December 29, 2022 Docket No. 52-050

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of the NuScale Standard Design Approval Application Part 2 – Final Safety Analysis Report, Chapter 16, "Technical Specifications," Revision 0

REFERENCES:
1. NuScale letter to NRC, "NuScale Power, LLC Submittal of Planned Standard Design Approval Application Content," dated February 24, 2020 (ML20055E565)
2. NuScale letter to NRC, "NuScale Power, LLC Requests the NRC staff to conduct a pre-application readiness assessment of the draft, 'NuScale Standard Design Approval Application (SDAA),'" dated May 25, 2022 (ML22145A460)
3. NRC letter to NuScale, "Preapplication Readiness Assessment Report of the NuScale Power, LLC Standard Design Approval Draft Application," Office of Nuclear Reactor Regulation dated November 15, 2022 (ML22305A518)
4. NuScale letter to NRC, "NuScale Power, LLC Staged Submittal of Planned Standard Design Approval Application," dated November 21, 2022 (ML22325A349)

NuScale Power, LLC (NuScale) is pleased to submit Chapter 16 of the Standard Design Approval Application, "Technical Specifications," Revision 0. This chapter supports Part 2, "Final Safety Analysis Report," (FSAR) of the NuScale Standard Design Approval Application (SDAA) (Reference 1). NuScale submits the chapter in accordance with requirements of 10 CFR 52 Subpart E, Standard Design Approvals. As described in Reference 4, the enclosure is part of a staged SDAA submittal. NuScale requests NRC review, approval, and granting of standard design approval for the US460 standard plant design.

From July 25, 2022 to October 26, 2022, the NRC performed a pre-application readiness assessment of available portions of the draft NuScale FSAR to determine the FSAR's readiness for submittal and for subsequent review by NRC staff (References 2 and 3). The NRC staff reviewed draft Chapter 16. The NRC did not identify readiness issues with the chapter.

Enclosure 1 contains SDAA Part 2 Chapter 16, "Technical Specifications," Revision 0. Enclosure 2 contains the associated technical report.

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

If you have any questions, please contact Mark Shaver at 541-360-0630 or at mshaver@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 29, 2022.

Sincerely,

Carrie Fosaaen
Senior Director, Regulatory Affairs
NuScale Power, LLC

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Enclosure 1: SDAA Part 2 Chapter 16, “Technical Specifications,” Revision 0
Enclosure 2: “US460 Standard Design Approval Technical Specifications Development,” TR-101310-NP, Revision 0

Enclosure 1:

SDAA Part 2 Chapter 16, “Technical Specifications,” Revision 0

NuScale US460 Plant
Standard Design Approval Application

Chapter Sixteen
Technical Specifications

Final Safety Analysis Report

Revision 0

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CHAPTER 16 TECHNICAL SPECIFICATIONS

16.1 Technical Specifications

16.1.1 Introduction to Technical Specifications

Technical Specification Content

The NuScale Power, LLC (NuScale), generic technical specifications (GTS) meet the 10 CFR 50.36 and 10 CFR 50.36a requirements. The technical specifications evolved from the GTS submitted with the NuScale US600 Design Certification Application (Reference 16.1-1) and approved by the NuScale US600 Standard Design Approval (Reference 16.1-2 and Reference 16.1-3). These revised GTS were developed consistent with the Improved Standard Technical Specification (ISTS) format and content typified in NUREG-1431, Revision 5 and NUREG-1432, Revision 5. The content differs from the ISTS and the previous NuScale US600 technical specifications as necessary to reflect technical differences between large light water reactor (LWR) and the NuScale US600 design, and the NuScale Power Plant US460 standard design. For example, Table 1.1-1 of the NuScale GTS lists five MODES that are distinct from those provided in ISTS for pressurized water reactor or boiling water reactor designs, and the US460 standard design incorporates an emergency core cooling system supplemental boron function.

The NuScale Power Plant US460 standard design is a single facility that is comprised of up to six individual NuScale Power Modules (NPMs), each of which constitutes a nuclear steam supply system as described in Section 1.2. Individual NPMs are installed in an operating position during power generation and transferred to a common refueling location when refueled as described in Section 9.1. The technical specifications, bases, and some related discussions use the term “unit” rather than the NuScale standard terminology of “NPM.” This term is used to maintain alignment with industry standard technical specification terminology.

The NuScale GTS are constructed to address the NuScale Power Plant US460 standard design by providing operating limitations for an individual NPM.

The majority of the NuScale GTS address conditions applicable to an individual NPM. However, some systems and parameters are applicable to multiple NPMs. While individually specified, these limits may be applicable to more than one NPM at the same time. Clarifications have been included in the associated bases to address multi-module technical specification interactions.

Selection Criteria for Limiting Conditions for Operation

Limiting conditions for operation (LCOs) are included in the NuScale GTS consistent with the screening criteria provided in 10 CFR 50.36(c)(2)(ii). These selection criteria are

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

2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

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

Information regarding the screening process employed during development and the differences in content between the US600 and US460 designs are provided in Technical Report TR-101310 (Reference 16.1-4). Where relevant to the operations of the facility, the results are incorporated into the bases of the technical specification.

NuScale considered risk-informed technical specification development approaches consistent with Regulatory Guide 1.177, and the guidance in NEI 04-10 and NEI 06-09 (Reference 16.1-5 and Reference 16.1-6). However, as a new design with low overall evaluated risk but no applicable operating experience, completion times and surveillance intervals are proposed that are generally consistent with corresponding industry systems or functions as included in the NUREG Standard Technical Specifications.

The NuScale Power Plant US460 standard systems used for MODE reduction are those used to respond to design basis events, e.g., ECCS and DHRS. This differs from those systems commonly used in large LWR designs. Therefore, LCO requirements for those large LWR systems are not required by the GTS.

For example, in typical large pressurized water reactors the safety-related shutdown cooling function is required to operate and reduce primary system temperatures, and to support low-temperature overpressure protection. Those shutdown cooling systems are cooled by an intermediate closed-loop system to transfer decay heat to other cooling systems that transfer the decay heat to the ultimate heat sink (UHS). Those safety-related functions are required to be operable and included in the technical specifications by criteria two or three of 10 CFR 50.36(c)(2)(ii).

NPM shutdown cooling is accomplished using passive convection and conduction of decay heat from the flooded portion of the containment through the containment vessel wall directly to the UHS. The low temperature overpressure protection function is provided by instrumentation and valves that perform the emergency core cooling system function during power operations. These NuScale systems are included in the proposed GTS in accordance with 10 CFR 50.36(c)(2)(ii). Distinct shutdown cooling pumps, valves, low temperature overpressure protection valves, and intermediate cooling loops are not credited or provided; therefore, they are not included for the NuScale GTS.

Completion Times and Surveillance Frequencies

When appropriate, the completion times and surveillance frequencies specified in the NuScale GTS are consistent with times and frequencies applied to similar actions and surveillance requirements (SRs) from the NuScale US600 technical specifications, NUREG-1431, and NUREG-1432. The bases for completion times and surveillance frequencies are described in the associated bases or in the Surveillance Frequency Control Program (SFCP) described in technical specification 5.5.11.

Table 16.1-1 provides the initial surveillance test frequencies to be incorporated into the Surveillance Frequency Control Program 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 of the surveillance test.

Incorporation of Improved Standard Technical Specification Change Travelers

Industry change travelers issued since publication of Revision 5 of the ISTS and available on the NRC Agencywide Documents Access and Management System (ADAMS) system were reviewed in the development of the NuScale GTS. Travelers were incorporated into the NuScale GTS or used as a basis for similar NuScale situations as described in the conformance report (Reference 16.1-4). The travelers considered in development of the NuScale GTS are listed in that report.

The GTS are intended to be used as a guide in the development of the plant-specific technical specifications. Preliminary information was provided in single brackets [ ]. Combined license applicants referencing the NuScale Power Plant US460 standard design are required to provide the final plant-specific information.

COL Item 16.1-1: 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.

COL Item 16.1-2: 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.

COL Item 16.1-3: 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.

16.1.2 References

16.1-1 NuScale Power, LLC, NuScale Standard Plant Design Certification Application, Rev. 5, July, 29, 2020 (ML20225A044), Portland, OR.

16.1-2 NuScale Power, LLC, "NuScale Power, LLC Request for Standard Design Approval based on the NuScale Standard Plant Design Certification Application," LO-0720-70936, July 13, 2020.

16.1-3 Anna H. Bradford, NRC, "Standard Design Approval for the NuScale Power Plant Based on the NuScale Standard Plant Design Certification Application," (ML20247J564) September 11, 2020.

16.1-4 NuScale Power, LLC, "Technical Specifications Regulatory Conformance and Development Technical Report," TR-101310-NP, Revision 0.

16.1-5 Nuclear Energy Institute, "Risk-Informed Technical Specifications Initiative 5b-Risk-Informed Method for Control of Surveillance Frequencies," NEI 04-10, Revision 1, April 2007.

16.1-6 Nuclear Energy Institute, “Risk-Informed Technical Specifications Initiative 4b-Risk-Managed Technical Specifications (RMTS) Guidelines," NEI 06-09, Rev. 0-A, November 2006.

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies

Surveillance Requirement	Base Frequency	Basis
3.1.1.1	24 hours	The Frequency of 24 hours is based on the generally slow change in required boron concentration and the low probability of an accident occurring without the required shutdown margin (SDM). This allows time for the operator to collect the required data, which includes performing a boron concentration analysis, and to complete the calculation.
3.1.2.1	31 effective full-power days (EFPDs)	The required subsequent Frequency of 31 EFPDs, following the initial 60 EFPDs after exceeding 5% rated thermal power (RTP), is acceptable based on the slow rate of core changes due to fuel depletion and the presence of other indicators (e.g. axial offset (AO)) monitored by the core monitoring system for prompt indication of an anomaly.
3.1.4.1	12 hours	Verification that individual control rod assembly (CRA) positions are within alignment limits at a 12 hour Frequency provides a history that allows the operator to detect a CRA that is beginning to deviate from its expected position. The specified Frequency takes into account other CRA position information that is continuously available to the operator in the control room so that during actual rod motion deviations can immediately be detected.
3.1.4.2	92 days	The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the CRAs.
3.1.5.1	12 hours	Because the shutdown CRAs are not moved during routine operation, except as part of planned surveillances, verification of shutdown CRA position at a Frequency of 12 hours is adequate to ensure that the shutdown CRAs are within their insertion limits. Also, the Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.
3.1.6.1	12 hours	Verification of the regulating group insertion limits at a Frequency of 12 hours is sufficient to detect a CRA that may be approaching the insertion limits because, normally, very little rod motion is expected to occur in 12 hours.
3.1.8.1	30 minutes	Verification that the THERMAL POWER is ≤ 5% RTP ensures that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS ensures that the initial conditions of the safety analyses are not violated.
3.1.8.2	24 hours	The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.
3.1.9.1	31 days	A 31 day Frequency is considered reasonable in view of other administrative controls that ensure a misconfiguration of the chemical and volume control system (CVCS) makeup pump demineralized water flow path is unlikely. Also, the Frequency takes into account other information available in the control room for the purpose of monitoring the status of CVCS makeup pump demineralized water flow path configuration.
3.1.9.2	24 months	The 24 month Frequency is based on the potential for unplanned plant transients if the surveillances were performed with the unit at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment. The actuation logic is tested as part of engineered safety features actuation system (ESFAS) actuation and logic testing, and valve performance is monitored as part of the Inservice Testing Program.

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies (Continued)

Surveillance Requirement	Base Frequency	Basis
3.1.9.3	31 days	The 31 day Frequency of this SR was developed considering the known stability of stored borated water and the low probability of an undesired source of diluting pure water. The Frequency takes into account administrative controls that ensure changes to boron concentration are performed in accordance with written procedures. This Frequency also considers the size of the boric acid storage tank and the normally expected demands of boric acid for plant operations.
3.1.9.4	24 months	The 24 month Frequency is based on the potential for unplanned plant transients if the surveillances were performed with the unit at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.
3.1.9.5	12 hours	A 12 hour Frequency is considered reasonable in view of other administrative controls that ensure a misconfiguration of the CVCS flow path is unlikely. Also, the Frequency takes into account other information available in the control room for the purpose of monitoring the status of CVCS configuration.
3.2.1.1	31 EFPD	The 31 EFPD Frequency is acceptable because the power distribution changes relatively slowly over this amount of fuel burnup. Accordingly, this Frequency is short enough that the enthalpy rise hot channel factor limit cannot be exceeded for any significant period of operation.
3.2.2.1	7 days	The Frequency of 7 days is adequate considering that the AO is monitored by a computer and any deviation from requirements is alarmed.
3.3.1.1	12 hours	The Frequency of 12 hours is based on industry operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.
3.3.1.2	12 hours	The Frequency of every 12 hours is adequate. It is based on industry operating experience, considering industry instrument reliability and operating history data for instrument drift. Together with engineering judgment, these factors demonstrate that a difference between the calorimetric heat balance calculation and the power range channel output of more than +1% RTP is not expected in any 12 hour period.
3.3.1.3	24 months	As appropriate, each channel's response must be verified every 24 months. This test measures the portion of the response time from the sensor to receipt in the digital module protection system. The digital processing portions of the module protection system are assumed to function in less than 1 second consistent with their design. Equipment actuation is measured through testing required by 3.3.2, 3.3.3, and LCO surveillance requirements associated with the actuated components. Response times cannot be determined during unit operation because sensor inputs are required to be varied to measure response times. Industry experience has shown that these components usually pass this surveillance when performed at the 24 months Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
3.3.1.4	24 months	The Frequency is based on consideration of the design reliability and performance characteristics of the equipment.
3.3.1.5	24 months	The 24 month Frequency is acceptable based on consideration of the design reliability of the equipment.

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies (Continued)

Surveillance Requirement	Base Frequency	Basis
3.3.2.1	24 months	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.
3.3.2.2	24 months	The Frequency of 24 months is justified based on consideration of the reliability of the equipment and industry operating experience with similar equipment.
3.3.2.3	24 months	The 24 month Frequency is based on the potential for unplanned plant transients if the surveillances were performed with the unit at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.
3.3.2.4	24 months	The Frequency of 24 months is based on the known reliability of similar functions in licensed designs and the multidivisional redundancy available, and has been shown to be acceptable through industry operating experience.
3.3.3.1	24 months	This Frequency of 24 months 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.
3.3.3.2	24 months	The 24 month Frequency is based on the potential for unplanned plant transients if the surveillances were performed with the unit at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.
3.3.3.3	24 months	This Frequency of 24 months 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.
3.3.3.4	24 months	The 24 month Frequency is based on the potential for unplanned plant transients if the surveillances were performed with the unit at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.
3.3.3.5	24 months	The Frequency of 24 months is based on the known reliability of similar functions in licensed designs and the multidivisional redundancy available, and has been shown to be acceptable through industry operating experience.
3.3.4.1	24 months	The Frequency of 24 months is based on the known reliability of similar functions in licensed designs and the potential for unplanned plant transients if the surveillances were performed with the unit at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.
3.4.1.1	12 hours	Because Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady-state condition following load changes and other expected transient operations. The 12 hour interval has been shown by industry operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies (Continued)

Surveillance Requirement	Base Frequency	Basis
3.4.1.2	12 hours	Because Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Frequency for reactor coolant system (RCS) cold temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by industry operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.
3.4.2.1	12 hours	The SR to verify all RCS temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine surveillances that are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and ensure that the minimum temperature for criticality is met as criticality is approached.
3.4.3.1	30 minutes	This Frequency of 30 minutes is considered reasonable in view of the control room indication available to monitor RCS status. Also, because temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.
3.4.5.1	72 hours	The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.
3.4.5.2	72 hours	The Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.
3.4.6.1	12 hours	The Frequency of 12 hours is based on the availability and reliability of the automatically monitored pressure alarms to detect a change in accumulator pressure and the similarity of the test to a CHANNEL CHECK as performed throughout existing large plant designs. The test verifies the accumulator pressure and thereby assures the OPERABILITY of the valves, as well as the status of the automatically monitored pressure alarms.
3.4.6.3	24 months	The Frequency of 24 months is based on equipment reliability and the need to perform this surveillance during periods in which the plant is shutdown for refueling to prevent any upsets of plant operation.
3.4.7.1	12 hours	The Frequency of 12 hours is based on industry operating experience. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.
3.4.7.2	12 hours	The Frequency of 12 hours is based on industry operating experience that demonstrates channel failure of pressure monitors is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.
3.4.7.3	12 hours	The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.
3.4.7.4	92 days	The Frequency of 92 days considers instrument reliability, and industry operating experience has shown that it is proper for detecting degradation.
3.4.7.5	92 days	The Frequency of 92 days considers instrument reliability, and industry operating experience has shown that it is proper for detecting degradation.
3.4.7.6	24 months	The Frequency of 24 months considers instrument reliability, and industry operating experience that has proven that this Frequency is acceptable.

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies (Continued)

Surveillance Requirement	Base Frequency	Basis
3.4.7.7	24 months	The Frequency of 24 months is based on the assumption of a 30 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.
3.4.7.8	24 months	The Frequency of 24 months considers instrument reliability, and industry operating experience that has proven that this Frequency is acceptable.
3.4.8.1	14 days	The 14 day Frequency is adequate to trend changes in the noble gas specific activity level and based on the low probability of an accident occurring during this time period.
3.4.8.2	14 days	The 14 day Frequency is adequate to trend changes in the iodine activity level and based on the low probability of an accident occurring during this time period.
3.4.10.1	24 months	The 24 month Frequency is based on equipment reliability and the need to perform these surveillances under the conditions that apply during a unit outage and the potential for unplanned plant transients if the surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.
3.5.1.1	24 months	The 24 month Frequency is based on equipment reliability and the need to perform these surveillances under the conditions that apply during a unit outage and the potential for unplanned plant transients if the surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.
3.5.2.1	12 hours	The Frequency of 12 hours is based on the availability and reliability of the automatically monitored pressure alarms to detect a change in accumulator pressure and the similarity of the test to a CHANNEL CHECK as performed throughout existing large plant designs. The test verifies the accumulator pressure and thereby assures the OPERABILITY of the valves, as well as the status of the automatically monitored pressure alarms.
3.5.2.2	24 hours	The 24 hour Frequency is based on equipment reliability and the expected low rate of gas accumulation and the availability of control room indication and alarm of decay heat removal system level in the control room.
3.5.2.3	12 hours	The SR to verify steam generator level is within limits every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine surveillances that are typically performed once per shift. In addition, operators are trained to be sensitive to steam generator level and ensure that the level is appropriately established and controlled.
3.5.2.4	24 months	The 24 month Frequency is based on equipment reliability and the need to perform these surveillances under the conditions that apply during a unit outage and the potential for unplanned plant transients if the surveillances were performed with the reactor at power. The 24 month Frequency is also acceptable based on consideration of the design reliability of the equipment.
3.5.3.1	24 hours	Because the UHS level is normally maintained at a stable level, and is monitored by main control indication and alarm, a 24 hour Frequency is appropriate. This Frequency also takes into consideration the high ratio of UHS volume change to UHS level change due to the UHS geometry.
3.5.3.2	24 hours	The Frequency of 24 hours is sufficient to identify a temperature change that would approach either the upper or lower limit of UHS bulk average temperature assumed in the safety analyses. Because the UHS bulk average temperature is normally stable, and is monitored by main control indication and alarm, a 24 hour Frequency is appropriate. This Frequency also takes into consideration the large heat capacity of the UHS in comparison to the magnitude of possible heat addition or removal mechanisms.

Table 16.1-1: Surveillance Frequency Control Program Base Frequencies (Continued)

Surveillance Requirement	Base Frequency	Basis
3.5.3.3	31 days	Because the UHS volume of borated water is large compared to potential dilution sources, the 31 day Frequency is acceptable. In addition, the relatively frequent surveillance of the UHS water volume provides assurance that the UHS boron concentration is not changed significantly.
3.6.2.1	12 hours	The Frequency of 12 hours is based on the availability and reliability of the automatically monitored pressure alarms to detect a change in accumulator pressure and the similarity of the test to a CHANNEL CHECK as performed throughout existing large plant designs. The test verifies the accumulator pressure and thereby assures the OPERABILITY of the valves, as well as the status of the automatically monitored pressure alarms.
3.6.2.2	31 days	Because verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions.
3.6.2.4	24 months	The 24 month Frequency is based on the need to perform this surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the surveillance were performed with the reactor at power. Industry operating experience has shown that these components usually pass this surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.
3.6.3.1	7 days	The 7 day Frequency is based on the de-energized and closed condition of automatic containment isolation valves, the limited accessibility during MODE 4, and administrative controls over configuration of other containment penetrations.
3.7.1.1	12 hours	The Frequency of 12 hours is based on the availability and reliability of the automatically monitored pressure alarms to detect a change in accumulator pressure and the similarity of the test to a CHANNEL CHECK as performed throughout existing large plant designs. The test verifies the accumulator pressure and thereby assures the OPERABILITY of the valves, as well as the status of the automatically monitored pressure alarms.
3.7.2.1	12 hours	The Frequency of 12 hours is based on the availability and reliability of the automatically monitored pressure alarms to detect a change in accumulator pressure and the similarity of the test to a CHANNEL CHECK as performed throughout existing large plant designs. The test verifies the accumulator pressure and thereby assures the OPERABILITY of the valves, as well as the status of the automatically monitored pressure alarms.
3.8.1.1	12 hours	The Frequency of 12 hours is based on equipment reliability and is consistent with the CHANNEL CHECK Frequency specified for similar neutron detector instruments in LCO 3.3.1.
3.8.1.2	24 months	Industry operating experience has shown that similar components usually pass this surveillance when performed at the 24 month Frequency.

Enclosure 2:

“US460 Standard Design Approval Technical Specifications Development,” TR-101310-NP, Revision 0

Licensing Technical Report

US460 Standard Design Approval Technical Specifications Development

December 2022

Revision 0

Docket: 52-050

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Department of Energy Acknowledgement and Disclaimer

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

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Abstract

This report describes the development process of the NuScale Power Plant US460 Standard Design Approval (SDA) technical specifications (TS) to conform with regulatory requirements and expectations regarding scope, content, and format. This report also provides the basis for including the requirements chosen for the NuScale TS.

This report describes the development process with emphasis on the differences from the NuScale US600 Design Certification Application (DCA) Technical Specifications, Revision 5 (Reference 5.1.1 and Reference 5.1.2). The development process for the NuScale DCA Technical Specifications is described in TR-1116-52011-NP, Revision 4 (Reference 5.1.3).

1.0 Introduction

1.1 Purpose

The purpose of this report is to outline the development process of the NuScale Power Plant technical specifications (TS) to conform with the applicable regulatory requirements and expectations regarding scope, content, and format. This report also provides the basis for including the specifications chosen for the NuScale TS. The report focuses on three aspects:

- Conformance with Title 10 Code of Federal Regulations Section (10 CFR) 50.36 (Reference 5.1.4) and 10 CFR 50.36a (Reference 5.1.5), including considerations in 58 FR 39132, Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (Reference 5.1.6).
- Conformance with regulatory expectations as expressed by the precedent established in the NuScale US600 Design Certification Application (DCA) TS (Reference 5.1.1 and Reference 5.1.2). Consideration of Nuclear Regulatory Commission (NRC)-published standard technical specifications (STS), approved generic technical specifications (GTS), and recent changes as delineated in industry change travelers is also described where applicable.
- Conformance with the technical specification format and content guidance established by TSTF-GG-05-01, Revision 1, Writer's Guide for Plant Specific Improved Technical Specifications, August 2010 (Reference 5.1.7).

The report supplements the descriptions in TR-1116-52011, Rev. 4 (Reference 5.1.3) describing the development of the NuScale US600 DCA technical specifications and Bases. It identifies significant changes from the US600 DCA technical specifications requirements, and identifies NRC/industry Improved Standard Technical Specifications (ISTS) travelers considered in the development of the US460 SDA technical specifications.

1.2 Scope

This report addresses the development of the TS applicable to an individual module installed in a US460 NuScale power plant. The NuScale US460 SDA technical specifications are drafted in the context of the US460 SDA application for a NuScale facility containing up to six modules, however the content is applicable to an individually licensed module. The content of this report and the NuScale GTS are generally applicable to any NuScale facility containing any number of individually licensed modules with variations to address facility specific design details.

1.3 Abbreviations

Table 1-1 Acronyms

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

Table 1-2 Definitions

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

Table 1-2 Definitions (Continued)

Term	Definition
Emergency core cooling system (ECCS) actuation	Emergency core cooling system actuation describes the signal that permits the ECCS valves (reactor vent valves (RVVs) and reactor recirculation valves (RRVs)) to open. The RVVs open immediately upon receipt of an actuation signal. The RRVs may not immediately open in response to actuation depending on the function of the pressure interlock feature that compares reactor coolant pressure with the pressure in the containment in accordance with the descriptions provided in the US460 SDA.
k_{eff}	effective neutron multiplication factor, k_{eff} = 1 is the critical configuration

2.0 Background

2.1 Approach

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

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

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

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

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

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

Section 3.0 of TR-1116-52011 (Reference 5.1.3) describes the assessment of each TS chapter and its incorporation into the NuScale GTS. Section 3.0 of this report describes substantive changes to the content of the TS from the DCA generic technical specifications to the US460 SDA technical specifications, including criterion for inclusion where relevant.

2.2 Regulatory Requirements

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

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

Subpart E of 10 CFR 52, Standard Design Approvals, does not require submittal of Technical Specifications for consideration. However, NuScale determined the TS and Bases support the US460 SDA application by providing a basis for the evaluation of how the design and its analyses will be implemented consistent with their descriptions in the US460 Final Safety Analysis Report (FSAR) and elsewhere. Additionally, the US460 SDA technical specifications provide a basis for development of generic standard TS for the NuScale design.

2.3 Design Specific Review Standard

A design specific review standard (DSRS) was issued for the NuScale small module reactor design (ML15355A312), primarily for use in evaluating the US600 design application and any subsequent combined license applications. No specific mention of use of the DSRS during review of an SDA was included, however the guidance was generally applicable to, and used in, the development of the US460 SDA technical specifications.

The DSRS provided a review path that is based on the evolution of existing operating PWR into the NuScale TS. The preparation, review, and subsequent approval of the DCA technical specifications was based on that evolution as described in TR-1116-52011 (Reference 5.1.3). However with the approved US600 technical specifications baseline available, the US460 SDA development and the description in this report focuses on the changes needed from the US600 DCA technical specifications content to the US460 SDA-specific content. The majority of the DSRS content remains applicable with this approach.

3.0 Changes to the Content of Standard NuScale Technical Specifications

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

3.1 Modifications to Chapter 1, Use and Application

3.1.1 1.1 Definitions

LEAKAGE

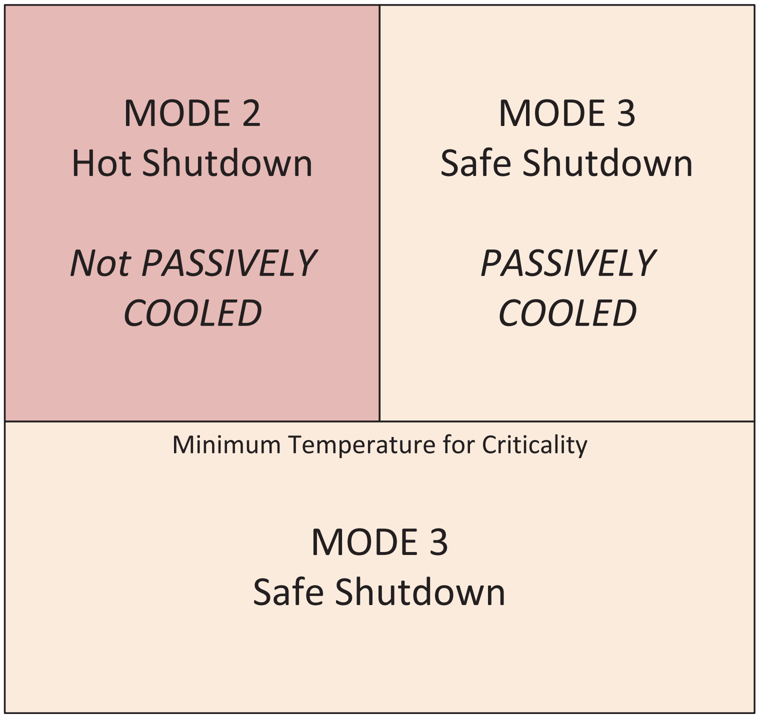
The definition of Pressure Boundary LEAKAGE is modified consistent with the applicable portions of the LEAKAGE definition provided in NUREG-1431, Revision 5 as appropriate for the NuScale design. The change includes modifications to the punctuation used in this definition, consistent with the Writer's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

MODE

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

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

Figure 3-1 MODES 2 and 3 Illustration



PASSIVELY COOLED - PASSIVE COOLING

This definition is revised to reflect the new design of the ECCS that only requires one or more RVVs and one or more RRVs to be open to perform its safety function. A related change to the OPERABILITY requirements for the system is provided in LCO 3.5.1, Emergency Core Cooling System.

RATED THERMAL POWER

The rated thermal power (RTP) for the US460 design is increased from 160 MWt to 250 MWt, consistent with the plant design and safety analyses.

SHUTDOWN MARGIN

The reference temperature used to establish the shutdown margin is reduced from 420 degrees F to 345 degrees F to maintain alignment with LCO 3.4.2, RCS Minimum Temperature for Criticality and used in the safety analyses.

3.1.2 Sections 1.2 through 1.4

No changes to these sections from that provided as NuScale US600 DCA content.

3.2 Modifications to Chapter 2, Safety Limits

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

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

3.3.1 Modification of Limiting Condition of Operation 3.0.3

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

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

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

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

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

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

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

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

<u>Modification of Emergency Core Cooling System Actuation Signals</u>

The design of the ECCS and the associated actuation signals have been changed. These changes to the ECCS components are described in FSAR Chapter 7, Section 6.3, and further addressed in LCOs 3.4.10 and 3.5.1. The analysis of ECCS actuation, including consideration of boron mixing features and the ECCS supplementary boron (ESB) system required by the new LCO 3.5.4, resulted in use of reactor pressure vessel riser level signals to initiate ECCS response. The revised actuation logic uses two level setpoints - Low, and Low-Low, to ensure appropriate response depending on RCS conditions as indicated by the RCS T_{cold} signal (T-5). The new design and actuation logic are consistent with the safety analyses in the FSAR. The new actuation signals are included because they satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

<u>Addition of Pressurizer Line Isolation Actuation</u>

This change adds a Pressurizer Line Isolation actuation that isolates the pressurizer spray and high point vent line containment isolation valves in response to a low pressurizer level signal. This isolation provides improved plant response to postulated small line breaks occurring on the pressurizer spray and high point vent lines. The new actuation signal is included because it satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

<u>Addition of Reactor Trip on High Reactor Coolant System Average Temperature</u>

This change adds a High T_{avg} actuation at lower powers. The actuation provides earlier response to conditions similar to the high RCS temperature trip when the event begins at a temperature below the full power T_{hot} temperature. The new trip is included because it satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

<u>Addition of Emergency Core Cooling System Actuation Timer on Reactor Trip</u>

A delayed actuation is added to ensure ECCS actuation and subsequent ECCS supplemental boron (ESB) function after reactor trip for non-loss-of-coolant accident events. Requirements for, and a description of the ESB function, is provided in new LCO 3.5.4 and the associated Bases. The new actuation signal is included because it satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

<u>Bypass Low-Low Pressurizer Pressure Reactor Trip and DWSI when Reactor Coolant System Temperature Below T-3</u>

Reactor trip and demineralized water system isolation actuation is modified so that the signal is not active when RCS temperature is below the T-3 interlock temperature. The RCS temperature T-3 bypass is active when RCS temperature is less than approximately 340 degrees F. This change supports startup of the reactor, while maintaining the credited safety function of the trip and actuation.

<u>Other Table 3.3.1-1, Module Protection System Instrumentation Changes</u>

Footnotes referring to capability to withdraw a control rod assembly (CRA), are modified to indicate the requirement applies when capable of withdrawing more than a single CRA. This change is necessary to allow energization of a portion of the control rod drive system (CRDS) in MODE 3 when preparing for module disassembly.

This capability is required to verify the CRA is disconnected from its extension rod. The CRDS design and administrative control ensure that no more than one CRA may be manipulated, and the definition of, and limits on SHUTDOWN MARGIN specified in LCO 3.1.1 ensure the plant remains safely shutdown. A description of the functional design of the CRDS is provided in FSAR Section 4.6. This change results in changes to the Bases discussions for the associated Functions.

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

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

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

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

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

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

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

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

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

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

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

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

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

3.3.9 Modification of Surveillance Requirement 3.4.4.1 - Reactor Safety Valve Setpoints

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

3.3.17 Addition of Limiting Condition of Operation 3.6.3, Containment Closure

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

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

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

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

3.3.19 Other Bases Changes

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

3.4 Chapter 4, Design Features

Section 4.3 Fuel Storage

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

3.5 Chapter 5, Administrative Controls

Section 5.2.2, Facility Staff

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

Section 5.5.9, Containment Leakage Rate Testing Program

The description of the Containment Leakage Rate Testing Program is revised to provide alternatives to adopt Option A or Option B of 10 CFR 50, Appendix J.

Section 5.6.3, Core Operating Limits Report

This section is modified to align with the safety analyses, referencing technical specification limits, and topical reports that describe the limits that will be included in the COLR.

Section 5.6.4, Reactor Coolant System PRESSURE AND TEMPERATURE LIMITS REPORT

Modified to align with the safety analyses and topical report that describes the limits that will be included in the Pressure and Temperature Limits Report.

4.0 Conformance with Industry Standard Technical Specifications and Standard Technical Specification Writer's Guide

The US460 Standard design, analyses, and therefore the technical specifications are significantly different from those provided in the STS for large LWRs. However, to the extent appropriate, conventions and content have been adopted and incorporated to parallel the NUREG standard technical specifications. For example, the writer's guide is generally used with regard to format and content of individual specifications and bases discussions.

For example, Chapter 1, Use and Application, and Sections 3.0, LCO and Surveillance Requirement Applicability contain content generally consistent with legacy industry content. This approach supports internal and external understanding and application of the balance of the contents of the technical specifications and bases.

Industry travelers that were publicly available on the NRC's Agencywide Documents Access and Management System since issuance of the DCA technical specifications and NUREG standard technical specifications revisions, have been monitored for potential applicability to the NuScale TS and bases. Where conceptually appropriate NuScale-specific content is developed and incorporated. A summary of actions taken to incorporate changes similar to those publicly available industry travelers is provided in the table below.

Table 4-1 Standard Technical Specifications Traveler Adaptation

Traveler No.	Addressed	Comments
571	N/A	T-traveler or otherwise not available
572	N/A	T-traveler or otherwise not available
573	No	Boiling water reactor (BWR)-specific
574	N/A	T-traveler or otherwise not available
575	N/A	T-traveler or otherwise not available
576	No	BWR-specific
577	Yes	Addressed in Section 5.5 to extent appropriate
578	No	Not applicable to NuScale design
579	No	Not applicable to NuScale TS
580	No	Not applicable to NuScale design
581	N/A	T-traveler or otherwise not available
582	No	Not applicable to NuScale design
583	N/A	T-traveler or otherwise not available
584	No	BWR-specific
585	Yes	NuScale incorporated revision 0 like content in anticipation of industry and regulatory adoption of this change. Adjustments will be considered for incorporation as industry and regulatory issues are resolved.
586	N/A	T-traveler or otherwise not available
587	N/A	T-traveler or otherwise not available
588	Pending	NuScale is monitoring this proposal for future consideration and adoption.
589	No	Not applicable to NuScale design
590	N/A	T-traveler or otherwise not available
591	No	Not applicable to NuScale TS
592	No	Not applicable to NuScale design
593	N/A	T-traveler or otherwise not available
594	N/A	T-traveler or otherwise not available
595	N/A	T-traveler or otherwise not available
596	Pending	NuScale is monitoring this proposal for future consideration and adoption.

5.0 References

5.1 Referenced Documents

5.1.1 NuScale Power, LLC, “Generic Technical Specifications, NuScale Nuclear Power Plants, DCA Part 4, Volume 1: Specifications,” Revision 5.

5.1.2 NuScale Power, LLC, “Generic Technical Specifications, NuScale Nuclear Power Plants, DCA Part 4, Volume 2: Bases,” Revision 5.

5.1.3 NuScale Power, LLC, “Technical Specifications Regulatory Conformance and Development,” TR-1116-52011, Revision 4, May 2020.

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

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

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

5.1.7 Technical Specification Task Force, “Writer’s Guide for Plant-Specific Improved Technical Specifications,” TSTF-GG-05-01, Revision 1, Rockville, MD, August 2010.