Gap Analysis Summary Report

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Nonproprietary

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Introduction

This report provides the results of a regulatory gap analysis performed by NuScale Power, LLC. (NuScale) as part of pre-application activities in preparation for submitting to the U.S. Nuclear Regulatory Commission (NRC) its application for standard design certification pursuant to 10 CFR 52, Subpart B. As such, the NuScale regulatory gap analysis provided herein involved a detailed reconciliation of existing light-water reactor (LWR) regulatory requirements and guidance with the characteristics of the NuScale power plant design. Specifically, the analysis involved a detailed review of the NRC regulatory requirements contained in Title 10 of the Code of Federal Regulations (10 CFR), Parts 1 through 199 (Reference 5.1), as well as the guidance stipulated in the NRC Standard Review Plan (SRP) (Reference 5.2) and those documents referenced within the SRP, which will henceforth be referred to as “sub-tier” documents. The focus of this report is meant to be on the regulatory process, rather than the technical merit, associated with the NuScale design effort.

Gap Analysis Results

The results of the NuScale regulatory gap analysis are summarized in Section 3.0 of this report. The detailed results are available for NRC review in the NuScale electronic reading room. The results of the NuScale gap analysis effort confirm that the NRC’s existing LWR-based regulations and regulatory guidance, with specific modifications as discussed herein, represent a valid regulatory framework to be applied to the development, submission, and acceptance of a complete design certification application for the NuScale power plant design.

The gap analysis assessment of 10 CFR 1 through 199 resulted in the identification of a number of regulations that, due to features unique to the NuScale power plant design, are not relevant and thus would be inappropriate to apply, either in whole or in part, to the NuScale power plant design. These “gaps” in the LWR-based regulatory framework warrant further consideration to develop a licensing path forward, such as regulatory departure/exemption, alternative interpretation, or other form of NRC approval/concurrence to be defined during pre-application deliberations between NuScale and NRC. The specific regulations determined to warrant further consideration are summarized in Table 3-1 of this report.

The detailed gap analysis assessment of the SRP revealed that a number of SRP acceptance criteria and sub-tier guidance documents are not relevant and thus would be inappropriate to apply, either in whole or in part, to the NuScale power plant design. Portions of the SRP that are inapplicable to the NuScale design warrant further consideration to facilitate understanding of the review process that the NRC intends to implement with regard to these NuScale specific issues. The creation of NuScale design-specific review standard (DSRS) sections will enhance NuScale’s understanding of the NRC’s review process, clarify the level of detail expected for unique features of the NuScale design, enhance the quality of the NuScale design certification application (DCA), and ultimately facilitate an efficient and timely NRC review of the NuScale design. SRP sections that would be especially beneficial to obtain NuScale-specific DSRS sections for were highlighted in Section 3.2 of this report to facilitate communication between the NRC and NuScale with regard to SRP guidance that describes methods/approaches that the staff has found acceptable for meeting NRC requirements and the methods/approaches that NuScale intends to pursue.

Conclusion

The results summarized in this report provide a strong foundation to facilitate deliberations between NuScale and the NRC during pre-application activities. Specifically, as part of further pre-application activities, NuScale will continue to seek to reach consensus with the NRC on the:

(1) applicability of the regulatory framework as assessed in this gap analysis; and (2) the path forward towards addressing “regulatory gaps” identified in Section 3.0 of this report.

NuScale believes that the regulatory gap analysis results represent a useful tool for the development of a design-specific review standard (DSRS) to be used by the NRC in its review of the NuScale application for design certification. NuScale remains committed to assisting the NRC as necessary and appropriate to facilitate the creation of the NuScale specific DSRS sections.

The results of the NuScale regulatory gap analysis reflect existing knowledge based on the current stage of engineering design, and as such represent NuScale’s best-effort assessment of applicability and relevance of current LWR-based requirements and guidance, in literal language or intent, to the NuScale power plant design. As the ongoing engineering design effort progresses, the relevance of portions of the requirements and guidance considered in this gap analysis may warrant reconsideration. Accordingly, the NuScale regulatory gap analysis results summarized herein are not intended to preclude NuScale from proposing in its application for design certification certain design features, analytical techniques, and procedural measures that would be different than those given in the design-specific review standard to be developed for the NuScale design based on the results of this gap analysis. Any such differences would be evaluated in accordance with 10 CFR 52.47(a)(9) to confirm that any proposed alternative provides an acceptable method of complying with the underlying NRC requirements.

This document contains no regulatory commitments.

1.0 Introduction

1.1 Objectives

The primary objective of the NuScale Power, LLC (NuScale) regulatory gap analysis is to provide an evaluation of the regulatory framework that should be applied to the development, submission, and acceptance of a complete design certification application for the NuScale power plant design. Per Regulatory Guide 1.206 Section C.IV.7, pre-application interactions "should focus on what would be most beneficial to the review, and what would achieve the best and most efficient use of staff and applicant resources" and focus on expected deviations from staff guidance and expected exemptions from the regulations. This regulatory gap analysis is intended to establish a documented, clear delineation of NRC expectations for completeness for the NuScale power plant design certification application in those areas that materially differ from existing LWR requirements and guidance. This should facilitate insights into the most efficient use of NRC and NuScale resources throughout the rest of the pre-application phase of the NuScale design certification effort and development of a high quality DCA that can be efficiently and effectively reviewed.

To meet these objectives, the gap analysis identifies existing LWR-based regulations and guidance, or portions thereof, that are not relevant and thus would be inappropriate to apply to the NuScale power plant design or design certification application specifically due to features, functions, and capabilities unique to the NuScale power plant. The gap analysis also identifies those areas wherein existing LWR-based regulations and guidance may be appropriately applied to the NuScale design with very little to no alterations. Additionally, the gap analysis describes areas wherein the inherent safety features and unique operational features of the NuScale design justify departures from existing LWR-based regulations.

1.2 Intended Uses

The results summarized in this report provide a strong foundation to facilitate deliberations between NuScale and the NRC during pre-application activities. Specifically, as part of further pre-application activities, NuScale will seek to reach consensus with the NRC on

1. the applicability of the regulatory framework as assessed in this gap analysis.
2. the path forward towards addressing "regulatory gaps" identified in Section 3.0 of this report.

At a minimum, the results of the NuScale regulatory gap analysis summarized in Section 3.0 of this report are anticipated to be used as input to

1. NuScale's development of

- engineering design and analysis requirements.
- the NuScale plant licensing basis.

2. The NRC's

- determination of relevance/applicability of regulations and guidance to the NuScale power plant design.
- development of a design-specific review standard for the NuScale power plant design, consistent with NUREG-0800 "Introduction - Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Small Modular Reactor Edition", Revision 0 (Reference 5.3) and SECY-11-0024 (Reference 5.4).

1.3 Abbreviations and Definitions

Table 1-1. Abbreviations

Term	Definition
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BTP	branch technical position
BWR	boiling water reactor
CFR	Code of Federal Regulations
CFS	containment flooding system
COL	combined operating license
COLA	combined operating license application
CSDRS	certified seismic design response spectra
CVC	chemical and volume control
DAC	design acceptance criteria
DC	design certification
DCD	design control document
DHR	decay heat removal
DHRS	decay heat removal system
DI&C	digital instrumentation and control
DSRS	design-specific review standard
ECCS	emergency core cooling system
EP-ITAAC	emergency planning inspections, tests, analyses, and acceptance criteria
EPR	evolutionary power reactor
EPRI	Electric Power Research Institute
ESBWR	economic simplified boiling water reactor
ESF	engineered safety feature
FPGA	field programmable gate array
GDC	general design criterion
I&C	instrumentation and control

Term	Definition
ISG	interim staff guidance
ITAAC	inspections, tests, analyses, and acceptance criteria
LOCA	loss-of-coolant accident
LWR	light-water reactor
MPS	module protection system
NEI	Nuclear Energy Institute
NQA	nuclear quality assurance
NRC	U.S. Nuclear Regulatory Commission
NSIR/DPR	Nuclear Security and Incident Response, Office of the NRC/Division Of Preparedness and Response
NUREG	NRC technical report designation (“Nuclear Regulatory Commission”)
NUREG/CR	NUREG contractor report
NuScale	NuScale Power, LLC
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
QA	quality assurance
QAPD	quality assurance program description
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RTNSS	regulatory treatment of non-safety systems
SBO	station blackout
SECY	Secretary of the Commission, Office of the (NRC)
SPDS	safety parameter display system
SRM	staff requirements memorandums
SRP	Standard Review Plan
SSC	structure, system, and component
TBV	turbine building ventilation
TMI	Three Mile Island

2.0 Scope and Methodology

2.1 Scope

To achieve the objectives of this report laid out in Section 1.1, the NuScale regulatory gap analysis methodology involved a detailed review of existing LWR regulations and guidance for applicability and technical relevance to the NuScale power plant design. The scope of this review included the following:

1. the body of NRC regulations contained in Title 10 of the Code of Federal Regulations (CFR), parts 1 through 199, with particular focus on 10 CFR 52 and those parts specified in 10 CFR 52.48 as standards for review of design certification applications (i.e., 10 CFR Parts 20, 50, 51, 73, and 100, and appendices thereto)
2. the NRC Standard Review Plan (NUREG-0800) for nuclear power plants, including branch technical positions (BTPs)
3. sub-tier guidance to the NRC Standard Review Plan (i.e., guidance documents referenced within the SRP), including the following:

- regulatory guides (RGs) including RG 1.206 (Reference 5.5) [Note: This represents a review of the RGs (or portions thereof) with respect to the context of the SRP that is referencing it]
- NUREG reports
- unresolved and generic safety issues
- NRC documents such as SECYs and associated staff requirements memorandums (SRMs)
- NRC generic communications (e.g., Inspection and Enforcement (IE) bulletins, circulars, generic letters, administrative letters, information notices, regulatory issue summaries, etc.)
- Three Mile Island (TMI) action plan items
- industry codes and standards

4. Interim Staff Guidance (ISG) with potential relevance to applicants for and holders of a design certification (i.e., ISGs designated as “DC/COL-ISG,” “DI&C-ISG,” “JLD-ISG,” and “NSIR/DPR-ISG”)
5. Division 1, 4, 5, and 8 regulatory guides[1] [Note: This represents a review of the entire regulatory guide as a standalone document]
6. The draft mPower DSRS

The evaluation of the set of regulatory documents considered in the NuScale regulatory gap analysis has been performed based on the current state of engineering design. However, both the design and identification of applicable regulations and guidance may evolve over time. Any new gaps resulting from this evolution will be identified as part of the NuScale design process, with finality anticipated upon approval and issuance of the design certification for the NuScale power plant design.

[1] Evaluation of these groups of regulatory guides is consistent with the scope specified in RG 1.206, Section C.I.1.9.1, “Conformance with Regulatory Guides.”

2.2 Methodology

The methodology used in the NuScale regulatory gap analysis was developed to

1. conform to current regulations and guidance to the extent practicable, considering that many of the current regulations and guidance are based on large LWR technology.
2. use a decision-making process that determines the relevance of existing LWR-based regulations and guidance to the NuScale power plant design. NuScale design information was considered in an effort to identify those design functions and characteristics unique to the NuScale power plant design, i.e., those that differ significantly from design functions and characteristics for a typical large LWR. These design functions and characteristics were then compared to the NRC regulations and guidance for relevance and applicability.
3. establish whether a NuScale-specific DSRS section would be desirable and to make a recommendation as to whether the SRP or mPower DSRS would be the more appropriate starting document for the creation of the NuScale DSRS.

Of the regulations and guidance evaluated, a determination of relevance resulted in one of four possible outcomes:

1. *Applicable* – The regulation or regulatory guidance is relevant and applicable to the NuScale application for design certification, and can be applied "as-is."
2. *Partially Applicable*– The underlying purpose or intent of the requirement or guidance is relevant to the NuScale power plant design but cannot be appropriately applied as written, or some portion of the requirement or guidance is applicable to the NuScale application for design certification while other portions are not applicable. The following are examples:

- The regulatory requirement or guidance is applicable except for aspects that are specific to combined license or early site permit applicants, or to boiling water reactor (BWR) designs, etc.
- A portion of the regulatory requirement or guidance is literally applicable, but the specific language refers to a different type of LWR design or a structure, system, and component (SSC) that is not part of the NuScale design.
- The intent of a regulatory requirement or guidance is applicable, but the specific language refers to one of the following:
  - a different type of LWR design
  - an SSC that is not part of the NuScale design, but for which a substantively equivalent function is served by other SSCs within the NuScale design

3. *Not Applicable*– The regulation or guidance is not appropriate to apply to the NuScale application for design certification. The following are examples:

- The regulatory requirement or guidance is applicable only to BWR designs.
- The regulatory requirement or guidance is applicable only to large pressurized-water reactor (PWR) designs.
- The regulatory requirement or guidance is applicable to the NuScale design, but is the responsibility of the combined license applicant.

4. *NuScale Unique Feature or Requirement* – NuScale plant design basis features or requirements are identified that do not appear to be addressed by existing regulations or guidance. Any such instances may require new requirements, guidance, or other form of regulatory approval, as appropriate, to be developed during the design certification pre-application process.

For items found within the code of federal regulations (CFR) a fifth possible outcome was also implemented:

5. Departure – The regulation is literally applicable, however NuScale intends to depart from the regulation based on the design or safety basis of the NuScale design. That is to say that conformance to the regulation would have a minimal, or even negative, impact on safety of the NuScale power plant design and hence a departure from the regulation (that was originally created for traditional LWRs) is warranted. The form of the departure may be through an exemption request under 10 CFR 52.7 or through a specific process available for a set of regulations. For example, the introduction to 10 CFR 50, Appendix A provides a departure from the General Design Criteria as explained in section 3.1, and the TMI action items may be identified and justified as "not technically relevant" to the design consistent with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f).[2]
6. The results of these assessments of applicability are summarized in Section 3.0 of this report.

[2] Note that some TMI action items are categorized as "partially applicable" or "not applicable" rather than "departure". The difference is that those requirements are not applicable by their own terms, for example because they apply to BWRs or to an SSC that the NuScale design lacks. On the other hand, a departure from a TMI requirement is appropriate where the requirement is literally applicable but is inappropriate to apply to the NuScale design.

3.0 Gap Analysis Summary Results

As discussed in Section 2.2 of this report, for each regulatory requirement and individual guidance criterion reviewed, the NuScale regulatory gap analysis effort includes a determination of relevance and applicability in the form of one of five dispositions: “Applicable,” “Partially Applicable,” “Not Applicable,” “NuScale Unique Feature or Requirement,” or “Departure.” The detailed results of this determination are available for NRC review in the NuScale electronic reading room.

Each regulatory requirement and guidance document was assessed to determine whether it could be practically applied by the NRC in its review of the NuScale application for design certification. As summarized below, and in Table 3-1 and Table 3-2 of this report, the results of this assessment indicate a number of “gaps” that need to be resolved with NRC during the pre-application phase to enhance the probability of an efficient and favorable review of the NuScale DCA. The results of the NuScale gap analysis effort confirm that the NRC’s existing LWR-based regulations and regulatory guidance, with specific modifications as discussed herein, represent a valid regulatory framework to be applied to the development, submission, and acceptance of a complete design certification application for the NuScale power plant design.

The results of the NuScale regulatory gap analysis reflect existing knowledge based on the current stage of engineering design, and as such represent NuScale’s best-effort assessment of applicability and relevance of current LWR-based requirements and guidance to the NuScale power plant design. As the ongoing engineering design effort progresses, the relevance of portions of the requirements and guidance considered in this gap analysis may warrant reconsideration. Accordingly, the NuScale regulatory gap analysis results summarized herein are not intended to preclude NuScale from proposing in its application for design certification certain design features, analytical techniques, and procedural measures different than those in the design-specific review standard to be developed for the NuScale design based on the results of this gap analysis. Any such differences would be evaluated in accordance with 10 CFR 52.47(a)(9) to confirm that any proposed alternative provides an acceptable method of complying with the underlying NRC requirements.

3.1 NRC Regulations

As discussed in Section 2.1 of this report, the NuScale regulatory gap analysis included a detailed review of the entire body of NRC regulations (10 CFR Parts 1 through 199). Both administrative regulations as well as design-related regulations were considered, with particular focus on 10 CFR 52 and those parts specified in 10 CFR 52.48 as standards for review of design certification applications (i.e., 10 CFR Parts 20, 50, 51, 73, and 100, and appendices thereto). Documentation of the detailed results of the NuScale gap analysis review of the NRC regulations is provided in the NuScale electronic reading room.

From this review, it was determined that due to design features, functions, and capabilities unique to the NuScale power plant design, a number of regulations are not relevant and thus would be inappropriate to apply, either in whole or in part, to the NuScale power plant design. These “gaps” in the LWR-based regulations warrant further consideration to develop a regulatory path forward, such as regulatory departure/exemption, alternative interpretation, or other form of NRC approval/concurrence to be defined during pre-application deliberations between NuScale and the NRC. The specific regulations determined to warrant further consideration are summarized in Table 3-1 of this report.

As part of the NuScale gap analysis review of NRC regulations, consideration was given to NuScale power plant design features that potentially could not be addressed by existing regulations, thus requiring new requirements or approvals to be established during the design certification process. One such instance identified related to NuScale design provisions that warrant a new design criterion as an alternative to GDC 33. This item is discussed in detail in Section A.10, and Table 3-1 of this report.

The assessment of NRC regulations for applicability to the NuScale design included a detailed review of General Design Criteria (GDCs) codified in 10 CFR 50, Appendix A. There were various instances in which the NuScale advanced passive design features were determined to be substantively different in certain specific areas, from those design features considered when the GDCs were formulated. In these instances, the affected GDCs have been determined to be unnecessary or inappropriate to apply, either in whole or in part, to the NuScale power plant design.

The introduction section of 10 CFR 50, Appendix A, states that certain GDCs may not be appropriate to apply to advanced reactor plant designs (such as the NuScale design). Accordingly, whereas the GDCs are regarded as minimum requirements for establishing principal design criteria for LWR designs similar to existing operating reactor designs, the GDCs are "guidance" for other types of reactor designs. This is explained in the second paragraph of 10 CFR 50, Appendix A, which states,

> "These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units."

The final paragraph of the introduction section of 10 CFR 50, Appendix A, further demonstrates the NRC's recognition that the GDCs may be insufficient, unnecessary, or inappropriate for application to some LWR designs, including advanced LWR designs. For occasions in which GDCs are determined to be unnecessary or inappropriate to apply, allowance is provided for establishing departures from the GDC.

> "There will be some water-cooled nuclear power plants for which the General Design Criteria are not sufficient and for which additional criteria must be identified and satisfied in the interest of public safety. In particular, it is expected that additional or different criteria will be needed to take into account unusual sites and environmental conditions, and for water-cooled nuclear power units of advanced design. Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified."

As indicated above, the NuScale gap analysis results identify a number of GDCs that are unnecessary or inappropriate to apply, either in whole or in part, to the NuScale advanced power plant design. Table 3-1 of this report identifies and provides a summary justification for those GDCs for which departures appear to be necessary. Consistent with the introduction to 10 CFR 50, Appendix A, as excerpted above, departures from these GDCs are warranted to accommodate the NuScale design. However, to the extent the GDCs in 10 CFR 50, Appendix A

represent guidance for the NuScale design, as opposed to requirements as would be the case for typical LWR designs, these departures would not require exemptions as contemplated by 10 CFR 52.7 and 10 CFR 50.12. Rather, these departures would be described and evaluated in the design control document (DCD)[3] to be submitted as part of the NuScale application for design certification pursuant to 10 CFR 52.47(a). Notwithstanding this conclusion, it is anticipated that the appropriate form of any departures will be confirmed during deliberations between NuScale and NRC as part of pre-application activities.

[3] The NuScale design control document is intended to represent the final safety analysis report that is required by 10 CFR 52.47(a) to be submitted with an application for design certification.

Table 3-1 NRC regulations requiring further consideration

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
1.	50.34(f)(1)(ii)	Evaluation and Design Review of AFW System	Regulatory Requirement 10 CFR 52.47(a)(8) requires a design certification applicant to provide, "The information necessary to demonstrate compliance with any *technically relevant* portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v) [emphasis added]." This requirement is repeated in the introduction to 10 CFR 50.34(f), including the "technically relevant" limitation. 10 CFR 50.34(f)(1)(ii) requires the applicant to, "Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWRs only) (II.E.1.1): (A) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques. (B) A design review of AFWS. (C) An evaluation of AFWS flow design bases and criteria." Summary Basis for Gap Determination The NuScale plant design does not contain an AFW system as would be found at a typical large LWR and therefore 10 CFR 50.34 (f)(1)(ii) is not literally applicable to the NuScale design. However, as discussed in Section A.2 of this report, the NuScale decay heat removal (DHR) system fulfills some of the functions of an AFW system at a large PWR. This requirement addresses the reliability of AFW based on its use in then-current CE, B&W, and Westinghouse plant-specific design bases. The underlying purpose of the requirement is related to the design basis functions for those AFW systems intended to prevent and mitigate small-break LOCAs. The NuScale DHR design and plant response to small-break LOCAs differs from those designs. Therefore, in addition to being literally not applicable, the underlying purpose of this requirement also is not applicable to NuScale. The NuScale DHR system is a safety-related system and is designed to be appropriately reliable and is fully considered within the NuScale PRA, but 10 CFR 50.34(f)(1)(ii) is not considered applicable to the NuScale DHR system. Because the literal language and intent of 10 CFR 50.34(f)(1)(ii) do not apply, the requirement is not technically relevant to the NuScale design. Therefore, an exemption would be unnecessary because 10 CFR 50.34(f)(1)(ii) only applies to the "technically relevant" portions of the Three Mile Island requirements. This conclusion appears to be supported by the lack of an exemption from 10 CFR 50.34(f)(1)(ii) for the AP1000 design, which also does not utilize a traditional AFW system. Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that the AFW system evaluation specified by 10 CFR 50.34(f)(1)(ii) is not technically relevant to the NuScale DHR system and no exemption is needed. (Note: NuScale suggests the creation of a section within the NuScale-specific DSRS to specifically address the DHR system.)

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
2.	50.34(f)(2)(iv)	Safety Parameter Display System (SPDS)	Regulatory Requirement 10 CFR 50.34(f)(2)(iv) requires the design certification applicant to, "Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. (I.D.2)." Summary Basis for Gap Determination This rule has been applied to previous design certification applicants as requiring an SPDS console separate from other control room displays. Specifically, because these applicants proposed integrating the SPDS function into the control room design rather than providing a separate console, NRC design certification approvals have included specific exemptions to 10 CFR 50.34(f)(2)(iv), as documented in Section V.B of 10 CFR 52, Appendices A through D. Additionally, GE-Hitachi requested as part of its economic simplified boiling water reactor (ESBWR) design certification application, and the NRC approved in the ESBWR final safety evaluation report, a similar exemption based on the lack of a separate console for the SPDS. Similar to those design certification holders for which exemptions have been granted, the NuScale SPDS will be integrated into the control room human-system interface design rather than having a separate console. Notwithstanding the above, it appears that the NRC position on the need for an exemption in these instances has changed. Specifically, during the NRC review of the pending evolutionary power reactor (EPR) design certification application, AREVA withdrew a similar exemption request on June 22, 2011, stating that the NRC had requested withdrawal of the request. Although the NRC's instructions to withdraw the exemption request do not appear to be publicly available, AREVA's revised response to the request for additional information related to the exemption request states, "The U.S. EPR design integrates the SPDS requirements into the design requirements for the [Process Information and Control System (PICS)] rather than a stand-alone, add-on system as is used at most currently operating plants. The language of the rule does not require that the console be standalone." NuScale agrees that the rule language does not require a separate SPDS console. Consistent with this recent precedent, NuScale has concluded that integration of the SPDS into the control room human-system interface design will not require an exemption from 10 CFR 50.34(f)(2)(iv). Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that integration of the NuScale SPDS into the control room human-system interface design will not require an exemption from 10 CFR 50.34(f)(2)(iv).
3.	50.34(f)(2)(xii)	Auxiliary Feedwater System Actuation and Flow Indication	Regulatory Requirement 10 CFR 50.34(f)(2)(xii) requires the design certification applicant to, "Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWRs only) (II.E.1.2)." Summary Basis for Gap Determination As discussed with respect to 50.34(f)(1)(ii), above, the NuScale plant design does not involve an AFW system as would be found at a typical large LWR. Also, while the DHR system performs some of the functions of an AFW system at a large PWR, the NuScale DHR system is designed for NuScale-specific transients and system characteristics, and its actuation and indication is designed accordingly.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			Specifically with regard to the portion of this requirement specifying control room flow indication, the DHR system operation involves passive natural circulation flow, with flow characteristics that inherently vary with system conditions, which makes DHR system flow a less useful measurement. For the NuScale design, control room indication for system parameters other than DHR system flow are more appropriate to ensure operators have the information necessary to adequately monitor DHR system operation and reactor core cooling. These parameters include DHR system pressure, DHR passive condenser level, DHR system valve position indication, and reactor coolant system pressure and temperature. As discussed with respect to 50.34(f)(1)(ii), above, 10 CFR 50.34(f)(2)(xii) is not considered applicable to the NuScale DHR system. Because the literal language and intent of 10 CFR 50.34(f)(1)(ii) do not apply, the requirement is not technically relevant to the NuScale design. Therefore, an exemption would be unnecessary because 10 CFR 50.34(f)(1)(ii) only applies to the "technically relevant" portions of the Three Mile Island requirements. See also gap analysis results for SRP Section 7.5. <u>Further Consideration</u> As part of pre-application activities, NuScale will seek NRC concurrence that the AFW actuation and indication specified by 10 CFR 50.34(f)(2)(xii) is not technically relevant to the NuScale DHR system and no exemption is needed. (Note: NuScale suggests the creation of a section within the NuScale-specific DSRS to specifically address the DHR system.)
4.	50.34(f)(2)(xv)	Containment Purging/Venting Capability and Isolation	<u>Regulatory Requirement</u> 10 CFR 50.34(f)(2)(xv) requires the design certification applicant to, "Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)." <u>Summary Basis for Gap Determination</u> The NuScale containment vessel design does not require or incorporate a purge/venting system function as contemplated by this requirement. The issues that led to the codification of this requirement are not technically relevant to the NuScale design. A typical LWR containment is a massive structure with subcompartments housing numerous power plant SSCs. These containment structures require purge/vent capability to allow personnel access and in some designs, to address combustible gas control and/or maintain containment pressure for emergency core cooling system (ECCS) performance. As discussed in Section A.7 of this report, the compact NuScale containment vessel is significantly smaller than a typical containment building, and its design is such that personnel access during reactor operation and purge/vent capability for combustible gas control is not needed. In addition, the NuScale ECCS design does not include pumps, and does not involve a typical PWR ECCS recirculation mode (i.e., ECCS pump suction is switched from water storage tank(s) to containment sumps) where ECCS pump performance relies on containment pressure. Thus, purge/vent capability as prescribed by 10 CFR 50.34(f)(2)(xv) is neither required nor included in the NuScale design. With no purge/vent system providing large diameter open paths to the environs, the concerns (underlying the requirement of 10 CFR 50.34(f)(2)(xv)) with the isolation capability of the large isolation valves in these lines are not germane to the NuScale design. Based on the above, it is concluded that this requirement is not technically relevant to the NuScale design. Thus, consistent with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), NuScale compliance with this provision is neither appropriate nor required, and no exemption is necessary.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that 10 CFR 50.34(f)(2)(xv) is not technically relevant to the NuScale design, and consistent with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), NuScale compliance with this provision is neither appropriate nor required, and no exemption is necessary.
5.	50.34(f)(3)(iv)	Spare containment penetration	Regulatory Requirement 10 CFR 50.34(f)(3)(iv) requires the applicant to, "Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. (II.B.8)." Summary Basis for Gap Determination This TMI requirement is based on traditional large LWR containment designs and the potential, as of the time of the requirement, need for future containment venting systems to accommodate severe accidents. As discussed in Section A.7, the NuScale containment design is substantially different from traditional designs because of its high-pressure capability. Moreover, the NuScale design accounts for severe accidents and does not require containment venting to safely mitigate them. A 3-foot opening relative to the NuScale containment would be infeasible to accommodate and is unnecessary. Furthermore, should any future development identify a need for a new penetration, adding such a penetration to the NuScale vessel is a substantially different process versus the typical containment. Based on the above, it is concluded that this requirement is not technically relevant to the NuScale design. Thus, consistent with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), NuScale compliance with this provision is neither appropriate nor required, and no exemption is necessary. Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that 10 CFR 50.34(f)(3)(iv) is not technically relevant to the NuScale design, and consistent with 10 CFR 52.47(a)(8) and 10 CFR 50.34(f), NuScale compliance with this provision is neither appropriate nor required, and no exemption is necessary.
6.	50.44	Combustible Gas Control	Regulatory Requirement 10 CFR 50.44(c)(2) states, "All containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features." 10 CFR 50.44(c)(4)(ii) states, "Equipment must be provided for monitoring hydrogen in the containment. Equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design-basis accident for accident management, including emergency planning." 10 CFR 50.44(d) "*Requirements for .. certain water-cooled reactor applicants and licensees"* The requirements in this paragraph apply to….. water-cooled reactors that do not fall within the description in paragraph (c), footnote 1 of this

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			section….Applications subject to this paragraph must include: (1) Information addressing whether accidents involving combustible gases are technically relevant for their design, and (2) If accidents involving combustible gases are found to be technically relevant, information (including a design-specific probabilistic risk assessment) demonstrating that the safety impacts of combustible gases during design-basis and significant beyond design-basis accidents have been addressed to ensure adequate protection of public health and safety and common defense and security. <u>Summary Basis for Gap Determination</u> Pursuant to 10 CFR 52.47(a)(12), an application for a design certification must include an analysis and description of the equipment and systems for combustible gas control as required by 10 CFR 50.44. For all new water-cooled reactors that fall within the description in paragraph (c), footnote 1, 10 CFR 50.44(c) requires, in part, that all containments have an inerted atmosphere, or limit hydrogen concentrations in containment to less than 10 percent (by volume) following a postulated design basis accident. Application of this requirement to the NuScale design is not necessary to achieve the underlying purpose of the rule. During normal operation, the NuScale containment vessel is at partial vacuum conditions. The act of introducing steam to the containment vessel from the reactor coolant system (via reactor coolant system (RCS) leakage or actuation of the ECCS system) causes the containment vessel to become inerted following postulated accidents related to combustible gas control concerns. As discussed in Section A.7 of this report, a postulated worst-case hydrogen combustion would have no significant adverse effect on plant safety functions. Accordingly, the NuScale containment vessel design does not actively control combustible gases. Because of these unique design features, and the lack of combustible gas control systems, the use of continuous oxygen and hydrogen monitors per 10 CFR 50.44(c)(4) would provide minimal safety benefits and therefore the NuScale design does not include them. The NuScale design meets the underlying intent of 10 CFR 50.44, which is to prevent a hydrogen combustion event that could result in a loss of containment structural integrity or accident mitigation features. Because the introduction of steam creates an inert containment, the inert containment requirements could be applied to NuScale and largely satisfied. However, NuScale anticipates at least a partial exemption from aspects of the monitoring requirements of 10 CFR 50.44(c)(4). Additionally, the framework for traditional inert containments does not fit the NuScale approach to combustible gases well. To avoid the complications of partial exemptions from 10 CFR 50.44(c) and to better tailor the safety analysis to the NuScale design, NuScale believes a more appropriate path forward is to meet 10 CFR 50.44 through the use of 10 CFR 50.44(d). The applicability of paragraph (d) to the NuScale design is based on the description of applicability in paragraph (c), which states “The requirements [paragraph (c)] apply only to water-cooled reactor designs with characteristics (e.g., type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to light water reactor designs licensed as of October 16, 2003.” NuScale utilizes cladding of a type and quantity similar to existing designs. However, the potential for production of combustible gases is not comparable due to the accident response characteristics. Moreover, the amount of combustible gases is not comparable viewed in relation to the high pressure design of the NuScale containment. Thus, NuScale believes that paragraph (d) can be applied to the NuScale design. Application of 10 CFR 50.44(d) would allow much greater flexibility in the design considerations and analysis needed to satisfy combustible gas control requirements, which would lead to the greatest assurance that the intent of 10 CFR 50.44 is met. If in the course of pre-application activities with the NRC it is discovered that the NRC interpretation of 10 CFR 50.44 is that paragraph (c) is applicable to the NuScale design, then NuScale would request a departure that would allow the approach of paragraph (d) to still be applied in lieu of paragraph (c) given the NuScale combustible gas control strategy.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			Further Consideration NuScale will seek NRC concurrence on the proper regulatory vehicle to utilize [either applying paragraph (c) with exemptions or verifying an express allowance to apply paragraph (d)] to ensure that the NuScale design is indeed meeting the underlying intent of, and is therefore compliant with, 10 CFR 50.44.
7.	50.46a & 50.34(f)(2)(vi)	Reactor Coolant System Venting	Regulatory Requirement 10 CFR 52.47(a)(4) requires for design certification applicants, “Analysis and evaluation of emergency core cooling system (ECCS) cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46a of this chapter.” 10 CFR 50.46a states in part: “Each nuclear power reactor must be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensable gases would cause the loss of function of these systems.” Substantively similar requirements for reactor coolant system venting capability are codified in 10 CFR 50.34(f)(2)(vi). Summary Basis for Gap Determination The underlying purpose of these requirements was to resolve post-TMI concerns that an accumulation of noncondensable gases could interfere with post-accident natural circulation or pump operation that might inhibit long-term cooling following an accident (see 68 FR 54123, at 54129, September 16, 2003). For the NuScale design, there is considerable evidence that an accumulation of noncondensable gases in the RCS or reactor pressure vessel could not inhibit post-accident core cooling flow. For this reason, the venting of noncondensable gases is unnecessary to ensure long term core cooling capability as contemplated for traditional LWRs by 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi). Therefore, NuScale intends to request an exemption from the requirements of 50.46a. See Section A.6 of this report for additional information. Further Consideration NuScale believes the appropriate approach is to request an exemption from 10 CFR 50.46a because the collection of noncondensable gases will not inhibit post-accident core cooling. 10 CFR 50.34(f)(2)(vi) is considered not technically relevant to the NuScale design in accordance with 10 CFR 52.47(a)(8) which requires compliance with the technically relevant portions of the Three Mile Island requirements. NuScale will seek NRC concurrence on this issue in the pre-application phase.
8.	50.54(m)(2)(i) and (iii)	Minimum Licensed Operator Staffing Requirements	Regulatory Requirement 10 CFR 50.54(m)(2)(i) states that “[e]ach licensee shall meet the minimum licensed operator staffing requirements” in the table specified in Section 50.54(m)(2)(i). 10 CFR 50.54(m)(2)(iii) states that “[w]hen a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times.”

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			Summary Basis for Gap Determination As detailed in Section A.1 of this report, licensee decisions regarding operator staffing levels, including the number, composition, and qualifications of licensed personnel, are more appropriately based on advanced features unique to the NuScale design rather than on large LWR-based staffing levels prescribed in 10 CFR 50.54(m)(2)(i) and (iii). As NRC recognized in NUREG-1791 (Ref. 5.23), "The design features and concepts of operation for new generations of advanced reactors, as well as the increased use of advanced, automated, and digital systems in existing plants, may lead applicants to request variations in the prescribed number, composition, or qualifications of licensed personnel." Through the use of passive systems and advanced control systems, the NuScale design should allow licensees to significantly reduce reliance on operators to safely operate the plant under normal and transient conditions. Therefore, consistent with NUREG-1791, reduced staffing levels at a NuScale plant is expected to "provide adequate assurance that public health and safety will be maintained at a level that is comparable to compliance with the current regulations." The final results of the human factors engineering (HFE) analysis based on the guidance provided in NUREG-0711, which is used to support a staffing exemption request per NUREG-1791 and justify the staffing plan for the NuScale design, will not be available until after initial DCA submittal (consistent with the industry norm in this area). However, some aspects of the NuScale Control Building design are based on significantly smaller staffing sizes than those required by this regulation, such that NRC approval of the Control Building design relates to acceptance of a licensee's smaller operator staffing sizes. Therefore, to support the design certification process and reduce COL applicant uncertainty, NuScale seeks to resolve both Control Building design and operator staffing levels at the design certification stage to the greatest extent possible. With respect to the Control Building design, while the final results of the HFE analysis will not be available until after DCA submittal, NuScale intends to provide intermediate information (e.g., HFE element implementation plans and results summary reports per NUREG-0711) throughout the pre-application phase and during the DCA review process. NuScale's plans for HFE deliverables at initial DCA submittal and throughout the review process were presented to the NRC in April 2014. The intent is that this information will provide assurance that the Control Building design is appropriately based on realistic expected staffing levels. With respect to operator staffing, 10 CFR 50.54 imposes conditions on operating licenses and thus is not directly applicable to a NuScale design certification. However, given the generic nature of staffing levels and the interrelated nature of staffing requirements and NuScale design features, resolving operator staffing levels within the design certification would be appropriate and beneficial. Resolving staffing within the NuScale design certification by exemption would be consistent with the approach outlined in SECY-11-098 (Reference 5.21), as well as NRC's discussion of exemptions during the 2007 Part 52 rulemaking (72 FR 49372) in which it stated "if the nature of the technical requirement is such that a subsequent applicant referencing the design certification would need an exemption from compliance with the requirement as applied to the applicant, then the Commission would include the exemption in the design certification rule itself." Further Consideration As part of pre-application activities, NuScale will seek concurrence with NRC regarding the appropriate process for submittal, review, and approval of control room staffing and related design issues. Important factors to consider in developing the path forward are the regulatory framework, the NuScale plan and schedule for relevant HFE analyses, and obtaining sufficient finality to support future license applications. The technical basis for the staffing plan will be based on HFE analysis of the NuScale plant using the guidance provided in NUREG-0711 (Reference 5.22) and NUREG-1791 (Reference 5.23).

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
9.	50.62(c)(1)	Reduction of Risk from ATWS Events	Regulatory Requirement 10 CFR 52.47(a)(15) requires a design certification applicant to include, “Information demonstrating how the applicant will comply with requirements for reduction of risk from anticipated transients without scram events in § 50.62.” 10 CFR 50.62(c)(1) states in part, “Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS.” Summary Basis for Gap Determination The design features required by 10 CFR 50.62(c)(1) were required for large LWRs based on the risk reduction it offered for ATWS events. These design features are either not included in the NuScale design or they are not required to reduce the risk from ATWS events as anticipated by the rule. The NuScale design does not include an auxiliary feedwater system. Therefore the portion of 50.62(c)(1) that requires a diverse capability to initiate auxiliary feedwater is not applicable to the NuScale design. A trip of the turbine on a module during ATWS conditions is not required to reduce the risk for the NuScale design, but because the rule is literally applicable to NuScale an exemption will be required. The NuScale design relies on diversity within the module protection system (MPS) to reduce the risk associated with ATWS events. Internal diversity within the MPS is a simpler approach that addresses the ‘diverse scram’ aspects of the ATWS rule, even though the ”diverse scram” aspect of paragraph (c)(2) of the ATWS rule is not literally applicable to NuScale. Additionally, the NuScale design was not restrained by an existing reactor trip system design (without internal diversity) that resulted in the additional diverse scram system required by 50.62(c)(2). Thus, internal diversity within the MPS is relied on to reduce the risk from ATWS events. NuScale submitted a white paper that describes the underlying purpose of the rule which is to reduce the risk from ATWS events (“NuScale Power Plant Design for ATWS and 10 CFR 50.62 Regulatory Compliance”, NP-ER-0000-2196, September 18, 2013). The proposed treatment of the rule was described in a presentation to the NRC, “Design for ATWS and 10 CFR 50.62 Regulatory Compliance”, PM-0114-5922-P, Feb 26, 2014. See also the detailed gap analysis results for SRP Section 10.4.9, Acceptance Criterion II.8, and SRP Section 15.8 in the NuScale electronic reading room for more information. Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that the underlying purpose of the rule is satisfied by reliance on diversity within the MPS to reduce the risk associated with ATWS events. Further, NuScale will seek to establish that • the portion of 50.62(c)(1) requiring diverse actuation of the auxiliary feedwater system is not applicable to the NuScale design because the NuScale design does not include an auxiliary feedwater system. • an exemption from the portion of 50.62(c)(1) that requires diverse capability to trip the turbine is appropriate because the underlying purpose of the ATWS rule is met by an alternative approach consistent with 10 CFR 50.12(a)(1).II.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
10.	50, App. A, GDC 17	Electric Power Systems	<u>Regulatory Requirement</u> GDC 17 requires in part, “Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions.” <u>Summary Basis for Gap Determination</u> As discussed further in Section A.3 of this report, the NuScale plant design supports a departure from GDC 17 by providing 1. safety-related passive systems designed to achieve safe shutdown and maintain core cooling and containment integrity, independent of AC or DC power sources, for an indefinite duration by requiring all SSCs to transition to their safety state upon loss of control or motive power. 2. sufficient battery capacity for other plant functions, including post-accident and pool monitoring, for a minimum of 72 hours following the onset of a design-basis event. 3. multiple nonsafety-related onsite and offsite electrical power sources for other functions such as post-accident monitoring, prevention of unnecessary challenges to safety systems, and emergency lighting. <u>Further Consideration</u> NuScale intends to pursue a departure from the criteria of GDC 17. The portion of GDC 17 for which departure will be sought is that requiring two physically independent offsite power supply circuits. As discussed in Section 3.1, the departure from general design criteria (GDC) 17 described herein would be identified and justified within the NuScale DCD submitted as part of the NuScale application for design certification pursuant to 10 CFR 52.47(a). Notwithstanding the above, NuScale recognizes that precedents exist whereby formal exemptions were issued for departures from GDC 17, the nature and justification of which were substantively equivalent to that described herein. Specifically, the NRC design certification approvals for the Westinghouse AP600 and AP1000 reactor designs included specific exemptions to the portion of GDC 17 requiring two physically independent offsite power supply circuits. In light of these precedents, NuScale will seek NRC concurrence during pre-application activities regarding the appropriate form of this departure as applied to the NuScale advanced reactor design.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
11.	50, App. A, GDC 27	Combined Reactivity Control Systems Capability	Regulatory Requirement GDC 27 states, "*Combined reactivity control systems capability.* The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained." Summary Basis for Gap Determination Unlike an ECCS at a typical PWR, the NuScale ECCS does not perform a poison addition safety function or provide any makeup function. NuScale interprets GDC 27 as allowing ECCS poison addition to be credited within the combined reactivity control capability, but not requiring it. The reactivity control systems associated with the NuScale design meet the requirements of GDC 27 with regards to reliably controlling reactivity changes and maintaining the capability of cooling the core without poison addition by the ECCS. NuScale does not believe a departure is needed from GDC 27. Note that NuScale does not interpret GDC 26 as requiring two safety related means of reactivity control. One of the independent reactivity control systems used to meet the requirements of GDC 26 in the NuScale design is the chemical volume control system, which is not safety related. Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that the rule is satisfied by NuScale's reactivity control systems, and no departure from GDC 27 is needed.
12.	50, App. A, GDC 33	Reactor Coolant Makeup	Regulatory Requirement GDC 33 states, "*Reactor coolant makeup.* A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation." Summary Basis for Gap Determination As discussed in Section A.10 of this report, the NuScale plant incorporates specific design provisions assuring adequate reactor coolant inventory to ensure that leaks do not result in core uncovery or loss of core cooling. Thus, a coolant makeup system as contemplated by GDC 33 is not appropriate for the NuScale design. Rather, a NuScale-specific principal design criterion for the assurance of adequate reactor coolant inventory is warranted as an alternative to GDC 33. The intent of this criterion would be to require that the reactor coolant pressure boundary and associated systems and components be designed to limit loss of reactor coolant so that an inventory adequate to perform the safety functions of the core decay heat removal systems (including the DHR system and the ECCS) is maintained under normal operation (including anticipated operational occurrences [AOO]) and postulated accident conditions. It is noted that a similar alternative design criterion to GDC 33 has been determined by the NRC to be acceptable in other applications, albeit to substantially different reactor technologies (References 5.6 and 5.7).

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that a departure from GDC 33 is warranted and appropriate, and that a NuScale-specific principal design criterion for the assurance of adequate reactor coolant inventory is warranted as an alternative to GDC 33. The proposed new criterion, to be finalized during design certification application activities, would represent a NuScale-specific principal design criterion to ensure that the NuScale design provides sufficient retention of coolant inventory in the event of a leak to maintain a decay heat removal path.
13.	50, App. A, GDC 40	Testing of Containment Heat Removal System	Regulatory Requirement GDC 40 states, "*Testing of containment heat removal system.* The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system." Summary Basis for Gap Determination As discussed in Section A.4 of this report, the passive NuScale containment heat removal system simply consists of the containment vessel steel walls and the heat transfer medium exterior to the containment vessel. The periodic functional and operational testing specified by GDC 40 is not relevant to the NuScale design. The passive design of the NuScale containment heat removal system provides assurance of adequate containment heat removal, with no reliance on electrical power, valve actuation, cooling water flow, or other active system/component operations. As detailed in Section A.4 of this report, even in the absence of nonsafety-related AC power or other active component operations, containment heat removal is assured for an indefinite duration. With no active components, the periodic functional and operational testing specified in this GDC is not relevant to the NuScale design. As part of design testing for design certification, NuScale intends to conduct performance tests of the containment heat removal function. This initial design testing will confirm operability of the passive containment heat removal system as a whole. Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that a departure from GDC 40 is appropriate to document that the periodic functional and operational testing specified in GDC 40 is not relevant to the NuScale containment heat removal system design.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
14.	50, App. A, GDC 41	Containment atmosphere cleanup	Regulatory Requirement GDC 41 states, "*Containment atmosphere cleanup.* Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure." Summary Basis for Gap Determination For the NuScale design, active systems contemplated by this criterion are not necessary to reduce fission product release to the environment or to ensure containment integrity following postulated accidents. As discussed in Appendix A, Section A.7, of this report, a postulated worst-case uncontrolled hydrogen-oxygen recombination would not challenge the integrity of the containment vessel. As discussed in Sections A.7 and A.8, of this report, the NuScale containment vessel design does not require an engineered safety feature (ESF) atmosphere clean-up system or pressure suppression systems that serve a fission product removal/dose mitigation function. NuScale design features provide assurance that, with no reliance on a containment ESF atmosphere cleanup system, the calculated dose is less than the criteria of 10 CFR 100.21, 10 CFR 50.34(a)(1)(ii)(D), and 10 CFR 52.47(a)(2)(iv). With consideration for the "as necessary" provision of GDC 41, and the determination that such systems are not necessary for the NuScale design, the NuScale design meets this GDC by having no such systems necessary to achieve the specified functions. Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that given the determination that the systems specified in GDC 41 are not necessary in the NuScale design, the absence of such systems is consistent with the "as necessary" provision of GDC 41, and thus no departure is needed.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
15.	50, App. A, GDC 42	Inspection of containment atmosphere cleanup systems	Regulatory Requirement GDC 42 states, "*Inspection of containment atmosphere cleanup systems.* The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems." Summary Basis for Gap Determination The underlying purpose of this GDC is to ensure the performance and reliability of containment atmosphere cleanup systems provided "as necessary" per GDC 41 to reduce the concentration of released fission products and assure containment integrity following postulated accidents. As indicated in the comments above for GDC 41, for the NuScale design, containment atmosphere cleanup systems are not necessary to ensure containment integrity or to reduce fission product release to the environment following postulated accidents. Thus, to design for periodic inspection of important components of such systems as specified in GDC 42 is not relevant to the NuScale design, particularly given that GDC 42 only requires "appropriate" inspections. Because the NuScale design lacks the systems that would be subject to the requirements of GDC 42, the NuScale design meets GDC 42 without any such inspections. Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that given the determination that the systems specified in GDC 41 are not necessary in the NuScale design, the absence of such systems is consistent with the "as necessary" provision of GDC 41, and thus GDC 42 is met for the NuScale design.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
16.	50, App. A, GDC 43	Testing of containment atmosphere cleanup systems	Regulatory Requirement GDC 43 states, "*Testing of containment atmosphere cleanup systems.* The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems." Summary Basis for Gap Determination The underlying purpose of this GDC is to ensure the performance and reliability of containment atmosphere cleanup systems provided "as necessary" per GDC 41 to reduce the concentration of released fission products and assure containment integrity following postulated accidents. As indicated in the comments above for GDC 41, for the NuScale design, containment atmosphere cleanup systems are not necessary to ensure containment integrity or to reduce fission product release to the environment following postulated accidents. Thus, to design for periodic pressure and functional testing of such systems as specified in GDC 43 is not relevant to the NuScale design, particularly given that GDC 43 only requires "appropriate" testing. Because the NuScale design lacks the systems that would be subject to the requirements of GDC 43, the NuScale design meets GDC 43 as there are no systems to design for periodic pressure and functional testing. Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that given the determination that the systems specified in GDC 41 are not necessary in the NuScale design, the absence of such systems is consistent with the "as necessary" provision of GDC 41, and thus GDC 43 is met for the NuScale design.
17.	50, App. A, GDC 52 and 50, App. J	Capability for containment leakage rate testing	Regulatory Requirement GDC 52 states, "The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure." Appendix J offers a prescriptive requirements option ("Option A") of meeting the containment leakage testing requirements described within Appendix J. "Option A" includes the performance of Type A, Type B, and Type C tests. Type A tests are intended to measure the primary reactor containment overall integrated leakage rate after the containment has been completed and is ready for operation and at periodic intervals thereafter. Type B tests are intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for primary reactor containment penetrations. Type C tests are intended to measure containment isolation valve leakage rates. Summary Basis for Gap Determination Due to the high pressure design of the NuScale containment vessel, compared to a typical LWR containment, Type A integrated leakage rate testing at containment design pressure presents challenges. NuScale does not intend to perform Type A testing as envisioned by Appendix J. NuScale proposes an alternative to Type A testing that consists of various inspection techniques that would have been impractical to implement at a typical LWR containment, but are practically deployable within the context of the NuScale design. Type A tests have historically been of little value to the operating fleet compared to the

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			reportable events detected by Type B and Type C leakage testing, which leads NuScale to believe that the NuScale alternative to Type A testing, in conjunction with Type B and Type C testing, will meet the underlying purpose of the regulations (i.e., ensuring a leak-tight barrier to the environment) to a greater extent and with fewer complications compared to conducting Type A testing. <u>Further Consideration</u> As part of pre-application activities, NuScale will seek NRC concurrence that the NuScale proposed alternative to GDC 52 and Appendix J integrated leakage rate testing still meets the underlying purpose of the regulations. The regulatory path forward in terms of the nature of the departure from these regulations will also be sought.
18.	50, App. A, GDC 55, 56, and 57	Containment Isolation	<u>Regulatory Requirement</u> GDC 55 states in part, “Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.” GDC 56 states in part, “Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:” and goes on to list the same four containment isolation valve provisions described in GDC 55 above. GDC 57 states “Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.” <u>Summary Basis for Gap Determination</u> NuScale has very few lines penetrating containment. The placement of containment isolation valves inside of containment was found to be technically unfavorable and would require the use of design features that deviate from designs with proven service experience. The harsher containment internal environment would likely reduce the reliability of valve isolation, even with rigorous environmental qualification, and the use of containment isolation valves inside of containment introduces other complications with regards to maintenance, likelihood of leakage, common cause failure, and increased number of penetrations. Therefore, the NuScale approach is to place all containment isolation valves outside of containment. The assessment of acceptability of alternate isolation provisions is documented generically in SRP Section 6.2.4. For lines that interface with the reactor coolant pressure boundary (RCPB) or containment atmosphere (GDCs 55 and 56),

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			the NuScale design features two containment isolation valves in series wherein the inboard valve is welded to the nozzle outside containment. In the context of the NuScale containment vessel this approach is anticipated to produce a higher assurance of reliable containment isolation compared to the “one valve inside and one valve outside of containment” traditional approaches detailed in GDC 55 and GDC 56. This configuration meets the underlying intent of GDC 55 and 56 with regards to assuring reliable containment isolation. The NuScale decay heat removal system (DHRS) represents a closed system inside and outside containment that does not contain containment isolation valves. The DHRS is an ESF system that is required to be operable when containment is isolated, which means that the absence of containment isolation valves in the DHR system increases system reliability by preventing the possibility of a malfunctioning valve disrupting the operation of the DHR during a transient. The DHR contains two passive barriers (the steam generator tubes and the DHR heat exchanger piping) in series that are designed to prevent coincident failure and a single failure does not cause bypass. This design approach is consistent with the underlying philosophy of having two barriers and preference for passive barriers and ESF system operability as described in SRP 6.2.4. <u>Further Consideration</u> As part of pre-application activities, NuScale will seek NRC concurrence that the NuScale design meets the underlying intent of GDCs 55, 56 and 57. Consistent with previous DCAs, these departures will be identified and justified in the DCD on a case-by-case basis.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
19.	50, App. K	ECCS Evaluation Models	Regulatory Requirement 10 CFR 52.47(a)(4) requires for design certification applicants, “Analysis and evaluation of emergency core cooling system (ECCS) cooling performance and the need for high-point vents following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §§ 50.46 and 50.46a of this chapter.” 10 CFR 50.46(a) allows the use of either a realistic (best-estimate) evaluation model pursuant to 10 CFR 50.46(a)(1)(i) or a conservative evaluation model pursuant to 10 CFR 50.46(a)(1)(ii) and 10 CFR 50, Appendix K. Summary Basis for Gap Determination NuScale intends to use the conservative evaluation model pursuant to 10 CFR 50.46(a)(1)(ii) and 10 CFR 50, Appendix K. The NuScale Power Module is designed to greatly reduce the consequences of design-basis LOCAs. Consequently, many of the phenomena that are the subject of Appendix K requirements are not encountered in the design-basis LOCA for the NuScale design. Thus, only a subset of the phenomena that are addressed in Appendix K will be encountered in design-basis LOCAs. NuScale submitted a white paper to the NRC entitled “Satisfaction of Appendix K Requirements for the NuScale Power Module” (WP-1013-5124) detailing NuScale’s anticipated satisfaction of Appendix K requirements. As detailed therein, the NuScale ECCS evaluation model is based on the Appendix K model characteristics and phenomena that are relevant to design-basis LOCAs, while excluding requirements that are not applicable and phenomena that do not occur. Where phenomena are not expected to occur, limitations are placed on application of the methodology to maintain conservative margins that assure that the excluded phenomena are not encountered. For example, Appendix K requirement C.6 states “The characteristics of rotating primary system pumps (axial flow, turbine, or centrifugal) shall be derived from a dynamic model…” As there are no primary coolant pumps in the NuScale Power Module, there are no pumps subject to requirement C.6 and the requirement is not applicable. Likewise, Appendix K paragraph B, states “Each evaluation model shall include a provision for predicting cladding swelling and rupture…” Calculated cladding temperatures for NuScale design-basis LOCAs are well below the threshold for clad swelling and rupture. Hence, neither clad swelling nor rupture will occur during any of the design-basis LOCA events. To account for the required provision in the evaluation model, limitations on clad temperature and differential pressure will be established in the model that provide sufficient margin to preclude the need for a clad rupture model With this approach, no exemptions to Appendix K requirements are necessary. See the white paper entitled “Satisfaction of Appendix K Requirements for the NuScale Power Module” (WP-1013-5124) for more information. Further Consideration Based on the methodology and examples provided above and detailed in WP-1013-5124, the NuScale evaluation model is expected to fully comply with Appendix K requirements. NuScale will seek NRC concurrence during pre-application activities regarding the appropriate application of Appendix K requirements.

3.2 Standard Review Plan (NUREG-0800), Branch Technical Positions, and Sub-Tier Guidance; Interim Staff Guidance for Design Certification Applications

NuScale performed a detailed review of the SRP, recognizing that this guidance will most directly impact preparation and regulatory review of the NuScale application for design certification. This review included branch technical positions and guidance and standards referenced within, and thus sub-tier to, the SRP. The gap analysis review for applicability was directed towards the acceptance criteria of each SRP section. However, the review considered the relevance of sub-tier guidance whether referenced in the acceptance criteria or in other portions of the SRP section being reviewed.

As documented in the detailed tables available in the NuScale electronic reading room, it was determined that a number of SRP acceptance criteria and sub-tier guidance documents are not relevant and thus would be inappropriate to apply, either in whole or in part, to the NuScale power plant design. Keeping this determination in mind, each SRP section was then assessed to determine whether or not a NuScale-specific DSRS would be desirable to enhance NuScale's understanding of the NRC's review process and facilitate communication between the NRC and NuScale with regard to SRP guidance that describes methods/approaches that the staff has found acceptable for meeting NRC requirements and the methods/approaches that NuScale intends to pursue. Based on these considerations, each SRP section was assigned one of the following dispositions:

1. *"No, SRP is sufficient"*- A NuScale specific DSRS is not desired for this section. The SRP is sufficient.

2. *"No, not applicable"*- An aspect of the NuScale design (such as the nonexistence of a particular system) renders the section "not applicable" and hence a NuScale-specific DSRS would not be appropriate.

3. *"Yes, base on SRP"*- A NuScale-specific DSRS is desired for this section and the SRP appears to be the most relevant document for the NRC to use as a starting point during the creation of the NuScale specific DSRS.

4. *"Yes, base on mPower DSRS"*- A NuScale-specific DSRS is desired for this section and the draft mPower DSRS appears to be the most relevant document for the NRC to use as a starting point during the creation of the NuScale specific DSRS.

5. *"Yes"*- A NuScale specific DSRS is desired for this section, but there was no existing document that stood out as a particularly pertinent starting point for the creation of the NuScale-specific DSRS. This could be due to a NuScale unique design feature or immaterial differences between the SRP and draft mPower DSRS resulting in a lack of preference.

The results of this NuScale-specific DSRS desirability assessment are summarized in Table 3-2 of this report.

From pre-application discussions with the NRC, NuScale understands that the NRC intends to use the NuScale design-specific review standard, not only in its review of the design certification application, but also in review of combined license applications referencing the NuScale design. Thus, NuScale's assessment of each SRP section also considered the extent to which a particular section may only be applicable to the NRC's review of combined license applications that reference the NuScale design and not be applicable to the design certification effort. This added consideration was taken since there are a number of SRP sections that govern site-specific information that are not relevant to (i.e., would not be part of) a certified standard plant design, but are germane to a license application review since such site-specific information is

available to the license applicant. In such instances, a note was made in the comments within the detailed tables available in the NuScale electronic reading room and within the comments contained within Table 3-2 of this report.

For example, SRP Sections 2.1.1 and 2.1.3 govern the review of power plant site location and description and site population distribution, respectively, which is site-specific information not available to an applicant for design certification. These SRP sections would not be relevant to the NRC's review of the NuScale application for design certification, but would be pertinent to the review of a combined license application where such site-specific information would be available.

Note: It is NuScale's understanding that Chapter 7 of the SRP is to be superseded by regulatory approaches substantively similar to the ones described in Chapter 7 of the initial draft of the mPower DSRS. NuScale is currently utilizing Chapter 7 of the draft mPower DSRS in house for DCA production related activities. Chapter 7 of the draft mPower DSRS was unique in how different its review approach was from Chapter 7 of the SRP. Therefore it was pertinent for NuScale to direct Chapter 7 of its gap analysis assessment towards the draft mPower DSRS rather than to the SRP, which was the bulk of the rest of the gap analysis effort. For historical purposes, NuScale's assessment of Chapter 7 of the SRP remains in the detailed tables available for NRC review in the electronic reading room. Having reconciled Chapter 7 of the draft mPower DSRS with the NuScale design, NuScale recommends the following items be kept in mind by the NRC during the creation of Chapter 7 of the NuScale specific DSRS. NuScale anticipates a NuScale specific DSRS for Chapter 7 based on the draft mPower DSRS.

- The NuScale I&C development lifecycle is different than that of the conceptual waterfall lifecycle listed in RG 1.152. NuScale will map the applicable tasks from the RG 1.152 lifecycle model to the NuScale I&C development lifecycle. Compliance with IEEE 7-4.3.2 clause 5.5 will be conditioned by the choice of FPGA technology, which makes some tests not applicable (e.g., calculation tests, watchdog timer tests, etc.).
- For RG 1.168, the requirements of IEEE 1012 will be tailored to the NuScale I&C development lifecycle, which is different than that of the conceptual waterfall lifecycle listed in IEEE 1012. NuScale will map the applicable tasks from IEEE 1012 to the NuScale I&C development lifecycle. NuScale will also take exception to some of the administrative mandatory requirements in the standard that conflict with established engineering or quality assurance (QA) documentation requirements. The requirements of IEEE 1028 will be tailored to the NuScale I&C development lifecycle. NuScale will identify the specific tasks from IEEE 1028 to the NuScale I&C development lifecycle. NuScale will also take exception to some of the administrative mandatory requirements in the standard that conflict with established engineering or QA documentation requirements.
- For RG 1.169, the requirements of IEEE 828 will be tailored to the NuScale I&C development lifecycle, which is different that the conceptual waterfall lifecycle listed in RG 1.152. NuScale will map the applicable tasks from IEEE 828 to the NuScale I&C development lifecycle. NuScale will also take exception to some of the administrative mandatory requirements in the standard that conflict with established engineering or QA documentation requirements.
- For RG 1.170, the requirements of IEEE 829 will be tailored to the NuScale I&C development lifecycle, which is different than that of the conceptual waterfall lifecycle listed in RG 1.152. NuScale will map the applicable tasks from IEEE 829 to the NuScale I&C development lifecycle. NuScale will also take exception to some of the administrative mandatory requirements in the standard that conflict with established engineering or QA documentation requirements.
- For RG 1.171, NuScale will take exception to some of the administrative mandatory requirements in IEEE 1008 that conflict with established engineering or QA documentation requirements.

- For RG 1.172, NuScale will take exception to some of the administrative mandatory requirements in IEEE 830 standard that conflict with established engineering or QA documentation requirements.
- For RG 1.173, the requirements of IEEE 1074 will be tailored to the NuScale I&C development lifecycle, which is different than that of the conceptual waterfall lifecycle listed in RG 1.152. NuScale will map the applicable tasks from IEEE 1074 to the NuScale I&C development lifecycle. NuScale will also take exception to some of the administrative mandatory requirements in the standard that conflict with established engineering or QA documentation requirements.
- For RG 1.180, aspects of this guidance related to the design of standard plant SSCs to address effects of electromagnetic and radio-frequency interference (EMI/RFI) are applicable to the NuScale application for design certification. Aspects of this guidance related to the design of site-specific SSCs and installation and testing practices for addressing the effects of EMI/RFI and power surges on safety-related I&C systems are the responsibility of the combined license applicant and are not within the scope of the NuScale application for design certification. The NuScale DCD will identify the general design requirements for electromagnetic compatibility protection for I&C systems, using the applicable guidance from IEEE 665. For RG 1.180, the NuScale DCD will identify the general design requirements for lightning protection for I&C systems, using the applicable guidance from IEEE Std 1050-1996 and IEEE 665-1995. NuScale will apply the guidance for EMI/RFI protection from IEEE Std 1050-1996 Sections 5.2.6, 5.3, 5.4, 5.5, 6, and 7.5 to the design of I&C systems. IEEE Std 665 only provides general guidance and does not contain specific mandatory design requirements. The NuScale DCD will describe the use of this guide as a reference document rather than a compliance document. Sections 5.5.5.6 and 5.5.6 of IEEE Std 665-1995 are applicable to I&C systems. NuScale will not use IEEE Std 518 as a guidance document, since it has been withdrawn by IEEE.
- For RG 1.204, the NuScale DCD will identify the general design requirements for lightning protection for I&C systems, using the applicable guidance from IEEE Std. 1050-1996, IEEE Std. 665-1995, and IEEE Std. C62.23-1995. NuScale will apply the guidance for EMI/RFI protection from IEEE Std. 1050-1996 Sections 5.2.6, 5.3, 5.4, 5.5, 6, and 7.5 to the design of I&C systems. IEEE Std. 665 and IEEE Std. C62.23 only provide general guidance and do not contain specific mandatory design requirements. The NuScale DCD will describe the use of this guide as a reference document rather than a compliance document. The sections of the IEEE guide applicable to I&C systems are identified as: IEEE Std. 665-1995 Sections 5.5.5.6 and 5.5.6 and IEEE Std. C62.23 Sections 6.2.1.1, 6.2.2, and 6.3. IEEE Std. 666-1991 is not applicable for lightning protection for I&C systems.
- NuScale's diversity and defense in depth strategy will preclude the need for a separate diverse actuation system for ATWS mitigation for 10 CFR 50.62.
- NuScale design does not have interlocks for emergency core cooling system (ECCS) accumulator valves, since the NuScale design does not have a traditional ECCS.
- 10 CFR 50.34(f)(2)(xxiii) requires an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. This is not applicable to the NuScale design. This rule explicitly states it is applicable to B&W designed plants. Such an action is not beneficial for the NuScale design.
- 10 CFR 50.34(f)(2)(xii), which requires auxiliary feedwater (AFW) system flow indication in the control room is not directly applicable to the NuScale design. See item 3 of Table 3-1 in this report for more information.
- Compliance with IEEE 7-4.3.2 clause 5.5 will be conditioned by the choice of field programmable gate array (FPGA) technology, which makes some tests not applicable (e.g., calculation tests, watchdog timer tests, etc.).

- With respect to draft mPower DSRS Section 7.2.9, note that maintenance support equipment may access the safety equipment via a network connection made for the maintenance activity in lieu of a local panel connection.
- With respect to draft mPower DSRS Section 7.2.11, note that the NuScale design has both safety and nonsafety systems that are shared and controlled by either the plant protection system or plant control system, respectively.

Table 3-2 Assessment of desirability for a NuScale specific DSRS

Section	Section Title	DSRS desired by NuScale?	Comments
1.0	Introduction and Interfaces	No, SRP is sufficient	
2.0	Site Characteristics and Site Parameters	No, SRP is sufficient	
2.1.1	Site Location and Description	No, SRP is sufficient	Identification of site location and description is not applicable for standard design certification reviews. Therefore, this section will only be utilized for the COL application.
2.1.2	Exclusion Area Authority and Control	No, SRP is sufficient	Exclusion area authority and control information is not applicable for standard design certification reviews. Therefore, this section will only be utilized for the COL application.
2.1.3	Population Distribution	No, SRP is sufficient	Identification of population distribution is not applicable for standard design certification reviews. Therefore, this section will only be utilized for the COL application.
2.2.1-2.2.2	Identification of Potential Hazards in Site Vicinity	No, SRP is sufficient	Identification of potential hazards is not applicable for standard design certification reviews. Therefore, this section will only be utilized for the COL application.
2.2.3	Evaluation of Potential Accidents	No, SRP is sufficient	Evaluation of potential accidents is not applicable for standard design certification reviews. Therefore, this section will only be utilized for the COL application.
2.3.1	Regional Climatology	No, SRP is sufficient	
2.3.2	Local Meteorology	No, SRP is sufficient	
2.3.3	Onsite Meteorological Measurements Programs	No, SRP is sufficient	There are no postulated site parameters for a design certification related to an onsite meteorological program. Therefore, this section will only be utilized for the COL application.

Section	Section Title	DSRS desired by NuScale?	Comments
2.3.4	Short-Term Dispersion Estimates for Accident Releases	No, SRP is sufficient	
2.3.5	Long-Term Atmospheric Dispersion Estimates for Routine Releases	No, SRP is sufficient	
2.4.1	Hydrologic Description	No, SRP is sufficient	
2.4.2	Floods	No, SRP is sufficient	
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	No, SRP is sufficient	
2.4.4	Potential Dam Failures	No, SRP is sufficient	
2.4.5	Probable Maximum Surge and Seiche Flooding	No, SRP is sufficient	
2.4.6	Probable Maximum Tsunami Hazards	No, SRP is sufficient	
2.4.7	Ice Effects	No, SRP is sufficient	
2.4.8	Cooling Water Canals and Reservoirs	No, SRP is sufficient	
2.4.9	Channel Diversions	No, SRP is sufficient	
2.4.10	Flooding Protection Requirements	No, SRP is sufficient	
2.4.11	Low Water Considerations	No, SRP is sufficient	

Section	Section Title	DSRS desired by NuScale?	Comments
2.4.12	Groundwater	No, SRP is sufficient	
2.4.13	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	No, SRP is sufficient	
2.4.14	Technical Specifications and Emergency Operation Requirements	No, SRP is sufficient	
2.5.1	Basic Geologic and Seismic Information	No, SRP is sufficient	There are no postulated site parameters for a standard design certification related to basic geologic and seismic information. Therefore, this section will only be utilized for the COL application.
2.5.2	Vibratory Ground Motion	No, SRP is sufficient	
2.5.3	Surface Faulting	No, SRP is sufficient	
2.5.4	Stability of Subsurface Materials and Foundations	No, SRP is sufficient	
2.5.5	Stability of Slopes	No, SRP is sufficient	
SRP CHAPTER 3			
3.2.1	Seismic Classification	No, SRP is sufficient	
3.2.2	System Quality Group Classification	No, SRP is sufficient	

Section	Section Title	DSRS desired by NuScale?	Comments
3.3.1	Wind Loadings	Yes, base on mPower DSRS	The specific language of this SRP section specifies design based on site-specific historical wind speed information. The NuScale design certification application will be based on a postulated site parameter value for extreme wind speed that is intended to bound the majority of candidate sites.
3.3.2	Tornado Loadings	Yes, base on mPower DSRS	The specific language of this SRP section specifies design based on site-specific historical tornado wind speed information. The NuScale design certification application will be based on a postulated site parameter value for tornado wind speed that is intended to bound the majority of candidate sites.
3.4.1	Internal Flood Protection for Onsite Equipment Failures	Yes, base on mPower DSRS	
3.4.2	Analysis Procedures	Yes, base on mPower DSRS	The specific language of this SRP section specifies design based on site-specific historical flood and groundwater level information. The NuScale design certification application will be based on postulated site parameter values for flood and groundwater levels that are intended to bound the majority of candidate sites. A portion of this guidance is applicable only to designs sited in locations where the maximum flood level is higher than the proposed plant grade. The NuScale design will be based on a postulated site parameter value for maximum flood level that is at or lower than the proposed plant grade.
3.5.1.1	Internally Generated Missiles (Outside Containment)	Yes, base on mPower DSRS	
3.5.1.2	Internally-Generated Missiles (Inside Containment)	Yes, base on mPower DSRS	

Section	Section Title	DSRS desired by NuScale?	Comments
3.5.1.3	Turbine Missiles	Yes, base on SRP	As discussed in Section A.5 of this report, compared to a large LWR, the NuScale plant design includes plant SSC designs and layouts that result in considerably reduced exposure of essential SSCs to potential turbine missiles. Specifically, in the NuScale design, essential SSCs are located within the reactor building, such that the reactor building represents the engineered barrier for protection of these SSCs. The design of the reactor building ensures that the probability of barrier perforation (P_2) is less than or equal to 10^{-7} per year per plant. Thus, the probability of unacceptable damage from turbine generated missiles (i.e., P_4) will be less than or equal to 10^{-7} per year per plant as specified in this acceptance criterion. As a result of these design features, adequate turbine missile protection does not rely on management of turbine missile generation probability (P_1) or SSC damage probability (P_3). Rather, consistent with RG 1.115, Revision 2, NuScale will satisfy the criteria of GDC 4 by the appropriate placement of the turbine generators, combined with the proper design and use of missile barriers (i.e., the reactor building) to protect essential SSCs against potential turbine-generated missiles.
3.5.1.4	Missiles Generated by Tornadoes and Extreme Winds	Yes, base on mPower DSRS	
3.5.1.5	Site Proximity Missiles (Except Aircraft)	No, SRP is sufficient	This guidance specifies information that is site-specific and as such is applicable only to applicants for a construction permit/operating license, an early site permit, or a combined license. Verification of the capability of essential SSCs to withstand site proximity missile effects requires site-specific information that is the responsibility of the applicant for a construction permit/operating license, an early site permit, or a combined license. Consistent with SRP Section 1.0, Appendix A, RG 1.206, Regulatory Position C.III.4, and ESP/DC/COL-ISG-015, the NuScale design certification application will contain combined operating license (COL) information items, as appropriate, that describe the information (such as that governed by this acceptance criterion) that is deferred to the license/permit applicant referencing the certified design.
3.5.1.6	Aircraft Hazards	No, SRP is sufficient	Applications for design certifications do not contain general descriptions of site characteristics because this information is site-specific and will be addressed by the combined license applicant.

Section	Section Title	DSRS desired by NuScale?	Comments
3.5.2	Structures, Systems, and Components to be Protected from Externally-Generated Missiles	Yes, base on mPower DSRS	This SRP section refers to RG 1.115, Rev. 1. Revision 2 to RG 1.115 was issued in January 2012. NuScale will apply the current RG 1.115, Rev. 2, to design activities in support of its application for design certification.
3.5.3	Barrier Design Procedures	Yes, base on mPower DSRS	The NuScale design does not include composite or multi-element barriers. This SRP section makes reference to ANSI/AISC N690-1994 with Supplement 2 (2004). NuScale intends to apply the 2006 version of this standard.
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	No, SRP is sufficient	
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	No, SRP is sufficient	
3.6.3	Leak-Before-Break Evaluation Procedures	No, SRP is sufficient	NuScale intends to use leak–before-break for piping inside containment because of the limited space. NuScale has done a study based on the current SRP information. Based on work to date, NuScale doesn't believe a NuScale-specific DSRS is necessary.
3.7.1	Seismic Design Parameters	Yes, base on mPower DSRS	A portion of this SRP section is applicable only to non-standard designs, and thus is not applicable to the NuScale application for design certification or combined license applications referencing the NuScale design. Certain aspects of this guidance require site-specific information (that is the responsibility of the combined license applicant) and/or specify the use of generic response spectra. The NuScale certified seismic design response spectra (CSDRS) is being developed to envelop a number of spectra from various reactor sites. Thus, in certain instances, the CSDRS incorporates assumptions or parameters more conservative than those specified in RG 1.60. For example, the NuScale CSDRS considers the generic response spectra provided in RG 1.60, while broadening the peak of the spectra up to 16Hz.

Section	Section Title	DSRS desired by NuScale?	Comments
3.7.2	Seismic System Analysis	Yes, base on mPower DSRS	Site-specific site investigation activities are the responsibility of the combined license applicant referencing the certified design, and are not applicable to the design certification application.
3.7.3	Seismic Subsystem Analysis	No, SRP is sufficient	
3.7.4	Seismic Instrumentation	No, SRP is sufficient	This SRP section is applicable except for aspects that 1. govern programmatic/operational activities that are not within the scope of design certification. 2. refer to SSC configurations that are not part of the NuScale design. For the latter (Item 2), examples include reference to the "containment structure" and specification of accelerograph locations at the "containment foundation," and "two elevations… on a structure inside the containment." A typical large LWR containment is a massive permanent structure requiring a Seismic Category I foundation and involving multiple levels, subcompartments, and internal structures. As discussed in Section A.7 of this report, the NuScale containment vessel is a portable steel component, as opposed to a building/structure. As such, the containment vessel does not have levels, subcompartments, or a foundation as contemplated by this guidance.
3.8.1	Concrete Containment	No, not applicable	This SRP section is applicable only to LWRs whose design includes concrete containments or steel and concrete containments. As discussed in Section A.7 of this report, the NuScale containment vessel is a steel containment (i.e., it does not use concrete in its design).

Section	Section Title	DSRS desired by NuScale?	Comments
3.8.2	Steel Containment	Yes, base on SRP	The NuScale containment vessel design is different compared to a typical containment structure, and in some ways is similar to a typical reactor vessel. The design of the NuScale containment vessel is such that the codes cited in this SRP section should be supplemented by ASME Code, Section III, Division 1, Subsection NB, and ASME Code, Section XI, Subsection IWB, where these sections are more conservative. A portion of this guidance pertains to combustible gas control systems installed in containment. As discussed in Section A.7 of this report, the NuScale containment design is such that its integrity does not rely on combustible gas control systems. Thus, the NuScale design does not include combustible gas control systems. SEI/ASCE Std. 37 02 governs the effects of temporary construction loads and environmental loads on containment. The NuScale containment vessel will be constructed in an enclosed fabrication facility protected from environmental effects and shipped to the plant site. Hence, this standard is not applicable to the NuScale design. Sub-tier NUREG/CR 6906 is only applicable to free-standing steel containments, steel-lined reinforced concrete containments, and steel lined pre-stressed concrete containments, and thus is not applicable to the NuScale design. ASME Code Case N-284, Revision 1, has been superseded. NuScale intends to apply the current ASME Code Case N-284, Revision 2, unless superseded by a later endorsed revision.
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	Yes, base on SRP	Aspects of this SRP section related to concrete containments or the use of safety-related concrete support structures, anchoring components, and radiation shields inside containment are not applicable, since the NuScale design does not involve the use of concrete inside the containment vessel. Additional details of the NuScale containment vessel design are provided in Section A.7 of this report.

Section	Section Title	DSRS desired by NuScale?	Comments
3.8.4	Other Seismic Category I Structures	Yes, base on SRP	This SRP section, via reference to RG 1.206, specifies a description of containment enclosure buildings, fuel storage buildings, control buildings, and diesel generator buildings, which are Seismic Category I structures typically found at large LWR plant sites. The NuScale design does not include many of these buildings. Nevertheless, the NuScale application will contain descriptive information of all Seismic Category I structures as specified by this guidance. This SRP section references RG 1.142 and RG 1.199, which endorse the 1997 and 2001 versions (or portions thereof) of ACI 349, respectively. NuScale intends to use the 2006 version of the ACI 349 standard. This SRP section references ANSI/AISC N690-1994 with Supplement 2 (2004). NuScale intends to use the 2006 version of ANSI/AISC N690. NuScale will perform code reconciliation, as appropriate and necessary, to support the use of ACI 349 2006 and ANSI/AISC N690 2006. Aspects of this SRP section related to earth retaining walls are not applicable to the NuScale design certification application since the NuScale standard plant design does not involve the use of earth retaining walls. Implementation of inservice inspection programs as specified by this guidance is site-specific and therefore is to be addressed by the combined license applicant.
3.8.5	Foundations	Yes, base on SRP	Portions related to containment foundations are not applicable to the NuScale design. As discussed in Section A.7 of this report, the NuScale containment vessel is a portable steel component, as opposed to a building/structure. As such, the containment vessel does not have a “foundation” as contemplated by this guidance. This SRP section references RG 1.142 and RG 1.199, which endorse the 1997 and 2001 versions (or portions thereof) of ACI 349, respectively. NuScale intends to use the 2006 version of the ACI 349 standard. This SRP section references ANSI/AISC N690-1994 with Supplement 2 (2004). NuScale intends to use the 2006 version of ANSI/AISC N690. NuScale will perform code reconciliation, as appropriate and necessary, to support the use of ACI 349 2006 and ANSI/AISC N690 2006. Implementation of inservice inspection programs as specified by this guidance is site-specific and therefore is to be addressed by the combined license applicant.

Section	Section Title	DSRS desired by NuScale?	Comments
3.9.1	Special Topics for Mechanical Components	No, SRP is sufficient	
3.9.2	Dynamic Testing and Analysis of Systems, Structures, and Components	No, SRP is sufficient	Aspects related to test performance and associated corrective actions (as required) are the responsibility of the combined license applicant/holder referencing the certified design.
3.9.3	ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures	No, SRP is sufficient	
3.9.4	Control Rod Drive Systems	Yes, base on SRP	This SRP section refers to RG 1.29, which is not applicable in this context, i.e., it is not related to descriptive information to be provided for control rod drive systems. It appears that the intended sub-tier guidance reference was RG 1.206, Section C.I.3.9.4.1, which is substantively similar in content to the description in Section I, Areas of Review, of SRP Section 3.9.4, Item 1.
3.9.5	Reactor Pressure Vessel Internals	Yes, base on SRP	There is no guidance in existence for SGs and pressurizers mounted inside the RPV. That guidance should be developed and agreed upon prior to DC submittal. Need to make sure there is no overlap (or conflicting requirements) in the Chapter 4 DSRS/SRPs.

Section	Section Title	DSRS desired by NuScale?	Comments
3.9.6	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	No, SRP is sufficient	Safety-related pumps are not used in the NuScale design. The only pumps that fall within the scope of this guidance in the NuScale design are the chemical and volume control (CVC) system pumps. These pumps are ASME Class III because they contain reactor coolant during normal operation, but they serve no safety function. Therefore, relief from some testing requirements that are intended to confirm pumping capability may be requested in accordance with Acceptance Criterion II.5 of this SRP section. This SRP section refers to guidance applicable only to reactor licensees/applicants that are developing/revising a risk-informed, performance-based inservice testing program for pumps, valves, and dynamic restraints. Development and implementation of a risk-informed, performance-based inservice testing program would be the responsibility of combined license applicants that reference the NuScale certified design (upon NRC approval), and that elect to implement such a program. A portion of this section specifies operational activities, including implementation of preservice testing, inservice testing and inspection, and motor-operated valve testing programs, that are the responsibility of the combined license applicant referencing the certified design.
3.9.7	Risk-Informed Inservice Testing	No, SRP is sufficient	Development and implementation of a risk-informed, performance-based inservice testing program would be the responsibility of combined license applicants that reference the NuScale design, and that elect to implement such a program.
3.9.8	Risk-Informed Inservice Inspection of Piping	No, SRP is sufficient	Development and implementation of a risk-informed, performance-based inservice inspection program for piping would be the responsibility of combined license applicants that reference the NuScale design, and that elect to implement such a program.

Section	Section Title	DSRS desired by NuScale?	Comments
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	No, SRP is sufficient	RG 1.100 and RG 1.148 and related standards have been superseded by more current revision/deletion. Per Federal Register notice dated January 19, 2010 (75 FR 2894), ANSI/ASME N278.1-1975 is superseded by ASME QME-1. As endorsed by RG 1.100, Rev. 3, ASME QME-1-2007 is applicable to the NuScale application for design certification. Aspects related to qualification records developed for standard plant SSCs during initial design are applicable to the NuScale application for design certification. Maintaining and updating these records, and the development of qualification records for site-specific SSCs outside the scope of the NuScale standard plant, are the responsibility of the combined license applicant.
3.11	Environmental Qualification of Mechanical and Electrical Equipment	Yes, base on mPower DSRS	Portions of this SRP section are applicable only to reactor designs that use continuous duty Class 1E motors. The NuScale design does not use continuous duty Class 1E motors. SRP Section 3.11 refers to RG 1.131 as containing NRC endorsement of IEEE Std. 383-1974. By Federal Register notice dated April 20, 2009 (74 FR 18000), the NRC announced the withdrawal of RG 1.131 because its guidance is replaced by RG 1.211. RG 1.211 endorses IEEE Std. 383-2003. NuScale intends to implement IEEE Std. 383-2003 as endorsed by RG 1.211 (April 2009). SRP Section 3.11 refers to RG 1.156 as containing NRC endorsement of IEEE Std. 572-1985. RG 1.156 has been revised, and the new Revision 1 endorses IEEE Std. 572-2006. NuScale intends to implement IEEE Std. 572-2006 as endorsed by RG 1.156, Rev. 1. IEEE Std. 323-1971 is applicable only to Category II criteria of NUREG 0588, which is explicitly stated in Acceptance Criterion II.1 as not applicable to any future plants.
3.12	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports	No, SRP is sufficient	
3.13	Threaded Fasteners - ASME Code Class 1, 2, and 3	No, SRP is sufficient	

Section	Section Title	DSRS desired by NuScale?	Comments
BTP 3-1	Classification of Main Steam Components Other than the Reactor Coolant Pressure Boundary for BWR Plants	No, not applicable	Applies only to BWRs.
BTP 3-2	Classification of BWR/6 Main Steam and Feedwater Components Other than the Reactor Coolant Pressure Boundary	No, not applicable	Applies only to BWRs.
BTP 3-3	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	No, SRP is sufficient	
BTP 3-4	Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	No, SRP is sufficient	
SRP CHAPTER 4			
4.2	Fuel System Design	Yes	
4.3	Nuclear Design	Yes	
4.4	Thermal and Hydraulic Design	Yes, base on mPower DSRS	References to reactor coolant pump related issues are not applicable to the NuScale design.
4.5.1	Control Rod Drive Structural Materials	No, SRP is sufficient	This guidance specifies use of ASME NQA 1 1994. The NuScale quality assurance program description (QAPD) will be based on ANSI/ASME NQA 1-2008 with NQA 1a 2009 addenda, as endorsed by RG 1.28, Rev. 4.
4.5.2	Reactor Internal and Core Support Structure Materials	Yes	
4.6	Functional Design of Control Rod Drive System	Yes, base on SRP	

Section	Section Title	DSRS desired by NuScale?	Comments
BTP 4-1	Westinghouse Constant Axial Offset Control	No, not applicable	NuScale does not intend to use the constant axial offset control operating scheme.
SRP CHAPTER 5			
5.2.1.1	Compliance w/ Codes and Standards Rule, 10 CFR 50.55a	Yes, base on mPower DSRS	A NuScale specific DSRS, based on the mPower DSRS, is needed due to the deviations in methodology needed to demonstrate compliance with the codes and standards.
5.2.1.2	Applicable Code Cases	Yes, base on mPower DSRS	A NuScale specific DSRS, based on the mPower DSRS, is needed due to the deviations in methodology needed to demonstrate compliance with the codes and standards.
5.2.2	Overpressure Protection	Yes	The NuScale design does not contain secondary side overpressure protection inside the containment isolation valves. The low temperature overpressure protection will consist of the ECCS valves, which is unique.
5.2.3	Reactor Coolant Pressure Boundary Materials	Yes, base on mPower DSRS	Refinements to the guidance found within the SRP contained within the mPower DSRS, as well as NuScale specific definitions of reactor coolant system and reactor coolant pressure boundary, lead to the desirability of a NuScale-specific DSRS section based upon the mPower DSRS. See NuScale comments on draft mPower DSRS section 5.2.3 found within NP-LO-0913-4695 for more details.
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection & Testing	No, SRP is sufficient	Since the NuScale design does not have a traditional reactor coolant system with reactor coolant loops, the reactor coolant loop related requirements of SRP section 5.2.4 may not be applicable. However, NuScale does not believe that this warrants a NuScale specific DSRS section.
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	Yes, base on mPower DSRS	To allow for the unique NuScale design that precludes the practical separation of identified and unidentified leakage within the containment vessel, any leakage into the containment vessel will be conservatively assumed to be unidentified leakage. See Section A.7 of this report for more information.
5.3.1	Reactor Vessel Materials	Yes, base on mPower DSRS	

Section	Section Title	DSRS desired by NuScale?	Comments
5.3.2	Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	Yes, base on mPower DSRS	
5.3.3	Reactor Vessel Integrity	Yes, base on mPower DSRS	
5.4	Reactor Coolant System Component and Subsystem Design	Yes, base on mPower DSRS	The NuScale design does not have traditional reactor coolant loops. SRP requirements related to the reactor coolant loops may not apply to the NuScale design.
5.4.1.1	Pump Flywheel Integrity (PWR)	No, not applicable	The NuScale design does not contain reactor coolant pumps.
5.4.2.1	Steam Generator Materials	Yes, base on mPower DSRS	
5.4.2.2	Steam Generator Program	Yes, base on mPower DSRS	While the NuScale and mPower designs both have steam generators integral to the reactor vessel, the design of the steam generators is different.
5.4.6	Reactor Core Isolation Cooling System (BWR)	No, not applicable	Applies only to BWRs.
5.4.7	Residual Heat Removal (RHR) System	No, not applicable	The NuScale design does not contain a residual heat removal system. The NuScale design incorporates systems that fulfill design functions similar to those served by a typical RHR system. These systems will be described in other portions of the NuScale DCD. NuScale DCD Section 5.4.7 will provide references to these other portions of the DCD, as appropriate. Therefore a NuScale specific DSRS is not necessary.
5.4.8	Reactor Water Cleanup System (BWR)	No, not applicable	Applies only to BWRs.
5.4.11	Pressurizer Relief Tank	No, not applicable	No pressurizer relief tank in NuScale design.

Section	Section Title	DSRS desired by NuScale?	Comments
5.4.12	Reactor Coolant System High Point Vents	No, not applicable	Because of the integral reactor coolant system configuration, noncondensables accumulating in the pressurizer space will not interfere with core cooling during or after design basis accidents. As described in Table 3-1 and Section A.6, NuScale intends to seek an exemption from 50.46a and considers that 10 CFR 50.34(f)(2)(vi) is not technically relevant to the NuScale design. Therefore, a NuScale specific DSRS is not needed.
5.4.13	Isolation Condenser System (BWR)	No, not applicable	Applies only to BWRs.
5-1 BTP	Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	Yes, base on SRP	Unlike a traditional once-through steam generator, the NuScale helical coil steam generator does not contain an inventory of water at the bottom of the heat exchanger that could be sampled or blown down. Therefore, some of the sampling requirements dictated by the referenced EPRI PWR Secondary Water Chemistry Guidelines and NEI 97-06 guidelines will require a different approach for the NuScale design. Therefore, NuScale believes that the creation of a NuScale-specific DSRS section would be desirable.
5-2 BTP	Overpressure Protection of Pressurized-Water Reactors While Operating at Low Temperatures	No, SRP is sufficient	
5-3 BTP	Fracture Toughness Requirements	No, SRP is sufficient	
5-4 BTP	Design Requirements of the Residual Heat Removal System	Yes, base on SRP	As discussed in Section A.11, of this report, the NuScale design does not have a typical RHR system, but has other SSCs that fulfill similar design functions; the functional criteria of BTP 5-4 are applicable to these NuScale SSCs, but the other system criteria are not relevant due to unique NuScale design features.
SRP CHAPTER 6			
6.1.1	Engineered Safety Features Materials	Yes, base on mPower DSRS	

Section	Section Title	DSRS desired by NuScale?	Comments
6.1.2	Protective Coating Systems (Paints) - Organic Materials	Yes, base on mPower DSRS	
6.2.1	Containment Functional Design	Yes, base on mPower DSRS	
6.2.1.1.A	PWR Dry Containments, Including Sub-atmospheric Containments	Yes, base on mPower DSRS	As discussed in Section A.7 of this report, the NuScale containment vessel 1. fulfills its functions without reliance on restoring pressure to subatmospheric conditions following a postulated design basis accident. 2. does not have subcompartments housing high-energy piping.
6.2.1.1.B	Ice Condenser Containments	No, not applicable	No ice condenser containment in the NuScale design.
6.2.1.1.C	Pressure-Suppression Type BWR Containments	No, not applicable	Applies only to BWRs.
6.2.1.2	Subcompartment Analysis	No, SRP is sufficient	
6.2.1.3	Mass and Energy Release Analysis for Postulated LOCAs	Yes, base on mPower DSRS	Portions addressing core refill and reflood are not applicable, since postulated design basis accidents are not anticipated to result in core uncovery.
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	Yes, base on mPower DSRS	
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	No, not applicable	Core reflood, including consideration of the effects of containment pressure during core reflood, is not relevant to the evaluation of NuScale ECCS performance capability. Therefore, SRP section 6.2.1.5 is not applicable to the NuScale design and a NuScale-specific DSRS is not appropriate.

Section	Section Title	DSRS desired by NuScale?	Comments
6.2.2	Containment Heat Removal Systems	Yes, base on mPower DSRS	As discussed in Sections A.4 and A.7, of this report, the NuScale containment heat removal system design simply consists of the containment vessel and the heat transfer medium surrounding the vessel which, except for cases of pool boil-off after extended operation of the ECCS with no AC electrical power available, would be the reactor pool water in which the containment vessel is submerged. The NuScale containment heat removal system design does not use active systems such as fan coolers, spray systems, etc. Thus, the portions of this SRP section related to active systems and components are not applicable to the NuScale design. The portion of this SRP section specifying periodic functional and operability testing per GDC 40 is not applicable to the NuScale design. Additional details regarding GDC 40 applicability are provided in Table 3-1 of this report.
6.2.3	Secondary Containment Functional Design	No, not applicable	The NuScale containment vessel design does not include a secondary containment. The reactor building is not credited for fulfilling functions served by a secondary containment at a dual containment reactor plant. Rather, the reactor pool and the reactor building and its ventilation systems provide defense-in-depth – in addition to credited barriers including the containment vessel itself – to fission product release. Based on the above, this section is not relevant to the NuScale design and a NuScale specific DSRS section is unnecessary.
6.2.4	Containment Isolation System	Yes, base on mPower DSRS	As discussed in Section A.7 of this report, while the NuScale containment includes an evacuation system, it serves a different purpose than a purge system and does not provide a direct open path to the environs.
6.2.5	Combustible Gas Control in Containment	Yes, base on mPower DSRS	As discussed in Section A.7 of this report, the NuScale containment vessel design does not use combustible gas control systems, nor is an inerted atmosphere maintained that would be credited for combustible gas control pursuant to 10 CFR 50.44(c)(2). See the discussion on 10 CFR 50.44 in Table 3-1 of this report for more information.
6.2.6	Containment Leakage Testing	Yes, base on mPower DSRS	See discussion on GDC 52 in Table 3-1 of this report.

Section	Section Title	DSRS desired by NuScale?	Comments
6.2.7	Fracture Prevention of Containment Pressure Boundary	Yes	A new acceptance criterion is warranted for the review of the NuScale containment vessel due to its increased susceptibility to radiation embrittlement compared to the pressure boundary of a typical LWR containment structure. Specifically, due to its close proximity to the reactor core, the NuScale containment vessel is subject to radiation embrittlement (although to a lesser extent than the reactor vessel itself). Thus, for the NuScale containment vessel, fracture toughness requirements similar to those described for the reactor coolant pressure boundary in 10 CFR 50, Appendix G, and a material surveillance program similar to that described for the reactor vessel in 10 CFR 50, Appendix H, are anticipated to be implemented to ensure that the NuScale containment vessel satisfies the provisions of GDC 16 and GDC 51 over its 60 year design life.
6.3	Emergency Core Cooling System	Yes, base on SRP	As discussed in Section A.4 of this report, the NuScale ECCS is a passive, closed loop system, the design and operation of which is significantly different than a typical ECCS for which this guidance was developed.
6.4	Control Room Habitability System	Yes, base on mPower DSRS	As discussed in Section A.8 of this report, the NuScale control room habitability system neither relies on nor uses emergency filtration to protect operators during postulated accident conditions; rather, clean air is provided using compressed air tanks.
6.5.1	ESF Atmosphere Cleanup Systems	No, SRP is sufficient	As discussed in Section A.8 of this report, the NuScale design does not rely on ESF filter and atmosphere cleanup systems in response to a postulated accident.
6.5.2	Containment Spray as a Fission Product Cleanup System	No, not applicable	As discussed in Section A.7 of this report, the NuScale design does not include a containment spray system.
6.5.3	Fission Product Control Systems and Structures	No, SRP is sufficient	As discussed further in Sections A.7 and A.8 of this report, the NuScale containment vessel does not contain fission product clean-up systems, nor does it include or require pressure suppression systems (e.g., suppression pools or active containment heat removal systems such as containment spray) that serve a fission product removal/dose mitigation function.

Section	Section Title	DSRS desired by NuScale?	Comments
6.5.4	Ice Condenser as a Fission Product Cleanup System	No, not applicable	No ice condenser containment in the NuScale design.
6.5.5	Pressure Suppression Pool as a Fission Product Cleanup System	No, not applicable	Applies only to BWRs.
6.6	Inservice Inspection and Testing of Class 2 and 3 Components	Yes, base on mPower DSRS	
6.7	Main Steam Isolation Valve Leakage Control System (BWR)	No, not applicable	Applies only to BWRs.
BTP 6-1	pH For Emergency Coolant Water for Pressurized Water Reactors	Yes, base on SRP	
BTP 6-2	Minimum Containment Pressure Model for PWR ECCS Performance Evaluation	No, not applicable	Core reflood, including consideration of the effects of containment pressure during core reflood, is not relevant to the evaluation of NuScale ECCS performance capability. Therefore, this guidance is not applicable to the NuScale design and a NuScale-specific DSRS is not necessary.
BTP 6-3	Determination of Bypass Leakage Paths in Dual Containment Plants	No, not applicable	BTP 6-3 is applicable only to large LWRs that incorporate both a primary and secondary containment. The NuScale containment vessel design does not include a secondary containment. Therefore, this guidance is not applicable to the NuScale design and a NuScale-specific DSRS is not necessary.
BTP 6-4	Containment Purging During Normal Plant Operations	No, not applicable	While the NuScale containment vessel design includes an evacuation system, it serves a different purpose than a purge system and does not provide a direct open path to the environs. Therefore, this guidance is not applicable to the NuScale design and a NuScale-specific DSRS is not necessary.

Section	Section Title	DSRS desired by NuScale?	Comments
BTP 6-5	Currently the Responsibility of Reactor Systems Piping From the RWST (or BWST) and Containment Sump(s) to the Safety Injection Pumps	No, not applicable	As discussed in Section A.4 of this report, there are no safety injection pumps and no refueling water storage tank in the NuScale ECCS design.
mPower DSRS CHAPTER 7			
mPower DSRS 7.0	I & C Introduction and Overview of Review Process	Yes, base on mPower DSRS	See note on I&C related items in Section 3.2 of this report for more information.
mPower DSRS 7.0 - App A	Hazard Analysis	Yes, base on mPower DSRS	See note on I&C related items in Section 3.2 of this report for more information.
mPower DSRS 7.0 - App B	System Architecture	Yes, base on mPower DSRS	See note on I&C related items in Section 3.2 of this report for more information.
mPower DSRS 7.0 - App C	Simplicity	Yes, base on mPower DSRS	See note on I&C related items in Section 3.2 of this report for more information.
mPower DSRS 7.0 - App D	References	Yes, base on mPower DSRS	See note on I&C related items in Section 3.2 of this report for more information.
mPower DSRS 7.1	Fundamental Design Principles	Yes, base on mPower DSRS	See note on I&C related items in Section 3.2 of this report for more information.
mPower DSRS 7.2	System Characteristics	Yes, base on mPower DSRS	See note on I&C related items in Section 3.2 of this report for more information.
SRP CHAPTER 8			
8.1	Electric Power - Introduction	Yes, base on SRP	

Section	Section Title	DSRS desired by NuScale?	Comments
8.2	Offsite Power System	Yes, base on mPower DSRS	Consistent with the NRC response dated January 23, 2009, to an industry position on applicability of GDCs 2, 4, and 5 to the offsite power system, GDCs 2 and 4 are not applicable to the NuScale offsite power system design. Because GDCs 2, 4, and 5 only apply to SSCs that are important to safety, the basis for concluding that GDCs 2 and 4 are not applicable to the offsite power system also supports a conclusion that GDC 5 is not applicable. (See Dominion Energy, Inc. (Dominion), response to NRC Request for Additional Information (RAI) No. 08.02-42, provided as Enclosure 1 to Dominion Letter No. NA3-11-003RA, "SRP 08.02: Response to RAI Letter 54," dated May 12, 2011.) For the NuScale plant design, the offsite power system, interfaces between the offsite power system and the onsite AC power system, and the onsite AC power system itself are not safety-related. Thus, specific to the offsite power system, some of the guidance and codes and standards endorsed therein are not applicable to the NuScale design.
8.3.1	AC Power Systems (Onsite)	Yes, base on mPower DSRS	As discussed in Section A.3 of this report, the NuScale plant is designed to achieve safe shutdown and maintain core cooling and containment integrity, independent of AC or DC power sources, for an indefinite duration. Sufficient battery capacity is available to provide electrical power for other plant functions, including post-accident and pool monitoring, for a minimum of 72 hours following the onset of a design basis event. With this reduced reliance on AC power (compared to a typical LWR design), the NuScale onsite AC power system is not safety-related or important to safety, and GDCs 2, 4, 5, 17, and 18 do not apply to its design (although the design may still meet many of the requirements for commercial reasons).
8.3.2	DC Power Systems (Onsite)	Yes, base on mPower DSRS	Contrary to portions of this guidance, the NuScale design allows for the sharing of DC electrical power systems, but such sharing is specifically limited to the DC electrical power supply to monitoring functions. This design satisfies the intent of RG 1.81, since sufficient electrical power capacity is provided to preclude undesirable interactions and to assure the ability of SSCs to perform their safety functions.

Section	Section Title	DSRS desired by NuScale?	Comments
8.4	Station Blackout	Yes, base on mPower DSRS	The NuScale plant design meets the intent of this guidance largely in its passive design and associated reduced reliance on AC or DC power in coping with design basis events. Specifically, as discussed further in Section A.3 of this report, and consistent with NRC policy, this strong coping capability eliminates any significant safety benefit a typical large LWR gains by having an alternate AC power source (e.g., gas turbine generator) for station blackout (SBO).
App 8-A	General Agenda, Station Site Visits	No, SRP is sufficient	This appendix governs NRC visits to plant sites as part of licensing reviews during the operating or combined license stage. Therefore, this section will only be utilized for the COL application.
BTP 8-1	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	No, not applicable	As discussed further in Section A.4 of this report, the NuScale ECCS design does not use safety injection tanks (or equivalent) in response to a design basis accident. Design and operation of the NuScale ECCS also does not involve motor-operated valves.
BTP 8-2	Use of Diesel Generator Sets for Peaking	Yes, base on mPower DSRS	The NuScale plant design does not require or include safety related emergency diesel generators. With the NuScale plant's reduced reliance on AC power (compared to a typical LWR design), the concurrent loss of the preferred power source and the nonsafety-related diesel generators would have no significant adverse effect on plant safety. Notwithstanding, the NuScale backup diesel generators provide a defense-in-depth function. Therefore, this guidance will be considered applicable to the NuScale backup diesel generators, i.e., the backup diesel generators will not be used for peaking service.
BTP 8-3	Stability of Offsite Power Systems	Yes, base on mPower DSRS	The information governed by this guidance is site specific and will be addressed by the combined license applicant. Notwithstanding, consistent with SRP Section 1.0, Appendix A, RG 1.206, Regulatory Position C.III.4, and ESP/DC/COL-ISG-015, the NuScale design certification application will contain COL information items, as appropriate, that describe the information that is deferred to the license applicant referencing the certified design.
BTP 8-4	Application of the Single Failure Criterion to Manually Controlled Electrically Operated Valves	No, SRP is sufficient	

Section	Section Title	DSRS desired by NuScale?	Comments
BTP 8-5	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	No, SRP is sufficient	The draft mPower DSRS deleted this section and NuScale anticipates its deletion from the NuScale specific DSRS.
BTP 8-6	Adequacy of Station Electric Distribution System Voltages	Yes, base on mPower DSRS	As discussed in Section A.3 of this report, a loss of voltage or degraded voltage condition on the offsite power system would have no reasonable likelihood of adversely affecting the performance of plant safety functions. Based on the above, the under-voltage provisions contained in this guidance are not relevant to the NuScale plant design.
BTP 8-7	Criteria for Alarms and Indications Associated With Diesel-Generator Unit Bypassed and Inoperable Status	No, not applicable	As discussed in Section A.3 of this report, with its reduced reliance on AC power (compared to a typical LWR design), the NuScale plant does not require or include safety-related emergency diesel generators. Since the NuScale nonsafety related backup diesel generators are not relied upon for the performance of plant safety functions for at least 72 hours following the onset of a design basis event, the bypass or deliberately induced inoperable conditions addressed by this guidance would have no significant adverse impact on safety.
BTP 8-8	Onsite (Emergency Diesel Generators) and Offsite Power Sources Allowed Outage Time Extensions	No, not applicable	As discussed in Section A.3 of this report, with its reduced reliance on AC or DC power (compared to a typical LWR design), the operating restrictions (i.e., technical specifications allowed outage times) for inoperable AC power sources specified in this guidance are inappropriate to apply to the passive NuScale plant design to be described in the NuScale application for design certification.
SRP CHAPTER 9			
9.1.1	Criticality Safety of Fresh and Spent Fuel Storage and Handling	No, SRP is sufficient	
9.1.2	New and Spent Fuel Storage	Yes, base on SRP	

Section	Section Title	DSRS desired by NuScale?	Comments
9.1.3	Spent Fuel Pool Cooling & Cleanup System	Yes, base on mPower DSRS	For the NuScale design, the spent fuel pool is in association with the reactor coolant pool. This represents a design departure from typical LWRs. The NuScale spent fuel pool cooling system is classified as nonsafety-related, and is not designed to meet Quality Group C and Seismic Category I requirements as is specified by RG 1.26 and RG 1.29, respectively. This approach is consistent in intent with the acceptable alternative discussed in SRP Section 9.1.3, Section III.1.B, and RG 1.13. However, the acceptable alternative involves applying specific Quality Group C and Seismic Category I requirements to the spent fuel pool structure and liner, pool makeup and backup systems, and the building ventilation system to ensure adequate pool cooling, ventilation, and shielding are maintained. In the NuScale design, the building ventilation system is not relied upon to vent steam/moisture to the atmosphere to protect safety-related components from the effects of boiling in the spent fuel pool. Thus, contrary to the literal language of the acceptable alternative, Quality Group C and Seismic Category I requirements are not appropriate and will not be applied to the reactor building ventilation system.
9.1.4	Light Load Handling System (Related to Refueling)	No, SRP is sufficient	
9.1.5	Overhead Heavy Load Handling Systems	Yes, base on SRP	NuScale design specific module assembly equipment associated with its unique refueling process needs to be addressed within a NuScale-specific DSRS section. NuScale believes Section 9.1.5 could be a viable option for at least a portion of this expanded review scope, which is why a NuScale-specific DSRS Section 9.1.5 is desirable.
9.2.1	Station Service Water System	Yes, base on mPower DSRS	Refinements to the guidance found within the SRP contained within the mPower DSRS, as well as the lack of a safety-related service water system, lead to the desirability of a NuScale-specific DSRS section based upon the mPower DSRS. See NuScale comments on draft mPower DSRS Section 9.2.1 found within NP-LO-0913-4695 for more details.

Section	Section Title	DSRS desired by NuScale?	Comments
9.2.2	Reactor Auxiliary Cooling Water Systems	Yes, base on mPower DSRS	Unlike a typical reactor auxiliary cooling water system, the NuScale reactor component cooling water system does not serve a safety-related cooling (or heat transfer) function, and thus it is not considered to be a safety-related system. As discussed in Section A.7 of this report, the NuScale design does not use containment air coolers. The NuScale containment vessel also does not contain isolated water-filled piping sections, the overpressurization of which could jeopardize the performance of safety functions.
9.2.4	Potable and Sanitary Water Systems	Yes, base on mPower DSRS	
9.2.5	Ultimate Heat Sink	Yes, base on mPower DSRS	
9.2.6	Condensate Storage Facilities	Yes, base on mPower DSRS	Unlike the condensate system designs at typical large LWRs, no portion of the NuScale condensate system serves an essential safety function; i.e., the NuScale condensate system is not an essential source of cooling water to prevent or mitigate the consequences of accidents or to shut down the reactor and maintain it in a safe-shutdown condition. Accordingly, the NuScale condensate system is neither safety-related nor important to safety. The implementation of GDC 60, regarding control of radioactive releases, is applicable to the NuScale design. This section also specifies applicability of GDC 2. Consideration of GDC 2 is relevant only to the extent needed to ensure that operation or a failure of the system will not adversely impact important to safety SSCs. Based on the above, NuScale DSRS Section 9.2.6 would only contain review guidance related to those elements that are appropriate for review of a non-essential system. Those elements primarily are related to verifying: (1) that a potential water system failure will not adversely affect important-to-safety systems; and (2) adequate capabilities are provided to detect, control, and isolate system leakage, including radioactive leakage into and out of the system, and to prevent accidental releases to the environment.
9.3.1	Compressed Air System	No, SRP is sufficient	

Section	Section Title	DSRS desired by NuScale?	Comments
9.3.2	Process and Post-accident Sampling Systems	Yes, base on mPower DSRS	
9.3.3	Equipment and Floor Drainage System	Yes, base on mPower DSRS	
9.3.4	Chemical and Volume Control System (PWR) (Including Boron Recovery System)	Yes, base on SRP	The NuScale CVC system does not serve a safety-related reactor coolant makeup, emergency boration, or ECCS function. The safety related functions of the NuScale CVC system are limited to containment isolation, maintaining the RCPB, and isolation of the CVC system from the reactor coolant system (RCS). Performance of NuScale CVC system safety functions does not rely on AC power, and the CVC system is not relied upon to support SBO coping capability. The portion of this SRP section implementing GDC 33 is not applicable to the NuScale CVC system. As discussed in Section A.10 and Table 3-1 of this report, a coolant makeup system as contemplated by GDC 33 is not appropriate for the NuScale design. Rather, a NuScale-specific principal design criterion for the assurance of adequate reactor coolant inventory is warranted as an alternative to GDC 33.
9.3.5	Standby Liquid Control System (BWR)	No, not applicable	Applies only to BWRs.
9.3.6	Containment Evacuation System	Yes	
9.3.7	Containment Flooding System	Yes	See Section A.11.1 of this report for more information on the containment flooding system.
9.4.1	Control Room Area Ventilation System	Yes, base on SRP	As discussed in Section A.8 of this report, the NuScale control room habitability system neither relies on nor uses emergency filtration to protect operators during accident conditions. Rather, clean air is provided using compressed air tanks.

Section	Section Title	DSRS desired by NuScale?	Comments
9.4.2	Reactor Building and spent fuel pool area Ventilation system (new title)	Yes, base on SRP	Unlike a typical large LWR plant, the NuScale design does not rely on ESF ventilation systems to mitigate the consequences of a design basis accident. As discussed in Section A.8 of this report, nonsafety-related normal ventilation systems provide atmosphere cleanup capability, as necessary, that meets the design, testing, and maintenance guidelines specified in RG 1.140. These systems provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment during normal operations, anticipated operational occurrences, and postulated accident conditions. However, these systems are not required following an accident and, accordingly, receive no credit in the determination of the radiological consequences of an accident.
9.4.3	Radwaste Building Ventilation system (new title)	Yes, base on SRP	As discussed in Section A.8 of this report, the NuScale design does not rely on the radwaste building ventilation system as an ESF atmosphere cleanup system in response to a design basis accident.
9.4.4	Turbine Area Ventilation System	No, not applicable	The NuScale turbine building ventilation (TBV) system is not relied upon to control airborne radioactivity concentrations in the turbine building and/or gaseous effluents during normal operations (including anticipated operational occurrences) and after any accidents that result in a radioactive material release. Furthermore, there are no requirements for TBV system performance that are needed to preclude any adverse effect on safety-related functions during all conditions of plant operation.

Section	Section Title	DSRS desired by NuScale?	Comments
9.4.5	Engineered Safety Feature Ventilation System	No, not applicable	Unlike a typical large LWR plant, the NuScale design does not rely on ESF ventilation systems to mitigate the consequences of a design basis accident. As discussed in Section A.8 of this report, nonsafety-related normal ventilation systems provide atmosphere cleanup capability, as necessary, that meets the design, testing, and maintenance guidelines specified in RG 1.140. These systems provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment during normal operations, anticipated operational occurrences, and postulated accident conditions. However, these systems are not required following an accident and, accordingly, receive no credit in the determination of the radiological consequences of an accident. The NuScale design does not have an engineered safety feature ventilation system. Therefore, a NuScale-specific DSRS section is not necessary.
9.5.1.1	Fire Protection Program	No, SRP is sufficient	RG 1.189, Revision 2, is applicable except for aspects 1. directed toward a specific reactor design (e.g., BWR or non-LWR) or SSC conditions not relevant to the NuScale PWR design. 2. related to site-specific fire protection systems and equipment or programmatic and procedural activities that are the responsibility of the combined license applicant.
9.5.1.2	Risk-Informed, Performance-Based Fire Protection Program	No, SRP is sufficient	Development and implementation of a risk-informed, performance-based fire protection program would be the responsibility of combined license applicants that reference the NuScale certified design (upon approval by the NRC) and that elect to implement the provisions of 10 CFR 50.48(c) and it appears as though the SRP would be sufficient for those purposes. This section was deleted from the mPower DSRS.

Section	Section Title	DSRS desired by NuScale?	Comments
9.5.2	Communications Systems	Yes, base on mPower DSRS	Aspects of this SRP section related to the physical design of the power reactor and communication systems within the scope of the certified design are applicable to the NuScale application for design certification. Aspects related to site-specific design, procurement, fabrication, erection, construction, testing, and inspection of SSCs are the responsibility of the combined license applicant referencing the certified design. A portion of this guidance is applicable only to licensees subject to 10 CFR 73.45 and the general performance requirements of 10 CFR 73.20. Licensees referencing the NuScale certified design would not be subject to 10 CFR 73.20 and 10 CFR 73.45, but rather would be subject to the performance requirements of 10 CFR 73.55.
9.5.3	Lighting Systems	Yes, base on mPower DSRS	
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System	Yes, base on SRP	NuScale has backup diesel generators that are not safety related or important to safety and therefore much of the SRP (which was written for safety-related emergency diesel generators) is not applicable.
9.5.5	Emergency Diesel Engine Cooling Water System	Yes, base on SRP	NuScale has backup diesel generators that are not safety related or important to safety and therefore much of the SRP (which was written for safety-related emergency diesel generators) is not applicable.
9.5.6	Emergency Diesel Engine Starting System	Yes, base on SRP	NuScale has backup diesel generators that are not safety related or important to safety and therefore much of the SRP (which was written for safety-related emergency diesel generators) is not applicable.
9.5.7	Emergency Diesel Engine Lubrication System	Yes, base on SRP	NuScale has backup diesel generators that are not safety related or important to safety and therefore much of the SRP (which was written for safety-related emergency diesel generators) is not applicable.
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System	Yes, base on SRP	NuScale has backup diesel generators that are not safety related or important to safety and therefore much of the SRP (which was written for safety-related emergency diesel generators) is not applicable.

Section	Section Title	DSRS desired by NuScale?	Comments
SRP CHAPTER 10			
10.2	Turbine Generator	Yes, base on mPower DSRS	As discussed in Section A.5 of this report, compared to a large LWR, the NuScale plant design includes plant SSC designs and layouts that result in considerably reduced exposure of essential SSCs to potential turbine missiles. Specifically, in the NuScale design, essential SSCs are located within the reactor building such that the reactor building represents the engineered barrier for protection of these SSCs. Thus, consistent with RG 1.115, Revision 2, NuScale will satisfy GDC 4 by appropriate orientation and placement of the turbine generators, combined with proper design and use of missile barriers (i.e., the reactor building) to protect essential SSCs against potential turbine-generated missiles. The acceptability of this approach is reviewed under SRP Sections 3.5.1.3 and 3.5.3.
10.2.3	Turbine Rotor Integrity	Yes, base on mPower DSRS	See comment above for SRP Section 10.2.
10.3	Main Steam Supply System	Yes, base on mPower DSRS	
10.3.6	Steam and Feedwater System Materials	Yes, base on mPower DSRS	Refinements to the guidance found within the SRP, contained within the mPower DSRS, lead to the desirability of a NuScale specific DSRS section based upon the mPower DSRS. See NuScale comments on draft mPower DSRS section 10.3.6 found within NP-LO-0913-4695 for more details.
10.4.1	Main Condensers	Yes, base on mPower DSRS	
10.4.2	Main Condenser Evacuation System	Yes, base on mPower DSRS	
10.4.3	Turbine Gland Sealing System	Yes, base on mPower DSRS	
10.4.4	Turbine Bypass System	Yes, base on mPower DSRS	

Section	Section Title	DSRS desired by NuScale?	Comments
10.4.5	Circulating Water System	Yes, base on mPower DSRS	
10.4.6	Condensate Cleanup System	Yes, base on mPower DSRS	As discussed in Section A.9 of this report, secondary water chemistry requirements for the NuScale design may differ from those outlined in the specified EPRI report.
10.4.7	Condensate and Feedwater System	Yes, base on mPower DSRS	
10.4.8	Steam Generator Blowdown System (PWR)	No, not applicable	As described in Section A.9 of this report, the NuScale design does not involve the accumulation of secondary-side impurities in the steam generator to the extent that a typical PWR experiences; thus, the NuScale steam generator design does not include a blowdown system.
10.4.9	Auxiliary Feedwater System (PWR)	No, not applicable	The NuScale DHR system fulfills some functions similar to those performed by a typical AFW at a large PWR, and thus the intent of portions of this SRP section is generally applicable to the DHR system. However, as discussed in Section A.2 of this report, the DHR system is a passive, closed loop system, the design and operation of which is significantly different than a typical AFW system for which this guidance was developed. Therefore, it is more appropriate to create a new NuScale-specific DSRS section (10.4.12) for the DHR system.
10.4.10	Feedwater Treatment System	Yes	NuScale believes that, due to unique aspects of the NuScale design, the creation of a NuScale-specific DSRS section for the feedwater treatment system would enhance the quality of review. See Section A.9 of this report.
10.4.11	Auxiliary Boiler System	Yes	
10.4.12	Decay Heat Removal System	Yes, base on SRP	NuScale's unique decay heat removal system warrants a NuScale-specific DSRS section. Since it fulfills many of the same functions as an auxiliary feedwater system, it seems pertinent to base the NuScale-specific DSRS on SRP section 10.4.9

Section	Section Title	DSRS desired by NuScale?	Comments
BTP 10-1	Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants	No, not applicable	The NuScale design does not incorporate a traditional auxiliary feedwater system. The NuScale DHR system (which provides a substantially equivalent function as a traditional auxiliary feedwater system) is a passive system and does not use pumps.
BTP 10-2	Design Guidelines for Avoiding Water Hammers in Steam Generators	Yes	As described in Section A.9 of this report, the NuScale steam generator design minimizes potential water hammer issues without providing water through an externally mounted supply top discharge header as specified by this guidance.
SRP CHAPTER 11			
11.1	Source Terms	Yes, base on SRP	
11.2	Liquid Waste Management System	Yes, base on SRP	
11.3	Gaseous Waste Management System	Yes, base on SRP	
11.4	Solid Waste Management System	Yes, base on SRP	
11.5	Process Effluent Radiation Monitoring Instrumentation and Sampling Systems	Yes, base on SRP	
11.6	Guidance On Instrumentation and Control Design Features for Radiological Monitoring, and Area Radiation and Airborne Radioactivity Monitoring	Yes, base on mPower DSRS	
BTP 11-3	Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	No, SRP is sufficient	
BTP 11-5	Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure	No, SRP is sufficient	
BTP 11-6	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	No, SRP is sufficient	

Section	Section Title	DSRS desired by NuScale?	Comments
SRP CHAPTER 12			
12.1	Ensuring that Occupational Radiation Exposures Are As-Low-As-Reasonably Achievable (ALARA)	Yes, base on SRP	
12.2	Radiation Sources	Yes, base on SRP	Sub-tier ANSI/ANS 18.1 was withdrawn in 2009; NuScale will apply the guidance of NUREG 0017.
12.3-12.4	Radiation Protection Design Features	Yes, base on SRP	The aspects of this guidance related to ESF ventilation are not applicable to the NuScale design. As discussed in Section A.8 of this report, the NuScale design does not rely on ESF atmosphere cleanup systems to mitigate the consequences of a design-basis accident. The aspects of this guidance that pertain to site-specific operational/decommissioning activities are the responsibility of the combined license applicant. Aspects related to design features, facilities, functions, and equipment that are relevant to the NuScale standard plant design, are applicable to the NuScale application for design certification.
12.5	Operational Radiation Protection Program	Yes, base on SRP	This guidance governs operational programs, procedures, facilities, and organization that are site-specific and, accordingly, will be addressed by the combined license applicant referencing the NuScale certified design.
SRP CHAPTER 13			
13.1.1	Management and Technical Support Organization	No, SRP is sufficient	Use of this SRP section for applications for standard design certification is limited to development of COL information (action) items, as applicable and appropriate.
13.1.2-13.1.3	Operating Organization	No, SRP is sufficient	As discussed further in Section A.1 of this report, it is more appropriate that the operating organization for the NuScale power plant be based on features unique to the NuScale design rather than on staffing levels prescribed in 10 CFR 50.54(m)(2)(i). Thus, the content of SRP Sections 13.1.2 and 13.1.3 that implements 10 CFR 50.54(m)(2)(i) would not be applicable to the minimum operational staffing (appropriate for the NuScale plant) that will be described in the NuScale application for design certification.

Section	Section Title	DSRS desired by NuScale?	Comments
13.2.1	Reactor Operator Requalification Program; Reactor Operator Training	No, SRP is sufficient	Use of this SRP section for applications for standard design certification is limited to development of COL information (action) items, as applicable and appropriate.
13.2.2	Non-Licensed Plant Staff Training	No, SRP is sufficient	Use of this SRP section for applications for standard design certification is limited to development of COL information (action) items, as applicable and appropriate.
13.3	Emergency Planning	No, SRP is sufficient	
13.4	Operational Programs	No, SRP is sufficient	Use of this SRP section for applications for standard design certification is limited to development of COL information (action) items, as applicable and appropriate.
13.5.1.1	Administrative Procedures - General	No, SRP is sufficient	Use of this SRP section for applications for standard design certification is limited to development of COL information (action) items, as applicable and appropriate.
13.5.1.2	Administrative Procedures - Initial Test Program	No, SRP is sufficient	Use of this SRP section for applications for standard design certification is limited to development of inspections, test, analyses, and acceptance criteria/design acceptance criteria (ITAAC/DAC), interface requirements, and COL information (action) items, as applicable and appropriate.
13.5.2.1	Operating and Emergency Operating Procedures	No, SRP is sufficient	Use of this SRP section for applications for standard design certification is limited to development of ITAAC/DAC, interface requirements, and COL information (action) items, as applicable and appropriate.
13.5.2.2	Maintenance & Other Operating Procedures	No, SRP is sufficient	Use of this SRP section for applications for standard design certification is limited to development of ITAAC/DAC, interface requirements, and COL information (action) items, as applicable and appropriate.
13.6	Physical Security	No, SRP is sufficient	
13.6.1	Physical Security - Combined License	No, SRP is sufficient	Applies only to combined license applicants and applicants for and holders of operating licenses.

Section	Section Title	DSRS desired by NuScale?	Comments
13.6.2	Physical Security - Design Certification	No, SRP is sufficient	Applies only to applicants for design certification.
13.6.3	Physical Security - Early Site Permit	No, SRP is sufficient	Applies only to applicants for an early site permit.
13.6.4	Access Authorization	No, SRP is sufficient	
13.6.6	Cyber Security Program	No, SRP is sufficient	Applies only to combined license applicants and applicants for and holders of operating licenses.
13.7	Fitness for Duty- Introduction	No, SRP is sufficient	
13.7.1	Fitness for Duty - Operational Program	No, SRP is sufficient	
13.7.2	Fitness For Duty - Construction	No, SRP is sufficient	
SRP CHAPTER 14			
14.2	Initial Plant Test Program - Design Certification and New License Applicants	Yes, base on mPower DSRS	
14.2.1	Generic Guidelines for Extended Power Uprate Testing Programs	No, not applicable	Applies only to extended power uprate license amendment requests.
14.3	Inspections, Tests, Analyses, and Acceptance Criteria	No, SRP is sufficient	
14.3.2	Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria	Yes, base on mPower DSRS	
14.3.3	Piping Systems and Components - Inspections, Tests, Analyses, and Acceptance Criteria	No, SRP is sufficient	

Section	Section Title	DSRS desired by NuScale?	Comments
14.3.4	Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria	No, SRP is sufficient	
14.3.5	Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria	Yes, base on mPower DSRS	
14.3.6	Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria	Yes	This section will require significant NuScale-specific DSRS section development due to NuScale's lack of 1E power. As described in Section A.3 of this report, the strong coping capability of the NuScale design, with its reduced reliance on AC power, obviates the need for a normally available second offsite power circuit per GDC 17 or an alternate AC power source for station blackout.
14.3.7	Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria	Yes, base on mPower DSRS	
14.3.8	Radiation Protection Inspections, Tests, Analyses, and Acceptance Criteria	No, SRP is sufficient	
14.3.9	Human Factors Engineering - Inspections, Tests, Analyses, and Acceptance Criteria	No, SRP is sufficient	
14.3.10	Initial Test Program and D-RAP - Inspections, Tests, Analyses, and Acceptance Criteria	No, SRP is sufficient	Portions of the generic emergency planning inspections, test, analyses, and inspection criteria (EP-ITAAC) govern site-specific EP activities that are the responsibility of the combined license applicant/holder. The remaining portions represent an acceptable set of generic EP-ITAAC that NuScale may use to develop application-specific EP-ITAAC. However, in certain instances the generic ITAAC will need to be tailored to accommodate the NuScale design. For example, consistent with the staffing discussion provided in Section A.1 of this report, decisions regarding NuScale plant staffing levels for emergency response are more appropriately based on advanced design features and operational characteristics unique to the NuScale power plant rather than on the EP staffing provisions of NUREG-0654/FEMA-REP-1, Revision 1.
14.3.11	Containment Systems and Severe Accidents - Inspections, Tests, Analyses, and Acceptance Criteria	No, SRP is sufficient	

Section	Section Title	DSRS desired by NuScale?	Comments
14.3.12	Physical Security Hardware - Inspections, Tests, Analyses, and Acceptance Criteria	No, SRP is sufficient	
		SRP CHAPTER 15	
15.0	Introduction- Transient and Accident Analyses	Yes	
15.0.1	Radiological Consequence Analyses Using Alternative Source Terms	Yes, base on SRP	For the typical large LWR, the limiting dose consequence analysis corresponds to the design basis LOCA; however, for the NuScale design, core damage is not expected for a design basis LOCA event. Thus, the RG 1.183 guidance specified by this SRP section will only be partially applicable to the NuScale LOCA dose consequence analysis. NuScale intends to use the alternative source term non-LOCA or transient-specific guidance of RG 1.183 for Chapter 15 events that do not result in core damage.
15.0.2	Review of Transient and Accident Analysis Method	No, SRP is sufficient	
15.0.3	Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors	Yes	See comment above for SRP Section 15.0.1
15.1.1-15.1.4	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	Yes	Note: NuScale plans to use 15.1.4 for "inadvertent turbine bypass system opening" rather than "inadvertent opening of a steam generator relief or safety valve". Additionally, two more sections (15.1.XX) are needed for the NuScale specific increase in heat removal events "Inadvertent DHRS actuation" and "Loss of containment vacuum".
15.1.5	Steam System Piping Failures Inside and Outside of Containment	Yes	The NuScale design does not include reactor coolant pumps or multiple reactor coolant loops. As discussed in Section A.9 of this report, the “single loop” design of the NuScale reactor coolant system, combined with the intertwined helical coil steam generator tube configuration, eliminates the potential that a typical PWR design has for asymmetric core temperatures as a result of a steam line failure or isolation of a single steam generator.

Section	Section Title	DSRS desired by NuScale?	Comments
15.1.5.A	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	No, SRP is sufficient	
15.2.1-15.2.5	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	Yes	
15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries	Yes	
15.2.7	Loss of Normal Feedwater Flow	Yes	
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	Yes	The NuScale design does not include reactor coolant pumps. Portions of this SRP section directed towards AFW systems may be adapted for review of the NuScale DHR system, which is functionally similar to an AFW system found in large PWR designs.
15.3.1-15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	No, not applicable	Events involving loss of forced flow due to mechanical failure are not applicable since NuScale does not use reactor coolant pumps. The NuScale position is that potential decreases in flow are addressed as part of the plant response to other DBEs and a NuScale-specific DSRS section is not necessary.
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	No, not applicable	The NuScale design does not involve forced reactor coolant flow and the requisite pumps that would provide the motive force.
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	Yes	
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	Yes	
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	No, SRP is sufficient	NuScale believes that the SRP seemed sufficient. Determine why NRC is leaning towards DSRS.

Section	Section Title	DSRS desired by NuScale?	Comments
15.4.4-15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	Yes	The NuScale design does not use forced reactor coolant flow and does not have reactor coolant loops and pumps. Thus, the specific AOOs that result in an increase in core reactivity due to decreased moderator temperature, moderator boron concentration, or core void fraction addressed in this SRP section are not applicable to the NuScale design. Nevertheless, the potential for a postulated startup reactivity accident (e.g., initiated by abnormal startup sequence) has been identified as an event requiring consideration for the NuScale power plant design. Thus, this SRP section may be adapted for review of this postulated startup reactivity accident.
15.4.6	Inadvertent Decrease in Boron Concentration in the Reactor Coolant System	Yes	
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Yes	
15.4.8	Spectrum of Rod Ejection Accidents	No, SRP is sufficient	
15.4.8.A	Radiological Consequences of a Control Rod Ejection Accident (PWR)	Yes	This is one of the accidents being addressed by the NuScale source term. This, and other radiological accidents, will be addressed in section 15.0.3
15.4.9	Spectrum of Rod Drop Accidents (BWR)	No, not applicable	Applies only to BWRs.
15.4.9.A	Radiological Consequences of Control Rod Drop Accident (BWR)	No, not applicable	Applies only to BWRs.
15.5.1-15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	Yes	Inadvertent operation of the NuScale ECCS is not addressed because operation of the NuScale ECCS does not increase reactor coolant inventory.
15.6.1	Inadvertent Opening of a Pressurizer Relief Valve	Yes	The NuScale design does not use power-operated relief valves, which have the potential to open inadvertently. Rather, the NuScale design uses spring loaded ASME code safety relief valves, which do not have the power-operated relief valves' vulnerability to inadvertent operation.

Section	Section Title	DSRS desired by NuScale?	Comments
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	Yes	This is one of the accidents being addressed by the NuScale source term. This, and other radiological accidents, will be addressed in section 15.0.3
15.6.3	Radiological Consequences of Steam Generator Tube Failure (PWR)	Yes	This is one of the accidents being addressed by the NuScale source term. This, and other radiological accidents, will be addressed in section 15.0.3
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	No, not applicable	Applies only to BWRs.
15.6.5	Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Yes	The NuScale design does not include reactor coolant pumps. Aspects related to automatic AFW system initiation are applicable in intent, but as discussed in Section A.2 of this report, the NuScale design does not include an AFW system as would be found at a typical PWR plant. However, consistent with the intent of Acceptance Criterion II.3 of this SRP section, the NuScale LOCA analyses will account for automatic initiation of systems (e.g., DHR system and ECCS, as appropriate) relied upon for core cooling.
15.6.5.A	Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution	Yes	This is one of the accidents being addressed by the NuScale source term. This, and other radiological accidents, will be addressed in section 15.0.3
15.6.5.B	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment	Yes	This is one of the accidents being addressed by the NuScale source term. This, and other radiological accidents, will be addressed in section 15.0.3
15.6.5.D	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Main Steam Isolation Valve Leakage Control System (BWR)	No, not applicable	Applies only to BWRs.

Section	Section Title	DSRS desired by NuScale?	Comments
15.6.6	Inadvertent Operation of Emergency Core Cooling System (ECCS)	Yes	The nature of the NuScale emergency core cooling system (ECCS) is such that its operation does not increase reactor coolant inventory and is not considered a LOCA because adequate inventory in the passive containment vessel is retained to cover the core. Because of the unique nature of the NuScale ECCS, compared to a traditional LWR's ECCS, NuScale believes that a new NuScale-specific DSRS section would be desirable to enhance the effectiveness of the review of the NuScale DCA.
15.7.3	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	No, not applicable	The technical content of this SRP section has been moved to BTP 11-6
15.7.4	Radiological Consequences of Fuel Handling Accidents	Yes	This is one of the accidents being addressed by the NuScale source term. This, and other radiological accidents, will be addressed in section 15.0.3. As discussed in Section A.7 of this report, the NuScale design does not use a containment building. Rather, each NuScale power module has its own compact steel containment vessel. This containment vessel does not contain fuel storage and handling systems as contemplated by this SRP section. The provisions of this SRP section for containment isolation during fuel handling operations inside containment are not relevant to the NuScale containment vessel design. However, the intent of this guidance is appropriate to apply to the NuScale reactor building, where the operating reactor modules reside in the reactor pool and fuel handling operations are performed. During fuel handling operations, appropriate measures consistent with those described in this acceptance criterion will be established to ensure that the reactor building is or can be promptly isolated from the environment. Unlike a typical large LWR plant, the NuScale design does not rely on ESF ventilation systems to mitigate the consequences of a design basis accident. As discussed in Section A.8 of this report, nonsafety-related normal ventilation systems provide atmosphere cleanup capability, as necessary, that meets the design, testing, and maintenance guidelines specified in RG 1.140.
15.7.5	Spent Fuel Cask Drop Accidents	Yes	This is one of the accidents being addressed by the NuScale source term. This, and other radiological accidents, will be addressed in section 15.0.3

Section	Section Title	DSRS desired by NuScale?	Comments
15.8	Anticipated Transients Without Scram (ATWS)	Yes	This guidance provides an acceptable means for meeting 10 CFR 50.62(c)(1) and (c)(2). The design features required by 10 CFR 50.62(c)(1) were required for large LWRs based on the risk reduction it offered for ATWS events. These design features are either not included in the NuScale design or they are not required to reduce the risk from ATWS events as anticipated by the rule. The NuScale design does not include an auxiliary feedwater system. Therefore the portion 50.62(c)(1) that requires a diverse capability to initiate auxiliary feedwater is not applicable to the NuScale design. A trip of the turbine on a module during ATWS conditions is not required to reduce the risk for the NuScale design. The NuScale design relies on diversity within the module protection system (MPS) to reduce the risk associated with ATWS events. Internal diversity within the MPS is a simpler approach to addressing the 'diverse scram' elements of the ATWS rule. The NuScale design was not restrained by an existing reactor trip system design (without internal diversity) that resulted in an additional diverse scram system as required by 50.62(c)(2) to accomplish the same end. NuScale submitted a white paper that describes the underlying purpose of the rule, which is to reduce the risk from ATWS events ("NuScale Power Plant Design for ATWS and 10 CFR 50.62 Regulatory Compliance", NP-ER-0000-2196, September 18, 2013). The proposed treatment of the rule was described in a presentation to the NRC, "Design for ATWS and 10 CFR 50.62 Regulatory Compliance", PM-0114-5922-P, February 26, 2014. The NuScale design relies on diversity within the module protection system to reduce the risk associated with ATWS events. Because the NuScale reactor coolant system operates at a lower design pressure than a typical large PWR, the "22 MPa (3200 psig)" specified in this guidance is not applicable. The NuScale reactor coolant pressure limit value will be based on the NuScale reactor pressure vessel operating pressure.

Section	Section Title	DSRS desired by NuScale?	Comments
15.9	Stability	Yes	The mechanisms of flow instability are NuScale design specific and are not BWR flow stability phenomenon (core boiling/void reactivity). The intent of this guidance is applicable but the language is specific to BWR core designs. Specifically, there may be AOOs for the NuScale power plant design for which a density wave oscillation (Type I) flow instability would be conceivable under two-phase (subcooled boiling) natural circulation conditions. However, BWR-specific parameters and terminology are not applicable. In addition, the specific FABLE/BYPSS stability criteria were established for BWR core designs, and are not appropriate to apply to the NuScale core design.
SRP CHAPTER 16			
16.0	Technical Specifications	Yes, base on SRP	The improved standard technical specification guidance for LWRs specified in this SRP section is based on large LWRs with designs that differ significantly from the NuScale power plant design. Thus, it is anticipated that there will be a significant number of substantive (i.e., technical rather than editorial) differences between the NuScale proposed technical specifications and those presented in the improved standard technical specification guidance.
16.1	Risk-Informed Decision Making: Technical Specifications	Yes, base on SRP	The modular design of the NuScale power plant, wherein operating reactors are located in the same pool of water during the refueling of other modules, makes the SRP not entirely possible to follow. NuScale specific technical specifications for multi-module limiting conditions for operation are anticipated, which leads to the desire for a NuScale-specific DSRS section. This guidance is directed explicitly towards existing licensees seeking NRC approval of changes to their plant-specific licensing basis, and sub-tier NUREG/CR 6595 is based on large LWR designs that differ significantly from the NuScale power plant design. The extent of these differences is such that NUREG/CR 6595 is inappropriate to apply to the NuScale design.

Section	Section Title	DSRS desired by NuScale?	Comments
SRP CHAPTER 17			
17.1	Quality Assurance During the Design and Construction Phases	No, not applicable	Applies only to existing NRC approved quality assurance (QA) programs that are based on ANSI N45.2 and its daughter standards. The NuScale QAPD, to be included in Chapter 17 of the DCD, will be based on NQA-1-2008 and the NQA-1a-2009 addenda, as endorsed in RG 1.28, Rev. 4. Since the issuance of SRP Section 17.1, the NRC has issued SRP Section 17.5 (based on NQA 1) for the review of QAPDs for new reactor applicants, including applicants for design certification. Accordingly, SRP Section 17.5 is the appropriate guidance to be applied to the NuScale QAPD.
17.2	Quality Assurance During the Operations Phase	No, not applicable	Applies only to existing NRC approved operational QA programs that are based on ANSI N45.2 and its daughter standards. The operational QAPD is site-specific and for new reactors will be addressed by the combined license applicant using the guidance of SRP Section 17.5, which allows COL applicants to reference a QAPD approved by the NRC under SRP Section 17.2.
17.3	Quality Assurance Program Description	No, not applicable	Applies only to existing NRC approved QA programs. Since the issuance of this SRP section, the NRC has issued SRP Section 17.5 for the review of QAPDs for new reactor applicants – including applicants for design certification – under 10 CFR 52. Accordingly, SRP Section 17.5 is the appropriate guidance to be applied to the NuScale QAPD.
17.4	Reliability Assurance Program (RAP)	No, SRP is sufficient	Note that this section now incorporates DC/COL-ISG-18.

Section	Section Title	DSRS desired by NuScale?	Comments
17.5	Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	No, SRP is sufficient	Certain acceptance criteria of this SRP section are related to a power plant's construction or operational phases and thus are not applicable to the NuScale QA program to be applied during the design certification phase. These criteria will be addressed within the construction and operational QA programs, as appropriate, and developed and maintained by the combined license applicant referencing the certified design. This SRP section references Revision 3 of RG 1.28, which endorsed portions of ANSI/ASME NQA 1 1983. RG 1.28, Revision 4, has subsequently been issued that endorses (with additions and modifications) ANSI/ASME NQA 1 2008 with NQA 1a 2009 addenda. NuScale intends to apply RG 1.28, Revision 4, and its endorsed standards, to the NuScale QA program to be applied during the design certification phase.
17.6	Maintenance Rule	No, SRP is sufficient	This program is a site-specific operational program that will be addressed by the combined license applicant.
SRP CHAPTER 18			
18.0	Human Factors Engineering	No, SRP is sufficient	
Appendix 18-A	Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses	No, SRP is sufficient	
SRP CHAPTER 19			
19.0	Probabilistic Risk Assessment and Severe Accident Evaluation	No, SRP is sufficient	
19.1	Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	No, SRP is sufficient	
19.2	Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance	No, not applicable	Applies only to existing reactor licensees' requests for license amendments under 10 CFR 50.90 and exemptions under 10 CFR 50.11.

Section	Section Title	DSRS desired by NuScale?	Comments
19.3	Regulatory Treatment of Non-Safety Systems for Passive Advanced Light Water Reactors	No, SRP is sufficient	
19.4	Strategies and Guidance to Address Loss of Large Areas of the Plant Due to Explosions and Fires	No, SRP is sufficient	
19.5	Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts	No, SRP is sufficient	

3.3 Interim Staff Guidance

Interim staff guidance (ISG) with potential relevance to applicants for and holders of a design certification (i.e., ISGs designated as DC/COL-ISG, DI&C-ISG, and NSIR/DPR-ISG)[4] were also reviewed for applicability with respect to the NuScale application for design certification. The summarized results of the NuScale assessment of pertinent ISGs are shown in Table 3-3 and detailed information is available for review in the NuScale electronic reading room.

Note: Documentation of the NuScale gap analysis review of the NRC regulatory guides is available for review in the NuScale electronic reading room. The results of this assessment will be used in the development of the table of conformance with NRC regulatory guides that is specified in SRP Chapter 1, Section 1, Areas of Review, Item 9, to be included in the NuScale application for design certification, updated to reflect regulatory guides in effect six months before the submittal date of the NuScale application.

[4] References to “SRP” in the remainder of this section shall include NUREG-0800, associated branch technical positions, and the aforementioned ISGs.

Table 3-3 Applicability of interim staff guidance to the NuScale DCA effort:

Interim Staff Guidance	Title/Subject	Incorporate into NuScale Design-Specific Review Standard		Comments
		DC Application	COL Application	
INTERIM STAFF GUIDANCE DIRECTED TOWARDS DESIGN CERTIFICATIONS				
DC/COL-ISG-01	Seismic Issues Associated with High Frequency Ground Motion in Design Certification and Combined License Applications	Yes — Use As-Is	Yes — Use As-Is	–
DC/COL-ISG-03	Probabilistic Risk Assessment Information to Support Design Certification and Combined License Applications	Yes — Use With Modification	Yes — Use As-Is	The current revision of RG 1.200 endorses probabilistic risk assessment (PRA) standards that are not practicable for a design certification applicant to fully implement since doing so would require site-specific seismic hazard information not available at the design stage. As an alternative approach to seismic PRA, NuScale intends to use the PRA-based seismic margin analysis methodology described in DC/COL-ISG-20 to demonstrate acceptably low seismic risk.
DC/COL-ISG-05	Use of the GALE86 Code for Calculation of Routine Radioactive Releases in Gaseous and Liquid Effluents from Boiling-Water Reactors and Pressurized-Water Reactors to Support Design Certification and Combined License Applications	Yes — Use As-Is	Yes — Use As-Is	–
DC/COL-ISG-06	Evaluation and Acceptance Criteria for 10 CFR 20.1406 to Support Design Certification and Combined License Applications	Yes — Use As-Is	Yes — Use As-Is	–

Interim Staff Guidance	Title/Subject	Incorporate into NuScale Design-Specific Review Standard		Comments
		DC Application	COL Application	
DC/COL-ISG-07	Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures	Yes — Use As-Is	Yes — Use As-Is	–
DC/COL-ISG-08	Necessary Content of Plant-Specific Technical Specifications When a Combined License Is Issued	Yes — Use As-Is	Yes — Use As-Is	–
DC/COL-ISG-10	Review of Evaluation to Address Adverse Flow Effects in Equipment Other Than Reactor Internals	Yes — Use As-Is	Yes — Use As-Is	–
DC/COL-ISG-11	Finalizing Licensing-Basis Information	Yes — Use As-Is	Yes — Use As-Is	–
DC/COL-ISG-13	NUREG-0800 Standard Review Plan Section 11.2 and Branch Technical Position 11-6 Assessing the Consequences of an Accidental Release of Radioactive Materials from Liquid Waste Tanks for Combined License Applications Submitted under 10 CFR Part 52 (Issued for Comment)	Yes — Use As-Is	Yes — Use As-Is	–
DC/COL-ISG-14	Standard Review Plan Sections 2.4.12 and 2.4.13 Assessing Groundwater Flow and Transport of Accidental Radionuclide Releases (Issued for Comment)	No	Yes — Use As-Is	As a supplement to SRP Sections 2.4.12 and 2.4.13, this guidance governs site-specific hydrogeological data, site characteristics, and radiological analysis aspects that are the responsibility of the combined license applicant referencing the certified design.

Interim Staff Guidance	Title/Subject	Incorporate into NuScale Design-Specific Review Standard		Comments
		DC Application	COL Application	
ESP/DC/COL-ISG-15	Post-Combined License Commitments	Yes — Use As-Is	Yes — Use As-Is	–
DC/COL-ISG-16	Compliance with 10 CFR 50.54(hh)(2) and 10 CFR 52.80(d)	No	Yes — Use As-Is	Since this ISG was issued as need to know, official use only, and security-related, the details are characterized as Sensitive Unclassified Non-Safeguards Information and are not available for the public or for this gap analysis. From a review of the associated issuance notice dated June 9, 2010, it appears that this ISG governs site-specific information that is applicable to combined license applicants and is not within the scope of the NuScale application for design certification.
DC/COL-ISG-17	Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses	No	Yes — Use As-Is	This ISG governs site-specific information that is applicable only to combined license applicants.
DC/COL-ISG-18	Standard Review Plan, Section 17.4, “Reliability Assurance Program”	No	No	Note that DC/COL-ISG-18 was incorporated into SRP Section 17.4 Revision 1 and is no longer applicable.
DC/COL-ISG-19	Review of Evaluation to Address Gas Accumulation Issues in Safety Related Systems and Systems Important to Safety	No	No	Applicable only to reactor designs for which operation of emergency core cooling, residual heat removal, and containment spray systems relies on pumps (i.e., forced circulation); the NuScale ECCS and DHR system (the NuScale design does not include a containment spray system) operate via natural circulation and do not use pumps.
DC/COL-ISG-20	Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors	Yes — Use As-Is	Yes — Use As-Is	–

Interim Staff Guidance	Title/Subject	Incorporate into NuScale Design-Specific Review Standard		Comments
		DC Application	COL Application	
DC/COL-ISG-21	Review of Nuclear Power Plant Designs using a Gas Turbine Driven Standby Emergency Alternating Current Power System	No	No	This ISG is applicable only when a gas turbine-driven standby emergency AC power system is used (in lieu of emergency diesel generators) to supply power to safety-related or important-to-safety equipment for operational events and postulated accidents. As discussed in Section A.3 of this report, the NuScale design uses onsite backup diesel generators as opposed to gas turbine generators. Regardless of the type of standby AC generation used in the NuScale design, the onsite standby AC generation source and the onsite AC distribution system it serves are not safety-related, nor are they relied upon to fulfill safety functions.
COL-ISG-22	Impact of Construction (under a Combined License) of New Nuclear Power Plant Units on Operating Units at Multi-Unit Sites	No	Yes — Use As-Is	Applies only to combined license applicants/holders.
COL-ISG-25	Changes during Construction Under 10 CFR Part 52	No	Yes — Use As-Is	Applies only to combined license holders.
INTERIM STAFF GUIDANCE ASSOCIATED WITH EMERGENCY PLANNING				
NSIR/DPR-ISG-01	Emergency Planning for Nuclear Power Plants	No	Yes — Use As-Is	This guidance governs site-specific programmatic and design aspects of emergency planning that will be the responsibility of the combined license applicant referencing the NuScale design.

Interim Staff Guidance	Title/Subject	Incorporate into NuScale Design-Specific Review Standard		Comments
		DC Application	COL Application	
INTERIM STAFF GUIDANCE ASSOCIATED WITH DIGITAL INSTRUMENTATION AND CONTROL				
DI&C-ISG-01	Cyber Security	Yes — Use With Modification	Yes — Use With Modification	Since the issuance of DI&C-ISG-01, the NRC has issued revised guidance to that endorsed or referenced in this ISG. To the extent that NuScale addresses certain cyber security provisions of 10 CFR 73.54 through the use of specific design features in its standard plant design, the guidance of RG 1.152, Rev. 3, and RG 5.71 are considered applicable to the NuScale application for design certification. However, the portions of RG 5.71 that govern site-specific operational and programmatic activities (e.g., development and implementation of operational cyber security plans) are applicable only to the combined license applicant. Since the issuance of DI&C-ISG-01, the NEI has issued NEI 08-09, "Cyber Security Plan for Nuclear Power Reactors," intended to replace the industry guidance provided in NEI 04-04. SECY-10-0153 states that "…industry representatives have indicated that they will revise the cyber security plan template and guidance contained in NEI 08-09…, Revision 6, and request NRC endorsement. As part of the update to RG 5.71, the NRC will review the updates to NEI 08-09, Revision 6, and endorse it if it adequately incorporates the Commission's interpretations provided in the SRM." Subject to availability, NuScale intends to use the endorsed revision of NEI 08-09 (rather than NEI 04-04) and RG 5.71 revised to reflect the balance-of-plant issue as discussed above.
DI&C-ISG-02	Diversity and Defense-in-Depth Issues	Yes — Use With Modification	Yes — Use With Modification	Reference to "computer system qualification testing" applies to hardware that is not planned to be used in the NuScale protection system design. NuScale intends to apply BTP 7-19, Revision 6, instead of BTP 7-19, Revision 5.
DI&C-ISG-03	Review of New Reactor Digital Instrumentation and Control Probabilistic Risk Assessments	Yes — Use With Modification	Yes — Use With Modification	This ISG refers to data analysis standards of ASME/ANS Std. RA-S-2002 and 2003 addenda, Subsection 4.5.6. NuScale intends to use the current 2009 version of this standard. The substantive content of the data analysis standards contained in Subsection 4.5.6 of the 2002 version (with 2003 addenda) are contained in Subsection 2-2.6 of ASME/ANS Std. RA-S-2009.

Interim Staff Guidance	Title/Subject	Incorporate into NuScale Design-Specific Review Standard		Comments
		DC Application	COL Application	
DI&C-ISG-04	Highly-Integrated Control Rooms – Communications Issues	Yes — Use With Modification	Yes — Use With Modification	This ISG refers to Revision 2 of RG 1.152. NuScale intends to apply the current Revision 3 (unless superseded by a newer revision) of RG 1.152 to its application for design certification.
DI&C-ISG-05	Highly-Integrated Control Rooms – Human Factors Issues	Yes — Use With Modification	Yes — Use With Modification	This ISG refers to the 1998 version of ANSI/ANS 3.5. The current Revision 4 of RG 1.149 endorses, with clarifications, the 2009 edition of ANSI/ANS 3.5. NuScale intends to apply ANSI/ANS 3.5-2009 to its application for design certification.
DI&C-ISG-06	Licensing Process	No	No	This guidance is directed towards review of requests for licensing basis changes from existing licensees to implement digital IE upgrades.
DI&C-ISG-07	Digital Instrumentation and Control Systems in Safety Applications at Fuel Cycle Facilities	No	No	This guidance is directed towards review of proposed measures for protecting digital I&C equipment used as items relied on for safety at fuel cycle facilities from unintentional digital events.
JLD-ISG-12-01	Interim Staff Guidance for Compliance with Order EA-12-049 Concerning Mitigation Strategies	No	Yes	This ISG is applicable to holders of, and applicants for, operating licenses, construction permits, and combined licenses. Therefore, it is not applicable to the NuScale DCA; however, NuScale is using the guidance found within this ISG and NEI 12-06 to assist in its design efforts.
JLD-ISG-12-02	Interim Staff Guidance for Compliance with Order EA-12-050 Concerning Reliable Hardened Vents	No	No	This ISG is applicable to BWR licensees with Mark I and Mark II containments. The NuScale containment heat removal capabilities are passive and the hardened containment vent systems addressed by this ISG are not applicable to the NuScale design.
JLD-ISG-12-03	Interim Staff Guidance for Compliance with Order EA-12-051 Concerning Spent Fuel Pool Instrumentation	No	Yes	This ISG is applicable to holders of, and applicants for, operating licenses, construction permits, and combined licenses. Pool monitoring instrumentation that is capable of monitoring and providing indication of beyond-design-basis events (i.e., instrumentation that can monitor a wide range of spent fuel pool levels) is part of the NuScale design.
JLD-ISG-12-04	Interim Staff Guidance on Performing a Seismic Margin Assessment in Response to the March 2012 Request for Information Letter	No	No	This ISG is only intended for the purpose of response to the March 2012 50.54(f) request for information letter. DC/COL-ISG-020 remains the NRC's current guidance for application of an SMA to new reactors licensing. Therefore, this ISG is not applicable to the NuScale DCA effort.

Interim Staff Guidance	Title/Subject	Incorporate into NuScale Design-Specific Review Standard		Comments
		DC Application	COL Application	
JLD-ISG-12-05	Interim Staff Guidance on Performance of an Integrated Assessment for Flooding	No	No	This ISG is only intended for the purpose of response to the March 2012 50.54(f) request for information letter and is therefore not applicable to the NuScale DCA effort.
JLD-ISG-12-06	Interim Staff Guidance for Performing a Tsunami, Surge, or Seiche Hazard Assessment	No	No	This ISG is only intended for the purpose of response to the March 2012 50.54(f) request for information letter and is therefore not applicable to the NuScale DCA effort.
JLD-ISG-13-01	Interim Staff Guidance for Estimating Flooding Hazards due to Dam Failure	No	No	This ISG is only intended for the purpose of response to the March 2012 50.54(f) request for information letter and is therefore not applicable to the NuScale DCA effort.

4.0 Conclusion

The results summarized in this report provide a strong foundation to facilitate deliberations between NuScale and the NRC during pre-application activities. Specifically, as part of further pre-application activities, NuScale will continue to seek to reach consensus with the NRC on the applicability of the regulatory framework as assessed in this gap analysis and the path forward towards addressing “regulatory gaps” identified in Section 3.0 of this report.

Any changes to the applicability determinations and gap dispositions resulting from these anticipated deliberations with the NRC will be incorporated, as appropriate, into a future revision to this report.

NuScale believes that the regulatory gap analysis results represent a useful tool for development of a design-specific review standard to be used by the NRC in its review of the NuScale application for design certification. NuScale remains committed to assisting the NRC as necessary and appropriate to facilitate the creation of the NuScale-specific DSRS sections.

The results of the NuScale regulatory gap analysis reflect existing knowledge based on the current stage of engineering design and represent NuScale’s best-effort assessment of applicability and relevance of current LWR-based requirements and guidance, in literal language or intent, to the NuScale power plant design. As the ongoing engineering design effort progresses, the relevance of portions of the requirements and guidance considered in this gap analysis may warrant reconsideration. Accordingly, the NuScale regulatory gap analysis results summarized in this report are not intended to preclude NuScale from proposing in its application for design certification certain design features, analytical techniques, and procedural measures that would be different than those given in the design-specific review standard to be developed for the NuScale design based on the results of this gap analysis. Any such differences would be evaluated in accordance with 10 CFR 52.47(a)(9) to confirm that any proposed alternative provides an acceptable method of complying with the underlying NRC requirements.

5.0 References

5.1. *U.S. Code of Federal Regulations*, Parts 1 through 199, Chapter 1, Title 10, "Energy."

5.2. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800.

5.3. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition – Introduction Part 2: Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Small Modular Reactor Edition," (Revision 0), NUREG-0800, (ADAMS Accession No. ML13207A315).

5.4. U.S. Nuclear Regulatory Commission, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," SECY-11-0024, February 18, 2011 (ADAMS Accession No. ML110110688).

5.5. U.S. Nuclear Regulatory Commission, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Regulatory Guide 1.206, June 2007 (ADAMS Accession Nos. ML070630003 through ML070630044, and ML071450387).

5.6. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Construction of the Clinch River Breeder Reactor Plant," Volume 1, Main Report, NUREG-0968, March1983 (ADAMS Accession No. 8303300448).

5.7. U.S. Nuclear Regulatory Commission, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid Metal Reactor," NUREG-1368, February 1994 (ADAMS Accession No. ML063410561).

5.8. U.S. Nuclear Regulatory Commission, "A Study of Control Room Staffing Levels for Advanced Reactors," NUREG/IA-0137, November 2000 (ADAMS Accession No. ML003774060).

5.9. NuScale Power, LLC, "NuScale Design Overview," NP-ER-0000-1198, Rev. 2.

5.10. U.S. Nuclear Regulatory Commission, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants," Branch Technical Position 10-1, (ADAMS Accession No. ML070850410).

5.11. U.S. Nuclear Regulatory Commission, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," SECY-90-016, January 12, 1990, Approved in Staff Requirements Memorandum, dated June 26, 1990 (ADAMS Accession No. ML003707885).

5.12. U.S. Nuclear Regulatory Commission, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," SECY-94-084, March 28, 1994, Approved in Staff Requirements Memorandum, dated June 30, 1994 (ADAMS Accession No. ML003708098).

5.13. U.S. Nuclear Regulatory Commission, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," SECY-95-132, Approved in Staff Requirements Memorandum, dated June 28, 1995 (ADAMS Accession No. ML003708019).

5.14. U.S. Nuclear Regulatory Commission, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.52, Revision 3, June 2001 (ADAMS Accession No. ML011710176).

5.15. U.S. Nuclear Regulatory Commission, "Protection Against Turbine Missiles," Regulatory Guide 1.115, Revision 2, January 2012 (ADAMS Accession No. ML101650675).

5.16. U.S. Nuclear Regulatory Commission, “Combustible Gas Control in Containment,” Federal Register, Final Rule, Volume 68, Pages 54123 - 54143 (68 FR 54123 – 54143), September 16, 2003.

5.17. U.S. Nuclear Regulatory Commission, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Regulatory Guide 1.140, Revision 2, June 2001 (ADAMS Accession No. ML011710150).

5.18. U.S. Nuclear Regulatory Commission, “Design Guidelines for Avoiding Water Hammers in Steam Generators,” Branch Technical Position 10-2, Revision 4, March 2007 (ADAMS Accession No. ML070850324).

5.19. Electric Power Research Institute, “Pressurized Water Reactor Secondary Water Chemistry Guidelines,” (EPRI# 1008224), EPRI, Palo Alto, CA, 2004

5.20. Nuclear Energy Institute, “Steam Generator Program Guidelines,” NEI 97-06, Revision 3, January 2011 (ADAMS Accession No. ML111310708).

5.21. U.S. Nuclear Regulatory Commission, “Operator Staffing for Small or Multi-Module Nuclear Power Plant Facilities,” SECY-11-0098, July 22, 2011 (ADAMS Accession No. ML111870574).

5.22. U.S. Nuclear Regulatory Commission, “Human Factors Engineering Program Review Model,” NUREG-0711, Revision 2, February 2004 (ADAMS Accession No. ML040770540).

5.23. U.S. Nuclear Regulatory Commission, “Guidance for Assessing Exemption Requests from the Nuclear Power Plant Licensed Operator Staffing Requirements Specified in 10 CFR 50.54(m),” NUREG-1791, July 2005 (ADAMS Accession No. ML052080125).

Appendix A. Summary of Significant NuScale Design Functions and Characteristics

This appendix provides additional information regarding some of the more significant NuScale power plant design features that led to certain regulations and regulatory guidance being identified as “not applicable”, in whole or in part, to the NuScale application for design certification.[5] Such instances may warrant modifications to the regulatory framework that will be applied to the NuScale application, as discussed in Table 3-1 and Table 3-2 of this report.

A.0 Fail to Safest Configuration on Loss of Power

The NuScale design can shut down and self cool indefinitely with no operator action, no AC or DC power, and no additional water. The passive safety features of the NuScale design are valve actuated. These safety valves, along with containment isolation valves, align to a safe configuration on loss of all plant power, which leads to the shutdown of all the reactors and leaves all modules rejecting decay heat to the reactor pool. The reactor pool has sufficient water inventory to maintain the reactors in safe shutdown states until the decay heat is sufficiently small to allow air cooling to be adequate to maintain the reactors in a safe shutdown state indefinitely.

Operators and backup battery power are only relied upon for post-accident monitoring functions and are not credited for any safety functions or actuations during the first 72 hours of any chapter 15 safety analysis related to design basis accidents. This should lead to significant simplifications in the design, analysis, and regulatory reviews of the control room habitability system, electrical power system, and several other systems. Of course, for prudence and commercial reasons, the control room habitability systems, electrical power systems, and other systems are all still rigorously designed to assure high degrees of availability for defense-in-depth considerations; however, these capabilities need not be design basis requirements.

A.1 Licensed Operator Staffing

The current regulations contained in 10 CFR 50.54(m)(2)(i) and (iii) governing minimum licensed operator staffing are prescriptive and are based on the design and operation of existing large light-water reactors (LWRs). As discussed in NUREG-1791 (Reference 5.23), the current regulations incorporate implicit assumptions and specify requirements that are not appropriate to apply to the NuScale advanced small modular reactor design. Examples include but are not limited to the following:

1. There is a maximum of three units and three control rooms.
2. There are no more than two units per control room.
3. There is at least one senior operator on site at all times and at least one in the control room for each unit in operation.

As discussed in the NuScale Design Overview (Reference 5.9), there are significant differences between the modular design of a NuScale power plant and that of a typical large LWR. These differences (as well as the unique features of the NuScale design described above in Section A.0) are such that many of the assumptions underlying the minimum staffing requirements of 10 CFR 50.54(m)(2)(i) and (iii) are not valid for the NuScale plant. For example, the NuScale design allows for up to 12 units operating at a single plant. It is unclear whether the 10 CFR

[5] The results of the NuScale regulatory gap analysis effort to determine relevance of the current regulations and guidance is available for NRC review in the NuScale electronic reading room.

50.54(m)(2)(i) assumption that a licensee at a site would have no more than three units and three control rooms would allow for a 12-unit NuScale facility. Even absent this limitation, applying the 10 CFR 50.54(m)(2)(i) assumption for no more than two units per control room would require for a 12-unit NuScale power plant no less than 6 separate control rooms, with the requisite minimum staffing per control room specified in 10 CFR 50.54(m)(2)(i). With consideration for certain design features specific to the NuScale design, applying these requirements to the NuScale plant is neither appropriate nor necessary to achieve the underlying purpose of the rule.

Specifically, the NuScale power plant incorporates advanced, simplified design features resulting in roles, responsibilities, composition, and size of plant operating crews that are different than those prescribed by 10 CFR 50.54(m)(2)(i) and (iii). These design features include increased use of automation, state-of-the-art instrumentation and controls, passive safety features, function allocation, displays that better integrate control room information, and plant-specific operating characteristics that support the operation of multiple modular reactors from the same control room. Consistent with research conducted by the NRC at the Halden Reactor Project (NUREG/IA-0137, Reference 5.8) NuScale believes that decisions regarding operator staffing levels, including the number, composition, and qualifications of licensed personnel, for the NuScale power plant are more appropriately based on these advanced design features and on human factors engineering analysis using the methodology provided in NUREG-0711 (Reference 5.22) and NUREG-1791 (Reference 5.23) rather than on staffing levels prescribed in 10 CFR 50.54(m)(2)(i) and (iii).

As part of pre-application activities, NuScale will seek concurrence with NRC regarding the appropriate process for submittal, review, and approval of control room staffing and related design issues. Important factors to consider in developing the path forward are the regulatory framework, the NuScale plan and schedule for relevant HFE analyses, and obtaining sufficient finality to support future license applications. The technical basis for the staffing plan will be based on HFE analysis of the NuScale plant using the guidance provided in NUREG-0711 (Reference 5.22) and NUREG-1791 (Reference 5.23).

A.2 NuScale Decay Heat Removal (DHR) System

Existing NRC regulations and regulatory guidance specify requirements for design of the AFW system. These regulations and guidance were established based on large LWR designs. Due to features unique to the NuScale design, in some instances the existing LWR-based regulatory framework is not appropriate to apply to the NuScale design. Specifically, the NuScale plant design does not involve an AFW system as would be found at a typical large LWR. However, the NuScale design does include a DHR system that fulfills functions similar to those performed by a typical AFW. In some instances, as identified in Table 3-1 and Table 3-2 of this report, application of AFW requirements and guidance may warrant further consideration (e.g., departure from inappropriate regulations, utilization of NuScale DSRS sections, etc).

The function of the AFW system at a typical PWR is to provide a source of feedwater supply to the steam generators when the main feedwater system is unavailable. The AFW system is designed to provide AFW automatically following a loss of main feedwater for the removal of sensible heat and reactor core decay heat to prevent core damage. AFW systems are also relied on to prevent and mitigate small-break LOCAs. Design and operation of a typical AFW system involves pumps powered by electrical/steam sources and taking suction from external sources of water (i.e., condensate storage tank).

The NuScale DHR system safety function is to ensure core cooling by providing an alternate source of feedwater to the steam generators when main feedwater is not available. However, the NuScale DHR design and plant response to small-break LOCAs differs from typical large PWR designs. In addition, the DHR system is a simple, passive, closed-loop system, the design and operation of which is significantly different than a typical AFW system for which the current regulatory framework was developed. As discussed further in the NuScale Design Overview

(Reference 5.9), DHR system operation does not require pumps or external sources of feedwater. Rather, it simply involves redirection of the steam flow exiting the steam generators to the DHR heat exchangers, which are immersed in the reactor pool. The steam is condensed in the DHR heat exchangers, and the condensed steam is then introduced back into the steam generators (as feedwater) via natural circulation.

The NuScale gap analysis also identified regulatory guidance related to AFW systems that, due to the design differences summarized above, clearly has no relevance to the NuScale design. For example, BTP 10-1, “Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants” (Reference 5.10), has no relevance since DHR system design and operation does not involve pumps. This and similar instances are indicated in Table 3-1 and Table 3-2 of this report. NuScale recommends the creation of a new DSRS Section (Section 10.4.12) that will specifically address the DHR system.

A.3 Offsite and Onsite AC Power Sources and Distribution Systems

Unlike the electrical power supply and distribution requirements for a typical large LWR plant, the passive design of the NuScale power plant translates to a strong coping capability without reliance on alternating current (AC) or direct current (DC) power. Although the NuScale plant is designed with reliable nonsafety-related offsite and onsite AC power systems, the NuScale plant is designed to achieve safe shutdown and maintain core cooling and containment integrity, independent of AC or DC power sources, for an indefinite duration. In the event of failure of the preferred AC electrical power supply, all SSCs transition to their safety state upon loss of control or motive power and hence systems such as the ECCS and DHR system are able to actuate and operate without the need for AC or DC power. Sufficient nonsafety related battery capacity is available to provide electrical power for other plant functions, including post-accident monitoring and emergency lighting, for a minimum of 72 hours following the onset of a design basis event and also provides power that could be used to prevent the ECCS from actuating for a period of 24 hours to avoid unnecessary challenges to that safety system in the event that AC power can be restored within 24 hours. With this reduced reliance on AC power (compared to a typical LWR design), the NuScale plant does not require or include safety-related emergency diesel generators, and loss of the preferred power source (i.e., offsite power) would have no significant adverse effect on plant safety. The electric power systems are not relied on to achieve any safety functions in the NuScale design. The unavailability of electrical power does not have a significant adverse impact on the ability to achieve and maintain safe-shutdown conditions. The NuScale design features highly reliable AC and DC power systems for prudency and commercial reasons, not for safety related reasons, and the requirements placed on these systems should reflect the low risk to plant safety associated with the loss of these systems during design basis accidents.

The strong coping capability of the NuScale design, with its reduced reliance on AC power, eliminates any significant safety benefit a typical large LWR gains by having an alternate AC power source (e.g., gas turbine generator) for SBO. This conclusion is consistent with the NRC’s policy documented in SECY 90-016 (Reference 5.11), SECY-94-084 (Reference 5.12), and SECY 95-132 (Reference 5.13), and their associated staff requirements memorandums (SRMs). Specifically, SECY-90-016 establishes the policy that advanced LWR plants should have an alternate AC power source of diverse design and be capable of powering at least one complete set of normal shutdown loads in the event of an SBO. In SECY-94-084 and SECY-95-132, the NRC modified this criterion for advanced LWR plants that use passive safety systems (such as the NuScale power plant design). Specifically, as further documented in SRP Section 8.4, an alternate AC power source is not necessary for passive plant designs that (a) do not need AC power to perform safety-related functions for 72 hours following the onset of an SBO, and (b) meet the NRC guidelines for the regulatory treatment of nonsafety systems (RTNSS). As the NuScale design will meet both of these criteria, the NuScale plant does not require an alternate AC power source to satisfy the SBO rule.

The coping capability of the NuScale plant design also obviates the need for a normally available second offsite power circuit as would be required at a typical LWR plant per GDC 17. In the event of a loss of offsite power, power is supplied by onsite nonsafety-related backup diesel generators. These diesel generators are only on the order of 500-1,000 kilowatts in size and are nonsafety-related since they are not relied upon for safe shutdown, core cooling, or containment integrity. By providing safety-related passive systems for core cooling and containment integrity, and multiple nonsafety-related onsite and offsite electric power sources for other functions, no significant safety benefit is realized by providing a redundant offsite power circuit.

With its reduced reliance on AC and DC power, the NuScale plant design does not require undervoltage protection typically required to ensure that safety-related loads are transferred from the preferred power source (i.e., offsite power) to the emergency diesel generators when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. In the NuScale design, a loss of voltage or degraded voltage condition on the offsite power system would have no reasonable likelihood of adversely affecting the performance of plant safety functions. Thus, the undervoltage provisions included in a typical large LWR design and technical specifications (e.g., undervoltage and degraded voltage trip setpoints) are not relevant to the NuScale plant design as design basis requirements, although the NuScale design may meet many of these criteria for commercial reasons.

A.4 NuScale Emergency Core Cooling System and Containment Heat Removal System

A typical ECCS is a complex system of pumps, valves, piping, accumulators, and water storage tanks, the operation of which involves separate injection and recirculation modes. Operation of the ECCS typically includes

- an injection phase, when the pumps take suction from a large tank and pump the tank contents (i.e., borated water) into the reactor
- a recirculation phase when the pumps take suction from the containment sump.

These ECCS designs include numerous active components, including motor-operated valves and pumps, requiring reliable diverse electrical power sources to ensure system actuation and changeover from injection mode to recirculation mode.

The primary safety function of the ECCS at a typical PWR is to provide emergency core cooling, following a loss of reactor coolant, at a rate sufficient to ensure that the core remains in a coolable geometry and that the clad metal-water reaction is limited to negligible amounts. A second safety function provided by the ECCS at many plants is to rapidly inject negative reactivity (i.e., poison addition) in the event of a LOCA.

The NuScale ECCS safety function is equivalent to the primary safety function of an ECCS at a large PWR – to provide emergency core cooling under certain accident conditions. However, the NuScale ECCS is a simple closed-loop system, the design and operation of which is significantly different than a typical ECCS system for which the current regulatory framework was developed. Operation of the NuScale ECCS does not require pumps or external sources of core cooling water (e.g., refueling water storage tank), and does not have separate injection and recirculation modes. Rather, it provides core decay heat removal by steam condensation and natural reactor coolant recirculation.

As discussed further in the NuScale Design Overview (Reference 5.9), the NuScale ECCS is actuated by the opening of two reactor vent valves in lines exiting the top of the (pressurizer region of the) reactor pressure vessel, and two reactor recirculation valves for lines entering the reactor pressure vessel in the downcomer region at a height above the core. This is depicted graphically in the NuScale Design Overview (Reference 5.9). Opening these valves allows a natural circulation path to be established whereby primary water that is heated in the core leaves

as steam through the reactor vent valves, is condensed and collected in the containment vessel, and then flows into the reactor vessel downcomer through the reactor recirculation valves. This design eliminates ECCS components outside containment (as would be found at a typical PWR) that could contain radioactive material following an accident and, as such, would require a leakage control program or filtration in accordance with RG 1.52 (Reference 5.14).

This design also is unique in that the NuScale design does not include or require an active containment heat removal system that serves a heat removal function and a fission product removal/dose mitigation function. Rather, the steel walls of the NuScale containment vessel, together with the heat transfer medium surrounding the containment vessel, serve as a passive system to remove heat from containment (i.e., the containment heat removal system) pursuant to GDC 38. For a minimum of 72 hours following the onset of a postulated design basis event, the heat transfer medium for containment heat removal is the reactor pool water in which the containment vessel is immersed. With the defense-in-depth considerations applied to the NuScale electrical power system design, NuScale expects that AC power would be available well within the initial 72 hours following event onset, allowing for operation of the reactor pool cooling system and pool water level to be maintained. However, even in the absence of nonsafety-related AC power, containment heat removal is assured for an indefinite duration. Specifically, the NuScale design is such that pool water boil-off and, in the unlikely event that all pool water has boiled off, passive air cooling alone provide sufficient cooling for long-term decay and containment heat removal, with no reliance on AC power.

Unlike an ECCS at a typical PWR, the NuScale ECCS does not perform a poison addition safety function nor does it provide any makeup function. Even with consideration for the significant design differences between the NuScale ECCS and an ECCS at a typical PWR, the similarity in safety function between the two allows for applying to the NuScale ECCS much of the regulations and guidance directed towards ECCS. However, in some instances (indicated in Table 3-1 and Table 3-2 of this report), such direct application of the existing regulatory framework may be inappropriate. For example, GDC 27 specifies that "…reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained." NuScale interprets GDC 27 as allowing ECCS poison addition to be credited, but not requiring it. The reactivity control systems associated with the NuScale design meet the requirements of GDC 27 with regards to reliably controlling reactivity changes and maintaining the capability of cooling the core without poison addition by the ECCS. NuScale does not believe a departure is needed from GDC 27. Note that NuScale does not interpret GDC 26 as requiring two safety related means of reactivity control. One of the independent reactivity control systems used to meet the requirements of GDC 26 in the NuScale design is the chemical volume control system, which is not safety related.

A second example is reflected in SRP Section 6.3, which the NRC uses to review proposed ECCS designs and design changes. Due to the design differences discussed above, significant portions of this guidance are not relevant to the NuScale design. This guidance typically would be used for review of the capability of an ECCS to perform both the emergency core cooling safety function (using pumps and external water sources) as well as the poison addition safety function. It may be possible to apply this same guidance, with modification as appropriate, to the review of the NuScale ECCS (related to the emergency core cooling function). Such modifications to this guidance might include the following:

- revision of the SRP acceptance criteria and review procedures to allow for a passive ECCS design that does not use pumps or external water sources (e.g., Acceptance Criterion II.5 of SRP Section 6.3).

- elimination of certain actuation provisions for reactivity control systems (e.g., Acceptance Criterion II.4 of SRP Section 6.3) from the NuScale design review since, as stated above, NuScale does not believe GDC 27 requires ECCS poison addition capability..

Similarly, SRP Section 6.2.2 governs containment heat removal systems. However, as indicated above, the NuScale containment heat removal system design simply consists of the containment vessel and the heat transfer medium surrounding the vessel (which is typically the reactor pool water in which the containment vessel is submerged, except in extreme events wherein reactor pool water is unavailable and air cooling is utilized). This passive design ensures adequate heat removal from containment with no potential for malfunctions of (and the associated need to isolate) active system components. Thus, much of SRP Section 6.2.2 is not relevant to the NuScale design and a NuScale DSRS section is warranted. These examples and other potential modifications to the application of the regulatory framework for the NuScale ECCS and containment heat removal system design are captured in Table 3-1 and Table 3-2 of this report.

A.5 Turbine Missile Protection for Essential SSCs

Substantive differences exist between the NuScale power plant design and the design of a typical large LWR that warrant a different approach, albeit an approach consistent with current regulatory guidance, for providing turbine missile protection for essential SSCs. Typical large LWR designs include plant SSC designs and layouts that inherently result in high exposure of essential SSCs to potential turbine missiles. A large LWR addresses the resultant risk by a combination of the approaches specified in RG 1.115 (Reference 5.15) as being acceptable for meeting GDC 4 with respect to turbine missile protection. These approaches include

- appropriate orientation and placement of the turbine generator
- management of the probability of turbine missile generation or the probability of SSC failure
- the use of missile barriers

The second of these, management of turbine missile and SSC failure probability, is addressed primarily in the design of the main turbine (including turbine rotor) and turbine control system and main valves arrangement to minimize the possibility of turbine missile generation. NRC review of turbine generator and turbine rotor design to minimize missile generation probability is conducted under SRP Section 10.2 and SRP Section 10.2.3, respectively.

Similar to the typical large LWR, NuScale's approach for meeting the provisions of GDC 4 as it relates to turbine missile protection is consistent with the guidance of RG 1.115 (Revision 2). However, due to design features unique to the NuScale power plant, adequate turbine missile protection does not rely on management of turbine missile generation or SSC failure probabilities. Specifically, the NuScale modular design involves smaller, simplified SSC designs and arrangements compared to a large LWR. This allows for the placement of essential SSCs requiring protection from postulated missiles within the robust reactor building structure, which is designed to withstand the effects of postulated missile impacts (as well as a postulated aircraft impact). The design of the reactor building ensures that the probability of barrier perforation and unacceptable damage to essential SSCs from turbine-generated missiles is less than or equal to 10^{-7} per year per plant as specified in RG 1.115.

As indicated above, NuScale intends to satisfy GDC 4 as it relates to turbine missile protection in a manner consistent with the most recent revision of RG 1.115. Based on the above-described design features, unique to the NuScale design, NuScale does not anticipate any significant safety benefit associated with applying the measures specified in SRP Sections 10.2 and 10.2.3 with respect to turbine generator and rotor design to minimize missile generation probabilities.

Specifically, the NuScale turbine generator is an "off-the-shelf" design that includes standard overspeed protection and control features with a proven record of quality and reliability. Pre-service inspection, inservice inspection, and maintenance of turbine generator components would comply with manufacturer recommendations. These measures inherently will minimize the probability of turbine missile generation; however, it is noted that they are provided primarily for asset and personnel protection, and are not intended to be relied upon for turbine missile protection of essential SSCs. Rather, consistent with RG 1.115, Revision 2, NuScale will satisfy the criteria of GDC 4 by the appropriate orientation and placement of the turbine generators, combined with the proper design and use of missile barriers[6] to protect essential SSCs against potential turbine-generated missiles. The acceptability of this approach is reviewed under SRP Sections 3.5.1.3 and 3.5.3.

A.6 Reactor Coolant System High Point Vents

10 CFR 52.47(a)(4) requires that design certification applicants address the high point vent requirements of 10 CFR 50.46a. 10 CFR 50.46a requires high point vents for the reactor coolant system and reactor vessel head, and also for other systems required to maintain adequate core cooling if the accumulation of noncondensable gases would cause the loss of function of these systems. Substantively similar requirements for reactor coolant system venting capability are codified in 10 CFR 50.34(f)(2)(vi). The underlying purpose of these requirements was to resolve post-TMI concerns that an accumulation of noncondensable gases could interfere with post-accident natural circulation or pump operation that might inhibit long-term cooling following an accident (see 68 FR 54123, at 54129, September 16, 2003).

The NuScale reactor module design includes the capability for reactor coolant system venting during normal operations via a small vessel head vent line. However, as a result of significant differences in the NuScale advanced reactor design compared to a traditional large LWR, separate NuScale reactor coolant system venting capability is not needed to meet the underlying purpose of 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi). Specifically, as discussed further below, the NuScale reactor module design is such that there is no reasonable likelihood that an accumulation of noncondensable gases could interfere with post-accident natural circulation or otherwise inhibit long-term cooling following an accident. Thus, although high point venting capability is included in the NuScale design to periodically remove accumulated noncondensable gases during normal operations, it is not relied upon to perform a safety function specific to ensuring long-term core cooling as contemplated by 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi).

As indicated above, the NuScale design differs from that of large LWRs in that natural circulation core cooling cannot be inhibited in the NuScale design by the accumulation of noncondensable gases, whether such accumulation is in the reactor vessel head or other system (e.g., ECCS). This has certain implications for the NuScale design that differ in significant respects from traditional LWR designs. First, additional high point vents in "other systems required to maintain adequate core cooling" as specified in 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) are not required, since the NuScale design does not include such systems in which accumulation of noncondensable gases would cause a loss of function. As discussed in Section A.4, operation of the NuScale ECCS relies on passive, natural circulation to maintain core cooling. Its design is such that no credible accumulation of noncondensable gases would adversely affect its ability to provide adequate core cooling.

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[6] As indicated above, essential SSCs are located in the reactor building; thus, the reactor building structure would represent the barrier for missiles generated outside the reactor building, including turbine-generated missiles.

There is no reasonable likelihood that an accumulation of noncondensable gases in the RCS or reactor pressure vessel could inhibit post-accident core cooling flow. For this reason, the venting of noncondensable gases does not have a safety-related function specific to ensuring long-term core cooling as contemplated for traditional LWRs by 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi). NuScale believes the appropriate approach is to request an exemption from 10 CFR 50.46a because the collection of noncondensable gases will not inhibit post-accident core cooling. 10 CFR 50.34(f)(2)(vi) is considered not technically relevant to the NuScale design in accordance with 10 CFR 52.47(a)(8) which requires compliance with the technically relevant portions of the Three Mile Island requirements. NuScale will seek NRC concurrence on this issue in the pre-application phase.

A.7 Containment Vessel

A containment building at a typical large PWR is a massive structure – approximately 200 to 230 feet in height and 60 to 130 feet in diameter – that houses the reactor vessel and numerous reactor system components. A typical PWR containment consists of a coated steel liner surrounded by reinforced or pre-stressed concrete. The concrete component provides the pressure retaining capability of the containment and acts as a biological shield (against gamma radiation) while providing protection for the SSCs within containment from outside elements (e.g., tornado and hurricane missiles). Many system components are located (and high-energy lines routed) inside sub-compartments within these typical containment structures, thus requiring consideration of transient differential pressures due to postulated pipe breaks inside a sub-compartment. The ability of the containment to provide prompt isolation and contain the highest expected pressure ensures that the containment is able to act as a fission product barrier to prevent the release of radiological contaminants following a design basis accident. Integrity of a typical containment relies on pressure suppression systems (e.g., sub-atmospheric operation, suppression pools, or an active containment heat removal system such as containment spray or ice condenser) that also serve a fission product removal and dose mitigation function.

As with a typical PWR containment, the NuScale containment vessel serves to contain the release of radioactivity following postulated accidents, and to protect the reactor pressure vessel and its contents from external hazards. However, the NuScale containment vessel design differs significantly from the typical PWR containment design discussed above. The compact NuScale containment vessel is significantly smaller than a typical containment building, with a cylindrical shape and nominal dimensions of approximately 76 feet (height) and 15 feet (outer diameter).[7] Whereas a typical large PWR containment is a permanent structure housing extensive reactor systems and associated piping, the NuScale containment vessel is a portable steel component that forms the outer boundary of the NuScale power module.[8] The NuScale containment vessel has no interior sub-compartments, thus eliminating the potential for damaging transient differential pressures resulting from postulated high-energy pipe breaks within sub-compartments (or internal compartments as referred to in GDC 50).

In addition to the safety functions described above, the NuScale containment vessel also provides an interfacing medium for decay and containment heat removal. Specifically, the steel walls of the NuScale containment vessel, together with the heat transfer medium surrounding the containment vessel, serve as a passive system to remove heat from containment (i.e., the containment heat removal system) pursuant to GDC 38. This passive design configuration

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[7] These dimensions are based on the current stage of engineering design and are subject to change as design progresses.

[8] As described in the NuScale Design Overview (Reference 5.9), a NuScale power module comprises a reactor core, two steam generators, and a pressurizer all contained within a single reactor vessel, along with the containment vessel that immediately surrounds the reactor vessel.

contributes to ensuring effective passive, natural circulation ECCS flow during and following a postulated accident requiring ECCS operation (see description of ECCS flow and containment heat removal in Section A.4).

Typical containment designs include containment purge and vent lines that provide an open path from the containment to the environs. Purge and vent capability is intended to allow personnel access and, in some designs, to address combustible gas control and/or maintain containment pressure to ensure ECCS performance. The NuScale containment design does not include a containment purge and vent system or other system that provides a direct open path to the environs. As discussed above, the compact NuScale containment vessel is significantly smaller than a typical containment building, and its design is such that personnel access during reactor operation, and purge and vent capability for combustible gas control are not needed. In addition, the NuScale ECCS design does not include pumps, and does not involve a typical PWR ECCS recirculation mode (i.e., ECCS pump suction is switched from water storage tank(s) to containment sumps) where ECCS pump performance relies on containment pressure. Thus, purge and vent capability is neither required nor included in the NuScale design. With no purge and vent system providing large-diameter (24 inch to 60 inch) direct open paths to the environs, concerns with isolation capability (e.g., issues raised in 10 CFR 50.34(f)(2)(xv)) of the large isolation valves in these lines are not germane to the NuScale design. It is noted that while the NuScale containment vessel includes an evacuation system, it serves a different purpose than a typical containment purge system, and does not provide a direct open path to the environs.

Specifically, the containment evacuation system is used to establish the dry (i.e., no liquid water), partial vacuum conditions under which the NuScale containment vessel is designed to function during normal operations. Rather than providing a direct open path to the environs as would a typical containment purge system, the NuScale containment vessel evacuation system transfers removed gases directly to the gaseous waste management system, and liquids either to the liquid waste management system or to the reactor pool. The dry, evacuated condition is maintained in the containment vessel to realize specific benefits discussed in the NuScale Design Overview (Reference 5.9). However, the partial vacuum condition is neither intended nor relied upon as an inerted atmosphere that would be credited for combustible gas control pursuant to 10 CFR 50.44(c)(2). For the NuScale design, even with consideration for a postulated release of hydrogen in an amount that would be generated from a 100 percent fuel clad-coolant reaction as specified in 10 CFR 50.44(c), a postulated worst-case uncontrolled hydrogen-oxygen recombination would not challenge the integrity of the containment vessel. The resultant pressure effects due to the worst-case event would be well within the containment design internal pressure rating, thus assuring that containment vessel structural integrity is maintained.

The capability to ensure a mixed atmosphere as required by 10 CFR 50.44(c)(1) is inherent in the design of, and absence of, sub-compartments within the NuScale containment vessel, with no reliance on active systems or components (e.g., fans, fan coolers, or containment spray). This mixing ensures that there is no significant concentration of combustible gases in localized areas that would support combustion or detonation of a magnitude that could cause loss of containment integrity.

Based on the above, it is concluded that a postulated worst-case hydrogen combustion would have no significant adverse effect on plant safety functions. Thus, there is no significant safety benefit associated with maintaining an inert containment atmosphere or limiting hydrogen concentrations in the containment vessel to less than 10 percent (by volume) following a postulated design basis accident as required by 10 CFR 50.44(c)(2). Accordingly, the NuScale containment vessel design does not use combustible gas control systems or hydrogen/oxygen monitors, nor is an inerted atmosphere maintained that would be credited for combustible gas control pursuant to 10 CFR 50.44(c)(2). Nevertheless, the robust design of the NuScale containment vessel and physical limitation on available oxygen, as discussed above, satisfy the underlying purpose of 10 CFR 50.44(c)(2). Further discussion with regards to combustible gas control within the containment vessel can be found in Table 3-1 of this report.

As indicated above, the NuScale containment vessel is designed to accommodate, with sufficient safety margin, maximum anticipated pressure conditions without relying on reducing containment pressure to sub-atmospheric conditions following a postulated design-basis accident. This ensures that the containment vessel is able to act as a fission product barrier to prevent the release of radiological contaminants following a design-basis accident. Unlike a typical large LWR containment, the NuScale containment vessel design does not include or require an ESF atmosphere clean-up system or pressure suppression systems (e.g., suppression pools or active containment heat removal systems) that serve a fission product removal or dose mitigation function. Rather, for the NuScale power plant design, fission product control associated with containment design and operational characteristics include

- the robust design of the NuScale reactor module containment vessel, which as discussed above ensures its integrity as a fission product barrier under maximum anticipated pressure conditions.
- reactor module configuration wherein the compact steel containment vessel is submerged in the reactor pool, which in turn is housed within the reactor building (i.e., the reactor pool and reactor building provide defense-in-depth, in addition to credited barriers including the containment vessel itself, to fission product release).
- design, inspection, and testing of containment vessel isolation provisions.
- containment vessel design leakage rate.

As indicated above, the partially evacuated space between the NuScale containment vessel and the reactor vessel is dry (i.e., does not contain water) under normal operating conditions. The NuScale containment vessel does not include a containment spray system or containment sumps for recirculation water, and compared to a typical PWR containment structure, contains minimal equipment and system components. However, the potential presence of water in the containment vessel (e.g., upon ECCS actuation) requires design consideration, similar to that required for a typical large LWR, to minimize potential interaction of the water with materials and equipment within the containment vessel. Such considerations include pH control and material selection to prevent potential hydrogen generation from interaction of steam or water with materials within the containment vessel.

Due to its close proximity to the reactor core, the NuScale containment vessel has a greater susceptibility to radiation embrittlement (although to a lesser extent than the reactor vessel itself) as compared to a typical LWR containment structure. Thus, for the NuScale containment vessel, fracture toughness requirements similar to those described for the reactor coolant pressure boundary in 10 CFR 50, Appendix G, and a material surveillance program similar to that described for the reactor vessel in 10 CFR 50, Appendix H, will be implemented to ensure that the NuScale containment vessel satisfies the provisions of GDC 16 and GDC 51 over its 60-year design life.

Due to the partial vacuum condition within the containment vessel, any reactor coolant leakage (whether identified or unidentified) into the containment vessel would quickly vaporize and disperse within the containment vessel atmosphere. Upon vaporization, there is no feasible means to monitor separately the flow rates of identified and any unidentified leakage. To allow for the unique module design that precludes the practical separation of identified and unidentified leakage within the containment vessel, any leakage into the containment vessel will be conservatively assumed to be unidentified leakage, and the total leakage flow rate will be established and monitored as specified in Regulatory Guide 1.45, Position C.1.2(ii).

As a result of the design differences summarized above, portions of the SRP and other regulatory guidance directed towards containment design are not appropriate to apply to the NuScale containment vessel design. See Table 3-1 and Table 3-2 for additional details.

A.8 ESF Ventilation/Atmosphere Cleanup Systems

At a typical LWR plant, ESF ventilation systems are used to maintain a controlled environment in areas containing safety-related equipment essential for the safe shutdown of the reactor or necessary to prevent or mitigate the consequences of an accident. ESF ventilation systems also are used to ensure that suitable environmental conditions are maintained in areas containing equipment required to function for a station blackout. ESF atmosphere cleanup systems are designed for fission product removal in post-accident environments (i.e., to mitigate the consequences of accidents). These systems generally include in-containment recirculation, and secondary systems such as standby gas treatment systems and emergency or post-accident air-cleaning systems for the fuel-handling building, control room, shield building, and areas containing ESF components.

Unlike a typical large LWR plant, the NuScale design does not rely on ESF ventilation systems to mitigate the consequences of a design basis accident. Nonsafety-related normal ventilation systems provide atmosphere cleanup capability, as necessary, that meets the design, testing, and maintenance guidelines specified in RG 1.140 (Reference 5.17). These systems provide appropriate containment, confinement, and filtering to limit releases of airborne radioactivity to the environment during normal and postulated accident conditions. However, these systems are not required following an accident, and accordingly receive no credit in the determination of the radiological consequences of an accident.

In the NuScale design, a containment ESF atmosphere cleanup system is not needed to control fission products that may be released into the containment vessel, nor to reduce the concentration of fission products released to the environment after an accident. As discussed in Section A.7, unlike a typical large LWR containment, the NuScale containment vessel design does not include an ESF atmosphere clean-up system or pressure suppression systems that serve a fission product removal/dose mitigation function. Rather, for the NuScale power plant design, fission product control associated with containment design and operational characteristics include

- the robust design of the NuScale reactor module containment vessel, which, as discussed above, ensures its integrity as a fission product barrier under maximum anticipated pressure conditions.
- reactor module configuration wherein the compact steel containment vessel is submerged in the reactor pool, which in turn is housed within the reactor building (i.e., the reactor pool and reactor building provide defense-in-depth, in addition to credited barriers including the containment vessel itself, to fission product release).
- design, inspection, and testing of containment vessel isolation provisions.
- containment vessel design leakage rate.

When considered together with the significantly reduced source term that the NuScale design has compared to a typical large LWR, these features provide assurance that, with no reliance on a containment ESF atmosphere cleanup system, the calculated dose is less than the criteria of 10 CFR 100.21, 10 CFR 50.34(a)(1)(ii)(D), and 10 CFR 52.47(a)(2)(iv). Therefore, offsite radiation doses resulting from an accident will be within regulatory limits, and containment ESF filtration is not needed.

In the NuScale design, a main control room ESF ventilation system is not needed to provide assurance that personnel needed to monitor and control an accident will be able to perform those functions effectively. Specifically, the NuScale main control room habitability system neither relies on nor uses ESF emergency filtration to protect operators during accident conditions. Rather, clean air is provided using compressed air storage tanks. This eliminates the potential for radioactive material or toxic gases to enter the control room via ventilation system inlets. The air

storage tanks are sized to deliver the required air flow to the main control room to meet ventilation and pressurization requirements for a minimum of 72 hours. Furthermore, the NuScale design does not require operator intervention to achieve and sustain a safe shutdown state following a design basis accident (as discussed in Section A.0 of this report).

A.9 Steam Generators

The NuScale once-through helical-coil steam generator design differs significantly from existing technology for which current regulatory and industry guidance (e.g., SRP Section 10.4.8 [Reference 5.2], BTP 10-2 [Reference 5.18], EPRI PWR Secondary Water Chemistry Guidelines [Reference 5.19], and NEI 97-06 [Reference 5.20]) was developed. Specifically, for steam generator designs found at a typical large PWR, heated primary water flows from the reactor vessel through piping loops to the steam generators. There the primary coolant passes through the steam generator tubes and its heat is transferred to the secondary water on the outside (i.e., shell side) of the tubes.

In the NuScale design, two helical coil steam generators are located within each reactor vessel, such that the reactor coolant flow path is completely contained within the reactor vessel. The tubes of the two steam generators are intertwined in a DNA-like double-helix configuration. The heated reactor coolant flows upward from the core (via natural circulation) through a large diameter central riser, then downward around the intertwined steam generator tubes where its heat is transferred to the secondary water on the inside (i.e., tube side) of the tubes. The coolant flow then continues downward through the annular downcomer to the plenum where it reenters the core. Additional details of the NuScale steam generator design are provided in the NuScale Design Overview (Reference 5.9).

The NuScale design summarized above has a number of significant design, operational, and safety benefits compared to traditional PWR and steam generator designs. Having only a single reactor coolant “loop” entirely contained within the reactor vessel eliminates the reactor coolant system piping loops and associated potential piping break (i.e., large break LOCA) events associated with traditional PWR designs. The “single loop” design, combined with the intertwined steam generator tube configuration, also eliminates the potential that a typical PWR design has for asymmetric core temperatures as a result of a steam line failure or isolation of a single steam generator. Specifically, for PWR plant designs that involve multiple reactor coolant loops and steam generators, a postulated steam line failure or steam generator isolation potentially would result in asymmetric core temperatures. However, isolation or failure of one of the two NuScale helical coil steam generators would not introduce asymmetrical cooling in the reactor coolant system since, with the intertwined tube configuration, both steam generators exert an equal impact on the symmetric downcomer reactor coolant temperature profile.

As stated above, in the NuScale design the primary water is outside the tubes, and the secondary water is inside the tubes. With the primary system at a higher pressure than the secondary, this design results in the steam generator tubes being in compression, reducing the likelihood of a tube rupture and eliminating the potential for pipe whip (compared to a typical steam generator design). With all secondary water flowing within the steam generator tubes, the deposition of secondary-side impurities in the NuScale steam generator is significantly different than for a typical PWR steam generator. The NuScale design has no secondary side crevices or low flow regions that could accumulate soft sludge, which may be effectively removed via a blowdown system as in a typical PWR. Secondary impurities will deposit directly on the tube surfaces, as a scale or film, which will be removed through periodic chemical and/or mechanical cleanings performed during outage periods. Unlike a traditional once through steam generator, the once through NuScale steam generator does not contain a bulk reservoir of water at the bottom of the heat exchanger that could provide a representative grab sample or facilitate blow down. Therefore, some of the sampling requirements dictated by the referenced EPRI PWR Secondary Water Chemistry Guidelines and NEI 97-06 guidelines will require a different approach for the

NuScale design. For reasons such as these, NuScale recommends the creation of a “Feedwater Treatment System” DSRS section (see Table 3-2 section 10.4.10).

The NuScale steam generator design eliminates the component configurations and minimizes the hydraulic instabilities that in a typical large PWR steam generator introduce potential sources of water hammer. For example, in the NuScale design, the feedwater enters the steam generator tubes at their lowest point. As the feedwater rises through the tubes, it experiences a phase change and exits the steam generator tubes as superheated steam. This configuration keeps the steam-water interface fairly fluid and the superheated steam separated from the subcooled liquid at the bottom of the tubes. Additional details regarding the NuScale steam generator design, including the features that are expected to result in no significant potential for water hammer, are provided in the NuScale Design Overview (Reference 5.9).

As a result of the design differences summarized above, portions of the SRP and other guidance directed towards steam generator design are not appropriate to apply to the NuScale steam generator design. See Table 3-2 for additional details.

A.10 Design Provisions Assuring Adequate Reactor Coolant Inventory as Alternative to GDC 33

The NuScale design includes a chemical volume control (CVC) system that provides nonsafety-related reactor coolant makeup capability to accommodate minor leakage from the reactor coolant system and level changes during reactor heatup and cooldown. However, CVC system makeup is not relied upon to prevent uncovering the core or to ensure core cooling in the event of a postulated leak in the reactor coolant pressure boundary. Rather, the design of the NuScale reactor module and decay heat removal systems ensure that the core will not be uncovered and the core will be cooled in a postulated design basis event involving a leak in the reactor coolant pressure boundary.

Specifically, with the exception of a postulated steam generator tube leak, the NuScale reactor module is designed such that reactor coolant water from a postulated credible leak[9] in the reactor coolant pressure boundary would be isolated within the containment vessel. For a postulated steam generator tube leak, leakage water would be isolated within the affected steam generator(s) that reside within the reactor vessel. These design provisions and the passive design of the decay heat removal systems (i.e., the DHR system and the ECCS) provide assurance that adequate reactor coolant inventory is maintained to ensure that leaks do not result in uncovering the core or loss of core cooling.

For postulated reactor coolant leaks other than a steam generator tube rupture, the NuScale reactor module and containment vessel (isolation) design ensures that the leakage would be isolated and contained within the containment vessel. As described in Section A.4, upon ECCS actuation the water accumulated in the containment vessel would be passively returned to the reactor vessel (i.e., core) by natural circulation. Thus, the reactor module and containment vessel design, in conjunction with the passive design and operation of the ECCS, ensure that the core will not be uncovered and adequate core cooling will be maintained.

For a postulated steam generator tube rupture, leakage water would be contained within the affected steam generator(s) that reside within the reactor vessel by isolating the affected steam generator(s). For postulated tube leaks affecting only one of the two tube bundles within the reactor vessel, the affected steam generator (and thus the leak) would be isolated, and core

[9] The NuScale modular reactor design does not include large diameter piping as part of the reactor coolant pressure boundary; therefore, leakage as a result of a large break LOCA is not possible in the NuScale design.

decay heat removal would be provided by the unaffected steam generator via DHR system operation (see Section A.2 for a description of DHR system operation). For postulated tube leaks affecting both tube bundles within the reactor vessel, both steam generator tube bundles would be isolated, and core decay heat cooling would be provided by the ECCS as described in Section A.4. Thus, for postulated steam generator tube leaks, isolation of the steam generator(s) contained within the reactor vessel, in conjunction with the passive design and operation of the DHR system and/or ECCS, ensure that the core will not be uncovered and adequate core cooling will be maintained.

With these design provisions assuring adequate reactor coolant inventory to ensure that leaks do not result in uncovering the core or loss of core cooling, a coolant makeup system as contemplated by GDC 33 is not appropriate for the NuScale advanced reactor design. However, with the reliance on design provisions assuring adequate inventory control, a NuScale-specific principal design criterion for the assurance of adequate reactor coolant inventory is warranted as an alternative to GDC 33. The intent of this criterion would be to require that the reactor coolant pressure boundary and associated systems and components be designed to limit loss of reactor coolant so that an inventory adequate to perform the safety functions of the core decay heat removal systems (including the DHR system and the ECCS) is maintained under normal operation (including anticipated operational occurrences) and postulated accident conditions. It is noted that a similar alternative design criterion to GDC 33 has been determined to be acceptable by NRC in other applications (References 5.6 and 5.7).

A.11 Residual Heat Removal System Safety Functions

A typical RHR system is a complex system of pumps, valves, and piping that shares common piping and nozzles at the reactor coolant loop piping interface with the plant's emergency core cooling systems. The RHR system is used to cool the RCS during and following shutdown. Parts of the RHR system may also act to provide low-pressure emergency core cooling and containment heat removal capability. For these functions performed by a typical RHR system, the NuScale design includes systems that fulfill substantively equivalent or similar safety functions.

A.11.1 Safety Function Related to RCS Cooling

With respect to the function of providing RCS cooling during and following shutdown, the NuScale main feedwater system is available to provide an equivalent function. The design of the NuScale main feedwater system is substantively similar to that of a typical PWR main feedwater system, such that the existing NRC guidance (e.g., SRP Section 10.4.7) may be applied in its review.

Additionally, the NuScale design contains a nonsafety related containment flooding system (CFS). The CFS is utilized during the normal shutdown sequence and also provides a defense-in-depth emergency water injection capability. The CFS provides a means of flooding the containment by pumping borated water from the reactor pool into the containment via the containment evacuation system's penetration.

A.11.2 Safety Function Related to Providing Low-Pressure Emergency Core Cooling and Containment Heat Removal

As discussed in Section A.11, a typical RHR system may also act to provide low-pressure emergency core cooling and containment heat removal capability. As described in the NuScale Design Overview, the NuScale ECCS, operating in conjunction with the containment heat removal system, serves the function of providing emergency core cooling through the entire range of pressures that would be experienced as the plant is cooled from normal operating temperature to a cold shutdown condition. The relevance of existing LWR-based regulations and guidance for application to the NuScale ECCS and containment heat removal system designs is addressed in Section A.4.