Gap Analysis Summary Report

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Nonproprietary

NuScale Power, LLC

1100 NE Circle Blvd Suite 350

Corvallis, Oregon 97330

www.nuscalepower.com

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Executive Summary

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 reactor 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 its “sub-tier” documents.

Gap Analysis Results

The results of the NuScale regulatory gap analysis are summarized in Section 3.0 of this report. Additional details supporting the summary 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 reactor 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 reactor plant design, are not relevant and thus would be inappropriate to apply, either in whole or in part, to the NuScale reactor plant design. These “gaps” in the LWR-based regulatory framework warrant further consideration, such as regulatory departure and/or exemption, additional interpretation, or other form of NRC approval or concurrence to be defined during pre-application deliberations between NuScale and NRC. The specific regulations determined to warrant further consideration are summarized in Table 3-4 of this report.

From a detailed review of the SRP, 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 reactor plant design. Based on this determination, each SRP section then was assessed to determine whether or not it could be applied by the NRC, either “as-is” or with modification, in its review of the NuScale application for design certification. NuScale’s assessment of each SRP section also considered the extent to which the section could be used in the NRC’s review of combined license applications that reference the NuScale design. This added consideration was taken since there are a number of SRP sections that govern site-specific information that is not relevant to (i.e., would not be part of) a certified standard plant design but is germane to a license application review, since such site-specific information is available to the license applicant. The results of this assessment — essentially a proposed disposition of each SRP section with respect to how the section would be applied as part of a NuScale design-specific review standard — are provided in Table 3-5 of this report. The dispositions contained in Table 3-5 are intended as a foundation (i.e., starting point) to facilitate deliberations between NuScale and the NRC during pre-application activities.

Conclusion

The completion of the NuScale regulatory gap analysis represents a major milestone in the pre-application phase of the NuScale design certification effort. The results summarized in this report provide a strong foundation to facilitate deliberations between NuScale and NRC during pre-application activities. Specifically, as part of further pre-application activities, NuScale will seek to

reach consensus with the NRC on the: (1) applicability of the regulatory framework as assessed in this gap analysis; and (2) the disposition of “regulatory gaps” identified in Section 3.0 of this report.

NuScale believes that the regulatory gap analysis results, with revisions to reflect the NRC’s final determinations on applicability and gap dispositions, represent the necessary information for development of a design-specific review standard to be used by the NRC in its review of the NuScale application for design certification. Specifically, where gaps are identified, appropriate modifications to affected SRP sections, or their replacement with new design-specific sections, will be significant activities in the NRC’s development of a NuScale design-specific review standard. NuScale remains committed to assisting the NRC as necessary and appropriate to facilitate this effort.

NuScale intends to use the results of this gap analysis, with any revisions as discussed above, to prepare a proposed content outline for the design control document (DCD) to be submitted pursuant to 10 CFR 52.47(a) as part of the NuScale application for design certification. The content outline will have particular focus on those DCD sections that, due to unique NuScale design features, are anticipated to differ significantly from what would be provided for a typical LWR application. NuScale anticipates providing the proposed content outline to the NRC in the fourth quarter of 2012. The content outline should facilitate alignment between NuScale and the NRC on planned content and structure of the DCD and the NRC’s development of the NuScale design-specific review standard as discussed above.

Finally, 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 reactor plant design. As the ongoing engineering design effort progresses in support of the NuScale application for design certification, the relevance of all or 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.

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 reactor plant design. Upon U.S. Nuclear Regulatory Commission (NRC) concurrence, this regulatory gap analysis is intended to establish a documented, clear delineation of NRC expectations for completeness for the NuScale reactor plant design certification application in those areas that materially differ from existing LWR requirements and guidance. 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 reactor plant design or design certification application specifically due to features, functions, and capabilities unique to the NuScale reactor plant.

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. applicability of the regulatory framework as assessed in this gap analysis.
2. disposition of “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 format and content outline (to be provided to NRC during the pre-application process) of the DCD that will be submitted as part of the NuScale standard design certification application pursuant to 10 CFR 52.47(a).
    - the NuScale plant licensing basis.
2. the NRC’s
    - ultimate determination of relevance/applicability of regulations and guidance to the NuScale reactor plant design.
    - development of a design-specific review standard for the NuScale plant design, consistent with the introduction to NUREG-0800, draft Revision 3 (Reference 5.3) and SECY-11-0024 (Reference 5.4).

1.3 Abbreviations and Definitions

Table 1-1. Abbreviations

Term	Definition
AFW	auxiliary feedwater
ANS	American Nuclear Society

Term	Definition
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
COL	combined operating license
COLA	combined operating license application
CVC	chemical and volume control
DAC	design acceptance criteria
DC	design certification
DCD	design control document
DHR	decay heat removal
DI&C	digital instrumentation and control
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
GDC	general design criterion
I&C	instrumentation and control
ISG	interim staff guidance
ITAAC	inspections, tests, analyses, and acceptance criteria
LOCA	loss-of-coolant accident
LWR	light-water reactor
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

Term	Definition
NuScale	NuScale Power, LLC
PLC	programmable logic controller
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
SAS	shutdown accumulator system
SBO	station blackout
SECY	Secretary of the Commission, Office of the (NRC)
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

The primary objective of the NuScale regulatory gap analysis is to identify existing LWR-based regulations and guidance that are not technically relevant and thus would be inappropriate to apply to the NuScale plant design specifically due to features, functions, and capabilities unique to the NuScale plant. To achieve this objective, the NuScale regulatory gap analysis methodology involved a detailed review of existing LWR regulations and guidance for applicability and technical relevance to the NuScale reactor 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, including the following:
    - regulatory guides (RGs) including RG 1.206 (Reference 5.5)
    - 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) requirements
    - 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,” and “NSIR/DPR-ISG”)1
5. Division 1, 4, 5, and 8 regulatory guides[2] other than those that are sub-tier to the Standard Review Plan (see item 3・ above)

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 will evolve over time. Any new gaps resulting from this evolution will be identified as part of the NuScale design process, with

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1 The remaining five of the eight categories of currently effective ISGs are specifically directed towards license renewal, research and test reactors, fuel cycle facilities, high-level waste repositories, and activities conducted under 10 CFR 71 and 10 CFR 72. With no potential relevance to an application for design certification, these five categories of ISGs were not reviewed as part of the NuScale regulatory gap analysis effort.

2 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.”

finality anticipated upon approval and issuance of the design certification for the NuScale reactor plant design.

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 reactor plant design. NuScale design information was considered in an effort to identify those design functions and characteristics unique to the NuScale reactor plant design, i.e., those that differ significantly from design functions and characteristics for the typical large LWR. These design functions and characteristics were then compared to the NRC regulations and guidance for relevance and applicability.

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 reactor plant design but cannot be 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 agreements, as appropriate, to be developed during the design certification process.

Following the determination of relevance and applicability in the form of one of the four outcomes discussed above, each regulatory requirement or guidance document was assessed to determine whether it could be applied by the NRC in its review of the NuScale application for design certification. The results of this assessment are summarized in Section 3.0 of this report.

3.0 Gap Analysis Summary Results

As discussed in Section 2.2 of this report, the NuScale regulatory gap analysis effort included, for each regulatory requirement and individual guidance criterion reviewed, a determination of relevance and applicability in the form of one of four outcomes: “Applicable,” “Partially Applicable,” “Not Applicable,” or “NuScale Unique Feature or Requirement.” 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 applied by the NRC – either “as-is” or with modification – in its review of the NuScale application for design certification. As summarized below and in Tables 3-4 and 3-5 of this report, the results of this assessment indicate a number of “gaps” that form part of the regulatory issues that need to be resolved with NRC during the pre-application phase. Notwithstanding these gaps, 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 reactor 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 reactor plant design. As the ongoing engineering design effort progresses in support of the NuScale application for design certification, the relevance of all or 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 NuScale gap analysis review of the NRC regulations is provided in detail in the NuScale electronic reading room.

From this review, it was determined that due to design features, functions, and capabilities unique to the NuScale reactor 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 reactor plant design. These “gaps” in the LWR-based regulations warrant further consideration, such as regulatory departure and/or exemption, reinterpretation, or other form of NRC approval and concurrence to be defined during pre-application deliberations between NuScale and the NRC. Table 3-1 illustrates the extent of applicability that current NRC regulations were determined to have with respect to the NuScale application for design certification. The specific regulations determined to warrant further consideration are summarized in Table 3-4 of this report.

Table 3-1. Extent of applicability of NRC regulations

Regulation	Total Items[3] Reviewed	Applicability Determination Result				Further Consideration Needed
		Applicable	Partially Applicable	Not Applicable	NuScale Unique Feature or Req't	
10 CFR 20	79	56	17	6	0	0
10 CFR 50	201	44	30	127	0	10
10 CFR 50, Appendix A (GDCs)	55	45	6	4	1	7
10 CFR 51	91	34	2	55	0	0
10 CFR 52	101	24	1	76	0	0
10 CFR 73	48	17	3	28	0	0
10 CFR 100	11	9	0	2	0	0

As part of the NuScale gap analysis review of NRC regulations, consideration was given to NuScale reactor plant design features that potentially could not be addressed by existing regulations, thus requiring new requirements or agreements to be developed during the design certification process. As indicated in Table 3-1, one such instance was 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 3-4 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. As indicated in Table 3-1, there were seven 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 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,

[3] In most instances, an item is defined as an individual section (e.g., 10 CFR 20.1001, 10 CFR 50.65, etc.) of the regulation. For 10 CFR 50.34, 10 CFR 50.54, 10 CFR 50.72, 10 CFR 50.73, and Appendix A to 10 CFR 50, further breakdown of the section was warranted to differentiate between the applicability of specific paragraphs of each section. Portions of the regulations considered as "items" are indicated in the detailed gap analysis tables provided in the NuScale electronic reading room.

> 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 reactor plant design. Table 3-4 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, as it is concluded from the above discussion that 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)[4] 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.

3.2 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, guidance and standards referenced within and thus sub-tier to the SRP, and 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).[5] The gap analysis review for applicability was directed towards the acceptance criteria of each SRP

4 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.

5 References to "SRP" in the remainder of this section shall include NUREG-0800, associated branch technical positions, and the aforementioned ISGs.

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 and reflected in Table 3-5 of this report, 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 reactor plant design. Based on this determination, each SRP section then was assessed to determine whether or not it could be applied by the NRC, either "as-is" or with modification, in its review of the NuScale application for design certification. The results of this assessment — essentially a proposed disposition of each SRP section with respect to how the section would be applied as part of a NuScale design-specific review standard — are provided in Table 3-5 of this report.

From pre-application discussions with the NRC, NuScale understands 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 the section could be used in the NRC's review of combined license applications that reference the NuScale design. This added consideration was taken since there are a number of SRP sections that govern site-specific information that is not relevant to (i.e., would not be part of) a certified standard plant design, but is germane to a license application review since such site-specific information is available to the license applicant.

For example, SRP Sections 2.1.1 and 2.1.3 govern the review of reactor 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.

Derived from the summary information in Table 3-5 of this report, Table 3-2 illustrates the extent of applicability that current SRP sections were determined to have with respect to the NuScale design. In some instances, the dispositions of SRP guidance proposed in Table 3-5 do not represent the only reasonable approach for applying the existing SRP guidance to the NuScale design certification application review.

For example, an SRP section may be proposed in Table 3-5 to be used with modification as part of a NuScale design-specific review standard. However, an alternative approach may be to not use the SRP section for the NuScale application, but rather to create a new section to replace the existing SRP section in the NuScale design-specific review standard. Thus, the dispositions contained in Table 3-5 are intended as a foundation (i.e., starting point) to facilitate deliberations between NuScale and the NRC during pre-application activities.

Table 3-2. Extent of applicability of Standard Review Plan

SRP Chapter	Total Sections	DSRS Disposition							
		Use As-Is		Use With Modification		Do Not Use (N/A)		New Section	
		DCA	COLA	DCA	COLA	DCA	COLA	DCA	COLA
SRP Chapter 1	1	1	1	0	0	0	0	0	0
SRP Chapter 2	30	23	30	0	0	7	0	0	0
SRP Chapter 3	42	15	23	22	16	5	3	0	0
SRP Chapter 4	7	5	5	1	1	1	1	0	0
SRP Chapter 5	23	14	14	2	2	7	7	0	0

SRP Chapter	Total Sections	DSRS Disposition							
		Use As-Is		Use With Modification		Do Not Use (N/A)		New Section	
		DCA	COLA	DCA	COLA	DCA	COLA	DCA	COLA
SRP Chapter 6	30	6	6	10	10	14	14	0	0
SRP Chapter 7	35	11	12	18	18	6	5	0	0
SRP Chapter 8	14	3	5	5	5	6	4	0	0
SRP Chapter 9	29	10	11	10	10	9	8	0	0
SRP Chapter 10	15	8	8	3	3	4	4	0	0
SRP Chapter 11	8	8	8	0	0	0	0	0	0
SRP Chapter 12	4	1	2	2	2	1	0	0	0
SRP Chapter 13	15	11	12	1	1	3	2	0	0
SRP Chapter 14	14	11	11	2	2	1	1	0	0
SRP Chapter 15	37	20	20	9	9	8	8	0	0
SRP Chapter 16	2	0	0	2	2	0	0	0	0
SRP Chapter 17	6	0	0	1	2	5	4	0	0
SRP Chapter 18	2	2	2	0	0	0	0	0	0
SRP Chapter 19	3	2	2	0	0	1	1	0	0
DC/COL-ISGs	17	11	15	1	0	5	2	0	0
DI&C-ISGs	7	0	0	5	5	2	2	0	0
NSIR/DPR-ISG-01	1	0	1	0	0	1	0	0	0

3.3 Regulatory Guides Other than those that are Sub-tier to the Standard Review Plan

The NuScale regulatory gap analysis included a detailed review of Division 1, 4, 5, and 8 regulatory guides other than those that are sub-tier to the Standard Review Plan. Whereas the applicability of regulatory guides that are sub-tier to the SRP was determined based on the specific context of the SRP section, the overall assessment of the regulatory guides (i.e., as an effort separate from the review of SRP sub-tier guidance documents) considered the applicability of the guidance in its totality to the NuScale application for design certification.

Table 3-3 illustrates the extent of applicability that current NRC regulatory guides were determined to have with respect to the NuScale application for design certification. 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.

Table 3-3. Extent of applicability of regulatory guides (divisions 1, 4, 5, and 8)

Regulatory Guide	Total Regulatory Guides Reviewed	Applicability Determination Result				Further Consideration Needed
		Applicable	Partially Applicable	Not Applicable	NuScale Unique Feature or Req't	
Division 1	223	49	63	111	0	0
Division 4	21	0	1	20	0	0
Division 5	70	1	5	64	0	0
Division 8	38	1	3	34	0	0

Table 3-4. NRC regulations requiring further consideration

<table>
<tr><th></th><th>10 CFR</th><th>Subject</th><th>Summary Basis for Gap Determination; Further Consideration</th></tr>
<tr><td>1.</td><td>50.34(f)(1)(ii)</td><td>Evaluation and Design Review of AFW System</td><td><u>Regulatory Requirement</u>

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 PWR’s 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.”

<u>Summary Basis for Gap Determination</u>

The NuScale plant design does not involve an AFW system as would be found at a typical large LWR. However, as discussed in Section A.2 of this report, the NuScale decay heat removal (DHR) system fulfills a substantively similar function as an AFW system at a large PWR.
The underlying purpose of this requirement — to ensure adequate core cooling in the event of a loss of main feedwater — appears to be relevant to the NuScale design, albeit to a system of a different name. Specifically, a reasonable interpretation of this requirement may be that the specified AFW system evaluation was not intended to exclude other systems designated by any other names but designed to fulfill a substantively similar function. Based on this interpretation, the AFW system evaluation required by 10 CFR 50.34(f)(1)(ii) would be considered applicable to the NuScale DHR system.
NuScale does not believe an exemption is needed from 10 CFR 50.34(f)(1)(ii), because the AFW system evaluation would be considered applicable to the NuScale DHR system. Even if this was not the case, an exemption would be unnecessary because this regulation only applies to the “technically relevant” portions of the Three Mile Island requirements. Because the NuScale design does not include an AFW system, the requirements in 10 CFR 50.34(f)(1)(ii) would not be technically relevant to the NuScale design. 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.

<u>Further Consideration</u>

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 applicable to the NuScale DHR system, and no exemption is needed.</td></tr>
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	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
2.	50.34(f)(2)(iv)	Safety Parameter Display System (SPDS)	<u>Regulatory Requirement</u> 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).” <u>Summary Basis for Gap Determination</u> 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.” Based on 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). <u>Further Consideration</u> 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)(vi)	Reactor Coolant System Venting	<u>Regulatory Requirement</u> 10 CFR 50.34(f)(2)(vi) requires the design certification applicant to, “Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1).” <u>Summary Basis for Gap Determination</u> These requirements are substantively similar to those contained in 10 CFR 50.46a. Further consideration related to 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi) is addressed in Entry No. 7 of Table 3-4, with supporting information in Section A.6 of this report.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			<u>Further Consideration</u> See Entry No. 7 of this Table 3-4 and the supporting information in Section A.6 of this report.
4.	50.34(f)(2)(xii)	Auxiliary Feedwater System Actuation and Flow Indication	<u>Regulatory Requirement</u> 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 PWR’s only) (II.E.1.2).” <u>Summary Basis for Gap Determination</u> The NuScale plant design does not involve an AFW system as would be found at a typical large LWR. However, as discussed in Section A.2 of this report, the NuScale DHR system fulfills a substantively similar function as an AFW system at a large PWR. The underlying purpose of this requirement appears to be relevant to the NuScale design, albeit to a system of a different name. Specifically, a reasonable interpretation of this requirement may be that the specified AFW system requirements were not intended to exclude other systems designated by any other names but designed to fulfill a substantively similar function. Based on this interpretation, the technically relevant portions of the AFW system requirements of 10 CFR 50.34(f)(2)(xii) would be considered applicable to the NuScale DHR system. With regard to the portion of this requirement specifying automatic and manual initiation, the underlying purpose of 10 CFR 50.34(f)(2)(xii) is met by providing the specified automatic and manual initiation for DHR system operation. With regard to the portion of this requirement specifying control room flow indication, the underlying purpose similarly may be met by providing the specified flow indication for DHR system operation. However, pending further design progress, the literal language of the portion of this requirement specifying control room flow indication may be determined to be not technically relevant to the NuScale DHR system design. Specifically, the DHR system operation involves passive natural circulation flow, with flow characteristics that inherently vary with system conditions. For the NuScale design, control room indication for system parameters other than DHR system flow may be determined to be 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. Pending further design progress, it may be determined that provisions for control room indication for these parameters would ensure that the underlying purpose of the portion of 10 CFR 50.34(f)(2)(xii) specifying control room flow indication is satisfied. NuScale does not believe an exemption is needed from 10 CFR 50.34(f)(2)(xii), because the requirements are either satisfied by the NuScale DHR system or, in the case of the specified control room flow indication, may be determined to be not “technically relevant” to the NuScale design. In the former instance, this conclusion appears to be supported by the lack of an exemption from 10 CFR 50.34(f)(2)(xii) for the AP1000 design, which also does not utilize a traditional AFW system. 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 1. the technically relevant AFW system requirements specified in 10 CFR 50.34(f)(2)(xii) are applicable to the NuScale

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			DHR system. 2. for the portion of 10 CFR 50.34(f)(2)(xii) specifying control room flow indication, if design progress determines that 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: a. the literal language of this requirement 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 the literal language of this provision is not required. b. the underlying purpose of this requirement is satisfied by providing control room indication for system parameters other than DHR system flow that are more appropriate for the NuScale design. 3. based on Items 1 and 2 above, no exemption is needed.
5.	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 reactor 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. <u>Further Consideration</u> 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.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
6.	50.44(c)(2)	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." Summary Basis for Gap Determination 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. 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. Specifically, in the NuScale design, containment vessel structural integrity and appropriate accident mitigating features are assured without reliance on an inerted atmosphere or limiting hydrogen concentrations as specified in this requirement. 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 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). Given the plain language of the regulation, a partial exemption appears to be necessary. See also gap analysis results for SRP Section 6.2.5. Further Consideration NuScale intends to pursue a partial exemption from the combustible gas control regulations of 10 CFR 50.44(c)(2). The portion for which exemption will be sought is that requiring either that containment designs have an inerted atmosphere or limit hydrogen concentrations within containment, uniformly distributed, to 10 percent or less.
7.	50.46a	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 noncondensible 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 noncondensible gases could interfere with post-accident natural circulation or pump operation that might inhibit long-term cooling following an accident. As discussed further in Section A.6 of this report, the NuScale reactor module design includes reactor coolant system venting capability that satisfies the literal language of 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi). However, as a result of significant differences in the NuScale advanced reactor design compared to a traditional large LWR, the NuScale

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			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, the NuScale reactor module design is such that there is no reasonable likelihood that an accumulation of noncondensible gases could interfere with post-accident natural circulation or otherwise inhibit long-term cooling following an accident. Although high point venting capability is included in the NuScale design to periodically remove accumulated noncondensible 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). Therefore, as applied to the NuScale design, the safety functions contemplated by 10 CFR 50.46a(c)(1) would include the function related to the vent being an integral part of the reactor coolant pressure boundary (similar to traditional LWR designs), but would not include the function of ensuring long-term post-accident core cooling. In addition, unlike a traditional LWR, the high point vent on the NuScale reactor vessel discharges directly to the radioactive waste management system rather than to the containment vessel. This design introduces a safety function specific to the NuScale vent system that typically would not be relevant for an LWR design that vents to containment: the containment vessel isolation function. The net result of the above-described differences in safety functions is that, whereas at a typical LWR the high point vent valve safety functions include both opening and closing, in the NuScale design only valve closure is relied upon as a safety function. The NuScale reactor pressure vessel vent design is intended to meet the design and operational criteria specified in 10 CFR 50.46a(a), (b), and (c) and 10 CFR 50.34(f)(2)(vi). However, the 10 CFR 50.46a(c)(1) criterion will be applied with consideration for those safety functions relevant to the NuScale vent design, which as discussed above include only the (1) function related to the vent being an integral part of the reactor coolant pressure boundary, and (2) containment isolation function. See also gap analysis results for SRP Section 5.4.12. <u>Further Consideration</u> As part of pre-application activities, NuScale will seek NRC concurrence that the 10 CFR 50.46a(c)(1) criterion will be applied with consideration for those reactor coolant system vent safety functions relevant to the NuScale advanced reactor design, and no exemption is required.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
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." Summary Basis for Gap Determination As detailed in Section A.1 of this report, NuScale 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). See also gap analysis results for SRP Section 13.1.2-13.1.3. Further Consideration Although 10 CFR 50.54 generally imposes conditions on licensees, NuScale believes that an exemption at the design certification stage is appropriate and beneficial given the generic nature of this exemption and the interrelated nature of staffing requirements and NuScale design features. Seeking an exemption also would be consistent with the NRC's discussion of exemptions during the 2007 Part 52 rulemaking in which it stated, "Moreover, 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" (72 FR 49372). Based on the above and consistent with SECY-11-098 (Reference 5.21), NuScale intends to pursue an exemption from the current operator staffing regulations of 10 CFR 50.54(m)(2)(i) and (iii). The exemption request will be based on human factors engineering analysis of NuScale plant-specific human system integration features and a NuScale plant-specific staffing plan developed using the methodology 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	<u>Regulatory Requirement</u> 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." <u>Summary Basis for Gap Determination</u> The portion of this requirement related to automatic initiation of turbine trip under anticipated transient without scram (ATWS) conditions is fully applicable to the NuScale design. However, the NuScale plant design does not involve the type of AFW system that would be found at a typical large LWR. As discussed in Section A.2 of this report, the NuScale DHR system fulfills a function substantively equivalent to the AFW system function contemplated by this requirement. The underlying purpose of this requirement appears to be relevant to the NuScale design, albeit to a system of a different name. Specifically, a reasonable interpretation of this requirement may be that the specified AFW system requirement was not intended to exclude other systems designated by any other names but designed to fulfill a substantively similar function. Based on this interpretation, the AFW system requirement of 10 CFR 50.62(c)(1) would be considered applicable to the NuScale DHR system, in that the underlying purpose of the rule is satisfied by reliance on equipment that is diverse and independent from the reactor trip system to automatically initiate the DHR system under conditions indicative of an ATWS. Therefore, NuScale does not believe an exemption is needed from 10 CFR 50.62(c)(1). See also gap analysis results for SRP Section 10.4.9, Acceptance Criterion II.8, and SRP Section 15.8. <u>Further Consideration</u> As part of pre-application activities, NuScale will seek NRC concurrence that the underlying purpose of the rule is satisfied by reliance on equipment that is diverse and independent from the reactor trip system to automatically initiate the DHR system under conditions indicative of an ATWS, and no exemption is needed.

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
10.	50, App. A, GDC 17	Electric Power Systems	Regulatory Requirement 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." Summary Basis for Gap Determination As discussed further in Section A.3 of this report, the NuScale plant design supports a departure from the portion of GDC 17 requiring two physically independent offsite circuits by providing 1. safety-related passive systems designed to achieve safe shutdown and maintain core cooling and containment integrity, independent of nonsafety-related AC power sources, for an indefinite duration. 2. the Class 1E DC power supply system as the only safety-related power source required to monitor and actuate safety-related passive systems for 72 hours. 3. multiple nonsafety-related onsite and offsite electrical power sources for other functions. Further Consideration 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 further below, NuScale will seek NRC concurrence during pre-application activities regarding the appropriate form of this departure. The introduction section of 10 CFR 50, Appendix A, explicitly represents the GDCs as "guidance" for advanced reactor designs (such as the NuScale design), as opposed to "requirements" as would be the case for typical LWR designs. Accordingly, for the NuScale advanced reactor design, the departure from GDC 17 described above would not require an exemption as contemplated by 10 CFR 52.7 and 10 CFR 50.12. Rather, consistent with the final paragraph of the introduction section of 10 CFR 50, Appendix A, 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. The exemptions are documented for the AP600 and AP1000 designs in 10 CFR 52, Appendix C, Section V.B.6, and Appendix D, Section V.B.3, respectively. 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. Rather, as discussed further in Section A.4 of this report, although other alternatives are under consideration, the current design approach is for this safety function to be performed by the shutdown accumulator system (SAS). The underlying purpose of GDC 27 appears to be relevant to the NuScale design, albeit to a system of a different name. Specifically, a reasonable interpretation of this GDC may be that the specified ECCS poison addition function was not intended to exclude other systems designated by any other names but designed to fulfill a substantively equivalent poison addition function. Based on this interpretation, the ECCS poison addition function specified by GDC 27 would be considered applicable to the NuScale SAS, in that the underlying purpose of the rule is satisfied by the poison addition function of the SAS. Therefore, NuScale does not believe a departure is needed from GDC 27. Further Consideration As part of pre-application activities, NuScale will seek NRC concurrence that the underlying purpose of the rule is satisfied by the poison addition function of the SAS (or other alternatives under consideration), and no departure from GDC 27 is needed. However, even if a departure were determined to be needed, NuScale does not anticipate that this departure would require an exemption under 10 CFR 52.7 and 10 CFR 50.12. Rather, consistent with the final paragraph of the introduction section of 10 CFR 50, Appendix A, the departure from GDC 27 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).

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
12.	50, App. A, GDC 33	Reactor Coolant Makeup	<u>Regulatory Requirement</u> 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." <u>Summary Basis for Gap Determination</u> 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). <u>Further Consideration</u> 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. NuScale does not anticipate that this departure would require an exemption under 10 CFR 52.7 and 10 CFR 50.12. Rather, consistent with the final paragraph of the introduction section of 10 CFR 50, Appendix A, the departure from GDC 33 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).
13.	50, App. A, GDC 40	Testing of Containment Heat Removal System	<u>Regulatory Requirement</u> 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."

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			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 pressure testing specified by GDC 40 would be performed on the containment vessel as part of the overall containment leakage rate testing program. However, the periodic functional and operational testing specified by GDC 40 is not relevant to the NuScale design. Specifically, 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. This testing, in conjunction with the periodic pressure testing that will be performed on the containment vessel as part of the overall containment leakage rate testing program, ensures that the underlying purpose of GDC 40 is achieved. 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. NuScale does not anticipate that this departure would require an exemption under 10 CFR 52.7 and 10 CFR 50.12. Rather, consistent with the final paragraph of the introduction section of 10 CFR 50, Appendix A, the departure from GDC 40 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).
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, the systems specified 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

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			require an engineered safety feature (ESF) atmosphere clean-up system or pressure suppression systems that serve a fission product removal/dose mitigation function. Rather, for the NuScale reactor plant design, fission product control associated with containment design and operational characteristics include 1. the robust design of the NuScale reactor module containment vessel, which ensures its integrity as a fission product barrier under maximum anticipated pressure conditions. 2. the 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). 3. design, inspection, and testing of containment vessel isolation provisions. 4. 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). 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 the underlying purpose of this GDC. <u>Further Consideration</u> 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. Additionally, even if a departure from GDC 41 was necessary, then, as discussed in Section 3.1 of this report, NuScale would not anticipate that this departure would require an exemption under 10 CFR 52.7 and 10 CFR 50.12. Rather, consistent with the final paragraph of the introduction section of 10 CFR 50, Appendix A, the departure from GDC 41 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).

	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, the periodic inspection of important components of such systems specified in GDC 42 is not relevant to the NuScale design, particularly given that GDC 42 only requires "appropriate" inspections. As discussed in the comments above for GDC 41, containment integrity and fission product control are passively assured by the robust containment vessel design, reactor module configuration, and containment vessel isolation provisions and 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 ensure that the NuScale design achieves the underlying purpose of GDC 42. 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 from the inspection requirements in GDC 42. Additionally, even if a departure from GDC 42 was necessary, NuScale does not anticipate that this departure would require an exemption under 10 CFR 52.7 and 10 CFR 50.12. Rather, consistent with the final paragraph of the introduction section of 10 CFR 50, Appendix A, the departure from GDC 42 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).

	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, the periodic pressure and functional testing of such systems specified in GDC 43 is not relevant to the NuScale design, particularly given that GDC 43 only requires "appropriate" testing. As discussed in the comments above for GDC 41, containment integrity and fission product control are passively assured by the robust containment vessel design, reactor module configuration, and containment vessel isolation provisions and 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 ensure that the NuScale design achieves the underlying purpose of GDC 43. 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 from the testing requirements in GDC 43. Additionally, even if a departure from GDC 43 was necessary, NuScale does not anticipate that this departure would require an exemption under 10 CFR 52.7 and 10 CFR 50.12. Rather, consistent with the final paragraph of the introduction section of 10 CFR 50, Appendix A, the departure from GDC 43 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).
17.	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. Thus, the required and technically relevant features of Section I of Appendix K will be applied to the NuScale evaluation

	10 CFR	Subject	Summary Basis for Gap Determination; Further Consideration
			model, rather than the performance criteria of 10 CFR 50.46(b). Due to unique features specific to the NuScale advanced reactor design, portions of Appendix K are not technically relevant to the NuScale design. For example, much of Section I of Appendix K is germane to ECCS evaluations for reactor designs for which a postulated loss-of-coolant accident (LOCA) results in core uncovery. For such designs, recovery from a postulated LOCA involves a core refill phase and a reflood phase. Core refill and reflood are not relevant to the NuScale reactor plant design since postulated design basis accidents would not result in core uncovery. Thus, portions of Appendix K specifying evaluation methods and assumptions for these post-blowdown phases are not applicable to the NuScale design (e.g., §§ I.D.2, 3, 4, and 5). In addition, Appendix K specifies evaluation criteria that are pertinent only for design features found at a typical PWR, such as reactor coolant system piping loops for which cold leg breaks would be postulated/evaluated. The NuScale design does not have reactor coolant piping loops, and thus a cold leg break is not applicable to the NuScale design (e.g., § I.C.1.c). The NuScale design also does not incorporate the use of ECCS pumps; rather, ECCS flow is generated passively via natural circulation. Thus, aspects of Appendix K specifying criteria for pump modeling (e.g., § I.C.6) are not relevant to the NuScale design. <u>Further Consideration</u> Based on the examples provided above, it is clear that departure from the literal language of 10 CFR 50, Appendix K, is necessary to accommodate the NuScale advanced reactor design. The Appendix K criteria may be interpreted as specifying evaluation methodology only when a given condition (e.g., core reflood or cold leg piping break) is present or relevant, but allowing for not considering that given condition in an evaluation when it is not relevant (as is the case with core reflood and cold leg break in the NuScale design). Such an interpretation would obviate the need for an exemption as contemplated by 10 CFR 50.12 and 10 CFR 52.7, since NuScale would simply perform its ECCS evaluation applying only those portions of Appendix K technically relevant to the NuScale design. Given the uncertainty regarding the intent of these requirements, NuScale will seek NRC concurrence during pre-application activities regarding the appropriate form of the Appendix K departure.

Table 3-5. Proposed reconciliation of NUREG-0800 with NuScale Design-Specific Review Standard

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
SRP CHAPTER 1				
1.0	Introduction and Interfaces	Yes — Use As-Is	Yes — Use As-Is	–
SRP CHAPTER 2				
2.0	Site Characteristics and Site Parameters	Yes — Use As-Is	Yes — Use As-Is	–
2.1.1	Site Location and Description	No	Yes — Use As-Is	Identification of site location and description is not applicable for standard design certification reviews.
2.1.2	Exclusion Area Authority and Control	No	Yes — Use As-Is	Exclusion area authority and control information is not applicable for standard design certification reviews.
2.1.3	Population Distribution	No	Yes — Use As-Is	Identification of population distribution is not applicable for standard design certification reviews.
2.2.1-2.2.2	Identification of Potential Hazards in Site Vicinity	No	Yes — Use As-Is	Identification of potential hazards is not applicable for standard design certification reviews.
2.2.3	Evaluation of Potential Accidents	No	Yes — Use As-Is	Evaluation of potential accidents is not applicable for standard design certification reviews.
2.3.1	Regional Climatology	Yes — Use As-Is	Yes — Use As-Is	–
2.3.2	Local Meteorology	Yes — Use As-Is	Yes — Use As-Is	–
2.3.3	Onsite Meteorological Measurements Program	No	Yes — Use As-Is	There are no postulated site parameters for a design certification related to an onsite meteorological program.
2.3.4	Short-Term Atmospheric Dispersion Estimates for Accident Releases	Yes — Use As-Is	Yes — Use As-Is	–
2.3.5	Long-Term Atmospheric Dispersion Estimates for Routine Releases	Yes — Use As-Is	Yes — Use As-Is	–
2.4.1	Hydrologic Description	Yes — Use As-Is	Yes — Use As-Is	–
2.4.2	Floods	Yes — Use As-Is	Yes — Use As-Is	–
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	Yes — Use As-Is	Yes — Use As-Is	–
2.4.4	Potential Dam Failures	Yes — Use As-Is	Yes — Use As-Is	–
2.4.5	Probable Maximum Surge and Seiche Flooding	Yes — Use As-Is	Yes — Use As-Is	–
2.4.6	Probable Maximum Tsunami Hazards	Yes — Use As-Is	Yes — Use As-Is	–
2.4.7	Ice Effects	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
2.4.8	Cooling Water Canals and Reservoirs	Yes — Use As-Is	Yes — Use As-Is	–
2.4.9	Channel Diversions	Yes — Use As-Is	Yes — Use As-Is	–
2.4.10	Flooding Protection Requirements	Yes — Use As-Is	Yes — Use As-Is	–
2.4.11	Low Water Considerations	Yes — Use As-Is	Yes — Use As-Is	–
2.4.12	Groundwater	Yes — Use As-Is	Yes — Use As-Is	–
2.4.13	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	Yes — Use As-Is	Yes — Use As-Is	–
2.4.14	Technical Specifications and Emergency Operation Requirements	Yes — Use As-Is	Yes — Use As-Is	–
2.5.1	Basic Geologic and Seismic Information	No	Yes — Use As-Is	There are no postulated site parameters for a standard design certification related to basic geologic and seismic information.
2.5.2	Vibratory Ground Motion	Yes — Use As-Is	Yes — Use As-Is	–
2.5.3	Surface Faulting	Yes — Use As-Is	Yes — Use As-Is	–
2.5.4	Stability of Subsurface Materials and Foundations	Yes — Use As-Is	Yes — Use As-Is	–
2.5.5	Stability of Slopes	Yes — Use As-Is	Yes — Use As-Is	–
SRP CHAPTER 3				
3.2.1	Seismic Classification	Yes — Use As-Is	Yes — Use As-Is	–
3.2.2	System Quality Group Classification	Yes — Use With Modification	Yes — Use With Modification	This SRP section refers to RG 1.85, which was withdrawn in 2004 because its guidance was updated and incorporated into RG 1.84.
3.3.1	Wind Loadings	Yes — Use With Modification	Yes — Use As-Is	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 — Use With Modification	Yes — Use As-Is	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 — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
3.4.2	Analysis Procedures	Yes — Use With Modification	Yes — Use As-Is	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 — Use With Modification	Yes — Use With Modification	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.
3.5.1.2	Internally-Generated Missiles (Inside Containment)	Yes — Use As-Is	Yes — Use As-Is	–
3.5.1.3	Turbine Missiles	Yes — Use With Modification	Yes — Use With Modification	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 (P2) is less than or equal to 10^{-7} per year per plant. Thus, the probability of unacceptable damage from turbine-generated missiles (i.e., P4) 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 (P1) or SSC damage probability (P3). 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 (i.e., the reactor building) to protect essential SSCs against potential turbine-generated missiles.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
3.5.1.4	Missiles Generated by Tornadoes and Extreme Winds	Yes — Use As-Is	Yes — Use As-Is	–
3.5.1.5	Site Proximity Missiles (Except Aircraft)	Yes — Use With Modification	Yes — Use As-Is	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	Yes — Use With Modification	Yes — Use As-Is	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.
3.5.2	Structures, Systems, and Components to be Protected from Externally-Generated Missiles	Yes — Use With Modification	Yes — Use With Modification	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 — Use With Modification	Yes — Use With Modification	The NuScale design does not include composite or multi-element barriers. This SRP sections sub-tier 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	Yes — Use As-Is	Yes — Use As-Is	–
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	Yes — Use With Modification	Yes — Use With Modification	This SRP section references ANSI/ANS 58.2-1988. This standard was withdrawn in 1998. With consideration for the NRC concerns related to technical adequacy of this standard, it is considered to be not applicable.
3.6.3	Leak-Before-Break Evaluation Procedures	Yes — Use As-Is	Yes — Use As-Is	–

<table>
<tr><th rowspan="2">SRP Section</th><th rowspan="2">Title/Subject</th><th colspan="2">Include as NuScale Design-Specific Review Standard?</th><th rowspan="2">Comments</th></tr>
<tr><th>DC Application</th><th>COL Application</th></tr>
<tr><td>3.7.1</td><td>Seismic Design Parameters</td><td>Yes — Use With Modification</td><td>Yes — Use With Modification</td><td>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) envelops the generic response spectra provided in RG 1.60 anchored at 0.5g ZPA (zero period acceleration), while also broadening the spectra up to 16 Hz.</td></tr>
<tr><td>3.7.2</td><td>Seismic System Analysis</td><td>Yes — Use With Modification</td><td>Yes — Use As-Is</td><td>The NuScale CSDRS envelops the generic response spectra provided in RG 1.60 anchored at 0.5g ZPA (zero period acceleration), while also broadening the spectra up to 16 Hz.

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.

NuScale will not be performing both time history analysis and response spectrum analysis in its analysis of SSCs.</td></tr>
<tr><td>3.7.3</td><td>Seismic Subsystem Analysis</td><td>Yes — Use As-Is</td><td>Yes — Use As-Is</td><td>–</td></tr>
<tr><td>3.7.4</td><td>Seismic Instrumentation</td><td>Yes — Use With Modification</td><td>Yes — Use With Modification</td><td>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.</td></tr>
</table>

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
3.8.1	Concrete Containment	No	No	This SRP section is applicable only to LWRs whose design includes concrete containments or steel or 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).
3.8.2	Steel Containment	Yes — Use With Modification	Yes — Use With Modification	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 is applicable 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 — Use With Modification	Yes — Use With Modification	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.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
3.8.4	Other Seismic Category I Structures	Yes — Use With Modification	Yes — Use With Modification	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 these buildings. Nevertheless, the NuScale application will contain descriptive information of Seismic Category I structures as specified by this guidance. 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. 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.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
3.8.5	Foundations	Yes — Use With Modification	Yes — Use With Modification	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 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. 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.
3.9.1	Special Topics for Mechanical Components	Yes — Use As-Is	Yes — Use As-Is	–
3.9.2	Dynamic Testing and Analysis of Systems, Structures, and Components	Yes — Use As-Is	Yes — Use As-Is	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	Yes — Use As-Is	Yes — Use As-Is	–
3.9.4	Control Rod Drive Systems	Yes — Use With Modification	Yes — Use With Modification	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 — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
3.9.6	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	Yes — Use With Modification	Yes — Use With Modification	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 which 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	Yes — Use As-Is	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	Yes — Use As-Is	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.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	Yes — Use With Modification	Yes — Use With Modification	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 — Use With Modification	Yes — Use With Modification	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	Yes — Use As-Is	Yes — Use As-Is	–
3.13	Threaded Fasteners – ASME Code Class 1, 2, and 3	Yes — Use As-Is	Yes — Use As-Is	–
BTP 3-1	Classification of Main Steam Components Other than the Reactor Coolant Pressure Boundary for BWR Plants	No	No	Applies only to BWRs.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
BTP 3-2	Classification of BWR/6 Main Steam and Feedwater Components Other than the Reactor Coolant Pressure Boundary	No	No	Applies only to BWRs.
BTP 3-3	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	Yes — Use As-Is	Yes — Use As-Is	–
BTP 3-4	Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	Yes — Use As-Is	Yes — Use As-Is	–
SRP CHAPTER 4				
4.2	Fuel System Design	Yes — Use As-Is	Yes — Use As-Is	–
4.3	Nuclear Design	Yes — Use As-Is	Yes — Use As-Is	–
4.4	Thermal and Hydraulic Design	Yes — Use As-Is	Yes — Use As-Is	–
4.5.1	Control Rod Drive Structural Materials	Yes — Use With Modification	Yes — Use With Modification	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 — Use As-Is	Yes — Use As-Is	–
4.6	Functional Design of Control Rod Drive System	Yes — Use As-Is	Yes — Use As-Is	–
BTP 4-1	Westinghouse Constant Axial Offset Control	No	No	NuScale does not intend to use the Constant Axial Offset Control operating scheme.
SRP CHAPTER 5				
5.2.1.1	Compliance with the Codes and Standards Rule, 10 CFR 50.55a	Yes — Use As-Is	Yes — Use As-Is	–
5.2.1.2	Applicable Code Cases	Yes — Use As-Is	Yes — Use As-Is	–
5.2.2	Overpressure Protection	Yes — Use As-Is	Yes — Use As-Is	–
5.2.3	Reactor Coolant Pressure Boundary Materials	Yes — Use As-Is	Yes — Use As-Is	–
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	Yes — Use As-Is	Yes — Use As-Is	–
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	Yes — Use As-Is	Yes — Use As-Is	–
5.3.1	Reactor Vessel Materials	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
5.3.2	Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	Yes — Use As-Is	Yes — Use As-Is	–
5.3.3	Reactor Vessel Integrity	Yes — Use As-Is	Yes — Use As-Is	–
5.4	Reactor Coolant System Component and Subsystem Design	Yes — Use As-Is	Yes — Use As-Is	–
5.4.1.1	Pump Flywheel Integrity (PWR)	No	No	No reactor coolant pumps in NuScale design.
5.4.2.1	Steam Generator Materials	Yes — Use As-Is	Yes — Use As-Is	–
5.4.2.2	Steam Generator Program	Yes — Use As-Is	Yes — Use As-Is	–
5.4.6	Reactor Core Isolation Cooling System (BWR)	No	No	Applies only to BWRs.
5.4.7	Residual Heat Removal (RHR) System	No	No	NuScale systems that fulfill substantively equivalent functions as those served by a typical RHR system are reviewed under other SRP sections, including Sections 10.4.9 and 6.3; see Section A.2 of this report.
5.4.8	Reactor Water Cleanup System (BWR)	No	No	Applies only to BWRs.
5.4.11	Pressurizer Relief Tank	No	No	No pressurizer relief tank in NuScale design.
5.4.12	Reactor Coolant System High Point Vents	No	No	NuScale reactor vents are not relied on to ensure long-term core cooling and do not discharge to the containment vessel; capability of vents to ensure reactor coolant pressure boundary (RCPB) integrity is reviewed under other SRP sections, including SRP Sections 5.2, 3.2, 3.6, 3.7, 3.9, 3.10, 3.11, and 3.12; see Section A.6 of this report.
5.4.13	Isolation Condenser System (BWR)	No	No	Applies only to BWRs.
BTP 5-1	Monitoring of Secondary Side Water Chemistry in PWR Steam Generators	Yes — Use With Modification	Yes — Use With Modification	As discussed in Section A.9 of this report, the secondary chemistry requirements for the NuScale design may differ from those outlined in the specified Electric Power Research Institute (EPRI) and Nuclear Energy Institute (NEI) guidance.
BTP 5-2	Overpressurization Protection of Pressurized-Water Reactors While Operating at Low Temperatures	Yes — Use As-Is	Yes — Use As-Is	–
BTP 5-3	Fracture Toughness Requirements	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
BTP 5-4	Design Requirements of the Residual Heat Removal System	Yes — Use With Modification	Yes — Use With Modification	As discussed in Section A.2.2, 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 — Use As-Is	Yes — Use As-Is	–
6.1.2	Protective Coating Systems (Paints) - Organic Materials	No	No	No coatings in NuScale containment design; see Section A.7 of this report.
6.2.1	Containment Functional Design	Yes — Use As-Is	Yes — Use As-Is	–
6.2.1.1.A	PWR Dry Containments, Including Subatmospheric Containments	Yes — Use With Modification	Yes — Use With Modification	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	No	No ice condenser containment in the NuScale design.
6.2.1.1.C	Pressure-Suppression Type BWR Containments	No	No	Applies only to BWRs.
6.2.1.2	Subcompartment Analysis	No	No	No subcompartments in NuScale containment design; see Section A.7 of this report.
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)	Yes — Use With Modification	Yes — Use With Modification	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 — Use As-Is	Yes — Use As-Is	–
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	No	No	For the NuScale reactor plant design, a LOCA does not result in core uncovery. Therefore, core reflood, including consideration of the effects of containment pressure during core reflood, is not relevant to the evaluation of NuScale ECCS performance capability.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
6.2.2	Containment Heat Removal Systems	Yes — Use With Modification	Yes — Use With Modification	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 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. The periodic pressure testing specified by GDC 40 would be performed on the containment vessel as part of the overall containment leakage rate testing program. However, the periodic functional and operational testing specified by GDC 40 is not relevant to the NuScale design. Additional details regarding GDC 40 applicability are provided in Table 3-4 of this report.
6.2.3	Secondary Containment Functional Design	No	No	No secondary containment in NuScale design.
6.2.4	Containment Isolation System	Yes — Use With Modification	Yes — Use With Modification	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 an open path to the environs.
6.2.5	Combustible Gas Control in Containment	Yes — Use With Modification	Yes — Use With Modification	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).
6.2.6	Containment Leakage Testing	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
6.2.7	Fracture Prevention of Containment Pressure Boundary	Yes — Use With Modification; NuScale-Unique Feature or Requirement	Yes — Use With Modification; NuScale-Unique Feature or Requirement	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 — Use With Modification	Yes — Use With Modification	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 — Use With Modification	Yes — Use With Modification	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	Yes — Use With Modification	Yes — Use With Modification	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	No	No containment spray in the NuScale design; see Section A.7 of this report.
6.5.3	Fission Product Control Systems and Structures	Yes — Use With Modification	Yes — Use With Modification	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.
6.5.4	Ice Condenser as a Fission Product Cleanup System	No	No	No ice condenser containment in the NuScale design.
6.5.5	Pressure Suppression Pool as a Fission Product Cleanup System	No	No	Applies only to BWRs.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
6.6	Inservice Inspection and Testing of Class 2 and 3 Components	Yes — Use As-Is	Yes — Use As-Is	–
6.7	Main Steam Isolation Valve Leakage Control System (BWR)	No	No	Applies only to BWRs.
BTP 6-1	pH for Emergency Coolant Water for Pressurized Water Reactors	Yes — Use As-Is	Yes — Use As-Is	–
BTP 6-2	Minimum Containment Pressure Model for PWR ECCS Performance Evaluation	No	No	For the NuScale reactor plant design, a LOCA does not result in core uncovery. Therefore, core reflood – including consideration of the effects of containment pressure during core reflood – is not relevant to the evaluation of the NuScale ECCS performance capability.
BTP 6-3	Determination of Bypass Leakage Paths in Dual Containment Plants	No	No	No secondary containment in the NuScale design.
BTP 6-4	Containment Purging During Normal Plant Operations	No	No	As discussed in Section A.7 of this report, while the NuScale containment vessel design includes an evacuation system, it serves a different purpose than a purge system, and does not provide an open path to the environs.
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	No	As discussed in Section A.4 of this report, no safety injection pumps and refueling water storage tank in the NuScale ECCS design.
SRP CHAPTER 7				
7.0	Instrumentation and Controls – Overview of Review Process	Yes — Use As-Is	Yes — Use As-Is	–
Appendix 7.0-A	Review Process for Digital Instrumentation and Control Systems	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
7.1	Instrumentation and Controls -- Introduction	Yes — Use With Modification	Yes — Use With Modification	This SRP section is generally applicable to the NuScale design. However, it specifies that an applicant should meet the acceptance criteria and sub-tier guidance described in SRP Sections 7.2 through 7.9, SRP Chapter 7 BTPs, and associated appendices. Some of these acceptance criteria and sub-tier guidance have been determined to be inappropriate to apply to the NuScale design, as detailed in the gap analysis table entries for the individual SRP Sections 7.2 through 7.9, SRP Chapter 7 BTPs, and associated appendices. Certain sub-tier guidance or codes referenced therein are superseded by more current revisions. NuScale intends to apply the current versions.
7.1-T	Table 7-1 Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety	Yes — Use With Modification	Yes — Use With Modification	This SRP section is generally applicable to the NuScale design. However, it specifies that an applicant should meet the acceptance criteria and sub-tier guidance described in Table 7-1. Some of the acceptance criteria and sub-tier guidance have been determined to be inappropriate to apply to the NuScale design, as detailed in the gap analysis table entries for the individual SRP Sections 7.2 through 7.9, SRP Chapter 7 BTPs, and associated appendices.
Appendix 7.1-A	Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety	Yes — Use With Modification	Yes — Use With Modification	This appendix is generally applicable to the NuScale design. However, specific acceptance criteria and sub-tier guidance described in this SRP appendix have been determined to be inappropriate to apply to the NuScale design, as detailed in the gap analysis table entries for the individual SRP Sections 7.2 through 7.9 and SRP Chapter 7 BTPs, and associated appendices.
Appendix 7.1-B	Guidance for Evaluation of Conformance to IEEE Std 279	No	No	This guidance is applicable only to nuclear power plants that are permitted by 10 CFR 50.55(a)(h)(2) to use IEEE Std. 279-1971 in the design of protection systems. Per 10 CFR 50.55(a)(h)(2) and (3), the standards of IEEE Std. 603-1991 –rather than IEEE Std. 279-1971 – are the applicable criteria to be applied to the NuScale design of safety systems.
Appendix 7.1-C	Guidance for Evaluation of Conformance to IEEE Std 603	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
Appendix 7.1-D	Guidance for Evaluation of the Application of IEEE Std 7-4.3.2	Yes — Use With Modification	Yes — Use With Modification	This SRP appendix is applicable except for references to RG 1.152, Rev. 2,including those in Section 9 that cite it as acceptable guidance for providing cyber security protection for digital instrumentation and control (DI&C) systems used in safety-related applications. RG 1.152, Revision 2, has been superseded by Revision 3 of RG 1.152 (July 2011) and RG 5.71 (January 2010). Positions 2.1 through 2.5 of RG 1.152, Rev. 2, were retained with minor changes in Revision 3, while Positions 2.6 through 2.9 were eliminated as they are now addressed in RG 5.71. To the extent that NuScale may address 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 would be considered applicable to the NuScale application for design certification.
7.2	Reactor Trip System	Yes — Use With Modification	Yes — Use With Modification	Several sub-tier guidance or codes referenced therein are superseded by more current revisions. NuScale intends to apply the current versions. Guidance related to protection system trip point changes required for operation with reactor coolant pumps out of service is not applicable, since the NuScale design does not include reactor coolant pumps. Guidance related to programmable logic controller (PLC) systems is not applicable, since the NuScale reactor trip system will be a field programmable gate array system as opposed to a PLC system.
7.3	Engineered Safety Features Systems	Yes — Use With Modification	Yes — Use With Modification	See comment above for SRP Section 7.2. Guidance related to typical AFW systems is not applicable to the NuScale design or would be applied to the NuScale DHR system. Portions of sub-tier guidance are applicable only to LWR ECCS designs that involve actuation and changeover from injection mode to recirculation mode. Operation of the NuScale ECCS does not have separate injection and recirculation modes.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
7.4	Safe Shutdown Systems	Yes — Use With Modification	Yes — Use With Modification	Several sub-tier guidance or codes referenced therein are superseded by more current revisions. NuScale intends to apply the current versions. Guidance related to power operated relief valves and block valves is not applicable, since the NuScale design does not include these types of valves. Guidance related to PLC systems is not applicable, since the safety control and information system will be a field programmable gate array system as opposed to a PLC system.
7.5	Information Systems Important to Safety	Yes — Use With Modification	Yes — Use With Modification	Several sub-tier guidance or codes referenced therein are superseded by more current revisions. NuScale intends to apply the current versions. Guidance related to typical AFW systems is not applicable to the NuScale design or would be applied to the NuScale DHR system. Guidance related to power operated relief valves and block valves is not applicable, since the NuScale design does not include these types of valves. Guidance related to PLC systems is not applicable, since the safety control and information system will be a field programmable gate array system as opposed to a PLC system.
7.6	Interlock Systems Important to Safety	Yes — Use With Modification	Yes — Use With Modification	Several sub-tier guidance or codes referenced therein are superseded by more current revisions. NuScale intends to apply the current versions. Certain sub-tier guidance is applicable only to PWR designs that include ECCS safety injection tanks (or equivalent) with motor-operated valves between the tanks and the reactor coolant system. As discussed further in Appendix A, Section A.4, of this report, the NuScale reactor design differs from that of large PWRs in that 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 do not involve motor-operated valves. Guidance related to PLC systems is not applicable, since the safety control and information system will be a field programmable gate array system as opposed to a PLC system.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
7.7	Control Systems	Yes — Use With Modification	Yes — Use With Modification	Several sub-tier guidance or codes referenced therein are superseded by more current revisions. NuScale intends to apply the current versions. Guidance related to PLC systems is not applicable, since the safety control and information system will be a field programmable gate array system as opposed to a PLC system.
7.8	Diverse Instrumentation and Control Systems	Yes — Use With Modification	Yes — Use With Modification	Several sub-tier guidance or codes referenced therein are superseded by more current revisions. NuScale intends to apply the current versions. Guidance related to PLC systems is not applicable, since NuScale does not intend to use programmable logic controllers in the diverse actuation system.
7.9	Data Communication Systems	Yes — Use With Modification	Yes — Use With Modification	Several sub-tier guidance or codes referenced therein are superseded by more current revisions. NuScale intends to apply the current versions. Guidance related to PLC systems is not applicable, since NuScale does not intend to use programmable logic controllers in the safety control and information system.
Appendix 7-A	General Agenda, Station Site Visits	No	Yes — Use As-Is	This appendix governs NRC visits to plant sites as part of licensing reviews during the operating or combined license stage.
Appendix 7-B	Acronyms, Abbreviations, and Glossary	Yes — Use As-Is	Yes — Use As-Is	–
BTP 7-1	Guidance on Isolation of Low-Pressure Systems from the High-Pressure Reactor Coolant System	Yes — Use With Modification	Yes — Use With Modification	NuScale does not intend to use motor-operated valves in the design of interfaces between low-pressure systems and the high-pressure reactor coolant system. The issue addressed by sub-tier GL 87-12 and GL 88-17 is not germane to the NuScale design. These generic communications were related to concerns over loss of decay heat removal during “mid-loop” operation. Mid-loop operation is not relevant to the NuScale design.
BTP 7-2	Guidance on Requirements of Motor-Operated Valves in the Emergency Core Cooling System Accumulator Lines	No	No	This guidance is applicable only to PWR designs that include ECCS safety injection tanks (or equivalent, a.k.a., accumulators or flooding tanks) with motor-operated valves between the tanks and the reactor coolant system. As discussed further in Appendix A, Section A.4, of the NuScale

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
				gap analysis report, the NuScale reactor design differs from that of large PWRs in that 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 do not involve motor-operated valves. The NuScale design uses the shutdown accumulator system (SAS) – a system separate from the ECCS system – to provide emergency boration to the reactor coolant system during an accident when reactivity control is necessary. The SAS actuates passively via check valves, and thus does not use motor operated valves or require actuation signals.
BTP 7-3	Guidance on Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service	No	No	This guidance is applicable only to PWR designs that use reactor coolant pumps, such that protection system trip point changes would be required for operation with reactor coolant pumps out of service. The NuScale design does not include reactor coolant pumps.
BTP 7-4	Guidance on Design Criteria for Auxiliary Feedwater Systems	Yes — Use With Modification	Yes — Use With Modification	As discussed in Section A.2.1, of this report, the NuScale design does not use an AFW system as contemplated by this guidance. However, the intent of BTP 7-4 is applicable to the NuScale DHR system, which fulfills a substantively similar function as that served by a typical AFW system.
BTP 7-5	Guidance on Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors	Yes — Use As-Is	Yes — Use As-Is	–
BTP 7-6	Guidance on Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode	No	No	This guidance is applicable only to LWR ECCS designs that involve actuation and changeover from injection mode to recirculation mode. As discussed further in Appendix A, Section A.4, of this report, the NuScale reactor design differs from that of large LWRs in that the NuScale ECCS design does not use safety injection tanks (or equivalent) or have separate injection and recirculation modes in response to a design basis accident. The NuScale design uses the SAS – a system separate from the ECCS system – to provide emergency boration to the reactor coolant system during an accident when reactivity control is necessary. The SAS actuates passively via check valves, and thus does not require actuation signals.
BTP 7-8	Guidance for Application of Regulatory Guide 1.22	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
BTP 7-9	Guidance on Requirements for Reactor Protection System Anticipatory Trips	Yes — Use As-Is	Yes — Use As-Is	–
BTP 7-10	Guidance on Application of Regulatory Guide 1.97	Yes — Use As-Is	Yes — Use As-Is	–
BTP 7-11	Guidance on Application and Qualification of Isolation Devices	Yes — Use With Modification	Yes — Use With Modification	This BTP refers to Revision 2 of RG 1.152. NuScale intends to use the current Revision 3 of RG 1.152. This BTP refers to ANSI Std. C84.1-1989, which has been withdrawn.
BTP 7-12	Guidance on Establishing and Maintaining Instrument Setpoints	Yes — Use As-Is	Yes — Use As-Is	–
BTP 7-13	Guidance on Cross-Calibration of Protection System Resistance Temperature Detectors	Yes — Use As-Is	Yes — Use As-Is	–
BTP 7-14	Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems	Yes — Use With Modification	Yes — Use With Modification	This BTP refers to RG 1.28, Revision 3, which endorses (with modifications) ANSI/ASME NQA-1-1983 with ANSI/ASME NQA-1a-1983 Addenda. NuScale intends to use the current Revision 4 (unless superseded by a newer revision) of RG 1.28 dated June 2010, which endorses ANSI/ASME NQA-1-2008 with NQA-1a-2009 addenda. This BTP refers to Revision 2 of RG 1.152. NuScale intends to use the current Revision 3 of RG 1.152. This BTP refers to IEEE Std. 1028-1988, which has been superseded by IEEE Std. 1028-1997 (endorsed by NRC RG 1.168, Revision 1). NuScale intends to apply the current endorsed IEEE Std. 1028-1997. This BTP refers to the 1998 version of IEEE Std. 1058, which is not endorsed by the NRC. NuScale intends to use the endorsed IEEE Std. 1058.1-1987.
BTP 7-17	Guidance on Self-Test and Surveillance Test Provisions	Yes — Use With Modification	Yes — Use With Modification	The monitoring memory and memory reference integrity self-tests are not applicable to the software logic-based reactor protection system.
BTP 7-18	Guidance on the Use of Programmable Logic Controllers in Digital Computer-Based Instrumentation and Control Systems	No	No	NuScale does not intend to use programmable logic controllers in the safety control and information system design.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
BTP 7-19	Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems	Yes — Use As-Is	Yes — Use As-Is	–
BTP 7-21	Guidance on Digital Computer Real-Time Performance	Yes — Use With Modification	Yes — Use With Modification	This BTP refers to Revision 2 of RG 1.152. NuScale intends to use the current Revision 3 of RG 1.152. This BTP refers to RG 1.168, Revision 1. RG 1.168 refers to Revision 1 of RG 1.152 as containing NRC endorsement of IEEE Std. 7-4.3.2-1993. The current Revision 3 of RG 1.152 endorses (with exceptions) IEEE Std. 7-4.3.2-2003. As indicated above, NuScale intends to apply RG 1.152, Revision 3 (unless superseded by a newer revision), and the 2003 version of IEEE Std. 7-4.3.2 that it endorses.
SRP CHAPTER 8				
8.1	Electric Power – Introduction	Yes — Use As-Is	Yes — Use As-Is	–
8.2	Offsite Power System	Yes — Use With Modification	Yes — Use With Modification	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.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
8.3.1	AC Power Systems (Onsite)	Yes — Use With Modification	Yes — Use With Modification	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 nonsafety-related AC power sources, for an indefinite duration. The Class 1E DC power supply system is the only safety-related power source required to actuate the safety-related passive systems. Sufficient battery capacity is available to provide electrical power for other plant safety 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.
8.3.2	DC Power Systems (Onsite)	Yes — Use With Modification	Yes — Use With Modification	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. Specifically, the NuScale DC power system design ensures that in the event of a loss of offsite power, 1. sufficient capacity is provided to energize important-to-safety equipment to attain a safe and orderly shutdown of all units in the event of a worst-case design basis event and a single failure. 2. single failure (including a false or spurious accident signal at the system level in one unit) will not preclude the capability to automatically supply minimum ESF loads in any one unit and safely shut down the remaining units. **3.** there is no interconnection between each unit’s ESF power and control circuits, which ensures that with any combination of maintenance and test operations, power is automatically supplied to minimum ESF loads in any unit.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
8.4	Station Blackout	Yes — Use With Modification	Yes — Use With Modification	The NuScale plant design meets the intent of this guidance largely in its passive design and associated reduced reliance on AC 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	Yes — Use As-Is	This appendix governs NRC visits to plant sites as part of licensing reviews during the operating or combined license stage.
BTP 8-1	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	No	No	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. The NuScale design uses the SAS – a system separate from the ECCS system – to provide emergency boration to the reactor coolant system during an accident when reactivity control is necessary. The SAS actuates passively via check valves, and thus does not use motor operated valves or require actuation signals.
BTP 8-2	Use of Diesel Generator Sets for Peaking	Yes — Use With Modification	Yes — Use With Modification	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 standby diesel generators provide a defense-in-depth function such that consideration of GDC 17 is appropriate. Therefore, this guidance will be considered applicable to the NuScale standby diesel generators, i.e., the standby diesel generators will not be used for peaking service.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
BTP 8-3	Stability of Offsite Power Systems	No	Yes — Use As-Is	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	Yes — Use As-Is	Yes — Use As-Is	–
BTP 8-5	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	Yes — Use As-Is	Yes — Use As-Is	–
BTP 8-6	Adequacy of Station Electric Distribution System Voltages	No	No	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	No	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 standby 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.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
BTP 8-8	Onsite (Emergency Diesel Generators) and Offsite Power Sources Allowed Outage Time Extensions	No	No	As discussed in Section A.3 of this report, with its reduced reliance on AC 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	Yes — Use As-Is	Yes — Use As-Is	–
9.1.2	New and Spent Fuel Storage	Yes — Use As-Is	Yes — Use As-Is	–
9.1.3	Spent Fuel Pool Cooling and Cleanup System	Yes — Use With Modification	Yes — Use With Modification	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)	Yes — Use As-Is	Yes — Use As-Is	–
9.1.5	Overhead Heavy Load Handling Systems	Yes — Use As-Is	Yes — Use As-Is	–
9.2.1	Station Service Water System	No	No	The NuScale design neither requires nor uses a service water system or other system that serves an equivalent function.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
9.2.2	Reactor Auxiliary Cooling Water Systems	Yes — Use With Modification	Yes — Use With Modification	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 — Use As-Is	Yes — Use As-Is	–
9.2.5	Ultimate Heat Sink	Yes — Use As-Is	Yes — Use As-Is	–
9.2.6	Condensate Storage Facilities	Yes — Use With Modification	Yes — Use With Modification	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. Thus, the only portion of this SRP section applicable to the NuScale design is that implementing GDC 60 regarding control of radioactive releases.
9.3.1	Compressed Air System	Yes — Use As-Is	Yes — Use As-Is	–
9.3.2	Process and Post-Accident Sampling Systems	Yes — Use As-Is	Yes — Use As-Is	–
9.3.3	Equipment and Floor Drainage System	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
9.3.4	Chemical and Volume Control (CVC) System (PWR) (including Boron Recovery System)	Yes — Use With Modification	Yes — Use With Modification	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 3-4 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	No	Applies only to BWRs.
9.4.1	Control Room Area Ventilation System	Yes — Use With Modification	Yes — Use With Modification	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.
9.4.2	Spent Fuel Pool Area Ventilation System	Yes — Use With Modification	Yes — Use With Modification	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	Auxiliary and Radwaste Area Ventilation System	Yes — Use With Modification	Yes — Use With Modification	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.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
9.4.4	Turbine Area Ventilation System	No	No	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.
9.4.5	Engineered Safety Feature Ventilation System	Yes — Use With Modification	Yes — Use With Modification	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.
9.5.1.1	Fire Protection Program	Yes — Use With Modification	Yes — Use With Modification	NuScale intends to apply the current Revision 2 of RG 1.189 (unless superseded by a new revision) rather than Revision 1 cited in SRP Section 9.5.1.1. 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	Yes — Use As-Is	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).

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
9.5.2	Communications Systems	Yes — Use With Modification	Yes — Use With Modification	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 — Use As-Is	Yes — Use As-Is	–
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System	No	No	As discussed in Section A.3 of this report, nonsafety-related AC power is not relied upon for the performance of NuScale plant safety functions; thus, there are no safety-related emergency diesel generators in NuScale design.
9.5.5	Emergency Diesel Engine Cooling Water System	No	No	See comment above for SRP Section 9.5.4.
9.5.6	Emergency Diesel Engine Starting System	No	No	See comment above for SRP Section 9.5.4.
9.5.7	Emergency Diesel Engine Lubrication System	No	No	See comment above for SRP Section 9.5.4.
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System	No	No	See comment above for SRP Section 9.5.4.
SRP CHAPTER 10				
10.2	Turbine Generator	No	No	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

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
				of this approach is reviewed under SRP Sections 3.5.1.3 and 3.5.3.
10.2.3	Turbine Rotor Integrity	No	No	See comment above for SRP Section 10.2.
10.3	Main Steam Supply System	Yes — Use As-Is	Yes — Use As-Is	–
10.3.6	Steam and Feedwater System Materials	Yes — Use As-Is	Yes — Use As-Is	–
10.4.1	Main Condensers	Yes — Use As-Is	Yes — Use As-Is	–
10.4.2	Main Condenser Evacuation System	Yes — Use As-Is	Yes — Use As-Is	–
10.4.3	Turbine Gland Sealing System	Yes — Use As-Is	Yes — Use As-Is	–
10.4.4	Turbine Bypass System	Yes — Use As-Is	Yes — Use As-Is	–
10.4.5	Circulating Water System	Yes — Use As-Is	Yes — Use As-Is	–
10.4.6	Condensate Cleanup System	Yes — Use With Modification	Yes — Use With Modification	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 — Use As-Is	Yes — Use As-Is	–
10.4.8	Steam Generator Blowdown System (PWR)	No	No	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)	Yes — Use With Modification	Yes — Use With Modification	The NuScale DHR system fulfills a similar function as the AFW system at a large PWR, and thus the intent 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.
BTP 10-1	Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants	No	No	The NuScale DHR system (equivalent function to AFW system) is a passive system and does not use pumps.
BTP 10-2	Design Guidelines for Avoiding Water Hammers in Steam Generators	Yes — Use With Modification	Yes — Use With Modification	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 — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
11.2	Liquid Waste Management System	Yes — Use As-Is	Yes — Use As-Is	–
11.3	Gaseous Waste Management System	Yes — Use As-Is	Yes — Use As-Is	–
11.4	Solid Waste Management System	Yes — Use As-Is	Yes — Use As-Is	–
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	Yes — Use As-Is	Yes — Use As-Is	–
BTP 11-3	Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	Yes — Use As-Is	Yes — Use As-Is	–
BTP 11-5	Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure	Yes — Use As-Is	Yes — Use As-Is	–
BTP 11-6	Postulated Radioactive Releases Due to Liquid-containing Tank Failures	Yes — Use As-Is	Yes — Use As-Is	–
SRP CHAPTER 12				
12.1	Assuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)	Yes — Use As-Is	Yes — Use As-Is	–
12.2	Radiation Sources	Yes — Use With Modification	Yes — Use With Modification	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 — Use With Modification	Yes — Use With Modification	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. This SRP section refers to RG 1.21, Rev. 1. NuScale intends to apply the current Revision 2 of RG 1.21 (unless superseded). 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 technically relevant to the NuScale standard plant design, are applicable to the NuScale application for design certification.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
12.5	Operational Radiation Protection Program	No	Yes — Use As-Is	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	Yes — Use As-Is	Yes — Use As-Is	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	Yes — Use With Modification	Yes — Use With Modification	As discussed further in Section A.1 of this report, it is more appropriate that the operating organization for the NuScale reactor 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 Section 13.1.2-13.1.3 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.
13.2.1	Reactor Operator Requalification Program; Reactor Operator Training	Yes — Use As-Is	Yes — Use As-Is	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	Yes — Use As-Is	Yes — Use As-Is	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	Yes — Use As-Is	Yes — Use As-Is	–
13.4	Operational Programs	Yes — Use As-Is	Yes — Use As-Is	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	Yes — Use As-Is	Yes — Use As-Is	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	Yes — Use As-Is	Yes — Use As-Is	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.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
13.5.2.1	Operating and Emergency Operating Procedures	Yes — Use As-Is	Yes — Use As-Is	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 and Other Operating Procedures	Yes — Use As-Is	Yes — Use As-Is	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	Yes — Use As-Is	Yes — Use As-Is	–
13.6.1	Physical Security – Combined License and Operating Reactors	No	Yes — Use As-Is	Applies only to combined license applicants and applicants for and holders of operating licenses.
13.6.2	Physical Security – Design Certification	Yes — Use As-Is	No	Applies only to applicants for design certification.
13.6.3	Physical Security – Early Site Permit	No	No	Applies only to applicants for an early site permit.
13.6.6	Cyber Security Plan	No	Yes — Use As-Is	Applies only to combined license applicants and applicants for and holders of operating licenses.
SRP CHAPTER 14				
14.2	Initial Plant Test Program – Design Certification and New License Applicants	Yes — Use As-Is	Yes — Use As-Is	–
14.2.1	Generic Guidelines for Extended Power Uprate Testing Programs	No	No	Applies only to extended power uprate license amendment requests.
14.3	Inspections, Tests, Analyses, and Acceptance Criteria	Yes — Use As-Is	Yes — Use As-Is	–
14.3.2	Structural and Systems Engineering – Inspections, Tests, Analyses, and Acceptance Criteria	Yes — Use As-Is	Yes — Use As-Is	–
14.3.3	Piping Systems and Components – Inspections, Tests, Analyses, and Acceptance Criteria	Yes — Use As-Is	Yes — Use As-Is	–
14.3.4	Reactor Systems – Inspections, Tests, Analyses, and Acceptance Criteria	Yes — Use As-Is	Yes — Use As-Is	–
14.3.5	Instrumentation and Controls – Inspections, Tests, Analyses, and Acceptance Criteria	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
14.3.6	Electrical Systems – Inspections, Tests, Analyses, and Acceptance Criteria	Yes — Use With Modification	Yes — Use With Modification	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 — Use As-Is	Yes — Use As-Is	–
14.3.8	Radiation Protection – Inspections, Tests, Analyses, and Acceptance Criteria	Yes — Use As-Is	Yes — Use As-Is	–
14.3.9	Human Factors Engineering – Inspections, Tests, Analyses, and Acceptance Criteria	Yes — Use As-Is	Yes — Use As-Is	–
14.3.10	Emergency Planning – Inspections, Tests, Analyses, and Acceptance Criteria	Yes — Use With Modification	Yes — Use With Modification	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 reactor plant rather than on the EP staffing provisions of NUREG-0654/FEMA-REP-1, Revision 1.
14.3.11	Containment Systems – Inspections, Tests, Analyses, and Acceptance Criteria	Yes — Use As-Is	Yes — Use As-Is	–
14.3.12	Physical Security Hardware – Inspections, Tests, Analyses, and Acceptance Criteria	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
SRP CHAPTER 15				
15.0	Introduction – Transient and Accident Analyses	Yes — Use As-Is	Yes — Use As-Is	–
15.0.1	Radiological Consequence Analyses Using Alternative Source Terms	Yes — Use With Modification	Yes — Use With Modification	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 which do not result in core damage.
15.0.2	Review of Transient and Accident Analysis Methods	Yes — Use As-Is	Yes — Use As-Is	–
15.0.3	Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors	Yes — Use With Modification	Yes — Use With Modification	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 — Use As-Is	Yes — Use As-Is	–
15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)	Yes — Use With Modification	Yes — Use With Modification	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.
15.1.5.A	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	Yes — Use As-Is	Yes — Use As-Is	–
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 — Use As-Is	Yes — Use As-Is	–
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
15.2.7	Loss of Normal Feedwater Flow	Yes — Use As-Is	Yes — Use As-Is	–
15.2.8	Feedwater System Pipe Break Inside and Outside Containment (PWR)	Yes — Use With Modification	Yes — Use With Modification	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	No	The NuScale design does not involve forced reactor coolant flow and the requisite pumps that would provide the motive force.
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	No	No	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 — Use As-Is	Yes — Use As-Is	–
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	Yes — Use As-Is	Yes — Use As-Is	–
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	Yes — Use As-Is	Yes — Use As-Is	–
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 — Use With Modification	Yes — Use With Modification	The NuScale design does not use forced reactor coolant flow and 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. Notwithstanding, 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 reactor 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 (PWR)	Yes — Use As-Is	Yes — Use As-Is	–
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Yes — Use As-Is	Yes — Use As-Is	–
15.4.8	Spectrum of Rod Ejection Accidents (PWR)	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
15.4.8.A	Radiological Consequences of a Control Rod Ejection Accident (PWR)	Yes — Use As-Is	Yes — Use As-Is	–
15.4.9	Spectrum of Rod Drop Accidents (BWR)	No	No	Applies only to BWRs.
15.4.9.A	Radiological Consequences of Control Rod Drop Accident (BWR)	No	No	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 — Use As-Is	Yes — Use As-Is	–
15.6.1	Inadvertent Opening of a PWR Pressurizer Relief Valve or a BWR Pressure Relief Valve	No	No	The NuScale design does not use power-operated relief valves (PORVs), which have the potential to open inadvertently. Rather, the NuScale design uses spring-loaded ASME code safety relief valves, which do not have the PORV's vulnerability to inadvertent operation.
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	Yes — Use As-Is	Yes — Use As-Is	–
15.6.3	Radiological Consequences of Steam Generator Tube Failure (PWR)	Yes — Use As-Is	Yes — Use As-Is	–
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	No	No	Applies only to BWRs.
15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Yes — Use With Modification	Yes — Use With Modification	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 — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
15.6.5.B	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage From Engineered Safety Feature Components Outside Containment	Yes — Use As-Is	Yes — Use As-Is	–
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	No	Applies only to BWRs.
15.7.3	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	No	No	The technical content of this SRP section has been moved to BTP 11-6.
15.7.4	Radiological Consequences of Fuel Handling Accidents	Yes — Use With Modification	Yes — Use With Modification	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 — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
15.8	Anticipated Transients Without Scram	Yes — Use With Modification	Yes — Use With Modification	As discussed in Section A.2 of this report, the NuScale design does not include an AFW system. However, the NuScale DHR system fulfills a substantively equivalent function, and will be the system that is automatically actuated by diverse equipment as specified. 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.
15.9	Boiling Water Reactor Stability	Yes — Use With Modification	Yes — Use With Modification	The intent of this guidance is applicable but the language is specific to BWR core designs. Specifically, there may be AOOs for the NuScale reactor 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. Thus, NuScale plans to develop its own frequency-domain linear stability analysis code and evaluate potential stability issues.
SRP CHAPTER 16				
16.0	Technical Specifications	Yes – Use With Modification	Yes – Use With Modification	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 reactor 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 – Use With Modification	Yes – Use With Modification	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 reactor plant design. The extent of these differences is such that NUREG/CR-6595 is inappropriate to apply to the NuScale design.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
SRP CHAPTER 17				
17.1	Quality Assurance During the Design and Construction Phases	No	No	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	No	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	No	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	No	Superseded by DC/COL-ISG-18.
17.5	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	Yes — Use With Modification	Yes — Use With Modification	Certain acceptance criteria of this SRP section are related to a reactor 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, 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.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
17.6	Maintenance Rule	No	Yes — Use As-Is	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	Yes — Use As-Is	Yes — Use As-Is	–
Appendix 18-A	Crediting Manual Operator Actions in Diversity and Defense-in-Depth (D3) Analyses	Yes — Use As-Is	Yes — Use As-Is	–
SRP CHAPTER 19				
19.0	Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors	Yes — Use As-Is	Yes — Use As-Is	–
19.1	Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities	Yes — Use As-Is	Yes — Use As-Is	–
19.2	Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance	No	No	Applies only to existing reactor licensees' requests for license amendments under 10 CFR 50.90 and exemptions under 10 CFR 50.11.
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.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
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	–
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.
ESP/DC/COL-ISG-15	Post-Combined License Commitments	Yes — Use As-Is	Yes — Use As-Is	–

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
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”	Yes — Use As-Is	Yes — Use As-Is	–
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	–
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 standby 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.

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
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 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 BTP7-19, Revision 5.
DI&C-	Review of New Reactor Digital	Yes — Use With	Yes — Use With	This ISG refers to data analysis standards of ASME/ANS

SRP Section	Title/Subject	Include as NuScale Design-Specific Review Standard?		Comments
		DC Application	COL Application	
ISG-03	Instrumentation and Control Probabilistic Risk Assessments	Modification	Modification	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.
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.

4.0 Conclusion

The completion of the NuScale regulatory gap analysis represents a major milestone in the pre-application phase of the NuScale design certification effort. 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 the

- applicability of the regulatory framework as assessed in this gap analysis.
- disposition of “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 revision to this report.

NuScale believes that the regulatory gap analysis results, with revisions to reflect the NRC’s final determinations on applicability and gap dispositions, represent necessary information for development of a design-specific review standard to be used by the NRC in its review of the NuScale application for design certification. A number of SRP acceptance criteria were found to be irrelevant, either in whole or in part, to the NuScale reactor plant design. In these instances, appropriate modifications to the affected SRP sections, or their replacement with new design-specific sections, will be significant activities in the NRC’s development of a NuScale design-specific review standard. NuScale remains committed to assisting the NRC as necessary and appropriate to facilitate this effort.

NuScale intends to use the results of this gap analysis, with any revisions as discussed above, to prepare a proposed content outline for the DCD to be submitted pursuant to 10 CFR 52.47(a) as part of the NuScale application for design certification. The content outline will have particular focus on those DCD sections that due to unique NuScale design features are anticipated to differ significantly from what would be provided for a typical LWR application. NuScale anticipates providing the proposed content outline to the NRC in the fourth quarter of 2012. The content outline is hoped to facilitate alignment between NuScale and NRC on planned content and structure of the DCD and the NRC’s development of the NuScale design-specific review standard as discussed above.

Finally, it is emphasized that 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 reactor plant design. As the ongoing engineering design effort progresses in support of the NuScale application for design certification, the relevance of all or 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,” (Draft Revision 3), NUREG-0800, (ADAMS Accession No. ML110110701).

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.

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. SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” 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 (EPRI), "Pressurized Water Reactor Secondary Water Chemistry Guidelines."

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

5.21. SECY-11-0098, "Operator Staffing for Small or Multi-Module Nuclear Power Plant Facilities," 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 reactor plant design features that led to certain regulations and regulatory guidance being identified as irrelevant – in whole or in part – to the NuScale application for design certification.[6] Such instances may warrant modifications to the regulatory framework that will be applied to the NuScale application, as discussed in Table 3-4 and Table 3-5 of this report.

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 such, the current regulations incorporate 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 reactor plant and that of a typical large LWR. These differences 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. Applying the 10 CFR 50.54(m)(2)(i) assumption for a maximum of three units and three control rooms would not 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 reactor 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 reactor 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 reactor 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).

6 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.

Based on the above and consistent with SECY-11-098 (Reference 5.21), NuScale intends to pursue an exemption from the current operator staffing regulations of 10 CFR 50.54(m)(2)(i) and (iii). The exemption request will be based on human factors engineering analysis of NuScale plant-specific human system integration features and a NuScale plant-specific staffing plan developed using the methodology provided in NUREG-0711 and NUREG-1791.

A.2 NuScale Decay Heat Removal (DHR) System Fulfilling Functions of Typical Auxiliary Feedwater and Residual Heat Removal Systems

Existing NRC regulations and regulatory guidance specify requirements and criteria, respectively, for design of the AFW system and RHR system. These regulations and guidance were established based on large LWR designs. Due to design features unique to the NuScale design, in some instances the existing LWR-based regulatory framework is inappropriate or irrelevant to apply to the NuScale design. Specifically, the NuScale plant design does not involve an AFW system or RHR system as would be found at a typical large LWR. However, the NuScale design does include systems that fulfill safety functions substantively equivalent or similar to those performed by a typical AFW and RHR system. In some instances, this similarity in safety functions may allow for application of the existing LWR-based regulatory framework to the NuScale systems that fulfill these safety functions. However, as identified in Table 3-4 and Table 3-5 of this report, other instances may warrant further consideration (e.g., exemption to or reinterpretation of regulations or revision of SRP sections to be used as design-specific review standards).

A.2.1 Auxiliary Feedwater System Safety Functions

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. Design and operation of the typical AFW system involves pumps powered by electrical and 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. As such, the DHR system fulfills a substantively similar function as the AFW system at a large PWR. However, 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, 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.

Even with consideration for the significant design differences between the NuScale DHR system and a typical AFW system, the similarity in safety function between the two allows for applying to the NuScale DHR system some of the regulations and guidance directed towards AFW systems. However, in some instances, such application of the regulatory framework may require further consideration. For example, 10 CFR 50.62(c)(1) requires 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 condition. The NuScale design provides for diverse automatic actuation of the DHR system and turbine trip. Since the DHR system serves a similar function as a typical AFW system, the NuScale design's reliance on the DHR system in an ATWS condition meets the underlying purpose of the rule. However, as the NuScale design does not involve an auxiliary (or emergency) feedwater system as was contemplated in the development of 10 CFR 50.62(c)(1), further consideration is warranted to determine whether regulatory exemption (full or partial) or

reinterpretation is needed. This and other potential modifications to the regulatory framework that will be applied to the NuScale DHR system design – specifically with respect to its safety function similar to that of a typical AFW system – are discussed in Table 3-4 and Table 3-5 of this report.

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 3-5 of this report.

A.2.2 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.2.2.1 Safety Function Related to RCS Cooling

With respect to the function of providing RCS cooling during and following shutdown, the NuScale main feedwater system and the DHR system are 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. However, as discussed in the NuScale Design Overview, the DHR system is a passive, closed loop system that, unlike a typical RHR system, has no direct interface with the reactor coolant system (i.e., primary side). As such, the design and operation of the NuScale DHR system is significantly different than that of a typical RHR system. Nevertheless, even with consideration for the significant design differences between the NuScale DHR system and a typical RHR system, the similarity in the core cooling safety function between the two may allow for applying to the NuScale DHR system select portions of the guidance directed towards RHR systems. Table 3-5 of this report identifies those instances where it was determined that application of existing LWR-based RHR system guidance to the review of the NuScale DHR system design may require further consideration (e.g., elimination of SRP sections, or revised or new SRP sections to be used as design-specific review standards)..

A.2.2.2 Safety Function Related to Providing Low-Pressure Emergency Core Cooling and Containment Heat Removal

As discussed in Section A.2.2, a typical RHR system may also act to provide low-pressure emergency core cooling and containment heat removal capability. Unlike the RHR system at a large PWR, the NuScale DHR system is not part of the ECCS or containment heat removal system. 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.

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 reactor plant translates to a strong coping capability without

reliance on alternating current (AC) power. Although the NuScale plant is designed with reliable nonsafety-related offsite and onsite AC power systems that are the preferred source of electrical power for important plant functions, the NuScale plant is designed to achieve safe shutdown and maintain core cooling and containment integrity, independent of nonsafety-related AC power sources, for an indefinite duration. In the event of failure of the preferred AC electrical power supply, a safety-related (i.e., Class 1E) direct current (DC) electrical power supply system provides the necessary Class 1E AC power through inverters to ensure continuous operation of safety-related plant systems and components. As such, the Class 1E DC power supply system is the only safety-related power source required to actuate and monitor the safety-related passive systems. Sufficient battery capacity is available to provide electrical power for other plant safety 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 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 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 capable of powering at least one complete set of normal shutdown loads in the event of a 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 reactor 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 a SBO, and (b) meet the NRC guidelines for the regulatory treatment of non-safety 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, AC power is supplied by onsite nonsafety-related standby diesel generators. These diesel generators are nonsafety-related since they are not relied upon for safe shutdown, core cooling, or containment integrity. Rather, as discussed above, the Class 1E DC power supply system is the only power source required to monitor and actuate plant safety-related passive systems. 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 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. The NuScale plant safety-related loads are electrically separated from the nonsafety-related AC power system (whether powered from the offsite power system or the onsite nonsafety-related diesel generators) through battery chargers and inverters. Given this design configuration, 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 and application for design certification.

A.4 NuScale Emergency Core Cooling System, Shutdown Accumulator 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, 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. 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. Rather, although other alternatives are under consideration, the current design approach is for this safety function to be performed by the SAS – a system separate from the ECCS system. The SAS actuates passively via check valves, and thus does not require actuation signals.

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 3-4 and Table 3-5 of this report), such application of the regulatory framework may require further consideration. 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…." As indicated above, poison addition in the NuScale reactor plant design is the function of the SAS rather than the ECCS. Thus, as described in Table 3-4, as part of pre-application activities, NuScale will seek NRC concurrence that the underlying purpose of the rule is satisfied by the poison addition function of the SAS, and no departure from GDC 27 is needed.

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) and the SAS (for the poison addition safety 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, the SAS actuates passively via check valves, and thus does not require actuation signals.

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 except for extended operation of the ECCS with no AC electrical power available, would be the reactor pool water in which the containment vessel is submerged). 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 further consideration is warranted to determine whether this guidance – with modification – can be applied to the containment heat removal system design proposed in the NuScale application for design certification, or if new design-specific guidance is warranted. These examples and other potential modifications to the regulatory framework that will be applied to the NuScale ECCS, SAS, and containment heat removal system design are captured in Table 3-4 and Table 3-5 of this report.

A.5 Turbine Missile Protection for Essential SSCs

Substantive differences exist between the NuScale reactor 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 reactor 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 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[7] 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.

[7] 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.

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 noncondensible 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 noncondensible gases could interfere with post-accident natural circulation or pump operation that might inhibit long-term cooling following an accident. This intent was clarified in the statements of consideration (Reference 5.16) for a final rule effective October 16, 2003 that, in part, relocated the high point vent requirements from 10 CFR 50.44 to 10 CFR 50.46a (68 FR 54123 - 54143, dated September 16, 2003):

> The NRC is relocating these requirements because high point vents are relevant to emergency core cooling system (ECCS) performance during severe accidents, and the final § 50.44 does not address ECCS performance. The requirement to install high point vents was adopted in the 1981 amendment to § 50.44. This requirement permitted venting of noncondensible gases that may interfere with the natural circulation pattern in the reactor coolant system. This process is regarded as an important safety feature in accident sequences that credit natural circulation of the reactor coolant system. In other sequences, the pockets of noncondensible gases may interfere with pump operation. The high point vents could be instrumental for terminating a core damage accident if ECCS operation is restored. Under these circumstances, venting noncondensible gases from the vessel allows emergency core cooling flow to reach the damaged reactor core and thus, prevents further accident progression.
>
> – 68 FR 54129, September 16, 2003

> Other information… adequately defines the purpose of high point vents by acknowledging their usefulness both for forced circulation scenarios and in the natural circulation mode.
>
> – 68 FR 54134, September 16, 2003

The NuScale reactor module design includes reactor coolant system venting capability that satisfies the literal language of 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi). However, as a result of significant differences in the NuScale advanced reactor design compared to a traditional large LWR, the 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 noncondensible 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 noncondensible 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 described in the NuScale Design Overview, a NuScale power module comprises a reactor core, two steam generator tube bundles, and a pressurizer all contained within a single reactor pressure vessel, along with the containment vessel that immediately surrounds the reactor vessel. A nozzle on the upper head of the (pressurizer region of the) reactor pressure vessel

provides a connection for a high point vent valve[8] that allows for venting the reactor coolant system. In the NuScale design, this single valve satisfies the literal language of 10 CFR 50.46a requiring high point vents on both the reactor coolant system and the reactor vessel head. Specifically, in the NuScale design, the reactor coolant system is entirely contained within the reactor vessel, i.e., there are no reactor coolant piping loops, reactor coolant pumps, or separate pressurizer with its associated piping as would be found with traditional PWR designs. Thus, the reactor vessel head vent represents the high point venting capability for the reactor coolant system.

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 noncondensible 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 noncondensible 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 noncondensible gases would adversely affect its ability to provide adequate core cooling.

Similarly, there is no reasonable likelihood that an accumulation of noncondensible gases in the reactor pressure vessel could inhibit post-accident core cooling flow. For this reason, the reactor vessel high point vent valve and associated piping and components do 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). Thus, as applied to the NuScale design, the safety functions contemplated by 10 CFR 50.46a(c)(1) would include that related to the vent being an integral part of the reactor coolant pressure boundary (similar to traditional LWR designs), but would not include the function of ensuring long-term post-accident core cooling.

Further differentiation between the NuScale design and that of a traditional LWR is that the high point vent on the NuScale reactor vessel discharges directly to the radioactive waste management system rather than to the containment vessel. Thus, considerations that would be addressed for a typical high point vent system design that discharges to containment – specifically related to ensuring that vent operation does not challenge containment integrity – are not germane to the NuScale design. Specifically, the NuScale design intrinsically ensures that use of the vent during and following a postulated accident would not aggravate the challenge to containment or the course of the accident. This design does introduce a safety function specific to the NuScale vent system that typically would not be relevant for an LWR design that vents to containment – the containment vessel isolation function. Thus, in addition to the function related to the vent being an integral part of the reactor coolant pressure boundary as discussed above, the safety functions that NuScale would consider within the scope of 10 CFR 50.46a(c)(1) includes the containment isolation function.

In summary, the NuScale reactor module design includes reactor coolant system venting capability that satisfies the literal language of 10 CFR 50.46a and 10 CFR 50.34(f)(2)(vi). Accordingly, the NuScale reactor pressure vessel vent design is intended to meet the design and operational criteria specified in 10 CFR 50.46a(a), (b), and (c) and in and 10 CFR 50.34(f)(2)(vi).

[8] It should be noted that this valve is separate and distinct from the reactor vent valves discussed in Section A.4. The reactor vent valves are not intended for venting noncondensible gases, but rather operate in conjunction with the reactor recirculation valves to actuate the ECCS.

However, the 10 CFR 50.46a(c)(1) criterion will be applied with consideration for those safety functions relevant to the NuScale vent design, which as discussed above include only the (1) function related to the vent being an integral part of the reactor coolant pressure boundary, and (2) containment isolation function.

A.7 Containment Vessel

A containment building at a typical large PWR is a massive structure – approximately 200 to 230 feet in height and 130 to 60 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 steel component provides a leak-tight barrier to contain radiological releases, while the concrete component acts as a biological shield (against gamma radiation) and 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 subcompartments within these typical containment structures, thus requiring consideration of transient differential pressures due to postulated pipe breaks inside a subcompartment. 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., subatmospheric 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 65 feet (height) and 15 feet (outer diameter).[9] 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.[10] The NuScale containment vessel has no interior subcompartments, thus eliminating the potential for damaging transient differential pressures resulting from postulated high-energy pipe breaks within subcompartments (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 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

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

[10] As described in the NuScale Design Overview, 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.

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 an 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) 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 an open path to the environs.

Specifically, the containment vessel 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 an 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. 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 subcompartments 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, nor is an inerted atmosphere maintained that would be credited for combustible gas control pursuant to 10 CFR 50.44(c)(2). Notwithstanding, the robust design of the NuScale containment vessel and physical limitation on available oxygen as discussed above satisfy the underlying purpose of and thus support a partial exemption to 10 CFR 50.44(c)(2). The need for partial exemption to 10 CFR 50.44(c)(2) – specifically the portion requiring either that containment designs must have an inerted atmosphere or must limit hydrogen concentrations within containment, uniformly distributed, to 10 percent or less – is discussed in Table 3-4 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 subatmospheric 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 reactor 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, 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.

As a result of the design differences summarized above, portions of the SRP and other guidance directed towards containment design are not appropriate to apply to the NuScale containment vessel design. In some instances (described in Table 3-4 and Table 3-5 of this report), such application of the regulatory framework may require further consideration.

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 reactor 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.

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 flowpath 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.

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 secondary water flowing within the steam generator tubes, 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 require or use a blowdown system that typically is needed to remove the accumulated impurities and assist in maintaining acceptable secondary water chemistry in the steam generators. The secondary chemistry requirements for the NuScale design also are anticipated to differ from those outlined in industry guidance.

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.

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. In some instances (described in Table 3-5 of this report), such application of the regulatory framework may require further consideration.

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

The NuScale design includes a 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 core uncovery or to assure 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[11] 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 core uncovery 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 decay heat removal would be provided by the unaffected steam generator via DHR system operation (see Section A.2.1 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 core uncovery 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

[11] 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.

(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).