.1.6). In addition, extensive discussions during the development of the STS ensured that the application of the TS criteria and the joint industry and staff plant-specific improved TS writer's guide (which also applies to STS and GTS) would consistently reflect detailed system configurations and operating characteristics for all nuclear steam supply system designs. As such, Bases documents include an abundance of information on the STS model requirements necessary to adequately protect public health and safety.

On July 22, 1993, the NRC issued its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 FR 39132), expressing the view that satisfying the guidance in the policy statement also satisfies section 182a. of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.36. In the final policy statement, the NRC described the safety benefits of the STS and encouraged licensees, to the extent applicable, to use the STS for plant-specific TS amendments and for complete conversions to improved TS. The NRC published major revisions to the STS in 1995 (Revision 1), 2001 (Revision 2), 2004 (Revision 3), 2012 (Revision 4), and 2021 (Revision 5). The W-AP1000-STS, published in April 2016 (Revision 0), incorporated (1) selected applicable TSTF travelers approved since the issuance of W-STS, Revision 2, and (2) improvements to the COL plant-specific TS approved for Vogtle Electric Generating Station, Units 3 and 4, into Revision 19 of the GTS and Bases, which are included in the AP1000 design certification rule, Appendix D, "Design Certification Rule for the AP1000 Design," to 10 CFR Part 52. Revision 1 of W-AP1000-STS was published in January 2024 and reflects improvements to the plant-specific TS for the AP1000 lead plant, Vogtle Electric Generating Plant, Unit 3, that were incorporated by COL amendments approved during unit construction.

The format and content of proposed GTS and Bases prepared for an SDAA may use applicable provisions of the STS and STS Bases to the extent practicable to realize the safety benefits of standardization, taking into account design-specific characteristics. Before design approval, the staff reviews, as applicable, the SDAA in detail to verify that the applicant includes sufficient technical justification for any appropriate deviation from conventions and precedents presented in STS, as well as any deviation in content based on design-specific characteristics.

Generic changes to STS, known as TSTF travelers, which the NRC has approved since the issuance of STS, Revision 5, are considered needed improvements or corrections to STS. Should an SDA applicant provide technical specifications, the staff would recommend considering such travelers, where applicable, for inclusion, with suitable design-related modifications, in the proposed GTS and Bases, to further realize the safety benefits of standardization. Section 16.4.11 of this SER discusses the disposition of TSTF travelers for the SDAA.

16.4 Technical Evaluation

The staff evaluated the GTS according to the guidance in DSRS Chapter 16 to confirm that they will preserve the validity of the plant design, as described in the FSAR, by ensuring that the plant will be operated (1) within the required conditions bounded by the FSAR and (2) with operable equipment that is essential to prevent NuScale postulated design-basis events (DBEs) or mitigate their consequences.

The staff also reviewed the GTS Bases to verify that their technical content, level of detail, and format are consistent with the STS Bases and that they accurately provide the technical basis for each provision in GTS Chapter 2 and Chapter 3, consistent with the FSAR. Note that references to a Bases subsection for a Specification subsection are prefixed with an uppercase letter "B" (e.g., the Bases for GTS Subsection 3.3.1 are labeled Subsection B 3.3.1). Each Chapter 3, Sections 3.1 through 3.8 Bases subsection includes several sections, which are labeled as follows: Background, Applicable Safety Analyses, Limiting Condition for Operation (LCO), Applicability, Actions, Surveillance Requirements (SRs), and References.

The staff's review of the GTS and Bases included the following 12 topics of evaluation:

(1) Application of LCO Selection Criteria

(2) Use and Application (GTS Chapter 1), Definitions (Section 1.1), Logical Connectors (Section 1.2), Completion Times (Section 1.3), and Frequency (Section 1.4)

(3) SLs (GTS Chapter 2)

(4) LCO and SR Use and Applicability (GTS Chapter 3, Section 3.0)

(5) LCO Statements (GTS Chapter 3, Sections 3.1 to 3.8)

(6) Applicability Statements (GTS Chapter 3, Sections 3.1 to 3.8)

(7) Action Requirements (GTS Chapter 3, Sections 3.1 to 3.8)

(8) SRs (GTS Chapter 3, Sections 3.1 to 3.8)[2]

(9) Design Features (GTS Chapter 4, Sections 4.1 to 4.3)

(10) Administrative Controls (GTS Chapter 5, Sections 5.1 to 5.7)

(11) TSTF Traveler Disposition

(12) SDAA Technical Issues Affecting GTS and Bases

16.4.1 Application of Limiting Condition for Operation Selection Criteria

The applicant evaluated the NuScale US460 design and safety analyses against the LCO selection criteria in 10 CFR 50.36(c)(2)(ii) and determined the LCOs that must be established for the NuScale design. Each subsection of the GTS Bases discusses the applicable section of the safety analyses and states the LCO selection criterion that each of these LCOs satisfies. The applicant also summarized a comparison of the selected LCOs to those LCOs included in the W-STS, CE-STS, and W-AP1000-STS in Table B-1, "Comparison of Standard Technical Specifications with NuScale Generic Technical Specifications," of the NuScale US600 DCA Regulatory Conformance and Development Report.

For each LCO listed under Criterion 2 or 3, the Bases document describes the principal DBA or transient analysis that credits the specified SSC or parameter limit. The staff compared the Bases for consistency with accident and transient analysis descriptions in FSAR Chapter 15, "Transient and Accident Analyses." As stated in FSAR Table 15.0-1, "Design Basis Events," the

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[2] For instrumentation Channel Calibration SRs, the following parameter values are required to be calculated in accordance with the NRC-approved instrumentation setpoint methodology referenced in GTS Subsection 5.5.10, "Setpoint Program": (1) the limiting trip setpoints, which are the LSSS and derived from the safety analysis analytical limits, (2) the nominal trip setpoints, which are derived from the limiting trip setpoints; and (3) the acceptance criteria for Channel Calibration SRs. The setpoint program requires maintaining the current values of these parameters in a document controlled under 10 CFR 50.59, "Changes, Tests and Experiments," and in accordance with the approved setpoint methodology.

safety analysis considers DBEs in the following categories classified according to their expected frequency of occurrence: postulated accident, infrequent event, and AOO. Two beyond-design-basis special events are also considered but are not addressed by TS requirements, which ensure the validity of analyses of postulated accidents, infrequent events, and AOOs. These special events are the anticipated transient without scram (ATWS) in Section 15.8 and the core damage event with an associated core damage source term in Section 15.10. Unless otherwise pointed out, DBEs cited in this SER subsection are designated as AOO, which is the most common category.

16.4.1.1 Limiting Conditions for Operation Required by Criterion 1—Installed Instrumentation that Is Used to Detect, and Indicate in the Control Room, a Significant Abnormal Degradation of the Reactor Coolant Pressure Boundary

- LCO 3.4.7 Reactor Coolant System (RCS) Leakage Detection Instrumentation

This LCO requires two of the three RCS leakage detection methods to be operable and, like STS, it is the only LCO satisfying Criterion 1. For the leakage detection instrumentation to be operable, the non-safety-related containment evacuation system (CES) must be in operation and must maintain a low pressure in the containment vessel (CNV). Each of the three detection methods has sufficient sensitivity and response time to provide control room operators with an early warning of the detection of significant degradation of the reactor coolant pressure boundary (RCPB), which results in reactor coolant leakage into the containment. By alerting operators to take effective remedial measures as soon after occurrence as practicable, this instrumentation minimizes the potential for propagation to gross failure of the RCPB.

Since this system is designed and credited for detection of RCPB leakage within the CNV, and the proposed LCO is consistent with the STS, the staff finds that the GTS satisfy Criterion 1.

16.4.1.2 Limiting Conditions for Operation Required by Criterion 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

For each LCO for a process variable, design feature, or operating restriction listed below, the staff compared, for consistency, the Bases discussion of the applicable safety analyses and the FSAR description of each postulated accident, infrequent event, and AOO for which the process variable, design feature, or operating restriction is an initial condition. The staff finds that the following LCOs satisfy Criterion 2.

- LCO 3.1.1 Shutdown Margin (SDM)

The minimum required SDM is assumed as an initial condition process variable for all safety analyses, including analyses of inadvertent boron dilution (FSAR Section 15.4.6.3.2, "Input Parameters and Initial Conditions"), uncontrolled control rod assembly (CRA) withdrawal from a subcritical or low-power condition (FSAR Section 15.4.1.3.2, "Input Parameters and Initial Conditions"), and CRA ejection (FSAR Section 15.4.8.3.2, "Input Parameters and Initial Conditions"). The LCO on SDM ensures that specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with the assumption of the highest worth CRA stuck out of the core on a reactor trip.

- LCO 3.1.2 Core Reactivity

This LCO ensures core reactivity behavior following initial fuel loading or refueling is consistent with the design core reactivity behavior for the duration of the fuel cycle. Every 31 effective full power days (EFPD) of fuel burnup, SR 3.1.2.1 verifies the overall core reactivity balance is within one percent of the predicted values, by comparing the RCS boron concentration and the predicted reactor coolant letdown curve boron concentration. If this one percent criterion is not met, actions are required to restore core reactivity balance to within limits and determine that the reactor core is acceptable for continued operation. This includes a re-evaluation of the core design and safety analyses and may involve implementing appropriate operating restrictions and normalizing the predicted change in RCS boron concentration to the beginning of the fuel cycle measured steady-state RCS critical boron concentration, so that core reactivity relative to predicted values can be continually monitored and evaluated as core conditions change during the fuel cycle. This LCO protects the validity of the accident and transient analysis initial condition assumption that the core is operating within acceptable design limits by comparing predicted and measured steady-state RCS critical boron concentrations to ensure the measured reactivity[3] is maintained within one percent above or below the predicted reactivity (i.e., to ensure that a reactivity balance within ±1% $\Delta k/k$ of the normalized predicted values is maintained). This LCO must be met in Mode 1.

- LCO 3.1.3 Moderator Temperature Coefficient (MTC)

This LCO ensures that the MTC is maintained within the upper and lower limits specified in the core operating limits report (COLR). In Mode 1, the upper limit (least negative value) on the MTC must be maintained to ensure that any core overheating accidents will not violate the design assumptions of the accident analysis. The limits must also be maintained to ensure startup and subcritical accidents, such as the uncontrolled CRA withdrawal from zero power, will not violate the assumptions of the accident analysis. The lower moderator temperature coefficient limit (most negative value) must be maintained in Modes 1 and 2, and in Mode 3 with any RCS temperature greater than or equal to 93.3 degrees Celsius (°C) (200 degrees Fahrenheit (°F)), to ensure that core overcooling accidents at the end of cycle will not violate the assumptions of the accident analysis.

- LCO 3.1.4 Rod Group Alignment Limits

In Mode 1, complying with the requirements that all shutdown bank and regulating bank CRAs be operable and that individual CRA positions be within six steps (2.25 in) of their shutdown or regulating group position, ensures that the CRA groups maintain the correct core power distribution and satisfy the SDM requirements of LCO 3.1.1. These CRA alignment limits protect the validity of the initial conditions assumed in the analysis of CRA misalignment accidents.

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[3] The effective multiplication factor, k_{eff}, equals the neutron production from fission in one neutron population generation divided by the sum of the preceding neutron population generation's absorption by the fuel and leakage from the core. If $k_{eff} = 1.0$, the core is said to be critical and the neutron population from generation to generation stays the same; if $k_{eff} > 1.0$, the core is said to be supercritical and the neutron population increases; and if $k_{eff} < 1.0$, the core is said to be subcritical and the neutron population decreases. For power reactors, core reactivity is the deviation of k_{eff} from one and equals $(k_{eff} - 1)/k_{eff}$; reactivity is expressed as a dimensionless number with units of percent of k_{eff} (% $\Delta k/k$).

- LCO 3.1.5 Shutdown Bank Insertion Limits

In Mode 1, complying with shutdown bank CRA group insertion limits protects initial assumptions in all safety analyses that assume shutdown bank CRA insertion upon reactor trip. These insertion limits protect assumptions of initial core power distribution, available SDM, ejected shutdown CRA worth, and initial reactivity insertion rate.

- LCO 3.1.6 Regulating Bank Insertion Limits

In Mode 1 with $k_{eff} \geq 1.0$, complying with regulating bank CRA group insertion limits protects initial assumptions in the safety analyses for loss-of-coolant accidents (LOCAs), loss of flow, ejected CRA, or other accidents that assume regulating bank CRA insertion upon reactor trip. These insertion limits protect assumptions of initial core power and fuel burnup distributions, available SDM, ejected regulating CRA worth, and initial reactivity insertion rate.

- LCO 3.1.7 Rod Position Indication

In Mode 1, the CRA position indication systems are required to be operable to determine CRA positions and thereby ensure compliance with the CRA alignment and power-dependent insertion limits. CRA positions must be known with sufficient accuracy to verify the core is operating within the required withdrawal and insertion sequences for the shutdown bank and regulating bank CRA groups, CRA group position overlap limits, design core power distribution peaking limits, ejected CRA reactivity addition worth limit, and minimum SDM (the reactivity addition needed to reach criticality after a reactor trip).

- LCO 3.1.9 Boron Dilution Control (chemical and volume control system (CVCS) demineralized water isolation valve operability, boric acid supply boron concentration limits, RCS CVCS makeup flowrate limits, and isolation of module heatup system (MHS) flow paths to and from other unit CVCS during MHS operation)

In Modes 1, 2, and 3, this LCO ensures that the boron addition system is not a source of reactor coolant boron dilution and that the flowrate for the makeup pump demineralized water flow path does not exceed the COLR-specified flowrate assumed in the analysis of the inadvertent decrease in the RCS boron concentration AOO (FSAR Section 15.4.6, “Inadvertent Decrease in Boron Concentration in the Reactor Coolant System”). The Applicable Safety Analyses section of Bases Subsection B 3.1.9 states, “The boron concentration in the boric acid supply, the CVCS makeup pump demineralized water flow path flowrate, and isolation of MHS flow paths between units satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).”

- LCO 3.2.1 Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)

FSAR Section 4.3.2.2.1, “Power Distribution—Definitions,” states the following:

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

FSAR Section 4.3.2.2.2, “Power Distribution—Radial Power,” indicates that $F_{\Delta H}$ is an indication of radial flux peaking.

FSAR Section 4.3.2.2.6, “Power Distribution—Limiting Power Distributions,” states, in part, the following:

> The radial power distribution is primarily determined by the cycle design. Each cycle core design is required to adhere to the maximum allowed $F_{\Delta H}$. This design limit is then applied in the subchannel analysis as described in Section 4.4. The transient analysis applies constant radial power distributions through the evolution of transient events. For those events where radial peaking can increase during the event, an augmentation factor is used to modify the radial power distribution.

The limiting value for $F_{\Delta H}$ is specified in the COLR and is an initial condition of all DBE analyses for which limits on the initial core power distribution are assumed. This LCO is applicable in Mode 1 with thermal power ≥ 20% Rated Thermal Power (RTP).

- LCO 3.2.2 AXIAL OFFSET (AO)

Axial Offset (AO) is the ratio of the difference in power between the top half of the core and the bottom half of the core to the total core power. This parameter is an indication of axial flux peaking, and the limiting values specified in the COLR are initial conditions of all DBE analyses for which limits on the initial core power distribution are assumed. This LCO is applicable in Mode 1 with thermal power ≥ 20% RTP.

FSAR Section 4.3.2.2.6, “Power Distribution—Limiting Power Distributions,” states, in part, the following:

> The core average axial flux shape can be affected by operator action. The core average axial power profile is affected by power level, control rod motion, load changes, xenon distribution, temperature changes, and cycle burnup. Axial power peaking (F_Z) does not have a limit associated with it, but the axial power distribution is limited by the allowable [Axial Offset (AO)] window in the technical specifications.
>
> An analysis of the possible axial power shapes is performed to identify the bounding axial power shapes for use in the CHF and transient analyses. These shapes are generated as a function of power level, cycle burnup, control rod position, xenon distribution, and core thermal-hydraulic conditions. Bounding axial power shapes are held constant through the evolution of transients.

FSAR Section 4.3.2.2.9, “Power Distribution—Monitoring,” states, in part, the following *(emphasis added*):

> The ICIS [incore instrumentation system] is used to synthesize core-wide three-dimensional power distributions. These power distributions are compared to predicted core power distributions to verify the core is operating as designed. *Axial power distributions are continuously monitored to validate the AO operating window, and actions required by the technical specifications are initiated based on this information.* Power distributions from the ICIS are used to calibrate the ex-core neutron flux detectors. When the rod position indication system is not

working properly, the ICIS has the capability to determine the relative position of a stuck or misaligned control rod.

FSAR Section 4.3.2.2.1 states, in part, the following:

> The heat flux hot channel factor (or total peaking factor), F_Q, is the ratio of maximum local heat flux on the surface of a fuel rod to the average fuel rod heat flux. F_Q is the product of axial peaking factor (F_Z) and the enthalpy rise hot channel factor ($F_{\Delta H}$), The maximum F_Q is used to calculate the peak linear hear generation rate ([PL]HGR).

FSAR Section 4.4.2.2 states, in part, the following:

> In safety analyses, PLHGR is determined using enthalpy rise hot channel factor ($F_{\Delta H}$) and maximum axial peaking for value (F_Z) allowable within AO [Axial Offset] window.

As such, F_Q is not used as an initial condition for any transient or design basis accident, including loss-of-coolant accident.

The staff observed that, even though the limit on maximum F_Q is used to ensure that none of the fuel design criteria are exceeded, NuScale US600 DC GTS Section 3.2 and proposed NuScale US460 SDA GTS Section 3.2 do not include an LCO for F_Q. In previous reactor design reviews, the staff has relied upon such an LCO to establish a finding that the reactor unit will be operated within the bounds of the safety analyses. By letter dated June 12, 2018 (ML18163A417), during the US600 DCA review, the applicant provided the following explanation, which the staff found acceptable during its review of the US600 DCA:

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

Although the US460 design linear heat rate is greater than that of the US600 design, the above explanation is valid for the US460 design. The staff concludes that for the US460 design an LCO for F_Q is not needed to ensure that the core SL for peak fuel centerline temperature is not violated in the event of a postulated accident. SER Section 4.3 discusses the basis of this conclusion.

- LCO 3.4.1 RCS Pressure, Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits

The limits of this LCO protect the initial condition assumptions of the FSAR Chapter 15 safety analyses. This LCO is applicable in Mode 1.

- LCO 3.4.2 RCS Minimum Temperature for Criticality

This LCO ensures the minimum temperature for criticality (MTC) is within the accident analysis bounds, the RCS instrumentation is operating within its nominal operating envelope, the pressurizer (PZR) operating characteristics are within the envelope assumed in the transient and accident analysis, and the reactor vessel is above nil-ductility limits when the reactor is operating. This LCO is applicable in Mode 1.

- LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits

This LCO protects the RCPB and must be met at all times.

- LCO 3.4.5 RCS Operational LEAKAGE

The leakage limits ensure that RCPB degradation will be detected and corrected before the flaw results in a LOCA. This LCO must be met in Modes 1 and 2, and in Mode 3 with RCS hot temperature at or above 93.3°C (200°F). As described in the Bases for Specification 3.4.5, the leakage detection system design lacks the capability to determine the source of leakage into the CNV; this results in relying on the unidentified LEAKAGE definition and limit to restrict unit operation when RCS leakage (other than RCS pressure boundary LEAKAGE) into the CNV exists. The definition of identified LEAKAGE and its limit are included to be consistent with W-STS and to account for a future design change to the leakage detection system that enables determining the source of leakage into the CNV.

- LCO 3.4.8 RCS Specific Activity

The dose equivalent iodine (I)-131 and the dose equivalent xenon (Xe)-133 activity limits are consistent with the design-basis failed-fuel fraction assumed in the design of radiation shielding in spaces with piping and vessels containing reactor coolant that are accessible to plant operators. These activity limits are also consistent with the assumed initial RCS specific activity in the accident radiological dose consequence analyses in FSAR Chapter 15. These activity limits reflect a specific activity resulting from expected fuel pin cladding defects that are much less severe than typically considered in the W-STS. With these much smaller specific activity limits, the contribution of the assumed initial RCS specific activity to dose consequences of DBAs, such as a steam generator (SG) tube rupture (or failure), is also much smaller. This LCO must be met in Modes 1 and 2.

- LCO 3.4.9 Steam Generator (SG) Tube Integrity

This LCO, in conjunction with the SG program of GTS Subsection 5.5.4, ensures the SG tubes are maintained such that a postulated accident involving an SG tube failure (SGTF) (double-ended failure of a single tube) is unlikely to occur. The safety analyses of postulated accidents and AOOs other than an SGTF assume the maintenance of tube structural integrity. SER Section 16.4.10.3 gives an additional evaluation of the SG tube requirements of GTS Subsections 3.4.9 and 5.5.4. This LCO must be met in Modes 1 and 2, and in Mode 3 when not passively cooled.

- LCO 3.5.3 Ultimate Heat Sink (UHS)

The LCO for the UHS, which is the water in the reactor pool, must be met at all times.

In Modes 1, 2, and 3, this LCO protects the UHS bulk average temperature upper limit of 48.9°C (120°F), which is assumed and credited, directly or indirectly, as an initial condition in postulated accidents that require the operation of the decay heat removal system (DHRS) and the emergency core cooling system (ECCS), for both LOCA and non-LOCA DBEs. This LCO also protects the UHS bulk average temperature lower limit of 18.3°C (65°F) assumed in the long-term cooling analyses.

This LCO protects the minimum reactor pool level limit of 15.85 meters (m) (52 feet (ft)) by providing margin above the minimum level required to support DHRS and ECCS operation in response to LOCA and non-LOCA DBEs. The 15.85 m (52 ft) minimum pool level also ensures the CNV wall temperature initial condition assumed in the peak containment pressure analysis. The upper limit of 16.46 m (54 ft) for the maximum pool level is an initial condition that ensures long-term cooling analyses assumptions are protected.

In Mode 5, or during irradiated fuel movement within the spent fuel pool, this LCO ensures the initial condition assumptions of the analysis of a postulated fuel handling accident during irradiated fuel movement are satisfied (FSAR Section 15.0.3.7.5, “Fuel Handling Accident”). The minimum reactor pool water level of 15.85 m (52 ft) ensures the assumption of 7.0 m (23 ft) of water above the top of a fuel assembly sitting on the weir wall, which is the most limiting, most shallow location of a dropped fuel assembly.

The UHS bulk average boron concentration lower limit is established to ensure adequate shutdown margin during unit shutdowns that are not associated with events resulting in DHRS or ECCS actuation, when the module containment is filled with reactor pool inventory using the containment flood and drain system (CFDS) and the RRVs are opened. This lower limit also ensures adequate shutdown margin when the module is configured with the UHS inventory in contact with the reactor core, specifically in Mode 4 when the upper containment vessel is disassembled for removal.

FSAR Section 15.4.6.3.2, for the analysis of an inadvertent decrease in reactor pool boron concentration AOO (FSAR Section 15.4.6) states, “A conservatively smaller pool volume is used to provide a limiting boron dilution for Mode 5.”

- LCO 3.8.2 Decay Time

This LCO is applicable during movement of irradiated fuel in the reactor vessel. In Mode 5, this LCO ensures irradiated fuel movement within the reactor vessel does not occur until 48 hours after reactor shutdown, as assumed by the postulated design basis fuel handling accident analysis.

Based on the above evaluation, the staff finds that the proposed GTS satisfy 10 CFR 50.36(c)(2)(ii)(B), Criterion 2.

16.4.1.3 Limiting Conditions for Operation Required by 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

For each LCO subsystem and instrumentation function listed below, the staff compared, for consistency, the Bases discussion of the applicable safety analyses and the FSAR description of each postulated accident, infrequent event, and AOO that credits the subsystem or function. The staff finds that the following LCOs satisfy Criterion 3.

- LCO 3.1.9 Boron Dilution Control (CVCS DWSI valve operability, boric acid storage tank boron concentration limits, CVCS makeup pump flow path flowrate limit, and module heatup system (MHS) flow path)

This LCO requires two CVCS demineralized water supply isolation (DWSI) valves to be operable to ensure that there will be redundant means available to automatically terminate an inadvertent boron dilution event in Modes 1, 2, and 3. This LCO also ensures that the boron addition system is not a source of reactor coolant boron dilution and that makeup pump flow does not exceed the COLR-specified flowrate assumed in the analysis of an inadvertent decrease in RCS boron concentration (boron dilution event), which is an AOO (FSAR Section 15.4.6). In discussing the applicable safety analyses, Bases Subsection B 3.1.9 states, “CVCS demineralized water isolation valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).” The staff concludes that the CVCS DWSI valves satisfy Criterion 3 because the analysis of an inadvertent boron dilution event credits them with automatically terminating the event.

- LCO 3.3.1 Module Protection System (MPS) Instrumentation

SER Table 16.4.1-1 lists each MPS instrumentation Function with its Modes of Applicability for each supported reactor trip system (RTS) or engineered safety features (ESF) actuation system (ESFAS) logic and actuation Function. For each supported Function, one or more of the events crediting the MPS Function are also listed, including a reference to the section of the FSAR describing each event. SER Table 16.4.1-1 uses italics to denote requirements included in GTS Subsection 3.3.1 for MPS instrumentation Functions. Regular font denotes requirements included in GTS Subsection 3.3.2 for RTS Logic and Actuation Functions; requirements included in GTS Subsection 3.3.3 for ESF Logic and Actuation Functions; and references to safety analyses described in FSAR Chapter 15.

By letter dated April 15, 2019 (ML19105B292), during the NuScale US600 DCA review, the applicant revised the ESFAS design by changing which MPS instrumentation Functions initiate partial trip signals to the ESF actuation coincidence logic divisions for the DHRS and ECCS ESFAS Functions. The previous MPS instrumentation Functions for DHRS actuation, which resulted in secondary system isolation (SSI) but also an unnecessary automatic RCS cooldown, were redesignated to initiate a new ESFAS Function, SSI, with one exception. The DHRS actuation on low pressurizer pressure with RCS hot temperature greater than the T-4 interlock was removed because the instrumentation Function for reactor trip on low pressurizer pressure above T-4 provides adequate RCS subcooled margin protection. The letter also removed the MPS instrumentation Function for ECCS actuation on low reactor pressure vessel (RPV) riser level because no analyzed event for the US600 certified design relied on this ECCS initiation. However, the US460 SDAA MPS instrument Functions and the ESF systems they initiate differ from those in the US600 certified design as shown in the following list.

MPS Instrument Functions for ECCS and DHRS Actuation

DCA Rev. 5	SDAA Rev. 2
<u>ECCS actuation</u>	<u>ECCS actuation</u>
22.a High Containment Water Level	—
23.a Low RCS Pressure	—
—	23.a Low Reactor Pressure Vessel Riser Level

—	24.a Low Low Reactor Pressure Vessel Riser Level
—	25.h Low AC Voltage to EDAS [augmented DC power system] Battery Chargers
<u>DHRS actuation</u>	<u>DHRS actuation</u>
7.b High Pressurizer Pressure	7.b High Pressurizer Pressure
—	11.c Low Pressurizer Level
13.b High Narrow Range RCS Hot Temperature	13.b High Narrow Range RCS Hot Temperature
16.b High Main Steam Pressure	17.b High Main Steam Pressure
—	22.c High Narrow Range Containment Pressure
25.b Low AC Voltage to Low Voltage AC Electrical Distribution System [ELVS] Battery Chargers	25.c Low AC Voltage to EDAS Battery Chargers
—	26.c High Under-the-Bioshield Temperature

The May 20, 2020, update to US600 DCA Revision 4 (ML20141L787) added the low RCS pressure - ECCS ESFAS instrumentation Function for ECCS actuation (Function No. 23.a of DCA GTS Table 3.3.1-1) to ensure that ECCS actuation occurs before significant accumulation of water with reduced boron concentration can occur in the containment. This ensures an unanalyzed reactivity transient will not occur during small loss-of-coolant events in the containment following initiation of coolant flow from containment through the reactor recirculation valves (RRVs) into the RPV downcomer and through the core. The US460 SDAA addressed this concern by the addition of the ECCS supplemental boron (ESB) system in the CNV and the associated GTS LCO 3.5.4. Accordingly, US460 SDAA GTS LCO 3.3.1 omits the US600 instrument Function for ECCS actuation on low RCS pressure. The US460 SDAA also added MPS instrument Functions for ECCS actuation on Low and Low Low RPV riser level. Also added is a timer function that initiates ECCS actuation approximately 8 hours following generation of an automatic or manual reactor trip signal, in case ESB is needed to prevent recriticality that may otherwise result from the low boron content of reactor coolant in containment—which is vented as steam from the RPV through the reactor vent valves (RVVs), or possibly through RSVs, condensed in containment, and returned to the RPV through the RRVs—that dilutes the boron concentration of the reactor coolant flowing through the reactor core combined with reactivity addition from RCS cool down and xenon decay. This post reactor trip ECCS timer function is implemented in the scheduling and voting module (SVM) of the MPS as a part of the ECCS logic and actuation function in LCO 3.3.3 (Function 1).

In US460 SDAA Part 7, Section 5, the applicant provided rationale to support an exemption request from GDC 33, “Reactor coolant makeup,” which requires a system to supply reactor coolant makeup for protection against small breaks in the RCPB. The NuScale US460 design’s CVCS is a reactor coolant makeup system; however, it is not a safety-related system. To justify this exemption, the applicant provided rationale that would demonstrate that the underlying purpose of GDC 33 could be met using only safety-related systems. FSAR Section 6.3.1, “Design Basis,” describes how the ECCS actuation instrumentation setpoints are chosen to support this justification. SER Sections 6.3 and 9.3 provide the staff’s evaluation of the ECCS design and how the rationale provided satisfies the underlying purpose of GDC 33. In

Section 6.3.1, the staff determined that the ECCS actuation on Low and Low Low RPV riser water level and Low alternating current (AC) voltage to augmented direct current (DC) power system (EDAS) battery chargers with the 24-hour time delay were used, in part, to support justification that the ECCS design would be sufficient to meet the underlying purpose of GDC 33. The staff considered NuScale's exemption requests and determined that those exemptions, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

The ECCS actuation on low AC voltage to the EDAS battery chargers is included as a TS LCO-required MPS instrumentation Function in GTS Table 3.3.1-1, even though no FSAR Chapter 15 accident or transient analysis that assumes ECCS automatic actuation depends upon or credits the low AC voltage to EDAS battery chargers' instrumentation ECCS actuation signal as part of the event response success path. This Function is modeled in some events because it would be part of the transient progression and was included because ECCS actuation, depending on the timing, could increase the severity of predicted unit conditions. Therefore, the staff concludes that the ECCS actuation on low AC voltage to EDAS battery chargers MPS instrumentation Function signal does not satisfy any of the LCO selection criteria of 10 CFR 50.36(c)(2)(ii).

Table 16.4.1-1 MPS (RTS and ESFAS) Instrumentation Functions

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	*APPLICABLE MODES*		
1. High Power Range Linear Power			
• *Designed to protect against exceeding CHF limits for reactivity and overcooling events (Table 15.0-7)*			
a. RTS • Loss of feedwater heating (15.1.1, 10.4.6.2) • Decrease in feedwater temperature AOO (15.1.1.2) (5% RTP penalty added) (Table 15.1-1) • Increase in feedwater flow AOO (15.1.2.2) (5% RTP penalty added) • Increase in steam flow AOO (nonlimiting CHFR 125% steam flow case) (15.1.3.2) (5% RTP penalty added) • Steam piping failures inside and outside CNV (15.1.5.2) (5% RTP penalty added) • Uncontrolled CRA withdrawal from a subcritical or low-power or startup condition (at 25% RTP for startup conditions) (15.4.1.2) • Uncontrolled CRA withdrawal at power (15.4.2.2) • Control rod misoperation (system malfunction or operator error) (15.4.3.2) • Spectrum of rod ejection accidents (15.4.8.2.4)	*1*	*2(a)*	*3(a)*
b. DWSI • Inadvertent decrease in RCS boron concentration (15.4.6.2) • DWSI designed to occur when Function 1.a, RTS, occurs	*1*	*2(a)*	*3(b)*
(a) When capable of withdrawal of more than one CRA.			
(b) When capable of withdrawal of more than one CRA and the RCS temperature is			

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	*APPLICABLE MODES*		
above the T-3 interlock.			
2. High Power Range Positive and Negative Rate • *Designed to protect against exceeding CHF limits for reactivity and overcooling events (Table 15.0-7)*			
a. RTS • Decrease in feedwater temperature AOO *(not credited)* (15.1.1.2) • Increase in feedwater flow AOO *(not credited)* (15.1.2.2) • Increase in steam flow AOO *(not credited)* (15.1.3.2) • Steam piping failures inside & outside CNV (*not credited*) (15.1.5.2) • Uncontrolled CRA withdrawal at power (15.4.2.2) • Spectrum of rod ejection accidents (15.4.8.2.4)	*1(c)*	-	-
b. DWSI • DWSI designed to occur when Function 2.a, RTS, occurs	*1(b)*	-	-
(b) When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock *(c) With power above the N-2H Interlock.*			
3. High Intermediate Range Log Power Rate • *Designed to protect against exceeding CHF and energy deposition limits during startup power excursions (Table 15.0-7)*			
a. RTS • Uncontrolled CRA withdrawal from a subcritical or low power or startup condition (15.4.1.2)	*1(d)*	*2(a)*	*3(a)*
b. DWSI • DWSI designed to occur when Function 3.a, RTS, occurs	*1(d)*	*2(a)*	*3(b)*
(a) When capable of withdrawal of more than one CRA. *(b) When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock.* *(d) With power below the N-2L interlock.*			
4. High Source Range Count Rate • *Designed to protect against exceeding CHF and energy deposition limits during rapid startup power excursions; this instrumentation Function is bypassed once the intermediate range log power signal is established (Table 15.0-7)*			
a. RTS • Uncontrolled CRA withdrawal from a subcritical or low-power or startup condition (15.4.1.2)	*1(e)*	*2(a)*	*3(a)*
b. DWSI • DWSI designed to occur when Function 4.a, RTS, occurs	*1(e)*	*2(a)*	*3(b)*
(a) When capable of withdrawal of more than one CRA. *(b) When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock* *(e) When Intermediate Range Log Power less than N-1 interlock.*			

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	*APPLICABLE MODES*		
5. *High Source Range Log Power Rate* • *Designed to protect against exceeding CHF and energy deposition limits during startup power excursions (Table 15.0-7)*			
a. RTS • Uncontrolled CRA withdrawal from a subcritical or low-power or startup condition (15.4.1.2)	*1(e)*	*2(a)*	*3(a)*
b. DWSI • DWSI designed to occur when Function 5.a, RTS, occurs	*1(e)*	*2(a)*	*3(b)*
(a) When capable of withdrawal of more than one CRA. *(b) When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock* *(e) When Intermediate Range Log Power less than N-1 interlock.*			
6. *High Subcritical Multiplication* • *Designed to detect and mitigate inadvertent subcritical boron dilutions in operating Modes 2 and 3 (Table 15.0-7)*			
a. DWSI • Inadvertent decrease in RCS boron concentration (15.4.6.2)	*1(e)*	*2*	*3*
(e) When Intermediate Range Log Power less than N-1 interlock.			
7. *High Pressurizer Pressure* • *Designed to protect against exceeding RPV pressure limits for reactivity and heatup events (Table 15.0-7)*			
a. RTS • Rod ejection accident—maximum RCS pressure case, (15.4.8, Table 15.4-21) • Increase in steam flow AOO (15.1.3.2) • Loss of external load AOO (15.2.1.2) • Turbine trip AOO (15.2.2.1) • Loss of condenser vacuum AOO (15.2.3.1) • Closure of main steam isolation valves (MSIVs) AOO (15.2.4.2) • Loss of nonemergency AC power to the station auxiliaries (15.2.6.2, Table 15.2-15, RCS Overpressurization, Table 15.2-16, SG Peak Pressure) • Loss of normal feedwater flow AOO (15.2.7.2, 10.4.7.2) • Feedwater system pipe breaks inside and outside of containment (15.2.8.1) • Inadvertent DHRS actuation (15.2.9, 10.4.7.2) • Uncontrolled CRA withdrawal at power (15.4.2.2) • Control rod misoperation (15.4.3.2) • Spectrum of rod ejection accidents (15.4.8.2.4) • CVCS malfunction AOO (15.5.1.2) (no pressurizer spray flow) • Failure of small lines carrying primary coolant outside containment (CVCS makeup line break + loss of normal AC) infrequent event (15.6.2.3.3, CVCS letdown line break + loss of normal AC, Table 15.6-3 maximum RPV pressure) • LOCAs resulting from a spectrum of postulated piping breaks within the RCPB postulated accident (15.6.5.3.3)	*1*	*2(a)*	*3(a)*

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	APPLICABLE MODES		
• SGTF postulated accident—limiting RPV pressure (15.6.3.3.3, Table 15.6-4)			
b. DHRS [4] • Increase in steam flow AOO (15.1.3.2) • Loss of feedwater flow (15.2.7, 10.4.7.2) • Loss of external load AOO (15.2.1.2) • Turbine trip AOO (15.2.2.1) • Loss of condenser vacuum AOO (15.2.3.1) • Closure of MSIV(s) AOO (15.2.4.2) • Loss of nonemergency AC power to the station auxiliaries (15.2.6.2, Table 15.2-15, RCS Overpressurization, Table 15.2-16, SG Peak Pressure) • Loss of normal feedwater flow AOO (15.2.7.2) • Uncontrolled CRA withdrawal from a subcritical or low-power or startup condition (15.4.1.2) • Feedwater system pipe breaks inside and outside of containment (15.2.8.1) • Uncontrolled CRA withdrawal at Power (15.4.2.2) • Control rod misoperation (15.4.3.2) • Failure of small lines carrying primary coolant outside containment (CVCS makeup line break + loss of normal AC) infrequent event (15.6.2.3.3, CVCS letdown line break + loss of normal AC, Table 15.6-3 maximum RPV pressure) • CVCS malfunction AOO (15.5.1.2) (Table 15.5-1 limiting RCS pressure case, no pressurizer spray) • SGTF postulated accident—limiting RPV pressure (15.6.3.3.3, Table 15.6-4) • LOCAs resulting from a spectrum of postulated piping breaks within the RCPB postulated accident (15.6.5.3.3)	*1*	*2*	*3(f)*
c. SSI • Loss of external load AOO (15.2.1.2) • CVCS malfunction AOO (15.5.1.2) (Table 15.5-1 limiting RCS pressure case, no pressurizer spray) • Steam Generator Tube Failure (Thermal Hydraulic) (15.6.3.3.3 Table 15.6-4 Limiting Reactor Pressure Vessel Pressure) • Events listed under DHRS actuation	*1*	*2*	*3(f)*
d. DWSI • DWSI designed to occur when Function 7.a, RTS, occurs.	*1*	*2(a)*	*3(b)*
e. Pressurizer Heater Trip (PHT) • SGTF postulated accident—limiting RPV pressure (15.6.3.3.3, Table 15.6-4), limiting SG pressure (Table 15.6-5) • Pressurizer heaters are tripped on all automatic DHRS actuation signals.	*1*	*2(g)*	*3(g)*

[4] DHRS actuation includes isolation of each feedwater line using the feedwater isolation valve (FWIV) and the feedwater regulating valve (FWRV), and each main steamline using the MSIVs and the main steam isolation bypass valves.

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	*APPLICABLE MODES*		
(a) When capable of withdrawal of more than one CRA. *(b) When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock* *(f) When not PASSIVELY COOLED.* *(g) With pressurizer heater breakers closed.*			
8. Low Pressurizer Pressure • *Designed to detect and mitigate high-energy line break (HELB) events from the pressurizer vapor space and protect RCS subcooled margin for protection against instability events (Table 15.0-7)*			
a. RTS • Decrease in feedwater temperature AOO (15.1.1.2) • Steamline break outside containment (15.1.5, 10.4.7.2) • Increase in steam flow AOO (15.1.3.2) • Steam piping failures inside and outside CNV (15.1.5.2) • Failure of small lines carrying primary coolant outside containment infrequent event (15.6.2.3.2/3) • SGTF postulated accident (15.6.3.3.2,10.4.7.2) • LOCAs resulting from a spectrum of postulated piping breaks within the RCPB postulated accident (15.6.5.3.3) • Subcooled margin protection to protect against flow instability events above T-4 interlock	*1(h)*	-	-
b. SSI • Steam Piping Failure - Limiting Reactor Coolant System Pressure Case (15.1.5, Table 15.1-11) • Steam Generator Tube Failure (Thermal Hydraulic) (15.6.3.3.3, Table 15.6-5 Limiting Steam Generator Pressure case)	*1(h)*	-	-
c. DWSI • DWSI designed to occur when Function 8.a, RTS, occurs • Subcooled margin protection above T-4 interlock	*1(h)*	-	-
d. Pressurizer Line Isolation • LOCAs resulting from a spectrum of postulated piping breaks within the RCPB postulated accident (15.6.5.3.3)	*1*	*2*	*3(i)*
(h) With RCS temperature above the T-4 interlock. *(i) With RCS temperature above the T-3 interlock.*			
9. Low Low Pressurizer Pressure • *Designed to protect RCS subcooled margin for protection against instability events (Table 15.0-7)*			
a. RTS • Decrease in RCS inventory that causes a slow RCS depressurization (15.9.3.6) at the beginning of cycle; trip occurs before loss of riser inlet subcooling leading to flow oscillations or instability • Subcooled margin protection below T-4 interlock	*1*	*2(a)*	*3(b)*
b. DWSI • DWSI is designed to occur when Function 9.a, RTS, occurs.	*1*	*2(a)*	*3(b)*

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	*APPLICABLE MODES*		
• Subcooled margin protection below T-4 interlock *(a) When capable of withdrawal of more than one CRA.* *(b) When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock*			
10. High Pressurizer Level • *Designed to detect and mitigate CVCS malfunctions to protect against overfilling the pressurizer (Table 15.0-7)*			
a. RTS • Inadvertent DHRS actuation AOO (15.2.9, 10.4.7.2) • Loss of feedwater flow AOO (15.2.7, 10.4.7.2) • CVCS malfunction AOO (15.5.1.2) (Table 15.5-1 limiting SG pressure case, pressurizer spray available)	*1*	*2(a)*	*3(a)*
b. DWSI • DWSI is designed to occur when Function 10.a, RTS, occurs.	*1*	*2(a)*	*3(b)*
c. CVCSI • CVCS malfunction AOO (15.5.1.2) (Table 15.5-1 limiting SG pressure case, pressurizer spray available) • Increased RCS inventory events and inadvertent operation of the module heatup system are terminated by isolation of RCS injection line (5.2.2.2.2) *(a) When capable of withdrawal of more than one CRA.* *(b) When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock*	*1*	*2*	*3*
11. Low Pressurizer Level • *Designed to detect and mitigate pipe breaks to protect RCS inventory and ECCS functionality during LOCAs, primary HELB outside containment events, or SGTF, and to protect the pressurizer heaters from uncovering and overheating during decrease in RCS inventory events (Table 15.0-7)*			
a. RTS • Decrease in feedwater temperature AOO (15.1.1.2) • Steam piping failures inside and outside containment (15.1.5.2, 10.4.7.2) • Failure of small lines carrying primary coolant outside containment infrequent event (15.6.2.3.2) • SGTF postulated accident (15.6.3,10.4.7.2), Table 15.6-6 limiting mass release • LOCAs resulting from a spectrum of postulated piping breaks within the RCPB postulated accident (15.6.5.3.3)	*1*	*2(a)*	*3(a)*
b. CIS • Decrease in feedwater temperature AOO (15.1.1.2, Table 15.1-1) • SGTF postulated accident (15.6.3,10.4.7.2) • LOCAs resulting from a spectrum of postulated piping breaks within the RCPB postulated accident (15.6.5.3.3)	*1(h)*	-	-
c. DHRS	*1(h)*	-	-

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	*APPLICABLE MODES*		
• Steam Generator Tube Failure (Thermal Hydraulic) (15.6.3.3.3, Table 15.6-5 Limiting Steam Generator Pressure			
d. SSI • Steam Generator Tube Failure (Thermal Hydraulic) (15.6.3.3.3, Table 15.6-5 Limiting Steam Generator Pressure case)	*1*	*2*	*3(j)*
e. DWSI • DWSI is designed to occur when Function 11.a, RTS, occurs.	*1*	*2(a)*	*3(b)*
f. CVCSI • Decrease in feedwater temperature AOO (15.1.1.2, Table 15.1-1) • Failure of small lines carrying primary coolant outside containment infrequent event (15.6.2.3.2)	*1(h)*	-	-
g. PHT • SGTF (15.6.3,10.4.7.2 Table 15.6-4 limiting RPV pressure case, Table 15.6-5 limiting SG pressure)	*1*	*2(g)*	*3(g)*
(a) When capable of withdrawal of more than one CRA. *(g) With pressurizer heater breakers closed.* *(j) With RCS temperature above the T-2 interlock and containment water level below the L-1 interlock.*			
12. Low Low Pressurizer Level • *Pipe breaks to protect RCS inventory and ECCS functionality during LOCAs, primary HELB outside containment events, or SGTF (Table 15.0-7)*			
a. Containment Isolation System (CIS) • Decrease in feedwater temperature AOO (15.1.1.2, Table 15.1-1) • SGTF postulated accident (15.6.3,10.4.7.2) • LOCAs resulting from a spectrum of postulated piping breaks within the RCPB postulated accident (15.6.5.3.3)	*1*	*2*	*3(j)*
b. CVCSI • Decrease in feedwater temperature AOO (15.1.1.2, Table 15.1-1) • Failure of small lines carrying primary coolant outside containment infrequent event (15.6.2.3.2)	*1*	*2*	*3(j)*
(j) With RCS temperature above the T-2 interlock and containment water level below the L-1 interlock.			
13. High Narrow Range RCS Hot Temperature • *Designed to protect against exceeding CHF limits for reactivity and heatup events (Table 15.0-7).*			
a. RTS • Decrease in feedwater temperature AOO (15.1.1.3, Table 15.1-1) • Increase in steam flow AOO—limiting CHFR 114% steam flow case (15.1.3.2) • Increase in steam flow AOO *(*15.1.3.2) • Loss of normal feedwater flow AOO (15.2.7.2) • Inadvertent operation of DHRS AOO (15.2.9.2, Table 15.2-31) • Uncontrolled CRA withdrawal at power (15.4.2.2) • Control rod misoperation (15.4.3.2)	*1*	-	-

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	*APPLICABLE MODES*		
• Spectrum of rod ejection accidents (15.4.8.2.4)			
b. DHRS • Decrease in feedwater temperature AOO (15.1.1.2) (Table 15.1-1 limiting minimum CHFR) • Increase in steam flow (15.1.3.2) • Increase in steam flow AOO *(*15.1.3.2) • Loss of normal feedwater flow AOO (15.2.7.2) • Inadvertent operation of DHRS AOO (15.2.9.2) • Uncontrolled CRA withdrawal at power (15.4.2.2) • Control rod misoperation (15.4.3.2)	*1*	*2*	*3(f)*
c. SSI • Decrease in feedwater temperature AOO (15.1.1.2) (Table 15.1-1 limiting minimum CHFR)	*1*	*2*	*3(f)*
d. DWSI • DWSI is designed to occur when Function 13.a, RTS, occurs.	*1*	-	-
e. PHT • Uncontrolled CRA withdrawal at power (15.4.2.2) • Pressurizer heaters are tripped on all automatic DHRS actuation signals. *(f) When not PASSIVELY COOLED.* *(g) With pressurizer heater breakers closed.*	*1*	*2(g)*	*3(g)*
14. High RCS Average Temperature • *Designed to protect against exceeding CHF limits for reactivity events (Table 15.0-7)*			
a. RTS • Failure of the non-safety-related module heatup system during startup condition reducing RCS flow (9.3.4)	*1*	-	-
b. DWSI • DWSI is designed to occur when Function 15.a, RTS, occurs.	*1*	-	-
15. Low RCS Flow • *Designed to ensure boron dilution cannot be performed at low RCS flow rates where the loop time is too long to be able to detect the reactivity change in the core within sufficient time to mitigate the event (Table 15.0-7)*			
a. DWSI • Inadvertent *decrease* in RCS boron concentration (15.4.6.2)	*1*	*2*	*3*
16. Low Low RCS Flow • *Designed to ensure flow remains measurable and positive during low-power startup conditions (Table 15.0-7)*			
a. RTS • Failure of the non-safety-related module heatup system during startup condition reducing RCS flow (9.3.4)	*1*	*2(a)*	*3(a)*

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	*APPLICABLE MODES*		
b. DWSI • DWSI is designed to occur when Function 16.a, RTS, occurs. *(a) When capable of withdrawal of more than one CRA.*	*1*	*2*	*3*
17. High Main Steam Pressure • Designed *to detect and mitigate loss of main steam demand to protect primary and secondary pressure limits during heatup events (Table 15.0-7).*			
a. RTS • Loss of external load AOO (15.2.1.2; limiting minimum CHFR) • Turbine trip AOO (15.2.2.1) • Loss of condenser vacuum AOO (15.2.3.1) • Closure of MSIV(s) AOO (15.2.4.2) • Inadvertent DHRS actuation (one valve opens, turbine load controller ineffective; turbine bypass not credited) (15.2.9.2, 10.4.7.2) • Loss of nonemergency AC power to the station auxiliaries (15.2.6; Table 15.2-17, Limiting Minimum CHFR) • Spectrum of rod ejection accidents (15.4.8.2.4) • Failure of small lines carrying primary coolant outside containment infrequent event (15.6.2.3.2) (CVCS letdown line break with loss of normal AC, Table 15.6-1, Maximum Mass Release)	*1*	*2(a)*	*-*
b. DHRS • Decrease in feedwater temperature AOO (15.1.1.2) • Increase in feedwater flow AOO (15.1.2.2) • Loss of external load AOO (15.2.1.2; limiting minimum CHFR) • Turbine trip AOO (15.2.2.1) • Loss of condenser vacuum AOO (15.2.3.1) • Closure of MSIV(s) AOO (15.2.4.2) • Loss of nonemergency AC power to the station auxiliaries (15.2.6, Table 15.2-17, Limiting Minimum CHFR) • CVCS malfunction AOO (15.5.1.2) (Table 15.5-1 limiting SG pressure case, pressurizer spray available) • Failure of small lines carrying primary coolant outside containment infrequent event (15.6.2.3.2) • SGTF postulated accident (SGTF limiting SG pressure case (Table 15.6-5)) • Double-ended CVCS letdown line break outside containment with coincident loss of normal AC power (15.6.2.3.3) • Failure of primary coolant carrying piping outside containment (Table 15.6-1)	*1*	*2*	*3(f)*
c. SSI • Decrease in feedwater temperature AOO (15.1.1.2) • CVCS malfunction AOO (15.5.1.2) (Table 15.5-1 limiting SG pressure case, pressurizer spray available)	*1*	*2*	*3(f)*
d. DWSI • DWSI is designed to occur when Function 17.a, RTS, occurs.	*1*	*2(a)*	*-*

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	APPLICABLE MODES		
e. PHT • Pressurizer heaters are tripped on all automatic DHRS actuation signals. *(a) When capable of CRA withdrawal.* *(f) When not PASSIVELY COOLED.* *(g) With pressurizer heater breakers closed.*	*1*	*2(g)*	*3(g)*
18. Low Main Steam Pressure • *Designed to detect and mitigate secondary HELB outside containment to protect steam generator inventory and DHRS functionality (Table 15.0-7)*			
a. RTS • Increase in steam flow AOO (15.1.3.2) • Steam piping failures inside and outside containment (15.1.5.2) • Inadvertent DHRS actuation AOO (15.2.9.2, 10.4.7.2) • Steamline break outside containment (15.1.5, 10.4.7.2)	*1(h)*	-	-
b. SSI • Increase in steam flow AOO (15.1.3.2) • Feedwater system pipe breaks inside and outside of containment (15.2.8.1)	*1(h)*	-	-
c. DWSI • DWSI is designed to occur when Function 18.a, RTS, occurs. *(h) With narrow range RCS hot temperature above the T-4 interlock.*	*1(h)*	-	-
19. Low Low Main Steam Pressure • *Designed to detect and mitigate secondary HELB outside containment to protect steam generator inventory and DHRS functionality (Table 15.0-7)*			
a. RTS • Steamline break outside containment (15.1.5, 10.4.7.2) during low-power startup conditions	*1*	*2(a)*	*3(a)*
b. SSI • Steamline break outside containment (15.1.5, 10.4.7.2)	*1*	*2(k)*	*3(k)*
c. DWSI • DWSI is designed to occur when Function 19.a, RTS, occurs. *(a) When capable of withdrawal of more than one CRA.* *(b) When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock.* *(k) With containment water level below the L-1 interlock with RCS temperature above the T-3 interlock, or with containment water level below the L-1 interlock with V-1 not active (both FWIVs open).*	*1*	*2(a)*	*3(b)*
20. High Steam Superheat • *Designed to detect and mitigate steam generator boil off to protect DHRS functionality during at power and post trip conditions (Table 15.0-7)*			
a. RTS	*1*	-	-

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	*APPLICABLE MODES*		
• Decrease in feedwater temperature AOO (15.1.1.2) • Increase in feedwater flow AOO *(not* credited*)* (15.1.2.2) (Table 15.1-4 limiting minimum CHFR) • Increase in steam flow AOO *(not credited)* (15.1.3.2) • Steam piping failures inside and outside containment (15.1.5.2) • Feedwater system pipe breaks inside and outside of containment (15.2.8.1) • Inadvertent DHRS actuation (one valve opens, turbine load controller ineffective; turbine bypass not credited) (15.2.9.2) • Spectrum of rod ejection accidents (15.4.8.2.4) • Failure of primary coolant carrying piping outside containment (Table 15.6-1) • Double-ended CVCS letdown line break outside containment with coincident loss of normal AC power (15.6.2.3.3)			
b. SSI • Decrease in feedwater temperature AOO (15.1.1.2) • Increase in feedwater flow (15.1.2) (Table 15.1-4 limiting minimum CHFR) • Feedwater system pipe breaks inside and outside of containment (15.2.8.1) • Double-ended CVCS letdown line break outside containment with coincident loss of normal AC power (15.6.2.3.3) • Failure of primary coolant carrying piping outside containment (Table 15.6-1)	*1*	-	-
c. DWSI • DWSI is designed to occur when Function 20.a, RTS, occurs.	*1*	-	-
21. Low Steam Superheat • *Designed to detect and mitigate steam generator overfilling to protect DHRS functionality during at power and post trip conditions (Table 15.0-7)*			
a. RTS • Steam piping failures inside and outside containment (5.1.5.2)	*1(l)*	-	-
b. SSI • Increase in feedwater flow AOO (15.1.2.2)	*1(l)*	*2(m)*	*3(m)*
c. DWSI • DWSI is designed to occur when Function 21.a, RTS, occurs. *(l) With V-1 not active (both FWIVs open).* *(m) With containment water level below the L-1 interlock and with V-1 not active (both FWIVs open).*	*1(l)*	-	-
22. High Narrow Range Containment Pressure • *Designed to detect and mitigate RCS or secondary leaks above the allowable limits to protect RCS inventory and ECCS function during these events (Table 15.0-7)*			
a. RTS • Main steamline break inside and outside containment postulated accident (6.2.1.4.4, 15.1.5.1, 15.1.5.2) • Feedwater line break postulated accident (6.2.1.4.4, 15.1.5.1)	*1*	*2(a)*	*3(a)*

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	APPLICABLE MODES		
• Feedwater system pipe breaks inside and outside of containment (15.2.8,15.2.8.3) • Steam piping failures inside and outside containment (15.1.5.2) • Loss of CNV vacuum, or CNV flooding (15.1.6.3) • LOCAs resulting from a spectrum of postulated piping breaks within the RCPB postulated accident (15.6.5.2, 15.6.5.3.3) • Inadvertent operation of ECCS—RRV opens with loss of normal AC and DC power (reactor vent valve (RVV) opening bounds reactor safety valve (RSV) opening) AOO (15.6.6.3.2/3)			
b. CIS • Main steamline break inside containment postulated accident (15.1.5.1) • Loss of CNV vacuum, or CNV flooding (15.1.6.3) • Feedwater system pipe breaks inside and outside of containment (15.2.8, 15.2.8.3) • Inadvertent operation of ECCS (RVV opening bounds RSV opening) AOO (15.6.6.3.2) • LOCA (15.6.5.2)	*1*	*2*	*3(i)*
c. DHRS • Loss of CNV vacuum, or CNV flooding (15.1.6.3) • Feedwater system pipe breaks inside and outside of containment (15.2.8, 15.2.8.3) • Inadvertent operation of ECCS (RVV opening bounds RSV opening) AOO (15.6.6.3.2) • LOCA (15.6.5.2)	*1*	*2*	*3(i)*
d. SSI • Main steamline break inside containment postulated accident (6.2.1.4.4, 15.1.5.1) • Loss of CNV vacuum, or CNV flooding (15.1.6.3) • Feedwater line break postulated accident (6.2.1.4.4, 15.1.5.1) • Feedwater system pipe breaks inside and outside of containment (15.2.8, 15.2.8.3) • Inadvertent operation of ECCS (RVV opening bounds RSV opening) AOO (15.6.6.3.2/3) • LOCA (15.6.5.2)	*1*	*2*	*3(i)*
e. DWSI • DWSI is designed to occur when Function 22.a, RTS, occurs.	*1*	*2(a)*	*3(b)*
f. CVCSI • RCS or secondary leaks above the allowable limits to protect RCS inventory and ECCS function by isolating part of the boundary (15.6.5)	*1*	*2*	*3(i)*
g. PHT • Main steamline break inside containment postulated accident (6.2.1.4.4, 15.1.5.1)	*1*	*2(g)*	*3(n)*

(a) When capable of withdrawal of more than one CRA.
(b) When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock.
(g) With pressurizer heater breakers closed.

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	*APPLICABLE MODES*		
(i) With RCS temperature above the T-3 interlock. *(n) With wide range RCS cold temperature above the T-3 interlock and pressurizer heater breakers closed.*			
23. Low Reactor Pressure Vessel Riser Level • *Designed to actuate ECCS upon riser uncovery for LOCA events (Table 15.0-7)*			
a. ECCS • LOCAs resulting from a spectrum of postulated piping breaks within the RCPB postulated accident (15.6.5.3.3) • Inadvertent operation of ECCS (RVV opening bounds RSV opening) AOO (15.6.6.3.2)	*1(o)*	*2(o)*	-
(o) With RCS temperature above the T-5 interlock.			
24. Low Low Reactor Pressure Vessel Riser Level • *Designed to actuate ECCS before the upper riser holes uncover (Table 15.0-7)*			
a. ECCS • LOCAs resulting from a spectrum of postulated piping breaks within the RCPB postulated accident (15.6.5.3.3) • Inadvertent operation of ECCS (RVV opening bounds RSV opening) AOO (15.6.6.3.2)	*1*	*2*	*3(f)*
(f) When not PASSIVELY COOLED.			
25. Low AC Voltage to Augmented DC Power System (EDAS) Battery Chargers • Designed to ensure appropriate load shedding occurs to EDAS in the event of extended loss of normal AC power to the EDAS battery chargers			
a. RTS • SGTF postulated accident (with a coincident loss of normal AC power (15.6.3.3.2) (Table 15.6-4)	*1*	*2(a)*	*3(a)*
b. CIS • SGTF postulated accident (with a coincident loss of normal AC power (15.6.3.3.2) (Table 15.6-4)	*1*	*2*	*3*
c. DHRS • Loss of external load (15.2.1) • DHRS extended passive cooling following a design basis event (15.0.5, 15.0.5.2.2) • SGTF postulated accident (with a coincident loss of normal AC power (15.6.3.3.2)	*1*	*2*	*3(f)*
d. SSI • Loss of external load (15.2.1) • SGTF postulated accident (with a coincident loss of normal AC power (15.6.3.3.2)	*1*	*2*	*3(f)*
e. DWSI • DWSI is designed to occur when Function 25.a, RTS, occurs	*1*	*2(a)*	*3(b)*
f. CVCSI • SGTF postulated accident (with a coincident loss of normal AC power (15.6.3.3.2)	*1*	*2*	*3*

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)	APPLICABLE MODES		
g. PHT • Pressurizer heaters are tripped on all automatic DHRS actuation signals.	*1*	*2(g)*	*3(g)*
h. ECCS • 24 hour timer delays actuation to conserve EDAS battery capacity • 8 hour timer after RTS actuation causes ECCS actuation to ensure ESB functions to ensure the boron concentration of reactor coolant collected in containment, which flows through the RRVs to the RPV downcomer, will preclude a return to criticality by compensating for the positive reactivity added by the core power reduction, xenon decay and RCS cooldown; this is implemented as a part of Actuation Function 1, ECCS," of LCO 3.3.3, "ESFAS Logic and Actuation" • Extended passive cooling following a DBE—ECCS extended passive cooling) (15.0.5.3) • LOCA resulting from postulated piping breaks within RCPB (15.6.5) • SGTF postulated accident (15.6.3.3.2)—with a coincident loss of normal AC power is limiting for RCS pressure (Table 15.6-4)	*1*	*2*	*3(f)*
(a) When capable of withdrawal of more than one CRA. *(b) When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock.* *(f) When not PASSIVELY COOLED.* *(g) With pressurizer heater breakers closed.*			
26. High Under-the-Bioshield Temperature • *Designed to detect high energy leaks or breaks at the top of the NPM under the bioshield to reduce the consequences of the harsh temperature conditions from high energy line breaks (HELBs) on the safety-related equipment located on top of the module (e.g., exposure of MSIVs and DHRS valves to sustained elevated temperatures)*			
a. RTS • High-energy line breaks under the bioshield (3.6)	*1*	*2(a)*	*3(a)*
b. CIS • High-energy line breaks under the bioshield (3.6)	*1*	*2*	*3*
c. DHRS • High-energy line breaks under the bioshield (3.6)	*1*	*2*	*3(f)*
d. SSI • High-energy line breaks under the bioshield (3.6)	*1*	*2*	*3(f)*
e. DWSI • DWSI is designed to occur when Function 26.a, RTS, occurs.	*1*	*2(a)*	*3(b)*
f. CVCSI • High-energy line breaks under the bioshield (3.6)	*1*	*2*	*3*
g. PHT • High-energy line breaks under the bioshield (3.6)	*1*	*2(g)*	*3(g)*
(a) When capable of withdrawal of more than one CRA. *(b) When capable of withdrawal of more than one CRA and the RCS temperature is above*			

MPS INSTRUMENTATION FUNCTION SUPPORTED RTS AND ESFAS LOGIC AND ACTUATION FUNCTION(S)			*APPLICABLE MODES*
the T-3 interlock. *(f) When not PASSIVELY COOLED.* *(g) With pressurizer heater breakers closed.*			
27. High RCS Pressure – Low Temperature Overpressure Protection (LTOP)			
a. LTOP • Low temperature overpressure events, limiting event is spurious actuation of the pressurizer heaters (5.2.2.2.2) *(p) With RCS temperature below the LTOP enable temperature specified in the PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (T-1 interlock) and both reactor vent valves closed.*	-	-	*3(p)*

- LCO 3.3.2 Reactor Trip System (RTS) Logic and Actuation

The following MPS instrumentation functions initiate an RTS actuation. SER Table 16.4.1-2 uses italics to denote requirements included in GTS Subsection 3.3.1 for MPS instrumentation Functions. Regular font denotes requirements included in GTS Subsection 3.3.2 for the RTS Logic and Actuation Function and discussions of interlocks referenced in footnotes in GTS Table 3.3.1-1.

Table 16.4.1-2 RTS Logic and Actuation Function

RTS LOGIC AND ACTUATION FUNCTION *ASSOCIATED MPS INSTRUMENTATION FUNCTIONS*	APPLICABLE MODES		
LCO 3.3.2 **Reactor Trip System**	1	2*	3*
* When capable of CRA withdrawal.			
1.a. High Power Range Linear Power	*1*	*2(a)*	*3(a)*
2.a. High Power Range Positive and Negative Rate	*1(c)*	*-*	*-*
3.a. High Intermediate Range Log Power Rate	*1(d)*	*2(a)*	*3(a)*
4.a. High Source Range Count Rate	*1(e)*	*2(a)*	*3(a)*
5.a. High Source Range Log Power Rate	*1(e)*	*2(a)*	*3(a)*
7.a. High Pressurizer Pressure	*1*	*2(a)*	*3(a)*
8.a. Low Pressurizer Pressure	*1(h)*	*-*	*-*
9.a. Low Low Pressurizer Pressure	*1*	*2(a)*	*3(b)*
10.a. High Pressurizer Level	*1*	*2(a)*	*3(a)*
11.a. Low Pressurizer Level	*1*	*2(a)*	*3(a)*
13.a. High Narrow Range RCS Hot Temperature	*1*	*-*	*-*
14.a High RCS Average Temperature	*1*	*-*	*-*
16.a. Low Low RCS Flow	*1*	*2(a)*	*3(a)*
17.a. High Main Steam Pressure	*1*	*2(a)*	*-*
18.a. Low Main Steam Pressure	*1(h)*	*-*	*-*
19.a. Low Low Main Steam Pressure	*1*	*2(a)*	*3(a)*
20.a. High Steam Superheat	*1*	*-*	*-*
21.a. Low Steam Superheat	*1(l)*	*-*	*-*
22.a. High Narrow Range Containment Pressure	*1*	*2(a)*	*3(a)*
25.a. Low AC Voltage [on Low Voltage AC Electrical Distribution System (ELVS)] to EDAS Battery Chargers	*1*	*2(a)*	*3(a)*
26.a. High Under-the-Bioshield Temperature	*1*	*2(a)*	*3(a)*

RTS LOGIC AND ACTUATION FUNCTION *ASSOCIATED MPS INSTRUMENTATION FUNCTIONS*	APPLICABLE MODES

(a) When capable of withdrawal of more than one CRA.

(b) When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock.

(c) With power above the N-2H Interlock.

The power range linear power interlock N-2H automatically bypasses (enables) MPS RTS Function 2.a below (above) 15% RTP (bypass requires 3 of 4 channels < 15% RTP).

(d) With power below the N-2L interlock.

The power range linear power interlock N-2L automatically bypasses (enables) MPS RTS Function 3.a above (below) 15% RTP (bypass requires 3 of 4 channels > 15% RTP).

The power range linear power permissive N-2L allows manual bypass of (automatically enables) MPS RTS Function 1.a, High-1 above (below) 15% RTP (permissive requires 3 of 4 channels > 15% RTP).

(e) When Intermediate Range Log Power less than N-1 interlock.

The intermediate range log power permissive N-1 allows manual bypass of (automatically enables) MPS RTS Functions 4.a and 5.a above (below) 1 x 10^5 cps (one decade above channel lower range limit).

(h) With RCS temperature above the T-4 interlock.

The narrow range RCS hot temperature interlock T-4 automatically enables (bypasses) MPS RTS Function 8.a above (below) 260°C (500°F).

(l) With V-1 not active (both FWIVs open).

The FWIV closed interlock V-1 automatically bypasses MPS RTS Function 21.a when at least one FWIV does not indicate open.

The FWIV closed interlock V-1 automatically enables MPS RTS Function 21.a when both FWIVs indicate open (V-1 not active).

- LCO 3.3.3 Engineered Safety Features Actuation System (ESFAS) Logic and Actuation

The following listed MPS instrumentation Functions initiate the indicated ESFAS Logic and Actuation Functions. In Table 16.4.1-3, this SER uses italics to denote requirements included in GTS Subsection 3.3.1 for MPS instrumentation Functions. Regular font denotes requirements included in GTS Subsection 3.3.3 for the ESFAS Logic and Actuation Functions and discussions of interlocks referenced in footnotes in Table 3.3.1-1.

Table 16.4.1-3 ESFAS Logic and Actuation Functions

ESFAS LOGIC AND ACTUATION FUNCTION *ASSOCIATED MPS INSTRUMENTATION FUNCTION(S)*	APPLICABLE MODES		
1. **Emergency Core Cooling System** (ECCS)	1	2	3(a)

(a) When not PASSIVELY COOLED.

ESFAS LOGIC AND ACTUATION FUNCTION *ASSOCIATED MPS INSTRUMENTATION FUNCTION(S)*	APPLICABLE MODES		
23.a. Low Reactor Pressure Vessel Riser Level	*1(o)*	*2(o)*	*-*
24.a. Low Low Reactor Pressure Vessel Riser Level	*1*	*2*	*3(f)*
25.h. Low AC Voltage to Augmented DC Power System (EDAS) Battery Chargers			
(f) When not PASSIVELY COOLED.			
(o) With RCS temperature above the T-5 interlock.			
2. **Decay Heat Removal System** (DHRS)	1	2	3(a)
(a) When not PASSIVELY COOLED			
7.b. High Pressurizer Pressure	*1*	*2*	*3(f)*
11.c Low Pressurizer Level	*1(h)*	*-*	*-*
13.b. High Narrow Range RCS Hot Temperature	*1*	*2*	*3(f)*
17.b. High Main Steam Pressure	*1*	*2*	*3(f)*
22.c High Narrow Range Containment Pressure	*1*	*2*	*3(i)*
25.c. Low AC Voltage to EDAS [augmented DC power system] Battery Chargers	*1*	*2*	*3(f)*
26.c High Under-the-Bioshield Temperature	*1*	*2*	*3(f)*
(f) When not PASSIVELY COOLED.			
(h) With RCS temperature above the T-4 interlock. • The narrow range RCS hot temperature interlock T-4 automatically enables (bypasses) MPS ESFAS Function 11.c above (below) 260°C (500°F).			
(i) With RCS temperature above the T-3 interlock. • When T-3 interlock is active (3 of 4 channels of wide range RCS hot temperature are < 171.1°C (340°F)), T-3 automatically bypasses MPS ESFAS Function 22.c. • When T-3 interlock is not active (2 or more channels of wide range RCS hot temperature are > 171.1°C (340°F), T3 automatically enables MPS ESFAS Function 22.c.			
3. **Containment Isolation System** (CIS)	1	2	3
11.b. Low Pressurizer Level	*1(h)*	*-*	*-*
12.a. Low Low Pressurizer Level	*1*	*2*	*3(j)*
22.b. High Narrow Range Containment Pressure	*1*	*2*	*3(i)*
25.c. Low AC Voltage to EDAS [augmented DC power system] Battery Chargers	*1*	*2*	*3*
26.b. High Under-the-Bioshield Temperature	*1*	*2*	*3*
(h) With RCS temperature above the T-4 interlock. • The narrow range RCS hot temperature interlock T-4 automatically enables (bypasses) MPS ESFAS Function 11.b above (below) 260°C (500°F).			
(i) With RCS temperature above the T-3 interlock. • When T-3 interlock is active (3 of 4 channels of wide range RCS hot temperature are < 171.1°C (340°F)), T-3 automatically bypasses MPS ESFAS Function 22.b. • When T-3 interlock is not active (2 or more channels of wide range RCS			

ESFAS LOGIC AND ACTUATION FUNCTION *ASSOCIATED MPS INSTRUMENTATION FUNCTION(S)*	APPLICABLE MODES

hot temperature are > 171.1°C (340°F), T3 automatically enables MPS ESFAS Function 22.b.

(j) *With RCS temperature above the T-2 interlock and containment water level below the L-1 interlock.*

- When the L-1 interlock is not active
(2 of 4 containment water level channels are ≤ 13.7 m (45 ft)),
and
the T-2 interlock is not active
(2 of 4 wide range RCS Thot channels are ≥ 93.3°C (200°F)),
MPS ESFAS Function 12.a is automatically enabled.

- When the L-1 interlock is active
(3 of 4 containment water level channels are > 13.7 m (45 ft)),
or
the T-2 interlock is active,
(3 of 4 wide range RCS Thot channels are < 93.3°C (200°F)),
MPS ESFAS Function 12.a is automatically bypassed.

4.	**Demineralized Water Supply Isolation** (DWSI)	1	2	3
1.b.	*High Power Range Linear Power*	*1*	*2(a)*	*3(b)*
2.b.	*High Power Range Positive and Negative Rate*	*1(b)*	*-*	*-*
3.b.	*High Intermediate Range Log Power Rate*	*1(d)*	*2(a)*	*3(b)*
4.b.	*High Source Range Count Rate*	*1(e)*	*2(a)*	*3(b)*
5.b.	*High Source Range Log Power Rate*	*1(e)*	*2(a)*	*3(b)*
6.a.	*High Subcritical Multiplication*	*1(e)*	*2*	*3*
7.d.	*High Pressurizer Pressure*	*1*	*2(a)*	*3(b)*
8.c.	*Low Pressurizer Pressure*	*1(h)*	*-*	*-*
9.b.	*Low Low Pressurizer Pressure*	*1*	*2(a)*	*3(b)*
10.b.	*High Pressurizer Level*	*1*	*2(a)*	*3(b)*
11.e.	*Low Pressurizer Level*	*1*	*2(a)*	*3(b)*
13.d.	*High Narrow Range RCS Hot Temperature*	*1*	*-*	*-*
14.b.	*High RCS Average Temperature*	*1*	*-*	*-*
15.a.	*Low RCS Flow*	*1*	*2*	*3*
16.b.	*Low Low RCS Flow*	*1*	*2*	*3*
17d.	*High Main Steam Pressure*	*1*	*2(a)*	*-*
18.c.	*Low Main Steam Pressure*	*1(h)*	*-*	*-*
19.c	*Low Low Main Steam Pressure*	*1*	*2(a)*	*3(b)*
20.c.	*High Steam Superheat*	*1*	*-*	*-*
21.c.	*Low Steam Superheat*	*1(l)*	*-*	*-*
22.e.	*High Narrow Range Containment Pressure*	*1*	*2(a)*	*3(b)*
25.e.	*Low AC Voltage to EDAS [augmented DC power system] Battery Chargers*	*1*	*2(a)*	*3(b)*
26.e.	*High Under-the-Bioshield Temperature*	*1*	*2(a)*	*3(b)*

(a) *When capable of withdrawal of more than one CRA.*

(b) *When capable of withdrawal of more than one CRA and the RCS temperature above the T-3 interlock.*

(d) *With power below the N-2L interlock.*

The power range linear power interlock N-2L automatically

ESFAS LOGIC AND ACTUATION FUNCTION *ASSOCIATED MPS INSTRUMENTATION FUNCTION(S)*	APPLICABLE MODES

bypasses (enables) MPS ESFAS Function 3.b above (below) 15% RTP (bypass requires 3 of 4 channels > 15% RTP).

The power range linear power permissive N-2L allows manual bypass of (automatically enables) MPS ESFAS Function 1.b, High-1 above (below) 15% RTP (permissive requires 3 of 4 channels > 15% RTP).

(e) When Intermediate Range Log Power less than N-1 interlock.

The intermediate range log power permissive N-1 allows manual bypass of (automatically enables) MPS ESFAS Functions 4.b and 5.b above (below) 1E5 cps (one decade above channel lower range limit) (permissive requires 3 of 4 channels > 1E5 cps).

The intermediate range log power interlock N-1 automatically bypasses (enables) MPS ESFAS Function 6.a above (below) 1E5 cps (bypass requires 3 of 4 channels > 1E5 cps).

(h) With RCS temperature above the T-4 interlock.

The narrow range RCS hot temperature interlock T-4 automatically enables (bypasses) MPS ESFAS Function 8.b above (below) 260°C (500°F).

(l) With V-1 interlock not active (both FWIVs open).

The V-1 interlock is active if at least one FWIV does not indicate open, and automatically bypasses MPS ESFAS Function 21.c in MODE 1.

The FWIV closed interlock V-1 automatically enables ESFAS Function 21.c when both FWIVs indicate open (V-1 interlock not active).

5.	**CVCS Isolation** (CVCSI)	1	2	3
10.c.	*High Pressurizer Level*	*1*	*2*	*3*
11.f.	*Low Pressurizer Level*	*1(h)*	*-*	*-*
12.b.	*Low Low Pressurizer Level*	*1*	*2*	*3(j)*
22.f.	*High Narrow Range Containment Pressure*	*1*	*2*	*3(i)*
25.f	*Low AC Voltage to EDAS [augmented DC power system] Battery Chargers*	*1*	*2*	*3*
26.f	*High Under-the-Bioshield Temperature*	*1*	*2*	*3*

(h) With RCS temperature above the T-4 interlock.

(j) With RCS temperature above the T-2 interlock and containment water level below the L-1 interlock.

- When the L-1 interlock is not active (below L-1) and the T-2 interlock is not active (above T-2) MPS ESFAS Function 12.b is automatically enabled.
- When the L-1 interlock is active (above L-1) or the T-2 interlock is active (below T-2) MPS ESFAS Function 12.b is automatically bypassed.

(i) With RCS temperature above the T-3 interlock.

- When T-3 interlock is active (3 of 4 channels of wide range RCS hot temperature are < 171.1°C (340°F)), T-3 automatically bypasses MPS ESFAS Function 22.f.

ESFAS LOGIC AND ACTUATION FUNCTION *ASSOCIATED MPS INSTRUMENTATION FUNCTION(S)*	APPLICABLE MODES		
• When T-3 interlock is not active (2 or more channels of wide range RCS hot temperature are > 171.1°C (340°F)), T-3 automatically enables MPS ESFAS Function 22.f.			
6. **Pressurizer Heater Trip**	1	2(b)	3(b)
(b) With pressurizer heater breakers closed.			
7.e. High Pressurizer Pressure	*1*	*2(g)*	*3(g)*
11.g. Low Pressurizer Level	*1*	*2(g)*	*3(g)*
13.e. High Narrow Range RCS Hot Temperature	*1*	*2(g)*	*3(g)*
17.e. High Main Steam Pressure	*1*	*2(g)*	*3(g)*
22.g High Narrow Range Containment Pressure	*1*	*2(g)*	*3(n)*
25.f. Low AC Voltage to Low Voltage AC Electrical Distribution System EDAS Battery Chargers	*1*	*2(g)*	-
(g) With pressurizer heater breakers closed.			
(n) With RCS temperature above the T-3 interlock and pressurizer heater breakers closed.			
7. **Low Temperature Overpressure Protection** (LTOP)	-	-	3(c)
(c) With RCS temperature below the LTOP enable temperature specified in the PTLR (T-1 interlock) and both reactor vent valves closed.			
27.a High RCS Pressure – Low Temperature Overpressure Protection	-	-	*3(p)*
(p) With RCS temperature below the LTOP enable temperature specified in the PTLR (T-1 interlock) and both reactor vent valves closed.			
With at least 3 of 4 wide range RCS cold temperature channels above the T-1 interlock setpoint (the nil ductility transition temperature) 143.3°C (290°F), T-1 is active and the LTOP actuation Function is automatically bypassed. Below T-1, the LTOP actuation Function is automatically enabled. The LTOP actuation Function opens the closed reactor vent valves on 2 of 4 MPS high wide range RCS pressure channels with indication above the LTOP setpoint, which is calculated for each channel as a function of wide range RCS cold temperature, as indicated by FSAR Table 5.2-5, “LTOP Pressure Setpoint as Function of Cold Temperature,” and FSAR Figure 5.2-3, “Variable LTOP Setpoint.”			
8. **Secondary System Isolation (SSI)**	1	2	3(a)
(a) When not PASSIVELY COOLED.			
7.c. High Pressurizer Pressure	*1*	*2*	*3(f)*
8.b. Low Pressurizer Pressure	*1(h)*	-	-
11.d Low Pressurizer Level	*1*	*2*	*3(j)*
13.c. High Narrow Range RCS Hot Temperature	*1*	*2*	*3(f)*
17.c. High Main Steam Pressure	*1*	*2*	*3(f)*
18.b. Low Main Steam Pressure	*1(h)*	-	-
19.b. Low Low Main Steam Pressure	*1*	*2(k)*	*3(k)*
20.b. High Steam Superheat	*1*	-	-

ESFAS LOGIC AND ACTUATION FUNCTION *ASSOCIATED MPS INSTRUMENTATION FUNCTION(S)*	APPLICABLE MODES		
21.b. Low Steam Superheat	*1(l)*	*2(m)*	*3(m)*
22.d. High Narrow Range Containment Pressure	*1*	*2*	*3(i)*
25.d. Low AC Voltage to Low Voltage AC Electrical Distribution System Battery Chargers	*1*	*2*	*3(f)*
26.d. High Under-the-Bioshield Temperature	*1*	*2*	*3(f)*
(f) When not PASSIVELY COOLED. *(h) With RCS temperature above the T-4 interlock.* *(i) With RCS temperature above the T-3 interlock.* *(j) With RCS temperature above the T-2 interlock and containment water level below the L-1 interlock.* *(k) With containment water level below the L-1 interlock with RCS temperature above the T-3 interlock, or with containment water level below the L-1 interlock with V-1 interlock not active (both FWIVs open).* *(l) With V-1 interlock not active (both FWIVs open).* *(m) With containment water level below the L-1 interlock and with V-1 not active (both FWIVs open).*			
9. **Pressurizer Line Isolation**	1	2	3(d)
(d) With RCS temperature above the T-3 interlock.			
8.d Low Pressurizer Pressure.	*1*	*2*	*3(i)*
(i) With RCS temperature *above the T-3 interlock.*			

- LCO 3.4.4 Reactor Safety Valves (RSVs)

The two RSVs must be operable to provide RCS overpressure protection in Modes 1 and 2, and in Mode 3 above the low temperature overpressure protection (LTOP) T-1 interlock setpoint, which is specified in the PTLR referenced by GTS Subsection 5.6.4, “PTLR.”

In discussing the applicable safety analyses, Bases Subsection B 3.4.4 lists the FSAR Chapter 15 analyses that credit the functioning of the two RSVs for overpressure protection not only of the RCS but also the SG system. One RSV, in conjunction with the MPS, can prevent RCS pressure from exceeding the RCS pressure SL of 16.685 megapascals (MPa) (absolute) (2,420 pound-force per square inch (psia)). The RSV minimum relief capacity is based on a postulated overpressure transient of a turbine trip without turbine bypass capability.

- LCO 3.4.6 Chemical and Volume Control System (CVCS) Isolation Valves

This LCO addresses the CVCSI function of the eight containment isolation valves (CIVs) in the CVCS. The CVCS isolation valves must be operable in Modes 1, 2, and 3. Under normal conditions with the NPM in MODE 1, 2, or 3, the two valves in each of the three flow paths of RCS injection, RCS discharge, and pressurizer spray are open, and the two valves in the RPV high point degasification line are closed. All of the valves receive a close signal from ESFAS when two of four channels of any of the MPS functions listed under Functions 3 and 5 (CIS and CVCSI) in Table 16.4.1-3 above are tripped.

The CVCSI Function’s actuated devices, the eight CVCS CIVs, are required to be operable in Modes 1, 2, and 3 to provide mitigation for the SGTF postulated accident and the CVCS postulated break outside containment event. The CVCSI also prevents pressurizer overfill during non-LOCA transients. Automatic closure of these valves is assumed in the safety analyses of these events.

- LCO 3.4.10 Low Temperature Overpressure Protection (LTOP) Valves

This LCO is required to be met in Mode 3 with wide range RCS cold temperature below the (low temperature overpressure) T-1 interlock. SER Section 16.4.5 gives the staff’s evaluation of GTS Subsection 3.4.10.

- LCO 3.5.1 Emergency Core Cooling System (ECCS)

This LCO ensures the normally closed RVVs and RRVs will automatically open to mitigate the postulated LOCAs and the postulated SGTF accident when the unit is initially in Mode 1, Mode 2, or Mode 3 with RCS indicated temperature < 174°C (345°F) and not passively cooled. The ECCS, in combination with the required water level in the CNV, can also be used to conduct passive cooling in Mode 3 maintaining a safe-shutdown condition.

- LCO 3.5.2 Decay Heat Removal System (DHRS)

This LCO ensures the normally closed DHRS parallel actuation valves for each SG will automatically open, and that the normally open MSIVs and feedwater isolation valves (FWIVs) will automatically close to establish a natural circulation flow of DHRS water through the SG tubes, which transfers energy from the reactor coolant to the DHRS water flowing in the SG tubes, through the main steam header plenum to the inlets of the DHRS heat exchangers in the reactor building pool, which rejects the energy to the UHS, then to the outlet of the DHRS heat exchangers back to the feedwater SG inlet plenum, and again through the SG tubes. The DHRS passively removes core decay heat during non-LOCA events and cools down the RCS to the safe-shutdown condition of Mode 3 with passive cooling in operation, assuming the secondary heat sink is unavailable because of a concurrent loss of AC power to the condensate and feedwater system (FWS) and the air-cooled condenser system. The DHRS is relied upon to provide a passive means of decay heat removal in Modes 1 and 2 and must remain operable in Mode 3 until passive cooling is placed in operation, which can be done by placing the DHRS in operation.

- LCO 3.5.3 Ultimate Heat Sink

In Modes 1, 2, and 3, the minimum reactor pool water level of 15.85 m (52 ft) provides margin above the minimum level required to support DHRS and ECCS operation in response to LOCA and non-LOCA DBEs. The maximum pool water level of 16.46 m (54 ft) is an initial condition that ensures long-term cooling analyses assumptions. Pool water bulk average temperature and boron concentration must also be within limits to satisfy safety analysis assumptions. This LCO must be met at all times.

- LCO 3.5.4 Emergency Core Cooling System Supplemental Boron (ESB)

In Mode 1, this LCO requires the two ESB dissolvers to contain boron in a form and quantity within the limits specified in the COLR. This ensures the quantity of boron assumed in the accident analyses will be available to dissolve and ensure subcriticality as the module cools. The contents of both ESB dissolvers are credited with adding negative reactivity to the reactor by increasing the boron concentration of the reactor coolant as the coolant returns to the reactor vessel after an ECCS actuation. Each ESB dissolver must contain the form and quantity of boron specified in the COLR.

- LCO 3.6.1 Containment

The CNV functions to limit the release of fission products to the outside environment in the event of a postulated DBA involving fuel damage. In the NuScale US460 design, it is also an integral component of the ECCS system and is used in two methods of passive cooling. LCO 3.6.1 is applicable in Modes 1 and 2, and in Mode 3 when not passively cooled.

- LCO 3.6.2 Containment Isolation Valves (CIVs)

The CIVs automatically close in response to a CIS actuation signal and mitigate a DBA involving a main steamline break inside containment (FSAR Section 15.1.5, “Steam Piping Failures Inside and Outside of Containment”). LCO 3.6.2 is applicable in Modes 1 and 2, and in Mode 3 with an RCS temperature greater than or equal to 93.3°C (200°F).

- LCO 3.6.3 Containment Closure

Containment penetrations that provide direct access from the containment atmosphere to the outside atmosphere must be isolated on at least one side. Containment closure means potential inventory paths are closed or capable of being closed. Isolation may be achieved by a closed automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. The LCO requirements are referred to as “containment closure” rather than “containment OPERABILITY,” because the potential pressure transient in containment in the applicable modes is well below containment design pressure. Containment closure is required when an NPM is in Mode 3 and passively cooled, according to the Background section of the Bases for LCO 3.6.3, to protect “the coolant inventory necessary to ensure adequate core cooling and limits the potential for release of fission product radioactivity. The potential for module pressurization is limited by the amount of decay heat generated in the reactor core and the passive heat transfer capability of the module to the ultimate heat sink; therefore, requirements to isolate the containment can be less stringent.” Containment closure is also required when an NPM is in Mode 4 with the upper module assembly seated on lower CNV flange for the same reasons as in Mode 3 discussed above. TR-101310-NP, “US460 Standard Design Approval Technical Specifications Development,” Revision 1, Section 3.3.17, “Addition of Limiting Condition for Operation 3.6.3, Containment Closure,” states that containment closure is required in these modes to ensure “module liquid inventory to maintain core coverage and transfer decay heat from the reactor fuel to the ultimate heat sink. Containment closure ensures the inventory will remain available to perform this function during an extended loss of alternating current power or during delays in the transfer of the module between the operating location and the containment closure tool. Containment closure must be maintained until the containment is disassembled and the reactor vessel is thermally connected to the UHS via the de-energized ECCS valves.”

- LCO 3.6.4 Passive Autocatalytic Recombiner

The passive autocatalytic recombiner (PAR) in the upper CNV recombines hydrogen and oxygen to passively limit oxygen concentration in the containment atmosphere at less than four percent oxygen by volume to preclude formation of a combustible atmosphere in the CNV during both DBAs and significant beyond DBAs. LCO 3.6.4 is applicable in Modes 1 and 2, and in Mode 3 not passively cooled, which matches when the supported CNV is required to be operable by LCO 3.6.1.

- LCO 3.7.1 Main Steam Isolation Valves (MSIVs)

The MSIVs, including the backup MSIVs, automatically close in response to a CIS actuation signal, an SSI actuation signal (for affected SGs), and a DHRS actuation signal to mitigate the DBA involving a main steamline break outside containment (FSAR Section 15.1.5). LCO 3.7.1 is applicable in Modes 1 and 2, and in Mode 3 when not passively cooled.

- LCO 3.7.2 Feedwater Isolation

The FWIVs and the feedwater regulating valves (FWRVs) automatically close in response to a CIS actuation signal, an SSI actuation signal (for affected SGs), and a DHRS actuation signal to mitigate DBAs involving an SGTF, and a FWS pipe break, both inside and outside containment (FSAR Section 15.6.3, "Steam Generator Tube Failure (Thermal Hydraulic)," and Section 15.2.8, "Feedwater System Pipe Breaks Inside and Outside of Containment"). LCO 3.7.2 is applicable in Modes 1 and 2, and in Mode 3 when not passively cooled.

- LCO 3.8.1 Nuclear Instrumentation (refueling neutron flux [monitoring] channels)

In Mode 5, two of the three refueling neutron flux channels must be operable to ensure that redundant monitoring capability is available to detect changes in core reactivity during removal of the upper reactor vessel assembly and during movement of an irradiated fuel assembly in the reactor vessel. Each channel must provide visual indication in the control room. In addition, at least one of the two required channels must provide an operable audible count rate function to alert the operators to the initiation of a boron dilution event. This LCO is applicable in Mode 5, except when the reactor vessel upper assembly is seated on the reactor vessel flange.

Based on its evaluation of the SDAA rationale for identifying the operability of SSCs specified by the above listed LCOs as meeting Criterion 3, the staff finds that the GTS satisfy 10 CFR 50.36(c)(2)(ii)(C), Criterion 3.

16.4.1.4 Limiting Conditions for Operation Required by 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 following LCO requires operability of SSCs that provide backup to other LCO-required SSCs. Its inclusion in the GTS enhances the safe operation of the NPM by providing defense in depth.

- LCO 3.3.4 [RTS and ESF System] Manual Actuation Functions

GTS Table 3.3.4-1 of this LCO specifies the applicability requirements for each specified automatically actuated system's manual actuation function.

The staff concludes that designating the operability of the SSCs specified by the above-listed LCO as meeting Criterion 4 is beneficial to safety. Therefore, the staff finds that the GTS satisfy 10 CFR 50.36(c)(2)(ii)(D), Criterion 4.

16.4.1.5 Limiting Conditions for Operation Required by None of the Criteria

The following LCO specifies which LCO requirements may be suspended while testing the reactor, because physics testing requires exceeding the excepted LCO restrictions. This LCO also specifies other restrictions to ensure plant safety during physics testing. LCO 3.0.7

addresses physics testing by providing rules for entering and exiting the physics testing LCO and by providing an exception to LCO 3.0.1.

- LCO 3.1.8 PHYSICS TESTS Exceptions

Because this LCO and the associated LCO 3.0.7 are consistent with the W-STS and NuScale physics testing, which is described in NuScale US460 FSAR Chapter 14, "Initial Test Program and Inspections, Tests, Analyses and Acceptance Criteria," the staff finds that including LCO 3.0.7 and LCO 3.1.8 in the GTS is acceptable.

16.4.1.6 Non-Limiting Conditions for Operation-Required SSCs, Functions, and Process Variables Typically Addressed by Limiting Conditions for Operation in W-STS, CE-STS, or W-AP1000-STS

The staff reviewed the NuScale US460 design as described in the FSAR to confirm that omission of certain SSCs and parameter limits typically included in TS is justified.

- *LCO not* required *because parameter is implicitly ensured to be within limits by another LCO*

Containment Vessel Atmosphere Temperature and Pressure

STS Section 3.6 usually includes LCOs on containment temperature and pressure. The staff notes that the NuScale US460 design requires a very low CNV internal pressure for leakage detection instrumentation operability. Therefore, meeting LCO 3.4.7 maintains the initial value of CNV pressure in MODES 1 and 2, and in MODE 3 above 93.3°C (200°F), which is an assumption in the accident analysis. Accordingly, an LCO directly tied to the assumed initial value of CNV pressure is not necessary under Criterion 2. In addition, accident analysis conclusions are insensitive to the initial mass and energy content and the temperature of the CNV atmosphere because of the near-vacuum initial pressure of the CNV. Therefore, an LCO to ensure the validity of the initial value assumption for the CNV atmosphere temperature is also unnecessary. For these reasons, the staff finds that omitting explicit LCOs for initial containment pressure and temperature is acceptable.

- *LCO not required because system is classified as not safety-related in NuScale design*

Electrical Power (Includes Offsite (Preferred) AC Electrical Power Sources; Onsite Alternating AC (Standby) Electrical Power Sources; Onsite DC Electrical Power Sources (Batteries); Battery Chargers; DC-to-AC Inverters; Battery Parameters; AC and DC Electrical Power Distribution Systems)

SER Chapter 8, "Electric Power," provides the staff's evaluation of electrical power systems in the NuScale US460 standard design. All FSAR Chapter 15 analyses of design basis postulated anticipated operational occurrences (AOOs) (except for AOOs involving ECCS RVV opening at event initiation) assume both ECCS RVVs are initially closed and remain closed for the analyzed duration of the transient. SER Chapter 8 evaluates the augmented quality aspects of the non-safety-related module-specific augmented quality DC electrical power system (EDAS-MS). An LCO need not be established for EDAS-MS electrical power support of this RVV "ECCS hold" function to provide reasonable assurance of adequate protection of public health and safety because a COL holder (or licensee) for a NuScale US460 standard design NPM in a multi-module facility will control the reliability and availability of the EDAS-MS according to a licensee-controlled document established to fulfill COL Item 16.1-2. This item requires a COL application referencing the NuScale US460 standard design approval to provide an “owner-controlled requirements manual” as a part of the FSAR that will contain “owner-controlled limits and requirements described in the Bases of the Technical Specifications or as otherwise specified in the FSAR.”

<u>Control Room Habitability</u> (includes Bottled Air System; Radioactivity Filtration System; Control Room Envelope Passive Cooling System; Control Room Normal HVAC [Heating, Ventilation, and Air Conditioning] System, Control Room Envelope Air Temperature and Humidity Limits; Control Room Envelope Boundary Unfiltered Inleakage Limit; Automatic Initiation of Control Room Isolation Mode and Bottled Air System on Detection of High Radiation in Outside Air Intake and on Detection of Toxic Gas).

SER Chapter 6, “Engineered Safety Features,” and Chapter 9, “Auxiliary Systems,” provide the staff’s evaluation of standby and normal heating, ventilation, and air conditioning systems, respectively.

- *LCO not required because system or component is not part of the NuScale Design*

The following are LCO-required SSCs in operating PWRs that are not included in the NuScale US460 standard design:

> Remote Shutdown Station (RSS) [Monitoring Instrumentation]
> RCS Loops (external to the RPV)
> Pressurizer Power-Operated Relief Valves
> Containment Purge Supply and Exhaust Ventilation System and Isolation Dampers

- *LCO not required because the 1988 Split Report’s rationale for including an optional system is not satisfied* [5]

<u>Postaccident Monitoring Instrumentation</u>

FSAR Section 7.1.1.2.2, “Post-Accident Monitoring,” states the following, in part:

> Post-accident monitoring [PAM] is a non-safety-related function. The PAM instrumentation includes the required functions, range, and accuracy for each variable monitored. The selection of each type of variable follows the guidance provided in [Institute of Electrical and Electronics Engineers] IEEE [Standard] Std 497-2016, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations” (Reference 7.1-6), as modified by [Regulatory Guide] RG 1.97[, Revision 5 “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” issued April 2019].
>
> [PAM Type B and C] Variables and their type classification are based on their accident management function as identified in abnormal operating procedures, emergency operating procedures, and emergency procedure guidelines. Because the abnormal and emergency operating procedures and guidelines have not been developed, NuScale developed an approach to identify PAM

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[5] Thomas E. Murley, Director, Office of Nuclear Reactor Regulation, to Walter S. Wilgus, Chairman, The B&W Owners Group, “NRC Staff Review of Nuclear Steam Supply System Vendors Owners Groups’ Application of the Commission’s Interim Policy Statement Criteria to Standard Technical Specifications,” May 9, 1988, ML11264A057 (1988 Split Report).

> variables as described below. [SER Section 7.2.13 gives the staff's evaluation of the alternate approach for selecting PAM variables.]
>
> …
>
> The [NuScale US460] reactor design has no Type A variables because there are no operator actions credited in any Chapter 15 anticipated operational occurrences, infrequent events, or postulated accidents, nor the station blackout or ATWS [anticipated transient without scram] analysis.

TR-101310-NP, Revision 1, references DCA TR-1116-52011, Revision 4 (the US600 DCA Regulatory Conformance and Development Report), in which Table B-1 indicates that no Type A PAM variables were identified. Both DCA Part 2, Tier 2, Section 7.1.5.1.14, and SDAA Part 2, Section 7.1.5.1.14, "Guideline 14—Manual Operator Action," begin by stating the following:

> The critical safety functions are accomplishing or maintaining containment integrity, fuel assembly heat removal, and reactivity control; however, there are no Type A accident monitoring variables. Type A variables provide information essential for the direct accomplishment of critical safety functions that require manual action.

In discussing the applicable safety analyses, SDAA Part 4, Bases Subsection B 3.8.1, "Nuclear Instrumentation," states the following:

> The audible count rate from the refueling neutron flux channels provides prompt and definite indication of any change in reactivity. The count rate increase is proportional to subcritical multiplication and allows operators to promptly recognize any change in reactivity. Prompt recognition of unintended reactivity changes is consistent with the assumptions of the safety analysis and is necessary to ensure sufficient time is available to initiate action before SHUTDOWN MARGIN is lost (Ref. 2). The refueling neutron flux channels satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

FSAR Section 9.2.5.2.2, "System Operation," describes how a loss of SDM in Mode 5 is also prevented, as follows:

> Before refueling operations commence, personnel verify that the boron concentration in the UHS is at or above the minimum required to prevent core criticality during refueling operations. Personnel monitor the boron concentration in the UHS to verify that it remains above this minimum when the reactor vessel is open to the pool.

Since PAM instrumentation does not address postulated events during refueling operations, proposed LCO 3.8.1, "Nuclear Instrumentation," is adequate to ensure sufficient time is available for control room operators to initiate action to terminate a reactivity transient in Mode 5 before SDM is lost.

DCA Part 2, Tier 2, Section 7.1, and SDAA Part 2, Section 7.1, identify Type B and Type C PAM variables; however, these applications do not address the option of not including an LCO for equivalent (non-Type A, but Type B and C Category 1) variables described in the 1988 Split Report. That report states the following, in part:

> During the NRC Staff's review, several issues were raised concerning the proper interpretation or application of the criteria in the Commission's Interim Policy Statement. The NRC Staff has considered these issues and concluded the following:
>
> …
>
> (5) Post-Accident Monitoring Instrumentation that satisfies the definition of Type A variables in Regulatory Guide 1.97, "Instrumentation for Water-cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," meets Criterion 3 and should be retained in Technical Specifications. Type A variables provide primary information (i.e., information that is essential for the direct accomplishment of the specified manual actions (including long-term recovery actions) for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for DBAs or transients). Type A variables do not include those variables associated with contingency actions that may also be identified in written procedures to compensate for failures of primary equipment. Because only Type A variables meet Criterion 3, the STS should contain a narrative statement that indicates that individual plant Technical Specifications should contain a list of Post-Accident Instrumentation that includes Type A variables. Other Post-Accident Instrumentation (i.e., non-Type A Category I) is discussed on page 6.
>
> …
>
> The staff reviewed the methodology and results provided by each Owners Group to verify that none of the requirements proposed for relocation contains constraints of prime importance in limiting the likelihood or severity of accident sequences that are commonly found to dominate risk. For the purpose of this application of the guidance in the Commission [Interim] Policy Statement, the staff agrees with the Owners Groups' conclusions except in two areas. First, the staff finds that the Remote Shutdown [Station (RSS)] Instrumentation meets the Policy Statement criteria for inclusion in Technical Specifications based on risk; and second, the staff is unable to confirm the Owners Groups' conclusion that Category 1 Post-Accident Monitoring Instrumentation is not of prime importance in limiting risk. Recent PRAs have shown the risk significance of operator recovery actions which would require a knowledge of Category 1 variables. Furthermore, recent severe accident studies have shown significant potential for risk reduction from accident management. The Owners Groups should develop further risk-based justification in support of relocating any or all Category 1 variables from the Standard Technical Specifications.

By a letter dated October 18, 2017 (ML17291A482), NuScale, for the US600 DCA, addressed the above guidance from the 1988 Split Report and presented a risk-informed justification for omitting an LCO for Type B and C PAM variables. The essential reason NuScale concluded that control room indication of these variables is not a significant contributor to risk reduction was that no operator actions are necessary to manage the NPM automatic response to any DBE to place the NPM in a safe-shutdown condition with passive long-term decay heat removal and core cooling assured. The US600 DC SER Chapter 19 provides the NRC staff's finding that the NuScale US600 probabilistic risk assessment (PRA) is acceptable, and based on this, the staff concluded the risk-informed analysis of the Type B and C PAM variables is acceptable. For the NuScale US460 SDAA, NuScale confirmed in an audit question response (ML24326A095) that the "justification for not including an LCO for PAM instrumentation provided in response to DCA RAI 16-22 (NuScale letter RAIO-1017-56656, dated October 18, 2017) remains valid for the NuScale [US460] Generic Standard Technical Specifications (GTS)."

In both the NuScale US600 DCA and US460 SDAA, FSAR Section 7.1.1.2.4, "Safety Display and Indication System," the applicant also pointed out that the PAM variables are displayed using the safety display and indication system (SDIS), which is subject to NuScale augmented quality assurance (AQ), addressed in FSAR Table 7.0-3, "Classification of Instrumentation and Controls Systems." This table states that all components of the SDIS have an "SSC Classification" of B2 (non-safety-related, non-risk-significant); "[Quality Assurance] Program (QAP) applicability" of AQ; and "Augmented Design Requirements" of (1) IEEE Std 497-2016, as endorsed by RG 1.97. The SDIS is also listed in the table as Seismic Classification I. Note 2 of FSAR Table 7.0-3 in Part 2 of the US460 SDAA states that the QAP applicability of AQ "indicates that pertinent augmented quality assurance requirements for non-safety-related SSC[s] are applied to ensure that the function is accomplished when needed based on that functionality's regulatory requirements....

The 1988 Split Report quoted above states that "…the Remote Shutdown Instrumentation meets the Policy Statement criteria for inclusion in Technical Specifications based on risk; …" However, for the NuScale US460 design, the applicant concluded that the equivalent of remote shutdown station (RSS) monitoring instrumentation, which resides in the instrumentation and control (I&C) equipment rooms outside the main control room (MCR), is not risk significant because like the Type B and Type C PAM variable instrumentation, the NuScale US460 design's RSS-like monitoring instrumentation only includes parameter indications; it includes no operator manual controls to achieve and maintain safe shutdown.

FSAR Section 7.1.1.1, "Design Bases," discusses achieving safe shutdown of the NPM(s) in the event the MCR must be evacuated:

> Consistent with principal design criterion (PDC) 19, the I&C systems are designed to ensure the ability to control each NPM during normal and accident conditions. The main control room (MCR) is designed with the ability to place the reactors in safe shutdown in case of a fire requiring an MCR evacuation and for safe shutdown to be maintained without operator action thereafter. Before evacuating the MCR, operators trip the reactors, initiate decay heat removal, and initiate containment isolation. These actions result in passive cooling that achieves safe shutdown of the reactors. Operators can also achieve safe shutdown of the reactors from outside the MCR in the I&C equipment rooms within the Reactor Building (RXB). Following shutdown and initiation of passive cooling from either the MCR or the I&C equipment rooms, the design does not rely on operator action, instrumentation, or controls outside the MCR to maintain the safe shutdown condition. There are no remote displays, alarms, or controls necessary to monitor or maintain the modules in a safe shutdown condition.

FSAR Section 7.1.1.2.3, "Alternate Operator Workstation Controls and Monitoring," provides additional discussion about MCR evacuation:

> If the MCR is evacuated, the alternate operator work stations located at various locations provide confirmation for the operators to monitor the NPMs in a safe shutdown condition with DHRS in service for each NPM. The alternate operator

> workstations provide [Diversity and Defense-in-Depth (D3)] capability to monitor the plant from outside the MCR and control balance of plant equipment to support asset protection and long-term plant recovery in case the MCR becomes uninhabitable. An MCR evacuation occurrence is a special event and is not postulated to occur simultaneously with a [design basis event (DBE)]; it does not cause fuel damage or result in consequential loss of function of the RCPB or primary containment barriers.
>
> …
>
> At the onset of an MCR evacuation, the operators trip the reactors, and initiate decay heat removal and containment isolation for each reactor before leaving the MCR. Following evacuation of the MCR, the ability to isolate the MPS manual switches to prevent spurious actuations is provided outside the MCR. An alarm is annunciated in the MCR when the MCR hard-wired switches are isolated using the MCR isolation switches (Figure 7.1-1j).
>
> The MPS manual isolation switches are mounted in a Seismic Category I enclosure to allow them to remain functional following an earthquake. Controls are available outside the MCR in the associated I&C equipment rooms that provide the capability to trip the reactors, to initiate DHRS, and to initiate containment isolation, which initiates passive cooling, and places and maintains the NPMs in safe shutdown. The alternate operator workstations provide nonsafety-related human-system interfaces (HSIs) and direct readings of the process variables necessary to monitor the NPMs.

FSAR Section 7.2.13.3, "Alternate Operator Workstation Controls and Monitoring," states, in part:

> … There is a set of [Module Control System] and [Plant Control System] displays located at various locations throughout the plant (alternate operator workstations) that allow operators to monitor the NPMs if evacuation of the MCR is required. Safety display and indication system displays are not provided locally as there is no manual control of safety-related equipment allowed outside the control room.

Based on the rationale provided in the above quoted FSAR discussions about the NuScale US460 design for monitoring NPM status following an MCR evacuation, the staff concludes that Alternate Operator Workstation Controls and Monitoring instrumentation is not significant to safety and meets none of the LCO selection criteria. The staff therefore finds that omitting an LCO for Alternate Operator Workstation Controls and Monitoring instrumentation is acceptable.

The augmented DC power system—common (EDAS-C) powers the PAM instrumentation and the SDIS for control room indication. The source of electrical supply to the EDAS-C battery chargers is the 480-volt AC low-voltage AC electrical distribution system, through the low-voltage AC electrical distribution system motor control centers, which can be powered by the backup diesel generators. A total of two 125-volt DC batteries and four battery chargers (one battery and two independent and redundant chargers in Division I and one battery and two independent and redundant chargers in Division II) are in the EDAS-C subsystem. Upon a loss of power to all battery chargers, both the Division I and Division II EDAS-C batteries are capable of supplying their connected plant loads for 72 hours. The primary function of the backup diesel generators is to provide backup electrical power to certain loads in the post-72-hour period following a station blackout event.

Based on the above design information, the staff concludes that there is reasonable assurance that the control room indication of PAM Type B and Type C variables will be available for 72 hours after a postulated accident, infrequent event, or AOO, and that the NuScale US460 design has no PAM Type A variables. The staff therefore finds that omitting an LCO for PAM Type B and Type C variable instrumentation is acceptable.

16.4.1.7 Support System with Operability Requirement Implied by Surveillance Requirement

The Class 1E isolation devices serve to isolate the Class 1E MPS, RTS Logic and Actuation, and ESFAS Logic and Actuation electrical circuits from non-Class 1E electrical power circuits. The Channel Calibration surveillances of SR 3.3.1.5, SR 3.3.2.3, and SR 3.3.3.4, respectively, verify the operability of these isolation devices each refueling cycle. Meeting these SRs is necessary to meet LCO 3.3.1 for MPS instrumentation Functions 1 through 27, LCO 3.3.2 for the RTS Logic and Actuation Function, and LCO 3.3.3 for ESFAS Logic and Actuation Functions 1 through 9.

Similarly, meeting SR 3.3.2.4 ("Verify each RTB actuates to the open position on an actual or simulated actuation signal") is necessary for meeting LCO 3.3.2, and meeting SR 3.3.3.5 ("Verify each pressurizer heater breaker actuates to the open position on an actual or simulated actuation signal") is necessary for meeting LCO 3.3.3, Function 6, PHT Logic and Actuation.

The staff finds that implicitly requiring these components to be operable by specifying Surveillances for them in the LCO subsections of the systems they support is acceptable because it ensures the operability of these components when the supported systems are required to be operable. In addition, this approach is consistent with the W-AP1000-STS implicit support system operability requirements in Surveillances of W-AP1000-STS LCO 3.3.15 for pressurizer heater circuit breakers, reactor coolant pump breakers, chemical and volume control system (CVS) letdown isolation valves, feedwater pump breakers, and auxiliary spray and purification line isolation valves; and LCO 3.3.16 for reactor coolant pump breakers, CVS letdown isolation valves, and spent fuel pool cooling system CIVs. None of these components have explicit LCO operability requirements, but each supports operability of a specified ESF actuation function in the AP1000 design.

Conclusion for Application of LCO Selection Criteria

Based on its review of Revision 1 of TR-101310-NP, "US460 Standard Design Approval Technical Specifications Development;" SDAA Part 2 Chapters 4, 5, 6, 7, 8, 9, 11, 12, 15, 16, and 19; and SDAA Part 4, the staff finds that the NuScale US460 GTS include all of the LCOs required by the LCO selection criteria. Therefore, the staff concludes that the GTS satisfy 10 CFR 50.36(c)(2)(ii).

16.4.2 Use and Application (GTS Chapter 1), Definitions (Section 1.1), Logical Connectors (Section 1.2), Completion Times (Section 1.3), and Frequency (Section 1.4)

The GTS and Bases follow the STS in presenting defined terms in capitalized type. This SER section follows this convention in discussions of defined terms and their definitions.

16.4.2.1 Included W-STS or W-AP1000-STS Definitions with No Changes

The GTS include the following W-STS or W-AP1000-STS definitions without change:

ACTIONS

The proposed definition matches the W-STS definition and the W-AP1000-STS definition and is therefore acceptable.

DOSE EQUIVALENT I-131

The proposed definition matches the W-AP1000-STS definition and is therefore acceptable.

DOSE EQUIVALENT Xe-133

The proposed definition matches the W-AP1000-STS definition and is therefore acceptable.

INSERVICE TESTING PROGRAM

This defined term and definition ("The INSERVICE TESTING PROGRAM is the licensee program that fulfills the requirements of 10 CFR 50.55a(f)") match the change made to W-STS, Revision 4, by approved STS change traveler TSTF-545-A, Revision 3, which was incorporated in W-STS Revision 5. Therefore, this defined term and definition are acceptable.

PHYSICS TESTS

The proposed definition matches the W-AP1000-STS definition and is therefore acceptable.

RATED THERMAL POWER (RTP)

The proposed definition matches the W-AP1000-STS definition and is therefore acceptable.

THERMAL POWER

The proposed definition matches the W-AP1000-STS definition and is therefore acceptable.

16.4.2.2 Included W-STS or W-AP1000-STS Definitions with Proposed Changes

ACTUATION LOGIC TEST

SER Section 16.4.8.3 gives the staff's evaluation of the specified SRs for the ACTUATION LOGIC TEST and associated Bases. The definition, as initially proposed in the DCA for the NuScale US600, departed from the definition of the ACTUATION LOGIC TEST in the W-AP1000-STS. Following several interactions with NuScale for the US600 DCA, the applicant revised the definition of ACTUATION LOGIC TEST and the following Bases subsections to consistently use the terms "self-test" and "self-testing": the Background section of Bases Subsection B 3.3.1, the Applicable Safety Analyses section of Bases Subsection B 3.3.1, and the Surveillance Requirements section of Bases Subsections B 3.3.2 and B 3.3.3. The descriptions of the MPS self-testing features in the NuScale US600 DCA Part 2 and DCA Part 4 GTS and Bases are clear, as is the scope of the part of the ACTUATION LOGIC TEST that uses the self-testing features. The NRC staff confirmed that the US460 SDAA GTS and Bases are consistent with the DCA GTS and Bases regarding the ACTUATION LOGIC TEST. Therefore, these changes are acceptable for the US460 design.

The next-to-last sentence in the definition in the US460 GTS states, "The ACTUATION LOGIC TEST shall be conducted such that it provides component overlap with the actuated device." This is acceptable to the NRC staff since the W-AP1000-STS definition includes this sentence.

Regarding the part of the definition that states, “The ACTUATION LOGIC TEST may be performed by means of any series of sequential, overlapping, or total steps,” the staff noted that this part of the sentence is included in the W-AP1000-STS definitions of CHANNEL CALIBRATION, CHANNEL OPERATIONAL TEST, and ACTUATION LOGIC TEST. Based on the NuScale US460 MPS design, the staff concludes that this part of the sentence is also appropriate and acceptable for the US460 ACTUATION LOGIC TEST definition.

The second part of the last sentence in the definition states, “…and each step must be performed within the Frequency in the Surveillance Frequency Control Program for the devices included in the step.” As described below in the discussion of the CHANNEL OPERATIONAL TEST definition, this phrase is acceptable based on the approved STS change traveler TSTF-563, Revision 0, “Revise Instrument Testing Definitions to Incorporate the Surveillance Frequency Control Program,” issued May 2017, on adoption of an SFCP; therefore, it is acceptable for inclusion in the NuScale US460 ACTUATION LOGIC TEST definition (as well as for the definitions of CHANNEL OPERATIONAL TEST and CHANNEL CALIBRATION).

By a letter dated September 26, 2017 (ML17269A210), NuScale, as part of the US600 DCA, explained why the ACTUATION LOGIC TEST definition includes the new phrase “to test digital computer hardware” but does not include the phrase “to test digital computer software.” Specifically, NuScale stated the following:

> The definition specifies testing of digital hardware only because there is no operating software in the installed system which performs a safety related function. A software development process is used to develop the logic which is implemented in the digital hardware [(field programmable gate arrays)]. The requirements for software development quality assurance are described in [DCA Part 2,] Tier 2, Section 7.2.1.

The staff finds this explanation is acceptable and is applicable to the US460 SDAA because field programmable gate arrays are digital hardware implementing the logic of the scheduling and bypass modules, SVMs, and equipment interface modules (EIMs) without use of software, as described in US460 SDAA, Part 2, Section 7.2.15, “Capability for Test and Calibration,” and TR-1015-18653-P-A, “Design of the Highly Integrated Protection System Platform Topical Report,” Revision 2, issued September 2017 (ML17256A892), which also applies to the US460 SDA.

Based on the above discussions, the staff concludes that the ACTUATION LOGIC TEST definition is acceptable.

CHANNEL CALIBRATION

The proposed definition of CHANNEL CALIBRATION matches the W-STS definition and is consistent with approved STS change traveler TSTF-563, Revision 0. Therefore, the staff concludes that the CHANNEL CALIBRATION definition is acceptable.

CHANNEL CHECK

The applicant proposed changes to the W-STS definition of CHANNEL CHECK, as indicated in the following markup of the STS definition:

CHANNEL CHECK	A CHANNEL CHECK shall be the ~~qualitative assessment, by observation, of~~ verification through

> the absence of alarms from the automatic analog and binary process signal monitoring features used to monitor channel behavior during operation. Deviation beyond the established acceptance criteria is alarmed to allow appropriate action to be taken. This determination shall include, where possible, comparison of ~~the~~ channel indication and status to other indications or status derived from the independent ~~instrument~~ channels measuring the same parameter. This determination can be made using computer software or be performed manually.

The changes are consistent with the design features of the MPS and how the applicant intends this Surveillance to be performed; the CHANNEL CHECK is the principal means of monitoring channel performance and status between CHANNEL CALIBRATIONS. Therefore, the staff concludes that this definition is acceptable. SER Chapter 7 describes the capabilities of the digital platform and the self-testing features of the MPS.

CHANNEL OPERATIONAL TEST (COT)

The applicant proposed specifying a CHANNEL OPERATIONAL TEST only for LCO-required instrumentation functions implemented by the module control system. These are the RCS leakage detection instrumentation of the CES condensate monitor (two channels) and CES gaseous radioactivity monitor (one channel), which are required to be OPERABLE by LCO 3.4.7 in MODES 1 and 2, and in MODE 3 with RCS hot temperature at or greater than 93.3°C (200°F). The proposed CHANNEL OPERATIONAL TEST definition is consistent with the CHANNEL OPERATIONAL TEST definition in Revision 5 of the W-STS which includes a change consistent with approved STS change traveler TSTF-563, Revision 0. This traveler adds a phrase to the last sentence of the CHANNEL OPERATIONAL TEST definition (as well as the definitions of ACTUATION LOGIC TEST and CHANNEL CALIBRATION). Since the proposed CHANNEL OPERATIONAL TEST definition includes this change by approved traveler TSTF-563, the staff concludes that the CHANNEL OPERATIONAL TEST definition is acceptable.

CORE OPERATING LIMITS REPORT (COLR)

The proposed definition is consistent with the W-STS definition of the COLR, except that instead of the phrase in the last sentence "Plant operation within these limits...," the GTS uses, "Module operation within these parameter limits...." Because "module operation" is equivalent to the intended meaning of "plant operation" and both include operation of the reactor core, the staff concludes that using "module" is only an administrative difference and is therefore acceptable. In addition, instead of the phrase "cycle" or "reload cycle," the GTS uses, "fuel cycle." Because "fuel cycle" is equivalent to the intended meaning of "cycle" or "reload cycle" the staff concludes that using "fuel cycle" is only an administrative difference and is therefore acceptable.

LEAKAGE

The applicant's proposed definition of "Pressure Boundary LEAKAGE" matches the definition in the W-STS and is, therefore, acceptable. The applicant proposed to depart from the W-STS definitions of "identified LEAKAGE" and "unidentified LEAKAGE" by omitting references to leakage "such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank." This departure is appropriate because the leakage sources and associated leakage collection systems do not apply to the NPM design. Accordingly, this departure omits paragraph a.1 and revises paragraph a.2 of the W-STS definition of "identified LEAKAGE," as indicated in the markup provided below. The W-STS definition's paragraph a.2 reference to LEAKAGE "into the containment atmosphere" is also omitted from the GTS "identified LEAKAGE" definition's paragraph a.1. This is appropriate because only RCS LEAKAGE into containment, which is not pressure boundary LEAKAGE, can be collected and measured by RCS leakage detection instrumentation, such as the CES condensate monitor channels. Other than these differences, the applicant's definitions of "identified LEAKAGE" and "unidentified LEAKAGE" match the W-STS definitions of these terms. Thus, the staff concludes that the applicant's definitions are acceptable.

LEAKAGE — LEAKAGE shall be:

a. Identified LEAKAGE

~~1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;~~

~~2~~1. LEAKAGE ~~into the containment atmosphere~~ from sources that are both specifically located and known to not interfere with the operation of leakage detection systems; or

~~3~~2. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE ~~(except RCP seal water injection or leakoff)~~ that is not identified LEAKAGE; and

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a fault in an RCS component body, pipe wall, or vessel wall. LEAKAGE past seals, packing, and gaskets is not pressure boundary LEAKAGE.

OPERABLE – OPERABILITY

The NuScale US460 GTS OPERABILITY definition, like the US600 definition, is based on the W-STS definition with differences to account for the NuScale US460 design. The NuScale US460 design, like the US600 design, does not use the word "emergency" to describe the onsite electrical power system (sources and distribution) and does not use the phrase "seal water,” since an NPM has no reactor coolant pumps and therefore has no need for reactor coolant pump seal water. In addition to the word “channel,” NuScale uses the equivalent term “separation group” to describe a redundant MPS instrumentation loop. At the suggestion of the staff during the US600 DCA review, the applicant revised the definition to more accurately reflect the NuScale US600 design; this change is applicable to the US460 design and is retained in the US460 definition. Therefore, the staff finds the OPERABILITY definition, as quoted below, acceptable.

OPERABLE – OPERABILITY	A system, subsystem, separation group, channel, division, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling water, lubrication, and other auxiliary equipment that are required for the system, subsystem, separation group, channel, division, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PRESSURE AND TEMPERATURE LIMITS REPORT

The proposed PTLR definition in both the US600 and US460 designs matches the W-STS definition except that it omits the W-STS definition’s phrase, “and the low temperature overpressure protection arming temperature,” which is also not included in the W-AP1000-STS PTLR definition. The AP1000 design uses the relief valves in the normal residual heat removal system suction line for LTOP and has no valve operator to “arm” at a particular RCS temperature; as a result, the AP1000 design certification applicant concluded that this phrase is not applicable. However, the NuScale LTOP functionality of the two RVVs is automatically enabled by the wide range RCS cold temperature interlock T-1 (two of four channels less than or equal to LTOP enabling temperature specified in the PTLR, approximately 143.3°C (290°F)). The T-1 interlock LTOP enabling temperature appears analogous to an “LTOP arming temperature” as used in the W-STS, which is based on a typical LTOP system design, such as the design implemented at Vogtle Electric Generating Station, Units 1 and 2. Therefore, in the NuScale US600 DCA review, the staff suggested that NuScale consider adding an equivalent phrase, such as “and the low temperature overpressure protection enable temperature,” to the NuScale US600 GTS PTLR definition.

By a letter dated December 12, 2018 (ML18347A619), NuScale, as part of the US600 DC, declined to incorporate the suggested changes into the PTLR definition, stating that “the LTOP arming temperature is established and maintained as specified in [generic] TS 5.5.10, and described in the Bases for LCO 3.3.1, ‘Module Protection System.’” Taking NuScale’s response into account, the staff finds that its suggested changes, though intended to promote consistency with the PTLR definition in other STS, are not necessary to ensure that the NuScale T-1 interlock LTOP enabling temperature will be correctly set and maintained to support OPERABILITY of the RVV LTOP Function, in accordance with SR 3.3.1.4, CHANNEL CALIBRATION. The US460 SDAA PTLR definition matches the US600 DCA PTLR definition because the US460 LTOP design is sufficiently similar to the US600 LTOP design. Therefore, for the reasons discussed above, the staff finds the US460 SDAA PTLR definition acceptable.

SHUTDOWN MARGIN (SDM)

In the NuScale US600 DCA, the applicant's originally proposed definition of SDM departed from the W-STS definition as indicated by the following markup of the W-STS definition:

SHUTDOWN MARGIN (SDM) — SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. Moderator temperature is 420°F [216 °C]; and

b. All ~~rod cluster control assemblies (RCCAs)~~ CRAs are fully inserted except for the single ~~RCCA~~ assembly of highest reactivity worth, which is assumed to be fully withdrawn. However, with all ~~RCCAs~~ CRAs verified fully inserted by two independent means, it is not necessary to account for a stuck ~~RCCA~~ CRA in the SDM calculation. With any ~~RCCA~~ CRA~~(s)~~ not capable of being fully inserted, the reactivity worth of ~~the RCCA~~ the affected CRA must be accounted for in the determination of SDM~~, and~~

~~b.~~ ~~In MODES 1 and 2, the fuel and moderator temperatures are changed to the [nominal zero power design level]~~.

The change in the order of parts a and b, and the use of CRA instead of rod cluster control assembly (RCCA), are editorial administrative changes to reflect NuScale nomenclature and the applicant's preferred presentation. Since the acronym "CRA" is previously defined in the definition of "MODE," not defining it upon its first use in this definition is acceptable. However, the staff suggested defining the acronym again for clarity. Also, subsequent use of the word "assembly" and "assemblies" should have been changed to "CRA" and "CRAs" to conform to the improved TS writer's guide convention concerning acronyms. Also, the W-STS definition does not appear to consider more than one RCCA to be incapable of being fully inserted; however, the W-AP1000-STS SDM definition does consider more than one uninsertable RCCA. Revision 1 of the US600 DCA contained no justification for why NuScale needed to consider more than one CRA that cannot be fully inserted. Finally, the US600 DCA did not justify using the minimum temperature for criticality, 174°C (420°F), in place of the statement, "In MODE 1, the fuel and moderator temperatures are changed to the [nominal zero power design level]." (Note that the NuScale US460 MODE 1 corresponds to W-STS MODES 1 and 2; and the NuScale US460 MODE 2 corresponds to W-STS MODE 3 but with RCS average temperature ≥ 173.89°C (345°F)). As part of the NuScale US600 DCA, NuScale resolved these issues by providing the following SDM definition, which NuScale also revised for the US460 SDAA, which has a different value for the minimum temperature for criticality, as shown by markup to reflect the SDAA temperature value:

SHUTDOWN MARGIN (SDM) — SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

a. Moderator temperature is ~~420~~ 345°F; and

b. All control rod assemblies (CRAs) are fully inserted except for the single CRA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CRAs verified fully inserted by two independent means, it is not necessary to account for a stuck CRA in the SDM calculation. With any CRA not capable of being fully inserted, the reactivity worth of the affected CRA must be accounted for in the determination of SDM.

In SER Section 16.4.8.5, the staff describes the basis for concluding that this SDM definition is acceptable.

16.4.2.3 NuScale-specific definitions

The following defined terms and definitions are unique to NuScale because of design differences from large light-water PWRs. The staff concludes that these definitions are acceptable because they accurately reflect the NuScale US600 and US460 designs.

ACTUATION RESPONSE TIME

This defined term is defined as follows:

> The time from when the Module Protection System equipment interface module output initiates an actuation signal until the actuated valves or breakers reach their final actuated position.

It is used with CHANNEL RESPONSE TIME to define the TOTAL RESPONSE TIME. The staff discusses response times in SER Section 16.4.2.4.

AXIAL OFFSET (AO)

This defined term and definition (AXIAL OFFSET equals (power in top half of core minus power in bottom half of core) divided by (power in top half plus power in bottom half)) is similar to the W-STS defined term and definition of AXIAL FLUX DIFFERENCE, except that AXIAL FLUX DIFFERENCE is based on core power derived from excore power range neutron detectors. The AXIAL OFFSET is based on core power derived from the neutron detectors of the incore instrumentation system.

CHANNEL RESPONSE TIME

This defined term is defined as follows:

> The time from when the process variable exceeds its setpoint until the output from the channel analog logic reaches the input of the digital portion of the Module Protection System digital logic.

It is used with ACTUATION RESPONSE TIME to define the TOTAL RESPONSE TIME. The staff discusses response times in SER Section 16.4.2.4.

MODE – MODES

The operational MODE definition in the NuScale US460 GTS, as provided below, differs from the W-AP1000-STS definition:

MODE	A MODE shall correspond to any one inclusive combination of reactivity condition, indicated reactor coolant temperature, PASSIVE COOLING status, control rod assembly (CRA) withdrawal capability, Chemical and Volume Control System (CVCS) and Containment Flood and Drain System (CFDS) configuration, reactor vent valve electrical isolation, and reactor vessel flange bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

The staff verified that this definition and associated Table 1.1-1, "MODES," are appropriate for delineating practical ranges of the NPM operational states because they support how the NPM is designed to be operated and facilitate specifying the physical conditions of the NPM in which each LCO must be met. Therefore, the staff finds the defined term MODE and its definition acceptable. SER Section 16.4.6, "Applicability Statements," gives the staff's evaluation of Table 1.1-1.

PASSIVELY COOLED – PASSIVE COOLING

This definition applies during MODES 2 and 3 when the secondary heat sink is not available for removal of core decay heat from the reactor coolant. Although there are three stated methods for achieving PASSIVE COOLING, they all transfer core decay heat to the reactor building pool, the UHS. This definition states:

PASSIVELY COOLED – PASSIVE COOLING	A module is in PASSIVE COOLING or is being PASSIVELY COOLED when: a. One or more reactor vent valves are open and one or more reactor recirculation valves are open, or b. One or more trains of DHRS are in operation, or c. Water level in the containment vessel is > 45 ft.

In this definition, "one or more" means that PASSIVE COOLING is in operation with just one reactor vent valve and one reactor recirculation valve open (passive cooling method a—defined in item "a" in the definition above) because the Chapter 15 analysis assumed heat removal capacity only requires one vent valve and one recirculation valve. The module is also being PASSIVE COOLED with just one DHRS train in operation (method b) because one train is designed to provide the Chapter 15 analysis assumed heat removal capacity. The module design includes two vent valves and two recirculation valves to provide redundancy for PASSIVE COOLING method a, and two DHRS trains to provide redundancy for PASSIVE COOLING method b. Passive decay heat removal by conduction through the reactor vessel and containment vessel walls and by convection in the reactor vessel and the water-flooded containment vessel to the reactor pool (the UHS) constitutes passive cooling method c.

TOTAL RESPONSE TIME

This defined term is defined as follows:

> TOTAL RESPONSE TIME is the sum of the CHANNEL RESPONSE TIME, the allocated MPS digital time response, and the ACTUATION RESPONSE TIME. The TOTAL RESPONSE TIME is the time interval from when the monitored parameter exceeds its actuation setpoint at the channel sensor until the actuated component is capable of performing its safety function (i.e., the valves travel to their required positions, breakers are open, etc.)

The staff discusses response times in SER Section 16.4.2.4.

16.4.2.4 Response Time

The applicant proposed the three defined terms of CHANNEL RESPONSE TIME, ACTUATION RESPONSE TIME, and TOTAL RESPONSE TIME and their definitions in place of the W-STS defined terms of REACTOR TRIP SYSTEM (RTS) RESPONSE TIME and ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME and their definitions.

The NRC staff reviewed material in FSAR Chapter 7, "Instrumentation and Controls," related to RTS and ESFAS delay (or response) times, some of which is quoted below; italics are used to highlight that the stated RTS and ESFAS delay times, in both FSAR Table 7.1-6, "Design Basis Event Actuation Delays Assumed in the Plant Safety Analysis," and Table 15.0-7, "Analytical Limits and Time Delays," include RTB and pressurizer heater breaker opening times, respectively, but that the ESFAS delay times do not include valve stroke times. FSAR Section 7.1.1.2.1, "Protection Systems," states, in part, the following (emphasis added):

> The ESFAS delays assumed in the plant safety analysis [e.g., Table 15.0-7, "Analytical Limits and Time Delays"] are a combination of sensor response time, MPS timing budget allocation, and actuation device delays. The sensor response delays are defined in Table 7.1-6. *The delay times in Table 7.1-6 associated with ESFAS signals do not include the delay times associated with the actuation device (e.g., valve stroke times) except opening the pressurizer heater breakers*.

FSAR Section 7.1.4, "Predictability and Repeatability," states, in part, the following (emphasis added):

> The MPS response time analysis demonstrates the MPS performs and completes its required safety functions in a predictable and repeatable manner. Section 7.7 of TR-1015-18653-P-A describes the calculation used to determine worst-case digital time response for an MPS channel.
>
> … The RTS timing analysis is defined from the point in time when the monitoring process variable exceeds its predetermined setpoint to when the [reactor trip breakers] open. The MPS digital portion of the RTS function is accounted for in the safety analysis. *For the RTS protective function, the MPS response time is composed of the analog input delay plus the digital time response delay plus the analog output delay and includes the time for the [reactor trip breakers] to open…*

> *For the ESFAS protective functions, the actuation delays in Table 7.1-6 are assumed in the plant safety analysis and are defined as the time from when the monitored process variable exceeds the predetermined setpoint until the [equipment interface module] output is de-energized.* The MPS portion of the ESFAS functions is accounted for in the safety analysis. *This time allocation budget is comprised of the analog input delay plus the digital time response delay plus the analog output delay and is defined from the sensor input to the [safety function module] input terminals to the [equipment interface module] output command to the final actuation device*. *For the pressurizer heater trip function, this time requirement includes the time for the pressurizer heater breakers to open*.

The staff notes that the ACTUATION RESPONSE TIME includes the time after the equipment interface module output is deenergized until the valve or breaker is in the actuated safety position. With the exception of the opening time of the pressurizer heater breakers, the delay times stated in TR-1015-18653-P-A, Revision 2, omit the valve stroke time and breaker opening time. The ESF valve stroke times are not included in FSAR Table 15.0-7, because these assumed stroke time values are used by the INSERVICE TESTING PROGRAM as acceptance criteria for the valve exercise inservice test as listed in FSAR Table 3.9-1, “Summary of Design Transients.” FSAR Section 3.9.6.3.2, “Valve Testing,” under the heading “(3) Power-Operated Valve Tests,” states, in part, the following:

> The IST requirement for measuring stroke time for valves can be completed in conjunction with a valve exercise test. The exercise test identified in Table 3.9-18 [“Valve Inservice Test Requirements per ASME OM Code”] includes the stroke time test.
>
> All POVs fail to their safe position and are subject to a valve exercise test and a fail-safe test.
>
> The valve exercise and fail-safe tests are intended to verify that the valve repositions to its safe position on loss of actuator power.

Accordingly, each ACTUATION RESPONSE TIME surveillance for an ESF valve has a Surveillance Frequency of “In accordance with the INSERVICE TESTING PROGRAM.”

Meeting each ACTUATION RESPONSE TIME surveillance for an ESF valve requires verifying the valve stroke time from its normal position to its safety position is within the safety analysis assumed stroke time value. All NuScale US460 power-operated valves receive an actuation command signal from the equipment interface module output by removing electrical power from the actuation device for the valve.

For example, each ESF valve that uses the stored energy of a nitrogen-gas-filled accumulator (which is pressurized by opening the valve using the non-safety-related hydraulic actuator) automatically repositions when the command signal deenergizes the division’s solenoid hydraulic vent valve, which opens to release the hydraulic lock keeping the valve in its normal open or closed position, and allowing the accumulator pressure to quickly reposition the valve to its safety position. FSAR Table 3.9-17, “Active Valve List,” lists the ESF valves with this actuator design. The accumulator pressure of each of these valves is monitored with control room indication and low-pressure alarms; the associated system LCOs also have SRs to periodically verify proper accumulator pressure (SR 3.4.6.1 for CVCSI valves, SR 3.5.2.1 for DHRS actuation valves, SR 3.6.2.1 for CIVs, SR 3.7.1.1 for MSIVs, and SR 3.7.2.1 for FWIVs).

The valve stroke time for the two RVVs is the time at which the RPV riser water level is below the low or low low setpoint for the ECCS actuation on low or low low riser water level instrumentation functions until each valve is open. The valve stroke time for the two RRVs is the time at which the RPV riser level is below the low or low low setpoint for the ECCS actuation on low or low low riser water level instrumentation functions concurrently with the release of the RRV's inadvertent actuation block until each valve is open. The RRV inadvertent actuation block is released when the pressure difference between the RPV and the CNV is below the inadvertent actuation block release setpoint. The ECCS actuation on low or low low RPV riser water level instrumentation function opens the ECCS valves by deenergizing the output of the equipment interface module to the two solenoid trip valve actuators for each ECCS valve. Depending upon the event being analyzed, the time at which these NPM conditions are satisfied can vary; if the reactor vessel to CNV differential pressure remains above the inadvertent actuation block release setpoint, the TOTAL RESPONSE TIME for ECCS RRV actuation would not correspond to the time interval after the RPV riser level drops below the low or low low water level setpoint until the ECCS RRV valve is open. In such cases, the analyses in the NuScale US460 SDAA Part 2, Chapter 15, show that any delay in automatically opening ECCS RRVs caused by the inadvertent actuation block not having cleared has no adverse effect on the analysis results.

The RVVs also automatically open on LTOP actuation on high wide range RCS pressure, provided at least two of four wide range RCS cold temperature channels indicate below the T-1 interlock (less than 143.3°C (290°F)).

The surveillance statement for SR 3.3.1.3, which verifies the CHANNEL RESPONSE TIME is within limits for each MPS instrumentation function, which includes the monitored process variable sensor delay time, also requires verifying the TOTAL RESPONSE TIME. Likewise, the surveillance statement for each SR, which verifies the ACTUATION RESPONSE TIME is within limits, also requires verifying the TOTAL RESPONSE TIME.

In light of the foregoing discussion, the staff concludes that the total time assumed for the RTS or ESFAS response in the safety analyses is determined to be within the analysis assumptions through a combination of NRC-approved allocations and measurements of valve stroke times and breaker opening times.

Based on the above-cited information provided in the NuScale US460 SDAA Parts 2and 4; the above evaluation; and the proposed defined terms of CHANNEL RESPONSE TIME, ACTUATION RESPONSE TIME, and TOTAL RESPONSE TIME, as defined in GTS Section 1.1, and as used in the response time verification SRs, the staff concludes that the response time of each RTS and ESFAS Function, from process sensor input to completion of device actuation, will be adequately verified to be within the safety analysis assumptions. Therefore, the staff finds the GTS defined terms and definitions for MPS automatic actuation response times are acceptable.

16.4.2.5 Omitted W-STS Definitions

The following W-STS defined terms and definitions are not included in GTS because of NuScale US460 design differences from large light-water PWRs, differences that include the use of digital I&C platforms. The staff concludes that omission of these definitions is acceptable because they are not needed to clearly state any of the requirements in the GTS and for the reasons discussed for each defined term.

AXIAL FLUX DIFFERENCE

Since the AXIAL FLUX DIFFERENCE is based on core power derived from excore power range neutron detectors, it is not applicable to the NuScale US460 design, which uses a similar defined term and definition, AXIAL OFFSET. Therefore, omission of AXIAL FLUX DIFFERENCE is acceptable.

$\bar{E}$—AVERAGE DISINTEGRATION ENERGY

STS change traveler TSTF-490-A, Revision 0, “Deletion of E Bar Definition and Revision to RCS Specific Activity Tech Spec,” September 13, 2005 (ML052630462) removed the defined term and definition of $\bar{E}$—AVERAGE DISINTEGRATION ENERGY in W-STS, Revision 4, Section 1.1. Consistent with this traveler and Section 1.1 of W-STS Revisions 4 and 5, the US460 SDAA GTS Section 1.1 omits the defined term and definition of $\bar{E}$—AVERAGE DISINTEGRATION ENERGY, replaces the W-STS definition of the defined term DOSE EQUIVALENT I-131 with the W-AP1000-STS Revision 1 definition of this defined term, and includes the W-AP1000-STS, Revision 1, defined term and definition of DOSE EQUIVALENT XE-133. GTS Subsection 3.4.8, “RCS Specific Activity,” and associated Bases, use these two defined terms.

The proposed RCS specific activity limits in GTS Subsection 3.4.8 are consistent with the fuel defect level of 0.066 percent as assumed by the NuScale US460 design-basis source term in the analyses of DBA radiological consequences, which is referenced in FSAR Section 15.0.3 (Reference 15.0-6, NuScale Power, LLC, “Accident Source Term Methodology,” TR-0915-17565-P-A, Revision 4).

SER Section 12.2.4 gives the staff’s finding that the limits in GTS Subsection 3.4.8 and Bases Subsection B 3.4.8 are acceptable.

MASTER RELAY TEST

This W-STS defined term is not applicable to the NuScale US460 MPS because the NuScale US460 MPS has no master relay.

QUADRANT POWER TILT RATIO

This W-STS defined term is not used to define limits on asymmetry of the reactor core radial power distribution in the NuScale US600 and US460 designs. Table B-1 of the Regulatory Conformance and Development Report, Revision 4, issued May 2020 (ML20141L804), during the US600 DCA review, states that W-STS LCO 3.2.4, “QUADRANT POWER TILT RATIO,” is “not applicable to NuScale analysis methodology and design.” The staff agrees with this statement. This DCA report is incorporated by reference in TR-101310-NP, Revision 1, “US460 Standard Design Approval Technical Specifications Development.” Therefore, the staff finds that omitting the QUADRANT POWER TILT RATIO in the NuScale US460 SDAA GTS Section 1.1 is acceptable.

STAGGERED TEST BASIS

The GTS do not use this defined term to modify any Surveillance Frequency because staggered testing of redundant subsystems, trains, or instrumentation channels provides no safety benefit in the NuScale US460 design. The staff therefore finds that omitting STAGGERED TEST BASIS in GTS Section 1.1 is acceptable.

TRIP ACTUATING DEVICE OPERATIONAL TEST

The scope of testing applicable to the TRIP ACTUATING DEVICE OPERATIONAL TEST definition exceeds the testing needed by the NuScale US460 design for which the defined term could be used in the surveillance statement of the one SR for the MPS manual actuation functions. The GTS do specify an equivalent test in SR 3.3.4.1, which only applies to the MPS manual actuation functions; this surveillance requires performing an “actuation device operational test” for the nine manual functions for initiating RTS, ECCS, DHRS, CIS, DWSI, CVCSI, PHT, SSI, and LTOP. Also, if the GTS had included a TRIP ACTUATING DEVICE OPERATIONAL TEST with a pared down definition appropriate for this SR, this SR would have been the sole use of this defined term. The Bases for SR 3.3.4.1 adequately describe the specified actuating device operational test. The staff therefore finds that omitting TRIP ACTUATING DEVICE OPERATIONAL TEST in GTS Section 1.1 is acceptable.

Conclusion for GTS Section 1.1

The staff evaluated the defined terms and definitions in GTS Section 1.1 and determined they are appropriate and consistent with the STS (with justified exceptions) and with the NuScale US460 design and are correctly used in stating the requirements in the remainder of the GTS and in the Bases for the requirements in GTS Chapters 2 and 3. Based on its evaluation and review of FSAR Section 16, and SDAA Part 4, GTS Section 1.1, the staff finds that GTS Section 1.1 is acceptable.

16.4.2.6 Logical Connectors, Completion Times, and Frequency

W-AP1000-STS Sections 1.2, 1.3, and 1.4 provide examples depicting the use and application rules, which are specified in these sections, for logical connectors, required action completion times, and SR frequencies (test intervals), respectively. These examples reflect the AP1000 passive design features. Each example includes a discussion of the particular provision being illustrated. Because the NuScale US460 design also relies on passive design features, the examples in the GTS for these sections are similar to and consistent with the examples in the W-AP1000-STS. The staff therefore concludes that GTS Sections 1.2, 1.3, and 1.4 are acceptable.

Conclusion for GTS Chapter 1

Based on its review and the above evaluation, the staff concludes that GTS Chapter 1 is acceptable.

16.4.3 Safety Limits (GTS Chapter 2)

GTS SL Section 2.1.1.1 states the following:

> 2.1.1 <u>Reactor Core SLs</u>
>
> > 2.1.1.1 In MODE 1 the critical heat flux ratio shall be maintained at or above the following correlation safety limits:

Correlation	Safety Limit
NSP4	[1.21]
NSPN-1	[1.15]

FSAR Section 4.4.1, "Design Bases," provides the following design basis for the CHF:

> Consistent with General Design Criterion (GDC) 10, the thermal-hydraulic design of the reactor core includes sufficient margin to critical heat flux (CHF) to ensure adequate heat transfer with a 95 percent probability at a 95 percent confidence (95/95) level so that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs) and conditions.
>
> Consistent with GDC 12, the thermal-hydraulic design of the core includes design and operational limits that preclude power instability such that fuel design limits are not exceeded.

FSAR Section 4.4.1.2, "Fuel Temperature," states that "the fuel melting temperature is not exceeded. Analyses are performed at rated power and during transients up to the design limit burnup."

FSAR Section 4.4.2.1, "Critical Heat Flux," describes the CHF methodology used to ensure thermal margins for the fuel are maintained. The CHFR is a ratio between CHF and local heat flux. The CHFR correlation used is NSP4. The NRC staff reviewed the use of the NSP4 correlation and the associated safety limit in TR-107522-P-A, Revision 1, "Applicability Range Extension of NSP4 CHF Correlation," issued April 28, 2023 (ML23118A377). The applicant determined and the NRC staff confirmed that the limit for this correlation is 1.21 to meet the requirements of GDC 10, "Reactor design." The NSP4 correlation is used to evaluate thermal margins for normal operation, AOOs, infrequent events, and accidents, except for those that exhibit rapid depressurization of the reactor vessel. The NSPN-1 correlation is used for events that result in rapid depressurization of the reactor vessel.

FSAR Section 4.4.2.1 also states that the limits provided by the core design and the MPS ensure that the CHF design basis is met. The MPS automatically initiates protective actions to mitigate the effects of DBEs. During normal operation and AOOs, the thermal margin criteria are not exceeded for the limiting fuel rod in the core. For infrequent events and accidents some fuel rods may exceed limits. For these events these fuel rods are assumed to fail and assumed to contribute to the radiological dose source term.

The SDAA, Part 4 GTS Bases (Subsection B 2.1.1) state that the reactor core SLs ensure that there is a 95-percent probability at a 95-percent confidence level that the hot fuel rod in the core does not experience CHF and that the hot pellet in the core does not experience centerline fuel melting. To meet these criteria, MPS functions are defined for steady state, normal operational transients and AOOs.

Similar to the W-STS, the NuScale US460 SLs for the reactor core in GTS SL 2.1.1 are applicable when the reactor is critical. For the NuScale US460 reactor this is in Mode 1, which is equivalent to W-STS Modes 1 and 2.

The NRC staff reviewed reactor core SL 2.1.1.1 and concluded it is appropriate for the NuScale US460 design based on the discussion above and because it is consistent with the W-STS reactor core SLs.

The staff reviewed reactor core SL 2.1.1.2 (“In MODE 1 the peak fuel centerline temperature shall be maintained ≤ { 4901 - (1.37E-3 × Burnup, MWD/MTU) } °F”), which is identical to the US600 peak fuel centerline temperature SL, and determined it is appropriate for the NuScale US460 design. The staff finds SL 2.1.1.2 acceptable because it is below the centerline temperature limit as specified in “Framatome Fuel and Structural Response Methodologies Applicability to NuScale,” issued October 2022 (ML22292A312 (nonproprietary), ML22292A313 (proprietary)). TR 117605-P, Revision 1, provides detailed analyses of the NuScale fuel assembly using the approved methods. The fuel centerline temperature limit provides a reasonable assurance that the specified acceptable fuel damage limit (SAFDL) is met in accordance with the regulatory requirement of general design criterion 10 (GDC 10). The SAFDL for US460 design is no fuel failure. Chapter of this SER provides detailed discussion on the centerline temperature limit and DGC 10.

The staff reviewed RCS pressure SL 2.1.2 (“In MODES 1, 2, and 3 pressurizer pressure shall be maintained ≤ 2420 psia [16.685 MPa (absolute)]”) and determined it is appropriate for the NuScale US460 design because it is 110 percent of the RCS design pressure of 2200 psia [15.168 MPa (absolute)]”) in accordance with Section III of the ASME Code and is consistent with W-STS, and therefore finds SL 2.1.2 acceptable.

The staff reviewed GTS Section 2.2, “Safety Limit Violations,” and finds it acceptable because it is consistent with the NuScale US460 operational modes defined in GTS Table 1.1-1 and the W-STS Section 2.2 action requirements for SL violations.

Conclusion for GTS Chapter 2

Based on its review and the above evaluation, the staff finds that GTS Chapter 2 meets the requirements of 10 CFR 50.36(c)(1)(A) for SLs and is therefore acceptable.

16.4.4 Limiting Condition for Operation and Surveillance Requirement Use and Applicability (GTS Chapter 3, Section 3.0)

The staff reviewed the general LCO and SR usage rules of GTS Chapter 3, Section 3.0, for consistency with W-STS Section 3.0 except for departures needed to account for unique NuScale US460 design and operational features.

16.4.4.1 Limiting Condition for Operation Use and Applicability

As discussed below, the staff found that LCO 3.0.1, LCO 3.0.2, LCO 3.0.3, LCO 3.0.4, LCO 3.0.5, LCO 3.0.6, LCO 3.0.7, and LCO 3.0.8 are acceptable because they are consistent with the W-STS (with justified exceptions) and the NuScale US460 design.

<u>LCO 3.0.1</u>

This Specification defines the logical connection between an LCO statement and the associated Applicability statement. It states that “LCOs shall be met during the MODES or other specified conditions in the Applicability, except as provided in LCO 3.0.2, [and] LCO 3.0.7[, and LCO 3.0.8].” The list of LCOs containing exceptions to LCO 3.0.1 is appropriate for the NuScale US460 GTS and includes brackets to reflect the status of bracketed LCO 3.0.8 as a COL action item. Since this Specification matches the W-STS, LCO 3.0.1 is acceptable.

LCO 3.0.2

This Specification defines the logical connection between the LCO and Applicability statements and the associated Action statements; it also specifies that LCO 3.0.5 and LCO 3.0.6 provide exceptions to LCO 3.0.2, consistent with the W-STS.

The industry TSTF submitted Revision 1 of TSTF-565, "Revise the LCO 3.0.2 and LCO 3.0.3 Bases," on March 30, 2018, for the NRC staff's review. The staff accepted the STS Bases changes proposed in the traveler revision in a letter to the TSTF dated December 31, 2018 (ML18284A377). The staff compared the proposed changes in the revised traveler with SDAA Part 4, Bases for LCO 3.0.2 and LCO 3.0.3, and finds that the GTS Bases for these LCOs are consistent with the traveler. Therefore, the Bases for LCO 3.0.2 and LCO 3.0.3 are acceptable. The NRC staff notes that the following passage from the end of the second to last paragraph of the W-STS LCO 3.0.2 Bases is omitted in the GTS LCO 3.0.2 Bases because the GTS contain no applicable SR Frequency modifiers: "Individual Specifications may specify a time limit for performing an SR when equipment is removed from service or bypassed for testing. In this case, the Completion Times of the Required Actions are applicable when this time limit expires, if the equipment remains removed from service or bypassed."

LCO 3.0.3

This Specification is consistent with W-STS Section 3.0, LCO 3.0.3, with differences stemming from NuScale US460 unit operational mode definitions. The staff noted that this Specification allowed times to place the unit in Mode 2, and in various specified conditions in Mode 3, that appeared to be inconsistent with the allowed times in most other LCO subsections with shutdown action requirements.

During the NRC staff's review of LCO 3.0.3 in the NuScale US600 DCA, in a letter dated September 14, 2017 (ML17257A450), the applicant modified the Bases for LCO 3.0.3 by adding a paragraph (subsequently modified in US600 DCA Revision 2, as shown in the markup below) describing the reasons why the shutdown sequence Completion Times of LCO 3.0.3 are appropriate. In the corresponding paragraph in SDAA Part 4, on page B 3.0-5, two additional edits, indicated by italics, are included, as shown in the following markup:

> The Completion Times are established considering the limited likelihood of a design basis event during the *37* hours allowed to ~~reach~~ enter MODE 3 and be PASSIVELY COOLED. They also provide adequate time to permit evaluation of conditions and restoration of OPERABILITY without ~~unnecessarily~~ challenging plant systems during a shutdown. Analysis shows that 37 hours from entry into *LCO* 3.0.3 is a reasonable time to ~~reach~~ enter MODE 3 and be PASSIVELY COOLED using normal plant systems and procedures.

This additional Bases explanation clarifies the rationale for the shutdown sequence Completion Times, which seem reasonable because they are consistent with the allowed time intervals to reach safe-shutdown conditions in W-AP1000-STS LCO 3.0.3, which is 37 hours to establish normal shutdown cooling (normal residual heat removal system) in Mode 5 (cold shutdown), and are equivalent to the US460 standard design GTS LCO 3.0.3 allowance of 37 hours to establish passive cooling in Mode 3. Passive cooling may be established by placing the DHRS in operation, establishing a recirculation flowpath through the CNV by opening at least one RVV and at least one RRV, or filling the CNV to a water level at or above 13.72 m (45 ft) elevation using the CFDS with borated water from the reactor pool. Since the allowed time to reach a safe shutdown condition are well within the capacity of the passive cooling design, the staff concludes that the US460 GTS LCO 3.0.3 and Bases are acceptable. This includes the US460 LCO 3.0.3 Bases passages related to changes included in TSTF-565-A, Revision 1, March 30, 2018 (ML18089A064, ML18284A377).

LCO 3.0.4

This Specification defines the conditions that must be met to allow for entry into a mode or other specified condition in the Applicability of an LCO when the LCO is not met. The GTS LCO 3.0.4 is consistent with Revision 5 of W-STS LCO 3.0.4, which is based on traveler TSTF-359-A, Revision 9, "Increase Flexibility in Mode Restraints," dated April 2003 (ML031190607). The Bases for LCO 3.0.4 omits the following passage in the W-STS Bases for LCO 3.0.4 because the GTS contain no notes prohibiting the use of LCO 3.0.4.b: "However, there is a small subset of systems and components that have been determined to be more important to risk and use of the LCO 3.0.4.b allowance is prohibited. The LCOs governing these systems and components contain Notes prohibiting the use of LCO 3.0.4.b by stating that LCO 3.0.4.b is not applicable." Because LCO 3.0.4 otherwise matches the W-STS, LCO 3.0.4 is acceptable.

LCO 3.0.5

This Specification defines an exception to LCO 3.0.2: "Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment." Because this Specification matches the W-STS, LCO 3.0.5 is acceptable.

In a letter dated March 20, 2018 (ML18079B134), the applicant stated there are eight CVCSI valves, which are specified to be operable by US600 DCA GTS LCO 3.4.6; the US460 SDAA also has these eight isolation valves, but these valves have a different enumeration, as follows (NuScale US460 SDAA valve designators and enumeration are used, SDAA FSAR Table 3.9-17: "Active Valve List"):

CVC-HOV-0324	Pressurizer Spray Outboard Containment Isolation Valve
CVC-HOV-0325	Pressurizer Spray Inboard Containment Isolation Valve
CVC-HOV-0330	CVCS Injection Outboard Containment Isolation Valve
CVC-HOV-0331	CVCS Injection Inboard Containment Isolation Valve
CVC-HOV-0334	CVCS Discharge Inboard Containment Isolation Valve
CVC-HOV-0335	CVCS Discharge Outboard Containment Isolation Valve
CVC-HOV-0401	RPV High Point Degasification Vent Inboard Containment Isolation Valve
CVC-HOV-0402	RPV High Point Degasification Vent Outboard Containment Isolation Valve

In addition to the CVCSI function required by GTS LCO 3.4.6, the containment isolation function of these valves is required to be operable by LCO 3.6.2, "Containment Isolation Valves [CIVs]."

As a part of adopting TSTF-529, Revision 4, "Clarify Use and Application Rules," dated February 2016 (ML16060A455), the example in the third paragraph of the Bases for US600 DCA GTS LCO 3.0.5 was modified to differ from the corresponding paragraph of the Bases for W-STS LCO 3.0.5 to reflect the NuScale US600 design, which lacks RCS pressure isolation valves (PIVs). PIVs are specified in W-STS LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage." During reactor operation, the PIVs are closed to protect the low-design pressure portion of an RCS connected system from the high pressure of the RCS. The US600 DCA proposed Bases discussion for LCO 3.0.5 had referred to RCPB leakage isolation, to illustrate the application of LCO 3.0.5, instead of referring to the intermittent opening of a manual valve that was used to isolate an RCS PIV, which had been declared inoperable due to leakage through the PIV greater than the limit specified by W-STS SR 3.4.14.1, and had been closed to comply with ACTIONS.

The problem with the US600 GTS LCO 3.0.5 Bases referencing RCPB leakage isolation is that no required action of the US600 LCO 3.4.5, "RCS Operational LEAKAGE," explicitly mandates isolation of RCS pressure isolation valve leakage by closing another valve. Although LCO 3.4.6 does not address RCPB leakage, it does address CIV leakage for systems connected to the RCS, such as the four CVCS flow path lines listed above. Therefore, the staff's suggestion during the US600 DCA review that the LCO 3.0.5 Bases discussion should reference the inoperability of a CVCSI valve (also a CIV) requiring isolation, possibly because of valve leakage, was incorporated in the Bases for US600 GTS LCO 3.0.5, and is also included in the Bases for US460 GTS LCO 3.0.5.

The applicant confirmed in an audit item response (ML24326A095) that the "SDAA design retains two CVCS isolation valves for each of the four flow paths lines connected to the RCS"; and that the US600 DCA LCO 3.0.5 Bases example is applicable to the US460 standard design, and therefore is retained in the US460 SDAA Bases for LCO 3.0.5. The example states: "An example of demonstrating equipment is OPERABLE with the Required Actions not met is opening a manual valve that was closed to comply with Required Actions to isolate a CVCS flowpath with an inoperable CVCS isolation valve in order to perform testing to demonstrate that the CVCS isolation valve is now OPERABLE."

LCO 3.0.6

This Specification defines the actions required to be taken when a supported system LCO is not met solely because a support system LCO is not being met. Only the support system LCO actions are required to be entered, unless a specific exception is specified. The required actions include an evaluation performed in accordance with Specification 5.5.8, "Safety Function Determination Program (SFDP)." Because this Specification matches the W-STS, LCO 3.0.6 is acceptable.

LCO 3.0.7

This Specification defines the rules for applying the allowances of Subsection 3.1.8, "PHYSICS TESTS Exceptions." Because this Specification matches the W-STS, LCO 3.0.7 is acceptable.

LCO 3.0.8

GTS Section 3.0 and GTS Bases Section B 3.0 include LCO 3.0.8 and the Bases for LCO 3.0.8, respectively, which are consistent with approved traveler TSTF-427-A, Revision 2, "Allowance for Non-Technical Specification Barrier Degradation on Supported System OPERABILITY," dated May 3, 2006 (ML061240055), and LCO 3.0.9 of the W-STS, Revision 5. The applicant included a reviewer's note before LCO 3.0.8, and before the Bases for LCO 3.0.8, to provide information regarding the conditions for adoption of LCO 3.0.8. The text of LCO 3.0.8 and its Bases and the associated reviewer's notes are placed in square brackets to designate LCO 3.0.8, its Bases, and the reviewer's notes as part of COL Item 16.1-1. The LCO reviewer's note varies from the corresponding reviewer's note in W-STS LCO 3.0.9, as indicated in the following markup of the W-STS reviewer's note, and the text of GTS LCO 3.0.8:

[------------------------------REVIEWER'S NOTE--------------------------------
A COL applicant who wants to adopt LCO 3.0.8 must perform or reference a risk assessment for the NuScale design that has been submitted to ~~and accepted by~~ the NRC, and that was prepared consistent with the bounding generic risk assessment provided in TSTF-427-A, Rev. 2, "Allowance for Non-Technical Specification Barrier Degradation on Supported System OPERABILITY.~~,~~" ~~Revision 2.~~
--]

[LCO 3.0.8~~9~~ When one or more required barriers are unable to perform their related support function(s), any supported system LCO(s) are not required to be declared not met solely for this reason for up to 30 days provided that at least one train or subsystem of the supported system is OPERABLE and supported by barriers capable of providing their related support function(s), and risk is assessed and managed. This Specification may be concurrently applied to more than one train or subsystem of a multiple train or subsystem supported system provided at least one train or subsystem of the supported system is OPERABLE and the barriers supporting each of these trains or subsystems provide their related support function(s) for different categories of initiating events.

If the required OPERABLE train or subsystem becomes inoperable while this Specification is in use, it must be restored to OPERABLE status within 24 hours or the provisions of this Specification cannot be applied to the trains or subsystems supported by the barriers that cannot perform their related support function(s).

At the end of the specified period, the required barriers must be able to perform their related support function(s) or the supported system LCO(s) shall be declared not met.]

The GTS LCO 3.0.8 is identical to W-STS LCO 3.0.9. The noted differences in the associated reviewer's note do not lessen the conditions that must be met for adopting LCO 3.0.8. By designating GTS LCO 3.0.8 as a part of a COL item, the NRC must approve the risk assessment that must be submitted as part of a COL application that references the US460 standard design approval and includes GTS LCO 3.0.8. Designating LCO 3.0.8, its Bases, and the reviewer's note as part of COL Item 16.1-1 is acceptable because it minimizes the administrative burden on the NRC staff and a COL applicant if a future license applicant elects to omit LCO 3.0.8 from the plant-specific TS. Deferring the submission or referencing of an associated risk assessment to a COL application referencing the NuScale US460 standard design approval is acceptable because it will still ensure that the inclusion of LCO 3.0.8 in the plant-specific TS is technically justified and not adverse to safe operation of the unit.

During the NuScale US600 DCA review, by a letter dated September 14, 2017 (ML17257A450), the applicant adequately justified the omission of two other reviewer's notes, which are included in the Bases for W-STS LCO 3.0.9 and TSTF-427-A, from the US600 DCA GTS Bases Section B 3.0 for LCO 3.0.8. The staff determined that the justification in that letter is applicable to the US460 SDAA. As noted above, the applicant placed square brackets around LCO 3.0.8 and its Bases, as well as around the associated reviewer's note, to indicate that they are part of COL

Item 16.1-1. Therefore, the staff concludes that GTS LCO 3.0.8 and its Bases, and the associated reviewer's note, are acceptable because they are consistent with TSTF-427-A and W-STS LCO 3.0.9 (with justified exceptions).

16.4.4.2 Surveillance Requirement Use and Applicability

As discussed below, the staff found that SR 3.0.1, SR 3.0.2, SR 3.0.3, and SR 3.0.4 are consistent with the W-STS (with justified exceptions) and the NuScale US460 design.

<u>SR 3.0.1</u>

This Specification defines the logical connection that meeting the LCO and Applicability statements requires meeting the acceptance criteria and performance intervals of the associated SRs. Because this Specification matches the W-STS, SR 3.0.1 is acceptable.

<u>SR 3.0.2</u>

This Specification provides a 25-percent extension of the specified Surveillance performance interval (Frequency) "as measured from the previous performance or as measured from the time a specified condition of the Frequency is met." It also defines specific exceptions to this allowance for Frequencies specified as "once," and for the initial performance of a Completion Time that requires periodic performance of a required action on a "once per..." basis. Because this Specification matches the W-STS, SR 3.0.2 is acceptable.

<u>SR 3.0.3</u>

This Specification defines the actions required to be taken if it is discovered that a Surveillance was not performed within its specified Frequency and provides an exception to SR 3.0.1. This Specification in NuScale US600 GTS matched SR 3.0.3 of W-STS Revision 4, as revised by NRC-approved traveler TSTF-529-A, which was incorporated in W-STS Revision 5.

In the US600 GTS SR 3.0.3 Bases NuScale partially adopted editorial changes to the W-STS SR 3.0.3 Bases that had been proposed by unapproved traveler TSTF-530, Revision 0 (ML112620602). The NuScale US600 GTS SR 3.0.3 Bases incorporated some of this traveler's proposed editorial changes, for which the NRC staff had no objection; but did not incorporate the examples of applying SR 3.0.3, which were inserted after the first paragraph of the Bases, as follows (Passages deleted in the traveler but retained in the GTS Bases, and which the staff found unacceptable.

In the below markup of Revision 4 of the W-STS Bases for SR 3.0.3,

- Words proposed for deletion by TSTF-530 that were retained in the US600 GTS Bases are denoted by shading.
- Words proposed for deletion by TSTF-530 that were not included in the US600 GTS Bases are denoted by lineout.
- Words proposed for addition by TSTF-530 that were included in the US600 GTS Bases are denoted by underline.
- Words proposed for addition by TSTF-530 that were not included in US600 GTS Bases, are denoted by both underline and lineout.

- Word changes proposed in NRC staff approved traveler TSTF-529-A and that were incorporated into Revision 5 of W-STS are denoted by italicized words that are either underlined or lined out.

- The second paragraph was added to Revision 4 of W-STS SR 3.0.3 Bases by TSTF-545-A, Revision 3, which was incorporated into Revision 5 of W-STS SR 3.0.3 Bases.

- NuScale elected to include the current revision number (Revision 4) in the reference to RG 1.160 (ML18220B281) in the US460 GTS Bases for SR 3.0.3; Revision 5 of W-STS does not include the revision number. This change is denoted by bold style.

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been ~~*completed*~~ *performed* within the specified Frequency. A delay period of ~~up to~~ 24 hours or up to the limit of the specified Frequency, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed ~~within its specified Frequency~~ in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

When a Section 5.5, "Programs and Manuals," Specification states that the provisions of SR 3.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with SR 3.0.3 when the testing has not been performed within the testing interval (including the allowance of SR 3.0.2 if invoked by the Section 5.5 Specification).

This delay period provides adequate time to ~~*complete*~~ *perform* ~~the Surveillance~~ Surveillances that have been missed. This delay period permits the ~~*completion*~~ *performance* of a Surveillance before complying with Required Actions or other remedial measures that might preclude ~~completion~~ *performance* of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing ~~performing~~ the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a Frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR Part 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed ~~within the specified Frequency~~ when specified, SR 3.0.3 allows for the full delay period of up to the specified Frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 3.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of MODE changes imposed by Required Actions.

SR 3.0.3 is only applicable if there is a reasonable expectation the associated equipment is OPERABLE or that variables are within limits, and it is expected that the Surveillance will be met when performed. Many factors should be considered, such as the period of time since the Surveillance was last performed, or whether the Surveillance, or a portion thereof, has ever been performed, and any other indications, tests, or activities that might support the expectation that the Surveillance will be met when performed. An example of the use of SR 3.0.3 would be a relay contact that was not tested as required in accordance with a particular SR, but previous successful performances of the SR included the relay contact; the adjacent, physically connected relay contacts were tested during the SR performance; the subject relay contact has been tested by another SR; or historical operation of the subject relay contact has been successful. It is not sufficient to infer the behavior of the associated equipment from the performance of similar equipment. The rigor of determining whether there is a reasonable expectation a Surveillance will be met when performed should increase based on the length of time since the last performance of the Surveillance. If the Surveillance has been performed recently, a review of the Surveillance history and equipment performance may be sufficient to support a reasonable expectation that the Surveillance will be met when performed. For Surveillances that have not been performed for a long period or that have never been performed, a rigorous evaluation based on objective evidence should provide a high degree of confidence that the equipment is OPERABLE. The evaluation should be documented in sufficient detail to allow a knowledgeable individual to understand the basis for the determination.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used *repeatedly* ~~*as an operational convenience*~~ to extend Surveillance intervals. While up to 24 hours or ~~up to~~ the limit of the specified Frequency is provided to perform the missed Surveillance, it is expected that the ~~missed~~ Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," **Revision 4**. This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown.

The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances ~~not performed within the specified Frequency~~ for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances ~~not performed within the specified Frequency~~ will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed ~~performed and met~~ within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Completion Times of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

~~Performing and meeting~~ Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

These editorial changes from approved travelers mostly replaced phrases like "completing the surveillance" with "performing the surveillance," or vice versa, which clarified Revision 4 of the W-STS Bases for SR 3.0.3, and are consistent with the previous meaning of the affected passages. The editorial changes from unapproved TSTF-530 are acceptable because they clarify the intended meaning of the affected passages. The staff concludes that these editorial changes to Revision 4 of the W-STS LCO 3.0.3 Bases, which are proposed for the US460 SDAA GTS SR 3.0.3 Bases, improve the clarity of the reasons for the allowances for completing a missed surveillance. The staff therefore finds that US460 GTS SR 3.0.3 and associated Bases are acceptable because they are consistent with Revision 5 of the W-STS, and the remedial actions of SR 3.0.3 will assure that the associated LCO of a missed surveillance will be met as stipulated by 10 CFR 50.36(c)(3).

<u>SR 3.0.4</u>

The GTS SR 3.0.4 is consistent with Revision 5 of W-STS SR 3.0.4, which is based on traveler TSTF-359-A, Revision 9. Because this Specification matches Revision 5 of W-STS, SR 3.0.4 is acceptable.

Conclusion for GTS Section 3.0

Based on its review, the staff concludes that GTS Section 3.0 LCOs and SRs conform to the STS (with justified exceptions) and are appropriate for the NuScale US460 design, and therefore the staff finds that GTS Section 3.0 is acceptable.

16.4.5 Limiting Condition for Operation Statements (GTS Chapter 3, Sections 3.1 through 3.8)

The staff reviewed the LCO statement in each subsection of GTS Sections 3.1 through 3.8 for technical accuracy and consistency with the NuScale US460 design, as described in SDAA Part 2, regarding the number of subsystems, trains, channels, divisions, or separation groups of the specified SSC required to be operable; or the value of the limit that the specified process variable must be within. The staff also reviewed each LCO statement for clarity and consistency with the STS writer's guide phrasing, formatting, and punctuation conventions, and for overall nomenclature consistency with the GTS and Bases, and SDAA Part 2. In most cases, the staff determined that the LCO statements were clear and accurate and consistent with the NuScale US460 design. The following describes the resolution of issues for selected LCOs.

- LCO 3.1.9, "Boron Dilution Control"

This LCO specifies that in Modes 1, 2, and 3 with any dilution source flow path not isolated, the two CVCS DWSI valves shall be operable, boric acid supply boron concentration shall be within COLR limits, and CVCS makeup pump demineralized water flow path maximum flowrate shall be within COLR limits. In Mode 3 when the CVCS supply flow path is aligned through the shared MHS, which is done by opening the unit's four CVCS-MHS isolation valves and running a CVCS recirculation pump, this LCO also specifies that the CVCS-MHS flow paths to and from other module CVCS shall be isolated by a locked, sealed, or otherwise secured valve or device.

The makeup flowrate is limited to the capacity of one makeup pump when critical boron concentration is above 600 parts per million (ppm), according to FSAR Table 15.4-12, "Bounding Critical Boron Concentrations and Boron Reactivity Coefficients (15.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)," and Section 15.4.6.2, "Sequence of Events and Systems Operation." The critical boron concentration for operation of two makeup pumps at full capacity is appropriately maintained in the COLR. LCO 3.1.9 states that the "Maximum CVCS makeup pump demineralized water flow path flowrate shall be within the limits specified in the COLR." The staff concurs with this approach, provided the COLR specified flowrate limits are maintained consistent with the assumed flowrate limits stated in FSAR Section 15.4.6.2.

FSAR Section 15.4.6.3.2 (FSAR Section 15.4.6.3.4 in the DCA), "Input Parameters and Initial Conditions," states that "A minimum makeup temperature of 40 degrees F [4.4 degrees C] is assumed for the analysis of boron dilution of the RCS." During the US600 DCA review, as this temperature assumption is not explicitly surveilled or specified by GTS Subsection 3.1.9, by a letter dated January 29, 2019 (ML19029B572), NuScale provided a rationale for omitting this makeup water minimum temperature limit from both LCO 3.1.9 and the discussion of applicable safety analyses in Bases Subsection B 3.1.9. The applicant stated the following:

> The 40°F [4.4°C] used in the boron dilution analysis was chosen because the density of water is approximately at a maximum at that temperature. This maximizes the mass of unborated water injected and the dilution. This simplifies and provides a common baseline for boron effects analyses. Boron concentration limits are described and implemented in accordance with plant procedures so that the assumptions of the core operating limits report (COLR) are appropriately implemented. Consistent with industry practice, effects of temperature differences will be addressed by the implementing procedures required to implement the COLR and technical specification requirements.

The staff determined that this rationale provided by the applicant during the NuScale US600 DCA review is valid for the NuScale US460 SDAA. The staff concurs with LCO 3.1.9 not specifying and the Bases for LCO 3.1.9 not discussing the 4.4°C (40°F) makeup water temperature, which is an assumption used in the boron dilution analysis, because this temperature assumption maximizes the RCS boron dilution rate and the resulting positive reactivity insertion rate, which conservatively minimizes the time to criticality calculated by the safety analysis of inadvertent boron dilution events, as described in FSAR Section 15.4.6.2. The staff concludes this makeup water temperature assumption does not satisfy Criterion 2 and therefore finds that not including it in LCO 3.1.9 is acceptable.

- LCO 3.3.1, "Module Protection System (MPS) Instrumentation"

There is one CVCS-CIS manual override switch, O-1, per ESFAS Logic and Actuation division. This safety-related switch, the operation of which must be preceded by manual operation of the nonsafety enable switch, is used in a beyond-DBE involving a containment bypass leak of radioactivity. In SDAA Part 4, in the combined Applicable Safety Analyses, LCO, and Applicability sections of the Bases for LCO 3.3.1, on page B 3.3.1-20, the applicant included the following discussion:

> <u>Containment System Isolation Override Function, O-1, for the Containment Flood and Drain and CVCS Isolation Valves</u>
>
> The manual override switches (one per division) allow operations to bypass the containment system isolation signal for the containment flood and drain isolation valves and the CVCS isolation valves to allow control of containment water levels with an active containment system isolation actuation or CVCS isolation signal (except High Pressurizer Level).
>
> 1. If an automatic Containment System Isolation or automatic CVCS isolation signal (other than High Pressurizer Level) is present and RT-1 is active, the operator can activate the override function O-1 manually with the Override switch. This override will allow manual control of the containment flood and drain, [reactor coolant injection (RCI)] isolation, Pressurizer Spray isolation, and CES containment isolation valves from the module control system.
>
> 2. The Override switch must be manually restored when the override function O-1 is no longer needed.
>
> If the reactor trip breakers are closed, the override function O-1 is not available.

The staff finds that the above Bases passage adequately describes the override O-1 switch.

- LCO 3.4.10, "Low Temperature Overpressure Protection (LTOP) Valves"

This LCO specifies that each RVV that is in the closed position shall be operable, but does not state the implied requirement that both RVVs shall be closed and operable for LTOP. Two RVVs are required, since one RVV is sufficient to perform the overpressure prevention function; the other RVV accounts for the assumed worst case single active failure of an RVV to open on an LTOP actuation signal. This leads to a rather unconventional construction of the associated actions. The staff considered that a clearer presentation would be for the LCO to explicitly require two RVVs to be closed and operable for LTOP or at least one RVV be open. The applicant previously considered this observation during the NRC staff review of the DCA for the

NuScale US600 design, which needed two of three RVVs to perform the overpressure prevention function and so the LCO required the three RVVs in the US600 design to be closed and operable for LTOP; NuScale has maintained an equivalent position in the NuScale US460 SDAA as described above. As part of the US600 DCA review, in a letter dated December 12, 2018 (ML18347A619), the applicant declined to revise the LCO, Applicability, and Actions, as suggested by the staff, and stated the following:

> NuScale technical specifications are developed in close coordination and consultation with the operating staff. Experience with the technical specifications in simulator operations and in support of DCA development has not identified the need for a modified presentation of this LCO.
>
> The staff observation of the LCO as "unconventional" is accurate, however it is appropriate for the NuScale design. The proposed presentation would reduce clarity and introduce unnecessary complexity to the specifications. Therefore, the current construction of the LCO is being retained.

Based on the applicant's letter submitted as part of the US600 DCA review, and the NRC staff's review of the SDAA GTS Subsection 3.4.10, the staff accepts that the SDAA LCO as proposed will ensure that the LTOP function of the RVVs is operable when the wide range RCS cold temperature is below the T-1 interlock setting of approximately 143.3°C (290°F). Therefore, SDAA Specification 3.4.10 operability and action requirements, and associated Bases, are acceptable.

Conclusion for LCO statements

Based on its review, the staff concludes the LCO statements are acceptable.

16.4.6 Applicability Statements (GTS Chapter 3, Sections 3.1 through 3.8)

The applicability statements of LCOs for SSCs, process parameters, and other operational restrictions must be sufficiently broad to ensure that the safety-related function, initial condition, or other restriction specified by the LCO protects the validity of the transient and accident analyses, thereby ensuring safe operation of the reactor facility and adequate protection of public health and safety.

The operational mode definitions in SDAA GTS Section 1.1, Table 1.1-1, reflect the unique characteristics of the design and operation of an NPM. The mode definitions, therefore, differ from the mode definitions used in the STS. Section 3.1 of TR-101310-NP, Revision 1, describes the equivalence of STS and GTS mode definitions and states the following:

> The MODE definition used in the TS is changed to better align with the plant response behavior. Specifically, the upper temperature limit on MODE 3, Safe Shutdown, is removed and the operational region expanded to include temperatures above the minimum temperature for criticality. This is accomplished by including 'and' and 'or' requirements so that above the minimum temperature for criticality the plant is in MODE 3 if it is PASSIVELY COOLED, and in MODE 2 if it is not PASSIVELY COOLED. …
>
> This MODE definition clarifies that the plant is in a passively safe configuration once PASSIVE COOLING is established, regardless of the reactor coolant temperature relative to the minimum temperature for criticality.

For most GTS LCOs, the applicability statement includes one or more of Mode 1, Mode 2, and some part or all of Mode 3.

The GTS definition of Mode 1 (Operations) encompasses the W-AP1000-STS definitions of Mode 1 (Power Operation) and Mode 2 (Startup). GTS Mode 1 is defined by the core reactivity condition (k_{eff} ≥ 0.99). Although not specified in Table 1.1-1, being in Mode 1 effectively also includes all reactor coolant temperature indications being ≥ 173.89 (or 174)°C (345°F) as specified by LCO 3.4.2, "RCS Minimum Temperature for Criticality," which is applicable in Mode 1; W-AP1000-STS definitions of Modes 1 and 2 use the same core reactivity condition (k_{eff} ≥ 0.99), but use core thermal power as a percent of RTP (> 5 percent RTP for Mode 1 and ≤ 5 percent RTP for Mode 2) instead of an explicit reactor coolant temperature indication. Because the minimum RCS temperature for criticality (minimum temperature for criticality) specified by GTS LCO 3.4.2 is also 174°C (345°F), the GTS Mode 1 definition is seen to be equivalent to the W-AP1000-STS Mode 1 and Mode 2 definitions combined. The broader RCS temperature range for the NuScale US460 Mode 1 definition reflects the need to use the module heating system of the CVCS to reach the minimum temperature for criticality and then use core thermal power to reach RCS normal operating temperatures (beginning around 15 percent RTP) because core flow is by natural circulation. An AP1000 unit achieves the minimum temperature for criticality (T_{avg} ≥ 288.3°C (551°F)), which is the average RCS temperature near the RCS normal operating temperature, by adding heat with forced core flow from running reactor coolant pumps. Defining GTS Mode 1 in this way results in GTS applicability and action statements that appear different but that are generally no more or less restrictive than equivalent requirements in the W-AP1000-STS for Modes 1 and 2.

The GTS definition of Mode 2 (hot shutdown) is equivalent to the W-AP1000-STS definition of Mode 3 (hot standby, k_{eff} < 0.99, T_{avg} > 420°F). The GTS Mode 2 is defined by the core reactivity condition (k_{eff} < 0.99) and reactor coolant temperature indication (any temperature indication ≥ 174°C (345°F)); it also requires the module to be not passively cooled. The W-AP1000-STS definition of Mode 3 uses the same core reactivity condition (k_{eff} < 0.99), but a higher reactor coolant average temperature indication (T_{avg} > 216°C (420°F)). Defining GTS Mode 2 in this way results in GTS applicability and action statements that appear different but that are generally no more or less restrictive than equivalent requirements in the W-AP1000-STS for Mode 3.

The GTS definition of Mode 3 (safe shutdown) encompasses the W-AP1000-STS definitions of Mode 3 (hot standby, k_{eff} < 0.99, T_{avg} > 420°F), Mode 4 (safe shutdown) and Mode 5 (cold shutdown). The GTS Mode 3 is defined by the same core reactivity condition (k_{eff} < 0.99) and either the reactor coolant temperature indication (all temperature indications < 174°C (345°F)) or when the module is passively cooled. The W-AP1000-STS definition of Mode 4 uses the same core reactivity condition (k_{eff} < 0.99), and a part of the reactor coolant temperature indication range (T_{avg} ≤ 216°C but > 93°C (≤ 420°F but > 200°F)). The W-AP1000-STS definition of Mode 5 also uses the same core reactivity condition (k_{eff} < 0.99), but the lower part of the reactor coolant temperature indication range (T_{avg} ≤ 93.3°C (200°F)). The GTS definition of Mode 3 also includes when any CRA is capable of withdrawal, or any CVCS or CFDS connection to the module is not isolated. Defining GTS Mode 3 in this way results in GTS applicability and action statements that appear different but that are generally no more or less restrictive than equivalent requirements in the W-AP1000-STS for Modes 3, 4, and 5.

The GTS definition of Mode 3 also encompasses the same temperature range as the GTS definition of Mode 2 (any indicated reactor coolant temperature at or above 174°C (345°F)) if the module is being passively cooled. During the audit phase of the US460 SDAA (ML24211A089), NuScale's docketed response to an audit question indicated that entering Mode 3 from Mode 2 by initiating passive cooling would not be the preferred method of complying with Required Actions that direct the unit to be in Mode 3 within 36 hours.

In SDAA Part 4, regarding the GTS definition of Mode 3 in Table 1.1-1, in addition to meeting the reactor coolant temperature condition (all indications < 174°C (345°F)) and the core reactivity condition (k_{eff} < 0.99), being in Mode 3 also includes meeting one or more of three conditions, as specified by Table 1.1-1 footnote (a):

(a) Any CRA capable of withdrawal, or any CVCS or CFDS connection to the module not isolated.

To enter Mode 4 (Transition) from Mode 3, the unit must satisfy all of the following conditions: (1) k_{eff} < 0.95, (2) all CRAs are incapable of withdrawal, (3) all CVCS module connections are isolated, (4) all CFDS module connections are isolated, (5) all RVVs are electrically isolated, and (6) all reactor vessel flange bolts are fully tensioned. The latter five conditions are specified by table footnotes (b) and (c):

(b) All CRAs incapable of withdrawal, and CVCS and CFDS connections to the module isolated, and all reactor vent valves electrically isolated.

(c) All reactor vessel flange bolts fully tensioned.

The GTS definition of Mode 4, although equivalent to the portion of the W-AP1000-STS definition of Mode 5 in which the RCS is filled but vented to the containment, is unique to the NuScale US460 design because during Mode 4, the unit staff can move the module to the refueling pool. Also, the core reactivity condition in Mode 4 is more limiting, with k_{eff} < 0.95 instead of k_{eff} < 0.99 in the W-AP1000-STS.

The GTS definition of Mode 5 (refueling) is equivalent to the W-AP1000-STS definition of Mode 6 (refueling), during which the reactor vessel closure head is removed to permit movement of irradiated fuel assemblies in the reactor vessel, the refueling pool, and the spent fuel pool. Both the GTS Mode 5 and W-AP1000-STS Mode 6 require one or more reactor vessel flange bolts for the GTS (head closure bolts for the W-AP1000-STS) to be less than fully tensioned.

In addition to the reactivity condition and reactor coolant temperature indication, GTS applicability statements are defined in relation to the active or not active status of RTS and ESFAS operating bypass interlocks and permissives. These interlocks and permissive Functions use sensor signals of NPM process variables and the open or closed status of valves and circuit breakers, as well as the functional status of the CRA drive mechanisms. Typically, there are four sensor channels for each process variable. When at least two out of four channels send an enable signal to the coincidence logic in each of the two actuation logic divisions, each division of the interlock Function outputs a signal in the same division to the actuation logic of the RTS and ESFAS Functions that use the interlock signal. Interlock bypass signals typically require three out of four sensor channels indicating the RTS or ESFAS Function is no longer needed to be operable to bypass the Function. Depending on how an RTS or ESFAS Function is designed to use the interlock or permissive signal, the signal will (1) cause the Function to be automatically bypassed or allow it to be manually bypassed, or (2) cause the Function to be automatically unbypassed or enabled. Because the process variable instrument channels that are used for MPS Functions are also used for interlock and permissive Functions, GTS Section 3.3, “Instrumentation,” does not explicitly state duplicate operability requirements for process variable instrument channels that also generate interlock and permissive signals.

Thus, an MPS Function's applicability statement mode requirements, which are modified by whether a relevant process variable is above or below the trip setting of the associated interlock, implicitly requires the interlock to automatically bypass, or automatically permit the manual bypassing of, the associated RTS or ESFAS Function when the Function is not needed to be operable and to automatically unbypass or enable the Function when it is needed to be operable.

Applicability statements are specified using the modes defined in GTS Table 1.1-1 and interlock definitions, which are based on the values of the following listed variables, and the status of the following listed SSCs. This list only describes LCO applicability statements that modify the range of a defined MODE or include other specified conditions. The list also shows that the LCO applicabilities for instrumentation and actuation logic Functions are consistent with or bound the LCO applicabilities for the supported actuated devices.

(1) Core reactivity condition

(a) In MODE 1 with k_{eff} < 1.0, the SDM requirements of LCO 3.1.1 are applicable.

(b) In MODE 1 with k_{eff} ≥ 1.0, the regulating group CRA insertion limits of LCO 3.1.6 are applicable.

(2) Core power level

(a) In MODE 1 with THERMAL POWER ≥ 20% RTP, the following LCOs are applicable:

3.2.1 Enthalpy Rise Hot Channel Factor
3.2.2 AXIAL OFFSET

(b) N-2H—Power Range Linear Power Interlock

N-2H is active or established[6] if
at least 3 of 4 power range channels indicate < 15% RTP.

N-2H is not active if
at least 2 of 4 power range channels indicate ≥ 15% RTP.

When active, N-2H automatically bypasses the following MPS Function of LCO 3.3.1 in MODE 1 with THERMAL POWER < 15% RTP:

2.a RTS on High Power Range Positive and Negative Rate

When not active, N-2H automatically enables the above MPS Function of LCO 3.3.1 in MODE 1 with THERMAL POWER ≥ 15% RTP.

Regardless of whether the N-2H interlock is active, in MODE 1:

(1) The operability of the two divisions of the RTS Logic and Actuation

[6] In SDAA Part 4, when describing the state of an automatic interlock, the Bases for GTS Section 3.3 use "is active" in some cases and "is established" in other cases to indicate that the operational bypass generated by the interlock prevents the supported MPS function from actuating on an automatic actuation signal.

Function of LCO 3.3.2 is required. Note that this includes each RTB.

(2) The operability of the two divisions of the following Manual Actuation Function of LCO 3.3.4 is required:

1 RTS

(c) <u>N-2L—Power Range Linear Power Interlock</u> (increasing power interlock)

N-2L interlock is active if
at least 3 of 4 power range channels indicate > 15% RTP.

N-2L interlock is not active if
at least 2 of 4 power range channels indicate ≤ 15% RTP.

When active, N-2L interlock automatically bypasses the following MPS Functions of LCO 3.3.1 in MODE 1 with THERMAL POWER > 15% RTP:

3.a RTS on High Intermediate Range Log Power Rate
3.b DWSI on High Intermediate Range Log Power Rate

When not active, N-2L interlock automatically enables the above MPS Functions of LCO 3.3.1 in MODE 1 with THERMAL POWER ≤ 15% RTP.

Regardless of whether the N-2L interlock is active, in MODE 1:

(1) The operability of the two divisions of the RTS Logic and Actuation Function of LCO 3.3.2 is required. Note that this includes each RTB.

(2) The operability of the two divisions of the following ESFAS Logic and Actuation Function of LCO 3.3.3 is required:

4 DWSI

(3) The operability of the two divisions of the following Manual Actuation Functions of LCO 3.3.4 is required:

1 RTS
5 DWSI

(4) The operability of the two CVCS demineralized water isolation valves is required by LCO 3.1.9 with any dilution source flow path not isolated.

(d) <u>N-2L—Power Range Linear Power Permissive (increasing power permissive)</u>

N-2L permissive is active if
at least 3 of 4 power range channels indicate > 15% RTP.

N-2L permissive is not active if
at least 2 of 4 power range channels indicate ≤ 15% RTP.

When active, N-2L permissive allows a manual operating bypass of the following MPS Functions of LCO 3.3.1 in MODE 1 with THERMAL POWER > 15% RTP:

1.a RTS on High-1 Power Range Linear Power (High-1 setpoint)
1.b DWSI on High-1 Power Range Linear Power (High-1 setpoint)

Note: Manually bypassing Function 1.a, RTS on Power Range Linear Power, bypasses the High-1 setpoint without affecting the High-2 setpoint.

When not active, N-2L permissive automatically enables the above MPS Functions of LCO 3.3.1 in MODE 1 with THERMAL POWER ≤ 15% RTP.

(e) <u>N-1—Intermediate Range Log Power Permissive and Interlock</u>

N-1 permissive and interlock are active if
at least 3 of 4 intermediate range log power channels indicate approximately one decade above the channel lower range limit.

Intermediate Range Log Power instrument indicated span is six decades from 1 x 10^4 cps to 1 x 10^{10} cps (1 x 10^{-4} to 1 x 10^2 percent RTP) (FSAR Table 7.1-2—Intermediate Range Log Power nominal range).

N-1 permissive and interlock are not active when
interlock or permissive condition is no longer satisfied for
at least 2 of 4 intermediate range log power channels.

When active, N-1 permissive allows a manual operating bypass of the following MPS Functions of LCO 3.3.1 in MODE 1 with intermediate range log power > 1 x 10^5 cps:

4.a RTS on High Source Range Count Rate
4.b DWSI on High Source Range Count Rate
5.a RTS on High Source Range Log Power Rate
5.b DWSI on High Source Range Log Power Rate

When active, N-1 interlock automatically bypasses the following MPS Function of LCO 3.3.1 in MODE 1:

6.a DWSI on High Source Range Subcritical Multiplication

When not active, N-1 interlock automatically enables the above MPS Functions of LCO 3.3.1 in MODE 1.

Regardless of whether the N-2L and N-1 interlocks and permissives are active, in MODE 1:

(1) The operability of the two divisions of the RTS Logic and Actuation Function of LCO 3.3.2 is required. Note that this includes each RTB.

(2) The operability of the two divisions of the following ESFAS Logic and Actuation Function of LCO 3.3.3 is required:

4 DWSI

(3) The operability of the two divisions of the following Manual Actuation Functions of LCO 3.3.4 is required:

1 RTS
5 DWSI

(4) The operability of the two CVCS demineralized water isolation valves is required by LCO 3.1.9 with any dilution source flow path not isolated.

(3) <u>RTB position</u>

(a) <u>RT-1—Reactor Tripped Interlock</u>

RT-1 is active if
two of the two divisional RTBs indicate open.

RT-1 is not active if
one or two of the two divisional RTBs indicate closed.

The RT-1 Interlock is used in conjunction with the T-2, T-3, and L-1 interlocks, and the override function O-1, in LCO applicability statements described below.

When both divisional RTBs are open (RT-1 is active, reactor is tripped):

- L-1 is enabled to automatically bypass associated LCO 3.3.1 MPS Functions upon 3 of 4 containment water level channels indicating > 13.7 m (45 ft):

  11.d SSI on Low Pressurizer Level – Mode 3[j]
  12.a CIS on Low Low Pressurizer Level – Mode 3[j]
  12.b CVCSI on Low Low Pressurizer Level – Mode 3[j]
  19.b SSI on Low Low Main Steam Pressure – Modes 2[k], 3[k]
  21.b SSI on Low Steam Superheat – Modes 2[m], 3[m]

  The Table 3.3.1-1 footnotes containing L-1 are the following:

  (j) With RCS temperature above the T-2 interlock and *containment water level below the L-1 interlock*.

  (k) *With containment water level below the L-1 interlock* with RCS temperature above the T-3 interlock, or *with containment water level below the L-1 interlock* with V-1 not active (both FWIVs open).

  (m) *With containment water level below the L-1 interlock* and with V-1 not active (both FWIVs open).

- T-2 is enabled to automatically bypass associated LCO 3.3.1 MPS Functions upon 3 of 4 wide range RCS hot temperature channels indicating < 93.3°C (200°F):

  12.a CIS on Low Low Pressurizer Level
  12.b CVCSI on Low Low Pressurizer Level

- O-1 switch is enabled to allow manual control of CIVs, even if CIS and CVCSI actuation signals are present, except for LCO 3.3.1 MPS Function 10.c, CVCSI on High Pressurizer Level.

- T-5 is enabled to automatically bypass the associated LCO 3.3.1 MPS Function upon 3 of 4 wide range RCS cold temperature channels indicating < 226.7°C (440°F):

  23.a ECCS on Low Reactor Pressure Vessel Riser Level

(4) Containment vessel water level

(a) L-1—Containment Water Level Interlock

L-1 is active if
at least 3 of 4 containment water level channels indicate > 13.7 m (45 ft)
and RT-1 is also active (two divisional RTBs indicate open).

L-1 is not active if
at least 2 of 4 containment water level channels indicate ≤ 13.7 m (45 ft)
or RT-1 is not active (one or two divisional RTBs indicate closed).

The L-1 interlock is used in conjunction with the T-2 (93.3°C (200°F)) and T-3 (171.1°C (340°F)) interlocks, in LCO applicability statements described below.

In MODE 3, the normally open CVCS CIVs (inboard and outboard, respectively) on the injection line (CVC-HOV-0331 and CVC-HOV-0330), the discharge line (CVC-HOV-0334 and CVC-HOV-0335), and pressurizer spray line (CVC-HOV-0325 and CVC-HOV-0324) are required to be operable by LCO 3.4.6 in MODE 3, and by LCO 3.6.2 in MODE 3 with RCS hot temperature ≥ 93.3°C (200°F).

(5) Reactor coolant temperature

°C	°F		
93.3	200	T-2	Wide Range (WR) RCS Hot Temperature (Thot) T-2 interlock (active below T-2)
143.3	290	T-1	WR RCS Cold Temperature (Tcold) T-1 interlock, LTOP enable temperature (active above T-1)
171.1	340	T-3	WR RCS Thot T-3 interlock (active below T-3)
173.9	345	Lowest RCS temperature for being in Mode 2 (Any Indicated Reactor Coolant Temperature (RCT_{indic}) ≥ 345°F); also, the RCS minimum temperature for criticality (LCO 3.4.2)	
226.7	440	T-5	WR RCS Tcold T-5 interlock (active below T-5)
260.0	500	T-4	Narrow Range RCS Thot T-4 Interlock (active below T-4)
290.6	555	Approximate setpoint for MPS Functions	

14.a, RTS on High RCS Average Temperature (Tavg), and
14.b, DWSI on High RCS Tavg

Tavg = ½ × (Thot + Tcold);

Thot is the coolant riser outlet narrow range temperature;
Tcold is the coolant temperature near the top of the reactor vessel downcomer below the steam generator outlet

326.7 620 Approximate setpoint for MPS Functions
13.a, RTS on High Narrow Range RCS Thot,
13.b, DHRS on High Narrow Range RCS Thot,
13.c, SSI on High Narrow Range RCS Thot,
13.d, DWSI on High Narrow Range RCS Thot,
13.e, PHT on High Narrow Range RCS Thot;
and
ECCS on High High RCS Tavg

(a) MODE 3 with any RCS temperature ≥ 93.3°C (200°F), the lower moderator temperature coefficient limit of LCO 3.1.3 is applicable.

(b) MODE 3 with RCS hot temperature ≥ 93.3°C (200°F) and all ECCS valves closed, the following LCOs are applicable:

3.4.5 RCS Operational LEAKAGE
3.4.7 RCS Leakage Detection Instrumentation (except during containment flooding operations)

(c) MODE 3 with RCS hot temperature ≥ 93.3°C (200°F), the following LCO is applicable:

3.6.2 Containment Isolation Valves

(d) T-2—Wide Range RCS Hot Temperature Interlock

T-2 interlock is active if at least 3 of 4 wide range RCS hot temperature channels indicate < 93.3°C (200°F) *and* RT 1 is also active (two divisional RTBs indicate open)

T-2 interlock is not active if at least 2 of 4 wide range RCS hot temperature channels indicate ≥ 93.3°C (200°F) *or* RT-1 is not active (one or two divisional RTBs indicate closed)

When active, the T-2 interlock automatically bypasses the following MPS Functions of LCO 3.3.1 in MODE 3 with wide range RCS hot temperature below 93.3°C (200°F):

11.d SSI on Low Pressurizer Level
12.a CIS on Low Low Pressurizer Level
12.b CVCSI on Low Low Pressurizer Level

When not active, T-2 Interlock automatically enables the above MPS Functions of LCO 3.3.1 in MODE 3 provided wide range RCS hot temperature is ≥ 93.3°C (200°F).

(e) T-1—Wide Range RCS Cold Temperature Interlock

T-1 is active if
3 of 4 wide range RCS cold temperature channels indicate > LTOP enable temperature specified in the PTLR (approximately 143.3°C (290°F)).

T-1 is not active if
2 of 4 wide range RCS cold temperature channels indicate ≤ LTOP enable temperature specified in the PTLR (approximately 143.3°C (290°F)).

When active, T-1 interlock automatically bypasses the following MPS Function of LCO 3.3.1 in MODE 3 with wide range RCS cold temperature > approximately 143.3°C (290°F) *or with one or both RVVs open*:

27.a LTOP on High RCS Pressure

and the operability of the two divisions of the following ESFAS Logic and Actuation Function of LCO 3.3.3 is not required in MODE 3 with T-1 active *or with one or both RVVs open*:

7 LTOP

and the operability of the two divisions of the following Manual Actuation Function of LCO 3.3.4 is not required in MODE 3 with T-1 active *or with one or both RVVs open*:

8 LTOP

but the operability of the two RSVs of LCO 3.4.4 is required in MODE 3 with T-1 active,

and the operability of the two closed RVVs of LCO 3.4.10 is not required in MODE 3 with T-1 active.

When not active, T-1 Interlock automatically enables the above MPS Function 27.a of LCO 3.3.1 in MODE 3 with wide range RCS cold temperature below approximately 143.3°C (290°F), and both RVVs closed (LTOP RVV lift setting, in terms of wide range pressurizer pressure, is a function of wide range RCS cold temperature),

and in MODE 3 with wide range RCS cold temperature below approximately 143.3°C (290°F) *and with both RVVs closed*, Function 7 of LCO 3.3.3 and Function 8 of LCO 3.3.4 are required to be operable,

but the operability of the two RSVs of LCO 3.4.4 is not required in MODE 3 with T-1 not active,

and the operability of both closed RVVs of LCO 3.4.10 is required in MODE 3 with T-1 not active.

(f) T-3—Wide Range RCS Hot Temperature (Thot) Interlock

T-3 is active if
3 of 4 wide range RCS hot temperature channels indicate < 171.1°C (340°F).

T-3 is not active if
2 of 4 wide range RCS hot temperature channels indicate ≥ 171.1°C (340°F).

When active, T-3 interlock automatically bypasses the following MPS Functions of LCO 3.3.1 in MODE 3 (These functions are required in Mode 3 as modified by Table 3.3.1-1 footnote (i) "With RCS temperature above the T-3 interlock"):

8.d	Pressurizer Line Isolation on Low Pressurizer Pressure	
22.b	CIS	on High Narrow Range Containment Pressure
22.c	DHRS	on High Narrow Range Containment Pressure
22.d	SSI	on High Narrow Range Containment Pressure
22.f	CVCSI	on High Narrow Range Containment Pressure

and in MODE 3 below T-3 or with [only one CRA capable of withdrawal] (These functions are required in MODE 3 as modified by Table 3.3.1-1 footnote (b) "When capable of withdrawal of more than one CRA and the RCS temperature is above the T-3 interlock."):

7.d	DWSI	on High Pressurizer Pressure
9.a	RTS	on Low Low Pressurizer Pressure
9.b	DWSI	on Low Low Pressurizer Pressure
10.b	DWSI	on High Pressurizer Level
11.e	DWSI	on Low Pressurizer Level
19.c	DWSI	on Low Low Main Steam Pressure
22.e	DWSI	on High Narrow Range Containment Pressure

and in MODE 3 below T-3 or above L-1, CNV water level ≥ 45 ft (This function is required in MODE 3 as modified by Table 3.3.1-1 footnote (k) "With containment water level below the L-1 interlock with reactor narrow range temperature above the T-3 interlock …"):

19.b	SSI	on Low Low Main Steam Pressure

and in MODE 3 below T-3 or with pressurizer heater breakers open (This function is required in Mode 3 as modified by Table 3.3.1-1 footnote (n) "With RCS temperature above the T-3 interlock and pressurizer heater breakers closed."):

22.g	PHT	on High Narrow Range Containment Pressure

When not active (Wide Range Thot ≥ 171.1°C (340°F)), T-3 interlock automatically enables the following MPS Functions of LCO 3.3.1 in MODE 3 (as modified by Table 3.3.1-1 footnote (i) "With RCS temperature above the T-3 interlock"):

22.b	CIS	on High Narrow Range Containment Pressure
22.c	DHRS	on High Narrow Range Containment Pressure

22.d SSI on High Narrow Range Containment Pressure
22.f CVCSI on High Narrow Range Containment Pressure

and in MODE 3 (as modified by Table 3.3.1-1 footnote (k) "With containment water level below the L-1 interlock with reactor narrow range temperature above the T-3 interlock …") provided CNV water level is < 45 ft:

19.b SSI on Low Low Main Steam Pressure

and provided more than one CRA is capable of withdrawal:

19.c DWSI on Low Low Main Steam Pressure
22.e DWSI on High Narrow Range Containment Pressure
26.e DWSI on High Under-the-Bioshield Temperature

and provided pressurizer heater breakers are closed:

22.g PHT on High Narrow Range Containment Pressure

(g) T-4—Narrow Range RCS Hot Temperature Interlock

T-4 is active if
3 of 4 narrow range RCS hot temperature channels indicate < 260°C (500°F).

T-4 is not active if
2 of 4 narrow range RCS hot temperature channels indicate ≥ 260°C (500°F).

When active, T-4 Interlock automatically bypasses the following MPS Functions of LCO 3.3.1 in MODE 1 (These functions are required in Mode 1 as modified by Table 3.3.1-1 footnote (h) "With RCS temperature above the T-4 interlock."):

8.a RTS on Low Pressurizer Pressure
8.b SSI on Low Pressurizer Pressure
8.c DWSI on Low Pressurizer Pressure
11.b CIS on Low Pressurizer Level
11.c DHRS on Low Pressurizer Level
11.f CVCSI on Low Pressurizer Level
18.a RTS on Low Main Steam Pressure
18.b SSI on Low Main Steam Pressure
18.c DWSI on Low Main Steam Pressure

When not active, T-4 Interlock automatically enables the above MPS Functions of LCO 3.3.1 in MODE 1.

(6) Status of Passive Cooling. The NPM is Passively Cooled if (a) one or more RVVs are open and one or more RRVs are open; (b) DHRS is in operation; or (c) CNV water level is > 13.7 m (45 ft) (L-1 active). In MODE 3 with Passive Cooling in operation, the safety function of the ECCS and DHRS is being fulfilled.

(a) LCO 3.0.3 is only applicable in MODES 1 and 2, and in MODE 3 when not Passively Cooled.

(b) If Passive Cooling is in operation, the operability of the four channels of the following MPS Functions of LCO 3.3.1 is not required in MODE 3 when Passively Cooled (These functions are required in Mode 3 as modified by Table 3.3.1-1 footnote (f) “When not PASSIVELY COOLED.”):

7.b	DHRS	on High Pressurizer Pressure
7.c	SSI	on High Pressurizer Pressure
13.b	DHRS	on High Narrow Range RCS Hot Temperature
13.c	SSI	on High Narrow Range RCS Hot Temperature
17.b	DHRS	on High Main Steam Pressure
17.c	SSI	on High Main Steam Pressure
24.a	ECCS	on Low Low RPV Riser Level
25.c	DHRS	on Low AC Voltage to EDAS Battery Chargers
25.d	SSI	on Low AC Voltage to EDAS Battery Chargers
25.h	ECCS	on Low AC Voltage to EDAS Battery Chargers
26.c	DHRS	on High Under-the-Bioshield Temperature
26.d	SSI	on High Under-the-Bioshield Temperature

and the operability of the two divisions of the following ESFAS Logic and Actuation Functions of LCO 3.3.3 is not required in MODE 3 (These functions are required in Mode 3 as modified by Table 3.3.3-1 footnote (a) “When not PASSIVELY COOLED.”):

1	ECCS
2	DHRS
8	SSI

and the operability of the two divisions of the following Manual Actuation Functions of LCO 3.3.4 is not required in MODE 3 (These functions are required in Mode 3 as modified by Table 3.3.4-1 footnote (b) “When not PASSIVELY COOLED.”):

2	ECCS
3	DHRS
9	SSI

and the requirements of LCO 3.4.9, “SG Tube Integrity,” are not required to be met in MODE 3,

and the requirements of LCO 3.5.1, ECCS, for the two RRVs and the two RVVs to be operable, are not required to be met in MODE 3,

and the requirement of LCO 3.5.2, DHRS, for the two DHRS loops to be operable, is not required to be met in MODE 3,

and the requirement of LCO 3.6.1 for the containment to be operable, is not required to be met in MODE 3,

but the requirement of LCO 3.6.3 for containment closure, is required to be met in MODE 3, and the requirement of LCO 3.7.1, for two MSIVs and two MSIV bypass valves in each main steam line to be operable, is not required to be met in MODE 3,

and the requirement of LCO 3.7.2, for one FWIV and one FWRV for each SG to be operable, is not required to be met in MODE 3.

If Passive Cooling is not in operation, the above LCOs are required to be met in MODE 3, except for LCO 3.6.3, which is not required to be met.

(7) Position of pressurizer heater breakers (open or closed). If pressurizer heater breakers are open, the operability of the four channels of the following MPS pressurizer heater trip (PHT) Functions is not required in MODES 2 and 3 (These functions are required in Modes 2 and 3 as modified by Table 3.3.1-1 footnote (g) “With pressurizer heater breakers closed.”):

7.e	PHT	on High Pressurizer Pressure
11.g	PHT	on Low Pressurizer Level
13.e	PHT	on High Narrow Range RCS Hot Temperature
17.e	PHT	on High Main Steam Pressure
22.g	PHT	on High Narrow Range Containment Pressure; or if RCS temperature is below T-3 interlock in Mode 3. (This function is required in Mode 3 as modified by Table 3.3.1-1 footnote (n) “With RCS temperature above the T-3 interlock and pressurizer heater breakers closed.”)
25.g	PHT	on Low AC Voltage to EDAS Battery Chargers

and the operability of the two divisions of the following ESFAS Logic and Actuation Function of LCO 3.3.3 is not required in MODES 2 and 3 (This function is required in Modes 2 and 3 as modified by Table 3.3.3-1 footnote (b) “With pressurizer heater breakers closed.”):

6 PHT

and the operability of the two divisions of the following Manual Actuation Function of LCO 3.3.4 is not required in MODES 2 and 3 (This function is required in Modes 2 and 3 as modified by Table 3.3.4-1 footnote (c) “With pressurizer heater breakers closed.”):

7 PHT

(8) Status of CRA withdrawal capability (more than one capable; none or one capable). If no more than one CRA is capable of withdrawal, the operability of the four channels of the following MPS Functions of LCO 3.3.1 is not required in MODES 2 and 3:

1.a	RTS	on High-1 Power Range Linear Power
1.b	DWSI	on High-1 Power Range Linear Power
3.a	RTS	on High Intermediate Range Log Power Rate
3.b	DWSI	on High Intermediate Range Log Power Rate
4.a	RTS	on High Source Range Count Rate
4.b	DWSI	on High Source Range Count Rate
5.a	RTS	on High Source Range Log Power Rate
5.b	DWSI	on High Source Range Log Power Rate

7.a	RTS	on High Pressurizer Pressure
7.d	DWSI	on High Pressurizer Pressure
9.a	RTS	on Low Low Pressurizer Pressure
9.b	DWSI	on Low Low Pressurizer Pressure
10.a	RTS	on High Pressurizer Level
10.b	DWSI	on High Pressurizer Level
11.a	RTS	on Low Pressurizer Level
11.e	DWSI	on Low Pressurizer Level
16.a	RTS	on Low Low RCS Flow
17.a	RTS	on High Main Steam Pressure (Mode 2 only)
17.d	DWSI	on High Main Steam Pressure (Mode 2 only)
19.a	RTS	on Low Low Main Steam Pressure
19.c	DWSI	on Low Low Main Steam Pressure
22.a	RTS	on High Narrow Range Containment Pressure
22.e	DWSI	on High Narrow Range Containment Pressure
25.a	RTS	on Low AC Voltage to EDAS Battery Chargers
25.e	DWSI	on Low AC Voltage to EDAS Battery Chargers
26.a	RTS	on High Under-the-Bioshield Temperature
26.e	DWSI	on High Under-the-Bioshield Temperature

If more than one CRA are capable of being withdrawn, the above LCOs <u>are required to be met in the specified MODES</u>.

If no CRA is capable of withdrawal, the operability of the two divisions of the RTS Logic and Actuation Function of LCO 3.3.2 <u>is not required</u> in MODES 2 and 3;

and the operability of the two divisions of the following Manual Actuation Function of LCO 3.3.4 <u>is not required</u> in MODES 2 and 3:

1 RTS

If one or more CRAs are capable of being withdrawn, the above LCOs <u>are required to be met in the specified MODES</u>.

The operability of the two CVCS demineralized water isolation valves <u>is required</u> by LCO 3.1.9 in MODES 2 and 3, regardless of CRA withdrawal capability, because MPS Function 15.a, DWSI on Low RCS Flow, is also required in MODES 2 and 3 regardless of CRA withdrawal capability.

(9) <u>Position of Feedwater Isolation Valves (FWIVs).</u> The valve position indication of the two FWIVs must be operable in MODE 1 and indicate that both FWIVs are open to enable Function 21.a, RTS on Low Steam Superheat, and Function 21.c, DWSI on Low Steam Superheat. If at least one FWIV does not indicate open, the FWIV closed interlock V-1 automatically bypasses these functions.

(10) <u>Position of reactor vessel upper assembly.</u> The two refueling neutron flux channels and the neutron flux audible count rate channel are required to be operable in MODE 5 by LCO 3.8.1, except when the reactor vessel upper assembly is seated on the reactor vessel flange.

The staff compared the Applicable Safety Analyses and Applicability sections of the Bases of each LCO subsection to check that each SSC credited by an analysis of a postulated accident, infrequent event, or AOO is required to be operable for the range of operational modes and other specified conditions in which the analysis assumes the analyzed event could occur and require mitigation by the RTS and the ESF systems.

The staff verified that the limiting applicability of each MPS instrumentation Function spans the applicability of all associated RTS and ESFAS Logic and Actuation Functions. Although the applicability of an MPS Function may span a smaller range of unit operational conditions for one associated RTS or ESFAS Function, the Actuation Function with the broadest applicability determines the limiting applicability for the MPS instrumentation Function.

The staff verified that the applicability of each LCO-required support system bounds the LCO applicability of all its LCO-required supported systems.

Conclusion for GTS Chapter 3 LCO Applicability Statements

Based on its review, the staff finds that the LCO applicability statements are acceptable.

16.4.7 Action Requirements (GTS Chapter 3, Sections 3.1 through 3.8)

The staff reviewed the Actions table for each LCO subsection to determine whether the action requirements (Actions) are appropriate for the safety significance of each Condition where the associated LCO is not met.

For each LCO requiring operability of redundant trains of a safety system, the staff verified that the Actions table includes (1) a Condition for one train inoperable (loss of redundancy) with a Required Action (or, in some cases, an implied Required Action) to restore the affected train to operable status within an appropriate associated Completion Time and (2) a Condition for failure to meet the Required Action and associated Completion Time with Required Actions to place the unit in a Mode in which meeting the LCO is not required. An Actions table of an LCO for a safety system may also include alternative Required Actions, which if completed, would allow unit operation to continue indefinitely in a loss of redundancy Condition. The staff assessed such Required Actions for whether they provide an equivalent level of safety to that provided by meeting the LCO. An Actions table of an LCO for a safety system may also include a Condition for both redundant trains being inoperable (loss of function); in such cases, the staff verified that the Completion Times of the Required Actions for unit shutdown are consistent with the times allowed for unit shutdown by LCO 3.0.3.

For each LCO that requires staying within limits specified for a unit process variable, such as SDM, core reactivity, moderator temperature coefficient, CRA position alignment deviation, core power distribution, RCS pressure and temperatures, UHS water level and bulk average temperature, and UHS boron concentration, the staff verified that the Actions table includes (1) an appropriate Condition for each variable outside its specified limits, with a Required Action to restore the variable to within limits within an appropriate associated Completion Time, and (2) a Condition for failure to meet the Required Action and associated Completion Time with Required Actions to place the unit in a Mode in which meeting the LCO is not required.

It is possible that some LCO subsections have an Actions table with Conditions and Required Actions for a unit status not captured by the kinds of LCOs described above. The following is an example of such a Condition (and the Completion Time to restore redundancy or to complete another remedial action):

- LCO 3.1.7 Condition C. One or more control rod drive mechanisms (CRDMs) with inoperable position indicators have been moved in excess of 6 steps in one direction since the last determination of the CRA's position. (Verify position of affected CRAs by using the MCS [module control system] within 4 hours.)

It is also possible that some LCO subsections include Surveillances for which it is unclear how meeting the Surveillance supports meeting the LCO. In such cases, the Actions table may need to include a Condition for when the Surveillance is not met. Some LCOs specify that a particular subsystem or component be in operation, or in a standby configuration, in addition to being operable; such an LCO subsection may have an Actions table with a Condition addressing when the subsystem or component is not in operation with appropriate Required Actions to restore operation, or other measures to compensate for the out-of-operation system or component, including a unit shutdown. The following is an example of such a Condition (and the Completion Time to restore redundancy or to complete another remedial action):

- LCO 3.1.7 Condition D. CRA CPI [counter position indicator] inoperable for one or more CRAs. (Verify by administrative means all RPIs [rod position indicators] for the affected groups are OPERABLE once per 8 hours, and verify the most withdrawn CRA and least withdrawn CRA of the affected groups are less than or equal to 6 steps apart once per 8 hours.)

The staff also reviewed the Actions table of each LCO to ascertain consistency with the use and application rules of GTS Sections 1.2, 1.3, and 3.0.

16.4.7.1 Conditions for a Loss of Redundancy

The staff reviewed the US460 GTS Chapter 3 Actions Conditions, in SDAA Part 4, that involve a loss of protection from a single active failure (i.e., a loss of system functional redundancy). In most instances, 72 hours are allowed to restore redundancy, which is consistent with STS. These Conditions (and the Completion Time to restore redundancy or to complete another remedial action) are as follows:

- LCO 3.1.7 Condition A. One RPI [rod position indication] per CRDM inoperable for one or more CRDMs. (Verify the position of the CRA with the MCS [Module Control System] once per 8 hours.)
- LCO 3.1.9 Condition A. One CVCS demineralized water isolation valve inoperable. (72 hours to restore DWI valve operability)
- LCO 3.3.1 Condition A. One or more Functions with one channel inoperable. (6 hours to place channel in trip or bypass)

  Condition B. One or more Functions with two channels inoperable. (6 hours to place one inoperable channel in trip and the other inoperable channel in bypass)

•	LCO 3.3.2	Condition A.	One reactor trip breaker (RTB) inoperable. (48 hours to open or restore RTB)
		Condition B.	One division of RTS Logic and Actuation inoperable. (6 hours to restore division to operable status)
•	LCO 3.3.3	Condition A.	LTOP Actuation Function with *one* or both Logic and Actuation divisions inoperable. (Open one or more RVVs within 1 hour.) This Condition addresses both a loss of LTOP actuation function redundancy (denoted by italics) and a complete loss of the LTOP actuation function; since each condition has the same required action, combining them into one Action is appropriate.
		Condition B.	One or more Actuation Functions, other than the LTOP Actuation Function, with one ESFAS Logic and Actuation division inoperable. (Enter the Condition Referenced in Table 3.3.3-1 for affected actuation Function within 6 hours.)
	ECCS, DHRS, SSI	Condition C(1)	As required by Required Action B.1 and referenced in Table 3.3.3-1. (Be in Mode 2 in 6 hours; and be in Mode 3 and Passively Cooled within 36 hours.)
	CIS	Condition D(1)	As required by Required Action B.1 and referenced in Table 3.3.3-1. (Be in Mode 3 with containment isolated within 48 hours.)
	DWSI	Condition E(1)	As required by Required Action B.1 and referenced in Table 3.3.3-1. (Isolate the dilution source flow paths in the CVCS makeup line within 1 hour.)
	CVCSI	Condition F(1)	As required by Required Action B.1 and referenced in Table 3.3.3-1. (Isolate the flow paths between the CVCS and the RCS within 1 hour.)
	PHT	Condition G(1)	As required by Required Action B.1 and referenced in Table 3.3.3-1. (Open pressurizer heater breakers within 6 hours.)

The staff notes that the Actions for one inoperable division of ESFAS Logic and Actuation are the same as the Actions for two inoperable divisions, except that entry into the referenced Condition for the affected ESFAS Logic and Actuation Function is required within 6 hours instead of immediately.

•	LCO 3.3.4	Condition A.	One or more Functions with one manual actuation division inoperable. (Enter the Condition referenced in Table 3.3.4-1 for the affected Function within 48 hours.)
	RTS	Condition C	As required by Required Action A.1 or B.1 and referenced in Table 3.3.4-1. (Immediately open RTBs.)

ECCS, DHRS, SSI	Condition D	As required by Required Action A.1 or B.1 and referenced in Table 3.3.4-1. (Be in Mode 2 within 24 hours; be in Mode 3 and Passively Cooled within 72 hours.)
DWSI	Condition E	As required by Required Action A.1 or B.1 and referenced in Table 3.3.4-1. (Isolate dilution source flow paths in the CVCS makeup line within 1 hour.)
CVCSI	Condition F	As required by Required Action A.1 or B.1 and referenced in Table 3.3.4-1. (Isolate the flow paths between the CVCS and the RCS within 1 hour.)
PHT	Condition G	As required by Required Action A.1 or B.1 and referenced in Table 3.3.4-1. (Open pressurizer heater breakers within 24 hours.)
LTOP	Condition H	As required by Required Action A.1 or B.1 and referenced in Table 3.3.4-1. (Immediately open one or more RVVs.)
CIS	Condition I	As required by Required Action A.1 or B.1 and referenced in Table 3.3.4-1. (Be in Mode 3 with containment isolated within 48 hours.)

The staff notes that the Actions for one inoperable manual actuation division are the same as the Actions for two inoperable manual actuation divisions, except that entry into the referenced Condition for the affected Manual Actuation Function is required within 48 hours for one inoperable manual actuation division (Condition A) instead of 6 hours for two inoperable manual actuation divisions (Condition B). In addition, compared to the Required Action Completion Times of LCO 3.3.3, the corresponding Completion Times for the Manual Actuation Functions are longer. During the staff review of the NuScale US600 DCA, in a letter dated September 14, 2017 (ML17257A450), the applicant provided the rationale for the time differences to achieve similar shutdown conditions in Mode 3 for Containment Isolation and Manual Actuation Functions compared to the other automatic RTS and ESFAS Functions. SER Section 16.4.4.1, “Limiting Condition for Operation Use and Applicability,” under the discussion of LCO 3.0.3, gives the staff’s evaluation of this rationale. The staff determined that this discussion is valid for the NuScale US460 SDAA because the affected systems in the two designs have no differences that would warrant differences in the Actions. Further, SER Section 16.4.7.6, “Shutdown Required Actions and Completion Times,” provides an additional discussion about relaxing shutdown action Completion Times for specified systems of lesser safety significance. These SER sections provide the staff’s rationale for finding the applicant’s proposed action requirements acceptable and concluding that the shutdown action Completion Times are acceptable.

- LCO 3.4.4 Condition A. One RSV inoperable. (Restore to operable status in 72 hours.)
- LCO 3.4.6 Condition A. One or more CVCS flow paths with one CVCS valve inoperable. (Isolate the affected CVCS flow path within

72 hours AND Verify the affected CVCS flow path is isolated once per 31 days.)

- LCO 3.4.7 Condition A. One or more required leakage detection instrumentation methods with one required channel inoperable. (Perform SR 3.4.5.1 (RCS water inventory balance) once per 24 hours AND Restore required channel(s) to operable status within 14 days.)

  Condition B. One required leakage detection instrumentation method with all required channels inoperable. (Restore one channel of affected required leakage detection instrumentation method to operable status within 72 hours.)

- LCO 3.4.10 Condition A. One closed RVV inoperable. (Within 72 hours, either restore RVV to operable status or open one RVV.)

- LCO 3.5.1 Condition A. One RVV or one RRV inoperable. (Within 72 hours, restore valve to operable status.)

- LCO 3.5.2 Condition A. One DHRS loop inoperable. (Within 72 hours, restore DHRS loop to operable status.)

- LCO 3.6.2 Condition A.

  -------------------------NOTE----------------------
  Only applicable to penetration flow paths with two containment isolation valves.

  One or more penetration flow paths with one containment isolation valve inoperable. (Isolate the affected penetration flow path within 72 hours AND Verify the affected penetration flow path is isolated once per 31 days.)

- LCO 3.6.3 Condition A. One or more containment penetrations not closed. (Immediately initiate Action to close penetrations.)

- LCO 3.7.1 Condition A.

  -------------------------NOTE----------------------
  Separate Condition entry is allowed for each valve.

  One or more [main steam isolation] valves inoperable. (Isolate [each] affected [MSIV and MSIV bypass] flow path within 72 hours AND Verify affected flow path is isolated once per 7 days.)

The Actions of Subsection 3.7.1 reflect the NuScale US460 design and are consistent with the main steam system description in the FSAR, the credited function, and the writer’s guide. Each of the two main steamlines contain a safety-related isolation valve and isolation bypass valve in parallel upstream of the module steamline disconnect, and a non-safety-related isolation valve and isolation bypass valve in parallel downstream of the module steamline disconnect. Each valve is considered to be in its own flow path. Isolation of a main steamline requires closure of at least one of the two pairs of isolation valves in the steamline.

The Note to Condition A, allowing the condition to be entered with separate Completion Times of Required Actions A.1 and A.2 being tracked for each affected MSIV and MSIV bypass valve, is appropriate because each valve is considered to be in its own flow path. Omitting the designators “main steam isolation” and “MSIV bypass” is appropriate because the meaning of Condition A is clear; therefore, Condition A is acceptable.

Condition B. Steam line that cannot be isolated.
(Isolate the affected main steam line within 8 hours.)

Only allowing a separate Condition entry for Condition A is appropriate, since Condition B would require a unit shutdown if one valve in each pair of valves in a steamline cannot be isolated automatically, and the associated, affected flow path cannot be isolated “by use of at least one closed and de-activated automatic valve, closed manual valve, or blind flange.” The staff concludes that Conditions A and B are acceptable because they adequately account for (1) having up to all eight steamline isolation valves inoperable, and (2) an inability to isolate a valve in each pair of valves on one or both main steam lines.

- LCO 3.7.2 Condition A. *One* or two FWIVs inoperable. (Isolate the affected FWIV flow path within 72 hours AND Verify FWIV [flow] path [is] isolated once per 7 days.)

  (Actions table Note 1, “Separate Condition entry is allowed for each valve.”)

  Condition B. *One* or two FWRVs inoperable. (Isolate the affected FWRV flow path within 72 hours AND Verify FWRV [flow] path [is] isolated once per 7 days.)

  (Actions table Note 1, “Separate Condition entry is allowed for each valve.”)

- LCO 3.8.1 Condition A. *One required refueling neutron flux channel inoperable. OR Required refueling neutron flux audible count rate channel inoperable*. (Immediately suspend positive reactivity changes and suspend operations that would cause introduction of water into UHS with boron concentration less than specified in the COLR.)

Conclusion for Loss of Redundancy Conditions

The staff finds that the restoration actions and remedial actions, and the associated Completion Times for the loss of redundancy Conditions, are appropriate for the NuScale US460 design, provide an adequate level of safety during operation within each Condition, and are consistent with STS. Therefore, the staff concludes that these Actions are acceptable.

16.4.7.2 Conditions for When a Required Action and Associated Completion Time Are Not Met

The staff reviewed the GTS Chapter 3 Actions Conditions, in SDAA Part 4, that require a unit power reduction in MODE 1 or a unit shutdown from MODE 1 as a default action whenever a Required Action for another entered Condition of the LCO subsection is not met within the associated Completion Time. In a few instances, the initially entered Condition specifies an Action to exit the applicability (e.g., LCO 3.1.4 Required Action A.2). SER Table 16.4.7-1 summarizes the default Required Actions and Completion Times, in SDAA Part 4, to facilitate comparison of times to reach different RCS temperatures in Mode 3, based on the affected systems, parameter limits, and instrumentation functions, and their relative importance to safety.

<u>Actions to Isolate Flow Paths between the CVCS and the RCS and Reactor Coolant Boron Concentration Dilution Source Flow Paths in the CVCS Makeup Line and the Module Heatup System (MHS)</u>

SDAA Part 4 states these flow path isolation requirements as follows:

LCO 3.1.9			-----------------------------------NOTE--------------------------------- Flow paths may be unisolated intermittently under administrative controls. --
	Required Action DWSI	B.1	Isolate dilution source flow paths in the CVCS makeup path by use of at least one closed manual or one closed and de-activated automatic valve. | 1 hour
	CVCS-MHS	B.2	Isolate MHS flow paths to and from cross-connected systems by use of at least one locked, sealed, or otherwise secured valve or device. | 1 hour
LCO 3.3.1	Required Action CVCSI	F.1	-----------------------------NOTE------------------------------- Flow paths may be unisolated intermittently under administrative controls. --- Isolate the flow paths between the CVCS and the Reactor Coolant System by use of at least one closed manual or one closed and de-activated automatic valve. | 6 hours
	DWSI	H.1	Isolate dilution source flow paths in the CVCS makeup line by use of at least one closed manual or one closed and de-activated automatic valve. | 1 hour
	DWSI	L.4	Isolate dilution source flow paths in the CVCS makeup line by use of at least one closed manual or one closed and de-activated automatic valve. | 96 hours
LCO 3.3.3	Required Action DWSI	E.1	-----------------------------NOTE------------------------------ Flow paths may be unisolated intermittently under administrative controls.

			-- Isolate dilution source flow paths in the CVCS makeup line by use of at least one closed manual or one closed and de-activated automatic valve. | 1 hour
	Required Action CVCSI	F.1	-----------------------------NOTE------------------------------ Flow paths may be unisolated intermittently under administrative controls. -- Isolate the affected flow paths between the CVCS and the Reactor Coolant System by use of at least one closed manual or one closed and de-activated automatic valve. | 1 hour
LCO 3.3.4	Required Action DWSI	E.1	-----------------------------NOTE------------------------------ Flow paths may be unisolated intermittently under administrative controls. -- Isolate dilution source flow paths in the CVCS makeup line by use of at least one closed manual or one closed and de-activated automatic valve. | 1 hour
	Required Action CVCSI	F.1	-----------------------------NOTE------------------------------ Flow paths may be unisolated intermittently under administrative controls. -- Isolate the flow paths between the CVCS and the Reactor Coolant System by use of at least one closed manual or one closed and de-activated automatic valve. | 1 hour

The staff finds that the phrasing of the above action statements is consistent among the listed action statements and associated Notes. These remedial actions to isolate CVCS flow paths and the associated Completion Times for the Conditions where automatic and manual Functions for isolating the CVCS flow paths are inoperable are appropriate for the NuScale US460 design, provide an adequate level of safety during operation within each Condition, and are consistent with STS.

The staff notes that LCO 3.1.9, "Boron Dilution Control," Action B, requires isolating the dilution source, which completes the safety function of the DWSI makeup isolation valves. The applicant revised the Applicability of LCO 3.1.9 to state, "MODES 1, 2, and 3 <u>with any dilution source flow path not isolated</u>." The staff finds this is acceptable because with the underlined phrase, the Applicability more accurately captures the conditions requiring the operability of the DWSI makeup isolation valves.

Based on its review and evaluation of Required Actions and associated Completion Times for Conditions in which a Required Action and associated Completion Time are not met, which are summarized in SER Table 16.4.7-1, and because these Actions are appropriate for the NuScale

US460 design and consistent with Actions for equivalent Conditions in the W-STS, the staff concludes that these GTS Actions are acceptable.

Table 16.4.7-1 Default Shutdown Action Completion Times

LCO ACTION	Be in Mode 1 below 20% RTP *Initiate action to close [CNV] penetrations	Be in Mode 1 below 15% RTP (< N-2H) *Initiate action to be in Mode 2	Be in Mode 1 with k_{eff} < 1.0	Open reactor trip breakers **Open all RTBs *Open pressurizer heater breakers	Be in Mode 2 *Open one or more reactor vent valves **Reduce T_{hot} < 260°C (500°F) (< T-4)	Be in Mode 3 and Passively Cooled *Be in Mode 3	Be in Mode 3 with T_{hot} < 93.3°C (200°F) *with T_{cold} < 143.3°C (290°F) (< T-1)	Be in Mode 3 with all RCS temperatures < 93.3°C (200°F) (< T-2)	Be in Mode 3 with T_{hot} < 93.3°C (200°F) (< T-2) *with T_{hot} < 171°C (340°F) (< T-3)	Isolate dilution source flow paths: (1) CVCS makeup line; (2) to - from RCS; or (3) to - from MHS cross connected systems
LCO 3.0.3					7 hours	37 hours				
3.1.2 Action B					6 hours					
3.1.3 Action B					6 hours			48 hours		
3.1.4 Action A					6 hours					
3.1.5 Action B					6 hours					
3.1.6 Action B			6 hours							
3.1.7 Action E					6 hours					
3.1.9 Action B										(1) 1 hour (3) 1 hour
3.2.1 Action A	6 hours									
3.2.2 Action A	6 hours									
3.3.1 Actions C & **D** for RTS Functions 1a,3a,4a,5a, 7a, 9a, 10a, 11a, 13a, 14a, 16a,17a,19a,20a,21a,22a				6 hours						
3.3.1 Actions C & **E** for RTS Function 2a		6 hours								
3.3.1 Actions C & **F** for CVCSI Functions 10c, 11f, 12b, 22f										(2) 6 hours
3.3.1 Actions C & **G** for PHT Functions 7e, 11g, 13e, 17e, 22g				* 6 hours						
3.3.1 Actions C & **H** for DWSI Functions 1b, 2b, 3b, 4b, 5b, 6a, 7d, 8c, 9b, 10b, 11e, 13d, 14b, 15a, 16b, 17d, 19c, 20c, 21c										(1) 1 hour
3.3.1 Actions C & **I** for ECCS Functions 23a, 24a DHRS Functions 7b,13b,17b SSI Functions 7c, 11d, 13c, 17c, 19b, 20b, 21b					6 hours	36 hours				
3.3.1 Actions C & **J** for LTOP Function 27a					* 1 hour					
3.3.1 Actions C & **K** for CIS Function 12a					6 hours				48 hours	
3.3.1 Actions C & **L** for RTS Functions 25a, 26a CIS Functions 25b, 26b DHRS Functions 25c, 26c SSI Function 25d DWSI Functions 25e, 26e CVCSI Functions 25f, 26f PHT Functions 25g, 26g ECCS Function 25h				L.5 *96 hours	L.1 72 hours	L.2 96 hours			L.3 96 hours	L.4 (1) 96 hours

LCO ACTION	Be in Mode 1 below 20% RTP *Initiate action to close [CNV] penetrations	Be in Mode 1 below 15% RTP (< N-2H) *Initiate action to be in Mode 2	Be in Mode 1 with k_{eff} < 1.0	Open reactor trip breakers **Open all RTBs *Open pressurizer heater breakers	Be in Mode 2 *Open one or more reactor vent valves **Reduce T_{hot} < 260°C (500°F) (< T-4)	Be in Mode 3 and Passively Cooled *Be in Mode 3	Be in Mode 3 with T_{hot} < 93.3°C (200°F) *with T_{cold} < 143.3°C (290°F) (< T-1)	Be in Mode 3 with all RCS temperatures < 93.3°C (200°F) (< T-2)	Be in Mode 3 with T_{hot} < 93.3°C (200°F) (< T-2) *with T_{hot} < 171°C (340°F) (< T-3)	Isolate dilution source flow paths: (1) CVCS makeup line; (2) to - from RCS; or (3) to - from MHS cross connected systems
3.3.1 Actions C & **M** for Pressurizer Line Isolation Function 8d CIS Function 22b DHRS Function 22c SSI Function 22d					6 hours				*48 hours	
3.3.1 Actions C & **N** for CIS Function 11b DHRS Function 11c RTS Function 18a SSI Function 18b DWSI Function 18c					** 6 hours					
3.3.2 Action C for RTS Actuation Function				**Immediately						
3.3.3 Actions B & **C** for ECCS Actuation Function 1 DHRS Actuation Function 2 SSI Actuation Function 8					6 hours	36 hours				
3.3.3 Actions B & **D** for CIS Actuation Function 3								48 hours Note 1		
3.3.3 Actions B & **E** for DWSI Actuation Function 4										(1) 1 hour
3.3.3 Actions B & **F** for CVCSI Function 5 Pressurizer Line Isolation Actuation Function 9										(2) 1 hour
3.3.3 Actions B & **G** for PHT Actuation Function 6				*6 hours						
3.3.4 Action **C** for RTS Manual Function 1				Immediately						
3.3.4 Action **D** for ECCS Manual Function 2 DHRS Manual Function 3 SSI Manual Function 9					24 hours	72 hours				
3.3.4 Action **E** for DWSI Manual Function 5										(1) 1 hour
3.3.4 Action **F** for CVCSI Manual Function 6										(2) 1 hour
3.3.4 Action **G** for PHT Manual Function 7				*24 hours						
3.3.4 Action **H** for LTOP Manual Function 8					*Immediately					
3.3.4 Action **I** for CIS Manual Function 4								48 hours Note 1		
3.4.1 Action C					6 hours					
3.4.2 Action A					30 minutes					
3.4.3 Action B					6 hours	36 hours Note 2				
3.4.3 Action D		D.1 *Immediately				36 hours Note 3				

LCO ACTION	Be in Mode 1 below 20% RTP *Initiate action to close [CNV] penetrations	Be in Mode 1 below 15% RTP (< N-2H) *Initiate action to be in Mode 2	Be in Mode 1 with k_{eff} < 1.0	Open reactor trip breakers **Open all RTBs *Open pressurizer heater breakers	Be in Mode 2 *Open one or more reactor vent valves **Reduce T_{hot} < 260°C (500°F) (< T-4)	Be in Mode 3 and Passively Cooled *Be in Mode 3	Be in Mode 3 with T_{hot} < 93.3°C (200°F) *with T_{cold} < 143.3°C (290°F) (< T-1)	Be in Mode 3 with all RCS temperatures < 93.3°C (200°F) (< T-2)	Be in Mode 3 with T_{hot} < 93.3°C (200°F) (< T-2) *with T_{hot} < 171°C (340°F) (< T-3)	Isolate dilution source flow paths: (1) CVCS makeup line; (2) to - from RCS; or (3) to - from MHS cross connected systems
3.4.4 Action B					6 hours		*36 hours Note 4			
3.4.5 Action C					6 hours		48 hours			
3.4.6 Action C					6 hours		48 hours			
3.4.7 Action C					6 hours		48 hours			
3.4.8 Action C					6 hours	*36 hours				
3.4.9 Action B					6 hours	36 hours				
3.5.1 Action B					6 hours	36 hours				
3.5.2 Action B					6 hours	36 hours				
3.5.3 Action C					6 hours	*36 hours				
3.5.4 Action A					24 hours					
3.6.1 Action B					6 hours	48 hours				
3.6.2 Action C					6 hours		48 hours			
3.6.3 Action A	*Immediately									
3.6.4 Action B					6 hours	48 hours				
3.7.1 Action C					6 hours	36 hours				
3.7.2 Action D					6 hours	36 hours				

Note: Use of bold typeface for Actions listed above in the first column for GTS Section 3.3 instrumentation functions denotes which Action is associated with the stated completion time(s). The Section 3.3 Actions tables typically use an Action with a Required Action that directs immediately entering another Action, which is listed in the Condition column for the affected function, in the associated table (e.g., Table 3.3.1-1) that lists the functions included in the LCO subsection.

Note 1: The applicability of CIS and SSI in 3.3.3 and 3.3.4 is MODES 1, 2, and 3. For CIS, 3.3.3 Required Action D.1 and 3.3.4 Required Action I.1 say, "Be in Mode 3 with containment isolated."

Note 2: Required Action B.2 of LCO 3.4.3 says, "Be in MODE 3 with RCS pressure < 500 psia [3,450 kPa (absolute)]."

Note 3: Required Action D.2 of LCO 3.4.3 says, "Be in MODE 3 with RCS temperature less than or equal to the containment flooding RCS temperature limit allowed by the PTLR."

Note 4: Required Action B.2 of LCO 3.4.4 says, "Be in MODE 3 with RCS cold temperature below LTOP enable interlock T-1 temperature."

16.4.7.3 Conditions for a Loss of Function

The staff reviewed the US460 GTS Chapter 3 Actions Conditions, in SDAA Part 4, that involve a loss of function caused by all redundant trains of a system being inoperable, or because fewer than the minimum number of trains needed to perform the system function are operable. In most cases, the associated Required Actions and Completion Times are identical to those listed in SER Table 16.4.7-1 in SER Section 16.4.7.2. In the following quotations of Condition statements, for statements that address both loss of redundancy and loss of function, italics are used to emphasize the portion of the statement corresponding to a loss of function condition. In a few cases, the associated required actions and completion times are summarized in parentheses, as well as in SER Table 16.4.7-1.

A. One or more of a set of nonredundant but identical components inoperable

- LCO 3.1.4 Condition A(1). One *or more* CRAs inoperable. (Verify SDM to be within COLR limits or initiate boration within 1 hour and be in Mode 2 in 6 hours.)

- LCO 3.4.9 Condition A. One *or more* SG tubes satisfying the tube plugging criteria and not plugged in accordance with the Steam Generator Program. (Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection within 7 days, and plug the affected tube(s) in accordance with the Steam Generator Program prior to entering MODE 3 following the next refueling outage or SG tube inspection.)

  Condition B(2). SG tube integrity not maintained. (SER Table 16.4.7-1 gives the associated Required Actions and Completion Times.)

B. Redundant trains of a two-train system inoperable, or more than two of four channels inoperable, two of two divisions inoperable, or the single train of a one-train system is inoperable

- LCO 3.1.7 Condition B. *More than one* RPI per CRDM inoperable. (Place affected CRA under manual control immediately; verify position of affected CRA using in-core neutron detectors once per 8 hours; and restore at least one rod position indicator on each CRA within 24 hours.)

- LCO 3.1.9 Condition B(2). *Two* CVCS demineralized water isolation valves inoperable. (Isolate dilution source flow paths by use of at least one closed manual or one closed and de-activated automatic valve within 1 hour.)

- LCO 3.3.1 Condition C(2). One or more Functions with *three or more channels inoperable*. (Immediately enter Condition referenced in Table 3.3.1-1 for the channel(s). SER Table 16.4.7-1 gives the associated Required Actions and Completion Times.)

- LCO 3.3.2 Condition C(2). Both divisions of RTS Logic and Actuation inoperable. (Immediately open all RTBs.)

  Condition C(3). More than one RTB inoperable. (Immediately open all RTBs.)

- LCO 3.3.3 Condition A. LTOP actuation Function with one *or both* Logic and Actuation divisions inoperable. (Open one or more RVVs within 1 hour.)

The staff verified that FSAR Section 3.9.1.1.2, “Service Level B Conditions,” describes the cold overpressure protection transient under the heading “Service Level B Transient 11—Cold Overpressure Protection.” FSAR Table 3.9-1 also lists this transient. In addition, FSAR Section 5.2.2.2.2, “Low Temperature Overpressure Protection System,” states the following:

> Selection of the LTOP setpoint considers the worst case low temperature overpressure transient, which is the spurious actuation of the PZR

heaters while below the LTOP enabling temperature. The LTOP … analysis results indicate the peak pressure remains below the brittle fracture stress limit.

FSAR Section 5.2.2.4.2, “Reactor Vent Valves,” states the following:

> Assuming a single active component failure, the RVVs, associated actuators, and controls maintain the LTOP function. The RVVs have sufficient pressure relief capacity to accommodate the most limiting single active failure assuming the most limiting allowable operating condition and system configuration.

The staff concludes from the above passages that the automatic opening of one of the two RVVs at the LTOP actuation pressurizer pressure setting (which is a function of RCS cold temperature) will limit RCS peak pressure to below the low temperature-pressure limit for the limiting RCS cold overpressurization event. Since the CNV is designed to accommodate the pressure transient of this limiting event, the action requirement for manually opening one or more RVVs when the LTOP automatic actuation capability is inoperable is acceptable.

- LCO 3.3.3 Condition C(2). Both divisions of ECCS Actuation Function inoperable. (Be in Mode 2 in 6 hours and Mode 3 and Passively Cooled within 36 hours.)

  Condition C(3). Both divisions of DHRS Actuation Function inoperable. (Be in Mode 2 in 6 hours and Mode 3 and Passively Cooled within 36 hours.)

  Condition C(4). Both divisions of SSI Actuation Function inoperable. (Be in Mode 2 in 6 hours and Mode 3 and Passively Cooled within 36 hours.)

  Condition D(2). Both divisions of Containment Isolation Actuation Function inoperable. (Be in Mode 3 with containment isolated within 48 hours.)

  Condition E(2). Both divisions of Demineralized Water Supply Isolation Actuation Function inoperable. (Isolate the dilution source flow path in CVCS makeup line within 1 hour.)

  Condition F(2). Both divisions of CVCS Isolation Actuation Function inoperable. (Isolate the flow paths between the CVCS and the RCS within 1 hour.)

  Condition F(3). Both divisions of Pressurizer Line Isolation Function inoperable. (Isolate the flow paths between the CVCS and the RCS within 1 hour.)

  Condition G(2). Both divisions of Pressurizer Heater Trip Actuation Function inoperable. (Open pressurizer heater breakers within 6 hours.)

- LCO 3.3.4 Condition B. One or more Functions with two manual actuation divisions inoperable. (Enter the Condition referenced in Table 3.3.4-1 for the affected Function within 6 hours.)

The staff notes that the Actions for two inoperable manual actuation divisions are the same as the Actions for one inoperable manual actuation division, except that entry into the referenced Condition for the affected manual actuation function is required within 6 hours instead of 48 hours.

- LCO 3.4.4 Condition B(2). Two RSVs inoperable. (Be in Mode 2 in 6 hours, and be in Mode 3 with RCS cold temperature below LTOP enable interlock T-1 temperature in 36 hours.)

- LCO 3.4.6 Condition B. One or more CVCS flow paths with two CVCS valves inoperable. (Isolate the affected CVCS flow path within 1 hour.)

- LCO 3.4.7 Condition C(2). Two required leakage detection instrumentation methods with all required channels inoperable. (Be in Mode 2 in 6 hours, and be in Mode 3 with RCS hot temperature less than 93°C (200°F) in 48 hours.)

- LCO 3.4.10 Condition B. Two closed RVVs inoperable. (Within 2 hours initiate action to depressurize RCS and open one RVV.)

SER Section 16.4.5 provides an additional discussion of GTS Subsection 3.4.10.

- LCO 3.5.1 Condition B(2). Two RVVs inoperable. (Be in Mode 2 within 6 hours and in Mode 3 and Passively Cooled within 36 hours.)

  Condition B(3). Two RRVs inoperable. (Be in Mode 2 within 6 hours and in Mode 3 and Passively Cooled within 36 hours.)

- LCO 3.5.2 Condition B(2). Both DHRS loops inoperable. (Be in Mode 2 within 6 hours and in Mode 3 and Passively Cooled within 36 hours.)

- LCO 3.5.4 Condition A(1). ESB [ECCS Supplemental Boron] operational limits specified in the COLR not met. (Be in Mode 2 within 24 hours.)

  Condition A(2). ESB inoperable for any other reason. (Be in Mode 2 within 24 hours.)

- LCO 3.6.1 Condition A. Containment inoperable. (Restore containment to OPERABLE status within 1 hour.)

- LCO 3.6.2 Condition B. One or more penetration flow paths with two containment isolation valves inoperable. (Isolate the affected penetration flow path within 1 hour.)

- LCO 3.6.4 Condition A. PAR inoperable. (Restore PAR to OPERABLE status within 72 hours.)

- LCO 3.7.1 Condition A. -------------------------NOTE---------------------
Separate Condition entry is allowed for each valve.

One or more valves inoperable.
(72 hours to isolate affected [main steam line] flow path and verify flow path isolated once per 7 days.)

Condition B. Steamline that cannot be isolated. (Isolate affected main steam line within 8 hours.)

- LCO 3.7.2 Condition A. One or two FWIVs inoperable. (Within 72 hours isolate the affected FWIV flow path, and verify FWIV [flow] path isolated once per 7 days.) (Note, separate Condition entry is allowed for each valve.)

Condition B. One or two FWRVs inoperable. (Within 72 hours isolate affected FWRV flow path, and verify FWRV [flow] path isolated once per 7 days.) (Note, separate Condition entry is allowed for each valve.)

Note that, as long as one isolation valve in each feedwater flow path remains operable for closing, the situation is a loss of redundancy, not a loss of function. SER Section 10.4 provides the staff's evaluation of the suitability of the FWRVs to function as backup FWIVs.

Condition C. Two valves in the same flow path inoperable. (Within 8 hours isolate the affected [feedwater (FW)] flow path or, as explained in GTS Section 1.3, restore one of the affected valves to operable status.)

- LCO 3.8.1 Condition B. Two required refueling neutron flux channels inoperable. (Immediately initiate action to restore one channel to operable status and verify UHS bulk average boron concentration is within limits once per 12 hours.)

Based on its review and evaluation of Required Actions and associated Completion Times for Conditions in which a loss of function exists, and because these Actions are appropriate for the NuScale US460 design and consistent with Actions for equivalent Conditions in the W-STS, the staff concludes that these GTS Actions are acceptable.

16.4.7.4 Actions Notes Allowing Separate Condition Entry

Whenever a system contains two or more identical subsystems, which function independently of each other, and the system's LCO Actions table includes a Note allowing separate Condition entry for each subsystem, the control room operator may track a separate Completion Time for each subsystem, if they are concurrently inoperable. The Actions table Note defines the basis for separate Condition entry. Guidance for applying such an Actions table Note is provided in Section 1.3 by Example 1.3-5. The GTS includes the following Actions table Notes.

- LCO 3.1.7 Separate Condition entry is allowed for each CRDS rod position indicator and each CRA counter position indicator.
- LCO 3.3.1 (Note 1) Separate Condition entry is allowed for each [MPS instrumentation] Function.

  (Note 2) Separate Condition entry is allowed for each steam generator for Functions 17, 18, 19, 20 and 21.

  (Note 3) Separate Condition entry is allowed for each EDAS battery charger of Function 25.
- LCO 3.3.3 Separate Condition entry is allowed for each [ESFAS Logic and Actuation] Function.
- LCO 3.3.4 Separate Condition entry is allowed for each [RTS and ESF Manual Actuation] Function.
- LCO 3.4.6 (Note 2) Separate Condition entry is allowed for each [CVCS] flow path.
- LCO 3.4.7 Separate Condition entry is allowed for each [containment evacuation system (CES)] condensate channel and each [CES inlet] pressure channel.
- LCO 3.4.9 Separate Condition entry is allowed for each SG tube.
- LCO 3.5.1 (Condition A) Separate Condition entry is allowed for each [reactor vent valve and each reactor recirculation] valve.
- LCO 3.6.2 (Note 2) Separate Condition entry is allowed for each [containment vessel] penetration flow path.
- LCO 3.7.1 (Condition A) Separate Condition entry is allowed for each [main steam isolation valve (MSIV) and each MSIV bypass] valve.
- LCO 3.7.2 (Note 1) Separate Condition entry is allowed for each [feedwater isolation valve (FWIV) and each feedwater regulation] valve (FWRV)].

The staff finds that the above Actions Notes are appropriate because the basis for each separate Condition entry involves a set of independent components, Functions, valves, flow paths, or channels consistent with the W-STS. Therefore, the staff concludes that these Actions Notes that allow separate Condition entry are acceptable.

16.4.7.5 Conditions for Process Variable Outside Limits

When an LCO limit on a process variable is not met, the Actions specify a Completion Time to restore the variable to within limits. Such Actions are provided for the following Conditions, and the Completion Time to restore the variable within limits, or other remedial action, is also provided:

- LCO 3.1.1 Condition A. SDM not within limits. (15 minutes)
- LCO 3.1.2 Condition A. Core reactivity balance not within limit. (7 days)
- LCO 3.1.3 Condition A. MTC [Moderator Temperature Coefficient] not within limits. (Be in Mode 2 within 6 hours.)

  Condition B. MTC not within lower limit. (Be in MODE 3 with all RCS temperatures < 93.3°C (200°F) within 48 hours.)
- LCO 3.1.4 Condition A(2). One or more CRAs not within alignment limits. (Verify SDM within limits or initiate boration within 1 hour, and be in Mode 2 within 6 hours.)
- LCO 3.1.5 Condition A. One or more shutdown [bank] groups not within insertion limits. (Verify SDM is within the limits specified in the COLR or initiate boration to restore SDM to within limit within 1 hour, and restore shutdown [bank] groups to within limits within 2 hours.)
- LCO 3.1.6 Condition A. One or more regulating [bank] groups not within insertion limits. (Verify SDM is within the limits specified in the COLR or initiate boration to restore SDM to within limits within 1 hour, and restore regulating [bank] groups to within limits within 2 hours.)
- LCO 3.1.8 Condition A. SDM not within limit. (Initiate boration to restore SDM to within limit within 15 minutes and suspend Physics Tests exceptions within 1 hour.)

  Condition B. THERMAL POWER not within limit. (Open reactor trip breakers immediately.)
- LCO 3.1.9 Condition B(3). Boric Acid supply boron concentration not within limits. (Isolate dilution source flow paths in the CVCS makeup path within 1 hour.)

  Condition B(4). CVCS makeup pump demineralized water flow path not configured to ensure maximum flowrate is within limits. (Isolate dilution source flow paths in the CVCS makeup path within 1 hour.)

  Condition B(5). MHS [module heatup system] flow paths to or from cross-connected systems not isolated by a locked, sealed, or otherwise secured valve or device. (Isolate MHS flow paths to and from cross connected systems by use of at least one locked, sealed, or otherwise secured valve or device within 1 hour.)
- LCO 3.2.1 Condition A. $F_{\Delta H}$ [enthalpy rise hot channel factor] not within limit. (Reduce Thermal Power to < 20% RTP within 6 hours.)

- LCO 3.2.2 Condition A. AO [AXIAL OFFSET] not within limits. (Reduce Thermal Power to < 20% RTP within 6 hours.)

- LCO 3.4.1 Condition A. RCS pressurizer pressure or RCS cold temperature CHF [Critical Heat Flux] parameters not within limits. (Restore RCS CHF parameter(s) to within limit within 2 hours.)

  Condition B. RCS flow resistance not within limits. (Evaluate flow resistance effect on safety analysis and verify that the reactor coolant system flow rate is acceptable for continued operation within 7 days.)

- LCO 3.4.2 Condition A. One or more RCS temperatures not within [minimum temperature for criticality] limit [of ≥ 174°C (345°F)]. (Be in Mode 2 within 30 minutes.)

- LCO 3.4.3 Condition A. Requirements of LCO [for PTLR limits on RCS pressure, temperature, and heatup and cooldown rates] not met in MODE 1, 2, or 3. (Restore parameters to within limits in 30 minutes, and determine RCS is acceptable for continued operation within 72 hours.)

  Condition C. Requirements of LCO not met any time in other than MODE 1, 2, or 3. (Immediately initiate action to restore parameter(s) to within limits, and determine RCS is acceptable for continued operation prior to entering MODE 3 [from MODE 4].)

  Condition D. Containment flooding initiated while RCS temperature greater than allowed by PTLR. (Initiate action to be in MODE 2 immediately; be in MODE 3 with RCS temperature at or below the PTLR limit within 36 hours; and determine RCS is acceptable for continued operation prior to entering MODE 2 from MODE 3.)

- LCO 3.4.5 Condition A. Pressure boundary LEAKAGE exists. [See SER Table 16.4.7-1] (Isolate affected component, pipe, or vessel from the RCS within 4 hours.)

  Condition B. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE. (Reduce LEAKAGE to within limits within 4 hours.)

  Condition C(2). Primary to secondary LEAKAGE not within limit. (Be in Mode 2 within 6 hours, and Mode 3 with RCS hot temperature less than 93°C (200°F) within 48 hours. [See SER Table 16.4.7-1]

These Conditions and the Required Actions and Completion Times of these Conditions are consistent with the W-STS and the W-AP1000-STS. Therefore, the Actions of LCO 3.4.5 are acceptable.

- LCO 3.4.8 Condition A. DOSE EQUIVALENT I-131 > 0.058 μCi/gm [microcuries per gram (2.146 kilobecquerels per gram (kBq/gm))]. (Verify DOSE EQUIVALENT I-131 ≤ 3.5 μCi/gm (129.5 kBq/gm) once per 4 hours and restore DOSE EQUIVALENT I-131 to within limit within 48 hours.)

  Condition B. DOSE EQUIVALENT Xe-133 > 16 μCi/gm [592 kBq/gm]. (Restore DOSE EQUIVALENT Xe-133 to within limit in 48 hours.)

  Condition C. DOSE EQUIVALENT I-131 > 3.5 μCi/gm [129.5 kBq/gm]. (Be in MODE 2 in 6 hours and MODE 3 in 36 hours.) [See SER Table 16.4.7-1]

  These Conditions and the Required Actions and Completion Times for these Conditions are consistent with the W-STS and the W-AP1000-STS. Therefore, the Actions of LCO 3.4.8 are acceptable. SER Section 16.4.2.5 and SER Section 12.2.4.2 provide additional discussion of the basis for the LCO 3.4.8 specific activity limits.

- LCO 3.5.3 Condition A. Ultimate Heat Sink level not within limits. [≥ 15.84 m and ≤ 16.46 m (≥ 52 ft and ≤ 54 ft)]. (Immediately suspend module movements, suspend movement of irradiated fuel assemblies in the refueling area, and initiate action to restore Ultimate Heat Sink level to within limits, and within 24 hours restore level to within limits.)

  Condition B. Ultimate Heat Sink bulk average temperature not within limits. (Immediately suspend module movements and initiate action to restore Ultimate Heat Sink bulk average temperature to within limits [(≥ 18°C and ≤ 49°C (≥ 65°F and ≤ 120°F)], and within 14 days restore Ultimate Heat Sink bulk average temperature to within limits.)

  Condition D. Ultimate Heat Sink bulk average boron concentration not within limits. (Immediately initiate action to restore concentration to within [COLR] limits and other remedial actions.)

- LCO 3.8.2 Condition A. Reactor subcritical for < 48 hours. (Immediately suspend movement of irradiated fuel in the RPV.)

Based on its review and evaluation of Required Actions and associated Completion Times for Conditions where a process variable is outside limits, and because these Actions are appropriate for the NuScale US460 design and consistent with Actions for equivalent Conditions in the W-STS, the staff concludes that the GTS Actions are acceptable.

16.4.7.6 Shutdown Required Actions and Completion Times

As previously discussed in the Section 16.4.4 evaluation of GTS LCO 3.0.3 shutdown Completion Times, and the Section 16.4.6 evaluation of GTS Section 1.1 definitions of operational modes, the times allowed by shutdown action requirements to place the unit in Mode 2 and Mode 3 appeared to be inconsistent.

As part of the NuScale US600 DCA review, by letter dated September 14, 2017 (ML17257A450), NuScale indicated that GTS Required Action Completion Times are based on consideration of the following:

- the NPM design
- operational processes required to perform the associated evolutions
- operating experience of legacy nuclear power plants
- the relative significance of the affected safety function and the availability of alternative means to compensate for a reduced or lost capability to perform the safety function
- industry standard Completion Times reflected in STS
- the reliability and capability of remaining (i.e., redundant) operable specified SSCs to perform required safety functions
- the low probability of a DBA occurring with the LCO not met during the specified Required Action Completion Time
- the time needed to perform the Required Action, including reaching the prescribed plant conditions, collecting data, completing evaluations, and performing surveillances
- the urgency of exiting the emergent plant conditions

In general, the NuScale GTS Conditions and associated Required Actions and Completion Times are comparable to the action requirements in the STS. In particular, LCO 3.0.3 specifies a 1-hour period to begin initiating action for placing the unit in Mode 2 within 7 hours, and in Mode 3 and passively cooled within 37 hours of entry into LCO 3.0.3. Individual LCO Actions tables specify similar shutdown requirements but without the 1-hour period. The standard Completion Times are 6 hours to Mode 2, and 36 hours to Mode 3 and passively cooled. Longer times are provided for systems such as Manual Actuation Functions of LCO 3.3.4. Also, a longer time of 48 hours is specified to reach the final state in Mode 3 for inoperable CIS instrumentation Functions, and CIS logic and actuation Functions, because the RCS temperature must be taken below the wide range RCS hot temperature interlock, T-2, which is active below 93.3°C (200°F). In contrast, the standard end state in Mode 3 with passive cooling in operation may be reached at a much higher RCS temperature, with the DHRS in operation.

The applicant modified the Bases for LCO 3.0.3 by adding a paragraph describing the reasons the shutdown sequence Completion Times are appropriate. The staff finds the explanation acceptable, as described in SER Section 16.4.4.1.

Based on its review and the explanation and additional justification provided by the applicant, the staff concludes that the Completion Times for the shutdown actions in the US460 SDAA

GTS are also appropriate, consistent with the STS, as well as the US600 GTS, and therefore acceptable.

Conclusion for Action Requirements

Based on its review and the above evaluation, the staff concludes that the GTS action requirements are acceptable.

16.4.8 Surveillance Requirements (GTS Chapter 3, Sections 3.1 through 3.8)

The staff reviewed the SRs specified for each LCO subsection to verify they satisfy 10 CFR 50.36(c)(3), which states that SRs "are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions for operation will be met." For each LCO on an SSC or parameter, the associated SRs verify the capability of the SSC to perform its specified safety function, or that the parameter is within specified limits. The choice of an SR frequency for an SSC considers past precedents, the operating and test history for similar TS-required SSCs, and recommendations of the SSC's manufacturer. Based on consistency with surveillance test intervals for similar SSCs specified in plant-specific TS of currently operating reactors, the staff believes the proposed surveillance frequencies provide reasonable assurance that the reliability and availability of GTS-required SSCs assumed in the NuScale design's PRA are appropriate. The staff reviewed the basis for each SR frequency provided in the Bases of each GTS Chapter 3 LCO subsection. Table 16.1-1 of FSAR Section 16.1, "Technical Specifications," provides the Bases for the initial or base frequency of each SR for which the frequency is specified to be governed by Subsection 5.5.11, "Surveillance Frequency Control Program (SFCP)." SER Table 16.4.8-1 lists surveillances with frequencies governed not by the SFCP but according to the inservice testing program, the containment leakage rate testing program, or the steam generator program, or by an event-driven surveillance performance requirement.

16.4.8.1 Surveillance Statements

The staff verified that the SRs of each LCO are adequate to ensure the LCO is being met. The proposed surveillances are phrased in a manner consistent with the phrasing of equivalent kinds of STS SRs involving SSC performance tests, inspections, and status checks and verification that the unit is operating within the specified limits of selected process variables. Surveillances that are unique to the NuScale US460 design are quoted below (surveillance frequency is indicated in parentheses):

- SR 3.1.9.1 Verify that CVCS makeup pump demineralized water flow path is configured to ensure that the maximum demineralized water flowrate remains within the limits specified in the COLR. (SFCP/31 days)

  SR 3.1.9.4 Verify each CVCS makeup pump maximum flowrate is ≤ 1.58 L/s [liters per second] (25 gpm [gallons per minute]). (SFCP/24 months)

  SR 3.1.9.5 -----------------------------------NOTE[S]--------------------------------------
  1. Only required to be met when CVCS flow is aligned through the module heatup system (MHS).

2. Isolation devices that are locked, sealed, or otherwise secured may be verified by use of administrative means.

Verify MHS flow paths to and from cross-connected systems are isolated by a locked, sealed, or otherwise secured valve or device. (SFCP/12 hours)

Surveillances 3.1.9.1 and 3.1.9.4 verify that, when thermal power is below the limit specified in the COLR, the two makeup pumps are aligned so that just one pump can supply the RCS through the CVCS injection line to satisfy initial conditions of the inadvertent RCS boron dilution event analysis. SR 3.1.9.5 verifies MHS flow paths are isolated because CVCS flow between units via the MHS headers could result in unplanned changes to the boration of a unit aligned to the MHS.

- SR 3.3.1.3 Verify CHANNEL RESPONSE TIME is within limits. The CHANNEL RESPONSE TIME is combined with the allocated MPS digital time response and the ACTUATION RESPONSE TIME to determine and verify the TOTAL RESPONSE TIME. (SFCP/24 months)

SER Sections 16.4.2.3 and 16.4.2.4 discuss response time testing and the associated defined terms and definitions.

- SR 3.3.1.5 Perform CHANNEL CALIBRATION on each required Class 1E isolation device. (SFCP/24 months)

Instrumentation Subsections 3.3.1, 3.3.2, and 3.3.3 include this Surveillance to ensure that the MPS is protected from an electrical fault in the non-safety-related electrical power system by isolating the MPS on overcurrent or undervoltage.

- SR 3.3.2.1 Perform ACTUATION LOGIC TEST. (SFCP/24 months)

The definition of this Surveillance in Section 1.1 differs from the W-AP1000-STS definition of Actuation Logic Test. SER Sections 16.4.2.2 and 16.4.8.4 provide additional discussion of this definition.

- SR 3.3.2.2 Verify ACTUATION RESPONSE TIME is within limits. The ACTUATION RESPONSE TIME is combined with the allocated MPS digital time response and the CHANNEL RESPONSE TIME to determine and verify the TOTAL RESPONSE TIME. (SFCP/24 months)

SER Section 16.4.2.4 discusses response time testing and the associated defined terms and definitions.

- SR 3.3.2.3 Perform CHANNEL CALIBRATION on each Class 1E isolation device. (SFCP/24 months)

- SR 3.3.3.1 Perform ACTUATION LOGIC TEST. (SFCP/24 months)

- SR 3.3.3.2 Verify pressurizer heater breaker ACTUATION RESPONSE TIME is within limits. The ACTUATION RESPONSE TIME is combined with the allocated MPS digital time response and the CHANNEL RESPONSE TIME to determine and verify the TOTAL RESPONSE TIME. (SFCP/24 months)

- SR 3.3.3.3 Verify ECCS actuation time delay is within limits. (SFCP/24 months)

- SR 3.3.3.4 Perform CHANNEL CALIBRATION on each Class 1E isolation device. (SFCP/24 months)

- SR 3.3.3.5 Verify each pressurizer heater breaker actuates to the open position on an actual or simulated actuation signal. (SFCP/24 months)

This Surveillance verifies that the pressurizer heater breakers will open if the MPS detects reactor conditions that could lead to the uncovering of the heaters. The ESFAS PHT Function is designed to protect the pressurizer heaters from uncovering, overheating, and potentially compromising the RCS pressure boundary. SER Section 16.4.1, Table 16.4.1-3, “ESFAS Logic and Actuation Functions,” lists the MPS Functions that initiate a PHT.

- SR 3.3.4.1 Perform actuation device operational test. (SFCP/24 months)

This Surveillance exercises manual switches that actuate the two divisions of RTBs and ESF-actuated valves and pressurizer heater breakers. It is equivalent to the manual actuation testing part of the TRIP ACTUATING DEVICE OPERATIONAL TEST (TADOT) included in W-AP000-STS; but NuScale US460 GTS do not include this defined term. SER Section 16.4.2.5 provides the staff’s reasoning for finding that omitting the W-STS TADOT definition from the GTS is acceptable.

- SR 3.4.1.3 Verify RCS flow resistance is within the limits specified in the COLR. (Once prior to exceeding 75% RTP after each refueling)

This Surveillance verifies that the flow resistance of the reactor coolant flow paths inside the reactor vessel is still within limits following operation in Mode 5 with module disassembly for refueling and subsequent reassembly. The 12-hour surveillance to verify RCS flow is above a lower limit, which is specified in the TS of PWRs using forced circulation, is not appropriate for the NuScale US460 design because reactor coolant flow is by natural circulation. Reactor coolant flow in the NuScale US460 design is caused by coolant density differences across the core and the SG tubes and is a function of core thermal power (reactor coolant mass flow rate is proportional to the cube root of core thermal power).

- SR 3.4.6.1 Verify [required] valves accumulator pressures are within limits. (SFCP/12 hours)

This Surveillance ensures that valves necessary to initiate the ESF Function of CVCSI have sufficient accumulator pressure to actuate to the required position. The applicant proposed placing “required” in brackets as shown above, to account for a COL applicant using a different type of actuator on a CVCS automatic isolation valve. The term “required” in valve actuator accumulator pressure surveillance requirements and the associated bases were modified by placing the terms in square brackets. This indicates that the content is required to be completed by a COL applicant when final plant-specific information is finalized. FSAR Chapter 16 describes the requirement to provide final plant-specific information identified by square brackets as COL Item 16.1-1.

- SR 3.5.1.3 Verify the inadvertent actuation block setpoints are within limits, and the inadvertent actuation block function of each RRV. (INSERVICE TESTING PROGRAM)

  This Surveillance ensures that the mechanical block of the two RRVs will prevent opening of these two valves upon an ECCS ESFAS signal until the pressure difference between the RCS and the CNV is below the specified setting. This block also prevents inadvertent opening of the RRVs in Modes 1, 2, and 3.

- SR 3.5.2.1 Verify [required] valves accumulator pressures within limits. (SFCP/12 hours)

  This Surveillance ensures that valves necessary to initiate the ESF Function of the DHRS have sufficient accumulator pressure to actuate to the required position. The applicant proposed placing “required” in brackets as shown above, to account for a COL applicant using a different type of actuator on a DHRS automatic actuation valve.

- SR 3.5.2.2 Verify DHRS heat exchangers are filled. (SFCP/24 hours)

  This SR is appropriate because, with the DHRS in standby, the DHRS level sensors monitor the level in the heat exchanger inlet piping downstream of the closed DHRS actuation valves.

- SR 3.5.2.3 -------------------------------------NOTE--------------------------------------
  Not required to be performed for DHRS loop with associated FWIV open.

  Verify SG level is > 5% and ≤ 50%. (SFCP/12 hours)

  This SR is appropriate to assure the SG contains inventory adequate to support actuation and operability of the associated DHRS loop when the SG’s associated FWIV is closed. When the associated FWIV is open, normal FWS controls will ensure that the SG will support DHRS loop actuation and operability.

- SR 3.5.3.3 Verify the Ultimate Heat Sink bulk average boron concentration is within limits. (SFCP/31 days)

  This Surveillance reflects the multiple Functions of the NuScale US460 reactor pool; besides serving as the UHS for the decay heat of the NPMs following shutdown, the reactor pool also provides reactivity control during refueling.

- SR 3.5.4.1 Verify the form and quantity of boron in the ESB dissolvers are within the limits specified in the COLR. (Once prior to entering

MODE 1 after operations that could affect the form or quantity of boron in the ESB dissolvers)

This Surveillance verifies that the form and quantity of boron in the ESB dissolvers are within limits specified in the COLR. The specified minimum quantity of boron ensures that a sufficient mass of boron will be available for dissolution by ECCS condensate flow to maintain the required SDM if the ECCS is actuated. The specified maximum quantity of boron ensures no boron precipitation will occur during the extended passive cooling period. The Frequency requires verification after operations, which are specified in the COLR, that could affect the form or quantity of boron available in the ESB dissolvers. The Frequency is appropriate because ESB dissolvers are physically isolated from changes by their location in the CNV. This ensures they are unaffected during normal operations and the form and quantity of boron will remain unchanged once established.

- SR 3.6.2.1 Verify [required] valves accumulator pressures are within limits. (SFCP/12 hours)

  The applicant proposed placing "required" in brackets as shown above, to account for a COL applicant using a different type of actuator on a CIV.

- SR 3.6.3.1 Verify each containment penetration is closed. (SFCP/7 days)

- SR 3.7.1.1 Verify [required] valves accumulator pressures are within limits. (SFCP/12 hours)

  The applicant proposed placing "required" in brackets as shown above, to account for a COL applicant using a different type of actuator on an MSIV and main steamline bypass isolation valve.

- SR 3.7.2.1 Verify [required] FWIV accumulator pressures are within limits. (SFCP/12 hours)

  SR 3.7.1.1 and SR 3.7.2.1 ensure that MSIVs and FWIVs, which are necessary to initiate the ESF Functions of CIS and DHRS, have sufficient accumulator pressure to actuate to the required position. The applicant proposed placing "required" in brackets as shown above, to account for a COL applicant using a different type of actuator on an FWIV.

- SR 3.8.1.2 Perform CHANNEL CALIBRATION. (SFCP/24 months)

Based on its review, the staff determined that the above NuScale US460-specific Surveillance statements clearly describe the Surveillances, are appropriate for assuring the associated LCOs are met, and are consistent with the NuScale US460 design and the W-STS. Therefore, the staff finds that these Surveillance statements are acceptable.

16.4.8.2 Surveillance Frequencies Not Governed by the Surveillance Frequency Control Program

The staff reviewed the Surveillances with a performance Frequency that is contingent on (1) having exceeded a specified thermal power level, (2) not having exceeded a specified thermal power level, (3) having exceeded a specified fuel expenditure expressed as a number of EFPD, (4) not having exceeded a specified number of EFPD, (5) having entered a specified mode or other specified condition, (6) not having entered a specified mode or other specified condition, (7) having completed a specified task, (8) not having completed a specified task, (9) a specified event having occurred, (10) a specified event not having occurred, or (11) a specified time interval having elapsed or not elapsed. In addition, there are Surveillances governed by other requirements, such as the steam generator program and the inservice testing program. These Frequencies are stated below in Table 16.4.8-1 and may not be changed in accordance with the SFCP.

Table 16.4.8-1[7] Surveillance Frequencies Not Governed by the Surveillance Frequency Control Program

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify overall core reactivity balance is within ±1% Δk/k of predicted values.	Once prior to exceeding 5% RTP after each refueling AND In accordance with the SFCP
SR 3.1.3.1	Verify moderator temperature coefficient (MTC) is within the upper limit.	Once prior to exceeding 5% RTP after each refueling
SR 3.1.3.2	Verify MTC is within the lower limit.	Once within 7 EFPD after reaching 40 EFPD fuel burnup from beginning of cycle (BOC) AND Once within 7 EFPD after reaching 2/3 fuel burnup from BOC AND ----------------------NOTE------------------------ Only required when projected end of cycle MTC is not within limit. --- 7 EFPD thereafter
SR 3.1.4.3	Verify each control rod assembly (CRA) drop time is ≤ 2.2 seconds.	Prior to reactor criticality after each removal of the upper reactor pressure vessel section
SR 3.1.7.1	Verify each RPI channel agrees within 6 steps of the group counter position indication for the full indicated range of CRA travel.	Prior to criticality after coupling a CRA to the associated CRDM for one or more CRAs
SR 3.2.1.1	Verify Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$) is within the limits specified in the COLR.	Once after each refueling prior to THERMAL POWER exceeding 20% RTP AND In accordance with the SFCP

7 In this table, material is added to define acronyms in the quoted passage.

	SURVEILLANCE	FREQUENCY
SR 3.4.1.3	-------------------NOTE----------------- Not required to be performed until 96 hours after exceeding 50% RTP. -- Verify RCS flow resistance is within the limits specified in the COLR.	Once prior to exceeding 75% RTP after each refueling
SR 3.4.4.1	Verify each [reactor safety valve (RSV)] is OPERABLE in accordance with the INSERVICE TESTING PROGRAM. Following testing, lift settings shall be within 1% of the nominal setpoints of 2200 psid [15.168 MPa (differential)] and 2290 psid [15.789 MPa (differential)] as shown below: Valve 1 Setpoint: ≥ 2178 psid and ≤ 2222 psid [≥ 15.017 MPa (differential) and ≤ 15.320 MPa (differential)]. Valve 2 Setpoint: ≥ 2268 psid and ≤ 2312 psid [≥ 15.637 MPa (differential) and ≤ 15.941 MPa (differential)].	In accordance with the INSERVICE TESTING PROGRAM
SR 3.4.6.2	Verify the isolation ACTUATION RESPONSE TIME of each automatic power operated CVCS valve is within limits. The ACTUATION RESPONSE TIME is combined with the allocated MPS digital time response and the CHANNEL RESPONSE TIME to determine and verify the TOTAL RESPONSE TIME.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.4.8.2	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity ≤ 5.8E-2 μCi/gm.	In accordance with the SFCP AND Between 2 and 6 hours after a THERMAL POWER change of ≥ 15% of RTP within a 1 hour period
SR 3.4.9.1	Verify steam generator (SG) tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.9.2	Verify that each inspected SG tube that satisfies the tube plugging criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 3 following a SG tube inspection

SURVEILLANCE		FREQUENCY
SR 3.4.10.2	Verify the open ACTUATION RESPONSE TIME of each RVV is within limits. The ACTUATION RESPONSE TIME is combined with the allocated MPS digital time response and the CHANNEL RESPONSE TIME to determine and verify the TOTAL RESPONSE TIME.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.1.2	Verify the open ACTUATION RESPONSE TIME of each RVV and RRV is within limits. The ACTUATION RESPONSE TIME is combined with the allocated MPS digital time response and the CHANNEL RESPONSE TIME to determine and verify the TOTAL RESPONSE TIME.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.1.3	Verify the inadvertent actuation block setpoints are within limits, and the inadvertent actuation block function of each RRV.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.2.5	Verify the open ACTUATION RESPONSE TIME of each DHRS actuation valve is within limits. The ACTUATION RESPONSE TIME is combined with the allocated MPS digital time response and the CHANNEL RESPONSE TIME to determine and verify the TOTAL RESPONSE TIME.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.5.4.1	Verify the form and quantity of boron in the ESB dissolvers are within the limits specified in the COLR.	Once prior to entering MODE 1 after operations that could affect the form or quantity of boron in the ESB dissolvers[8]
SR 3.5.4.2	Verify RCS boron concentration is within the ESB operational limits specified in the COLR.	Prior to THERMAL POWER increase ≥ 5% RTP
SR 3.6.1.1	Perform required visual examinations and leakage rate testing in accordance with the	In accordance with the Containment Leakage Rate Testing Program

[8] The Bases for SR 3.5.4.1 state that “operations that could affect the form or quantity of boron in the ESB dissolvers are specified in the COLR.” In a letter dated February 4, 2025 (ML25035A088), the applicant clarified that “an RSV lift would be an example of such an operation.”

	SURVEILLANCE	FREQUENCY
	Containment Leakage Rate Testing Program.	
SR 3.6.2.3	Verify the isolation ACTUATION RESPONSE TIME of each automatic containment isolation valve is within limits. The ACTUATION RESPONSE TIME is combined with the allocated MPS digital time response and the CHANNEL RESPONSE TIME to determine and verify the TOTAL RESPONSE TIME.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.6.4.1	Visually examine PAR enclosure and ensure there is no obstruction or blockage of the inlets or outlets.	During each refueling
SR 3.6.4.2	Test a sample of PAR catalytic plates to confirm catalyst performance.	During each refueling
SR 3.7.1.2	Verify isolation ACTUATION RESPONSE TIME of each MSIV and MSIV bypass valve is within limits on an actual or simulated actuation signal. The ACTUATION RESPONSE TIME is combined with the allocated MPS digital time response and the CHANNEL RESPONSE TIME to determine and verify the TOTAL RESPONSE TIME.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.1.3	Verify each MSIV and MSIV bypass valve leakage is within limits.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.2.2	Verify the closure ACTUATION RESPONSE TIME of each FWIV and FWRV is within limits on an actual or simulated actuation signal. The ACTUATION RESPONSE TIME is combined with the allocated MPS digital time response and the CHANNEL RESPONSE TIME to determine and verify the TOTAL RESPONSE TIME.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.7.2.3	Verify each FWIV and FWRV leakage is within limits.	In accordance with the INSERVICE TESTING PROGRAM

	SURVEILLANCE	FREQUENCY
SR 3.8.2.1	Verify reactor has been subcritical for ≥ 48 hours.	Once prior to movement of irradiated fuel assemblies in the reactor pressure vessel

In a letter dated December 12, 2018 (ML18347A619), regarding NuScale US600 DCA GTS SR 3.7.1.3 and SR 3.7.2.3, NuScale explained how the SSI valve leakage acceptance criteria would be determined, as follows:

> Specific valve leakage limits for the individual valves will be developed as required by the applicable ASME Code requirements. Development of acceptance criteria will be based on the functions of the valves as described in FSAR Tables 3.9-16 and 3.9-17 including forming a portion of the decay heat removal system boundary. Specific values will be developed and implemented in the inservice testing program developed by a COL applicant as required by COL Item 3.9-5.

As part of the NuScale US460 SDAA, FSAR Section 3.9, "Mechanical Systems and Components," includes an equivalent COL item, which is labeled COL Item 3.9-8 and states the following:

> An applicant that references the NuScale Power Plant US460 standard design will establish Preservice and Inservice Testing Programs. These programs are to be consistent with the requirements in the latest edition and addenda of the American Society of Mechanical Engineers (ASME) Operation and Maintenance (OM) Code incorporated by reference in 10 CFR 50.55a.

The staff concludes that COL Item 3.9-8 will ensure that, in accordance with the inservice testing program, a COL applicant will develop appropriate leakage acceptance criteria for the secondary system MSIVs and associated bypass valves, and FWIVs, FWRVs, and feedwater check valves to include the safety-related valves and the backup non-safety-related valves. Limiting valve leakage to within the SR leakage acceptance criteria, established under the inservice testing program, will ensure that each passive decay heat removal subsystem will maintain sufficient water inventory to perform its specified safety function in case an event requiring DHRS heat exchanger operation occurs.

The staff determined that the Frequencies of the above Surveillances, which are not included in the SFCP, are appropriate for the NuScale US460 design and consistent with W-STS and W-AP1000-STS and are therefore acceptable.

16.4.8.3 Surveillance Frequencies Governed by the Surveillance Frequency Control Program

FSAR Table 16.1-1 states the initial base Frequencies of SRs with Frequencies governed by GTS Subsection 5.5.11, and the basis for each Frequency. FSAR Section 16.1.1 states the following:

> Table 16.1-1 provides the initial surveillance test frequencies to be incorporated into the SFCP required by NuScale GTS 5.5.11. The table identifies each GTS surveillance test requirement that references the SFCP, the base testing frequency for evaluation of future changes to the surveillance test frequency, and

> the basis for that initial base test frequency. Base test frequencies in Table 16.1-1 include consideration of the rules of applicability for surveillance testing including, when applicable, up to 1.25 times the specified interval as permitted by technical specification SR 3.0.2. For example, a base frequency of 24 months implies consideration of up to 30 months between performance[s] of the surveillance test. The SFCP ensures surveillance requirements specified in the Technical Specifications are technically justified for a plant specific design and performed at intervals sufficient to assure the associated limiting conditions for operation are met.

Therefore, FSAR Table 16.1-1 is part of the program documentation required by GTS Subsection 5.5.11. This programmatic specification is acceptable because it conforms to the W-STS and TSTF-425-A, “Relocate Surveillance Frequencies to Licensee Control - [Risk-Informed] TSTF Initiative 5b.” Based on its evaluation of the Surveillance Frequencies governed by the SFCP, the staff determined that the base Frequencies, and the rationales of the base Frequencies for the SRs within the SFCP scope are consistent with the guidance in Nuclear Energy Institute (NEI) 04-10, “Risk-Informed Method for Control of Surveillance Frequencies,” Revision 1, dated April 2007 (ML071360456). To apply test experience to justify a longer test interval, the guidance assumes the validity of the basis for the existing Frequency. The proposed bases for the initial Frequencies for the GTS SRs to be controlled by the SFCP are valid because they are consistent with STS Frequencies for equivalent SRs and are appropriate for the NuScale US460 design. Therefore, the staff concludes that the initial Frequencies and associated Bases in FSAR Table 16.1-1, and GTS Subsection 5.5.11 are acceptable.

16.4.8.4 Instrumentation Surveillances

Channel Check

A channel check with a base performance frequency of 12 hours is specified for each MPS function listed in Table 3.3.1-1, which is consistent with the testing described in FSAR Section 7.2.15 for RTS and ESFAS instrumentation. FSAR Section 7.2.15.1, “System Calibration,” states the following:

> The MPS and the NMS are designed with the capability for calibration and surveillance testing, including channel checks, calibration verification, and time response measurements, as required by the technical specifications to verify that I&C safety systems perform required safety functions.

FSAR Section 7.2.15.2, “Instrumentation and Controls System Testing,” states the following, in part:

> The MPS and the NMS allow [structures, systems, and components (SSC)] to be tested while retaining the capability to accomplish required safety functions. The MPS uses modules from the [Highly Integrated Protection System (HIPS)] platform that are designed to eliminate non-detectable failures through a combination of built-in self-testing and periodic surveillance testing.
>
> Testing from the sensor inputs of the MPS through to the actuated equipment is accomplished through a series of overlapping sequential tests, and the majority of the tests can be performed with the NPM at power. Where testing final equipment at power has the potential to upset plant operation or damage

equipment, provisions are made to test the equipment when the NPM is shut down.

…

The MPS provides a means for checking the operational availability of the sense and command feature input sensors relied upon for a safety function during reactor operation.

This capability is provided by one of the following methods:

- perturbing the monitored variable
- cross-checking between channels that have a known relationship (i.e., channel check)
- introducing and varying a substitute input to the sensor.

The staff finds that the specified channel checks are consistent with the MPS testing described in FSAR Section 7.2.15 and are therefore acceptable.

Channel Operational Test

In the NuScale US600 DCA and the US460 SDAA , NuScale proposed specifying a Channel Operational Test only for LCO-required instrumentation functions implemented by the module control system. These are the RCS leakage detection instrumentation of the CES condensate monitor (two channels) and CES gaseous radioactivity monitor (one channel), which are evaluated below.

In the NuScale US600 DCA review, the staff requested that NuScale clarify its justification for not proposing to specify a Channel Operational Test for MPS instrumentation functions, the RTS logic and actuation function, ESFAS logic and actuation functions, and the CES inlet pressure monitor channels, which are used for RCS leak detection, in RAI 156-9031, Question 16-2, Subquestion c (ML17220A038); RAI 196-9050, Question 16-16, Subquestion e (ML17237C007); and RAI 197-9051, Question 16-25, Subquestion a4.2 (ML17237C008).

In its responses to Subquestion 16-2c (ML17269A210), Subquestion 16-16e (ML17291A482), and Subquestion 16-25.a4.2 (ML17291A299), NuScale stated that a Channel Operational Test would not add to the assurance of operability provided by the MPS continuous self-testing features, which verify sensor input to the output switching logic for MPS instrumentation functions, the RTS logic and actuation function, ESFAS logic and actuation functions, and the CES inlet pressure monitor channels. As part of the US460 SDAA review, NuScale confirmed in an audit question response (ML24326A095) that the justification provided in responses to DCA RAI 16-2 Item c, 16-16 item e, and 16-25 item a.4 for omission of Channel Operational Tests for MPS instrumentation Functions, and MPS supported RCS leakage detection instrumentation is valid for the NuScale US460 SDAA GTS. SER Chapter 7 contains the staff’s evaluation of the capability of MPS self-testing features to provide adequate assurance of MPS operability without performing manual Channel Operational Tests.

The staff recognizes that, upon receipt of an alarm generated by an MPS self-testing feature, the control room staff would promptly determine the operability of the affected MPS instrumentation function channel or MPS logic and actuation function division. This operability determination would include following the alarm response procedure, which is required by Specification 5.4.1. This procedure can be expected to account for the built-in redundancy of the MPS power supplies and the logic within each SFM, SVM, and communication module provided for each MPS function channel and actuation logic division. An alarm associated with a single redundant component within a module would likely not make the associated channel or division inoperable but would be addressed by the licensee's corrective action program. The combination of the MPS continuous self-testing capability and the 12-hour Frequency channel check, which includes verifying alarm status, provide adequate assurance that any component malfunction in an MPS channel or division will not go undetected for more than a brief period. This will ensure that an MPS degraded condition is identified and its effect on channel or division operability determined in a timely manner.

The US460 SDAA takes the same approach as the NuScale US600 DCA in not proposing to specify a Channel Operational Test for MPS instrumentation functions, the RTS logic and actuation function, ESFAS logic and actuation functions, and the CES inlet pressure monitor channels. For the reasons described above, the staff concludes that omission of a Channel Operational Test SR for MPS instrumentation functions and MPS logic and actuation functions, and MPS-supported RCS leakage detection instrumentation is acceptable.

A Channel Operational Test with a base Frequency of 92 days is specified for the CES gaseous radioactivity monitor channel by SR 3.4.7.4, and the two CES condensate monitor channels by SR 3.4.7.5. These SRs are needed because these monitors are implemented using the module control system, which lacks the MPS capability to perform automatic self-testing of the instrument loop. In addition, the US460 SDAA GTS Section 1.1 definition of Channel Operational Test for analog instrumentation is based on the W-STS definition of Channel Operational Test for analog instrumentation, which includes approved STS change traveler TSTF-563, Revision 0. As such, the staff concludes that this definition is acceptable. The proposed base test Frequency of 92 days is consistent with the Channel Operational Test Frequency of similar instrumentation in current use at operating power reactor facilities and also with the W-STS and the W-AP1000-STS. The rationale for this base Frequency, provided in FSAR Section 16.1, Table 16.1-1, states the following:

> The Frequency of 92 days considers instrument reliability, and industry operating experience has shown that it is proper for detecting degradation.

This is consistent with the basis for the 92-day Frequency of the Channel Operational Test for the F18 radioactive particulate monitor leakage detection instrumentation in W-AP1000-STS Subsection B 3.4.9, and therefore, acceptable. SER Section 16.4.2 describes the proposed modification of the Channel Operational Test definition, and other instrumentation surveillance definitions, based on recently approved STS change traveler TSTF-563, on adoption of an SFCP. Therefore, the staff concludes that the proposed GTS Channel Operational Test SRs are acceptable.

Channel Calibration

A Channel Calibration with a base Frequency of 24 months is specified for each MPS instrumentation function listed in Table 3.3.1-1, which is consistent with the testing described in FSAR Section 7.2.15 for RTS and ESFAS instrumentation. The proposed base test Frequency is consistent with the Channel Calibration Frequency of similar instrumentation in current use at operating power reactor facilities and also with the W-STS and the W-AP1000-STS.

The rationale for this base Frequency, provided in FSAR Section 16.1, Table 16.1-1, for SR 3.3.1.4, states the following:

> The Frequency is based on consideration of the design reliability and performance characteristics of the equipment.

A Channel Calibration with a base Frequency of 24 months is specified for each RCS leakage detection instrument function channel. The proposed base test Frequency is consistent with the Channel Calibration Frequency of similar instrumentation in current use at operating power reactor facilities and also with the W-STS and the W-AP1000-STS.

The rationale for this base Frequency, provided in FSAR Section 16.1, Table 16.1-1, for SR 3.4.7.6 for the CES condensate channel, and SR 3.4.7.8 for the CES gaseous radioactivity monitor channel, states the following:

> The Frequency of 24 months considers instrument reliability, and industry operating experience that has proven that this Frequency is acceptable.

The rationale for this base Frequency, provided in FSAR Section 16.1, Table 16.1-1, for SR 3.4.7.7 for the CES inlet pressure channel, states the following:

> The Frequency of 24 months is based on the assumption of a 30 month calibration interval in determination of the magnitude of equipment drift in the setpoint methodology.

A Channel Calibration with a base Frequency of 24 months is specified for SR 3.8.1.2 for the two refueling neutron flux channels and one refueling neutron flux audible count rate channel. The rationale for this base Frequency, provided in FSAR Section 16.1, Table 16.1-1, states the following:

> Industry operating experience has shown that similar components usually pass this surveillance when performed at the 24-month Frequency.

The staff concludes that the above rationales are acceptable because they are consistent with STS Bases and are appropriate for the NuScale US460 reactor coolant leakage detection instrumentation.

SER Chapter 7 describes the individual components comprising a measurement channel of an MPS instrument loop and subject to calibration. However, for the present evaluation of the Channel Calibration surveillance, the staff considered the descriptions provided in the NRC-approved US600 DC TR-1015-18653-P-A, Revision 2, and the NuScale US460 SDAA TR-122844-P, "NuScale Instrument Setpoint Methodology," Revision 0.

The staff notes that Bases Subsection B 3.3.1 includes discussions that clarify the scope and intent of the Channel Calibration. One discussion explains that, when determining the as-found trip setting at the beginning of the instrument Channel Calibration, as-found tolerances for the output signal of each sensor and device in the instrument loop must be satisfied for the sensor or device to be considered functioning normally. Provided the actual trip setting of the channel as a whole is within the as-found tolerance specified by the setpoint program (SP), the channel is considered operable. However, any sensor or device found to be outside its as-found tolerance should be entered into the corrective action program. Specifically, the Background section of Bases Subsection B 3.3.1 discusses the nominal trip setpoints (NTSPs) as follows *(emphasis added)*:

> The trip and actuation setpoints used in the [safety function module (SFM)] core logic function are based on the analytical limits derived from safety analysis (Ref. 5). The calculation of the [limiting trip setpoint (LTSP)] specified in the Setpoint Program (SP) is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those MPS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 6), the LTSP specified in the SP is conservative with respect to the analytical limits. The nominal trip setpoint (NTSP) is the LTSP with margin added and is always equal to or more conservative than the LTSP. A detailed description of the methodology used to calculate the NTSPs is provided in the "NuScale Instrument Setpoint Methodology" (Ref. 7). The as-left tolerance and as-found tolerance band methodology is provided in the SP. The as-found OPERABILITY limit for the purpose of the CHANNEL CALIBRATION is defined as the as-left limit plus the acceptable drift about the NTSP.
>
> The NTSPs listed in the SP are based on the methodology described in Reference 7, which incorporates the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each NTSP. *All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes. Transmitter and signal processing equipment calibration tolerances and drift allowances must be specified in plant calibration procedures and must be consistent with the values used in the setpoint methodology.*
>
> The OPERABILITY of each transmitter or sensor can be evaluated when its "as-found" calibration data are compared against the "as-left" data and are shown to be within the setpoint methodology assumptions. The as-left and as-found tolerances listed in the SP define the OPERABILITY limits for a channel during a periodic CHANNEL CALIBRATION that requires trip setpoint verification.

Another discussion in the Background section of Bases Subsection B 3.3.1 explains that an RTS trip Function division requires its associated interlock to be in the correct state to be operable, likewise, for an ESFAS function that has an enabling interlock. The combined Applicable Safety Analyses, LCO, and Applicability sections of Bases Subsection B 3.3.1 (and by reference in Subsections B 3.3.2 and B 3.3.3) state the following, in part:

> …Proper operation of these permissive[s] and interlocks supports OPERABILITY of the associated reactor trip and ESF functions and/or the requirement for actuation logic OPERABILITY. The permissives and interlocks must be in the required state, as appropriate, to support OPERABILITY of the associated functions. The permissives and interlocks associated with each MPS Instrumentation Function channel, each Reactor Trip System (RTS) Logic and Actuation Function division, and each Engineered Safety Features Actuation System (ESFAS) Logic and Actuation Function division, respectively, must be OPERABLE for the associated Function channel or Function division to be OPERABLE.

In addition, since the sensors and transmitters for process variables used by the RTS and ESFAS are also used to generate the interlock and permissive signals, a Channel Calibration of an MPS sensor and transmitter satisfies the calibration requirement for the shared interlock sensor and transmitter. The combined Applicable Safety Analyses, LCO, and Applicability sections of Bases Subsection B 3.3.1 also state the following, in part:

> …The combination of the continuous self-testing features of the MPS and the CHANNEL CALIBRATION specified by SR 3.3.1.4 verify the OPERABILITY of the interlocks and permissives. Specification 5.5.10, Setpoint Program is used to control interlock and permissive setpoints.

The above quotation clarifies the relationship of the MPS instrumentation Functions, and their bypassing or enabling interlocks and permissives, to the SP controls and Channel Calibration surveillances.

US460 SDAA Part 4 contains similar clarifications throughout the combined Applicable Safety Analyses, LCO, and Applicability sections of Bases Subsection B 3.3.1, under the heading "Reactor Trip System and ESFAS Functions," for each MPS instrumentation Function that has an associated supporting interlock or permissive Function. In most cases, a sentence such as the following is appended to the appropriate paragraph: "Interlock and permissive setpoints are governed by the Setpoint Program." The applicant also included the phrase "in accordance with the Setpoint Program" with the statement of the Channel Calibration in SR 3.3.1.4 to ensure the SR is performed according to the requirements of the SP. Additionally, US460 SDAA, Part 4, GTS Subsection 5.5.10, paragraph b, references TR-122844-P (previously, the staff approved a similar setpoint methodology report, TR-1015-18653-P-A, Revision 2, as part of the NuScale US600 DC). Furthermore, the setpoint methodology report referenced in paragraph 5.5.10.b places the report number, NRC-approved version number, and title of the setpoint methodology report in square brackets to ensure that the SP reflects the most current approved version of the methodology when a COL applicant submits the plant-specific TS for approval. The staff finds the scope of the Channel Calibration SRs, as described in the Bases, and the associated SP consistent with the NuScale US460 design and the W-AP1000-STS. Therefore, the Channel Calibration SRs for MPS instrumentation functions are acceptable.

<u>MPS Class 1E Isolation Devices</u>

The staff determined that specifying a Channel Calibration for the Class 1E isolation devices is appropriate for ensuring that the device will actuate to protect the MPS upon an electrical fault in the electrical power supply, and that the 24-month initial Frequency is adequately justified in FSAR Chapter 16, Table 16.1-1, by the associated Bases. Therefore, the staff concluded that the Channel Calibration SRs for the Class 1E isolation devices are acceptable.

Actuation Logic Test

SER Section 16.4.2.2 discusses the NuScale US460 definition of the defined term Actuation Logic Test.

An Actuation Logic Test with a base Frequency of 24 months is specified for the two divisions of the RTS Logic and Actuation for the reactor trip Function, which is addressed by GTS Subsection 3.3.2, and the two divisions of the ESFAS Logic and Actuation for each ESFAS Actuation Function listed in Table 3.3.3-1.

The discussion of SR 3.3.2.1 in US460 SDAA Part 4, Bases Subsection B 3.3.2, states the following, in part:

> …The RTS Logic and Actuation circuitry functional testing is accomplished with continuous system self-testing features on the SVMs and [equipment interface modules (EIMs)] and the communication between them. The self-testing features are designed to perform complete functional testing of all circuits on the SVM and EIM, with the exception of the actuation and priority logic (APL) circuitry. The self-testing includes testing of the voting and interlock/permissive logic functions. The built-in self-testing will report a failure to the operator and place the SVM or EIM in a fail-safe state.
>
> …
>
> The ACTUATION LOGIC TEST includes testing of the APL on all RTS EIMs, the enable nonsafety control switches, and the operating bypass switches. The ACTUATION LOGIC TEST includes a review of any alarms or failures reported by the self-testing features.

In US460 SDAA Part 4, the SRs section of Bases Subsection B 3.3.3, for SR 3.3.3.1 similarly states, in part, the following:

> The ESFAS Logic and Actuation circuitry functional testing is accomplished with continuous system self-testing features on the SVMs and EIMs and the communication between them. The self-testing features are designed to perform complete functional testing of all circuits on the SVM and EIM, with the exception of the actuation and priority logic (APL) circuitry. The self-testing includes testing of the voting and interlock/permissive logic functions. The built-in self-testing will report a failure to the operator and place the SVM or EIM in a fail-safe state.
>
> …
>
> The ACTUATION LOGIC TEST includes testing of the APL on all ESFAS EIMs, the enable nonsafety control switches, the main control room isolation switches, the override switches, and the operating bypass switches. The ACTUATION LOGIC TEST includes a review of any alarms or failures reported by the self-testing features.

Based on its review as described in SER Section 16.4.2.2 and the above discussion, the staff concludes that the Actuation Logic Test SRs are appropriate for the NuScale US460 MPS and APL design. The staff concludes that, because of the continuous self-testing of the MPS components, the Actuation Logic Test will provide adequate assurance that the MPS RTS and ESF Logic and Actuation Functions are operable and that LCO 3.3.2 and LCO 3.3.3 are met. For the logic components, such as the APL, which are not covered by self-tests, manual testing under the Actuation Logic Test Frequency of 24 months is adequate to assure operability because of the component reliability and the need to perform this test when the unit is shut down to avoid an inadvertent operational transient.

FSAR Section 7.2.15.2 addresses testing that cannot be performed during normal power operation, as follows:

> Where testing final equipment at power has the potential to upset plant operation or damage equipment, provisions are made to test the equipment when the NPM is shut down.

In FSAR Table 16.1-1, the bases for the 24-month initial Frequency of SR 3.3.2.1 and SR 3.3.3.1 are similar and consistent with the above. The 24-month initial Frequency bases for SR 3.3.2.1 states the following:

> The 24-month Frequency is based on the potential for unplanned plant transients if the surveillances were performed with the unit at power. This Frequency is justified based on the system design, which includes the use of continuous diagnostic test features that report a failure within the logic and actuation system to the operator promptly. The only part of the actuation logic circuitry that is not continuously self-tested is the actuation and priority logic circuit, which consists of simple discrete components that are very reliable.

Based on the above information, the staff concludes that the Actuation Logic Test SRs and their 24-month Frequency are acceptable.

Based on its review as described in SER Section 16.4.2 and the above evaluation, the staff concludes that the instrumentation Surveillances are acceptable.

Conclusion for Surveillance Requirements

Based on its review and evaluation of the NuScale US460 SRs, the staff concludes that the Surveillance statements and Frequencies of Section 3.1 though Section 3.8 satisfy 10 CFR 50.36(c)(3) and, therefore, are acceptable.

16.4.9 Design Features (GTS Chapter 4, Sections 4.1 through 4.3)

In 10 CFR 50.36(c)(4), the NRC requires that TS include design features, which it states are “those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in [the] categories” of SLs, LCOs, or SRs.

GTS Chapter 4 addresses the requirement to include design features not covered in the LCOs of GTS Chapter 3. GTS Chapter 4 contains information about site location, core design, and fuel storage design.

16.4.9.1 GTS Section 4.1, Site Location

GTS Section 4.1 contains bracketed information regarding site-specific information about the facility’s location, site and exclusion boundaries, and the low-population zone, which must be provided by applicants for a COL referencing the NuScale US460 SDA. SER Section 16.5 summarizes the COL information items included in NuScale US460 SDAA Part 4.

16.4.9.2 GTS Section 4.2, Reactor Core

GTS Section 4.2 contains reactor core design requirements. GTS Subsection 4.2.1, “Fuel Assemblies,” specifies the design number of fuel assemblies and allowed composition of fuel rod cladding and fuel material, and it requires that fuel assemblies be limited to fuel designs that have been analyzed with applicable NRC staff-approved codes and methods and shown by tests or analyses to comply with fuel safety design bases. It also permits a limited number of lead test assemblies that have not completed representative testing to be placed in nonlimiting core regions.

GTS Subsection 4.2.2, “Control Rod Assemblies,” specifies the number of CRAs and the permitted control materials used in the CRAs.

16.4.9.3 GTS Section 4.3, Fuel Storage

GTS Section 4.3 contains bracketed information regarding the facility’s fuel storage soluble boron concentration, nominal center-to-center distance between fuel assemblies, spent fuel pool drainage design, and spent fuel pool storage capacity, which must be provided by applicants for a COL referencing the NuScale US460 SDA. SER Section 16.5 summarizes the COL information items included in NuScale US460 SDAA Part 4 and Part 2 Section 16.1.1.

GTS Subsection 4.3.1, “Criticality,” contains design requirements for the spent fuel storage racks to prevent criticality of the stored fuel assemblies. The spent fuel storage racks are designed and maintained with stored fuel assemblies with a maximum uranium-235 enrichment of 5.0 weight percent; a $k_{eff} \leq 0.95$ with the fuel storage pool fully flooded with borated water at a minimum soluble boron concentration of [800] parts per million; a $k_{eff} < 1.00$ with the fuel storage pool fully flooded with unborated water; and a nominal [25.4] cm ([10] inch) center-to-center distance between fuel assemblies placed in the spent fuel storage racks. To meet GDC 61, “Fuel storage and handling and radioactivity control,” NuScale is incorporating neutron-absorbing material into the design of the spent fuel racks to maintain the specified subcriticality and ensure safe operation. GTS Subsection 5.5.12, “Spent Fuel Rack Neutron Absorber Monitoring Program,” is included in GTS Section 5.5 to ensure safe operation by requiring periodic physical examination and neutron attenuation testing, and performance-based examinations. SER Section 16.4.10.3 further discusses GTS Subsection 5.5.12. GTS Subsection 4.3.2, “Drainage,” requires the spent fuel pool to be designed and maintained to prevent inadvertent draining of the pool below [5.8] m ([19] ft) above the spent fuel pool floor. GTS Subsection 4.3.3, “Capacity,” contains information regarding how the spent fuel pool shall be designed and maintained to hold no more than ([600]) fuel assemblies.

Conclusion for GTS Chapter 4

The staff found GTS Chapter 4 to be consistent with the W-STS and the NuScale US460 design as described in the FSAR. The staff concludes that GTS Chapter 4 satisfies 10 CFR 50.36(c)(4) and is therefore acceptable.

16.4.10 Administrative Controls (GTS Chapter 5, Sections 5.1 through 5.7)

16.4.10.1 GTS Sections 5.1, Responsibility; 5.2, Organization; and 5.3, Facility Staff Qualifications

The staff reviewed these sections and found that they are consistent with the W-STS with one acceptable exception, as noted below, and are therefore acceptable. For GTS Subsection 5.2.2, “Facility Staff,” SER Chapter 18 evaluates NuScale’s minimum licensed staffing requirements when there is fuel in any reactor vessel.

GTS Subsection 5.2.2 omits the W-STS requirement for a Shift Technical Advisor. This difference is acceptable as described in Sections 2.2 and 3.5 of the staff’s safety evaluation of the NRC-approved NuScale topical report, TR-0420-69456, Revision 1-A, “NuScale Control

Room Staffing Plan," dated June 2, 2021 (ML21231A286), and US460 SDAA SER Section 18.5.4. Note that an applicant for a COL that references the NuScale US460 SDA will need to request an exemption from 10 CFR 50.54(m) for three licensed operators (one licensed reactor operator and two licensed senior reactor operators) to operate up to six units, and an exemption from 10 CFR 50.120(b)(2)(iii) for Shift Technical Advisor training requirements.

16.4.10.2 GTS Section 5.4, Procedures

<u>GTS Subsection 5.4.1, Procedures</u>

Because the GTS Subsection 5.4.1 opening paragraph and paragraphs a, b, c, d, and e are consistent with the W-STS, the staff concludes these paragraphs are acceptable. In particular, paragraph 5.4.1.a states the following:

> 5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:
>
> a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 3, June 2013;

The staff noted that the US460 Standard Design Approval Technical Specifications Development report (TR-101310-NP), like the US600 DCA Regulatory Conformance and Development Report, does not describe how the availability and testing of NuScale non-safety-related SSCs, which NuScale has determined do not meet any of the four LCO selection criteria of 10 CFR 50.36(c)(2)(ii), are intended to be controlled by a NuScale Nuclear Power Plant COL holder, including references to any regulatory basis for the controls. As part of the US600 DCA review, NuScale, in a letter dated August 26, 2019 (ML19238A372), stated that including procedural controls for such non-safety-related SSCs in Specification 5.4.1 is not required by regulation and would be inconsistent with STS 5.4.1. In that response to the staff's RAI about controls for non-safety-related SSCs during the US600 DCA review, NuScale proposed establishing an owner-controlled requirements manual (OCRM) that would provide reasonable assurance that appropriate written procedures will be established, implemented, and maintained to satisfy the availability and reliability requirements for selected non-safety-related SSCs that are within the scope of the OCRM. In its US460 SDAA, NuScale similarly proposed the establishment of an OCRM by a COL applicant as COL Item 16.1-2, which FSAR Section 16.1.1 describes as follows:

> An applicant that references the NuScale Power Plant US460 standard design will prepare and maintain an owner-controlled requirements manual that includes owner-controlled limits and requirements described in the Bases of the Technical Specifications or as otherwise specified in the FSAR.

FSAR Section 6.4.5, "Testing and Inspection," states that the OCRM manual will include controls over the availability and reliability of the control room habitability system and the control room envelope.

Other SSCs, which are not safety related, are expected to have procedural controls appropriate to their classification. To the extent that the FSAR describes non-safety-related SSCs, 10 CFR 50.59 will govern changes to those descriptions. SSCs described in the FSAR as being included in the OCRM will be subject to whatever additional controls a COL applicant includes to resolve COL Item 16.1-2.

Based on the above discussion and its evaluation, the staff concludes GTS Section 5.4 is acceptable.

16.4.10.3 GTS Section 5.5, Programs and Manuals

GTS Subsection 5.5.1, Offsite Dose Calculation Manual (ODCM)
GTS Subsection 5.5.2, Radioactive Effluent Control Program
GTS Subsection 5.5.3, Component Cyclic or Transient Limit

These three Specifications are consistent with W-STS and are therefore acceptable.

GTS Subsection 5.5.4, Steam Generator (SG) Program

Together with GTS Subsection 3.4.9, “SG Tube Integrity,” and GTS Subsection 5.6.5, “Steam Generator Tube Inspection Report,” the SG Program ensures that SG tube integrity is maintained. The staff compared these subsections against the corresponding provisions in Revision 5 of the W-STS, which include changes made by STS change travelers TSTF-510-A, Revision 2, “Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection,” dated March 2011 (ML110610350), and TSTF-577-A, Revision 1, “Revised Frequencies for Steam Generator Tube Inspections,” dated March 2021 (ML21060B434). The staff also took into consideration the unique design of the NuScale US460 NPM SGs to evaluate the proposed (1) criteria for assessing the as-found condition of a SG tube following a tube inspection, (2) SG tube integrity performance criteria (tube structural integrity, accident-induced primary-to-secondary Leakage limits, and operational primary-to-secondary Leakage limits), (3) SG tube plugging criteria, (4) criteria for selection of SG tube inspection intervals and the tubes to be inspected, and (5) provisions for monitoring operational primary-to-secondary Leakage. SER Section 5.4.2, “SG Program,” further discusses the staff’s technical evaluation of SG-related GTS requirements.

The NRC staff determined that the GTS incorporate TSTF-510 and TSTF-577 as intended, consistent with the NuScale US460 design, and concludes that GTS Subsections 3.4.5, 3.4.9, 5.5.4, and 5.6.5 together satisfy 10 CFR 50.36(c) and Subsections (2), (3), and (5) and that Bases Subsections B 3.4.5 and B 3.4.9 satisfy 10 CFR 50.36(a); they are therefore acceptable.

GTS Subsection 5.5.5, Secondary Water Chemistry Program
GTS Subsection 5.5.6, Explosive Gas and Storage Tank Radioactivity Monitoring Program
GTS Subsection 5.5.7, Technical Specifications (TS) Bases Control Program
GTS Subsection 5.5.8, Safety Function Determination Program (SFDP)

These four program Specifications are consistent with the W-STS and are therefore acceptable.

GTS Subsection 5.5.9, Containment Leakage Rate Testing Program

The staff verified that this subsection is consistent with the W-STS and satisfies the requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.” This program includes two options, A and B, either of which can be chosen for meeting the requirements of Appendix J. This program is referenced by US460 SDAA Part 4, SR 3.6.1.1, “Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.” In Bases Subsection B 3.6.1, the Applicable Safety Analyses section defines P_a as “the calculated peak containment internal pressure [940] psia [6481 kPa (absolute)] (P_a) resulting from the limiting DBA”; the LCO section states, “Leakage integrity is ensured by performing local leak rate testing (LLRT) and containment inservice inspection. Total LLRT leakage is maintained < 0.60 L_a in accordance with 10 CFR Part 50, Appendix J (Ref. 1)."

GTS Subsection 5.5.10, Setpoint Program

SER Chapter 7 gives the staff's evaluation of TR-122844-P. GTS Specification 5.5.10, paragraph b, of US460 SDAA Part 4, includes the document revision number of the NRC-approved version of TR-122844-P.

Based on being consistent with the W-AP1000-STS SP, including the number of the setpoint methodology version approved by the NRC, and requiring calculation and documentation of the LTSP values in accordance with the approved setpoint methodology, the staff finds the SP satisfies the LSSS requirement of 10 CFR 50.36(c)(1)(ii)(A) and is therefore acceptable.

GTS Subsection 5.5.11, Surveillance Frequency Control Program

SER Section 16.4.8.3 gives the staff's evaluation of this program.

GTS Subsection 5.5.12, Spent Fuel Storage Rack Neutron Absorber Monitoring Program

This program Specification is bracketed to indicate it is a part of COL Item 16.1-1. Because the GTS adopt the content of the improved version of this program, as stated in STS change traveler TSTF-557-A, Revision 1, this Subsection is acceptable.

Conclusion for GTS Section 5.5

Based on the above evaluation, the staff concludes that GTS Section 5.5 is acceptable.

16.4.10.4 GTS Section 5.6, Reporting Requirements

GTS Subsection 5.6.1, Annual Radiological Environmental Operating Report
GTS Subsection 5.6.2, Radioactive Effluent Release Report

These report Specifications are consistent with the W-AP1000-STS report Specifications and are therefore acceptable.

GTS Subsection 5.6.3, Core Operating Limits Report (COLR)

This GTS Subsection lists the documents describing analytical methods previously reviewed and approved by the NRC and used to determine the core operating limits in GTS Subsection 5.6.3, paragraph b, with brackets to indicate the list is a COL item. Along with this bracketed list, the applicant also inserted the following bracketed reviewer's note:

> b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
>
> [---------------------------------REVIEWER'S NOTE---------------------------------
> The COL applicant shall confirm the validity of each listed document and the listed Specifications for the associated core operating limits, or state the valid NRC approved analytical method document and list of associated Specifications.

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

Each document is listed by title, revision number, and date, along with the LCOs supported by the analytical methods described in the document. Paragraph b is consistent with the W-AP1000-STS presentation regarding level of detail and format. SER Section 4.3 gives the staff's evaluation of the listed methodologies. The staff verified that the listed methodologies and associated LCOs accurately cite the FSAR sections and topical reports provided in the FSAR. Specification 5.6.3 encloses these methodology references and associated LCOs within brackets to indicate their designation as COL items to ensure that the COLR specification in the plant-specific TS of a COL application references the approved version of each methodology used by the COL applicant.

GTS Subsection 5.6.4, Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

The language of this GTS Subsection is consistent with the language of the W-STS PTLR Specification and is therefore acceptable. However, as discussed in SER Section 5.3, which gives the staff's evaluation of TR-130877, Revision 1, "Pressure and Temperature Limits Methodology," dated December 2022 (ML23304A341), the staff's approval of the pressure-temperature limits presented in the PTLR is based on preliminary transients (not final transients), and consequently, is predicated on acceptable application of NuScale SDAA COL Item 5.3-1 by an applicant referencing the US460 SDA. A COL applicant must also be subject to NuScale US460 SDAA Exemptions 6 and 15. Based on these conditions, the staff concludes that GTS Subsection 5.6.4 is acceptable.

GTS Subsection 5.6.5, Steam Generator Tube Inspection Report

This GTS Subsection is consistent with the W-STS Subsection for the SG tube inspection report and NRC staff-approved improvements described in TSTF-510-A and TSTF-577-A. It is therefore acceptable.

Omitted Reports Included in STS Section 5.6

As described in SER Section 16.4.1.6, the GTS do not include an LCO for PAM instrumentation; accordingly, Section 5.6 omits the PAM Report.

16.4.10.5 GTS Section 5.7, High Radiation Area

Because GTS Section 5.7 is identical to W-STS Section 5.7, it is acceptable.

Conclusion for GTS Chapter 5

Based on its review and the above evaluation, the staff concludes that GTS Chapter 5 satisfies 10 CFR 50.36(c)(5); therefore, the GTS administrative controls are acceptable.

16.4.11 Technical Specification Task Force Traveler Disposition

As part of the NuScale US600 DCA, the applicant presented its evaluation of TSTF travelers in Table C-1 of Appendix C of Revision 4 of the Regulatory Conformance and Development

Report. Appendix C of the Regulatory Conformance and Development Report stated the following:

> 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 [equivalence] 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 [were] considered are those that were issued as new or revised since the earliest manuscript date of [Revision 4 of] the [STS NUREGs], October 2011, and by comparison of the traveler content with the contents of the STS with the changes identified in the TSTF.

For the US450 SDAA, the applicant supplemented that report in Table 4-1 of TR-101310-NP, "US460 Standard Design Approval Technical Specifications Development." The staff reviewed the applicant's rationale for choosing or declining to incorporate applicable changes of each TSTF traveler that the applicant had evaluated for both the US600 DCA and US460 SDAA. Travelers that were not yet approved by the staff when Revision 2 of the NuScale US460 SDAA was submitted may be recognized by having no "-A" appended to the traveler number.

In SER Table 16.4.11-1 to Table 16.4.11-6, italics denote material quoted from DCA Regulatory Conformance and Development Report Table C-1, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development report (TR-101310-NP). SER Table 16.4.11-7 lists commitments that a COL applicant must make as a condition of NRC approval of plant-specific TS changes based on TSTF traveler changes already incorporated into Revision 5 of the STS. The status of each traveler listed in the tables below is based on its approval status when Revision 2 of US460 SDAA Part 4 was submitted to the NRC on April 9, 2025. In this SER section, the phrase "TSTF travelers proposed for incorporation" means TSTF traveler changes that the applicant has considered and found to be appropriate for the NuScale US460 design and has proposed for the GTS to the extent practicable.

- Table 16.4.11-1 *Approved TSTF travelers proposed for incorporation:* 490-A, 493-A, 510-A, 513-A, 523-A, 529-A, 545-A, 546-A, 554-A, 557-A, 563-A, 565-A, 577-A
- Table 16.4.11-2 *Approved TSTF travelers not proposed for incorporation:* 426-A, 432-A, 501-A, 505-A, 514-A, 522-A, 535-A, 541-A, 542-A, 547-A, 551-A, 567-A, 580-A, 582-A, 584-A, 589-A, 591-A

- Table 16.4.11-3 *Unapproved TSTF travelers under NRC staff review*
  Proposed for incorporation: None
  Not proposed for incorporation:
  521, 530, 531, 536, 537, 538, 540, 566, 568, 569, 576, 585, 592, 596 (Note that some acceptable editorial changes to STS Bases proposed in withdrawn traveler TSTF-530 were adopted.)

- Table 16.4.11-4 *Withdrawn, previously approved, or pending TSTF travelers:*
  Proposed for incorporation: None
  Not proposed for incorporation:
  454, 515, 525, 534, 553, 564, 588

- Table 16.4.11-5 *Disposition of T-travelers:*
  Proposed for incorporation:
  502-T, 548-T, 555-T
  Not proposed for incorporation:
  494-T, 504-T, 520-T, 524-T, 526-T, 527-T, 528-T, 532-T, 533-T, 539-T, 543-T, 549-T, 550-T, 556-T, 558-T, 559-T, 560-T, 561-T, 562-T, 571-T, 573-T, 574-T, 575-T, 578-T, 579-T, 581-T, 583-T, 586-T, 587-T, 594-T, 595-T

- Table 16.4.11-6 *Conditions for adoption of TSTF changes, which are included in STS Revisions 4 and 5,*
  Included in GTS: 359-A, 366-A, 425-A, 427-A
  Not included in GTS: 409-A, 422-A

(For the tables in this SER section, shading in the first column denotes that the listed traveler is incorporated, as described.)

16.4.11.1 Approved Technical Specification Task Force Travelers Proposed for Incorporation

Table C-1 of US600 DCA Regulatory Conformance and Development Report, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development report, indicated the following approved travelers were proposed for incorporation, consistent with the NuScale US460 design. Italics denote material quoted from Table C-1 of the US600 DCA Regulatory Conformance and Development Report, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development technical report.

Table 16.4.11-1 Approved TSTF Travelers Proposed for Incorporation

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Incorporation
490-A, Revision 1	*Deletion of E-Bar Definition and Revision to RCS Specific Activity Tech Spec*	*The proposed NuScale TS generally implement the traveler changes modified to reflect the NuScale specific limits.* Changes are reflected in GTS Section 1.1, "Definitions," and Subsection 3.4.8, "RCS Specific Activity"
493-A, Revision 4	*Clarify Application of Setpoint Methodology for LSSS Functions* in STS Section 3.3 and offer the option to	The proposed NuScale TS Sections 3.3 and 5.5 implement Option B of the traveler through inclusion of a Setpoint Program in Section 5.5 (GTS 5.5.10).

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Incorporation
	implement an STS Section 5.5 setpoint program (SP)	
510-A, Revision 2 (not addressed in DCA Regulatory Conformance and Development Report Table C-1)	"Revision to Steam Generator Program Inspection Frequencies and Tube Sample Selection."	SDAA GTS 1.1, 3.4.5, 3.4.9, 5.5.4, and 5.6.5 incorporate TSTF-510
513-A, Revision 3	*Revise PWR Operability Requirements and Actions for RCS Leakage* Detection *Instrumentation*, which affects W-STS 3.4.15.	*The contents of* this *traveler were considered during construction of proposed* GTS Subsection *3.4.7, "RCS Leakage Detection Instrumentation." The NuScale leakage detection methods are significantly different from those used in PWRs* accounted for in W-STS, CE-STS, and W-AP1000-STS.
523-A, Revision 2	*Generic Letter 2008-01, Managing Gas Accumulation*	Affects *GTS 3.5.2, "Decay Heat Removal System"* *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.* *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.*
529-A, Revision 4	Clarify Use and Application Rules. Affects [STS] Section 1.3, "Completion Times," and ISTS Bases Section B 3.0, "SR Applicability," of the B&W-STS, W-STS, and CE-STS.	The changes to W-STS by this traveler are included as appropriate in the [DCA and SDAA] GTS, in • Section 1.3, See response (ML17269A210) to DCA RAI 156-9031, Question 16-3 • LCO 3.0.2 and Bases • LCO 3.0.3 Bases • LCO 3.0.4 Bases • LCO 3.0.5 Bases, See response (ML19072A287) to DCA RAI 157-9033, Question 16-12 and SER Section 16.4.4.1. • SR 3.0.2 Bases • SR 3.0.3 Bases, See responses (ML17257A450, ML19072A287) to DCA RAI 157-9033, Question 16-15, and DCA SER Section 16.4.4.2.
545-A, Revision 3	*TS Inservice Testing* [IST] *Program Removal & Clarify SR*	*The changes described in the TSTF were implemented in appropriate locations*

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Incorporation
	Usage Rule Application to Section 5.5 Testing	*throughout the NuScale US600 DCA and proposed US460 SDAA GTS (Section 1.1, "Definitions," and Subsections 3.1.9, "Boron Dilution Control," 3.4.4, "Reactor Safety Valves (RSVs)," 3.4.6, "CVCS Isolation Valves," 3.5.1, "ECCS," 3.5.2, "Decay Heat Removal System," 3.6.2, "CIVs," 3.7.1, "MSIVs," 3.7.2, "Feedwater Isolation.") The [Inservice Testing] program was incorporated into the GTS Definitions section. SRs applicable to similar components associated with functions or SSCs in the GTS were revised to be consistent with the traveler. Consistent with the TSTF traveler, the [Inservice Testing] program description is not provided in* [GTS Section] *5.5 Programs.* The following statement is included after the first paragraph of [Bases] Subsection B 3.0 SR Applicability: *"SR 3.0.2 and SR 3.0.3 apply in Chapter 5 only when invoked by a Chapter 5 Specification."* See DCA responses to RAI 157-9033, Question 16-14 (ML17257A450) and RAI 512-9634, Question 16-60, Subquestion 49 (ML19010A409).
546, Revision 0	*Revise* average power range [neutron flux] monitor *(APRM) Channel Adjustment SR;* affects *BWR* STS reactor protection system *(RPS) instrumentation.*	*The NuScale design does not incorporate APRMs, however the excore neutron monitoring system that provides a similar function includes 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.* *The allowances provided by the TSTF traveler are incorporated in the proposed NuScale GTS Subsection 3.3.1, "Module Protection System Instrumentation,"* SR 3.3.1.2, surveillance column Note 3b, which applies to Functions: 1a, "Reactor Trip Signal (RTS)" on "High Power Range Linear Power"; and 1b, "Demineralized Water System Isolation (DWSI)" on "High Power Range Linear Power."

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Incorporation
554-A, Revision 0 (not addressed in Regulatory Conformance and Development Report Table C-1)	Revise Reactor Coolant Leakage Requirements	The definition of Pressure Boundary LEAKAGE is modified consistent with the applicable portions of the LEAKAGE definition provided in NUREG-1431, Revision 5 as appropriate for the NuScale design. This change also affects LCO 3.4.5, reactor coolant system Operational LEAKAGE and associated Bases. The NRC staff verified that this traveler applies to NuScale.
557-A, Revision 1	*Spent fuel storage rack neutron absorber monitoring program*	*NuScale design includes spent fuel racks that use neutron absorber material.*
563-A, Revision 0	*"Revise Instrument Testing Definitions to Incorporate the Surveillance Frequency Control Program"*	The proposed modification of the [Channel Operational Test] definition, and other instrumentation surveillance definitions, implements STS change traveler TSTF-563-A; see SER Section 16.4.2.2. The revised definitions of Channel Calibration and Channel Operational Test permit each segment of an instrument loop to have its own Frequency controlled by the SFCP.
565-A, Revision 1	Clarify the Term Operational Convenience in the LCO 3.0.2 Bases to correct an inconsistency between the LCO 3.0.2 and LCO 3.0.3 Bases, and to restore the original intent of the phrase described in Generic Letter (GL) 87-09.	*Addressed as described in [supplemental] response (ML18122A292) to DCA RAI [157-9033, Question] 16-9S1.* This response stated*, "Changes have been made to the proposed NuScale technical specifications to incorporate TSTF 565, Revision 1 into the Bases of LCO 3.0.2 and LCO 3.0.3.* In Revision 2 of DCA Part 4, the Bases for LCO 3.0.2 and LCO 3.0.3 incorporated the changes in this traveler. These Bases in the SDAA match the DCA Bases.
577-A, Revision 1	"Revised Frequencies for Steam Generator Tube Inspections,"	Addressed in GTS administrative controls (Programs and Reports)

16.4.11.2 Approved Technical Specification Task Force Travelers Not Proposed for Incorporation

Table C-1 in the Regulatory Conformance and Development Report, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development report, indicated the following approved travelers were not proposed for incorporation. Italics denote material quoted from Table C-1 of the Regulatory Conformance and Development

Report, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development report.

Table 16.4.11-2 Approved TSTF Travelers Not Proposed for Incorporation

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Non-incorporation
426-A, Revision 5	*Revise or Add Actions to Preclude Entry into LCO 3.0.3—[Risk-Informed] TSTF Initiatives 6b & 6c*	*The topical report does not apply to NuScale. 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* [TS] *End States (WCAP-16294)*, which affects W-STS action requirements	*The topical report does not apply to NuScale. The proposed NuScale TS including operational paradigm is significantly different from that addressed in the* [TSTF].
501-A, Revision 1	*Relocate Stored Fuel Oil and Lube Oil Volume Values to Licensee Control*, which affects W-STS 3.8.3, "Diesel Fuel Oil, Lube Oil, and Starting Air."	*The NuScale design does not require or include safety-related onsite diesel generators. Therefore, no corresponding specification is proposed, and the TSTF traveler is not applicable.*
505-A, Revision 1	*Provide Risk-Informed Extended Completion Times - [Risk-Informed TSTF] Initiative 4b,* which affects W-STS and CE-STS	*NuScale has chosen not to incorporate this traveler into the proposed GTS.*
514-A, Revision 3	*Revise BWR* [STS] *Operability Requirements and Actions for RCS Leakage Detection Instrumentation*	*NuScale leakage detection instrumentation and methods are not similar to those used in [General Electric (GE)] BWRs.* Therefore, changes related to this traveler are not applicable to the NuScale design.
522-A, Revision 0	*Revise Ventilation System [SRs] to Operate for 10 hours per Month*	*The NuScale design does not include credited ventilation systems and no TS are proposed.*
535-A, Revision 1	*Revise Shutdown Margin Definition to Address Advanced Fuel Designs* – Only affects BWR STS definition of SDM.	*Not applicable* to NuScale SDM definition.
541, Revision 1	*Add Exceptions to SRs When the Safety Function is Being Performed*	The applicant withdrew all SR notes motivated by the inferred intent of this traveler, because it identified no practical configurations where such a note would provide the desired relief.
542-A, Revision 1	*Reactor Pressure Vessel Water Inventory Control*	*Not applicable. The NuScale design and operating paradigm does not include operations at reduced inventories or water levels.* *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*

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Non-incorporation
		vessel and fuel remaining submerged in the reactor pool.
547-A, Revision 1	*Clarification of Rod Position Requirements;* affects W-STS Section *3.1 reactivity control specifications related to rod position requirements.*	*The NuScale core design is significantly different from that of large PWRs. The traveler was not incorporated* in Subsections 3.1.4, "Rod Group Alignment Limits"; 3.1.5, "Shutdown Bank Insertion Limits"; and 3.1.6, "Regulating Bank Insertion limits," *because the proposed changes are not necessary.*
551-A, Revision 3	*Revise Secondary Containment* [SRs]; affects *Bases for BWR STS Subsection 3.6.4, "Secondary Containment."*	This traveler is no*t applicable because the NuScale design does not include or credit a secondary containment or similar functional boundary and does not include a corresponding specification.*
567-A, Revision 1	*Add Containment Sump TS to Address GSI-191 Issues*	*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.*
580-A, Revision 1	"Provide Exception from Entering Mode 4 With No Operable RHR Shutdown Cooling"	*Not applicable to the NuScale design*
582-A, Revision 0	"RPV WIC Enhancements"	*Not applicable to the NuScale design*
584-A, Revision 0	"Eliminate Automatic RWCU System Isolation on SLC Initiation"	Not applicable to the NuScale design *(BWR-specific)*
589-A, Revision 0	"Eliminate Automatic Diesel Generator Start During Shutdown"	*Not applicable to the NuScale design*
591-A, Revision 0	"Revise Risk Informed Completion Time (RICT) Program"	*Not applicable to NuScale TS*

16.4.11.3 Unapproved Technical Specification Task Force Travelers under NRC Staff Review and Not Proposed for Incorporation

Table C-1 of the Regulatory Conformance and Development Report, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development report, indicated that no unapproved travelers that were under NRC staff review were proposed for incorporation. The following unapproved travelers that were under NRC staff review were not proposed for incorporation. Italics denote material quoted from Table C-1 of the Regulatory Conformance and Development Report, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development report.

Table 16.4.11-3 Unapproved TSTF Travelers under NRC Staff Review and Not Proposed for Incorporation

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Disposition
521, Revision 0	*Exclusion of Time Constants from Channel Operational Tests in* W-STS *Specifications 3.3.1,* "Reactor Protection System (RPS) Instrumentation," *and 3.3.2*, "Engineered Safety Features Actuation System (ESFAS) Instrumentation."	*The NuScale protective instrumentation does not include functions similar to the Westinghouse PWR design that this traveler is applicable to.*
530, Revision 0	*Clarify SR 3.0.3* and Section B 3.0, Bases for SR 3.0.3, *to be Consistent with Generic Letter 87-09*	The initial version of NuScale US600 DCA SR 3.0.3 had incorporated the content of this traveler, but that content was removed in the response (ML17257A450) to Question 16-15 of RAI 157-9033, because, in 2012, the staff did not accept this traveler for review (ML12207A564). See SER Section 16.4.4.2. In the SDA, the applicant partially adopted editorial changes proposed by this unapproved traveler: *The proposed NuScale SR 3.0.3* [Bases] *incorporates some of the editorial changes to the Bases contained in the Traveler* [ML112620602], except for the unacceptable examples of applying SR 3.0.3 inserted after the first paragraph. These changes mostly replace phrases like "completing the surveillance" with "performing the surveillance" which clarifies the W-STS Bases for SR 3.0.3, and is consistent with the intended meaning of the affected passages.
531, Revision 0	*Revision of Specification 3.8.1*, "AC Sources—Operating," *Required Actions B.3.1 and B.3.2.*	*The NuScale* [US600 and US460 designs do] *not depend on emergency AC power sources and there are no corresponding requirements in the proposed NuScale* [US460 SDAA GTS].
536, Revision 0	*Resolve CE Digital TS Inconsistencies Regarding [Core Protection Calculators] and [Control Element Assembly Calculators]*—affects CE-STS *instrumentation and control specifications*	*The NuScale digital control system does not include* [CE core protection calculators (CPCs) or control element assembly calculators (CEACs)], *however the underlying purpose of the traveler was considered in the development of the Actions and SRs applicable to the corresponding NuScale specifications* (3.3.1, "Module Protection System"; 3.3.2, "Reactor Trip System Logic and Actuation"; 3.3.3, "ESFAS Logic and Actuation"; and 3.3.4, "Manual Actuation Functions.") *The NuScale TS considered the reason for* the proposed changes *to the STS by the TSTF traveler. The specification Actions and*

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Disposition
		SRs do not include Conditions unrelated to system Operability. The staff concludes that this traveler is not adopted because it is not applicable.
537, Revision 0	*Increase containment isolation valve (CIV) Completion Times; update of TSTF-373;* affects CE-STS Subsection 3.6.3	TSTF *traveler is based on a risk-informed technical basis applicable to CE designed plants.* *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—Risk-Informed TSTF (RITSTF) Initiatives 6b & 6c.* Affects B&W-STS Specifications for *containment spray and cooling systems, and emergency ventilation systems.*	*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.*
540, Revision 0	Add Exceptions to SRs When the Safety Function is Being Performed*;* affects BWR specifications for secondary containment and control room ventilation and filtration systems.	*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 DBA.* The staff concludes that this traveler is not adopted because it is not applicable.
566, Revision 0	*Revise Actions for Inoperable BWR [Residual Heat Removal] Shutdown Cooling Subsystems*	*The NuScale passive shutdown cooling design does not include component or configuration issues similar to those addressed in this traveler.*
568, Revision 0	*BWR Drywell to Suppression Chamber pressure and Primary Oxygen Concentration LCO*	*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.*
569, Revision 0	*Revise Response Time Testing Definition*	*The NuScale protection system is different from existing plant designs. This results in the need for different testing boundaries and approaches.*
576, Revision 0	Revise Safety/Relief Valve Requirements	*BWR-specific.* The NuScale design does not include a BWR-like Safety/Relief Valve design.

TSTF Traveler No.	Purpose of Traveler	Applicant’s Rationale for Disposition
585, Revision 2	Provide an Alternative to the LCO 3.0.3 One-Hour Preparation Time	NuScale may consider incorporation as industry and regulatory issues are resolved
592, Revision 2	Revise Automatic Depressurization System (ADS) Instrumentation Requirements	*Not applicable to the NuScale design.* The NuScale design does not include a BWR ADS design.
596, Revision 0	Expand the Applicability of the Surveillance Frequency Control Program (SFCP)	*NuScale is monitoring this proposal for future consideration and adoption.*

16.4.11.4 Withdrawn Previously Approved or Pending Technical Specification Task Force Travelers

Table C-1 of the Regulatory Conformance and Development Report, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development report, indicated the following previously approved or pending travelers had been withdrawn by the TSTF. Italics denote material quoted from Table C-1 of the Regulatory Conformance and Development Report, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development report.

Table 16.4.11-4 Withdrawn Previously Approved or Pending TSTF Travelers

TSTF Traveler No.	Purpose of Traveler	Applicant’s Rationale for Disposition
454, Revision 3	*Staggered Integrated ESFAS Testing (WCAP-15830);* affects *CE-STS ESFAS and ESF surveillance tests*	*The topical report does not apply to NuScale* [design, which] *uses ESFAS and ESF systems that are not similar to those accounted for in the CE-STS.*
515, Revision 0	*Revise Post-Accident Monitoring Instrumentation based on Regulatory Guide 1.97, Rev. 4 and NEDO-33349*, which affects GE-BWR4-STS (NUREG-1433) and GE-BWR6-STS (NUREG-1434) Section 3.3.3.	*Withdrawn by TSTF. Also, the NuScale design does not include any PAM instrumentation that meets the threshold for inclusion in the TS,* as described in SER Section 16.4.1.6.
525, Revision 0	*Post-Accident Monitoring Instrumentation Requirements (WCAP-15981-NP-A).* The NRC declined to review this traveler in a letter dated March 7, 2011 (ML103420584).	Since this *TSTF Traveler is specific to PAM instrumentation selection for Westinghouse designs*, it would not apply to NuScale PAM instrumentation for Type B and C variables. The applicant found no *PAM instrumentation* for Type B and C variables meeting *the threshold for inclusion in GTS,* as described in SER Section 16.4.1.6.
534, Revision 0	*Clarify Application of Pressure Boundary Leakage Definition.* Affects W-STS Subsections 3.4.5 and B 3.4.5, “RCS Operational LEAKAGE.”	The initial version of GTS Section 1.1 (added a sentence to LEAKAGE definition), GTS Subsection 3.4.5, “RCS Operational LEAKAGE,” and Bases Subsection B 3.4.5 had incorporated this traveler, even though

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Disposition
		the TSTF had previously withdrawn it following receipt of NRC staff comments. In a letter dated April 22, 2019, (ML19112A378), NuScale removed the DCA changes associated with this traveler, as requested by the staff. The US460 SDAA GTS Subsection 3.4.5 and Bases Subsection B 3.4.5 are consistent with the DCA.
553, Revision 1	*Add Action for Two Inoperable* [Control Room Emergency Air Temperature Control System (CREATCS)] *Trains;* affects B&W-STS, W-STS, and CE-STS.	*The TSTF withdrew this traveler.* In addition, since the *NuScale design does not credit a CREATCS or a similar* function*, the* [GTS do] *not include a corresponding specification.*
564, Revision 1	*BWR 2.1.1, Safety Limits Safety Limit [Minimum Critical Power Ratio]*	The traveler is not applicable because it *is related to calculating the [Minimum Critical Power Ratio] limit at BWRs. The NuScale design uses a design-specific methodology for calculating core parameters and limits.*
588, Revision 0	Revise list of COLR Methods	This traveler has been withdrawn by TSTF.

16.4.11.5 Disposition of T-Travelers

TSTF travelers have documented some industry-proposed minor improvements to STS that the TSTF chose not to submit for NRC staff review. These travelers are identified with a "T" appended to the sequential TSTF number (e.g., TSTF-494-T). Such a T-traveler may become an approved traveler following staff approval of a license amendment request to incorporate the associated changes into an individual licensee's plant-specific TS. Following are the T-travelers evaluated by the applicant as described in Table C-1 of the Regulatory Conformance and Development Report, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development report. Italics denote material quoted from Table C-1 of the Regulatory Conformance and Development Report, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development report.

Table 16.4.11-5 Disposition of T-Travelers

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Disposition
494-T, Revision 2	*Correct Bases Discussion of Figure B 3.0-1*, which is related to Bases for W-STS LCO 3.0.6	*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.*
502-T, Revision 1	*Correct Containment Isolation Valve Bases Regarding Closed Systems*, which affects W-STS B 3.6.3, "Containment Isolation Valves"	*The proposed NuScale Bases* for US460 SDAA GTS Subsection 3.6.2, "Containment Isolation Valves," *incorporate the corrected wording.*

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Disposition
504-T, Revision 0	*Revised the Main Steam Isolation Valve (MSIV) and Main Feedwater Isolation Valve (MFIV) Specifications to Provide Actions for Actuator Trains*, which affects W-STS 3.7.1, "MSIVs" and 3.7.2, "MFIVs"	*The NuScale MSIV and feedwater isolation valve (FWIV) designs do not incorporate dual actuators such that the TSTF traveler changes should be incorporated*. Therefore, changes related to this traveler were not incorporated in GTS 3.7.1 "MSIVs" and GTS 3.7.2, "Feedwater Isolation."
520-T, Revision 0	*Correct* conflicting statements in CE-STS Subsection B *3.1.4,* "Control Element Assembly (CEA) Alignment," Actions section of Bases for *Required Action A.1.*	*The proposed NuScale TS Bases do not include the conflicting statements.* Therefore, this traveler is not applicable for incorporation in GTS Subsection B 3.1.4, "Rod Group Alignment Limits."
524-T, Revision 0	*Clarify the Application of SR 3.0.2 to SR 3.1.3.2, MTC;* affects Bases for W-STS *SR 3.1.3.2.*	*The NuScale moderator temperature coefficient (MTC) specification SR does not include Notes that correspond directly with those in* W-STS Subsection 3.1.3 and Subsection B 3.1.3, Surveillance Requirements section, *and the NuScale Bases are consistent with the proposed specifications.* Therefore, this traveler is not applicable for incorporation in GTS Subsection B 3.1.3, "Moderator Temperature Coefficient."
526-T, Revision 0	*Clarify* SR section of Bases for STS Subsection concerning surveillance column *Notes regarding momentary transients outside the load band.* Affects W-STS Subsection B 3.8.1, SRs section, discussion of emergency diesel generator load tests required by SR 3.8.1.3 Note 2, SR 3.8.1.14 Note 1, and SR 3.8.1.15 Note 1.	*The NuScale design does not depend on emergency AC power sources and there are no corresponding requirements in the proposed NuScale TS.* Therefore, this traveler is not applicable for incorporation in GTS.
527-T, Revision 0	*Incorporate* Commitments *in Model Applications* for TSTF travelers *as Reviewer's Notes* in Bases of affected STS.	*This traveler describes the use of Reviewer's Notes in the Bases of the published STS. The TSTF 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.* See SER Section 16.4.11.6.
528-T, Revision 0	*Bracket Accident Analysis Discussion in LCO 3.4.4.* Affects B&W-STS (NUREG-1430) Bases Subsection B 3.4.4, "RCS Loops—MODES 1 and 2."	*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.* Therefore, this traveler is not applicable for incorporation in GTS Bases.

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Disposition
532-T, Revision 0	*Eliminate Incorrect Reference to Appendix R in the Remote Shutdown System* (RSS) *Bases*; affects CE-STS Bases Subsection B 3.3.5	*The incorrect reference in the CE*-STS *Bases* Subsection B 3.3.5 References section *is not included in the NuScale Bases for the RSS,* G*TS* Subsection B *3.3.5.* Therefore, this traveler is not applicable for incorporation in the GTS Bases.
533-T, Revision 0	*Remove COLR and PTLR Revision and Date Relocation Provisions Added by TSTF-363, -408, and -419;* affects B&W-STS, W-STS, and CE-STS	Not included. *The NuScale administrative specifications in* [US460 SDAA] GTS Section 5.6 *that describe the COLR and PTLR will include the* current *number, title, date, and NRC-approved document describing the methodology* [by NRC letter and date]. This information in GTS Subsections 5.6.3 and 5.6.4 is bracketed to denote it as part of COL Item 16.1-1.
539-T, Revision 0	*Correction of Post-Accident Monitoring* (PAM) *Instrumentation Bases;* affects B&W-STS, W-STS and CE-STS PAM instrumentation Bases.	*The NuScale* PAM *design does not include any variables that result in inclusion of a PAM technical specification.* See discussion in SER Section 16.4.1.6.
543-T, Revision 0	*Clarify Verification of Time Constants;* affects W-STS Section 3.3, [Instrumentation]	*The NuScale protective instrumentation does not include functions similar to the Westinghouse PWR design to which this traveler is applicable.*
548-T, Revision 0	*Safety Function Determination Program (SFDP) Changes for Consistency;* affects W-STS *Subsection 5.5.8 program description.*	*The NuScale SFDP description provided in* GTS Subsection *5.5.8 is consistent with the intended content as previously described in* B&W-STS, CE-STS, and W-AP1000-STS.
549-T, Revision 0	*Correct* Actions section of Bases for W-STS Subsection 3.2.4, *"Quadrant Power Tilt Ratio (QPTR)."*	*The NuScale design does not include monitoring of a QPTR or QPTR-like variable. The TSTF is specific to an inappropriate wording that existed in the* [W-STS Bases]. This traveler is not incorporated because it does not apply to NuScale GTS Section 3.2 requirements for core operating limits.
550-T, Revision 1	*Correct Misleading Bases Statements in Systems not Required to be Operable in Shutdown Modes;* affects B&W-STS, W-STS, and CE-STS *Bases for* TS *systems that perform a support function for other* TS *systems required to be operable when the facility is shutdown. Specifically, cooling water systems.*	*The NuScale design uses a large reactor pool as the* Ultimate Heat Sink *(UHS) during operational modes and during transition and refueling operations.* The applicability of *Specification 3.5.3,* "UHS," *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* operational modes, which also *provide support functions during shutdown conditions.* Therefore, this traveler is not applicable to the NuScale design and the GTS and Bases.

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Disposition
555-T, Revision 0	*Clarify the Nuclear Instrumentation Bases Regarding the Detection of an Improperly Loaded Fuel Assembly;* affects B&W-STS, W-STS, and CE-STS *Section 3.9 nuclear instrumentation specifications*. This traveler removes the incorrect claim in the STS Bases Section B 3.9.	*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 flux monitors used at PWRs.* Therefore, the GTS includes Specification 3.8.1, "Nuclear Instrumentation." In that GTS Subsection B 3.8.1 does not include a description of an ability to detect an improperly loaded fuel assembly, this traveler is incorporated.
556-T, Revision 1	*Modify TS 3.8.1 and TS 3.8.2 Bases to Address an Open Phase Condition*	This traveler is not applicable to the GTS because the GTS have no LCOs for AC electrical power sources, since *the NuScale design does not credit offsite electrical power.*
558-T, Revision 0	*Clarify SR Bases added by TSTF-523; affects PWR and BWR specifications related to ECCS, decay heat removal,* [residual heat removal, shutdown cooling], *and containment Spray systems.*	Affects GTS *3.5.2, "Decay Heat Removal System (DHRS)"* *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.* *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.* The staff concludes that clarification of the Bases for the SRs to check for gas accumulation is not applicable to the US600 and US460 GTS Bases for SR 3.5.2.2.
559-T, Revision 0	*Revise Bases to Reflect Revised SL Pressure Limit;* affects *BWR STS Bases for Subsections 2.1.1, 3.3.1, and 3.3.6.*	*This traveler resolves an issue specific to the GE design that does not correspond to a NuScale SSC or function.* Therefore, this traveler is not applicable to NuScale GTS.
560-T, Revision 0	*Addition of SRs Note for Turbine Bypass System, LCO 3.7.7 (BWR4 STS) and LCO 3.7.6 (BWR6 STS.)*	This traveler is not applicable because *no corresponding SSC or function in the NuScale design is credited or otherwise would result in inclusion in the GTS.* There is no LCO for a Turbine Bypass System.
561-T, Revision 0	*Bracket LCO 3.5.1 LCO Note in the [improved STS];* affects BWR-STS Subsection 3.5.1, "ECCS," LCO Note	Addition of optional content or reviewer's notes to STS are not applicable or appropriate for DCA GTS submittal. *Only COL-specific content is presented as bracketed content to*

TSTF Traveler No.	Purpose of Traveler	Applicant's Rationale for Disposition
		be modified by applicants referencing the certified design.
562-T, Revision 0	PWR and BWR 3.8.1, AC Sources Operating *Bases Clarification for TS 3.8.1, Required Actions B.3.1 and B.3.2*	*NuScale design does not credit electrical power and therefore does not include corresponding TS.* The NRC staff concludes this traveler is not applicable to US600 and US460 designs.
571-T, Revision 0	Revise Actions for Inoperable Source Range Neutron Flux Monitor.	This traveler is not incorporated. Not applicable to NuScale design.
573-T, Revision 0	Correct RPS Bases for MSIV and TSV Functions.	This traveler is not incorporated. Not applicable to NuScale design.
574-T, Revision 0	Correct TS 3.1.7 Change Made by TSTF-547.	This traveler is not incorporated. Not applicable to NuScale design.
575-T, Revision 0	Revise TS 3.5.3, ECCS - Shutdown, Bases.	This traveler is not incorporated. Not applicable to NuScale design.
578-T, Revision 0	Remove TSTF-493 Option B Specifications.	This traveler is not incorporated. *Not* applicable *to NuScale design.*
579-T, Revision 0	RICT Program Update	This traveler is not incorporated. *Not* applicable *to NuScale TS.*
581-T, Revision 0	Replace References to RG 1.182 with RG 1.160.	This traveler is not incorporated because the NuScale to US600 and US460 GTS Bases already refer to RG 1.160. (There are no references to RG 1.182.)
583-T, Revision 0	TSTF-582 Diesel Generator Variation	This traveler is not incorporated. Not applicable to NuScale design.
586-T, Revision 0	Correct LCO 3.5.2 Note Bases.	This traveler is not incorporated. Not applicable to NuScale design.
587-T, Revision 0	Delete LCO 3.5.2 Note.	This traveler is not incorporated. Not applicable to NuScale design.
594-T, Revision 0	Revise the SR 3.8.3.3 Bases to be Consistent with the SR.	This traveler is not incorporated. Not applicable to NuScale design.
595-T, Revision 0	Correct the SR 3.8.1.9 Bases.	This traveler is not incorporated. Not applicable to NuScale design.

16.4.11.6 Conditions for Adoption of Technical Specification Task Force Changes Included in STS Revisions 4 and 5

The staff reviewed the previously approved TSTF travelers, which are included in the W-STS, the CE-STS, or both, and which the applicant determined contain changes appropriate for the NuScale US600 DC design and US460 SDAA design and associated GTS and Bases, to verify that the applicant had satisfied the associated conditions stated in the traveler for including the changes. These travelers, some of which are marked with an asterisk (*) because they are addressed by TSTF-527-T, are given below. Italics denote material quoted from Table C-1 of DCA Regulatory Conformance and Development Report.

Table 16.4.11-6 Conditions for Adoption of TSTF Changes Included in STS Revisions 4 and 5

TSTF Traveler No.	Purpose of Traveler	Conditions for Adoption/Reviewer's Note(s)
359-A, Revision 9	The STS Revision 2 version of LCO 3.0.4 is revised to allow entry into a MODE or other specified condition in the Applicability while relying on the associated ACTIONS, provided (a) the ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time, (b) a risk assessment has been performed which justifies the use of LCO 3.0.4, or (c) an NRC-approved allowance (i.e., a Required Actions Note) is provided in the Specification to be entered ("LCO 3.0.4.c is applicable"). The STS Revision 2 version of LCO 3.0.4 allows entry into a MODE or a specified condition in the Applicability, while relying on the associated ACTIONS, only if (a) the ACTIONS permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time, or (b) if an NRC-approved allowance is provided in the Specification to be entered (LCO 3.0.4 is not applicable"). SR 3.0.4 is also revised to reflect the concepts of the changes to LCO 3.0.4. The applicability of LCO 3.0.4 and SR 3.0.4 is expanded to include transition into all MODES or other specified conditions in the Applicability, except when required to comply with ACTIONS or that are part of a shutdown of the unit.	In conformance with the commitment required as a condition of adopting TSTF-359-A, in the US600 DCA review, the staff requested that NuScale "perform a qualitative risk assessment for NuScale; the scope of the PWR risk assessment should focus on the transition from MODE 5 to 4, MODE 4 to 3, MODE 3 to 2, and MODE 2 to 1. Also consider unique events to the MODE of interest, such as LTOP protection. Should address 'initiating events of interest' in each Mode of operation, and determine if any systems, if inoperable, or any parameters outside its limits, should preclude entering its Mode of Applicability as allowed by LCO 3.0.4.b." The NuScale US460 SDAA SER Section 16.4.4.1 discusses the adequacy of the applicant's risk assessment and the making of a NuScale design-related change to the Bases for STS LCO 3.0.4. NuScale revised the STS Bases for LCO 3.0.4 so that the US460 SDAA GTS Bases for LCO 3.0.4 state: "...In this context, a unit shutdown is defined as a change in MODE or other specified condition in the Applicability associated with transitioning from MODE 1 to MODE 2, MODE 2 to MODE 3, and MODE 3 not PASSIVELY COOLED to MODE 3 PASSIVELY COOLED." This change was also made to the GTS Bases for LCO SR 3.0.4.
*366-A, Revision 0	Elimination of Requirements for a Post-Accident Sampling System (PASS); inserts Reviewer's Notes in W-STS 5.5.3, PASS	This traveler does not apply to NuScale US600 and US460 designs, which have no PASS. The W-STS Reviewer's Notes state the

TSTF Traveler No.	Purpose of Traveler	Conditions for Adoption/Reviewer’s Note(s)
	CE-STS 5.5.3, PASS	following: “This program may be eliminated based on the implementation of WCAP-14986, Rev. 1, ‘Post Accident Sampling System Requirements: A Technical Basis,’ and the associated NRC Safety Evaluation, dated June 14, 2000, and implementation of the following commitments: 1. [LICENSEE] has developed contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, containment sump, and containment atmosphere. The contingency plans will be contained in [emergency plan implementing procedures] and [implemented with the implementation of the License amendment]. Establishment of contingency plans is considered a regulatory commitment. 2. The capability for classifying fuel damage events at the Alert level threshold has been established for [PLANT] at radioactivity levels of 300 mCi/cc dose equivalent iodine. This capability will be described in emergency plan implementing procedures and implemented with the implementation of the License amendment. The capability for classifying fuel damage events is considered a regulatory commitment. 3. [LICENSEE] has established the capability to monitor radioactive iodine that has been released to offsite environs. This capability is described in our emergency plan implementing procedures. The capability to monitor radioactive iodine is considered a regulatory commitment.” In its response (ML19010A409) to DCA RAI 512-9634, Question 16-60, Subquestion 44, the applicant stated the following, in part: NuScale has addressed this issue in the responses to eRAI listed below: 8837 dated May 18, 2018 (ML18138A383) 9044 dated October 31, 2017 (ML17304B483) 9278 dated May 16, 2018 (ML18136A870)

TSTF Traveler No.	Purpose of Traveler	Conditions for Adoption/Reviewer's Note(s)
		As discussed in DCA SER Section 9.3.2 and SDAA SER Section 9.3.2.4.10, the staff has determined that the requested exemption from 10 CFR 50.34(f)(2)(viii) for the requirement for a PASS should be granted. The exemption provides a regulatory basis for concluding that a PASS does not significantly contribute to NuScale (both US600 and US460 designs) plant safety or accident recovery. Therefore, no TS program is required.
*409-A, Revision 2	Containment Spray System Completion Time Extension (CE NPSD-1045-A); inserts Reviewer's Note in CE-STS Bases Subsection B 3.6.6.A, "Containment Spray and Cooling Systems (Atmospheric and Dual)," Actions section.	The CE-STS Bases Subsection B 3.3.6.A Reviewer's Note for Bases for Required Action A.1 states: "Utilization of the 7 day Completion Time for Required Action A.1 is dependent on the licensee adopting CE NPSD-1045-A (Ref. 6) and meeting the requirements of the Topical Report and the associated Safety Evaluation including the following commitment: '[LICENSEE] has enhanced its Configuration Risk Management Program, [as implemented under 10 CFR 50.65(a)(4), the Maintenance Rule,] to include a Large Early Release Fraction assessment to support this application.' Otherwise, a 72 hour Completion Time applies." This traveler does not apply to NuScale, which has no containment spray system.
*422-A, Revision 1	Change in Technical Specifications End States (CE NPSD-1186); inserts Reviewer's Note in CE-STS Bases Subsections.	Not applicable to NuScale US460 GTS action requirements. The NRC staff determined that the generic TS Chapter 3 default Actions for shutting down a unit after the time allowed for restoring compliance with the LCO has expired, and the final state of the unit, are acceptable. SER Table 16.4.7-1 summarizes these Actions.
425-A, Revision 3	Relocate Surveillance Frequencies to Licensee Control - [Risk-Informed] TSTF Initiative 5b. Affects B&W-STS, W-STS, and CE-STS. Relocates most periodic surveillance frequencies and associated Bases to documentation required by a new program specified in STS Section 5.5, "Administrative	SDAA Part 2, Section 16.1, includes a complete listing of SRs with Frequencies controlled by the Surveillance Frequency Control Program, along with the base Frequency, and the basis of each Frequency.

TSTF Traveler No.	Purpose of Traveler	Conditions for Adoption/Reviewer's Note(s)
	Controls—Programs and Manuals," Subsection 5.5.18, "Surveillance Frequency Control Program."	
*427-A, Revision 2	Allowance for Non-Technical Specification Barrier Degradation on Supported System OPERABILITY. Addition of LCO 3.0.9 and associated Bases includes a Reviewer's Note regarding commitments. Affects Section B 3.0, "LCO Applicability," of B&W-STS, W-STS, and CE-STS.	Adoption of LCO 3.0.9 requires the licensee to make the following commitments: 1. [LICENSEE] commits to the guidance of NUMARC 93-01, Revision 2, Section 11, which provides guidance and details on the assessment and management of risk during maintenance. 2. [LICENSEE] commits to the guidance of NEI 04-08, "Allowance for Non-Technical Specification Barrier Degradation on Supported System OPERABILITY (TSTF-427) Industry Implementation Guidance," March 2006. SER Section 16.4.4.1 discusses the adoption of this traveler. Adoption of W-STS LCO 3.0.9 as GTS LCO 3.0.8 is part of COL Item 16.1-1.

Conclusion for TSTF Traveler Disposition

Based on its review of the Regulatory Conformance and Development Report, as supplemented in Table 4-1 of the US460 Standard Design Approval Technical Specifications Development technical report, and the above evaluation, the NRC staff finds that NuScale's disposition of the listed TSTF travelers is acceptable.

16.4.12 SDAA Technical Issues Affecting GTS and Bases

TR-101310-NP, Revision 1, describes proposed substantive differences between the content of the NuScale US600 certified design (DC) GTS and Bases and the NuScale US460 SDA GTS and Bases. Most of these differences stem from design differences between the US600 certified design and the US460 proposed standard design. The GTS and Bases differences associated with these design differences and related unresolved audit issues are described below.

16.4.12.1 Revised Definitions of GTS Section 1.1 Defined Terms

The RCS minimum temperature for criticality (MTC) specified in the US600 DCA GTS LCO 3.4.2 decreased from 216°C (420°F) to 174°C (345°F) in the US460 SDAA. This expands the RCS temperature range for being in Mode 2, which includes when RCS temperature is at or above the MTC ("Any Indicated Reactor Coolant Temperature (RCT_{indic}) ≥ 345°F") in GTS Table 1.1-1, "MODES." This broadens the Applicability of LCOs that must be met in MODE 2. However, this lower MTC value does not result in changing the typical Completion Time of 36 hours for Required Actions that direct being in Mode 3 with RCS temperature below the MTC ("All RCT_{indic} < 345°F"). This time is based on an expectation that, in most instances, the MCR operator will use the unit's secondary system to cool down the RCS to below the MTC to enter Mode 3 in a controlled manner without challenging unit systems.

The Mode 3 definition also changed to include when RCS temperature is above the MTC if the NPM is being passively cooled. Therefore, there is no RCS temperature upper bound above the MTC for the unit being in Mode 3 safe shutdown with the core being passively cooled. During the audit, NuScale provided a description of the NPM operation in transitioning from full power operation to Mode 2, and to Mode 3, including a cooldown to below the T-2 interlock (< 93.3°C (< 200°F)) in preparation for entry into Mode 4 (letter dated August 2, 2024 (ML24215A190)). This description implied that performance of Required Actions that direct the unit to be in Mode 3 within 36 hours, would only involve entering Mode 3 above the MTC by initiating passive cooling using the DHRS under special circumstances, such as a loss of secondary heat sink. Initiating DHRS (which isolates the secondary system heat sink) in Mode 2 after a manual shutdown or trip of the reactor would quickly cool down the RCS below the MTC but without exceeding the RCS cooldown rate limits specified in the PTLR. SDAA FSAR Chapter 5, Figure 5.4-10, “Primary Coolant Temp with DHRS two trains” the temperature versus time curve shows the RCS cooldown with both DHRS trains in operation does not exceed the typical RCS cooldown rate limit (typically 100°F/hr).

The lower value of the MTC also affects LCO 3.1.1, “SDM,” because the minimum value of the SDM required in the COLR is calculated by assuming the RCS temperature is at the MTC.

16.4.12.2 Pressure-Temperature Limits Report Specification

As discussed in the US460 SDAA SER Section 5.3, which gives the staff’s evaluation of TR-130877, Revision 1, the review of the pressure-temperature limits methodology is incomplete because parts of NuScale’s supporting analyses are based on preliminary information, to be updated by a COL applicant under COL Item 5.3-1. Therefore, the NRC staff finds GTS Subsection 5.6.4 acceptable at the SDAA stage, as further described in SER Section 16.4.10.4.

16.4.12.3 Omission of US600 DCA GTS LCO 3.7.3 In-Containment Secondary Piping Leakage

The NRC staff identified a significant change from the US600 DCA design during its review of the US460 SDA application related to the elimination of breaks in high energy main steam system (MSS) and FWS piping inside the CNV and in the NPM bay. The US600 DCA design (TR-0818-61384-NP, Revision 2, “Pipe Rupture Hazards Analysis Technical Report,” dated July 2019 (ML19212A682)) used leak-before-break (LBB) methodology while the US460 SDA design (TR-121507-NP, Revision 1, “Pipe Rupture Hazards Analysis dated October 31, 2023 (ML23304A316)) does not use LBB.

NuScale’s approach in the US600 DCA credited LBB methods to justify exclusion of high energy line breaks in large-diameter secondary piping, namely MSS and FWS piping. This was noted in section 2.2.2.1.1 of TR-0818-61384-P, Revision 2. NuScale analyzed MSS and FWS piping for LBB and showed that this piping meets the criteria in the Section 2.2.5 discussion related to SRP Section 3.6.3, “Leak-Before-Break Evaluation Procedures,” as part of the US600 DCA. This was also reflected in Section 3.6.3.2 of the US600 DCA SER. However, in the US460 SDAA, NuScale did not use LBB methodology for MSS and FWS piping inside the CNV and in the NPM bay, or include an associated GTS LCO limit on secondary piping leakage in the GTS. This is a significant change in NuScale’s approach.

The staff asked NuScale (ML24062A001) to submit an additional discussion on the basis for break exclusion for the high energy piping (and particularly the MSS and FWS piping) inside containment. In its letter dated August 27, 2024 (ML24240A141), NuScale provided summary information for the requested stress analysis, which used branch technical position (BTP) 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment,” for the MSS and FWS piping, as described in the US460 SDAA. The results showed that MSS and FWS meet ASME stress criteria as well as BTP 3-4 stress criteria for break exclusion (dynamic effects-jet impingement, pipe whip, etc.). Therefore, based on the applicant’s analysis results, the staff finds that omission of an LCO subsection for a secondary piping leakage limit in GTS Section 3.7 is acceptable.

16.4.12.4 Decrease in Thermal Power at which the Requirements of LCO 3.2.1, Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$), and LCO 3.2.2, Axial Offset Are Applicable.

The US460 SDAA expands the applicability of GTS LCO 3.2.1 and LCO 3.2.2, compared to the applicability of the corresponding LCOs in the GTS of the US600 certified design. The US460 SDAA GTS require the LCO limits to be met at or above 20 percent RTP, rather than the 25 percent RTP value of the US600 DCA GTS, expanding the applicability of the requirements to a larger region of the operating power levels. Since this change is more restrictive than the applicability of the equivalent LCOs in the US600 certified design GTS, the staff concludes the change is acceptable.

16.4.12.5 Changes in LCO 3.3.1, “Module Protection System (MPS)”

Section 3.3.4, “Changes to Limiting Condition for Operation 3.3.1, Module Protection System,” in the US460 Standard Design Approval Technical Specifications Development report states, “Modifications are made to actuation logic to align with safety analyses and design changes consistent with the increased reactor rated thermal power, safety analyses, refinements in operational intentions, and lessons learned since the submittal of the [US600] DCA technical specifications.” One of these changes to the MPS requirements of the US600 DCA GTS proposes to modify the applicability footnotes of RTS instrumentation functions and reactor coolant boron dilution source isolation (DWSI) actuation instrumentation functions in Modes 2 and 3 in GTS Table 3.3.1-1. The affected MPS instrumentation functions are listed in SER Section 16.4.6(8).

Footnote (a) (“When capable of CRA withdrawal.”) in the US600 DCA GTS Table 3.3.1-1 is replaced by the corresponding US460 SDAA Footnotes (a) and (b), which allow a single CRA to be capable of withdrawal (the associated CRDM is energized) without requiring the associated RTS and DWSI actuation instrumentation functions to be operable. The US460 SDAA TS development report Section 3.3.4 cited an operational need to make a CRA capable of withdrawal while disconnecting the CRA from its associated control rod drive mechanism in preparation for entering Mode 4 and subsequent refueling operations in Mode 5, stating, “This change is necessary to allow energization of a portion of the control rod drive system (CRDS) in MODE 3 when preparing for module disassembly.” These MPS instrumentation functions are not required to mitigate a reactivity excursion transient when uncoupling and coupling a CRA to its control rod drive shaft because the reactor trip function is fulfilled by RCS boron concentration above the SDM requirements established in LCO 3.1.1 and the SDM definition. The applicant stated in an audit item response (ML24326A126) that:

> [V]erification that a control rod drive shaft is uncoupled or coupled is performed with the [RPV] and [CNV] fully assembled using the associated [CRDM]. A single CRDM is energized and utilized to attempt to lift an individual CRA to verify the CRA is unlatched before disassembly and latched after reassembly. During that evolution, current traces are used to determine the status of the control rod drive shaft, coupled or uncoupled. The phrasing of the CRA related footnotes [in the

GTS] allows [a] single CRDM to be energized to support verification of control rod drive shaft coupling, or uncoupling, respectively.

Based on the applicant's response, the staff concludes that the GTS footnote condition of "when capable of withdrawal of more than one CRA" is an acceptable design-specific applicability provision.

16.4.12.6 Steam Generator Program

The steam generator tubes in the US460 design have higher (reactor coolant) pressure on the outside of the tubes and are therefore susceptible primarily to collapse or buckling rather than burst. Steam generators in the operating fleet have reactor coolant inside the tubes and are susceptible primarily to burst. As a result, certain criteria in the US460 GTS are different than the corresponding criteria in the STS or are the same as the STS but are based on a design-specific analysis. The technical evaluations of these criteria are in SER Section 5.4.4.6.

GTS Section 5.5.4.b.1 (steam generator tube structural integrity performance criterion) has a safety factor of greater than 2.0 against tube collapse or buckling rather than the W-STS safety factor of 3.0 against burst. This is based on requirements related to external pressurization in the ASME Code. The tube plugging criterion in GTS Section 5.5.4.c, which is 40 percent through-wall, enclosed in brackets, is the same as in the STS and the US600 design. However, an analysis was performed by NuScale to justify this value for collapse based on the US460 design, and a COL applicant is required to confirm this value, or justify another value, to account for any differences from the US460 standard design or tube plugging analysis (COL Item 16.1-1). GTS Section 5.5.4.d.2 has a maximum interval of 72 effective full power months (EFPM) between inspections of 100 percent of the tubes. This is less than the 96 EFPM maximum interval in the STS for the same tube material due to the lack of operating experience for the US460 design. Based on evaluation of the criteria for steam generator tube integrity, as discussed in SER Section 5.4.4.6, the staff concludes these specifications are acceptable.

16.5 Combined License Information Items

FSAR Section 16.1.1 describes COL Item 16.1-1, as follows:

> An applicant that references the NuScale Power Plant US460 standard design will provide the final plant-specific information identified by [ ] in the generic Technical Specifications and generic Technical Specification Bases.

Table 16.5-1 of this SER lists and describes COL information sub-items, as enumerated by the staff, related to bracketed information in SDAA Part 4, GTS and Bases.

Table 16.5-1 NuScale COL Item 16.1-1

Sub-Item Number	COL Item 16.1-1 Sub-item Description	GTS or Bases Location
1	Confirm or update each listed critical heat flux ratio [reactor core Safety Limit correlation CHFR value]	2.1.1.1
2.1	Confirm Setpoints will be in owner-controlled requirements manual	B 2.1.1 App

Sub-Item Number	COL Item 16.1-1 Sub-item Description	GTS or Bases Location
3.1.1 3.1.2	Replace “[2017 edition]” with the site-specific edition of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code in reference 2 in the References section of the Bases .... Replace “[2017 edition]” with the site-specific edition of Section XI, Article IWA-5000 of the ASME, Boiler and Pressure Vessel Code in reference 3 in the References section of the Bases..	 B 2.1.2 Ref B 2.1.2 Ref
4.1 4.2 4.3 4.4 4.5 4.6 4.7 4.8	Confirm adoption of LCO 3.0.8 and correctly phrase B 3.0 LCOs .... Confirm adoption of LCO 3.0.8 and correctly phrase LCO 3.0.1 .... Confirm performance of risk assessment if adopting LCO 3.0.8..................... Confirm performance of risk assessment if adopting LCO 3.0.8..................... Confirm if adopting LCO 3.0.8, and include LCO 3.0.8, if applicable .... Confirm if adopting LCO 3.0.8, and include LCO 3.0.8, if applicable.............. Confirm initiating event categories if adopting LCO 3.0.8 .... Confirm licensee commitment to NEI 04-08 if adopting LCO 3.0.8.................	B 3.0 LCOs LCO 3.0.1 LCO 3.0.8 B LCO 3.0.8 LCO 3.0.8 B LCO 3.0.8 B LCO 3.0.8 B LCO 3.0.8
5.1 5.2	Update reference to NuScale Reload Safety Evaluation Methodology report . Update reference to NuScale Reload Safety Evaluation Methodology report .	B 3.1.8 ASA B 3.1.8 Ref
2.2 2.3 6 2.4 2.5	Confirm Setpoints will be in owner-controlled requirements manual .... Confirm Setpoints will be in owner-controlled requirements manual .... Confirm use of sensor response time allocations (COL Item 16.1-3) .... Confirm Setpoints will be in owner-controlled requirements manual .... Confirm Setpoints will be in owner-controlled requirements manual	B 3.3.1 Bkgd B 3.3.1 ASA B SR 3.3.1.3 B 3.3.2 B 3.3.3
3.2 20.1	Confirm applicable edition of ASME Boiler and Pressure Vessel Code........... Update reference to Pressure and Temperature Limits Methodology report...	B 3.4.3 Ref B 3.4.3 Ref
27.1 27.2	Confirm minimum relief capacity value of “[83,400 lb/hr]” for RSV1 .... Confirm minimum relief capacity value of “[87,500 lb/hr]” for RSV2	B 3.4.4 Bkgd B 3.4.4 Bkgd
3.3	Confirm applicable edition of ASME Boiler and Pressure Vessel Code.........	B 3.4.4 Ref
7 8.1 8.2 9 3.4	Confirm correct Revisions of references .... Confirm use of “required” for automatic actuation valves................................ Confirm use of valves with actuators with pressurized accumulators .... Confirm correct Revisions of references .... Confirm applicable edition of ASME, Boiler and Pressure Vessel Code..........	B 3.4.5 Ref SR 3.4.6.1 B SR 3.4.6.1 B 3.4.9 Ref B 3.4.9 Ref
21 8.3 8.4 22 8.5 8.6 8.7 8.9 23.1 8.10 8.11 23.2	Confirm value of “[eight]” hour timer for ECCS actuation after reactor trip ...... Confirm use of “required” for automatic actuation valves................................ Confirm use of valves with actuators with pressurized accumulators .... Confirm calculated peak containment internal pressure (P_a) .... Confirm use of “required” for automatic actuation valves................................ Confirm use of valves with actuators with pressurized accumulators .... Confirm use of “required” for automatic actuation valves................................ Confirm use of valves with actuators with pressurized accumulators .... Confirm applicable edition of ASME OM Code .... Confirm use of “required” for automatic actuation valves................................ Confirm use of valves with actuators with pressurized accumulators .... Confirm applicable edition of ASME OM Code	B 3.5.1 Bkgd SR 3.5.2.1 B SR 3.5.2.1 B 3.6.1 SR 3.6.2.1 B SR 3.6.2.1 SR 3.7.1.1 B SR 3.7.1.1 B 3.7.1 Ref SR 3.7.2.1 B SR 3.7.2.1 B 3.7.2 Ref
24	Update reference to Accident Source Term Methodology report.....................	B 3.8.2 Ref
10 11	Insert statement describing Site Location .... Insert statement describing Site and Exclusion Boundaries	4.1 4.1.1

Sub-Item Number	COL Item 16.1-1 Sub-item Description	GTS or Bases Location
12 25.1 25.2 25.3 25.4	Insert statement describing Low-Population Zone .... Confirm value of “[800]” ppm as minimum soluble boron concentration .... Confirm value of “[10.00]” as nominal center-to-center distance .... Confirm value of “[19]” feet related to inadvertent draining of spent fuel pool . Confirm use of “[600]” limit on number of fuel assemblies in spent fuel pool...	4.1.2 4.3.1.b 4.3.1.d 4.3.2 4.3.3
13.1.1 13.2.1 13.1.2 13.3 13.4 13.5 13.1.3 13.2.2	Replace “[Plant Manager]” with equivalent site-specific title .... Replace “[Shift Manager (SM)]” with equivalent site-specific title .... Replace “[Plant Manager]” with equivalent site-specific title .... Replace “[specified corporate officer]” with equivalent site-specific title .... Replace “[operations manager]” with equivalent site-specific title .... Update reference to NuScale Control Room Staffing Plan report................... Replace “[Plant Manager]” with equivalent site-specific title .... Replace “[shift manager]” with equivalent site-specific title.............................	5.1.1 5.1.2 5.2.1 5.2.1 5.2.2 5.3.1 5.5.1.c.2 5.7.2.a.1
14	Confirm percent flaw depth criterion for steam generator tube plugging	5.5.4.c
15.1 15.2 16 17	Confirm Containment Leakage Rate Testing Program (Option A or B) .... Confirm calculated peak containment internal pressure under Option B .... Confirm version of NuScale Instrument Setpoint Methodology....................... Confirm Spent Fuel Storage Rack Neutron Absorber Monitoring Program ...	5.5.9 5.5.9 5.5.10.b 5.5.12
18	Replace “[FSAR/QA Plan]” with equivalent site-specific title	5.2.1
19	Confirm or update each listed [document describing the NRC reviewed and approved analytical methods used to determine the core operating limits, and the supported LCOs that reference the limits in the COLR]............................	5.6.3.b
20.2	Confirm Revision to TR-130877, “PTL Methodology”	5.6.4
26	Confirm any plant-specific steam generator reporting requirements	5.6.5
Abbreviations: App Applicability section of a Bases Subsection ASA Applicable Safety Analyses section of a Bases Subsection B Bases Subsection label prefix Bkgd Background section of a Bases Subsection Ref References section of a Bases Subsection		

FSAR Section 16.1.1 describes COL Item 16.1-2 as follows:

> An applicant that references the NuScale Power Plant US460 standard design will prepare and maintain an owner-controlled requirements manual that includes owner-controlled limits and requirements described in the Bases of the Technical Specifications or as otherwise specified in the FSAR.

FSAR Section 16.1.1 describes COL Item 16.1-3 as follows:

> An applicant that references the NuScale Power Plant US460 standard design, and uses allocations for sensor response times based on records of tests, vendor test data, or vendor engineering specifications as described in the bases for Surveillance Requirement 3.3.1.3, will do so for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

The staff finds that the proposed COL information related to US460 SDAA Part 4 is appropriate for completion by a COL applicant referencing the NuScale Power Plant US460 standard design approval.

16.6 Conclusion

The staff finds that the NuScale US460 GTS and Bases comply with 10 CFR 50.34, “Contents of Applications; Technical Information”; 10 CFR 50.36; and 10 CFR 50.36a and are therefore acceptable.