6 ENGINEERED SAFETY FEATURES

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff (hereinafter referred to as the staff) review of Chapter 6, “Engineered Safety Features,” of the NuScale Power, LLC (hereinafter referred to as the applicant) Design Certification Application (DCA), Part 2, Tier 2, “Final Safety Analysis Report.”

6.1.1 Engineered Safety Features Materials

6.1.1.1 Introduction

To address the review of the selection, fabrication methods, and compatibility of materials with fluids for engineered safety feature (ESF) systems, NuScale submitted information in the DCA Part 2, Tier 1, “Certified Design Descriptions and Inspections, Tests, Analyses, & Acceptance Criteria (ITAAC),” and DCA Part 2, Tier 2, Section 6.1.1, “Engineered Safety Feature Materials.”

The information that NuScale provided can be found in Revision 2 to the DCA, dated October 30, 2018, (Agencywide Documents Management and Access System (ADAMS) Accession No. ML18311A006), and in letters dated July 11, 2017 (ADAMS Accession No. ML17192A869); August 3, 2017 (ADAMS Accession No. ML17215A977); August 21, 2017 (ADAMS Accession No. ML17233A364); November 20, 2017 (ADAMS Accession No. ML17324B389); November 27, 2017 (ADAMS Accession No. ML17331A994); December 18, 2017 (ADAMS Accession No. ML17352B263); April 20, 2018 (ADAMS Accession No. ML18110A359); July 3, 2018 (ADAMS Accession No. ML18184A239); July 12, 2018 (ADAMS Accession No. ML18193B178); and December 17, 2018 (ADAMS Accession No. ML18351A357).

The NRC staff evaluation considered materials and fabrication, composition and compatibility of ESF fluids, component and systems cleaning, and thermal insulation of the ESF systems.

6.1.1.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.1, “NuScale Power Module,” includes DCA Part 2, Tier 1, information associated with this section.

DCA Part 2, Tier 2: The applicant provided a description of the ESF materials in DCA Part 2, Tier 2, Section 6.1.1, “Engineered Safety Feature Materials,” which is summarized in the following discussion.

As described in DCA Part 2, Tier 2, Section 1.9, “Conformance with Regulatory Criteria,” and Section 6.1.1, the NuScale Design conforms to the guidance provided in the following Regulatory Guides (RGs):

- RG 1.28, “Quality Assurance Program Criteria (Design and Construction),” Revision 4
- RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” Revision 4
- RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components,” Revision 1
- RG 1.44, “Control of the Processing and Use of Stainless Steel,” Revision 1

- RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel,” Revision 1

- RG 1.71, “Welder Qualification for Areas of Limited Accessibility,” Revision 1

DCA Part 2, Tier 2, Table 1.9-3, “Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS),” summarizes the differences between the DCA and NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)” (SRP), Section 6.1.1.

Metallic Materials

DCA, Part 2, Tier 2, Section 6.1.1, stated that the NuScale ESF systems include the containment system (CNTS), emergency core cooling system (ECCS), and decay heat removal system (DHRS). The ECCS and CNTS are described in DCA Part 2, Tier 2, Section 6.3, “Emergency Core Cooling System,” and Section 6.2, respectively. The DHRS is described in DCA Part 2, Tier 2, Section 5.4.3.

DCA Part 2, Tier 2, Table 6.1-1, “Material Specifications for ESF Components,” lists the material grade and material type for the ESF pressure boundary materials; weld materials, including cladding materials; and associated supports. DCA Part 2, Tier 2, Table 6.1-3, “Pressure Retaining Materials for RCPB and ESF Valves,” lists the allowable materials for the components of reactor coolant pressure boundary (RCPB) and ESF valves, which include the reactor vent valves (RVV), reactor recirculation valves (RRV), reactor safety valves (RSV), DHRS actuator valves, reactor coolant system (RCS) check valves, RCS excess flow check valves, RCS injection and discharge isolation valves, and containment isolation valves (CIVs). The CIVs are listed in DCA Part 2, Tier 2, Table 6.2-5, “Containment Isolation Valve Information.”

The applicant stated that the material selection and fabrication methods ensure that the ESF components are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). The applicant also stated that ESF pressure-retaining components are fabricated of materials that have a low probability of abnormal leakage, rapidly propagating failure, or gross (nonbrittle) rupture.

The applicant stated that DCA Part 2, Tier 2, Section 6.1.1, also provides information on materials within the containment vessel (CNV) that are associated with non-ESF systems. These systems, which have components within the CNV, include the containment flood and drain system (CFDS), RCS, steam generator system (SGS), and control rod drive system (CRDS). DCA Part 2, Tier 2, Figure 6.2-4, “Containment System Piping and Instrumentation Diagram,” shows the CNTS piping and the system classification breaks. DCA Part 2, Tier 2, Figure 6.6-1, “ASME Class Boundaries for NuScale Power Module Piping Systems,” shows the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class Boundaries. DCA Part 2, Tier 2, Table 6.1-2, “Material Specifications for CNV Related non-ESF Components,” lists the material grade and material type for the non-ESF pressure boundary materials, weld materials, and associated supports of the aforementioned systems. The materials for these non-ESF portions of the CNTS are reviewed in this section of the SER.

The materials for these non-ESF systems were also selected and fabricated to be compatible with the environmental conditions associated with normal operation and postulated accidents within the CNV, including those that would expose the components to reactor coolant water chemistry. SRP Section 4.5.1, “Control Rod Drive Structural Materials,” discusses materials that are internal to the control rod drive mechanisms (CRDMs), or part of the CRDM pressure

boundary. Therefore, the materials of the CRDS listed in DCA, Part 2, Tier 2, Table 6.1-2 that are within the CNV are reviewed this section of the SER.

The materials of the portion of the SGS within the CNV are reviewed in Section 5.4.2.1 of the SER. The materials of the portion of the RCS within the CNV are reviewed in Section 5.2.3 of the SER. The CIVs of these aforementioned systems are part of the CNTS and are reviewed in this section of the SER.

Material Selection and Fabrication

DCA Part 2, Tier 2, Section 6.1.1.1, "Material Selection and Fabrication," stated that ESF pressure boundary materials, weld materials, and associated supports conform to the fabrication, construction, and testing requirements of ASME Code, Section II, "Materials," and Section III, "Rules for Construction of Nuclear Facility Components," including the requirements of ASME Code, Section III, Articles NB-2000, NC-2000, and NF-2000. The materials are compatible with the coolant system fluids and their selection is consistent with ASME Code, Section II, Parts A, B, and C; and ASME Code, Section III, Appendix I. The fracture toughness properties of the ferritic pressure-retaining ESF components and supports also comply with the requirements of the ASME Code. DCA Part 2, Tier 2, Section 6.2.1.1.1, "Design Bases," stated that the CNV is an ASME Code Class MC (steel) containment that is designed, analyzed, fabricated, inspected, tested, and stamped as an ASME Code Class 1 pressure vessel. DCA Part 2, Tier 2, Section 6.1.1.1, stated, consistent with DCA Part 2, Tier 2, Table 5.2-1, "American Society of Mechanical Engineers Code Cases," that Code Case N-759-2, "Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Class 1, 2, and 3 Section III, Division 1," is used for the reactor pressure vessel (RPV) and CNV. Code Case N-759-2 is not related to the materials selection of the ESF materials and not reviewed in this section of the SER.

The applicant stated that the lower portion of the CNV is solution annealed austenitic stainless steel, SA-965, Grade FXM-19, and weld filler metals E209 or ER209 will be used for welds between the SA-965, Grade FXM-19 components. The applicant stated that the peak fluence at the core region of the CNV is less than $1x10^{19}$ n/cm^2 (E > 1.0 MeV) for the 60-year design life. DCA Part 2, Tier 2, Section 6.2.1, contains more information related to the irradiation of the CNV and stated that the peak fluence in the lower CNV does not exceed 5.5 x 10^{18} n/cm^2 (E > 1.0 MeV).

The upper shell, top head, and associated supports of the CNV are fabricated from low alloy steel and clad with austenitic stainless steel. To avoid cracking of the base material, the stainless steel weld overlay cladding process conforms to the guidelines of RG 1.43, and the underlying low alloy steel satisfies fine grain requirements, specifically that the low alloy steel will be manufactured to an American Society for Testing and Materials (ASTM) grain size of 5 or finer. The weld cladding processes are also qualified in accordance with the ASME Code. The interior surface of the low-alloy steel portions of the CNV is clad with at least one layer of Type 309L austenitic stainless steel. The exterior surface of the low-alloy steel portions, including the CNV through-bolt holes on the CNV upper main assembly closure flange, is clad with at least two layers of austenitic stainless steel. The first layer consists of Type 309L stainless steel and the subsequent layers are Type 308L stainless steel. The applicant also stated that electroslag welding is only used for the austenitic stainless steel cladding of low-alloy steel.

"NuScale Containment Leakage Integrity Assurance Technical Report," Technical Report (TR)-1116-51962-NP, Revision 0, is incorporated by reference, in accordance with DCA Part 2,

Tier 2, Section 1.6, “Material Referenced,” and Table 1.6-2, “NuScale Referenced Technical Reports.”

Relevant to the ESF Materials, TR-1116-51962-NP, Revision 0, states the following:

- The maximum carbon content of SA-965 Grade FXM-19 and E209/ER209 weld filler metals is restricted to 0.04 percent

- RG 1.44 is applicable to FXM-19 related to confirming non-sensitization

- Weld filler metals contain 5 to 20 FN delta ferrite in accordance with RG 1.31

The applicant stated that unstabilized type 3XX series austenitic stainless steel materials meet the requirements of RG 1.44. The applicant also stated that, where austenitic stainless steel, which would include SA-965, Grade FXM-19, are subjected to sensitizing temperatures for greater than 60 minutes during postweld heat treatment, nonsensitizing of the materials is verified by testing in accordance with ASTM A262, Practice A or E. The applicant also stated that furnace-sensitized austenitic stainless steel is not used.

The applicant stated that the delta ferrite content of stainless steel weld filler material, which includes SA-965, Grade FXM-19, conforms to the guidelines stipulated in ASME Code, Section III, Paragraphs NB-2433, NC-2433, or NF-2433, as well as RG 1.31. However, while the applicant stated that RG 1.31 and ASME Code, Section III, Paragraph NB-2433, do not apply to the filler metals used to deposit cladding because the cladding is a corrosion-resistant layer and does not have a structural function, the applicant stated that the delta ferrite content in the austenitic stainless steel weld cladding is controlled between ferrite numbers 5 and 20.

The staff requested that information in TR-1116-51962-NP, Revision 0, related to controls on FXM-19, be incorporated into the next revision of the DCA and is tracking this as **Confirmatory Item 6.1.1-1.**

The applicant stated that the nickel-based weld filler metal used for Alloy 690 to low-alloy material welds are made with Alloy 52/152/52M.

DCA Part 2, Tier 2, Section 6.1.1.1, describes the pressure-retaining threaded fasteners and threaded inserts. Threaded fasteners and their associated components are further discussed in DCA Part 2, Tier 2, Section 3.13, “Threaded Fasteners (ASME Code Class 1, 2, and 3).” Threaded fasteners and their associated components, including materials, are reviewed in Section 3.13 of the SER.

The NuScale design does not use thermal insulation (metallic or nonmetallic) inside the CNV. Reflective metallic insulation is used for the external upper CNV head above the reactor pool water level. Fibrous material is not permitted. DCA Part 2, Tier 2, Section 6.1.2, “Organic Materials,” states that mineral (silicon dioxide) insulated cabling is within a Type 304L stainless steel jacket and is reviewed in Section 6.1.2 of the SER.

The applicant stated that the use of cold-worked austenitic stainless steel is avoided, and if it is used, then the yield strength (as determined by 0.2 percent offset method) is limited to 90 kilopound per square inch (ksi) maximum. The applicant also stated that cold working of austenitic steel from abrasive work is minimized, and when abrasive work is used, ferritic carbon steel contaminants are avoided. The applicant stated that controls will be established to meet

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, Criterion IX, “Control of Special Processes,” and Criterion XIII, “Handling, Storage and Shipping,” as well as the applicable requirements of ASME NQA-1, “Quality Assurance Requirements for Nuclear Facility Applications,” and RG 1.28.

The applicant stated that socket welds are not used on piping greater than ¾-inch nominal pipe size (NPS) in DCA Part 2, Tier 2, Table 6.1-1, “Material Specifications for ESF Components,”, and any socket weld used on piping less than ¾-inch NPS conforms to ASME B16.11. Components that are less than or equal to ¾-inch NPS include two small tubes connected to the ECCS trip/reset actuator valve. Further explanation is in DCA Part 2, Tier 2, Section 3.8.2.1.7, “Emergency Core Cooling System Trip/Reset Valve Penetrations,” and NuScale request for additional information (RAI) response letter dated May 29, 2018 (ADAMS Accession No. ML18149A651). In addition, socket welds are not used for piping NPS 2 or less in size in DCA Part 2, Tier 2, Table 6.1-2.

Composition and Compatibility of Core Cooling Coolants

The ECCS valves, their actuators, and their connecting hydraulic lines are designed to be compatible with the RCS chemistry that would be present under LOCA condition, as well as compatible with the ultimate heat sink (UHS) water.

The two trains of DHRS are designed to ASME Code Class 2 requirements and mostly submerged in the UHS. However, some of the DHRS piping is within the CNV. The DHRS is designed to be compatible with the RCS, secondary, and UHS water chemistries.

The non-ESF CFDS piping, supports, and components that are within the CNV and defined as part of the CNTS are also designed to be compatible with the aforementioned water chemistries. The other non-ESF piping, supports, and components that are associated with the RCS, CRDS, and SGS are also designed to be compatible.

The RCS chemistry is controlled by the Electric Power Research Institute (EPRI), Pressurized Water Reactor (PWR) Primary Water Chemistry Guidelines. Since the materials inside of the CNV are designed to be compatible with the RCS water, the applicant prohibits the use of materials within the CNV that could alter postaccident coolant chemistry. DCA Part 2, Tier 2, Section 5.2.3, contains additional information related to the RCS chemistry and reviewed Section 5.2.3 of the SER. The secondary water chemistry control program is described and reviewed in Section 10.4.6 of the SER. The UHS cleanup system is described in DCA Part 2, Tier 2, Section 9.1.3, “Spent Fuel Pool Cooling and Cleanup System,” and reviewed in Section 9.1.3 of the SER. Since the selected materials are compatible with the various water chemistries, none of the components are designed with a corrosion allowance.

ITAAC: The ITAAC associated with DCA Part 2, Tier 2, Section 6.1.1, are located in DCA Part 2, Tier 1, Section 2.1.

Technical Specifications: DCA Part 2, Tier 2, Chapter 16, “Technical Specifications,” does not contain technical specifications (TS) related to DCA Part 2, Tier 2, Section 6.1.1.

Technical Reports: The staff reviewed the following TR that is incorporated by reference in DCA Part 2, Tier 2, Table 1.6-2, “NuScale Referenced Technical Reports:”

- TR-1116-51962-NP, Revision 0

6.1.1.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” General Design Criteria (GDC) 1, “Quality Standards and Records,” requires that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

- 10 CFR 50.55a, “Codes and Standards,” lists the standards and documents approved for incorporation by reference.

- GDC 4, “Environmental and Dynamic Effects Design Bases,” requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

- GDC 14, “Reactor Coolant Pressure Boundary,” requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” requires that the RCPB be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized.

- 10 CFR Part 50, Appendix B, Criteria IX and XIII, require establishing measures to: (1) assure that special processes are controlled and accomplished in accordance with applicable codes, standards, specifications, criteria, and other special requirements, and (2) control cleaning of material and equipment in accordance with work and inspection instructions to prevent damage or degradation.

The NuScale PDC are discussed in DCA Part 7, “Exemptions,” and DCA Part 2, Tier 2, Section 3.1.

The guidance in SRP Section 6.1.1, “Engineered Safety Features Materials,” Revision 2, lists the acceptance criteria adequate to meet the above requirements as well as review interfaces with other SRP sections.

Branch Technical Position 6-1 (BTP 6-1), “pH For Emergency Coolant Water for Pressurized Water Reactors,” lists the controls on the post-accident coolant water pH and chemistry to meet the requirements of GDC 14.

The staff notes that RG 1.28 is not mentioned in SRP Section 6.1.1. RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” was withdrawn and replaced with RG 1.28. SRP Section 6.1.1 will be updated at a future time to reflect the new staff guidance.

6.1.1.4 Technical Evaluation

Materials Selection and Fabrication

To meet the requirements of GDC 1 and 10 CFR 50.55a to ensure that plant SSCs important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function they perform, the applicant must identify codes and standards and maintain records. Selection of the materials specified for use in these systems must be in accordance with the applicable provisions of ASME Code, Section III, Division 1 or RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III." ASME Code, Section III references applicable portions of ASME Code, Section II, Parts A, B, and C.

The applicant stated that materials for these systems comply with Parts A, B, and C of ASME Code Section II, and Appendix I to ASME Code, Section III. The applicant stated that the fracture toughness requirements for all ferritic ESF materials will comply with the requirements of ASME Code, Section III, Subarticles NB-2300, NC-2300, and NF-2300. The staff reviewed the material specifications listed in Table 6.1-1, Table 6.1-2, and Table 6.1-3, and verified that the materials are acceptable for use in accordance with ASME Code, Section II and Section III. The staff also verified the that the penetrations listed in DCA Part 2, Tier 2, Table 6.2-4, "Containment Penetrations," match the penetrations listed in Table 6.1-1.

Operational experience with cast austenitic stainless-steel components has shown that the base material can lose ductility when exposed to high temperatures or significant radiation fields over extended periods of time. DCA Part 2, Tier 2, Table 6.1-3, "Pressure Retaining Materials for RCPB and ESF Valves," states that cast austenitic stainless steel components may be used for ESF valves. Based on the location of these valves, the staff finds that the cast austenitic stainless-steel components in the ESF system are not susceptible to radiation embrittlement because of the distance from the reactor vessel.

Staff guidance is found in an NRC letter, "License Renewal Issue No. 98-0030, 'Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components,'" dated May 19, 2000 (ADAMS Accession No. ML003717179), which specifies a screening temperature of above 250 degrees Celsius (C) (482 degrees Fahrenheit (F)) for cast components. Note 6 of DCA Part 2, Tier 2, Table 6.2-5, states that all CIVs have a design temperature of 650 degrees F. DCA Part 2, Tier 2, Table 6.3-2, "Emergency Core Cooling System Valve and Actuator Design and Operating Parameters," states that the RRV, RVV, and valve actuators design temperature is 650 degrees F. Therefore, cast ESF valve components are subject to the screening criteria.

Note 2 of DCA Part 2, Tier 2, Table 6.1-3 states that carbon is limited to 0.03 percent maximum and delta ferrite is limited to 20 percent maximum, except to 14 percent maximum for CF3M and CF8M. CF3M and CF8M have a higher percentage of molybdenum (2.0 percent and 3.0 percent). Based on the guidance in the NRC letter, the staff finds that the delta ferrite limits are within the bounds to ensure that the cast austenitic stainless steel components in the ESF system are not susceptible to thermal embrittlement. As such, the staff finds their use acceptable.

The staff finds that the materials selected for the NuScale ESF systems satisfy the applicable requirements of ASME Code, Section II and Section III, and therefore satisfy GDC 1 and 10 CFR 50.55a.

Austenitic Stainless Steel

The NuScale design must meet the requirements of GDC 4, GDC 14, and the quality assurance requirements of Appendix B of 10 CFR Part 50. Designs may meet these requirements by following the guidance of RGs 1.28, 1.31, and 1.44. Designs must also provide controls over the use of cold-worked austenitic stainless steels.

The materials selected are compatible with the environmental conditions associated with the NuScale design. Therefore, components that interact with the primary, secondary, or UHS water chemistries are either fabricated out of nickel alloys, austenitic stainless steel, or stainless steel-clad ferritic steel. NuScale commits to following the guidance in RGs 1.28, 1.31, and 1.44 to ensure the quality assurance criteria for cleaning fluid systems, controlling delta ferrite content to mitigate microfissures, and avoiding sensitization that could lead to intergranular stress-corrosion cracking (SCC) for unstabilitized austenitic stainless steel American Iron and Steel Institute (AISI) Type 3XX and FXM-19, respectively. The applicant also stated that cold working of austenitic steel from abrasive work is minimized, and when abrasive work is used, ferritic carbon steel contaminants are avoided. NuScale stated that it is complying with RG 1.44, which discusses the use of low-carbon (L-grade) stainless steel. DCA Part 2, Tier 2, Section 6.1.1, Table 6.1-1, Table 6.1-2, and Table 6.1-3, state which materials will be used, either Grade or Type “304/304L,” and different variations such as forging (F) or typical (TP). As denoted by the Notes of the Tables, the procured material will be dual certified to ensure its compliance with RG 1.44 by containing less than 0.03 weight percent (wt%) carbon. The maximum carbon content of SA-965 Grade FXM-19 and E209/ER209 weld filler metals is restricted to 0.04 percent.

The ferritic components of the NuScale design are clad with either a single or double layer of stainless steel, based on normal exposure to UHS water. Because of the cladding of the low-alloy steel components, NuScale commits to following RG 1.43, which is related to stainless steel weld cladding of low-alloy steel components. The use of stainless steel and clad ferritic steel also eliminates flow-accelerated corrosion concerns.

The staff finds that the applicant controls over the use of cold-worked austenitic stainless steels, and compliance with RGs 1.28, 1.31, and 1.44, acceptable to reduce the susceptibility of degradation to stainless steel components. Based on the conformance to RG 1.44, the staff finds the use of dual certified stainless steel acceptable to ensure that stainless steel will contain less than 0.03 wt% carbon. The staff also find limiting the maximum carbon content of FXM-19 to 0.04 percent acceptable. Because of the specific NuScale design, the staff also finds it acceptable that NuScale will conform to the guidance in RG 1.43, and the additional delta ferrite requirements specifically for the stainless steel cladding.

NuScale stated that the lower portion of the CNV is solution-annealed austenitic stainless steel, SA-965, Grade FXM-19 and weld filler metals E209 or ER209 will be used for welds between the SA-965, Grade FXM-19, components. Weld filler metals E209 and ER209 are most often used to weld SA-965, Grade FXM-19, base metal. The lower portion of the CNV is located closer to the reactor core compared to a traditional light-water reactor (LWR). Because of the proximity to the reactor core, NuScale considered the impacts of radiation embrittlement on the CNV. NuScale used the same methodology to calculate the fluence on the CNV as the RPV. NuScale stated that the peak fluence in the lower CNV does not exceed 5.5×10^{18} n/cm^2 (E > 1.0 MeV). The staff reviewed NuScale‘s methodology. The staff finds the use of SA-965, Grade FXM-19, and its associated weld filler metals acceptable for use in the lower portion of the CNV as the calculated fluence to the CNV is lower than what is expected to cause

embrittlement, and the selection of SA-965, Grade FXM-19, an austenitic stainless steel, is resistant to radiation embrittlement. Additional detail related to the fracture prevention of the CNV is located in TR-1116-51962-NP, Revision 0; DCA Part 2, Tier 2, Section 6.2.7, and reviewed in Section 6.2.7 of the SER.

Ferritic Steel Welding

The NuScale design must meet the requirements of GDC 1; 10 CFR Part 50, Appendix B, Criterion IX; and 10 CFR 50.55a. The designers may meet these requirements by meeting ASME Code, Section III, Appendix D, Article D-1000, and RG 1.50. In addition, moisture control on low-hydrogen welding materials must conform to the requirements of ASME Code, Section III. Lastly, a designer may comply with RG 1.71 for welding in areas of limited accessibility to meet the requirements.

The applicant stated that it will comply with RG 1.50 and RG 1.71 to control preheat temperatures on low-alloy steel to prevent cracking during welding and provide supplemental performance qualification requirements for welding caused by restricted physical and visual access. The applicant also stated that it will comply with ASME Code Section III, Appendix D, Article D-1000, related to preheat temperatures, and ASME Code Section III for moisture controls for low-hydrogen welding materials. The applicant also stated that controls established for special processes for the CNV and ESF materials satisfy the applicable requirements of 10 CFR Part 50, Appendix B, Criterion IX.

The staff finds the applicant's compliance with RG 1.50 and RG 1.71 acceptable to ensure the quality of welds. Furthermore, since the applicant will follow the provisions of ASME Code, Section III, related to minimum preheat temperatures and moisture control, the staff finds this acceptable, as it will meet the requirements of GDC 1, 10 CFR Part 50, Appendix B, Criterion IX, and 10 CFR 50.55a.

Composition and Compatibility of Engineered Safety Feature Fluids

To meet the requirements of GDC 4, GDC 14, PDC 35, and PDC 41, SSCs important to safety should be designed to be compatible with the environmental conditions. The design should also assure that hydrogen generation caused by corrosion during a design-basis accident (DBA) is controlled to maintain containment integrity.

The NuScale design does not use any materials, paint, or coatings within CNV that contribute to corrosion-related hydrogen production or alter post-LOCA coolant chemistry. Furthermore, the materials selected are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs. The RCS, secondary, and UHS water chemistry programs are controlled to prevent degradation of components. Because of the NuScale design, ESF components can be subject to each of these various chemistries. Ensuring compliance with the water chemistry programs is essential for the NuScale design to prevent degradation of the ESF components. The RCS, secondary, and UHS water chemistry programs are reviewed in Section 5.2.3, Section 10.4.6, and Section 9.1.3 of the SER, respectively. For these reasons, NuScale does not incorporate a corrosion allowance. Since NuScale has selected materials that are compatible with the environmental conditions, as well as prohibited the use of materials that could contribute to corrosion-related hydrogen production or alter post-LOCA coolant chemistry, the staff finds not including a corrosion allowance acceptable.

Component and System Cleaning

To meet the requirements of 10 CFR Part 50, Appendix B, Criteria IX, and XIII, the applicant should establish measures to: (1) assure that special processes are controlled and accomplished in accordance with applicable codes, standards, specifications, criteria, and other special requirements; and (2) control cleaning of material and equipment in accordance with work and inspection instructions to prevent damage or degradation.

The applicant stated that controls will be established to meet 10 CFR Part 50, Appendix B Criterion XIII, as well as the applicable requirements of ASME NQA-1 and RG 1.28 related to cleanliness controls and cleaning fluid systems. Additionally, NuScale stated that controls for special processes will meet the applicable requirements of 10 CFR Part 50, Appendix B, Criterion IX.

Since the NuScale design complies with NQA-1 and RG 1.28, and the applicant stated that controls will be established for both special processes and the handling, storage, shipping, cleaning, and preservation of CNV and ESF materials and equipment to prevent damage or deterioration, the staff finds that NuScale meets the applicable requirements of 10 CFR Part 50, Appendix B, Criteria IX and XIII.

Thermal Insulation

To meet the requirements of GDC 1, 14, and 31, ESF systems should be designed, fabricated, erected, and tested such that there is an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The levels of leachable contaminants in nonmetallic insulation materials that come into contact with Type 3XX series austenitic stainless steels used in fluid systems important to safety should be controlled to mitigate SCC. In particular, the leachable chlorides and fluorides should be held to the lowest levels practical. The staff's position is that following the guidance in RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," is an acceptable method to control leachable contaminants in nonmetallic insulation materials. The applicant stated that thermal insulation (metallic or nonmetallic) is not used inside the CNV, and the insulation used on the exterior upper CNV head above the reactor pool is reflective metallic insulation. The applicant stated that the use of fibrous material is not permitted. Since nonmetallic thermal insulation will not be used in the CNV, the staff finds that RG 1.36 is not applicable. Furthermore, since the applicant stated that insulation will not be used inside the CNV, and the use of fibrous material is not permitted, the staff finds this approach acceptable to meet the requirements of GDC 1, 14, and 31.

Branch Technical Position 6-1

NRC Staff guidance includes BTP 6-1, "pH for Emergency Coolant Water for Pressurized Water Reactors," which describes the minimum pH range for the post-LOCA recirculating fluid. BTP 6-1 recommends a minimum fluid pH of 7.0 in order to prevent stress corrosion cracking (SCC) of austenitic stainless-steel, and it focuses on containment spray systems because they may be vulnerable to SCC in a post-LOCA environment that may contain sources of chloride ions. Standard grades of austenitic stainless-steel, such as Types 304 and 316, are susceptible to SCC in acidic chloride-containing water depending on the pH, temperature, and chloride concentration. Typical pressurized water reactor (PWR) containments have a pH buffer chemical that dissolves in the post-LOCA fluid and maintains the pH above 7. Another basis for pH 7 is iodine retention in the post-LOCA fluid, but this is addressed in SRP Sections 6.5.2 and 15.0.3 and is not the subject of BTP 6-1. BTP 6-1 also recommends consideration of hydrogen generation from aluminum if the pH is greater than 7.5. DCA, Part 2, Tier 2, Table 1.9-3 states

that although the NuScale design has no containment sprays or sumps, the pH criteria are nonetheless applicable for SCC, iodine retention, and hydrogen generation. These pH values all refer to pH at ambient temperature.

According to DCA, Part 2, Tier 2, Section 15.0.3, the at-temperature pH of the post-LOCA fluid would remain between 6.0 and 7.0 for a period of 30 days. Since the pH of water decreases with increasing temperature, the range of 6.0 to 7.0 at temperature corresponds to a higher pH range at ambient temperature. The staff finds the NuScale design acceptable with respect to BTP 6-1 because the ESF components are designed for operation in the high-temperature, low-pH, low-chloride conditions of the primary coolant, and there are not sources of chloride that would create conditions for SCC of austenitic stainless-steel following a LOCA. Therefore, even if the pH does not meet the 7.0 minimum, it would be in a range for which the ESF components are designed because it will be at the normal operating pH or higher. In addition, aluminum is not used in containment, so hydrogen generation from a pH potentially greater than 7.0 does not need to be considered. The restrictions on the use of materials in containment are described in FSAR Tier 2 Sections 6.1.1 and 6.1.2.

Tier 1 and ITAAC

The ITAAC related to the ESF materials include the ITAAC that ensure that the ESF systems comply with or conform to ASME Code, Section III, requirements, which include materials. The staff reviewed the proposed ITAAC and finds that the ITAAC and DCA Part 2, Tier 1 information adequately describe the requirements for the ESF materials. Further, the staff concludes that the ITAAC requirements are sufficient to demonstrate that if the ITAAC are satisfied, the ESF materials will be constructed in accordance with the design certification, the provisions of the Atomic Energy Act of 1954, as amended (AEA), and NRC regulations, as required by 10 CFR 52.47(b)(1).

Required ITAAC for the ESF systems (CNTS, ECCS, and DHRS) are described in each of those system subsections of this report. The ECCS and CNTS are evaluated in Section 6.3 and Section 6.2 of this report, respectively. The DHRS is evaluated in Section 5.4.3 of this report.

A discussion of all NuScale ITAAC is provided in Section 14.3 of this SER.

Technical Specifications

There are no TS requirements associated with the ESF materials. Required TS for the ESF systems (CNTS, ECCS, and DHRS) are evaluated in each of those system subsections of this report. The ECCS and CNTS are evaluated Section 6.3 and Section 6.2 of this report, respectively. The DHRS is evaluated in Section 5.4.3 of this report. Therefore, the staff finds this acceptable in accordance with 10 CFR 50.36, “Technical Specifications.”

6.1.1.5 Combined License Information Items

There are no combined license (COL) Items associated with DCA Part 2, Tier 2, Section 6.1.1.

6.1.1.6 Conclusion

Based on its review of the information provided by NuScale, and subject to the closure of **Confirmatory Item 6.1.1-1**, the staff concludes that the NuScale DCA for the materials to be used in the fabrication of the ESF systems is acceptable and meets the relevant requirements of GDC 1, 4, 14, and 31; PDC 35 and 41; Appendix B to 10 CFR Part 50; and 10 CFR 50.55a.

6.1.2 Organic Materials

6.1.2.1 Introduction

Organic and inorganic coatings are typically used at nuclear power plants to provide corrosion protection or facilitate surface decontamination. In some locations, coating failure can be a source of debris that could prevent the ECCS from performing its safety-related function. Conditions causing degradation and failure can be present during operation, maintenance, or accident conditions. Organic materials are also typically found in the form of coatings, cable jacketing, and cable insulation.

For plant designs with protective coatings and organic materials that could affect the ESF systems, the types of organic materials, the quality assurance applied to coatings, and the potential for physical and chemical decomposition products are evaluated according to the criteria in SRP Section 6.1.2, “Protective Coating Systems (Paints) - Organic Materials.”

6.1.2.2 Summary of Application

The information in the application on coatings and organic materials is summarized below. Based on the exclusion of coatings and organic materials in the CNV, the staff did not perform a technical evaluation of the information in DCA Part 2, Tier 2, Section 6.1.2, according to SRP Section 6.1.2, and there are no conclusions for this section of the application. Instead, the acceptability of Type 304 stainless steel for ESF systems is evaluated in Section 6.1.1 of this SER. The potential for debris generation from the cable insulation is evaluated in Section 6.2.2 of this SER.

DCA Part 2, Tier 1: There is no Tier 1 information for this topic.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 6.1.2, states that protective coatings are not permitted on the inside or outside surface of the CNV or on any other ESF or non-ESF system components located within the CNV. Section 6.1.2 also describes cable in the CNV as follows:

- mineral insulated with silicon dioxide;
- jacketed with unpainted, seamless Type 304 stainless steel; and
- free of organic material in both the insulation and jacketing.

ITAAC: There are no ITAAC items related to this review topic.

Technical Specifications (TS): There are no TS for this review topic.

6.2 Containment Systems

6.2.1 Introduction

This section describes the NRC staff review for conformance to RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” Revision 4, regarding component performance in the ECCS flowpath during long-term cooling. The NuScale design does not include pumps, piping, trash racks, debris interceptors, or sump screens.

6.2.1.1 Containment Structure

***Note:** Section 6.2.1.1 contains three unresolved issues without a mutually understood and clearly defined path toward resolution. Although these issues are presumptively labeled as Open Items in this section, they are highlighted to differentiate them from the "qualified" Open Items with a path to resolution, recognizing that these issues need to advance in developing a clear path toward resolution.*

Introduction

The primary functions of the reactor containment building are to protect the safety-related SSCs located within it and to establish an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment during normal plant operation and accidents. The containment encloses the reactor systems and is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure shall be designed to withstand, without loss of function, the impact of the postulated accidents involving the release of high energy fluids from the RCS and secondary systems. The containment structure shall also maintain functional integrity in the long term following a postulated accident (i.e., it shall remain a low-leakage barrier against the release of fission products for as long as postulated accident conditions require). The design and sizing of a CNTS are largely based on the pressure and temperature conditions that result from the release of the reactor coolant in the event of a LOCA or a non-LOCA design-basis event (DBE). The containment design basis includes considerations of the effects of stored energy in the RCS, decay energy, and energy from other sources, such as the secondary system, and metal-water reactions, including the recombination of hydrogen and oxygen. The CNTS is not required to be a complete and independent safeguard against a LOCA or a non-LOCA DBE by itself but functions to contain any fission products released while the ECCS cools the reactor core.

Summary of Application

DCA Part 2, Tier 1: The Tier 1 information associated with this evaluation is provided in Tier 1, Section 2.1; Section 2.3, "Containment Evacuation System"; and Section 3.6, "Ultimate Heat Sink."

DCA Part 2, Tier 2: The Tier 2 information associated with this evaluation is provided in Section 6.2.1.1. A summary of the technical information is as follows.

The CNV is a compact, steel pressure vessel that consists of an upright cylinder with top and bottom head closures. The CNV is partially immersed in a below-grade, borated-water-filled, stainless steel-lined, reinforced concrete reactor pool that provides a passive heat sink and is absent of internal sumps or subcompartments that could entrap water or gases. The CNV is an evacuated pressure vessel fabricated from a combination of low-alloy steel and austenitic stainless steel that houses, supports, and protects the RPV from external hazards and provides a barrier to the release of fission products to the environment (GDC 16, "Containment Design"), while accommodating the calculated pressures and temperatures resulting from postulated mass and energy (M&E) release inside containment with margin such that design leakage rates are not exceeded (GDC 50, "Containment Design Basis"). The CNV is an ASME Code Class MC (steel) containment and is rated as an ASME Code Class 1 pressure vessel. The CNV and the reactor pool are housed within a seismic Category 1 reactor building. The CNV design includes no internal subcompartments, which eliminates the potential for collection of combustible gases and differential pressures resulting from postulated high-energy pipe breaks within containment. The CNV design specifications also take into consideration the pressures

and temperatures associated with combustible gas deflagration described in DCA Part 2, Tier 2, Section 6.2.5, “Combustible Gas Control in the Containment Vessel.”

The CNV is designed to withstand the full spectrum of primary and secondary system M&E releases (LOCA and non-LOCA DBEs) while considering the worst case single active failure and loss-of-power conditions. Under these conditions, the CNV transfers the RCS coolant heat and core decay heat through its walls to the UHS and provides effective passive, natural circulation emergency core cooling flow. The integrated design of the RPV and CNV ensures that RCS leakage is collected within the CNV. In the event of primary system releases (e.g., LOCAs or ECCS valve opening events), the CNV provides for the retention of adequate reactor coolant inventory to prevent core uncovery or loss of core cooling. The reactor coolant water that is collected is cooled in the CNV and is passively returned to the reactor vessel by natural circulation through the ECCS described in Section 6.3. The analyses results are presented in Chapter 5 of the containment response analysis methodology (CRAM) TeR (ADAMS Accession No. ML17009A490) to show that the calculated peak containment pressures and temperatures remain below the CNV internal design pressure and temperature. The methodology addresses LOCAs, valve-opening non-LOCA DBEs, and secondary pipe breaks; and potential single failures have been considered in the methodology. The supporting M&E release analyses for the primary and secondary systems are presented in DCA Part 2, Tier 2, Sections 6.2.1.3, “Mass and Energy Release Analyses for Primary System Release Events,” and 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary-System Pipe Ruptures Inside Containment,” respectively. The containment response to the spectrum of breaks is specified in DCA Part 2, Tier 2, Table 6.2-2, “Containment Response Analysis Results.” The peak containment pressure occurs as a result of an inadvertent RRV actuation, and the peak containment temperature occurs as a result of an RCS injection line break. The peak pressure and CNV wall temperature results for secondary system line-break events are bounded by the LOCA results. The containment is also designed so that CNV pressure and temperature are rapidly reduced and maintained at acceptably low levels following postulated M&E releases, including LOCAs, into containment (GDC 38, “Containment Heat Removal”). The containment heat removal function is described in DCA Part 2, Tier 2, Section 6.2.2.

ITAAC: The ITAAC associated with this evaluation of NuScale DCA Part 2, Tier 2, Section 6.2.1.1 are provided in DCA Part 2, Tier 1, Table 2.3-1, “Containment Evacuation System Inspections, Tests, Analyses, and Acceptance Criteria,” and Table 2.1-4, “NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria.”

Technical Specifications: The TS associated with this evaluation of NuScale DCA Part 2, Tier 2, Section 6.2.1.1 are provided in Section 3.6.1, “Containment,” and Section 3.5.3, “Ultimate Heat Sink.”

Technical Reports: “Containment Response Analysis Methodology,” TR-0516-49084, Revision 0 (ADAMS Accession No. ML17009A490), January 2017, NuScale Power, LLC.

The unique nature of the NuScale power module (NPM) containment design and heat removal systems necessitates development of a specific CRAM that is described in TR-0516-49084. This TR describes the thermal-hydraulic analysis methodology for primary and secondary system M&E releases into the CNV of the NPM as well as the resulting pressure and temperature response of the CNV to support DCA Part 2, Tier 2, Chapter 6 and meet the applicable regulatory guidance, including the DSRS for the NuScale small modular reactor design, Section 6.2.1 (ADAMS Accession No. ML15118A922). A spectrum of design-basis M&E release events is analyzed that bounds all of the LOCAs and valve-opening transients in

the primary system and all secondary-system pipe-break accidents. The CRAM uses conservative initial conditions and boundary conditions. The limiting peak containment pressures and temperatures are shown to be less than the design pressure (1,050 psia) and the design temperature (550 degrees F) of the CNV. The qualification of the LOCA and non-LOCA methodologies are presented in the respective topical reports (ADAMS Accession Nos. ML17004A138 and ML17222A827), in particular the comparisons to separate effects tests and integral effects tests that are also applicable to the CRAM. The differences in the NRELAP5 simulation models used in the CRAM as compared to the LOCA and non-LOCA models, along with the rationale for the selection of conservative initial and boundary conditions, are the subject of this TR. Analysis results presented for the limiting cases, along with nominal condition case results, demonstrate conservatism in initial conditions. The longer term response for equipment qualification is not within the scope of this report.

Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 4, as it relates to SSCs important to safety to be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs

- GDC 5, “Sharing of Structures, Systems, and Components,” as it relates to SSCs important to safety, which shall not be shared among nuclear power units or modules in a single power unit unless it can be shown that such sharing will not significantly impair their ability to perform their safety or risk-significant functions, including, in the event of an accident in one unit or module, an orderly shutdown and cooldown of the remaining units or modules

- GDC 16, as it relates to the reactor containment and associated systems being designed to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require

- GDC 50, as it relates to the reactor containment structure and associated heat removal system(s) being designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any LOCA without exceeding the design leakage rate and with sufficient margin

- GDC 38, as it relates to the containment heat removal system(s) (CHRS) function to rapidly reduce the containment pressure and temperature following any LOCA and maintain them at acceptably low levels

- GDC 13, “Instrumentation and Control,” as it relates to instrumentation and control, requires instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation and for accident conditions as appropriate to assure adequate safety

- GDC 64, “Monitoring Radioactivity,” as it relates to monitoring radioactivity releases, which requires means be provided for monitoring the reactor containment atmosphere for radioactivity that may be released from normal operations and from postulated accidents

- 10 CFR 50.34, "Contents of Applications; Technical Information," paragraph (f)(3)(v)(A)(1), "Additional TMI Related Requirements," as it relates to containment integrity being maintained during an accident that releases hydrogen generated from a 100-percent fuel-clad metal-water reaction accompanied by hydrogen burning

- The regulation in 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission's (NRC's) regulations.

The guidance in DSRS for NuScale Design Section 6.2.1.1.A, "PWR Dry Containments, Including Sub-Atmospheric Containment," lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections.

Technical Evaluation

NuScale needs to demonstrate that the NPM containment design methodology treats the various safety-significant thermal-hydraulics phenomena conservatively, thus maximizing the calculated containment peak pressure and temperature while minimizing heat removal from the CNV. The following subsections incorporate the graded staff review narrative applicable to this area of review.

6.2.1.1.1.1 Peak Calculated Containment Pressure/Temperature (GDC 50, 16, and 38)

To satisfy the requirements of GDC 16 and 50 on sufficient design margin, the applicable NuScale DSRS criterion prescribes at least a 10-percent margin between the limiting peak calculated containment pressure and the containment design pressure, following a LOCA or a steam or feedwater line break (FWLB). Design margins of less than 10 percent may be sufficient, provided appropriate justification is provided. DCA Part 2, Tier 2, Section 3.1.4.9, states that because the NuScale CNTS design has the CNV essentially immersed, it is capable of removing the thermal energy from the containment for accident conditions. The application goes on to state that the design configuration of the CNV and UHS provides the capability to remove containment heat for accident conditions to establish low containment pressure and temperature and maintain these conditions below the design values for an indefinite period with no reliance on operator action, active components or electrical power. The CHRS supports the containment function by minimizing the duration and intensity of the pressure and temperature increase following an accident, thus lessening the challenge to containment integrity.

To meet GDC 16, 38, and 50 relevant to the containment design basis and guided by the NuScale DSRS Section 6.2.1.1.A, the staff reviewed the applicant's analytical models and assumptions used in the CRAM to determine if the licensing-basis safety analyses are acceptably conservative. Specifically, the staff assessed the conservatism of the licensing-basis models, constitutive/closure relations, model input parameters, and initial/boundary conditions used for the applicant's NPM containment response analyses, as well as the experimental data used to validate the accident phenomenology to determine whether the results are valid over the applicable range of DBE conditions. Validation of the NRELAP5 capabilities for modeling peak containment pressure and temperature through NuScale Integral System Test (NIST)-1 testing is a major part of the staff review.

6.2.1.1.1.1.1 NuScale Containment Heat Removal System Design

Following a DBE that results in containment pressurization, the containment heat removal safety function is accomplished with effective passive transfer of containment heat to the UHS through the steel wall of the CNV and natural circulation ECCS flow after the ECCS valves open. During a primary M&E release into containment, the released reactor coolant inventory condenses and accumulates within the CNV and, following actuation of the ECCS, flows back to the RPV and to the reactor core. The capability to remove heat from the CNV (depressurization rate) is determined by the heat transfer rate from the CNV to the reactor pool. The peak containment pressure and temperature resulting from an M&E release in containment depend upon the size and location of the postulated release. In concert with the CIVs and passive containment isolation barriers, the CNV serves as a final barrier to the release of fission products to the environment (GDC 16). The containment heat removal function is described in Section 6.2.2 of this SER.

6.2.1.1.1.1.2 Break Spectrum and Single Failures

The spectrum of postulated M&E release events analyzed for the NPM using the CRAM TeR (ADAMS Accession No. ML17009A490) is the same break spectrum as considered by the LOCA TR (ADAMS Accession No. ML17004A138). Consistent with DSRS Section 6.2.1.1.A, paragraph III.4, which guides the staff to review assumptions used in the containment response analysis to determine if the analyses are acceptably conservative, the staff reviewed the applicant's assumptions for power availability, initial and boundary conditions, and single failures. The staff noted that the analysis consideration did not assume single failure of an inadvertent actuation block (IAB) valve to keep an ECCS valve closed until the RPV to containment differential pressure is above the inadvertent block set point. Whether the IAB is subject to single failure is still unresolved, and the staff provided SECY-19-0036 (ADAMS Accession No. ML19060A162) requesting the Commission for guidance on how the staff should treat the IAB. If the Commission decides that the IAB is subject to the single-failure criterion and DCA Part 2, Tier 2, Chapter 15, should be analyzed consistent with past practices (e.g., not crediting non-Class 1E power), the sequence and consequence of many events, including that associated with peak containment pressure presented in DCA Part 2, Tier 2, Chapter 6, Revision 2, will be different. As the decision on how the single failure criterion applies to the IAB is still unresolved**, RAI 8815** is being tracked as **Open Item 15.0.0.5-1**.

For postulated secondary system pipe ruptures, a limiting single failure is considered, including main steam isolation valve (MSIV) or feedwater isolation valve (FWIV) failure. For the NuScale design, full power and the maximum break size at each break location are the limiting conditions. Initial and boundary conditions are selected to maximize containment pressure and temperature response. The applicant's sensitivity results show that, in some scenarios, the consideration of no single failure provides a more limiting result. The maximum break opening area is modeled such that M&E releases to containment are maximized. The staff noted that the RVV nozzles and reactor recirculation valve nozzles were not considered in the spectrum of possible break locations. A key staff concern was whether the break spectrum was adequate. Section 15.6.5 of this SER discusses the staff's review of this area and notes **RAI 8785, Question 15.06.05-1**, and **RAI 9358, Question 03.06.02-17**, remain unresolved and are being tracked together as **Open Item 03.06.02-1**.

To establish the peak calculated containment pressure and temperature, Table 3-3 in the CRAM TeR (ADAMS Accession No. ML17009A490) describes five cases of primary system M&E release used to calculate the maximum CNV pressure and maximum containment wall

temperature for each case. The staff performed smart-samples for confirmatory and sensitivity CNV analyses to support making safety findings regarding the NuScale design under DBA conditions, using the M&E release data submitted by the applicant (ADAMS Accession No. ML18310A090) pertaining to five primary-side DBEs (breaks in RCS discharge line, injection line, and high point vent line; and inadvertent openings of an RVV and RRV); and two secondary-side DBEs (main steamline break (MSLB) and main FWLBs). The transmitted spreadsheets were used to perform confirmatory calculations of the DBA containment pressure and temperature loads. The staff's confirmatory calculations predicted a lower peak CNV pressure and wall temperature than the corresponding DCA values for the respective pressure and temperature limiting cases. However, the applicant's response to **RAI 9482, Question 06.02.01.01.A-18,** dated January 28, 2019, (ADAMS Accession No. ML18304A132) reported a CNV response analysis model change to incorporate an additional 65-lb of noncondensables released from the RPV into the CNV during the event. If the RPV noncondensable gas is ignored, confirmatory analysis implies that the applicant has met the DBA requirements. However, as explained in Section 6.2.1.1.4.1.8 of this SER, the issue of noncondensable gas discharge to the containment following DBAs is being tracked as **Open Item 6.0.2-1**.

To confirm the accident progression caused by inadvertent opening of an RRV, the staff issued RAI **9357, Question 06.02.01.01.A-4,** requesting the reason for the increasing containment vapor temperature in CRAM TeR Figure 5-31 during the period of zero to 50 seconds that precede the transient. In its response (ADAMS Accession No. ML18310A090), the applicant explained that the cause of the steadily increasing CNV vapor temperature was caused by an error in the initial CNV atmosphere pressure conditions specified during the 50 seconds before the inadvertent opening of an RRV event. The applicant corrected this error and provided revised figures in the **RAI 9357, Question 06.02.01.01.A-4,** response, dated October 26, 2018, including markups to CRAM TeR Figure 5-31, which are being tracked as a **Confirmatory Item 06.02.01.01.A-4**.

6.2.1.1.1.1.3 NRELAP5 Design-Basis Modeling Decks

The CRAM to determine peak calculated CNV pressures and temperatures for both primary and secondary system events uses the NRELAP5 system thermal-hydraulic code with conservative initial and boundary conditions. The NRELAP5 code is a NuScale modified version of the Idaho National Laboratory (INL) RELAP5-3D computer code (see Section 15.0.2.2 of this SER) that is used for both LOCA and non-LOCA transient analyses. NuScale developed the CRAM to determine CNV peak pressure and temperature by using the NRELAP5 code to analyze the M&E release transient to predict the bounding containment pressures and temperatures.

The staff found that the NRELAP5 model used in the containment response analyses for primary system CNV response analyses is consistent with the model used in the LOCA TR (ADAMS Accession No. ML17004A138), and the NRELAP5 model used in the secondary system pipe break CNV response analysis is similar to the modeling described in the Non-LOCA transient analysis methodology (ADAMS Accession No. ML17222A827). The key differences between the primary system release and secondary system break CNV response analysis models are model biasing used to maximize containment peak pressure and peak temperature for each event. The LOCA TR describes benchmarks of the NRELAP5 code to separate and integral effects tests to demonstrate the capability of the code to model LOCA, which can be extended to support M&E release events. The LOCA TR is currently under staff review and is being tracked as **Open Item 15.0.2-2**.

6.2.1.1.1.1.4 Conservatism in the NuScale Power Module Containment Vessel Model and Initial Conditions (Plant-Specific Design Parameters)

Initial conditions for the spectrum of primary system release containment response analyses are selected to ensure conservative CNV peak pressure and peak temperature. The energy sources are maximized, and energy sinks are minimized. The CRAM includes the following elements for modeling heat transfer from the CNV inside to the reactor pool through the CNV wall, to ensure a bounding peak CNV pressure and temperature response following a LOCA or valve-opening event.

- radiative heating of CNV to maximize the inside surface temperature and thereby minimize the condensation rate
- high CNV pressure to maximize noncondensable gas concentration
- heat transfer from the CNV outer surface to the CNV liquid pool
- condensation on CNV inside diameter including the effects of noncondensable gas
- conservative low heat transfer from CNV outside diameter to the reactor pool
- low reactor pool level
- high reactor pool temperature

The CRAM considers all of the relevant sources of energy including the following:

- core power equal to rated thermal power plus uncertainties
- conservative decay heat model
- maximized RCS volume and fuel stored energy
- maximized secondary system stored energy and energy resulting from feedwater pump runout

GDC 50, among other things, requires that consideration be given to the potential consequences of degraded ESFs, such as the CHRS and the ECCS, the limitations in defining accident phenomena, and the conservatism of calculation models and input parameters in assessing containment design margins. The initial conditions and assumptions are based on the range of normal operating conditions with consideration given to maximizing the calculated peak containment pressure and temperature.

The staff reviewed the conservatisms in the NPM initial conditions and issued **RAI 9482, Question 06.02.01.01.A-18,** requesting that the applicant identify why 2 percent volume adjustment is justified while using an NRELAP5 base model that had been superseded (see Section 15.6.5 of this SER relative to RAI 9325). The applicant was also asked to update the base model and resubmit or justify how the peak CNV pressure and temperature results remain conservative with an outdated base model. In its response to **RAI 9482, Question 06.02.01.01.A-18** (ADAMS Accession No. ML18304A132), the applicant stated the minimum containment free volume is 6,000 ft^3 and reevaluated the analyses incorporating containment geometry changes and new design changes with allowance for RCS thermal

expansion. The applicant determined that the overall limiting peak CNV pressure, resulting from an inadvertent reactor recirculation valve opening with loss of normal alternating current (ac) and direct current (dc) power, was 986 psia. This also credited a reanalysis of the vessel stresses that increased the CNV design pressure to 1,050 psia, providing approximately 6 percent margin.

The NPM design changes included (1) reducing initial pool temperature from 140 degrees F to 110 degrees F, (2) increasing initial pool level from 55 ft to 65 ft, and (3) increasing the ECCS high CNV level actuation setpoint. Additionally, the applicant provided draft markups of changes to the DCA Part 2, Tier 2, Chapters 1, 3, and 6 and the TS, as well as Revision 1 of the CRAM TeR. The staff reviewed these results and determined that the thermal hydraulic M&E release analysis was acceptable; however, the staff found that the stress reanalysis included plastic deformation. In audit meetings in December 2018, the staff questioned the NRELAP5 modeling changes and discussed concerns with use of plastic deformation and noted other DCA Part 2, Tier 2, sections that would be affected by this change in design pressure. In its supplemental response **to RAI 9482, Question 06.02.01.01.A-18**, dated January 28, 2019 (ADAMS Accession No. ML19028A413), the applicant stated the CNV ultimate design pressure is 1,050 psia and provided draft markups of additional DCA Part 2, Tier 2, sections. The information supporting raising the CNV ultimate design pressure is detailed in Section 3.8.2.4.3 of this SER related to **RAI 9362, Question 03.08.02-16**, and is **Open Item 03.08.02-1**.

During the audit, the staff requested justifications for the initial CNV wall and head temperature distribution used for the limiting transient, and the pool-side natural convection heat transfer modeling. NuScale provided it; the staff audited the additional information and found it acceptable because it demonstrated conservatism in the methodology for initializing containment wall temperature under normal operation. However, NuScale was asked to supplement the response to **RAI 9482, Question 06.02.01.01.A-18**, with this information, and as such, this issue is being tracked as **Open Item 6.2.1-2**.

The staff noted that DCA Part 2, Tier 2, Table 6.2-1, on containment design and operating parameters did not include containment free volume, and that Table 6.5-1 reported containment free volume as a value that is higher than the nominal value used in the NRELAP5 base model, as confirmed by the staff through an audit (ADAMS Accession Nos. ML17087A077, and ML18177A087). Therefore, the staff issued **RAI 9482, Question 06.02.01.01.A-19,** to request that the applicant update DCA Part 2, Tier 2, to have a consistent containment free volume. In its response to **RAI 9482, Question 06.02.01.01.A-19** (ADAMS Accession No. ML18304A128), the applicant stated the minimum containment free volume is 6,000 ft^3, which allows for RCS thermal expansion, piping, valves, cabling, and miscellaneous components such as platforms and ladders, and provided markups of Table 1.8-2, “Combined License Information Items,” and Table 6.5-1, “Containment Vessel Key Attributes.” Based on the information provided, the staff accepts the response to **RAI 9482, Question 06.02.01.01.A-19,** and the incorporation of the associated DCA markups of the revised DCA Part 2, Tier 2, Tables 1.8-2 and 6.5-1, provided with the RAI response is being tracked as a **Confirmatory item 06.02.01.01.A-19**.

Further, in performing its review to confirm ITAAC had been appropriately identified for this review area, the staff issued **RAI 9467, Question 06.02.01.01.A-13,** to request that the applicant provide an ITAAC to verify its as-built value to conservatively bound the value assumed in the design-basis containment analyses. In its response to **RAI 9467, Question 06.02.01.01.A-13** (ADAMS Accession No. ML18156A578), the applicant stated that, because containment net free volume is not a top-level design feature in DCA Part 2, Tier 2, Section 14.3.2.1.1, “Tier 1 Design Descriptions Are Limited to the Top-Level Design Features,”

there is no ITAAC for this parameter. However, NuScale added COL Item 6.2-3 in response to **RAI 9482, Question 06.02.01.01.A-18** (ADAMS Accession No. ML18304A132) directing a COL applicant to use the as-built CNV free volume in an analysis confirming that the peak calculated CNV pressure and temperature are bounded by the containment design pressure and temperature values. The staff considers the COL item to be appropriate for this case, as it serves to remind the COL applicant that such an analysis is necessary at the COL stage to incorporate margin to account for the as-built configuration. Incorporation of the COL item is being tracked as **Confirmatory Item 06.02.01.01.A-18**.

6.2.1.1.1.1.5 Test Data Review

The staff assessed the ability of the applicant's analytical tools and conservatism of the models used in the licensing-basis safety analyses to meet the aspects of GDC 16, 38, and 50; and 10 CFR 52.47, "Contents of Applications; Technical Information," and 10 CFR 50.43(e), relevant to the containment design basis over the applicable range of DBE conditions. The thermal-hydraulic phenomena pertinent to NuScale DCA Part 2, Tier 2, Section 6.2, containment DBE analyses are the heat transfer from the CNV to reactor pool (including condensation on the inner surface of the CNV, conduction through the CNV wall (represented by the heat transfer plate (HTP) in the NIST-1 testing), and the convection to the reactor cooling pool in NIST-1).

The applicant constructed a scaled facility of the NPM at Oregon State University, called NIST-1, to assist in validation of the NRELAP5 system thermal-hydraulic code. The NIST-1 facility provides realistic test data for modeling validation, and, as there are no other counterpart tests, the NIST-1 testing is critical for NRELAP5 validation. Validation of NRELAP5 with the set of NIST-1 tests provides confidence in the code's ability to predict containment responses; however, there is an additional step of evaluating the scaling and distortions; these are addressed primarily in the LOCA TR (ADAMS Accession No. ML17004A138). The staff's review of the test data is being performed under the LOCA TR review, which is ongoing and is being tracked as **Open Item 15.0.2-2**. The staff conducted an audit of the applicant's test data and validation, and the relevant portions of the review are described in the subsections below.

6.2.1.1.1.1.5.1 NIST-1 Test Data Scaling Distortions Relevant to NuScale Containment Design

In the LOCA TR, the applicant describes the scaling portion of its methodology. Section 4.1 of the CRAM TeR (ADAMS Accession No. ML17009A490) addresses some scaling distortions for the primary and secondary system releases. During review, the staff identified several additional distortions and discrepancies that affect peak containment pressure, and some of them were not adequately addressed in audit discussions with the applicant. The staff issued several RAI questions regarding the scaling methodology described in the LOCA TR. Section 15.0.2 of this SER also provides a summary of the staff's evaluation of the code and methodology. The LOCA TR is currently under staff review and is being tracked as **Open Item 15.0.2-2.**

In addition to the scaling questions related to the LOCA TR, the staff issued **RAI 9494, Question 06.02.01.01.A-16**, to request that the applicant provide an integral estimate of the uncertainty of peak containment pressure in the NuScale design and address the concerns identified in **RAI 9494** in addition to **RAI 9208**. The staff concerns identified in **RAI 9494** mainly pertained to the differences between the NIST-1 and NPM initial and boundary conditions and procedures, as well as in the resulting measured and predicted containment pressure responses. The NIST-1 distortion and NPM scaling concerns are mainly about initial CNV and RPV pressures, timing of ECCS actuation, condensation heat transfer, and CNV wall thickness and material properties. The applicant was asked to provide an explanation for the

discrepancies in terms of distortions in the NIST-1 design or code's scaling up toward NPM. NuScale submitted a response to **RAI 9494, Question 06.02.01.01.A-16** (ADAMS Accession No. ML19067A286), which is currently under staff evaluation, and **RAI 9494, Question 06.02.01.01.A-16,** is being tracked as **Open Item 6.2.1-3**.

6.2.1.1.1.1.5.2 Applicability of the NIST-1 Validation to the CRAM Technical Report

Section 2.0 of the CRAM TeR (ADAMS Accession No. ML17009A490) states that the qualification of the LOCA and non-LOCA methodologies presented in both LOCA and non-LOCA TRs, and, in particular, the comparisons to separate effects tests and integral effects tests, are applicable for the CRAM. Thus, NuScale relies on the qualification documented in these two topical reports as part of its CRAM. The LOCA and non-LOCA methodologies discussed in Section 15.0.2 of this SER are currently under staff review and are being tracked as **Open Items 15.0.2-2** (LOCA TR) and **15.0.2-4** (non-LOCA TR).

In its response to **RAI 9494, Question 06.02.01.01.A-17** (ADAMS Accession No. ML19009A551), the applicant stated that, in the methodology described in the CRAM TeR, the qualification of the NRELAP5 code to predict the NPM containment response is based on the code qualification and the NPM plant modeling approach that is described in the LOCA TR (ADAMS Accession No. ML17004A138) for primary side pipe breaks and reactor valve opening events; and in the non-LOCA TR (ADAMS Accession No. ML17222A827) for MSLB and FWLB events. The response also included the associated markups of the CRAM TeR (ADAMS Accession No. ML17009A490) to document that the CRAM is an extension of the NuScale LOCA, valve opening event, and non-LOCA methodologies, and that the qualification of the NRELAP5 code to predict the NPM containment response also extended to valve opening events. The markups further clarified that newly added Appendices B and C to LOCA TR (ADAMS Accession No. ML17004A138) describe the methodology for analyzing the valve opening events, which is the starting point for developing the containment response analysis model. As discussed in Appendix B.7, there are no significant differences in physical phenomena between the LOCA and valve opening events for the NPM. Therefore, the phenomena identification and ranking tables (PIRTs) developed for the LOCA and non-LOCA methodologies are also applicable to the valve opening events and CRAM. The **RAI 9494, Question 06.02.01.01.A-17**, response and the associate markups also emphasized that the NRELAP5 simulation model used for the CRAM is also similar to the NRELAP5 simulation models used for the LOCA, valve opening event, and non-LOCA methodologies. These NPM plant models provide an appropriate initial model to which changes are made to maximize containment pressure and temperature response to primary and secondary system release events, as described in the CRAM TeR.

Based on the information provided in the response, the staff concludes that the qualification of the LOCA, valve opening event, and non-LOCA methodologies presented in the LOCA and non-LOCA TRs are applicable to the CRAM. In particular, this includes the comparisons to separate effects tests and integral effects tests. Further, the CRAM TeR references these methodologies and identifies and justifies differences for the containment response analysis when compared to these methodologies. Therefore, the staff accepts the response to **RAI 9494, Question 06.02.01.01.A-17**, and the incorporation of the associated CRAM TeR markups is being tracked as a confirmatory item.

6.2.1.1.1.1.5.3 Evaluation of Test Data

The NuScale DCA identifies an inadvertent RRV opening (a credible anticipated operational occurrence (AOO) and not a LOCA) as the most limiting DBE for the calculated peak CNV

pressure (986 psia) in the NPM caused by the quickest loss of coolant from the reactor vessel liquid space before the ECCS is actuated. NuScale performed integral testing of an RRV inadvertent opening, and the staff conducted an audit of the results. The evaluation of the results is part of the ongoing review of the LOCA TR, which is being tracked as **Open Item 15.0.2-2**.

In its audit of the HP-43 and HP-49 test data, the staff found an inconsistency in the NIST-1 NRELAP5 core rod modeling. The rod model was updated to use recent rod centerline temperature data but apparently used the maximum of the temperature data rather than their average value. This issue remains unresolved and will be addressed as a part of the LOCA TR SER. The LOCA TR is currently under staff review and is being tracked as **Open Item 15.0.2-2**.

6.2.1.1.1.1.6 Liquid Thermal Stratification inside the Containment Vessel

As shown in the NuScale DCA Part 2, Tier 2, Table 6.2-2, "Containment Analysis Response Results," the limiting transient for containment peak pressure is an inadvertent opening of a RRV. The NRELAP5 analysis of this transient shows that the pressurized RCS liquid flowing into the CNV flashes into steam. As the CNV pressure increases from approximately 3 psia at the start of the transient up to the peak containment pressure, a smaller fraction of the liquid would flash to steam, since the degree of superheat is reduced as the containment pressure increases. The liquid falls to the bottom of the containment where the condensate from the flashed steam that was condensed on the cold containment wall also accumulates. The condensate eventually becomes subcooled because of CNV pressurization and the heat transfer from the liquid to reactor pool through the CNV wall.

NRELAP5 is expected to calculate the flashing/separation of the steam and liquid entering the CNV and liquid water falling to the bottom of the CNV. As the liquid temperature of water entering the CNV increases with time, thermal stratification of this water accumulating in the CNV is expected. NRELAP5 should be able to accurately calculate this potentially safety-significant, nonequilibrium thermodynamic process. This is important because overestimating the temperatures of the stratified subcooled water inventory in the lower CNV could lead to a lower calculated containment pressure that would be a nonconservative result. During its review, the staff noted that the NRELAP5 model of the NPM uses only a few large-volume nodes to represent the portion of the CNV volume below the liquid-steam interface, and it is not clear whether NRELAP5 accurately simulates the temperature stratification phenomenon in the liquid water accumulated in the CNV. The NRELAP5 peak CNV pressure will be underpredicted if the NuScale NRELAP5 model overestimates the mixing and cooling of CNV steam by this relatively cool water in the lower CNV. Thus, a conservative NRELAP5 model for temperature stratification that minimizes the steam cooling by the water accumulating in the CNV, and thus leads to a conservative distribution of energy in the CNV liquid and vapor phases, would be required in a conservative CNV peak pressure analysis.

The staff needed a greater understanding to assess the safety significance of the thermally stratified water in the CNV of the NPM during blowdown out to the time of peak containment pressure. The concern was that inadequate CNV nodalization may lead to higher water temperatures and, thus, lower peak-containment pressure. The staff issued **RAI 9380, Question 06.02.01.01.A-5**, to request that NuScale assess the impact of subcooled liquid water temperature stratification on the calculated CNV peak pressure.

NuScale responded to **RAI 9380, Question 06.02.01.01.A-5** (ADAMS Accession Nos. ML18298A361 and ML19073A242) and presented a nodalization study. NuScale provided a revised NPM analysis for the limiting peak CNV pressure DBE, which evaluated the effect of

using a set of coarser and finer axial nodalizations for the CNV volume, a finer reactor pool nodalization, and a finer CNV heat structure radial nodalization to determine the most limiting nodal representation with respect to CNV peak pressure and temperature. Based on the submitted comparison plots and table resulting from the nodalization sensitivity study, the staff concluded that sufficiently fine nodings had been used in the licensing-basis analysis, specifically in the liquid region, and the NPM containment pressure response was not very sensitive to various axial and radial nodalizations to make a significant impact on the peak CNV pressure margin. However, NuScale's response to **RAI 9380, Question 06.02.01.01.A-6** (ADAMS Accession No. ML18256A344), issued to address similar concerns about the NIST-1 test facility CNV, showed significant sensitivity of peak containment pressure to CNV axial nodalization for the HP-02 test, using the same NRELAP5 code. The staff's audit of the HP-02 test data shows a temperature stratification of the liquid water that accumulates in the NIST-1 CNV during the course of this test.

In its response to **RAI 9380, Question 06.02.01.01.A-6**, NuScale renodalized the NIST-1 containment for the HP-02 test data analysis. The RAI response showed that a finer hydrodynamic noding of the NIST-1 containment led to an increase in the containment pressure overprediction for three different HP-02 test runs and a larger deviation from the measured HP-02 test data that is not in the direction an increase in nodalization should result in. The response did not explain why the NIST-1 HP-02 pressure transient is significantly sensitive to nodalization while the NPM DBE transient is not. The overall NRELAP5 bias for the containment peak pressure prediction would also include the impact of the rod model uncertainty issue as reflected in the HP-49 audit review and test data assessment described in Section 6.2.1.1.4.1.5.3 of this SER. The staff is therefore unable to confirm NRELAP5's capability to predict CNV liquid temperature stratification and, thus, peak CNV pressure accurately for the limiting DBE. Therefore, this issue is being tracked as **Open Item 6.2.1-4**.

6.2.1.1.1.1.7 Containment Wall Condensation Heat Transfer

The CRAM considers steam condensation on the CNV walls, as discussed in CRAM TeR Section 3.2.4.1. The CRAM TeR describes the condensation correlation used for heat transfer on the CNV inside diameter and inside the DHRS heat exchanger tubes that was added to NRELAP5. All related technical details are provided in the LOCA TR (ADAMS Accession No. ML17004A138). The staff had some concerns with use of the added condensation correlation, because it is a correlation developed for very small-diameter tubes, unlike the NPM CNV's substantially larger diameter with a large annular flow area. The applicant provided the justification for use of this correlation for its NPM in the LOCA TR. The staff's evaluation of this correlation is being performed as part of the LOCA TR review, which is still ongoing. This issue is being tracked as **Open Item 15.0.2-2**.

6.2.1.1.1.1.8 Effect of Noncondensable Gases on NRELAP5 Model Prediction

In the NuScale design, the CNV atmosphere is maintained at a near-vacuum initial condition during normal operation. However, DCA Part 2, Tier 2, Section 6.2.1.3, describes a high initial CNV pressure that is used to maximize the initial noncondensable gas concentration in the CNV, and the condensation modeled on CNV inside diameter includes the effects of noncondensable gas. The CRAM TeR explains that the NRELAP5 code modeled the deterioration of the interfacial heat transfer to the condensing film on CNV wall caused by the presence of noncondensable gases. A staff audit of the NRELAP5 NPM containment analysis decks confirmed an initial noncondensable gas concentration caused by air consistent with the initial CNV pressure. The staff found it conservative to assume a higher CNV initial pressure for

analysis to maximize the adverse effect of noncondensable gases on condensation on the CNV wall inner surface. However, the applicant's **RAI 9482, Question 06.02.01.01.A-18**, response (ADAMS Accession No. ML18304A132), reported that the CNV response analysis model was updated to incorporate the effects of an additional 65-lb of noncondensables released from the RPV into the CNV during an event. During the audit, the staff asked for additional information on the assumed composition (e.g., hydrogen, nitrogen, air), release location and starting time, and M&E release rates of the additional noncondensables into the CNV during the transient. The additional noncondensable gas release was not accounted for in the spreadsheets submitted with the RAI 9357 response. The staff's preliminary confirmatory calculations indicated significant sensitivity of the peak CNV pressure to these noncondensable gas release parameters. NuScale performed a sensitivity study investigating the impact of a conservative noncondensable gas release on peak pressure results that the staff is currently auditing to reassess the applicant's peak pressure and temperature calculations. The issue of noncondensable gas discharge to the containment following DBAs is being tracked as **Open Item 6.2.1-5**. Note that, irrespective of the open issue of noncondensable gas discharge, the containment pressure is expected to reduce by 50 percent from its peak value, well within 24 hours of the initiation of DBAs.

6.2.1.1.1.1.9 NRELAP5 Modeling Results

The CRAM TeR Section 5.1 provides the results of the NRELAP5 limiting analyses of the spectrum of the five primary system M&E release scenarios for the NPM that are determined using the CRAM. Additionally, the TeR discusses the insights obtained from the sensitivity analyses used to determine the bounding set of assumptions for the primary release and single failure for peak CNV pressure and wall temperature. Similarly, Sections 5.2 and 5.3 of the CRAM TeR present the results of NRELAP5 limiting CNV pressure and temperature results for MSLB and FWLB scenarios, respectively, along with the analysis assumptions that provide these limiting results. DCA Part 2, Tier 2, Table 6.2-2, presents the results of the base case and limiting CNV pressure and wall temperature analyses for primary release (LOCA and valve opening events), as well as limiting secondary system break scenarios.

The limiting primary system pipe break (LOCA) event calculated peak containment pressure is 959 psia, resulting from a double-ended break of the RCS injection line. The analysis assumes a combined simultaneous loss of normal ac power that occurs at event initiation, conservatively biased ECCS actuation setpoints, and the single failure of one RRV to open. The peak calculated CNV pressure for the limiting LOCA is 959 psia, which provides sufficient margin to the CNV design pressure of 1,050 psia. The peak calculated CNV wall temperature for this event is 526 degrees F, providing a margin of 24 degrees F to the CNV design temperature of 550 degrees F.

The overall limiting peak calculated containment pressure, based on the M&E release spectrum analyses, is postulated to occur as the result of an AOO caused by the inadvertent opening of an RRV. The analysis of the primary system release event models an expansion of the RCS fluid into the CNV volume and includes all relevant energy input from RCS, secondary, and fuel stored energy sources, along with conservatively modeled core power and decay heat. Additional assumptions accounting for the results of sensitivity analyses, include the loss of normal ac power and highly reliable dc power system (EDSS) postulated to occur at event initiation and an IAB release pressure of 1,000 psid and minimum primary system flow. As noted in Section 6.2.1.1.4.1.2 of this SER, the analysis consideration did not assume single failure of an IAB valve. Whether the IAB is subject to single failure is still unresolved, and RAI 8815 is being tracked as **Open Item 15.0.0.5-1**.

The peak calculated containment pressure resulting from an inadvertent RRV opening event is 986 psia, providing a 6.1 percent margin to the CNV design pressure of 1,050 psia. The CRAM TeR, Section 5.1, provides figures of the CNV pressure results, and Section 5.4 discusses the analytical and design margin incorporated into the CNV design pressure and temperature limits to satisfy the requirements of GDC 16 and 50. The overall peak CNV temperature is 526 degrees F, as discussed above.

The peak calculated containment pressure resulting from a secondary side M&E release is postulated as the result of a double-ended FWLB inside containment. The analysis assumes a loss of normal ac power and dc power that occurs simultaneously with a turbine trip, an IAB release pressure of 1,200 psid, with DHRS available, and a failure of the associated FWIV to close. The peak calculated pressure is 449 psia.

The peak calculated containment temperature resulting from a secondary side M&E release is postulated as the result of a double-ended steamline break inside containment. The analysis assumes normal ac and dc power available, with DHRS available, and a failure of the associated FWIV to close. The peak calculated temperature is 433 degrees F. The peak pressure and CNV wall temperature results for secondary system line break event results are bounded by the primary system M&E release events. The LOCA peak pressure bounds the steam or FWLB peak pressures.

The CRAM TeR Table 2-2 on compliance with DSRS Section 6.2.1.1.A notes that, to satisfy the requirements of GDC 16 and 50 on sufficient design margin, the containment design pressure should provide at least a 10-percent margin above the accepted peak calculated containment pressure following a LOCA, or a steamline break or FWLB. Because of the open items noted throughout Section 6.2.1.1.4.1 of this SER, the staff is not able to conclude that the NuScale results presented in the DCA, CRAM TeR, and audited NuScale documents demonstrate that the CNV design meets GDC 16 and GDC 50 by providing sufficient margin to the CNV design pressure of 1,050 psia and CNV design temperature of 550 degrees F.

The CRAM TeR shows that the CNV pressure decreases to less than 50 percent of the peak pressure within 24 hours to satisfy the requirements of GDC 38 for rapid reduction of containment pressure. Figure 5-29 demonstrates that, for the limiting peak pressure case (inadvertent reactor recirculation valve opening event), the CNV pressure is reduced to less than 50 percent of its peak value in less than 2 hours of the accident initiation. However, because of the open items noted throughout Section 6.2.1.1.4.1 of this SER, the staff is not able to conclude that the performance of NPM containment heat removal as modeled using the CRAM meets the requirements of GDC 38 for reducing the CNV pressure rapidly after an accident and providing adequate long-term cooling.

6.2.1.1.1.1.10 Postulated Secondary-System Pipe Ruptures inside the NuScale Containment

Section 6.2.1.4 of this SER evaluates the secondary system M&E releases following a MSLB and FWLB. NuScale CRAM TeR and non-LOCA transient analysis methodology TR (ADAMS Accession No. ML17222A827) provide details about the assumptions and results for the MSLB and FWLB transients as well as the NRELAP5 models used for the design-basis analyses. The staff audited the detailed MSLB and FWLB calculation files. NuScale conservatively biased the NRELAP5 inputs to maximize the M&E release while minimizing the performance of containment heat removal, to predict the maximum containment peak pressures and temperatures for MSLB and FWLB. As stated in DCA Part 2, Tier 2, Section 6.2.1.4, NuScale analyzed MSLB and FWLB events with double-ended ruptures of the largest main steamline and feedwater line pipes. NuScale analyses of MSLB and FWLB M&E releases included

releases from the steam generator with the break that blows down into the CNV until the feedwater supply or main steamline is isolated. NuScale calculated critical flow for MSLB and FWLB transients using the Moody and Henry-Fauske models and the heat transfer correlation package in the NRELAP5 computer code.

During the review of the NuScale MSLB and FWLB DBEs, the staff identified unique features of the NuScale design compared to typical PWRs. The NuScale design uses helical coil steam generators (HCSG) with secondary coolant inside the HCSG tubes. The secondary coolant available for release in an MSLB or FWLB is limited to the inventory inside the HCSG plus additional feedwater added before isolation of the steam and feedwater coolant lines. The other unique feature of the NuScale design is that the primary coolant is also released to containment during any MSLB or FWLB transient that actuates the ECCS. Thus, the largest M&E release for MSLB and FWLB transients is from the primary coolant released by ECCS actuation and not as a result of secondary coolant release to the CNV by the MSLB or FWLB.

After an MSLB or FWLB, the reactor trips and then the RCS is cooled down by the DHRS connected to the intact HCSG. For MSLB and FWLB scenarios where the ECCS is actuated by loss of power or high CNV liquid level, the ECCS valves do not open until the differential pressure drops below the IAB setpoint. During its audit, the staff noted that NuScale assumed an IAB pressure at the upper end of the range (1,200 psia) to minimize the time before the ECCS opened to release primary coolant to the CNV. Even with this IAB setpoint assumption, the minimum time after an MSLB or FWLB before the ECCS valves open is more than 2 hours. Therefore, the staff agrees with the NuScale statements and analyses showing that the immediate opening of a RRV and release of RCS M&E inventory is substantially more limiting for maximum CNV temperature and pressure than any MSLB or FWLB event. Therefore, because LOCA and inadvertent ECCS valves are clearly more limiting than MSLB and FWLB transients, the staff found that detailed MSLB and FWLB spectrum analyses are not necessary for the NuScale design to show compliance with GDC 50, 16, and 38.

6.2.1.1.1.2 External Pressure

To satisfy the requirements of GDC 38 and 50 with respect to the functional capability of the CHRSs and containment structure under LOCA conditions, provisions should be made to protect the containment structure against possible damage from external pressure conditions that may result. The NPM containment CNV external design pressure is 60 psia, which is based on an internal pressure of 0 psia and an external pressure resulting from 30.5 m (100 ft) of pool water static pressure. The normal operating level of the reactor pool is roughly 21.0 m (69 ft), and the pool is lined to support up to 22.9 m (75 ft) of water, less than the 30.5 m (100 ft) of designed static external pressure. The NuScale CNV is normally at near-vacuum conditions, and thus there is no minimum containment pressure analysis as the CNV is designed to conditions that are more conservative than could exist in the plant design basis, as stated above. The staff concluded that the NuScale containment structure is designed for external pressure conditions that could not be exceeded and, thus, meets the GDC 38 and 50 requirements under LOCA conditions with respect to the functional capability of the containment against possible damage from external pressure conditions.

6.2.1.1.1.3 Instrumentation and Control (GDC 13) and Monitoring Radioactivity Releases (GDC 64)

NuScale DCA Part 2, Tier 2, Section 6.2.1.1.1, reports that instrumentation is provided to monitor containment parameters for normal operation, AOOs, and accidents to include temperature, pressure, isolation valve position, and liquid level (GDC 13 and 64). Section 7.1

discusses the containment parameters monitored. The environmental qualification of mechanical and electrical equipment exposed to the containment environment following a primary or secondary system M&E release inside containment is discussed in Section 3.11. The CNV instrumentation provided to monitor and record the required containment parameters and the capability to operate in postaccident environments are discussed in Chapter 7 and Section 3.11. The portions of postaccident monitoring (PAM) equipment required to be environmentally qualified are discussed in Section 3.11.2.1. Section 7.1.1.2.2, "Post-Accident Monitoring (PAM)," describes the PAM function. The staff evaluations of the instrumentation and control system design described in the application and the compliance of the CNTSs with the requirements of GDC 13 are documented in Sections 7.1 and 7.2 of this SER. The staff's review relative to GDC 64 regarding monitoring radioactivity releases are provided in Chapter 11 of this SER.

6.2.1.1.1.4 Accidents Involving Hydrogen Release and Burning (10 CFR 50.34(f)(3)(v)(A)(1))

The staff evaluation of the TMI requirement, 10 CFR 50.34(f)(3)(v)(A)(1), and findings related to combustible gases within the containment are documented in Section 6.2.5 of this SER.

6.2.1.1.1.5 Environmental and Dynamic Effects Design Bases (GDC 4)

The staff's evaluation of the NuScale design relative to the requirements of GDC 4 is provided in Section 3.6.3 ("Leak-Before-Break Evaluation Procedures") and Section 3.6 of this SER. A safety evaluation of the determination of rupture locations and dynamic effects associated with the postulated rupture of piping is documented in Section 3.6.2 of this SER.

6.2.1.1.1.6 Sharing of Structures, Systems, and Components (GDC 5)

NuScale DSRS Section 6.2.1.1.A specifies an acceptance criterion to satisfy the requirements of GDC 5 that an accident that affects one CNV in the set of reactor modules should not impair the containment integrity of any other reactor module in the common reactor building pool. In the event of a prolonged station blackout (SBO), the reactor building pool should have sufficient capacity to accommodate cooling of all reactor modules the building pool contains, assuming simultaneous shutdown of all modules from full power.

The staff determined that the NuScale design considers the risk and safety effects of the multimodule plant operation with shared systems to ensure the independence and protection of the safety systems of each NPM during all operational modes. The NuScale CIVs and barriers are designed in a way that the containment isolation components are not shared among up to 12 NPMs that include the associated balance-of-plant support systems and structures. The plant is designed such that a failure of a shared system—which are not safety-related, with exception of the UHS—does not prevent the performance of NPM safety functions. The staff does not envision an accident within the CNV of one NPM that could propagate to the other NPMs.

A key SSC important to safety common among the NuScale NPMs in regard to GDC 5 requirements for sharing SSCs is the reactor pool or the UHS system. A DBA in one NPM concurrent with a loss of all ac power is assumed to result in a shutdown of all NPMs. According to the DCA Part 2, Tier 2, Section 9.2.5, "Ultimate Heat Sink," the CNV steel wall provides for direct (passive) containment heat removal (normal, transient, or accident conditions) to the UHS, which does not rely on active components or electrical power (ac or dc). The DCA states that the capability for containment heat removal (long-term ECCS operation) is maintained assuming a single failure, without operator action for at least 72 hours. The DCA

further states that the volume of water in the pool provides the inventory for the necessary heat removal immediately after a DBA to achieve safe shutdown and maintain core cooling and containment integrity for at least 30 days postaccident, independent of ac power sources, such as in the event of SBO, and that water makeup to the UHS is not required to achieve the UHS safety functions.

Section 9.2.5 of this SER addresses the conformance of the UHS with GDC 5 to ensure sufficient long-term cooling capacity of the reactor building pool in the event of an accident in one NPM and accommodating simultaneous safe shutdown and cooldown of the remaining NPMs from full power and maintaining them in a safe shutdown condition. SER Section 9.2.5 documents the review of the assumptions and initial and boundary conditions to ensure that the most conservative conditions are assumed for analyzing the UHS's capability. Also, all water levels and temperature limits are at their most limiting allowable value. So, in effect, the events discussed in Section 9.2.5 are more limiting than the scenario typically evaluated in the extended loss of ac power analysis. SER Section 9.2.5.4.4 concludes that "GDC 5 as it relates to capability of shared UHS to perform required safety function is satisfied."

6.2.1.1.1.7 Technical Specifications

Safety analyses in DCA Part 2, Tier 2, Section 6.2.1, rely on initial containment conditions, including leakage, pressure, and temperature being within an assumed range. Regular verification of these parameters through TS provides assurance that initial conditions at the outset of a postulated accident will remain within an acceptable range. The applicant presented TS for the NuScale design in DCA Part 4, Volume 1, "Generic Technical Specifications." The TS associated with this evaluation of NuScale DCA Part 2, Tier 2, Section 6.2.1.1, are provided in Section 3.6.1, "Containment," and Section 3.5.3, "Ultimate Heat Sink." The limiting condition for operation (LCO) 3.6.1 deals with the operability of the containment. LCO 3.5.3 is included to maintain the UHS within the specified limits that ensure that the reactor pool level is remains at least 68 ft and the bulk average temperature does not exceed 110 degrees F.

The CNV atmosphere is normally maintained at a near-vacuum initial condition in the NPM design. Per Table 5-1 of the CRAM TeR, a high initial CNV pressure of 2 psia is used for the primary system release event analyses to maximize the CNV pressure to maximize the adverse effect of noncondensable gases on condensation on the CNV inside diameter. The staff notes that LCO 3.5.3, in conjunction with LCO 3.4.5 (RCS Operational Leakage) restrict the containment pressure during operation to the curve defined by DCA Part 2, Tier 2, Figure 5.2-3, "Containment Leakage Detection Acceptability." Figure 5.2-3 plots the operational limits of acceptable RCS leak detection as a function of the containment pressure and pool water temperature, which defines their normal operation domain. Based on Figure 5.2-3, the staff notes that LCO 3.4.5 does, in fact, restrict operation to slightly less than 3 psia, which is higher than the 2 psia initial condition that had been used in the limiting peak CNV pressure/temperature analyses. The staff evaluated this difference based on the supplemental sensitivity information provided by the applicant (ADAMS Accession No. ML17265A825) and found the TS controls relative to initial containment pressure acceptable.

6.2.1.1.1.8 Combined License Information Items

Table 6.2-1 lists COL information item numbers and descriptions related to containment peak pressure, from DCA Part 2, Tier 2, Table 1.8-2.

Table 6.2-1 NuScale COL Information Items for Section 6.2.1.1

Item No.	Description	DCA Part 2, Tier 2 Section
6.2-3	A COL applicant will demonstrate the containment peak accident pressure and temperature remain within limits considering necessary margin to accommodate the as-built containment free volume.	6.2.1.1.1

As discussed in Subsection 6.2.1.1.1.1.4, the staff considers the COL item to be appropriate.

6.2.1.1.1.9 Conclusion

The staff's review is incomplete pending satisfactory resolution of the open items identified above.

6.2.1.2 Containment Subcompartments

As stated in the FSAR, the NuScale CNV has no interior subcompartments and therefore no containment subcompartment analysis.

6.2.1.3 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents

Introduction

A LOCA is defined as a breach of the RCS pressure boundary. For most standard PWR designs, there is a spectrum of break sizes and locations of the RCS pressure boundary that must be considered, up to and including the large-diameter hot and cold leg piping. In the NuScale design, no large-diameter piping exists. As such, potential breaches of the RCPB are limited to chemical and volume control system (CVCS) piping (which includes inlet piping, outlet piping and pressurizer spray supply piping) RPV vent, ASME Code safety relief valves, and ECCS valves. The ECCS valves consist of the three RVVs, located at the top of the reactor vessel, and the two reactor RRVs, located near the bottom of the reactor vessel. Any breach of the RCS pressure boundary results in a M&E release into the CNV. The methodology for the analysis of a LOCA includes examining the containment response, including calculating the peak pressure which ultimately determines the containment structural integrity for DBEs. The intent of the M&E release analyses for postulated LOCAs is to maximize the M&E release to produce a conservative input to the containment design-basis analyses that are discussed in Section 6.2.1.1 of this SER. M&E releases from secondary system piping ruptures are discussed in Section 6.2.1.4 of this SER.

Summary of Application

DCA Part 2, Tier 1: There are no areas of discussion directly related to this section in DCA Part 2, Tier 1.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 6.2.1.3, provides an overview of the CRAM as it applies to the calculation of the M&E release values. As stated by the applicant, the NRELAP5 code is used to produce the containment response analyses. The model used is a modification of the LOCA evaluation model used for transient analyses in Chapter 15. Section 6.2.1.3 and Section 6.2.1.3.2, "Energy Sources—Primary System Release Events,"

describe, at a high level, the conservatisms enforced on the model to produce a maximum M&E release while minimizing containment heat removal, including the heat transfer conditions used to minimize condensation, the high initial containment pressure and temperature that act to maximize initial energy and noncondensable gas concentrations inside containment, and the various conservatisms applied to the initial stored energy inside the primary system.

DCA Part 2, Tier 2, Section 6.2.1.3.3, "Description of the Blowdown Model—Primary System Release Events," provides a high-level description of the blowdown model used in the calculation of the M&E release from the primary side breaks. Regardless of where the initial break occurs, the applicant states if power is available, the reactor trips on high containment pressure, and when the ECCS setpoint is reached (low primary system inventory or high containment level, depending on the specific transient), the ECCS valves open and, shortly after that, the limiting containment conditions are reached. In general, the descriptions in Section 6.2.1.3 are high-level descriptions of the system response, and the detailed material the staff relied on to make its finding is located in TR-0516-49084, which has been incorporated by reference into the DCA as specified in DCA Part 2, Tier 2, Table 1.6-2, and is discussed in further detail below.

ITAAC: There are no ITAAC associated with the M&E release from the RCS. ITAAC associated with the containment and the ECCS system are located in DCA Part 2, Tier 1, Section 2.1 and Section 2.8.

Technical Specifications: TS associated with the analysis assumptions described in this section are located in TS Section 3.5, "Passive Core Cooling Systems," for the UHS and ECCS valves, and in TS Section 3.6, "Containment Systems" for the containment.

Technical Reports: TR-0516-49084 provides a detailed description of the evaluation model used to produce the M&E release values. The model as used in TR-0516-49084 is a modification of the NRELAP model used for the Chapter 15 transient analyses, as described in topical report TR-0516-49422, "LOCA Evaluation Model." Qualification of the NRELAP5 model is demonstrated in this topical report, and TR-0516-49084 relies on this qualification as part of the demonstration of NRELAP5's adequacy to model the phenomena present during a LOCA within a NuScale module.

Additionally, this report describes in detail the modeling choices used to produce the peak containment pressure and temperature models, including the sources of energy considered and their biases, the break locations and conditions used in the analyses, and the attendant modeling considerations used to bias the M&E release high and the containment heat removal low. The report provides a detailed description of the module response for each break location, including a sequence of events and plots of the figures of merit for each break. As an appendix to the report, the applicant presents tables of the M&E release for the limiting primary- and secondary-side break, as well as an inventory of the passive heat sinks and their properties as used in the analysis.

Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in SRP Section 6.2.1.3, Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," and are summarized below:

- GDC 50, which requires, in part, requires that the containment and its associated systems be designed to accommodate, without exceeding the design leakage rate, the

calculated pressure and temperature conditions resulting from any LOCA with sufficient margin

- Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, which provides requirements to assure that all the sources of energy during the LOCA have been considered

The guidance in SRP and DSRS Section 6.2.1.3 lists the acceptance criteria adequate to meet the above requirements. Review interfaces with other SRP sections can also be found in SRP Section 6.2.1.3.

Technical Evaluation

As part of demonstrating the suitability of the containment design with sufficient margin, NuScale documented the M&E release and associated methodology for a spectrum of DBEs. As discussed above, the applicant provided a high-level summary in DCA Part 2, Tier 2, Section 6.2.1.3, and a more detailed discussion of the inputs, methodology, and associated modeling nuances in TR-0516-49084.

The approach chosen by NuScale for calculating containment conditions following a high-energy line break differs slightly from the traditional approach in that NuScale uses the same computer code to generate the M&E release from the RCS and to calculate containment parameters following the release. The review in this section pertains only to the portion of the calculation that involves the release of M&E from the primary side; M&E releases resulting from an initiating event in the secondary system are discussed in Section 6.2.1.4 of this report, while the calculation of the containment pressure is reviewed in Section 6.2.1.1.A of this report.

The applicability of the NRELAP5 code to simulating a LOCA transient will be evaluated separately as part of the NRC staff's LOCA topical report evaluation. NuScale uses both a base LOCA transient model and base non-LOCA transient model to perform evaluations for the design-basis transient calculations. The containment response analysis uses the base LOCA transient model, modified to include additional conservatisms to maximize the resulting containment pressure. The staff is tracking review of the LOCA methodology as **Open Item 15.0.2-2**, pending the completion of the staff's review of the LOCA topical report.

Energy sources

As part of the containment response model in NRELAP5, the applicant modeled a conservative initial inventory of energy sources within the RCS. The primary energy source, stored energy in the reactor coolant, is maximized by modeling the RCS parameters at the maximum fluid quantity at the highest feasible operating temperature. Accordingly, the applicant used conditions corresponding to the limiting high initial average coolant temperature, high pressurizer level and pressure, and reactor power level as limited by the values set forth in DCA Part 2, Tier 2, Table 15.0-6, "Module Initial Conditions Ranges for Design Basis Event Evaluation." These parameters include additional margin for uncertainty, where applicable. Considering the above, the staff finds that NuScale has appropriately chosen a conservative set of initial RCS conditions such that the M&E release is conservative.

In reviewing the initial stored energy in the RCS, the NRC staff determined that, because of the nature of the NuScale design, there was the potential for the fluid enthalpy at low reactor power levels to be greater than that at high reactor power levels when only RCS temperature is accounted for. Ultimately, the staff determined that, when all effects from a bounding high

reactor power level are considered, the stored energy in the system is most conservatively calculated at full power.

For other reactor parameters, the applicant chose appropriately conservative values that bound operating conditions. These include (1) a maximum power level accounting for uncertainty, (2) decay heat values corresponding to the American Nuclear Standards Institute/American Nuclear Society (ANSI/ANS) 5.1 1979 standard decay heat curve (including 2 sigma uncertainty), (3) bounding high stored energy in the fuel, (4) a power shape peaked to increase the initial fuel stored energy, and (5) maximum normal RCS metal temperatures. The containment initial conditions were chosen similarly: the applicant chose a containment free air volume reduced from nominal to account for uncertainty; a bounding wall temperature based on the bounding, upper allowed reactor pool temperature; and an initial containment pressure higher than the expected value during operation to account for noncondensable gases.

Although the initial containment pressure is expected to be very low (less than 0.1 psia), there appeared to be no controls on the containment pressure upper limit during operation, save for the high containment pressure reactor trip. In its response to RAI 8793, dated July 28, 2017, (ADAMS Accession No. ML17209B011), the applicant referenced TS LCO 3.5.3 for UHS. Further review by the staff indicated that LCO 3.5.3, in conjunction with LCO 3.4.5, restricted the containment pressure during operation to the curve defined by DCA Part 2, Tier 2, Figure 5.2-3, “Containment Leakage Detection Acceptability,” but only to slightly less than 3 psia for a pool temperature of 140 degrees F, which is higher than that analyzed as part of the DCA. To evaluate the effect of this increase in initial conditions, the staff requested that the applicant provide a sensitivity study documenting the effect of increasing the initial pressure. In its supplemental response (ADAMS Accession No. ML17265A825), dated September 22, 2017, the applicant more clearly explained how TS limit its containment pressure and provided the results of a sensitivity study increasing initial pressure in containment to 3 psia, which resulted in a peak containment pressure 2 psia higher than that documented in TR-0516-49084. Based on the small, documented increase in containment pressure resulting from higher initial conditions, in conjunction with the very narrow operational range above the assumed initial containment pressure, the staff finds this response acceptable.

Based on the above initial conditions, the staff finds that NuScale has adequately either conformed directly with the guidance provided in DSRS Section 6.2.1.3 or used appropriately bounding initial conditions, the combination of which serve to conservatively maximize the energy release from the primary system. Therefore, the NRC staff finds the applicant adequately selected initial and boundary conditions for the primary system M&E releases such that a limiting pressure or temperature, or both, results.

Break spectrum

With regards to the break spectrum, the applicant examined breaks at three locations. For the NuScale design, breaks include both traditional line ruptures as well as opening of the ECCS valves, which cause a containment response similar to a LOCA. Further, all breaks of sufficient size eventually cause the transient to progress to an ECCS actuation, which, for the NuScale design, ultimately leads to the limiting containment condition. The NuScale ECCS valves are designed to actuate on a combination of a signal (resulting from either the pressurizer level or containment level setpoints being reached) and reaching the IAB differential pressure setpoint threshold (nominally an 1,100 psia difference between the containment and the RCS). Because of the potential variance in the differential pressure setpoint, the timing of the ECCS actuation may change and thus change the resultant M&E release. As such, the break spectrum

analyzed includes two components: the traditional break size and location considerations and the evaluation of the ECCS actuation condition and timing.

Break locations analyzed include the RCS discharge and injection lines, pressurizer spray line, and RPV high-point degasification line, as well as the opening of a reactor safety valve, RVV, and RRV. The pressurizer spray line, RPV high-point degasification line, and reactor safety valve are all located near the top of the reactor. The RPV high-point degasification line has the largest area and all three lines contain primary coolant from nearly the same conditions. The staff views this as an appropriate consideration under the context to narrow the scope of pressurizer-space breaks considered.

In total, the applicant examined five M&E release locations. These included breaks of the RCS discharge and injection lines, the break of the RPV high-point vent degasification line, and inadvertent openings of both an RRV and an RVV. The RCS discharge line break results in the downcomer blowing down into the containment, followed by an ECCS actuation to release the balance of the RCS fluid into containment; when compared with the other cases discussed below, neither containment pressure nor temperature is limiting.

The RCS injection line break results in the riser blowing down into the containment, followed by an ECCS actuation to release the rest of the RCS fluid that releases into containment. This case represents the limiting pipe break case (as opposed to the inadvertent RRV actuation that only resembles a break and is analyzed in this section), as well as the limiting containment wall temperature case. The limiting pressure for this break is produced from a slightly different event sequence from the peak temperature, so limiting values are produced for both separately. Detailed system response curves are provided for this transient in TR-0516-49084.

The RPV high-point degasification line break and the inadvertent opening of an RVV present as very similar transients with M&E releases high in the containment followed by ECCS actuations; neither produces a limiting pressure or temperature value for the most limiting sensitivity when compared with the RCS injection line break and the inadvertent RRV opening.

The final primary side M&E release event, the inadvertent RRV opening, results in the limiting peak containment pressure. This case is most conservative with a loss of ac power, and the peak pressure occurs after the other five ECCS valves open by reaching the IAB differential pressure setpoint a little over a minute into the transient. Only the limiting sensitivity for pressure is provided as part of TR-0516-49084, but the staff audited data for each of the sensitivity cases examined by the applicant and agrees the case represented in TR-0516-49084 is the limiting scenario for the RRV opening, based on the design-basis conditions.

As part of its evaluation to determine the limiting peak pressure and temperature cases, the applicant evaluated the effect of ECCS differential pressures ranging from 900 psi to 1,200 psi differential between the containment and the RCS (the design differential thresholds for the IAB range from 1,000 to 1,200 psi). Additionally, the applicant considered the effects of power availability and both signals that initiate ECCS if power is available. This resulted in approximately six sensitivity cases per break location for these parameters. In arriving at the limiting containment conditions for each break, the applicant determined that a loss of ac power results in more limiting containment conditions, attributed to an earlier feedwater pump trip, which yields lower secondary-side heat removal and therefore higher primary-side energy. A loss of dc power, in addition to a loss of ac power, was determined to be more limiting with respect to containment pressure for the cases of interest, as well. The differential IAB setpoint pressure that yielded the most conservative containment conditions varied with the break examined. The staff audited the sensitivity studies provided by the applicant and found that

NuScale chose the most conservative conditions to impose on each of the breaks with regards to ECCS opening timing within the plant design-basis values.

As part of the analysis, the applicant considered single failures of the MSIV or FWIV to close, as well as the failure of an ECCS valve (or a single RRV and RVV at the same time) to open. Failure of an IAB to maintain closure pressure was considered a passive failure and was not included in the analysis. The staff audited the applicant's sensitivity studies, which showed that a secondary-side failure of a valve to close has a negligible impact on pressure and agrees with the applicant's determination that a failure of an ECCS valve to open would result in a lower, longer M&E release to the containment and as such would not be conservative for containment peak pressure analyses. The staff, therefore, finds that single failures have been appropriately considered for primary side M&E release analyses.

Blowdown conditions

For two-phase water-steam mixture, the applicant stated it used the Moody critical flow model, and for subcooled water and superheated steam, the applicant used the Henry-Fauske critical flow model. The Moody model is consistent with the guidance associated with the model endorsed by the staff in DSRS Section 6.2.1.3; the Henry-Fauske model is generally accepted as suitable for subcooled flow, and the staff has found the model acceptable in previous analyses reviewed by the NRC. The applicant also conservatively used critical flow conditions at the break with a discharge coefficient of 1.0, which serves to maximize the energy release. These methods provide conservative M&E releases for the breaks analyzed in this section.

Additionally, the applicant evaluated the effect of entrainment of droplets in the release fluid on a case-by-case basis. The sensitivity studies showed either negligible liquid entrainment from the break flow, or that effectively reducing the entrainment did not have an impact on the peak containment pressure. The staff audited the detailed calculations performed by the applicant and determined the treatment of entrainment did not adversely impact the calculated M&E release values from the breaks examined. The staff's audit summary report will be issued in August 2019.

The staff evaluated the M&E release table for the limiting break as provided in the appendix to TR-0516-49084. The staff performed a scoping analysis of the integral energy release as well as confirming the validity of the Henry-Fauske correlation used in determining that the M&E release provided by the applicant represented an acceptable M&E input for the conditions provided.

Conclusion

The staff finds that NuScale has fully addressed the required information related to primary coolant system mass and energy release calculations for the design. However, because of the open items related to the LOCA and non-LOCA TRs, the staff was unable to finalize its conclusions as to acceptability.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment

Introduction

Similar to a LOCA, the rupture of a secondary system pipe can result in an M&E release into the CNV. The mechanics of such an M&E release are similar for both the NuScale design and a

traditional PWR, but for the NuScale design, the system response differs somewhat because of the potential for an eventual ECCS actuation. The ECCS system consists of the three RVVs, located at the top of the reactor vessel, and the two RRVs located near the bottom of the reactor vessel. This causes the transient to behave similarly to a combination secondary-side break event and a delayed primary M&E release. The methodology for the analysis of an M&E release includes the containment response for peak pressure and temperature, affecting the containment structural integrity and equipment qualification. The intent of these analyses is to maximize the M&E release to produce a conservative input to the containment design-basis analyses that are discussed in Section 6.2.1.1 of this SER. M&E releases from primary system piping ruptures are discussed in Section 6.2.1.3 of this SER.

Summary of Application

DCA Part 2, Tier 1: There are no areas of discussion directly related to this section in DCA Part 2, Tier 1.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 6.2.1.4, provides an overview of the CRAM resulting from the calculation of secondary-side M&E releases. As stated by the applicant, the NRELAP5 code is used to produce the containment response analyses. The model used is a modification of the LOCA evaluation model used for transient analyses in Chapter 15.

DCA Part 2, Tier 2, Section 6.2.1.4, describes, at a high level, the conservatisms enforced on the model to produce a maximum M&E release while minimizing containment heat removal. In general, as stated in DCA Part 2, Tier 2, Section 6.2.1.4.1, these conservatisms are identical to those used in the M&E release calculations from the primary system, with additional considerations on the secondary side. These considerations are detailed in DCA Part 2, Tier 2, Section 6.2.1.4.3, which provides high-level information related to the initial and boundary conditions considered for the analyses. Similar to the primary side M&E release conditions discussed in the previous section of this report, DCA Part 2, Tier 2, Section 6.2.1.4, provides only a high-level description of the system response. The applicant's docketed information that the staff reviewed to make its finding is in TR-0516-49084, which has been incorporated by reference into the DCA as specified in DCA Part 2, Tier 2, Table 1.6-2, "NuScale Referenced Technical Reports," and is discussed in further detail below.

ITAAC: There are no ITAAC associated with the M&E release from the secondary side. ITAAC associated with the containment and the ECCS system are located in DCA Part 2, Tier 1, Section 2.1 and Section 2.8.

Technical Specifications: TS associated with the analysis assumptions described in this section are located in TS Section 3.5, "Passive Core Cooling Systems", for the UHS and ECCS valves; in TS Section 3.6, "Containment Systems," for the containment; and in TS Section 3.7, "Plant Systems," for the MSIVs, main FWIVs, and feedwater regulation valves (FWRVs).

Technical Reports: TR-0516-49084 provides a detailed description of the evaluation model used to produce the M&E release values. The model as used in TR-0516-49084 is a modification of the NRELAP model used for the Chapter 15 transient analyses as described in topical report TR-0516-49422. Qualification of the NRELAP5 model is demonstrated in this topical report, and TR-0516-49084 relies on this qualification as part of the demonstration of NRELAP5's adequacy to model the phenomena present during a LOCA within a NuScale module. As part of the report, the applicant provides a full accounting of the modifications made to the model to incorporate the secondary-side components, which include modeling choices made in topical report TR-0516-49416, "Non-Loss-of-Coolant Accident Methodology."

Additionally, TR-0516-49416 describes in detail the modeling choices used to produce the limiting containment pressure and temperature models for a postulated secondary-side break, including the sources of energy considered and their biases, the break locations and conditions used in the analyses, and the attendant modeling considerations used to bias the M&E release high and the containment heat removal low. The report provides a detailed description of the module response for each postulated break, including a sequence of events and plots of the figures of merit for each break. As an appendix to the report, the applicant presents tables of the M&E release for the limiting primary- and secondary-side break, as well as an inventory of the passive heat sinks and their properties as used in the analysis.

Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in the SRP and NuScale DSRS Section 6.2.1.4 and are summarized below:

- GDC 50, which requires, in part, that the containment and its associated systems be designed to accommodate, without exceeding the design leakage rate, the calculated pressure and temperature conditions resulting from any LOCA with sufficient margin.

The guidance in SRP and DSRS Section 6.2.1.4 lists the acceptance criteria adequate to meet the above requirements. Review interfaces with other SRP sections can also be found in SRP and NuScale DSRS Section 6.2.1.4.

Technical Evaluation

As part of demonstrating the suitability of the containment design with sufficient margin, NuScale documented the M&E release and associated methodology for a spectrum of DBEs. As discussed above, the applicant provided a high-level summary in DCA Part 2, Tier 2, Section 6.2.1.3, and a more detailed discussion of the inputs, methodology, and associated modeling nuances in TR-0516-49084.

The approach chosen by NuScale for calculating containment conditions following a high-energy line break differs slightly from the traditional approach in that NuScale uses the same computer code to generate the M&E release and to calculate containment parameters following the release. The review in this section pertains only to the portion of the calculation that involves the release of M&E resulting from a secondary-side break plus any associated release from the RCS from an ECCS actuation; M&E releases resulting from an RCS rupture initiating event are discussed in Section 6.2.1.3 of this report, while the calculation of the containment pressure is reviewed in Section 6.2.1.1.1 of this report.

The applicability of the NRELAP5 code to simulating a LOCA transient will be evaluated separately as part of the NRC staff's LOCA topical report evaluation and non-LOCA topical report evaluation. NuScale uses both a base LOCA transient model and base non-LOCA transient model to perform evaluations for the design-basis transient calculations. The containment response analysis uses the base LOCA transient model, modified to include additional conservatisms to maximize the resulting containment pressure and adequately model the secondary system to produce a limiting M&E release. The staff is tracking its review of the LOCA methodology as **Open Item 15.0.2-2** and the non-LOCA methodology as **Open Item 15.0.2-4**, pending the completion of the staff's review of the topical reports.

Energy sources

As part of the containment response model in NRELAP5, the applicant modeled a conservative initial inventory of energy sources within the RCS. Although the RCS is not necessarily the primary energy source, it still plays a large role in the final containment conditions. Accordingly, the applicant used the same RCS conditions (limiting high initial average coolant temperature, high pressurizer level and reactor power level) discussed in the previous section with one exception. The applicant varied the value of pressurizer pressure as a sensitivity for both steam breaks and FWLBs, and these effects are discussed further below. These parameters include additional margin for uncertainty, where applicable. Reactor power level and stored energy in the RCS metal and fuel followed similarly to the previous section, and containment initial conditions used the same limiting values as discussed in the previous section. Considering the above, the staff finds that NuScale has appropriately chosen a conservative set of initial reactor and containment conditions such that the resulting total M&E release is conservative.

Secondary-side conditions present more complex considerations for the secondary-side breaks. To maximize the resultant M&E release to the containment, the applicant took steps to maximize feedwater flow and primary-to-secondary side heat transfer. DSRS Section 6.2.1.4 stipulates that secondary water inventory also be maximized to maximize the available break flow. In TR-0516-49084, the applicant states that the secondary system pressure, temperature, and inventory is determined in the NRELAP5 steady-state model balance based on the primary-side conditions. As a result, the applicant states that conservative primary-side conditions (which are used in this analysis) drive the secondary inventory, pressure, and temperature to conservative conditions. Additionally, the analysis accounts for measurement uncertainty and deadband where appropriate. While it may be possible for brief upsets on the secondary side of the plant to exhibit higher inventory or temperature conditions, imposing such conditions would be akin to analyzing a transient during another transient, which is not a design-basis scenario. The staff reviewed the analyses associated with these statements and determined that the NRELAP5 model adequately represents the limiting set of secondary-side conditions expected during steady-state operation.

The applicant has imposed conservatisms as specified by DSRS Section 6.2.1.4 on other considerations associated with the secondary side of the plant. Both the MSIVs and FWIVs are assumed to have the longest possible stroke time and delay before actuation to maximize the potential for inventory release to the containment. The feedwater flow rate has been specified to be as high as possible based on the pump performance curve.

The staff also reviewed the impact of DHRS performance on the calculated M&E release. The applicant assumes that the DHRS actuates and performs nominally on the unaffected train. This is a realistic assumption under the circumstances of the transient but has the effect of reducing the energy release slightly during the transient, as some of the system energy is rejected to the reactor building pool. In the case of the MSLB, this has an extremely small impact on the transient because of the nature of the break and timeline (peak containment pressure and temperature are reached less than 1 minute into the transient). For the FWLB, the DHRS plays a substantially larger role, as the transient takes place over hours. Additionally, the FWLB does not by itself increase containment pressure and/or decrease primary pressure to the point where the IAB differential pressure necessary to actuate ECCS is reached, so the role of the DHRS is important in driving the transient to the eventual peak containment pressure. Based on the above discussion, the staff views the applicant's treatment of the DHRS for the secondary-side breaks as appropriate.

As part of examining the initial conditions, the applicant performed sensitivity studies to examine the effect of initial pressurizer pressure on the transient. For the MSLB, reducing initial

pressurizer pressure has the effect of reducing the time to ECCS actuation, which drives up containment pressure and temperature after ECCS occurs but does not cause it to increase above the containment pressure and temperature for the limiting MSLB with no ECCS actuation. As such, the sensitivity parameter does not affect the licensing-basis analysis for the limiting design-basis parameters as specified. For the FWLB cases, the ECCS system behaves similarly, and reducing initial pressurizer pressure has the effect of reducing the time to ECCS actuation. Because an FWLB is a less energetic initial break, the RCS fluid from an ECCS actuation is the primary driver for containment pressure and temperature. As such, although the FWLB results in an initial input of M&E into containment, the main factor in the peak containment pressure and temperature is the M&E release resulting from the ECCS actuation. Sensitivity studies performed by the applicant indicate that the earliest (and thus hottest RCS conditions) ECCS actuation results in the limiting FWLB, and so initial pressurizer pressure is minimized to decrease the time to reach the IAB setpoint for ECCS actuation.

Based on the above initial conditions, the staff finds that NuScale has adequately either conformed directly with the guidance provided in DSRS Section 6.2.1.4 or used appropriately bounding initial conditions, the combination of which serves to conservatively maximize the resultant energy release. Therefore, the NRC staff finds the applicant adequately selected initial and boundary conditions for the secondary side M&E releases such that a limiting pressure and temperature results.

Break spectrum

There are two scenarios analyzed by the applicant for a secondary side M&E release: a double-ended rupture of the largest main steam line, which results in both steam generators blowing inventory down into the containment until the MSIVs close, and a double-ended rupture of the largest feedwater line, which results in a blow down of inventory from the affected line to the containment (plus associated effects on the steam generator that is not faulted before the MSIV closes) until the FWIV isolates.

The MSLB event results in a relatively rapid sequence of events. Within a few seconds, a low steam line pressure signal causes a reactor trip, turbine trip, containment isolation, and DHRS actuation, and begins the closure of the MSIVs and FWIVs. Maintaining ac power availability results in a higher M&E release as the feedwater pump continues to discharge inventory into containment. The NRC staff audited additional detail related to other parameters, including the effect of IAB differential pressures ranging from 900 psi to 1,200 psi, failure of either an MSIV or an FWIV, and power availability. Ultimately, the limiting case occurs for the greatest inventory of secondary-system release, which occurs when an FWIV fails and requires the associated FWRV, with its slower stroke time, to close. The staff audited the sensitivity cases and agrees with the applicant that the MSLB with power available and a failed FWIV results in the highest peak containment pressure and temperature for steam line breaks.

Because of the lower enthalpy of the feedwater, the FWLB event requires an ECCS actuation to yield the peak containment pressure and temperature for breaks of this type. As discussed above, to drive the limiting M&E release from the RCS for an FWLB, initial pressurizer pressure is minimized. As the primary source of M&E for an FWLB is the RCS, other inputs were selected to minimize the time to ECCS actuation. This includes the assumption a loss of ac power and loss of EDSS coincident with the turbine trip. Additionally, maximizing secondary fluid input into containment results in an effectively lower containment volume for the ECCS actuation (and an increase in pressure), so the limiting single failure is the failure of the affected train FWIV to close, as was the case for the MSLB. Even considering all these factors, the

FWLB does not represent a limiting transient for the containment, as it resembles a lower energy secondary break with a delayed ECCS actuation; the MSLB is the limiting secondary-side break, the inadvertent RRV actuation is the limiting primary-side release, and the RCS injection line break is the limiting LOCA, as discussed in the previous section.

As part of the analysis, the applicant considered single failures of the MSIV or FWIV to close, as well as the failure of an ECCS valve (or a single RRV and RVV at the same time) to open. As discussed in Section 5.4.3 of this report, the DHRS system is not susceptible to a single active failure condition based on its design. Failure of an IAB to maintain closure pressure on the ECCS valves was considered a passive failure and not included in the analysis. As discussed in the previous section, failure of an ECCS valve to open represents a smaller M&E release. Failure of an MSIV or FWIV was analyzed in sensitivity calculations. The staff audited these sensitivity calculations and determined that failure of an FWIV presented the most limiting results for both MSLB and FWLB cases. The staff therefore finds that single failures have been appropriately considered for primary side M&E release analyses.

Based on the above considerations, the staff found that NuScale chose the most conservative conditions to impose on each of the breaks with plant conditions within the plant design-basis values.

Blowdown conditions

For subcooled water and superheated steam, the applicant stated it used the Henry-Fauske critical flow model. DSRS Section 6.2.1.4 specifies that the applicant use the Moody model for saturated conditions or another model demonstrated to be suitably conservative. The Henry-Fauske model is generally accepted as suitably conservative for superheated steam and subcooled flow, and the staff has found the model acceptable in previous analyses of this type reviewed by the NRC. The applicant also conservatively used critical flow conditions at the break with a discharge coefficient of 1.0, which serves to maximize the energy release. These methods provide conservative M&E releases for the breaks analyzed in this section.

Additionally, the applicant discussed the effect of entrainment of droplets in the release fluid. The applicant stated that no sensitivity study on entrainment in the break flow was necessary in the case of the MSLB, as there was negligible entrainment in the break; based on the quality of the steam flow during the period of interest, in concert with the entrainment sensitivity studies discussed in the previous section, the staff agrees with this assertion. For the FWLB, the applicant stated that the secondary-side break flow was relatively insignificant when compared with the primary system release and therefore the effect of entrainment was also not significant. The staff judged that, because entrainment from the primary system was modeled adequately, as discussed in the previous section, and the FWLB did not represent a limiting case, it was not necessary for the applicant to further investigate the effect of entrainment from the FWLB.

Results for the MSLB case show a spike in containment vapor temperature well in excess of both the design value and the real, expected value. The applicant states that the value was calculated correctly but that the value itself is a result of the containment model used in NRELAP5 and not reflective of the real containment conditions. A large spike in containment vapor temperature is common across codes for models of a concentrated steam release, and some spike in vapor temperature is expected under these conditions, were such a steam release to occur. Although it is difficult to ascertain what the exact peak containment vapor temperature is for the MSLB, it is ultimately not the relevant parameter of interest. The containment wall temperature is modeled in NRELAP5 and remains below the design temperature of 550 degrees F.

Based on the above considerations, the staff found that NuScale chose appropriately conservative blowdown conditions resulting in a limiting M&E release from the secondary-side breaks.

Conclusion

The staff finds that NuScale has fully addressed the required information related to secondary side break mass and energy release calculations within the containment for the design. However, because of the open items related to the LOCA and non-LOCA TRs, the staff was unable to finalize its conclusions as to acceptability.

6.2.2 Containment Heat Removal

6.2.2.1 Introduction

To maintain containment integrity and conform with the requirements of GDC 38, LWRs are equipped with systems to remove heat from the containment. These systems take various forms in different designs. In the case of the NuScale design, the containment heat removal function is an inherent characteristic of the containment, as each CNV is largely submerged in a large reactor pool shared with up to eleven other modules for each facility. This configuration results in different review emphasis areas from the staff as compared to a traditional LWR, and it also prompted the applicant to request an exemption from GDC 40, “Testing of Containment Heat Removal System.” The NRC staff reviews the capability of the system to withstand a single failure, the heat removal capability of the system, the performance characteristics of the CNV under worst-case expected conditions, the proposed inspection and testing programs, and any impacts from accident-generated debris or chemical effects on long-term core cooling.

6.2.2.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.1, “NuScale Power Module,” describes the NPM and associated systems, one of which is the CNV. The CNV, which is specified as an ASME Code, Section III, Class 1 component in Table 2.1-2, “NuScale Power Module Mechanical Equipment,” is integral in transferring heat from the reactor module to the UHS. This function is described in DCA Part 2, Tier 1, Section 3.6, “Ultimate Heat Sink,” which states that the UHS supports the containment by providing for the removal of heat by direct water contact with the CNV.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 6.2.2, “System Design,” provides a functional description of the CHRS used in the NuScale design. During a design-basis transient, heat transfer to the UHS is accomplished by physical contact between the CNV and the water in the reactor pool. In the event of a postulated M&E release into containment, the released coolant inventory, whether primary or secondary, flashes and condenses, where it is collected at the bottom of the CNV. Release events in the containment of sufficient size lead to a containment isolation and ECCS actuation, which opens the RRVs and RVVs and enables natural circulation between the reactor core, where steam is generated, and the containment volume, where the steam condenses, fills the bottom of the CNV, and returns to the reactor through the RRVs.

Because of the nature of the CHRS, there is no reliance on active components or any sort of electrical power. The applicant states that there is sufficient inventory in the UHS to remove heat from the containment for at least 30 days. The UHS is discussed in DCA Part 2, Tier 2, Section 9.2.5.

During normal operation, the CNV is maintained dry at a very low pressure (less than 1 psia). In this configuration, the primary means of heat transfer from the reactor vessel to the CNV is radiation; the CNV continues to transfer heat by conduction and convection to the reactor pool. This vacuum condition is maintained by the containment evacuation system, which is discussed in DCA Part 2, Tier 2, Section 9.3.6.

ITAAC: There are no ITAAC associated with the containment heat removal function of the CNV.

Technical Specifications: TS associated with containment heat removal are located in TS Section 3.5.3, “Ultimate Heat Sink,” and Section 3.6.1, “Containment.” Section 3.5.3 sets limits on the inventory and temperature of the reactor pool, which are required to ensure adequate heat removal to the UHS, and Section 3.6.1 requires containment be operable (ensuring inventory control within the containment).

Technical Reports: TR-0516-49084-P describes the containment response behavior, including containment heat removal, for a selection of limiting transients. This TR is discussed further in Sections 6.2.1.1.A, 6.2.1.3, and 6.2.1.4 of this report.

6.2.2.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in SRP Section 6.2.2, Containment Heat Removal Systems,” and are summarized below.

- GDC 4 as it relates to the ability of SSCs (including pumps, valves, and strainers) important to safety to accommodate the effects of and to be compatible with the dynamic and environmental conditions associated with postulated accidents

- GDC 35, “Emergency Core Cooling,” as it relates to providing abundant emergency core cooling to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

- GDC 38, which requires that:

  – the containment heat removal system be capable of rapidly reducing the containment pressure and temperature following a LOCA and to maintain these parameters at acceptably low levels

  – the containment heat removal system performs in a manner consistent with the function of other systems

  – the safety-grade design of the containment heat removal system provides suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capability to ensure that the system safety function can be accomplished in the event of a single failure

- GDC 39, “Inspection of Containment Heat Removal System,” as it relates to the design of the CHRS to permit periodic inspection of components

- GDC 40, as it relates to (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 50.46(b)(5), as it relates to the requirements for long-term cooling, including in the presence of LOCA-generated and latent debris

The NRC staff notes that NuScale has proposed a PDC, rather than a GDC, for GDC 38. The PDC proposed by NuScale is functionally identical to the GDC with the exception of the discussion related to electric power. A discussion of NuScale's reliance on electric power and the related exemption to GDC 17, "Electric Power Systems," can be found in the staff's evaluation of TR-0815-16497, "Safety Classification of Passive Nuclear Power Plant Electrical Systems," issued December 13, 2017 (ADAMS Accession No. ML17340A524).

In addition to the aforementioned regulatory requirements, NuScale's request for an exemption to GDC 40 is evaluated in this section. Requirements associated with that review not specified above include the following:

- 10 CFR 52.47(a), which requires, in part, that the [design certification] application must contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:

    (3) The design of the facility including:

        (i) The principal design criteria for the facility. Appendix A to 10 CFR Part 50, general design criteria, establishes minimum requirements for the principal design criteria for watercooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units*;*

        (ii) The design bases and the relation of the design bases to the principal design criteria.

- 10 CFR 52.7, "Specific Exemptions," which states:

    > the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of the regulations of this part. The Commission's consideration will be governed by § 50.12 of this chapter, unless other criteria are provided for in this part, in which case the Commission's consideration will be governed by the criteria in this part. Only if those criteria are not met will the Commission's consideration be governed by § 50.12 of this chapter. The Commission's consideration of requests for exemptions from requirements of the regulations of other parts in this chapter, which are

applicable by virtue of this part, shall be governed by the exemption requirements of those parts.

- 10 CFR 50.12(a), which states, in part, that the two conditions that must be met for granting an exemption are the following:

  (1) Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security.

  (2) The Commission will not consider granting an exemption unless special circumstances are present. (Circumstances are enumerated in 10 CFR 50.12(a)(2)).

The guidance in SRP Section 6.2.2, supplemented by DSRS Section 6.2.2, lists the acceptance criteria adequate to meet the above requirements. Review interfaces with other SRP sections can also be found in SRP Section 6.2.2.

In addition, some provisions of the following documents provide guidance associated with acceptance criteria that confirm that the above requirements have been adequately addressed:

- Nuclear Energy Institute (NEI) Guidance Report NEI 04-07, “Pressurized Water Reactor Sump Performance Evaluation Methodology,” Revision 0, issued December 2004 (ADAMS Accession No. ML050550138)

- NEI 04-07, Volume 2, “Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02,” Revision 0, dated December 6, 2004 (ADAMS Accession No. ML050550156)

- “Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02, ‘Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,’” dated April 6, 2010 (ADAMS Accession No. ML100960495)

- Enclosure 2, “NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation,” to “Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, ‘Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,’” dated March 28, 2008 (ADAMS Accession No. ML080230234)

- RG 1.82, Revision 4, for guidance on debris evaluation and associated effects on component performance

6.2.2.4 Technical Evaluation

Because of the unique design of the CHRS, the NRC staff evaluation of the containment heat removal function differs significantly from a traditional LWR. While the CHRS is an inherently simple system because of its passive nature as a vessel largely submerged in a large pool, the analyses demonstrating its effectiveness are of comparable complexity to a traditional LWR. Because of the tight coupling of the containment heat removal parameters to the evaluation of limiting containment transients, the NRC staff reviewed the test program and simulation models applicable to the code in conjunction with the containment peak pressure and temperature analyses. As such, the efficacy of the CHRS to meet the requirements associated with GDC 35

and 38, which corresponds to the applicant's PDC 35 and PDC 38, is discussed in Section 6.2.1.1.1 of this report.

To maintain the containment heat removal function, containment isolation is required for inventory control. This function is evaluated in Section 6.2.4 of this report.

The guidance in GDC 40 requires that nuclear power plant designs have provisions to test the CHRS such that operability is demonstrated for the full spectrum of components. The intent behind GDC 40 is to ensure the continued operability of the CHRS and verify that the system remains within the performance specifications assumed in safety analyses. In past practice, compliance with GDC 40 has been achieved by testing the safety system performance characteristics (i.e., pump flow rate and system pressure for a containment spray system) and ensuring these values are bounded by the accident analyses. The licensee does not subject the plant to a transient to test the system performance. The expectation is that, by ensuring the measurable real system performance characteristics are bounded by the analysis assumptions, the licensee demonstrates that real system performance will meet or exceed the analysis acceptance criteria, and compliance with the GDC is therefore verified by a combination of testing and analysis.

In its exemption request (Part 7 of the DCA, "8. 10 CFR 50, Appendix A, GDC 40 Testing of Containment Heat Removal System"), the applicant requested an exemption from GDC 40, periodic pressure and functional testing of the CHRS. In the technical basis of its exemption request, the applicant stated that containment heat removal is an inherent characteristic of the system ensured by the materials and physical configuration of the CNV, which is partially submerged in the reactor pool.

For the NuScale design, containment heat removal for DBEs is achieved through heat transfer from the CNV to the reactor pool; the primary parameters driving the value of the heat transfer are related to the condensation inside the CNV, which is impractical to test using the as-built configuration, as doing so involves a transient, and the through-wall conduction and convection to the reactor pool. The condensation heat transfer parameter in the analysis is based on a correlation that was validated by experimental testing. The NRC staff has evaluated the acceptability of the applicant's implementation of the condensation correlation as part of Section 6.2 of this report. System performance characteristics for the conduction and convection to the pool are either not expected to change substantially from those assumed in the analysis or are verified within specifications through regular surveillances and inspection, as is the case for surface fouling of the CNV and reactor pool parameters. The design-basis safety analyses use a presumptive limiting value for fouling with respect to heat transfer, given that the inspections will reveal degraded heat transfer conditions beyond those assumed in the analyses.

NuScale's exemption request for GDC 40 states that periodic inspections of the containment heat removal surfaces will assess surface fouling or degradation that could potentially impede heat transfer from the containment and that further details of these inspections and the conformance with GDC 39 is provided in DCA Part 2, Tier 2, Section 6.2.2. DCA Section 6.2.2 states that periodic inspections of the containment heat removal surfaces will assess surface fouling or degradation that could potentially impede heat transfer from the containment and refers to the performance of "periodic inservice inspection of the containment heat removal surfaces." The inspection scope for the containment shell is defined in DCA Part 2, Tier 2, Table 6.2-3, "Containment Vessel Inspection Elements."

Additionally, as described in FSAR Tier 2 Section 6.2.2, a group of systems act in concert to provide containment heat removal—for design-basis transients, containment heat removal is not ensured without proper ECCS actuation, which provides the means for heat removal from the reactor to the containment, and an essentially leaktight containment, which ensures sufficient inventory is available for cooling the post-transient heat load. These requirements are addressed by other measures described elsewhere (GDC 36, "Inspection of Emergency Core Cooling System," and GDC 37, "Testing of Emergency Core Cooling System," for ECCS, evaluated in Section 6.3 of this report, and GDC 50, "Containment Design Basis"; GDC 51, "Fracture Prevention of Containment Pressure Boundary"; and GDC 53, "Provisions for Containment Testing and Inspection," for containment integrity, discussed in Section 6.2.1.6 of this report).

Accordingly, the NRC staff determined that the applicant has met the underlying purpose of the rule, to verify that the performance characteristics of the CHRS remain with acceptable parameters and to ensure operability. The staff concludes that the requested exemption will not impact the consequences of a design-basis event, nor will it provide for a new, unanalyzed event. The applicant has considered the impact of the system performance throughout the design lifetime and provided adequate justification for not testing the containment heat removal function. In accordance with 10 CFR 50.12(a)(1), the staff finds that the requested exemption to GDC 40 is authorized by law, will not present an undue risk to the public health and safety, and is consistent with the common defense and security. Therefore, the NRC has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) are present, as the underlying purpose of the rule is not necessary to achieve compliance with the intent of GDC 40. As discussed above, because of the nature of the NuScale design, compliance with GDC 39 (in conjunction with meeting the requirements associated with other coupled systems, such as the ECCS and containment isolation functions) is sufficiently demonstrated for continued operability of the CHRS without the need for testing.

Long-Term Cooling

To comply with the 10 CFR 50.46(b)(5) requirements for long-term cooling, the applicant performed an evaluation of the impact of debris on long-term cooling events, summarized in DCA Part 2, Tier 2, Section 6.3.3.1. Because there are no pumps, this evaluation takes a different form than is typical, as referenced in the NEI 04-07 guidance and the associated safety evaluation endorsed by the staff. In accordance with the NuScale DCA, the applicant must do the following to address debris concerns in a similar fashion to NEI 04-07:

- consider how a potential break could generate debris
- characterize the debris that could be generated in combination with the latent debris that exists within the containment
- evaluate the potential for chemical effects caused by debris
- ensure that any debris that exists does not impact the long-term core cooling function, whether in-core, impairing any ECCS functionality, or the containment heat removal capability of the system

In DCA Part 2, Tier 2, Section 6.3.3.1, the applicant states the design minimizes debris generation by restricting the use of insulation, paint, and coatings within containment and the zone of influence (ZOI) for debris generated by fluid jet forces is limited by integral jet

impingement shield and pipe whip restraint (ISR) devices. The NRC staff audited the analysis of NuScale Generic Safety Issue (GSI)-191, “Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance,” and agrees that the material choices (such as the lack of coatings), in concert with the debris limits imposed as part of the containment cleanliness program, provides a programmatic assurance that debris-generated material will remain within the bounds assessed in the analysis. The limits imposed on the cleanliness program were not clearly identified in the FSAR; therefore, this is an **Open Item 6.2.2-1**, pending a submittal from NuScale to address the lack of clarity in the FSAR. However, as part of Section 3.6.2 of this report, the NRC staff requested further justification for the basis claiming the integral jet impingement shield and pipe whip restraint devices act to reduce pipe break ZOIs. The staff’s audit summary report will be issued in August 2019.

Additional debris is likely to be present in the containment in the form of latent debris. The applicant estimated the amount of latent debris present in the NuScale containment by comparing the surface area present inside an operating plant containment to that of the NuScale design. Using this method, the applicant estimates that 2.86 pound-mass (lbm) of latent debris might exist inside containment. Of this 2.86 lbm, 0.46 lbm is comprised of fiber, and the rest is characterized as particulate. This corresponds to a 16-percent fiber, 84-percent particulate split, which is conservative with respect to the value associated with guidance (15-percent fiber). Further, in DCA Part 2, Tier 2, Section 6.3.3.1, the applicant states limits of 2.86 lbm of debris, split into 0.46 lbm of fiber and 2.72 lbm of particulate, and 27.1 lbm of chemical species. The applicant also states that analyses demonstrate acceptable core cooling up to higher limits.

The NRC staff agrees that the method used to calculate in-containment debris is reasonable, but no acceptance criteria was provided as part of the containment cleanliness program (COL Item 6.3-1) to control the debris inside containment within the limit specified. Using the values specified above as limits in the FSAR, including the total fiber and total particulate debris allowed for the design, the staff determined that there is reasonable assurance that consequential debris limited to the values specified in the DCA will not impair long-term core cooling functionality. Further debris impacts from specific equipment existing in the design are discussed below.

The staff evaluation of the overall long-term core cooling function is discussed in Section 6.3 of this report.

Chemical Effects Introduction

To determine the compliance of the NuScale design with the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5), as they relate to chemical debris (precipitates) formed in the post-LOCA containment pool, the staff reviewed the information in DCA Part 2, Tier 2, as supplemented by letters dated November 27, 2017 (ADAMS Accession No. ML17331A994) and June 7, 2018 (ADAMS Accession No. ML18158A226). Chemical effects are corrosion products, gelatinous material, or other chemical reaction products that form as a result of interaction between the PWR containment environment and containment materials after a LOCA. SRP Section 6.2.2 does not provide specific guidance for chemical effects evaluations but references RG 1.82, Revision 3; NEI Report 04-07, and the staff’s SER of NEI 04-07 for PWR sump debris evaluations. The NuScale design conforms to Revision 4 of RG 1.82, which contains the following guidance for PWRs:

Section 2.2 Chemical Reaction Effects

d. The Westinghouse report, WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," and the limitations discussed in the associated SER provide an acceptable approach for PWRs to evaluate chemical effects that may occur in a post-accident containment sump pool (ADAMS Accession No. ML073521294, December 21, 2007).

e. Plant-specific information should be used to determine chemical precipitate inventory in containment. However, plant-specific chemical effect evaluations should use a conservative analytical approach. Additionally, "NRC Staff Review Guidance Regarding Generic Letter 04-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations," provides a general approach for PWR licensees to conduct plant-specific chemical effect evaluations (ADAMS Accession No. ML080230234, March 28, 2008).

f. WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid," is still under review by the NRC staff. When approved by the staff, it, along with the SER, will provide guidance for evaluation of chemical debris within the reactor (ADAMS Accession No. ML11292A021, October 12, 2011).

The staff subsequently approved WCAP-16793-NP with an SER dated April 8, 2013 (ADAMS Accession No. ML13084A154), including an approach for evaluating the effects of chemical debris within the reactor. Therefore, at the time the NuScale application was submitted, RG 1.82, Revision 4, and the approved TR WCAP-16793-NP-A, contain the staff's guidance for addressing chemical effects in a post-LOCA containment pool and reactor vessel. Following this guidance generally involves addressing the following principal elements:

- Identify the materials that could generate chemical precipitates.
- Determine the temperature, pH, and containment spray characteristics to determine the rate of metallic corrosion or material dissolution.
- Calculate the amount of precipitate-forming elements released over the mission time.
- Identify the type and amount of chemical precipitates formed.
- Include the chemical precipitate load into strainer and fuel assembly head-loss testing.
- Include the chemical precipitate load in analysis of deposition on the fuel.

Applicant's Approach to Addressing Chemical Effects

In DCA Part 2, Tier 2, Section 6.3.3.1, "Debris Generation and Impact Evaluation," the applicant describes how the long-term cooling evaluation considered potential chemical precipitation in debris accumulation at the RRVs, core inlet, and heated core region. DCA Part 2, Tier 2, Section 6.1, describes metallic and nonmetallic materials used inside containment, as well as materials prohibited from use inside containment.

Rather than performing a typical evaluation using the steps listed above in this subsection, the NuScale design approach is to prevent chemical effects by avoiding materials and pH buffers that have been identified as potential precipitate formers. DCA Part 2, Tier 2, Section 6.3.3.1, states that the design minimizes debris generation by restricting the use of insulation, and that chemical buffering agents are not used in containment. To demonstrate margin with respect to chemical effects, the applicant assumed that a certain quantity of chemical precipitate would be present in the post-LOCA fluid. The chemical precipitate was included in calculations of design-basis debris deposition on the fuel rods with respect to blocking the space between fuel rods and overheating the fuel cladding. In addition, the applicant calculated the total amount of chemical precipitate that could be tolerated in the post-LOCA fluid without violating the acceptance criteria. Evaluation of the effect of debris on core cooling is described in Section 6.3, "Emergency Core Cooling System," of this SER.

Source Term for Chemical Effects

The chemical effects source term refers to the interaction of materials and environment (corrosion and dissolution) that results in dissolved species that could precipitate in the post-LOCA recirculating fluid. The NuScale design approach is to prevent chemical effects by avoiding materials and pH buffers that have been identified in WCAP-16530-NP-A as potential precipitate formers. These materials, including exposed aluminum, concrete, and nonmetallic insulation materials, will not be present in containment in the NuScale design. With respect to insulation, DCA Part 2, Tier 2, Section 6.1.1.1 states that fibrous materials are not permitted in the CNV. DCA Part 2, Tier 2, Section 6.3.3.1, states that buffering agents for post-LOCA pH control are not included in containment. DCA Part 2, Tier 2, Section 15.0.2.4.6, "NuScale pHT Code," describes the analysis of post-LOCA pH.

The electrical cables in containment are not expected to be a source of solid debris or chemical precipitates. DCA Part 2, Tier 2, Section 6.1.2, "states that cables in the CNV have Type 304L stainless steel jacketing with silicon dioxide mineral insulation and no organic material. The proposed cable type has been qualified and is in use in operational and safety systems at nuclear power plants. Environmental qualification of the cables is included in DCA Part 2, Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." The staff finds it acceptable to assume no chemical effects from cables inside containment because they will be jacketed with stainless steel and qualified for the environmental conditions.

Based on the exclusion of materials and chemicals considered most likely to contribute to significant chemical effects, the proposed NuScale design has no significant source term on which to base a calculation of the chemical precipitate type and quantity. Therefore, the staff finds it acceptable for the design to be based on an assumed chemical precipitate type and quantity intended to be conservative and bounding, as described below in the next section.

Type and Amount of Chemical Precipitates

Since a goal of the design is to exclude materials and pH buffers expected to produce chemical effects, the applicant did not have a basis for calculating an amount of precipitate based on source materials and conditions (pH, T). Instead, the applicant assumed a quantity of precipitate based on calculations of the total of all debris types that could be deposited on the fuel without overheating the cladding or blocking flow passages. The staff's expectation expressed in the March 2008 guidance is that input of plant parameters should be done in a manner that results in a conservative amount of precipitate formation (Section 3.7.c.i of the guidance).

DCA Part 2, Tier 2, Section 6.3.3.1, describes estimates and design limits for the post-LOCA latent debris. The estimated fiber quantity is 0.46 lbm (207.3 g, or 5.6 g/FA,) and the estimated particulate quantity is 2.72 lbm (1230 g or 33.4 g/FA). The design limits are 0.61 lbm fiber (7.5 g/FA) and 30 lbm total particulate. The June 7, 2018, letter (ADAMS Accession No. ML18158A226), explains that the 30 lbm particulate includes both solid particles and chemical precipitates. The DCA does not identify a specific amount or composition of chemical precipitate. Debris limits normally include separate limits for solid particles and chemical precipitates because these two forms of debris play different roles in forming and clogging fibrous debris beds on strainer and fuel inlets. Because NuScale has no ECCS strainers and a fiber quantity too low to form an effective filtering bed on the fuel inlet, the particles and chemicals do not contribute to clogging, and individual limits are not necessary to prevent clogging. NuScale bases this approach, in part, on AREVA fuel assembly inlet pressure drop (blockage) testing documented in WCAP-16793-NP-A. The staff finds this acceptable with respect to chemical effects on fuel inlet blockage, based on the staff's approval of WCAP-16793-NP-A. To evaluate the potential for deposition on the fuel, DCA Part 2, Tier 2, Section 6.3.3.1, assumed an additional 271 lbm (123 kg) of chemical precipitate with the 0.61 lbm of fiber and 30 lbm of particles/chemicals. Given the design approach of preventing chemical effects, the staff finds this large assumed quantity of chemical precipitate acceptable for the analysis. The staff evaluation of in-vessel debris effects on long-term cooling is discussed in Section 6.3 of this report.

In addition to the potential effects of chemical precipitates on fuel inlet blockage and deposition on the fuel, chemical precipitates are also considered in the analysis of components outside the core. Review of these analyses is below in this section of this report, under the heading, "*Capability of Mechanical Equipment in ECCS Flowpath during Post-LOCA Operation*."

Chemical Effects Summary

The NuScale design excludes materials such as insulation and aluminum associated with the formation of chemical precipitates in laboratory testing. The design includes a combined limit for debris in the form of solid particles or chemical precipitates. With respect to chemical effects on fuel inlet blockage, the staff finds this acceptable because the low fiber debris limit prevents the formation of a filtering bed that could be clogged by chemical precipitates. With respect to chemical effects on fuel deposits, the staff finds this acceptable because a large amount of chemical precipitate beyond the design limit was included in the analyses.

Capability of Mechanical Equipment in ECCS Flowpath during Post-LOCA Operation

The objective of this review is to evaluate the effects of LOCA-generated debris, latent debris, and chemical reaction products on component performance in the ECCS flowpath during long-term cooling.

The NRC staff reviewed DCA Part 2, Tier 2, Section 6.2.2, for the evaluation and effects of LOCA-generated debris, latent debris, and chemical reaction products for potential blockage at narrow flow passages (e.g., tight clearance valves) and wear and abrasion of components for consistency with applicable NRC regulations and guidance. The staff issued several RAIs to NuScale to resolve staff questions on the information provided in the original DCA Part 2 submittal. In response to the RAIs, NuScale clarified specific information with respect to the effects of LOCA-generated debris, latent debris, and chemical reaction products on component performance in the ECCS flowpath during long-term cooling. In this SER section, the staff

focuses on the revised DCA Part 2 and its compliance with the applicable NRC regulations and guidance rather than discussing each RAI and NuScale response.

The NuScale ECCS does not use pumps or dynamic restraints. The active mechanical equipment in the ECCS are RVVs and RRVs.

DCA Part 2, Tier 2, Section 6.2.2.2, states that Tier 2, Section 6.3.2.5, describes conformance with RG 1.82 (Revision 4) and the approach used to address GSI-191.

DCA Part 2, Tier 2, Section 6.3, describes the design of the ECCS that provides core cooling during and after AOOs and postulated accidents, including LOCAs. The ECCS includes three RVVs mounted on the upper head of the RPV and two RRVs that are mounted on the side of the RPV.

The staff review is described in the following paragraphs.

The applicant provided supplemental information regarding the type, quantity, and maximum size of LOCA-generated debris, latent debris, and chemical products, as referenced in a letter dated September 25, 2017 (ADAMS Accession No. ML17268A409). The applicant stated that no LOCA debris is generated in the NuScale plant CNV; latent debris in the NuScale containment consists of fibers and particulates that remain in containment after maintenance or testing; chemicals and precipitants that may form are typically soft, nonabrasive, low-shear and readily stay in solution because of the flow conditions present within the system and can be treated like a particulate. The applicant also specified the size of quantity of the debris as follows: Latent particulates have a diameter of 10 microns, density of 100 pounds mass per cubic foot (lbm/ft^3), with a limit of 30 lbm combined [particulate (2.7 lbm) and chemical (27.3 lbm)]; latent fibrous debris is characterized as 7 micron fiber diameter, density of 62.4 lbm/ft^3, with a limit of 0.61 lbm. The staff finds this information acceptable because the type, quantity, and maximum size of LOCA-generated debris, latent debris, and chemical products that are used to evaluate the blockage in the ECCS are specified, and the methodology to determine post-LOCA debris is consistent with RG 1.82 as determined by the staff in Section 6.3 of this SER.

The applicant provided supplemental information in a letter dated September 25, 2017 (ADAMS Accession No. ML17268A409), to address the potential of blockage or reduced flow caused by the effects of LOCA-generated debris, latent debris, and chemical products on tight-clearance valves (such as RVVs, RRVs, and any throttle valves or check valves in the flowpath during long-term cooling) that may not be in the fully open position during post-LOCA operation. The applicant stated the ECCS valves (RVVs and RRVs) are open/closed valves that do not have any throttling requirements during their operation and, therefore, will not be in a partially open position and that there are no other valves in the recirculation pathway. The applicant also stated that fluid passages in the valves are large compared to the debris size (10 micron diameter) and will provide ample room for the latent debris and chemicals that are in solution to pass through without clogging. Based on the above, the staff finds this information acceptable because the fluid passages in the valves are large compared to the debris size and will not result in blockage or reduced flow from the effects of LOCA-generated debris, latent debris, and chemical products on tight-clearance valves. This information is consistent with RG 1.82 (Revision 4) and meets the regulatory requirements in GDC 38, GDC 40, and 10 CFR 50.46(b)(5).

The applicant provided supplemental information in a letter dated September 25, 2017 (ADAMS Accession No. ML17268A409), to address the quantity and type of material that will settle,

locations where it will settle, and its impact on the performance of components in the applicable systems. The applicant stated that the quantity and type of material in the post-LOCA fluid is latent particulates with a limit of 30 lbm combined [particulate (2.7 lbm) and chemical (27.3 lbm)]; and latent fibrous debris with a limit of 0.61 lbm. The applicant also stated that the fluid velocity is capable of maintaining the debris in solution; therefore, debris settling will not occur in the ECCS components. Based on the above, the staff finds this information acceptable because debris will stay in solution and will not settle and affect the performance of the ECCS components. This information is consistent with RG 1.82 (Revision 4) and meets the regulatory requirements in GDC 38, GDC 40, and 10 CFR 50.46(b)(5).

The applicant provided supplemental information in a letter dated September 25, 2017 (ADAMS Accession No. ML17268A409), to identify all small-diameter tubing/piping such as instrument lines, sensing lines, and IAB valve pressure sensing lines in the ECCS system and long-term cooling flowpath and to evaluate the effects of LOCA-generated debris, latent debris, and chemical products for potential blockage that could affect component function. The applicant stated that there are no small-diameter tubing/instrument lines subject to blockage in the path of the ECCS during post-LOCA recirculation operation. The applicant also stated that during recirculation, the IAB valve pressure sensing passage is a dead end with a vertical orientation, preventing LOCA-generated debris, latent debris, and chemical products from accumulating in the IAB valve. Based on the above, the staff finds this information, that blockage will not occur in small-diameter instrument/sensing lines, acceptable because there is no small-diameter instrument/sensing lines subject to blockage in the path of the ECCS, and the IAB valve pressure sensing passage is a dead end with a vertical orientation. Further, this information is consistent with RG 1.82 and is therefore acceptable.

The applicant provided supplemental information in a letter dated September 25, 2017 (ADAMS Accession No. ML17268A409), to address the potential effects of wear and abrasion of components from LOCA-generated debris, latent debris, and chemical products during post-LOCA operation. The applicant stated the RRVs and RVVs are the only components of the ECCS and that these valves open in response to LOCA events and are not required to close or throttle. The applicant also stated that, because of the minimal amount of debris, minimal-to-no wear or abrasion of components is expected. Based on the above, the staff finds this information, that minimal-to-no wear or abrasion of components is expected for the RRVs and RVVs, acceptable because of the minimal amount of debris in the post-LOCA fluid. This information is consistent with RG 1.82 and meets the regulatory requirements in GDC 38, GDC 40, and 10 CFR 50.46(b)(5).

Mechanical Equipment Evaluation Summary

The NRC staff concludes that the provisions in the DCA Part 2, that the ECCS and its associated components will function as designed under post-LOCA fluid conditions for the required mission time, are acceptable and meet applicable NRC regulations and guidance. This conclusion is based on the applicant having specified provisions in the DCA Part 2 that the ECCS and its associated components will function as designed under post-LOCA fluid conditions for the required mission time.

The impact of debris transported within the reactor vessel is evaluated in Section 6.3 of this report.

6.2.2.5 Combined License Information Items

There are no COL information items specified as part of Section 6.2.2; however, COL Item 6.3-1 is relevant to the issue of adequate long-term core cooling.

Item No.	Description	DCA Part 2, Tier 2 Section
COL (6.3-1)	A COL applicant that references the NuScale Power Plant design certification will describe a containment cleanliness program that limits debris within containment. The program should contain the following elements: • Foreign material exclusion controls to limit the introduction of foreign material and debris sources into containment. • Maintenance activity controls, including temporary changes, that confirm the emergency core cooling system function is not reduced by changes to analytical inputs or assumptions or other activities that could introduce debris or potential debris sources into containment. • Controls that limit the introduction of coating materials into containment. • An inspection program to confirm containment vessel cleanliness prior to closing for normal power operation.	6.3

6.2.2.6 Conclusion

Based on the above evaluation, the staff finds that the requirements of 10 CFR 50.46(b)(5) with respect to long-term cooling in the presence of LOCA-generated and latent debris and GDC 38 and GDC 39 with regards to the functional design of the containment are met. The staff evaluation of the applicant's demonstration of adequate containment heat removal to meet GDC 38 is discussed in Section 6.2.1.1.A of this report. Further, the applicant has justified an exemption to GDC 40 based on the nature of the design and associated demonstrated compliance with GDC 39, as discussed above.

6.2.3 Secondary Containment Functional Design

This Section is not applicable to the NuScale design.

6.2.4 Containment Isolation System

6.2.4.1 Introduction

The containment isolation system (CIS) consists of isolation barriers, such as valves, closed systems, and the associated instrumentation and controls required for automatic or manual initiation of containment isolation. The purpose of the CIS is to permit the normal or postaccident passage of fluids through the containment boundary while protecting against

release of fission products to the environment that may be present in the containment atmosphere and fluids because of postulated accidents.

6.2.4.2 Summary of Application

DCA Part 2, Tier 1: In DCA Part 2, Tier 1, Section 2.1, the applicant specifies the Tier 1 design description for containment isolation. In Figure 2.1-1, "Containment System (Isolation Valves)," the applicant depicts the functional arrangement of the containment isolation equipment.

DCA Part 2 Tier 2: In DCA Part 2, Tier 2, Section 6.2.4, "Containment Isolation System," the applicant describes the design bases and general description for the CIS.

ITAAC: In DCA Part 2, Tier 2, Section 14.3, Table 14.3-1, "Module-Specific Structures, Systems, and Components Based Design Features and Inspections, Tests, Analyses, and Acceptance Criteria Cross Reference," the applicant describes the proposed ITAAC that address containment isolation. In DCA Part 2, Tier 1, Table 2.1-4, the applicant lists the ITAAC associated with containment isolation equipment. Table 2.1-4 includes ITAAC No. 8 for CIV closure time and ITAAC No. 9 for the length of piping between each penetration and its associated outboard CIV.

Initial Test Program: In DCA Part 2, Tier 2, Section 14.2, "Initial Plant Test Program," Table 14.2-43, "Containment System Test # 43," the applicant describes the testing related to the containment system, to include the CIS.

Technical Specifications: In DCA Part 4, "Generic Technical Specifications—NuScale Nuclear Power Plants," Section 3.6.2, "Containment Isolation Valves," the applicant describes the TS for the CIVs.

6.2.4.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 1, "Quality Standards and Records," requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

- GDC 2, "Design Bases for Protection against Natural Phenomena," requires safety-related SSCs be designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, tsunamis, and seiches, without loss of capability to perform safety functions.

- GDC 4, "Environmental and Dynamic Effects Design Bases," requires safety-related SSCs to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and these SSCs shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids.

- GDC 5, requires SSCs important to safety not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair the ability to perform their safety functions, including, in the event of an accident in one unit, and order shutdown and cooldown of the remaining units.

- GDC 16, requires that the reactor containment and its systems establish an essentially leak-tight barrier against the uncontrolled release of radioactive materials to the environment.

- GDC 54, “Systems Penetrating Containment,” requires that piping systems penetrating the containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect their importance to safety and as it relates to designing such piping systems, a capability to periodically test the operability of the isolation valves and associated apparatus and determine if valve leakage is within acceptable limits.

- GDC 55, “Reactor Coolant Pressure Boundary Penetrating Containment,” and GDC 56, “Primary Containment Isolation,” require, in part, isolation valves for lines penetrating the primary containment boundary as parts of the RCPB (GDC 55) or as direct connections to the containment atmosphere (GDC 56) as follows:

    1) One locked-closed isolation valve inside and one outside containment; or

    2) One automatic isolation valve inside and one locked-closed isolation valve outside containment; or

    3) One locked-closed isolation valve inside and one automatic isolation valve outside containment; or

    4) One automatic isolation valve inside and one outside containment

- GDC 57, “Closed Systems Isolation Valves,” requires, in part, that lines that penetrate the primary containment boundary and are neither part of the RCPB nor connected directly to the containment atmosphere have at least one locked-closed, remote-manual, or automatic isolation valve outside containment and located as close to the containment as practical.

- The regulation in 10 CFR 52.47(a)(8), as it relates to demonstrating compliance with any technically relevant portions of the Three Mile Island (TMI)-related requirements. For this review area, the following areas are assessed: 10 CFR 50.34(f)(2)(xiv), 10 CFR 50.34(f)(2)(xv), 10 CFR 50.34(f)(2)(xix), and 10 CFR 50.34(f)(3)(iv).

- The regulation in 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the DC has been constructed and will be operated in conformity with the DC, the provisions of the Atomic Energy Act (AEA), and the U.S. Nuclear Regulatory Commission’s (NRC’s) regulations.

- The regulation in 10 CFR 52.48, “Standards for review of applications,” which requires in part that applications filed under this subpart will be reviewed for compliance with the standards set out in Part 50 and its appendices (i.e., 50.12, “Specific exemptions”).

- The regulation associated with Station Blackout, 10 CFR 50.63(a)(2), as it relates to ensuring that appropriate containment integrity is maintained in the event of a station blackout for a specified duration.

The guidance in the NuScale Design Specific Review Standard (DSRS) Section 6.2.4, "Containment Isolation System," lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other Standard Review Plan (SRP) sections. The following NRC regulatory guides are also applicable for this review:

- Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems."

- Regulatory Guide 1.155, "Station Blackout."

6.2.4.4 Technical Evaluation

The staff reviewed the containment isolation system described in DCA Part 2, Tier 2, Section 6.2.4, using the staff's review guidance provided in NuScale DSRS Section 6.2.4. NuScale DSRS Section 6.2.4 identifies the staff's review methodology and acceptance criteria for evaluating compliance with requirements related to piping systems penetrating the containment.

The NuScale design departs from portions of GDC 55, GDC 56, and GDC 57, as described in DCA Part 2, Tier 2, Section 6.2.4. The design also departs from 10 CFR 50.34(f)(2)(xiv)(E), as described in DCA Part 2, Tier 2, Section 9.3.6, "Containment Evacuation System and Containment Flooding and Drain System." For each departure discussed above, the applicant seeks an exemption. The application's exemption requests reside in DCA Part 7, "Exemptions."

In general, GDC 55, GDC 56, and GDC 57 require each line that penetrates primary reactor containment to be provided with CIV(s). In GDC 55 and GDC 56 the isolation valves location is specified as one valve inside containment and one valve outside containment. The NuScale design departs from the requirements by locating both valves outside the containment. With respect to GDC 57, the isolation barrier outside containment is specified to be an isolation valve. The NuScale design departs from the requirement by using a closed system (i.e., decay heat removal system (DHRS)) as the isolation barrier outside containment instead of the specified CIV.

The provisions in 10 CFR 50.34(f)(2)(xiv)(E) require containment isolation systems, which include automatic closing on a high radiation signal for all systems that provide a path to the environs. NuScale's design departs from the requirement by providing alternate means to reliably isolate systems that provide a path to the environs.

In addition, there are also two TMI requirements associated with this review area that the applicant considers not technically relevant. These two TMI requirements are 10 CFR 50.34(f)(2)(xv) and 10 CFR 50.34(f)(3)(iv).

As part of the staff's review for containment isolation, the applicant submitted information that supplemented the DCA by letters dated October 24, 2017 (ADAMS Accession No. ML17297B243), November 22, 2017 (ADAMS Accession No. ML17326B443), December 21, 2017, (ADAMS Accession No. ML17355A511), February 8, 2018 (ADAMS Accession No. ML18039A975), March 15, 2018 (ADAMS Accession No. ML18074A342), March 27, 2018 (ADAMS Accession No. ML18086B093), April 13, 2018 (ADAMS Accession No. ML18103A161), and June 4, 2018 (ADAMS Accession No. ML18155A596).

The staff's review of the applicant's compliance with requirements and departures from requirements is provided in the following 6.2.4.4 subsections.

GDC 1 and GDC 2

The provisions in GDC 1 and GDC 2 are applicable to the review of containment isolation for its ability to perform its safety function and prevent the release of radioactive materials to the environment.

NuScale DSRS Section 6.2.4 identifies that Regulatory Guide (RG) 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and RG 1.29, "Seismic Design Classification," provide guidance applicable to the containment isolation system designs ability to meet GDC 1 and GDC 2.

In DCA Part 2, Tier 2, Section 6.2.4, the applicant states that all CIVs are designed to ASME Code Section III and meet Quality Group A or B specifications, which is consistent with RG 1.26. In addition, as described in DCA Part 2, Tier 2, Table 3.2-1, "Classification of Structures, Systems, and Components," and DCA Part 2, Tier 2, Section 6.2.4, CIVs are designed to meet Seismic Category I requirements to satisfy the guidance specified in RG 1.29. Furthermore, DCA Part 2, Tier 2, Section 6.2.4 describes that the CIV area is also protected from natural phenomena hazards (e.g., earthquakes, winds, tornadoes, and floods) by the reactor building (a Seismic Category 1 structure). Accordingly, because the applicant's DCA conforms to regulatory guidance, the staff finds the CIVs meet the requirements of GDC 1 and GDC 2. Additional discussion on protection against natural phenomena (GDC 2) is found in Section 3 of this report.

General Design Criterion 4

The provisions in GDC 4 are applicable to the review of the containment isolation system for its ability to perform its isolation function at all times, in any environmental condition (e.g., normal operations and postulated accidents) to which the system's components may be exposed, including dynamic effects (e.g., missiles and pipe whipping).

NuScale DSRS Section 6.2.4 identifies that the reviewer evaluates the containment isolation system to perform its function in any environmental condition to which the systems components may be exposed, including dynamic effects.

In DCA Part 2, Tier 2, Section 6.2.4 the applicant states that the containment isolation system meets the requirements of GDC 4. For example, CIVs and barriers are designed to accommodate and be compatible with applicable environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and, in addition, these containment isolation components are protected against dynamic effects of missiles and pipe whip. The applicant describes missile protection in DCA Part 2, Tier 2, Section 3.5, "Missile Protection," protection against dynamic effects in DCA Part 2, Tier 2, Section 3.6, "Protection against Dynamic Effects Associated with Postulated Rupture of Piping," and environmental conditions in DCA Part 2, Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." The staff confirmed that the safety-related containment isolation function was appropriately identified in these DCA sections. The staff's finding on GDC 4 is primarily made in these interfacing review sections (e.g., 3.5, 3.6, and 3.11 of this report).

In support of the finding on GDC 4 for the CIVs, the staff evaluated the development of the environmental conditions in the region that contains the CIVs. In DCA, Part 2, Tier 2, Section 3.11, the applicant requires equipment providing the containment isolation function to be environmentally qualified. The containment isolation function achieved by isolation valves occurs outside containment and under the bioshield. The environmental conditions under the bioshield are provided in DCA Part 2, Tier 2, Appendix 3C, "Methodology for Environmental Qualification of Electrical and Mechanical Equipment," assuming a vented bioshield. To better understand this assumption, the staff audited the applicant's modeling associated with the venting aspects of the bioshield and confirmed that the applicant appropriately modeled the vented bioshield (e.g., flow area) to include a stacked bioshield configuration (one bioshield affixed atop an adjacent bioshield to support refueling activities). The staff's audit summary report is scheduled to be issued in August 2019. Therefore, because the applicant appropriately modeled the vented bioshield, the staff finds the applicant's approach supports development of the environmental conditions in the region that contains the CIVs and is therefore, acceptable.

Based on the review discussion above, the staff finds that the containment isolation system meets the requirements in GDC 4 because environmental conditions and dynamic effects have been appropriately considered in the design of the safety-related containment isolation system.

General Design Criterion 5

The provisions in GDC 5 require that SSCs important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair the ability to perform their safety functions, including, in the event of an accident in one unit, and orderly shutdown and cooldown of the remaining units.

NuScale DSRS Section 6.2.4 guidance with regard to containment isolation directs that the CIVs should be essentially independent from the other units.

In DCA Part 2, Tier 2, Section 6.2.4, the applicant states that the CIVs or other containment barriers are not shared among the other modules. The staff reviewed the DCA information described above and confirmed that the CIVs or other containment barriers are not shared among other modules.

Based on the independence of the containment isolation provisions among modules, the staff finds that the containment isolation functional design is acceptable and meets the requirements of GDC 5.

General Design Criterion 16

The provisions in GDC 16 require that the reactor containment and its systems establish an essentially leak-tight barrier against the uncontrolled release of radioactive materials to the environment.

NuScale DSRS Section 6.2.4 provides guidance on design requirements for the containment isolation system. Specifically, the containment isolation system should allow the normal or emergency passage of fluids through the containment boundary while preserving the capability of the boundary to prevent or limit the escape of fission products from postulated accidents.

In DCA Part 2, Tier 2, Section 6.2.4 the applicant states that the containment isolation components are designed to provide an essentially leak tight barrier against the uncontrolled

release of radioactive materials to the environment. For example, the containment isolation components are designed consistent with staff guidance discussed above in Sections 6.2.4.4.1 and 6.2.4.4.2 of this report and function to prevent the release of radioactive materials to the environment. In addition, DCA Part 2, Tier 2, Section 6.2.4 provides information regarding the number and location of isolation valves, actuation features, signals and closure times (these elements are discussed in additional detail in subsequent subsections of this report) that are designed to support providing an essentially leak tight barrier. The containment isolation system is also designed to support containment leakage testing in order to verify the leak tight integrity of the containment isolation system. Furthermore, the design, qualification, and testing of the isolation valves is addressed in DCA Part 2, Tier 2, Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints." The NRC staff's review of leakage testing and valve design are found in Sections 6.2.6 and 3.9.6, respectively, of this report.

In DCA Part 2, Tier 2, Section 6.2.4, the applicant states that the isolation function is accomplished given the occurrence of a single active failure in the isolation provisions. Specifically, the applicant provides Table 6.2-6, "Failure Modes and Effects Analysis Containment System," to address system failure modes. Because the applicant provided information to demonstrate that one isolation barrier remains after the occurrence of a single active failure, the staff finds that the isolation valves support establishing an essentially leak-tight barrier as required by GDC 16.

The staff also reviewed the isolation provisions for passive containment barriers, described in DCA Part 2, Tier 2, Section 6.2.4, such as flange connection closures for access and inspection ports, manways, electrical penetration assemblies, and emergency core cooling valve actuator assemblies. Each of these connection closures have double isolation barriers (i.e., double seals) with a port between the seals for periodic testing of the seal leakage rate. Having redundant seals helps to establish an essentially leak-tight barrier against the uncontrolled release of radioactive materials to the environment, as required by GDC 16. Leakage testing of isolation barriers is evaluated in Section 6.2.6 of this report. The design of the steel containment, to include a discussion of these passive containment isolation barriers, is described in DCA Part 2, Tier 2, Section 3.8.2, "Steel Containment," and is evaluated in Section 3.8.2 of this report.

Based on the discussion above, the staff finds the containment isolation system meets the requirements of GDC 16 to provide an essentially leak tight barrier against the uncontrolled release of radioactive materials to the environment.

General Design Criterion 55

The provisions in GDC 55 require, in part, that each line that is part of the RCPB and penetrates primary reactor containment shall be provided with CIVs. The combination of valves, automatic or locked closed, and the location of valves, one inside and one outside containment, are specified in GDC 55. For the isolation valve function, redundant barriers are required to account for a single active failure in the isolation provisions. This is achieved by providing two isolation valves in series.

In DCA, Part 2, Tier 2, Section 6.2.4, the applicant states that the NuScale design contains four containment piping penetrations that are subject to GDC 55. These four piping penetrations are associated with reactor coolant system (RCS) injection, pressurizer spray, RCS discharge, and reactor pressure vessel high point vent lines. While the applicant provides a design that

complies with the requirements of GDC 55 in terms of the number of valves, there is a departure from the GDC requirements with regard to valve location.

The four GDC 55 piping penetration lines provide a containment isolation design consisting of two automatic isolation valves located outside containment in series rather than locating one of the CIVs inside containment as specified in GDC 55. Therefore, the applicant requested an exemption from the requirements of GDC 55 regarding valve location (refer to Part 7 of the DCA).

The staff concludes that with both automatic valves cited as CIVs, the design is adequate for assuring redundancy in achieving containment isolation. This conclusion is supported by the series arrangement of the isolation valves and the appropriate quality of the design (e.g., ASME Code, Section III, Class 1, Subsection NB and Seismic Category I criteria), as described in DCA Part 2, Tier 2, Sections 6.2.4 and 3.1.5.6, "Criterion 55-Reactor Coolant Pressure Boundary Penetrating Containment." In addition, in DCA Part 2, Tier 2, Section 6.2.4, the applicant describes these valves as remotely actuated by an automatic signal or operator action and fail closed on loss of power; where each valve in a pair has a separate instrumentation and control division to provide independence and redundancy. Furthermore, in NuScale DSRS Section 6.2.4, the staff states that containment isolation provisions different from the explicit requirements of GDC 55 are acceptable if the differences are justified. The applicant's justification (see DCA Part 7) is that the differences in the isolation provisions for these lines (i.e., locating both valves outside containment) meet the intent of provisions defined by NuScale DSRS Section 6.2.4 Acceptance Criterion 4. In particular, given that both valves are located outside the containment, the applicant evaluated the region from the containment vessel head to the isolation valves using requirements that are consistent with the NRC staff's position for precluding a breach of piping integrity and are in conformance with SRP Section 3.6.2, including associated Branch Technical Position (BTP) 3-4 (this characterization is discussed further in Section 3.6.2 of this report). Therefore, a break between the containment vessel and the isolation valves need not be considered. Although these lines are not part of an engineered safety feature system or required for safe shutdown (i.e., considered non-essential), the staff finds that the applicant's justification is acceptable because it meets the intent of the guidance provided in NuScale DSRS Section 6.2.4, Acceptance Criterion 4.

A provision in GDC 55 also requires that isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. In DCA Part 2, Tier 2, Section 6.2.4, the applicant describes that all GDC 55-related CIVs are welded directly to the containment nozzle via a safe-end and upon receipt of a closure signal or loss of actuating power reposition closed, if open. The staff finds connecting the isolation valves to the containment nozzle safe-end locates the valve as close to containment as practical because a safe-end is a short transition piece welded to the containment vessel nozzle. The staff also finds positioning the isolation valves closed upon receipt of a closure signal or loss of actuating power provides greater safety because flow through the GDC 55-related process lines penetrating the containment is not essential to prevent or mitigate the consequences of a LOCA.

The staff believes that given the NuScale small modular design no significant enhancement to plant safety would be achieved by modification of the isolation design to fully comply with GDC 55 regarding valve location. This is due, in part, to the isolation valve design sharing a single-body (eliminates piping and welds between valves) as described in DCA Part 2, Tier 2, Section 6.2.4 and appropriate consideration of design criteria, such as quality standards (see

GDC 1 discussion above), protection against natural phenomena (see GDC 2 discussion above), and environmental and dynamic effects (see GDC 4 discussion above). In addition, all isolation valves on these lines are outside the containment because of practical limitations (e.g., space and environment) inside containment, and as such they are not exposed to the more severe environmental conditions inside containment and are accessible for maintenance, inspection, and testing without entering containment.

Pursuant to 10 CFR 52.48, applications filed under this subpart will be reviewed, in part, for compliance with the standards set out in 10 CFR Part 50 and its appendices. The requirements of GDC 55 are set forth in Appendix A to 10 CFR Part 50. As described above, the applicant seeks an exemption, in part, from the requirements of GDC 55.

Pursuant to 10 CFR 50.12, "Specific Exemption," the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR 50 when:

> (a)(1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and
>
> (a)(2) when special circumstances are present.

Special circumstances are present whenever, according to 10 CFR 50.12(a)(2)(ii), "[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

The underlying purpose of the GDC 55 requirement is to provide containment isolation capability that supports the safety function of containment to provide a barrier to the release of radioactivity associated with each line that is connected to the RCPB and penetrates the primary reactor containment. This is generally accomplished by providing redundant means of isolation (two isolation barriers in series), physically separated by the primary containment boundary.

As discussed above, the staff finds that the NuScale design accomplishes this safety function by locating two valves outside containment, providing a containment isolation capability comparable to that required by GDC 55 and therefore, the underlying purpose of the rule is met without the need for one valve inside containment. The staff concludes that special circumstances exist, in that, the regulation (i.e., having a valve inside containment) need not be applied in this particular circumstance to achieve the underlying purpose of the rule. This meets the requirements for an exemption to GDC 55, as described in 10 CFR 50.12. Therefore, the staff concludes that an exemption to the requirements of GDC 55 is justified and that the underlying purpose of the rule will be met.

With regard to 10 CFR 50.12(a)(1), because the staff finds that the underlying purpose of the rule will be met, the staff also concludes that the proposed exemption request is acceptable in terms of public health and safety. In addition, the NRC has authority under law (e.g., 10 CFR 52.7) to grant exemptions from the requirements of this regulation, therefore the staff finds the proposed exemption request is authorized by law. Furthermore, because this exemption request does not affect security related matters, the staff also concludes that the requested exemption is consistent with the common defense and security.

General Design Criterion 56

The provisions in GDC 56 require, in part, that each line that is connected directly to the containment atmosphere and penetrates the primary reactor containment shall be provided with CIVs. The combination of valves, automatic or locked closed, and the location of valves, one inside and one outside containment, are specified in GDC 56. For the isolation valve function, redundant barriers are required to account for a single active failure in the isolation provisions. This is achieved by providing two isolation valves in series.

In DCA, Part 2, Tier 2, Section 6.2.4, the applicant states that the NuScale design contains four containment piping penetrations that are subject to GDC 56. These four piping penetrations are associated with the containment evacuation system (CES), the containment flood and drain system (CFDS) and cooling lines (i.e., supply and return) for the control rod drive system (CRDS). Although the CRDS supply and return lines penetrate primary reactor containment and are not connected directly to containment atmosphere, these lines are considered subject to GDC 56 (see DCA Part 2, Tier 2, Table 6.2-4, "Containment Penetrations") because the CRDS supply and return lines inside containment are not credited as barriers (i.e., closed loop) and are conservatively treated as if the lines connect directly to the containment atmosphere.

While the applicant provides a design that complies with the requirements of GDC 56 in terms of the number of valves, there is a departure from GDC requirements with regard to valve location. The four GDC 56 piping penetration lines provide a containment isolation design consisting of two automatic isolation valves located outside containment in series rather than locating one of the CIVs inside containment as specified in GDC 56. Therefore, the applicant requested an exemption from the requirements of GDC 56 regarding valve location (refer to Part 7 of the DCA).

The staff concludes that with both automatic valves cited as CIVs, the design is adequate for assuring redundancy in achieving containment isolation. This conclusion is supported by the series arrangement of the isolation valves and the appropriate quality of the design (e.g., ASME Code, Section III, Class 1, Subsection NB and Seismic Category I criteria) as described in DCA Part 2, Tier 2, Sections 6.2.4 and 3.1.5.7, "Criterion 56-Primary Containment Isolation." In addition, in DCA Part 2, Tier 2, Section 6.2.4, the applicant describes these valves as remotely actuated by an automatic signal or operator action and fail-closed on loss of power; where each valve in a pair has a separate instrumentation and control division to provide independence and redundancy. Furthermore, in NuScale DSRS Section 6.2.4, the staff states that containment isolation provisions different from the explicit requirements of GDC 56 are acceptable if the differences are justified. The applicant's justification (see DCA Part 7) is that the differences in the isolation provisions for these lines (i.e., locating both valves outside containment) meet the intent of provisions defined by NuScale DSRS Section 6.2.4 Acceptance Criterion 4. In particular, given that both valves are located outside the containment, the applicant evaluated the region from the containment vessel head to the isolation valves using requirements that are consistent with the NRC staff's position for precluding a breach of piping integrity and are in conformance with SRP Section 3.6.2, including associated BTP 3-4 (this characterization is discussed further in Section 3.6.2 of this report). Therefore, a break between the containment vessel and the isolation valves need not be considered. Although these lines are not part of an engineered safety feature system or required for safe shutdown (i.e., considered non-essential), the staff finds the applicant's justification acceptable because it meets the intent of the guidance provided in NuScale DSRS Section 6.2.4 Acceptance Criterion 4.

A provision in GDC 56 also requires that isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety. In DCA Part 2, Tier 2, Section 6.2.4 the applicant describes that all GDC 56-related CIVs are welded directly to the containment nozzle via a safe-end and upon receipt of a closure signal or loss of actuating power reposition closed, if open. The staff finds connecting the isolation valves to the containment nozzle safe-end locates the valve as close to containment as practical because a safe-end is a short transition piece attached to the containment vessel nozzle. The staff also finds positioning the isolation valves closed upon receipt of a closure signal or loss of actuating power provides greater safety because flow through the GDC 56-related process lines penetrating the containment is not essential to prevent or mitigate the consequences of a LOCA.

The staff believes that given the NuScale small modular design no significant enhancement to plant safety would be achieved by modification of the isolation design to fully comply with GDC 56 regarding valve location. This is due, in part, to the isolation valve design sharing a single-body (eliminates piping and welds between valves) as described in DCA Part 2, Tier 2, Section 6.2.4, and appropriate consideration of other design criteria, such as quality standards (see GDC 1 discussion above), protection against natural phenomena (see GDC 2 discussion above), and environmental and dynamic effects (see GDC 4 discussion above). In addition, all isolation valves on these lines are outside the containment because of practical limitations (e.g., space and environment) inside containment, and as such they are not exposed to the more severe environmental conditions inside containment and are accessible for maintenance, inspection, and testing without entering containment.

Pursuant to 10 CFR 52.48, applications filed under this subpart will be reviewed, in part, for compliance with the standards set out in 10 CFR Part 50 and its appendices. The requirements of GDC 56 are set forth in Appendix A to 10 CFR Part 50. As described above, the applicant seeks an exemption from the valve requirements, in part, of GDC 56.

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR 50 when (a)(1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (a)(2) when special circumstances are present. Special circumstances are present whenever, according to 10 CFR 50.12(a)(2)(ii), "[a]pplication of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

The underlying purpose of the GDC 56 requirement is to provide containment isolation capability that supports the safety function of containment to provide a barrier to the release of radioactivity associated with each line that is connected directly to the containment atmosphere and penetrates the primary reactor containment. This is generally accomplished by providing redundant means of isolation (two isolation barriers in series), physically separated by the primary containment boundary.

As discussed above, the staff finds that the NuScale design accomplishes this safety function by locating two valves outside containment, providing a containment isolation capability comparable to that required by GDC 56 and therefore, the underlying purpose of the rule is met without the need for one valve inside containment. The staff concludes that special circumstances exist, in that, the regulation (i.e., having a valve inside containment) need not be applied in this particular circumstance to achieve the underlying purpose of the rule. This meets

the requirements for an exemption to GDC 56, as described in 10 CFR 50.12. Therefore, the staff concludes that an exemption to the requirements of GDC 56 is justified and that the underlying purpose of the rule will be met.

With regard to 10 CFR 50.12(a)(1), because the staff finds that the underlying purpose of the rule will be met, the staff also concludes that the proposed exemption request is acceptable in terms of public health and safety. In addition, the NRC has authority under law (e.g., 10 CFR 52.7) to grant exemptions from the requirements of this regulation, therefore the staff finds the proposed exemption request is authorized by law. Furthermore, because this exemption request does not affect security related matters, the staff also concludes that the requested exemption is consistent with the common defense and security.

General Design Criterion 57

The provisions in GDC 57 require, in part, that each line that penetrates the primary containment and is neither part of the RCPB nor connected directly to the containment atmosphere shall have at least one CIV. As specified in GDC 57, the valve shall be outside containment. The containment isolation function for GDC 57 lines is accomplished by a closed system inside containment and a valve outside containment to achieve two isolation barriers.

In DCA, Part 2, Tier 2, Section 6.2.4, the applicant states that the NuScale design contains six containment piping penetrations that are subject to GDC 57. These six piping penetrations are associated with the main steam system (two lines), the feedwater system (two lines), and the decay heat removal system (DHRS, two lines). The applicant provides a design that complies with GDC 57 isolation valve requirements for feed and steam lines by providing an automatic isolation valve outside containment. However, the NuScale design departs from the requirement to provide a CIV outside containment for the DHRS piping lines.

There are two independent DHRS trains, each with a DHRS steam supply line (connects with main steam piping) and a DHRS condensate return line (connects with feedwater piping). The applicant proposes to use the closed-loop DHRS outside containment, which does not include CIVs, as an alternative to the isolation valve requirement. Therefore, the applicant requested an exemption from the requirements of GDC 57, that is use of an isolation valve outside containment as applied to the DHRS piping penetrations (refer to Part 7 of the DCA).

As described in the DCA, the applicant's GDC 57-related lines provide two containment isolation barriers:

> 1) the piping inside containment, including main steam, feedwater and DHRS, functions as the first isolation barrier, and
>
> 2) the main steam and main feed valves and DHRS lines outside containment, function as the second isolation barrier.

With regard to the adequacy of the first isolation barrier, in DCA Part 7, the applicant states the lines inside containment (i.e., main steam system, feedwater system, and DHRS) meet the requirements for a closed system inside containment by conforming to the provisions in NuScale DSRS Section 6.2.4, Acceptance Criterion 15 (e.g., protection from missiles and pipe whip, designed to Seismic Category I, classified as Quality Group B, and designed to withstand the environmental effects from a LOCA). The staff finds the applicant's justification regarding the first isolation barrier acceptable because the barrier (closed system inside containment) meets the provisions defined in NuScale DSRS Section 6.2.4 Acceptance Criterion 15 (see discussion

provided in the NuScale DCA Part 2, Tier 2, Section 3 related to classification and protection from missiles and pipe whip).

With regard to the adequacy of the second isolation barrier, achieved by the main steam and main feedwater isolation valves, in DCA Part 2, Tier 2, Section 6.2.4, the applicant states that these automatic isolation valves meet GDC 57 valve requirements for location and automatic isolation. The staff concludes that with these automatic valves cited as CIVs, the design is adequate for achieving containment isolation. This conclusion is supported by the appropriate consideration of other design criteria, such as quality standards (see GDC 1 discussion above), protection against natural phenomena (see GDC 2 discussion above), and environmental and dynamic effects (see GDC 4 discussion above). In addition, in DCA Part 2 Tier 2, Section 6.2.4, the applicant describes these valves as remotely actuated by an automatic signal or operator action and fail-closed upon receipt of a closure signal or on loss of power.

With regard to the adequacy of the second isolation barrier, achieved by the DHRS, in DCA Part 7, the applicant states that the design of the DHRS outside containment allows it to function as a suitable containment isolation barrier. Although use of closed systems outside containment as an alternative isolation provision is not addressed by GDC 57 or guidance, the applicant's basis is that the isolation provisions for the DHRS lines (i.e., closed system outside containment) otherwise meet the intent of the provisions described in NuScale DSRS 6.2.4 Acceptance Criterion 5. For example, the DHRS closed loop outside containment is missile protected, designed to Seismic Category I and Quality Group B standards, and has a design temperature and pressure rating at least equal to that of containment. As described in DCA Part 2, Tier 2, Section 5.4.3, "Decay Heat Removal System," the DHRS is a welded design with a design pressure equal to that of the reactor pressure vessel, which greatly exceeds the design pressure for containment. Specifically, the DHRS design pressure is 2100 pounds per square inch absolute (psia), whereas the containment design pressure is roughly half of DHRS design pressure. In addition, the applicant evaluated the DHRS outside containment using requirements that are consistent with the NRC staff's position for precluding a breach of piping integrity in conformance with SRP Section 3.6.2, including associated BTP 3-4 (this characterization is discussed further in Section 3.6.2 of this report). Therefore, the applicant proposes that a break in this line outside containment need not be considered. The staff finds the applicant's alternate approach to use a closed system outside containment in place of an isolation valve outside containment as required by GDC 57 is acceptable because the applicant provided a suitable basis (e.g., design provisions) to justify that the DHRS barrier outside containment will retain its integrity and therefore, provide containment isolation capability that supports the safety function of containment to provide a barrier to the release of radioactivity.

A provision in GDC 57 also requires that CIVs shall be located as close to containment as practical. In DCA Part 2, Tier 2, Section 6.2.4, the applicant states that all GDC 57-related CIVs are welded directly to the containment nozzle safe-end, except the main steam isolation valves. The main steam isolation valves, as described in DCA Part 2, Tier 1, Table 2.1-1 and DCA Part 2, Tier 2, Section 6.2.4, are four feet from the containment vessel. A distance of four feet accommodates the installation of two branch line connections (e.g., piping tees) for the decay heat removal system. Based on the discussion above, the staff finds the isolation valve location information provided in the DCA satisfies GDC 57 requirements for locating isolation valves outside containment as close to containment as practical.

In summary, the staff finds that the applicant's GDC 57-related lines described above are sufficient barriers to the release of radioactivity because of the existence of two redundant physical barriers: closed system inside containment (i.e., closed-loop steam generator system

and connecting piping) and automatic isolation valves (main steam and main feed) in conjunction with a closed system outside containment (DHRS).

Pursuant to 10 CFR 52.48, applications filed under this subpart will be reviewed, in part, for compliance with the standards set out in Part 50 and its appendices. The requirements of GDC 57 are set forth in Appendix A to Part 50. As described above, the applicant seeks an exemption, in part, from the requirements of GDC 57.

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR 50 when (a)(1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and area consistent with the common defense and security; and (a)(2) when special circumstances are present. Special circumstances are present whenever, according to 10 CFR 50.12(a)(2)(ii), "Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule." In addition, special circumstances are also present whenever, according to 10 CFR 50.12(a)(2)(vi), "There is present any other material circumstance not considered when the regulation was adopted or which it would be in the public interest to grant an exemption."

The underlying purpose of the GDC 57 requirement is to provide containment isolation capability that supports the safety function of containment to provide a barrier to the release of radioactivity associated with each line that is neither part of the RCPB nor connected directly to the containment atmosphere and penetrates primary reactor containment. This is generally accomplished by providing redundant means of isolation (two isolation barriers in series), physically separated by the primary containment boundary.

As discussed above, the staff finds that the NuScale design accomplishes this safety function for the DHRS lines outside containment by using a closed system that is designed to preclude a breach of integrity (e.g., Seismic Category I, ASME Section III, Class 2 with a design temperature and pressure equal to that of the reactor pressure vessel). Therefore, containment isolation is achieved without the need for DHRS CIVs outside containment. The staff concludes that special circumstances exist, in that, the regulation (i.e., having valves outside containment) need not be applied in this particular circumstance to achieve the underlying purpose of the rule. This meets the requirements for an exemption to GDC 57, as described in 10 CFR 50.12(a)(2). Therefore, the staff concludes that an exemption to the requirements of GDC 57 is justified and that the underlying purpose of the rule will be met.

With regard to 10 CFR 50.12(a)(1), because the staff finds that the underlying purpose of the rule will be met, the staff also concludes that the proposed exemption request is acceptable in terms of public health and safety. In addition, the NRC has authority under law (e.g., 10 CFR 52.7) to grant exemptions from the requirements of this regulation, therefore the staff finds the proposed exemption request is authorized by law. Furthermore, because this exemption request does not affect security related matters, the staff also concludes that the requested exemption is consistent with the common defense and security.

10 CFR 50.34(f)(2)(xiv)

The *Code of Federal Regulations* in 10 CFR 52.47(a) states in part, the [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:

...

(8) 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)...

10 CFR 50.34(f), “Additional TMI-related requirements,” states, in part:

In addition, each applicant for a design certification, design approval, combined license, or manufacturing license under Part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section…

10 CFR 50.34(f)(2)(xiv) requires containment isolation systems that: (II.E.4.2)

(A) Ensure all non-essential systems are isolated automatically by the containment isolation system,

(B) For each non-essential penetration (except instrument lines) have two isolation barriers in series,

(C) Do not result in reopening of the CIVs on resetting of the isolation signal,

(D) Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation, and

(E) Include automatic closing on a high radiation signal for all systems that provide a path to the environs.

In DCA Part 2, Tier 2, Table 1.9-5, the applicant indicates that 10 CFR 50.34(f)(2)(xiv), items (A) through (D) are technically relevant requirements for the NuScale design and that an exemption is sought for item (E).

The staff reviewed the applicant’s classifications for systems containing CIVs as essential or non-essential. Specifically, the staff reviewed DCA Part 2, Tier 2, Table 6.2-4, and finds that it provides the proper classifications consistent with 10 CFR 50.34(f)(2)(xiv) and RG 1.141. The staff also reviewed the normal, shutdown, and post-accident CIV positions shown in DCA Part 2, Tier 2, Table 6.2-5, and concludes that they are consistent with their classifications.

In DCA Part 2, Tier 2, Section 6.2.4, the applicant describes that CIVs serving in non-essential systems (i.e., not required to prevent, arrest or mitigate the consequences of an accident) are designed to close automatically on a containment isolation signal. Therefore, the 10 CFR 50.34(f)(2)(xiv)(A) requirement to ensure all non-essential systems are isolated automatically by the containment isolation system is satisfied. Additionally, in DCA Part 2, Tier 2, Section 6.2.4 the applicant provides information to conclude that each non-essential penetration has two isolation barriers in series, thereby meeting the requirements of 10 CFR 50.34(f)(2)(xiv)(B). Furthermore, in DCA Part 2, Tier 2, Sections 6.2.4 and 7, “Instrumentation and Controls,” the applicant states that resetting an isolation signal will not automatically reopen isolation valves, demonstrating that the 10 CFR 50.34(f)(2)(xiv)(C) requirement is satisfied.

Provisions in 10 CFR 50.34(f)(2)(xiv) item (D) require that an applicant provide containment isolation systems that utilize a containment set point pressure for initiating containment isolation

as low as is compatible with normal operation (additional background information can be found in NuScale DSRS Section 6.2.4 and NUREG-0737, "Clarification of TMI Action Plan Requirements"). As shown in DCA Part 2, Tier 2, Table 7.1-4, "Engineered Safety Feature Actuation System Functions," a containment system isolation actuation signal is initiated on containment pressure. In DCA Part 2, Tier 2, Table 1.9-5, the applicant states that the pressure setpoint is compatible with normal operating pressure (e.g., above the highest allowable pressure for leak detection acceptability given by DCA Part 2, Tier 2, Figure 5.2-3) and the containment isolation pressure signal is initiated while the containment pressure is sub-atmospheric, therefore isolation of containment occurs under partial vacuum conditions. Initiating containment isolation while containment is at a partial vacuum ensures non-essential piping penetrations are isolated before significant containment pressurization occurs or significant release of radionuclides into containment (see additional discussion related to 10 CFR 50.34(f)(2)(xiv)(E) in this report) thereby enhancing containment dependability. The staff finds that initiating containment isolation at a pressure that is compatible with normal operation (far enough away from expected pressure in containment so that inadvertent containment isolation does not occur during normal operation), while containment is at a partial vacuum and before significant release of radionuclides, meets the intent of this technically relevant requirement and is acceptable.

Therefore, based on the discussion above, the staff finds that the containment isolation system meets the requirements of 10 CFR 50.34(f)(2)(xiv) items (A) through (D).

The remaining discussion in this subsection of this report focuses on addressing the applicant's request for an exemption from the requirements provided in 10 CFR 50.34(f)(2)(xiv) item (E).

In response to the lessons learned from the accident at Three Mile Island (TMI), NRC added requirements to its power reactor safety regulations. In 10 CFR 50.34 and the *Federal Register* associated with the Final Rule for these additional TMI-related requirements the NRC refers to NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and NUREG-0737.

As described in NUREG-0660 and NUREG-0737 the purpose for the requirements within 10 CFR 50.34(f)(2)(xiv) is to improve containment isolation dependability and to improve the reliability and capability of nuclear power plant containment structures to reduce the radiological consequences and risks to the public from design basis events and degraded-core and core-melt accidents. In addition, as described in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," the containment is to provide a final barrier to the release of radioactivity in the event of an accident and isolation of non-essential systems penetrating the containment boundary before significant releases from the building is imperative.

In DCA Part 2, Tier 2, Section 9.3.6, the applicant indicates that the containment evacuation system (CES) connects the containment atmosphere to the environs and states that the NuScale design supports an exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the CES. In addition, DCA Part 2, Tier 2, Section 9.3.6 states:

> *[d]esign basis events meet their thermal and hydraulic acceptance criteria without reliance on isolating the CES on a high radiation signal. No design basis event results in degraded or damaged core conditions. Section 19.2 analyses demonstrate severe accident conditions, with resultant core damage, also result in generation of reliable containment isolation signals, without reliance on*

isolation on high containment radiation. An in-containment event resulting in core damage or degradation also results in containment isolation on low low pressurizer level and high containment pressure…These features provide a reliable alternative means to prevent radiological release from the CES to the environs."

In DCA Part 7, "Exemptions," Chapter 13, "10 CFR 50.34(f)(2)(xiv)(E) Containment Evacuation System Isolation," the applicant requests an exemption from 10 CFR 50.34(f)(2)(xiv) item (E) as applied to the CES. The applicant describes the purpose of the rule is to limit radiological releases by ensuring containment isolation for systems that provide a path to the environs during events where reliance on a high containment pressure isolation signal may not be sufficient (see NUREG-0578, NUREG-0660, and NUREG-0737). The applicant states the design meets the purpose of the rule by ensuring reliable and dependable isolation of the CES system (a non-essential system) upon any event involving radiological consequences inside the containment vessel. Therefore, the applicant concludes that alternate means are provided to prevent radiological release to the environment such that automatic isolation on a high radiation signal is not required to meet the underlying purpose of the rule.

In DCA Part 7, Chapter 13, the applicant describes that the NuScale design meets the underlying purpose of the rule by isolating CES using two automatic containment isolation signals: 1) high containment vessel (CNV) pressure signal and 2) low low pressurizer level. The applicant explains that the NuScale design differs from the traditional large light water reactor designs "…because reactor core uncovery, and resulting core damage, cannot occur without reaching a low low pressurizer [level] containment isolation setpoint." Therefore, "[a]n event similar to the TMI, Unit 2, accident is precluded by the NuScale plant design." The applicant also states that in the NuScale plant design "[t]he pressurizer is located well above the level of the reactor core and not connected to the reactor vessel by piping. Any decrease in reactor vessel inventory to the level of the core would result in complete emptying of the pressurizer and operation of the pressurizer level containment isolation signal." As such, the applicant describes that automatic isolation of the CES on a high radiation signal is not required to meet the underlying purpose of 10 CFR 50.34(f)(2)(xiv) item (E) because alternate means to preclude a path to the environs are provided in the NuScale plant design before core damage or degradation occurs.

The staff reviewed the information provided in the DCA against the requirement (item (E)) and TMI-related NUREGs. Based on the review, the staff concludes that non-essential systems (to include CES) penetrating the containment boundary would receive an isolation signal because of an in-containment event before any core damage or degradation occurring and therefore, before significant releases into the containment. Accordingly, containment isolation for systems that provide paths to the environs (i.e., CES) in order to limit radiological releases is accomplished without reliance on the features required by the rule.

Pursuant to 10 CFR 52.48, applications filed under this subpart will be reviewed, in part, for compliance with the standards set out in Part 50 and its appendices. As described above, the applicant seeks an exemption from the requirements of 10 CFR 50.34(f)(2)(xiv)(E).

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirements of 10 CFR 50 when (a)(1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security; and (a)(2) when special circumstances are present. Special circumstances are present whenever, according to 10 CFR 50.12(a)(2)(ii),

“Application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.”

The underlying purpose of the 10 CFR 50.34(f)(2)(xiv)(E) requirement is to limit radiological releases by ensuring containment isolation for systems that provide paths to the environs where reliance on a high containment pressure isolation signal may not be sufficient. As discussed above, the staff finds that the NuScale design accomplishes this safety function by having two containment isolation signals (containment pressure and pressurizer level) before any core damage or degradation occurring, preventing significant releases from the containment, and therefore, the underlying purpose of the rule is met without the need for isolation on high radiation. The staff concludes that special circumstances exist, in that, the regulation (i.e., having isolation on high radiation) need not be applied in this particular circumstance to achieve the underlying purpose of the rule. This meets the requirements for an exemption to 10 CFR 50.34(f)(2)(xiv)(E) as described in 10 CFR 50.12. Therefore, the staff concludes that an exemption to the requirements of 10 CFR 50.34(f)(2)(xiv)(E) is justified and that the underlying purpose of the rule will be met.

With regard to 10 CFR 50.12(a)(1), because the staff finds that the underlying purpose of the rule will be met, the staff also concludes that the proposed exemption request is acceptable in terms of public health and safety. In addition, the NRC has authority under law (e.g., 10 CFR 52.7) to grant exemptions from the requirements of this regulation, therefore the staff finds the proposed exemption request is authorized by law. Furthermore, because this exemption request does not affect security related matters, the staff also concludes that the requested exemption is consistent with the common defense and security.

10 CFR 50.34(f)(2)(xv)

The Code of Federal Regulations in 10 CFR 52.47(a) states in part, the [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:

> ...
>
> (8) 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)...

10 CFR 50.34(f), “Additional TMI-related requirements,” states, in part:

> *In addition, each applicant for a design certification, design approval, combined license, or manufacturing license under Part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section…*

10 CFR 50.34(f)(2)(xv) states:

> *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)*

Technical specifications for several operating plants, define purge or purging as the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration, or other operating condition, in such a manner that replacement air or gas is required to purify the confinement. In addition, staff guidance (Branch Technical Position 6-4, “Containment Purging during Normal Operations”), states that containment purge and vent systems provide plant operational flexibility during normal operations (e.g., facilitate personnel access into containment during reactor power operation). Additionally, whenever containment integrity is required, NUREG-0737 discusses limiting containment purge and venting operation to when there is an established need to improve working conditions to perform a safety-related surveillance or safety-related maintenance procedure.

In DCA Part 2, Tier 2, Table 1.9-5, “Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933),” the applicant provides a conformance status associated with TMI requirements to include 10 CFR 50.34(f)(2)(xv). The applicant states that their design does not require or incorporate a purge or venting system function as contemplated by this requirement and therefore, it is not technically relevant to the NuScale design. For example, the NuScale containment design is significantly smaller than a typical containment building and does not require personnel access during reactor operation (e.g., when containment is required to be operable – Modes 1, 2, and 3 with the reactor coolant temperature hot $\geq$ 200 ° Fahrenheit).

Based on technical specifications definition of purging, staff guidance contained in BTP 6-4 and the applicant’s information presented above (i.e. that the NuScale design does not require or incorporate the capability for containment purging during reactor operation) the NRC staff finds that the 10 CFR 50.34(f)(2)(xv) regulation is not technically relevant to the NuScale design. An exemption is not needed for this requirement because the applicant demonstrated that the requirement is not technically relevant and therefore, the NuScale design complies with 10 CFR 52.47(a)(8).

10 CFR 50.34(f)(3)(iv)

The Code of Federal Regulations in 10 CFR 52.47(a) states in part, the [design certification] application must contain a final safety analysis report (FSAR) that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:

> ...
>
> (8) 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)...

10 CFR 50.34(f), “Additional TMI-related requirements,” states, in part:

> *In addition, each applicant for a design certification, design approval, combined license, or manufacturing license under Part 52 of this chapter shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (3) of this section…*

10 CFR 50.34(f)(3)(iv) states,

Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. (II.B.8)

In DCA Part 2, Tier 2, Table 1.9-5, the applicant describes conformance with TMI requirement 50.34(f)(3)(iv) as not technically relevant. In addition, in Table 1.9-5 the applicant states that a 3-foot containment opening is not necessary since the NuScale design already addresses severe accident scenarios that could lead to containment failure. In DCA Part 2, Tier 2, Section 6.2, the applicant discusses 10 CFR 50.34(f)(3)(iv) as follows:

The NuScale CNV [containment vessel] does not include one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, to accommodate future installation of systems to prevent containment failure. As discussed in this section [Section 6.2], the calculated peak containment pressures for design basis events remain less than the CNV internal design pressure. As discussed in Section 19.2.3, peak containment pressures do not challenge vessel integrity for any analyzed severe accident progression. Therefore, 10 CFR 50.34(f)(3)(iv) is not technically relevant to the NuScale design.

Prior to submission of the NuScale application, in a March 24, 2016, letter (ADAMS Accession No. ML15266A264), the NRC staff responded to a report from NuScale Power entitled "Gap Analysis Summary Report," related to containment regulatory issues, to include the provision for a dedicated penetration. In Enclosure 2 to the March 24, 2016 letter, the staff responds to Report Table 3-1 Gap 5, "Spare Containment Penetration," and states the following, in part, in response to NuScale Powers position that 10 CFR 50.34(f)(3)(iv) is not technically relevant to the NuScale design:

The NRC staff will remain receptive to additional information from NuScale regarding NuScale's position that the regulation is not technically relevant if NuScale can provide technical justification in Tier 2 of the design certification (DC) application that the pressure in containment during beyond design basis accidents cannot exceed the allowable pressure of the containment structure.

The *Federal Register*, Volume 47, No. 10, dated Friday, January 15, 1982, describes a final rule that contained the provision for a dedicated penetration (10 CFR 50.34(f)(3)(iv)). The Commission's consideration of a comment related to this requirement is as follows:

(3)(iv)-Containment Penetration

Several commentors (OPS, Gilbert, W, CEC, TVA) centered on the asserted arbitrariness of the requirement for a 3-foot diameter penetration, the lack of technical justification, and the possibility that containment venting provisions may not provide a significant contribution to safety.

Discussion

The containment penetration size was selected so that it would be consistent with mitigation features designed to accommodate medium- and slow-rate pressure rises in

containments that would otherwise have failed. Among the features considered were filtered vented containment systems and passive containment cooling systems. Rapid-rate pressure rises from hydrogen burns, for example, were excluded from consideration. The 3-foot penetration was determined to be a conservative penetration size that would not preclude the eventual installation of one of the aforementioned features. Of course, there is the possibility that such penetrations will not be needed, but that will be known only after the completion of the degraded core rulemaking. Therefore, the Commission has retained this requirement so as not to preclude later installation of containment venting systems, if required.

As discussed above, the Commission acknowledges the possibility that dedicated penetrations may not be necessary. The applicant's position is that the requirement for a dedicated containment penetration is not technically relevant because the pressure in containment during beyond design basis accidents (DBAs) cannot exceed the allowable pressure of the containment structure (containment pressure control is accomplished through passive containment cooling systems - partial immersion of the containment vessel in a pool of water). By providing this technical justification, the applicant was responsive to the staff position communicated in Enclosure 2 to the March 24, 2016 letter.

As demonstrated in the State-of-the-Art Reactor Consequence Analyses (SOARCA) study (NUREG 1935, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," issued November 2012), large LWRs have the potential for long-term containment overpressure failure because of the following:

1) loss of containment heat removal; and

2) generation of noncondensable gases because of in-vessel cladding oxidation and ex-vessel corium-concrete interaction.

Containment venting through an external filter or through a BWR suppression pool is a severe accident mitigation feature intended to prevent catastrophic containment failure while limiting the release of radionuclides to the environment for long-term overpressure scenarios.

In comparison to item (1) above, the NuScale design is not vulnerable to loss of containment heat removal because the containment is submerged in the reactor pool. In comparison to item (2) above, NuScale's generation of noncondensable gases is limited to hydrogen generation from in-vessel cladding oxidation because there is no concrete in the NuScale containment. However, most importantly, the applicant performed simulations for a range of severe accident scenarios using the MELCOR code. The simulations, which are documented in DCA Part 2, Tier 2, Chapter 19.2, "Severe Accident Evaluation," show that long-term containment pressures in a severe accident stay below those that could fail the containment. In accordance with the SRP 19.0, "Severe Accidents," the staff performed independent analysis and confirmed the applicant's MELCOR simulations. Section 19.2 of this report documents the staff's review.

In summary, as discussed above, the staff reviewed the applicant's containment analysis results and finds that pressure in containment did not exceed the allowable pressure of the containment structure. Therefore, the staff finds that the 10 CFR 50.34(f)(3)(iv) requirement is not technically relevant to the NuScale design. An exemption is not needed because the applicant demonstrated that the requirement was not technically relevant and therefore, the NuScale design complies with 10 CFR 52.47(a)(8).

10 CFR 50.34(f)(2)(xix)

The provisions in 10 CFR 50.34(f)(2)(xix) require the applicant to provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

As described in DCA Part 2, Tier 2, Section 6.2.4, position indication for CIVs is provided in the main control room. In addition, as described in DCA Part 2, Tier 2, Section 7, “Instrumentation and Controls,” position indication for containment isolation is a post-accident monitoring variable (see also, DCA Part 2, Tier 2 Table 7.1-7).

The staff finds that providing position indication for CIVs is consistent with guidance contained in NuScale DSRS Section 6.2.4 and satisfies, in part, the regulations in 10 CFR 50.34(f)(2)(xix) requiring instrumentation to monitor plant conditions following an accident. See section 7.2.13.4.6, “Three Mile Island Action Items,” of this report for additional staff review of 10 CFR 50.34(f)(2)(xix).

General Design Criterion 54

The provisions in GDC 54 require piping systems that penetrate the primary reactor containment have leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect the safety importance of isolating these piping systems.

To address GDC 54, NuScale DSRS Section 6.2.4 directs the reviewer to evaluate the containment isolation system for whether valves in piping systems that penetrate the containment are designed to close reliably under accident conditions and prevent the uncontrolled release of radioactive materials. In addition, NuScale DSRS 6.2.4 identifies that there should be diversity in the parameters sensed for the initiation of containment isolation to satisfy the GDC 54 requirement for reliable isolation capability.

As discussed above (i.e., GDC discussions in Section 6.2.4.4 of this report), piping systems that penetrate containment have appropriate isolation and containment capabilities and meet redundancy requirements. In DCA Part 2, Tier 2, Section 6.2.4, the applicant describes the parameters sensed for the initiation of a containment isolation signal from the engineered safety feature actuation system (ESFAS, see DCA Part 2, Tier 2, Table 7.1-4). The staff finds that the diversity of parameters that are sensed for isolating containment (e.g., containment pressure, pressurizer level, high under-the-bioshield temperature, and low ac voltage to battery chargers) is sufficient to ensure timely isolation of all non-essential piping penetrations, in conformance with NuScale DSRS Section 6.2.4, Acceptance Criterion 12 and for meeting the GDC 54 requirement for reliable isolation capability.

The provisions in GDC 54 also require these piping systems to be designed with the capability to test periodically the operability of the isolation valves and to determine if valve leakage is within acceptable limits. As described in DCA Part 2, Tier 2, Section 6.2.4, the piping system design has the capability to test the operability of the isolation valve and determine if valve leakage is within acceptable limits (additional discussion on valve leakage is provided in Section 6.2.6 of this report). Because the design does have the capability to test periodically the operability of the isolations valves and determine if valve leakage is acceptable, the staff finds the design meets these GDC 54 requirements.

In DCA Part 2, Tier 2, Section 5.2.5.1, “Leakage Detection and Monitoring,” the applicant describes how leakage is monitored using pressure, temperature, level, and radioactivity

instrumentation. This, coupled with the methods available to detect intersystem leakage described in DCA Part 2, Tier 2, Section 5.2.5.5, “Chemical and Volume Control System Intersystem Leakage Monitoring,” supports the conclusion that there are adequate leakage detection provisions to enable the operators to detect leakage and identify lines that should be isolated. The staff finds the provisions for detecting leakage from the lines outside containment conforms to NuScale DSRS Section 6.2.4 Acceptance Criterion 8 and supports meeting the GDC 54 requirement for leak detection.

The staff reviewed the closure times for the CIVs provided in DCA Part 2, Tier 2 Table 6.2-5. As identified in the Table, the normally open CIVs close in less than or equal to seven seconds, assuming signal delay and valve stroke times. This staff finds this closure time is consistent with the guidance contained in NuScale DSRS 6.2.4, Acceptance Criterion 14 (i.e., rapid isolation of containment) for meeting GDC 54 requirements and is therefore acceptable. The staff also finds that the closure times associated with CIVs used in the containment evacuation system (i.e., path to the environs) are assessed in the radiological dose analyses. The radiological assessment for closing lines that provide a path to the environment (i.e., containment evacuation system isolation valves) is evaluated in Section 15.0.3, “Radiological Consequences of Design Basis Accidents,” of this report. In addition, the applicant’s evaluation model for the emergency core cooling system (ECCS) performance accounts for containment isolation closure times, consistent with the isolation times presented in the DCA Part 2, Tier 2, Table 6.2-5. ECCS performance is evaluated in Sections 6.3 and 15 of this report. Overall, the staff finds the valve performance capabilities reflect the safety importance of isolating these piping systems.

Sections 6.2.4.4.8 (TMI requirements that address reliable isolation capability) and 6.2.4.4.18 (Technical Specifications) of this report also address aspects that support meeting GDC 54 reliable isolation capability requirements that reflect the safety importance of isolating these piping systems.

Based on the discussion above, the staff finds that the containment isolation system meets the requirements of GDC 54.

Station Blackout (10 CFR 50.63) and Appropriate Containment Integrity

A station blackout (SBO) means the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). The provisions in 10 CFR 50.63, “Loss of all alternating current power,” require, in part, that each plant demonstrate sufficient capacity and capability to ensure that appropriate containment integrity is maintained in the event of an SBO for the specified duration. In accordance with RG 1.155, “Station Blackout,” appropriate containment integrity is ensured by providing the capability, independent of the preferred and blacked out unit's onsite emergency ac power supplies, for valve position indication and closure for CIVs that may be in the open position at the onset of a SBO. As described in DCA Part 2, Tier 2, Section 8.4, “Station Blackout,” CIVs automatically close following receipt of a module protection system actuation on a low ac voltage to battery charger signal. In addition, valve position indication (powered by highly reliable dc power system (EDSS)) is available in the control room for operators to verify valve closure. This arrangement satisfies the guidance provided in RG 1.155, position C.3.2.7 for appropriate containment integrity. The staff finds that the NuScale design meets 10 CFR 50.63(a)(2) with respect to the ability to maintain appropriate containment integrity because the NuScale design conforms to staff guidance.

Reliance on Electrical Power

As described in DCA Part 2, Tier 2, Section 6.2.4, the NuScale CIVs are designed to fail close upon a loss of power to the actuator. The staff finds the closed position for the NuScale CIVs provides greater safety because, for example, the flow through the lines is not relied upon to mitigate the consequences of a loss-of-coolant accident. Because the containment isolation safety function does not rely upon electrical power, this supports findings made in Section 8, "Electric Power," of this report regarding overall reliance on electrical power for the NuScale plant.

Overpressure Protection

In DCA Part 2, Tier 2, Section 6.2.4, the applicant addresses overpressure protection in the region between two closed primary CIVs because of heatup of fluid between the barriers. In particular, in response to increasing pressure between the two valves, a thermal relief device integral to the inboard valve relieves fluid pressure back to the containment vessel. The staff finds this approach acceptable because it is consistent with the thermally induced overpressure protection guidance for liquid-filled piping between containment isolation barriers discussed in RG 1.141.

Inspections, Tests, Analysis, and Acceptance Criteria

The staff evaluation of the ITAAC, summarized in Section 6.2.4.2 of this report, is located in Section 14.3 of this report.

Initial Test Program

The staff evaluation of the Initial Test Program summarized in Section 6.2.4.2 of this report is located in Section 14.2 of this report.

Technical Specifications

In DCA, Part 4, "Generic Technical Specifications – NuScale Nuclear Power Plants," Section 3.6.2, "Containment Isolation Valves," the applicant states the technical specifications requirements for the CIVs. These technical specifications provide limiting conditions for operation and surveillance requirements for the CIVs. In particular, the surveillances require periodic operability testing of the CIVs to include verification of isolation time and valve leakage. The staff evaluation of technical specifications and associated bases are located in Section 16 of this report.

6.2.4.5 Combined License Information Items

There are no COL items specified by the applicant for this area of review.

6.2.4.6 Conclusion

On the basis of its review, the staff concludes that the proposed NuScale containment isolation system, described in the DCA, conforms to the acceptance criteria of Section 6.2.4 of the NuScale DSRS, except where the applicant requests an exemption. Conformance with the criteria in Section 6.2.4 of the NuScale DSRS and justification for exemption requests, as described in this section, constitutes an acceptable basis for satisfying the containment isolation requirements of GDC 1, 2, 4, 5, 16, 54, 55, 56, and 57; 10 CFR 52.47(a)(8) and technically

relevant TMI-related requirements of 10 CFR 50.34(f)(2)(xiv) and 10 CFR 50.34(f)(2)(xix); and 10 CFR 50.63.

6.2.5 Combustible Gas Control in Containment

6.2.5.1 Introduction

Control of combustible gases in containment is described in DCA Part 2, Tier 2, Section 6.2.5. Following a postulated accident, hydrogen and oxygen may accumulate inside the containment. Combustible gases are predominantly generated within the containment as a result of reactions between the fuel-clad and reactor coolant, although some gases are generated by radiolytic decomposition during the course of the accident. In some accident scenarios, significant amounts of combustible gas can be generated; the NuScale plant is designed to limit scenarios that could produce a hydrogen combustion event and withstand deflagrations, detonations, and deflagration-to-detonation transitions such that the containment functions are not challenged.

6.2.5.2 Summary of Application

DCA Part 2, Tier 1: There are no items related to this section in DCA Part 2, Tier 1.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 6.2.5, details how the applicant addresses the potential buildup of combustible gases in the containment resulting from up to a 100-percent fuel clad-coolant interaction. The applicant states that by design, no provisions are required to prevent a hydrogen combustion event that could result in a loss of containment structural integrity. The design-basis conditions for the containment are to maintain this condition for at least 72 hours, with severe accident management guidelines (to be implemented by future licensees) acting to manage combustible gas concentrations after that period of time. Because of this, no combustible gas control system exists in the traditional sense.

The applicant’s determination that the design adequately controls combustible gas concentrations is based on an analysis of a spectrum of severe accidents assuming a previously intact containment boundary. In DCA Part 2, Tier 2, Section 6.2.5.1, the applicant states that the containment is adequately mixed for these events and that, in the event of combustion, a structural evaluation demonstrates the ability of the CNV to withstand the limiting loads with no impact on structural integrity.

The CNV itself is an ASME Code Class 1 pressure vessel, designed to control fission product releases from the RCPB, contain the inventory released from a LOCA, and support ECCS operation by acting as a heat transfer conduit to the UHS. As described in DCA Part 2, Tier 2, Section 6.2.5.2, these design functions include accommodation of the hydrogen generated by a 100-percent fuel-clad metal water reaction.

Regulations in 10 CFR 50.44(c)(2) require all containments to have an inerted atmosphere, or to limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen to that 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. In Exemption Request Number 2, NuScale requested an exemption from the requirements of 10 CFR 50.44(c)(2).

ITAAC: There are no ITAAC associated with combustible gas control in containment because no systems exist in the traditional sense to reduce combustible gas concentrations. ITAAC

associated with the containment, which is designed to withstand a postulated combustion event, are located in DCA Part 2, Tier 1, Section 2.1 and Section 2.8.

Technical Specifications: There are no TS associated with combustible gas control in containment.

Technical Reports: TR-0716-50424-P is referenced in DCA Part 2, Tier 2, Section 6.2.5, and is incorporated by reference in Tier 2, Table 1.6-2.

6.2.5.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in DSRS Section 6.2.5 and are summarized below:

- GDC 5, as it relates to providing assurance that the sharing of SSCs important to safety among nuclear power units will not significantly impair their ability to perform their safety functions

- GDC 41, "Containment Atmosphere Cleanup," as it relates to systems being provided to control the concentration of hydrogen or oxygen that may be released into the reactor containment following postulated accidents to ensure that containment integrity is maintained; that suitable redundancy in components and features exists such that the system safety function can be accomplished, assuming a single failure; and that systems are provided with suitable leak detection, isolation, and containment capability to ensure that system safety function can be accomplished

- GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," as it relates to the design of the systems to permit appropriate periodic inspection of components to ensure the integrity and capability of the system

- GDC 43, "Testing of Containment Atmosphere Cleanup Systems," as it relates to the systems being designed to permit periodic testing to ensure system integrity, and the operability of the systems and active components

- 10 CFR 50.44(c), as it relates to PWR plants being designed to accommodate hydrogen generation equivalent to 100-percent fuel clad-coolant reaction while limiting containment hydrogen to less than 10 percent and maintain containment structural integrity and appropriate accident mitigating features; and the capability to ensure a mixed atmosphere during DBAs and significant beyond-DBAs

- 10 CFR 50.12, "Specific Exemptions," states that the Commission may grant exemptions from the requirements of the regulations of this part, which are:

    (a)(1) consistent with the Atomic Energy Act of 1954, as amended, would not present an undue risk to the public health and safety, and would be consistent with the common defense and security

    (a)(2) The Commission will not consider granting an exemption unless special circumstances are present. Special circumstances are present whenever

(a)(2)(ii) denying the exemption would not serve the underlying purpose of the rule, or

(a)(2)(iii) exemption would result in undue hardship

The guidance in DSRS Section 6.2.5 lists the acceptance criteria adequate to meet the above requirements. Review interfaces with other SRP sections can also be found in DSRS Section 6.2.5. The staff notes NuScale proposed PDC 41 to satisfy the requirements associated with GDC 41, discussed further below.

6.2.5.4 Technical Evaluation

The main regulatory requirements for combustible gas control of nuclear power plants are specified in 10 CFR 50.44(c), namely:

- All designs shall ensure a mixed containment atmosphere; this applies to both DBAs and significant beyond-DBAs. A mixed atmosphere means that the concentration of combustible gases in any part of the containment is below a level that supports combustion or detonation that could cause loss of containment integrity.

- The concentration of hydrogen shall be limited, both globally and locally, to less than 10 percent, following an accident that releases an equivalent amount of hydrogen as would be generated from a 100-percent fuel clad-coolant reaction, and maintain containment structural integrity and appropriate accident mitigating features.

- Equipment and systems needed to maintain containment integrity shall be able to perform their functions during and after a hydrogen burn; detonations of hydrogen shall also be included unless it can be shown that such detonations are unlikely to occur.

- Equipment shall be provided for continuously measuring hydrogen concentration inside containment following a significant beyond-DBA.

- A structural analysis shall be completed that demonstrates containment integrity will be maintained during and after a hydrogen burn that ignites all of the hydrogen that is released by the fuel clad-coolant reaction.

The latter four criteria are based on the limiting conditions that are created by significant beyond-DBAs, which bound the conditions that are generated during a DBA. The significant beyond-design-basis analyses shall consider the amount of hydrogen that is equivalent to that generated from a 100-percent fuel clad-coolant reaction. These criteria are evaluated in detail below.

NuScale proposed a PDC in place of GDC 41 in DCA Tier 2, Section 1.9.2, Table 1.9-3, Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS). The two are functionally very similar, with only the provisions associated with electric power eliminated from the PDC. A discussion of NuScale's reliance on electric power and the related exemption to GDC 17 can be found in the staff's evaluation of TR-0815-16497 (ADAMS Accession No. ML17340A524). Compliance with the PDC for the design is met by ensuring that the containment and all associated systems required to maintain safe shutdown can withstand the effects of the limiting hydrogen/oxygen combustion event for at least 72 hours. In effect, NuScale relies on no SSCs to control fission products as it is not necessary to reduce the concentrations because containment integrity and all associated SSCs required to

maintain safe shutdown can survive the effects of the constituents of the containment atmosphere following postulated accident conditions. As such, the staff finds the proposed PDC 41 meets the regulatory intent of GDC 41 and is acceptable.

Containment Mixing for Beyond-Design-Basis Accidents

The NuScale containment represents the final barrier to a radionuclide release and also serves as the means of inventory retention and as the primary heat transfer mechanism to the UHS, the reactor pool. As such, it is important to maintain containment integrity, especially in beyond-design-basis scenarios where the containment is initially intact.

Because of the unique nature of the NuScale design, a hydrogen combustion event requires a specific, stylized sequence of events to occur. Initially, containment pressure is well below atmospheric, as the containment is maintained at a near-vacuum during normal operations. This results in a limited initial inventory of oxygen. Additional oxygen can be added to containment through radiolysis. The applicant neglects inleakage, and states in TR-0716-50424-P, "Flow Visualization of Natural Convection in a Tall, Air-Filled Vertical Cavity," issued in 2006, that inleakage is negligible because of the leakage rate requirements and the relatively small potential differential pressures between a potential air source and the containment. In addition, the applicant states that shortly after the onset of the accident, containment pressure will likely be above atmospheric pressure, resulting in outleakage rather than inleakage, were leakage to occur. The staff agrees that, in comparison to the initial inventory and radiolysis, inleakage represents a relatively small potential term and can be encapsulated through the choice of slightly more bounding initial and generation terms, as done by the applicant.

Because of the aforementioned design choices, the NuScale containment is effectively inerted for many accident sequences. However, hydrogen combustion events are possible through the selection of appropriate hydrogen/oxygen mixtures. As such, although the applicant considers hydrogen generation up to the amount of hydrogen that is equivalent to that generated from a 100-percent fuel clad-coolant reaction plus that generated from hydrolysis, oxygen is generally the limiting reactant for producing a bounding combustion loading in the scenarios examined by the applicant.

Irrespective of the accident sequence, because of the nature of the design, a limiting amount of oxygen can be calculated, based on the chosen initial conditions and event timing. The applicant calculated a limiting amount of oxygen that could be present in the containment at 72 hours, then determined the amount of hydrogen that would result in the mixture in containment being flammable. The amount of hydrogen assumed is less than that which would be generated from a 100-percent fuel clad-coolant reaction, assuming the full inventory of hydrogen resulting from such a reaction would inert the containment. For this design, the staff considers the hydrogen and oxygen quantity assumption to be appropriately conservative, as the applicant has considered the limiting combustion event and provided a discussion on the effects of considering additional hydrogen generation.

Because the quantity of oxygen analyzed does not depend on the severe accident scenario (for a substantive set of severe accident scenarios), the calculation performed by the applicant is stated to be a bounding hydrogen combustion at 72 hours. The staff reviewed the severe accident scenarios and determined that there were no scenarios that maintained containment integrity that could result in an oxygen concentration exceeding that analyzed by the applicant. As such, the staff finds that the scenario constructed by the applicant represents the limiting

combustion event possible for the NuScale containment, given containment integrity is maintained before the combustion.

The guidance in 10 CFR 50.44(c) requires, in part, both that containment be adequately mixed and that hydrogen concentrations be limited to less than 10 percent. The applicant requested an exemption from 10 CFR 50.44(c)(2), related to combustible gas control and the 10 percent requirement. The staff's evaluation of the exemption request is located below. While the staff evaluated whether the NuScale design meets the intent of the combustible gas control requirement with a concentration exceeding 1 percent, mixing is still important, as the analysis is based on concentrations in a well-mixed containment, and inadequate mixing could result in conditions that are more limiting than those analyzed.

Consideration of mixing in the NuScale containment is relatively straightforward based on the single volume that makes up containment, although complicated somewhat by the relatively low driving forces within containment. Initially, the containment is at an extremely low gas inventory compared to later stages of the transient, and the addition of gas and steam from the RCS will be highly turbulent. Therefore, the initial degree of mixing has practically no impact on the development of the transient. For demonstrating mixing during the transient, the applicant chose to evaluate conditions at 72 hours. Earlier than 72 hours, conditions are generally more turbulent, and the containment is at a lower pressure, and after 72 hours, the applicant states that this represents a reasonable period of time to implement severe accident management guidelines to mitigate the accumulation of combustible gases. This time period aligns with that used in current regulatory precedent and is therefore acceptable.

The applicant evaluated mixing by calculating the degree of turbulence present in the containment at 72 hours. Because the containment is expected to be in a natural convection condition, one representation of the measure of turbulence in the fluid is the nondimensional Rayleigh number (Ra). Depending on the fluid geometry, the value of the Ra represents whether the fluid heat transfer is primarily convective or conductive. Reference 5.1.22 in TR-0716-50424-P, "Wright, J.L., Jin, H., Hollands K.G.T., Naylor, D., "Flow Visualization of Natural Convection in a Tall, Air-Filled Vertical Cavity," Int. Journal of Heat and Mass Transfer, Vol. 49, Issues 5–6, pp. 889–904 (2006)" states that a transition to bulk turbulent conditions in a tall vertical cavity with a hot surface and a cool surface (in air) occurs somewhere between Ra = 10,000 and Ra = 100,000. The staff reviewed this reference and determined it provides a reasonable basis for mixing for the specified geometry. This configuration is a reasonable approximation for the annular region that makes up the bulk of the containment volume.

Based on the calculated parameters at 72 hours, the applicant determined that the Ra would exceed this transition regime by at least one order of magnitude, likely more, depending on the exact hydrogen and air concentrations. Conditions within the containment at 72 hours may range over a variety of concentrations, temperatures, and pressures, depending on the nuances of the specific severe accident scenario, as discussed above in considering the spectrum of applicable accidents. Therefore, the staff independently calculated Ras for a range of expected conditions beyond those provided by the applicant at 72 hours. Although the applicant's calculated Ra range included parameter choices that may not have been conservative for some severe accident conditions, the values calculated by the applicant bounded those calculated by the staff. The NRC staff agrees that, in the annular region, bulk turbulent mixing is likely to occur for at least 72 hours.

The staff notes that both the values calculated by the applicant and those calculated by the staff are only representative of the annular region, not necessarily the dome region, where the

geometry is different caused both by the structure and other components that may impact the flow, and the temperature difference between the CNV (which is not submerged in this region) and the reactor vessel may not be as great.

Because of these potential discrepancies, the staff issued **RAI 8862, Question 06.02.05-1** and requested that the applicant provide a more complete justification for mixing in the dome region. In the response (ADAMS Accession No. ML17226A373), the applicant explained that the conditions in the containment atmosphere were evaluated for the first 72 hours of a transient. The applicant conservatively neglected the effects from flow out of the reactor vessel and steam condensation; this results in a simpler problem to evaluate: a rectangular region with a hot lower region and a cold upper region. Although the CRDMs and other piping do exist in the region, the volume occupied by the components is not substantial enough to obstruct flows or prevent mixing, according to the applicant.

Under these conditions, the applicant calculated an Ra substantially larger than the value indicative of the onset of turbulence, based on the available literature. The staff independently calculated conditions based on estimated conditions following an event likely to generate hydrogen and arrived at values within an order of magnitude of the applicant. Given the large margin between the calculated Ra and conditions indicative of turbulence, the staff finds it reasonable to conclude that the entirety of containment will be mixed (even before considering the effect of additional flow stimulated by steam flow from the RVVs and condensation on the walls). The applicant revised TR-0716-50424 to reflect the new discussion in the RAI response, and combined with the above considerations, the staff finds the response acceptable.

Exemption Request for control of hydrogen concentration in containment

Regulations in 10 CFR 50.44(c)(2) require that the concentration of hydrogen be limited, both globally and locally, to less than 10 percent, following an accident that releases an equivalent amount of hydrogen as would be generated from a 100-percent fuel clad-coolant reaction, and maintain containment structural integrity and appropriate accident mitigating features. The staff evaluated the beyond-design-basis scenarios identified in NuScale DCA Part 2, Tier 2, Table 19.2-2, "Core Damage Simulations for Severe Accident Evaluation," which could result in generating and releasing hydrogen into an intact containment and concludes that these scenarios do not have sufficient oxygen to support combustion. The staff also reviewed a bounding combustion analysis for a scenario at 72 hours that could support combustion and potentially detonation and confirmed that containment integrity would be maintained.

The staff concludes that, for likely beyond-DBA scenarios, preventing a loss of containment structural integrity, a loss of safe shutdown functions, or a loss of accident mitigation features does not require providing a hydrogen control system. These features are supported by the containment design. Therefore, the staff approves NuScale's proposed exemption from the requirements of 10 CFR 50.44(c)(2).

NuScale identified a related beyond-design-basis bypass scenario, whereby hydrogen is released from the containment to the space under the bioshield. Since NuScale has recently redesigned the bioshield, the staff is reevaluating this scenario to determine whether or not it could potentially lead to combustible or detonable conditions under the bioshield. The NRC staff's discussion of potential multimodule implications is in Section 19.2.4.2 of this report. The NRC staff is currently awaiting NuScale's response to RAI 9447, Question 03.11-19 (ADAMS Accession No. ML18320A253).

Equipment Survivability

A discussion of equipment survivability following a hydrogen combustion event is located in DCA Part 2, Tier 2, Section 19.2.3.3.8, “Equipment Survivability”. In part, 10 CFR 50.44(c) requires that all systems required to maintain containment integrity and safe shutdown must continue to perform their functions during DBAs and significant beyond-DBAs, including any effects resulting from the hydrogen generated following a fuel clad-coolant reaction involving 100 percent of the fuel cladding.

The equipment and functions identified by the applicant in Section 19.2.3.3.8 include the containment, containment isolation, and postaccident monitoring equipment. No further information was identified. Therefore, the staff issued RAI 8862, Question 06.02.05-2, requesting the applicant provide an explicit inventory of systems and components required to maintain and establish safe shutdown and support containment structural integrity.

In its response (ADAMS Accession No. ML17226A373), the applicant referenced a list in TR-0716-50424 of SSCs and their required state to establish safe shutdown and maintain containment integrity. The staff audited the design specifications, and requirements for these components in detail and determined that, during a postulated combustion event of the nature discussed above, these components will not be subject to environmental conditions that will inhibit the required function. The staff’s audit summary report will be issued in August 2019. Further, the applicant revised DCA Part 2, Tier 2, adding a statement to point back to the list of SSCs which were added to Section 19.2.4.

In defining the scope of the components required to operate under 10 CFR 50.44(c), this response is acceptable. The staff reviewed the provided list of components and views it as adequate, as it corresponds to the functional arrangement required during a combustion event, as discussed in TR-0716-50424 and DCA Part 2, Tier 2, Section 6.2.5. Further discussion of equipment survivability and containment performance as related to beyond-DBEs is located in Section 19.2 of this report.

Equipment must be demonstrated to survive environmental conditions, including but not limited to temperature and pressure resulting from a hydrogen burn. Although the temperature effects of combustion are explored in Section 3.3.5.6 of TR-0716-50424, no discussion related to the effect of short duration temperature pulses for equipment other than steel exist. The equipment required to function includes both the steel containment and the containment penetrations, which vary in material composition. Additionally, the calculation performed by NuScale examines the effect of temperature increases from hydrogen combustion events at only a very high level. The equipment qualification envelope defined by NuScale is either a combustion temperature increase of 75 degrees F (as defined in DCA Part 2, Tier 2, Section 19.2.3.3.8) or 300 degrees F (as defined in Section 3.3.5.6 of the TR), and both are stated to remain within the bounds of the containment design parameters. Accordingly, in NuScale RAI 9191, Question 06.02.05-8 (30820), the staff requested that NuScale provide additional discussion for temperature equipment survivability for all containment components for temperature pulses. In its response, the applicant described how the components will see very little heating because of the energy absorption from a short-term pulse event. As such, only the outer skin (less than 1/20th of an inch) of containment components is expected to exceed 300 degrees F, and only for a very short duration. Because the components walls are substantially thicker than the skin thickness, the average component temperature will remain below the design temperature; additionally, for a postulated combustion event where containment begins at a temperature of

250 degrees F, temperature increases are not expected to exceed the design temperature for in-containment components (550–600 degrees F). The staff finds this to be a reasonable description of the physical behavior expected following the pulse, and as the bulk of all containment components is expected to remain below their qualification temperature, finds this response acceptable.

Ultimately, containment integrity is the primary function that must be satisfied for the NuScale design to successfully retain the radionuclide inventory in the event of an accident. The applicant states this is accomplished either because of hydrogen effectively inerting the containment vapor space, with insufficient oxygen available for a combustion, or because the containment is capable of withstanding a combustion event resulting from a small amount of hydrogen (such that containment conditions are not inert). Discussion related to the quantity of oxygen and its contribution to containment inerting is located above in the containment mixing section, and the staff evaluation of the containment pressure resulting from a combustion event is discussed below in the structural analysis section.

As part of the review, the staff also audited the adiabatic isochoric complete combustion analysis performed by the applicant to provide additional scoping to confirm containment integrity during a severe accident. Although some inputs to the calculation could result in a lower calculated final pressure, the applicant's choice of initial containment pressure was sufficiently conservative (especially in the context of a severe accident) to bound these inputs.

Hydrogen Monitoring

With regards to hydrogen monitoring, the NuScale design provides for hydrogen and oxygen monitoring through the process sampling system, which draws from the discharge of the CES pumps. Oxygen monitoring is important, because as mentioned above, a postulated combustion event is generally limited by the oxygen concentration rather than the hydrogen concentration. The applicant states that the process sampling system is continuously available in the control room during normal operation, and that following an event, the CES is isolated. Further discussion of the process sampling system is located in Section 9.3 of this report.

Availability of the process sampling system is intended to satisfy the requirement to provide hydrogen and oxygen monitoring that is functional and reliable in the event of a significant beyond-DBA. Although the system is isolated on receipt of a containment isolation signal, the applicant states that CNV isolation valves for CES and CFDS could be opened in the event of a need for sampling postaccident, and the containment sampling system sample pump would be aligned to take a suction from the CES vacuum pump bypass line. This configuration allows for continuous gas concentration monitoring. Staff is currently preparing an RAI to determine that this configuration could be safely established in the context of a beyond-DBA in order to meet the requirements associated with monitoring for 10 CFR 50.44(c)(4). This is **Open Item 6.2.5-1**.

Structural Analysis of Containment Integrity

As discussed above, in the event of an accident resulting in combustible concentrations of gas inside the containment, the applicant elected to perform a bounding evaluation for a combustion event at 72 hours. This involves demonstrating the design is able to establish and maintain safe shutdown during a severe accident by showing that the containment can withstand the effects from a detonation involving a limiting mixture of hydrogen and oxygen. NuScale examined the effects of a detonation pressure pulse, using different methods, and chose the limiting pressure to evaluate the containment stresses.

One method the applicant used involved calculating a Trinitrotoluene (TNT)-equivalent pressure for the detonation. The applicant used this method to calculate a pulse duration for the estimated detonation. This method did not yield a limiting pressure, but in reviewing the calculated value, the staff determined it was possible that the TNT-equivalent pressure may not have been representative of containment conditions, and that selecting more appropriate parameters for the analysis could result in substantial differences in the resultant pressure parameters. Accordingly, the staff issued **RAI 9047, Question 06.02.05-3**, requesting that the applicant justify the chosen parameters for the TNT-equivalence calculation. In the response, the applicant provided a justification for the yield efficiency factor, containment conditions, and impacts on other SSCs within the containment. The justification included a discussion of related literature for the TNT-equivalent calculation and expanded on the conservative aspects of the NuScale analysis.

The applicant also noted that only the pulse-duration is obtained from the TNT-equivalent methodology, while peak pressure is derived from chemical kinetics and equilibrium of the gas mixture using the Chapman-Jouguet method. This is an important distinction, and the staff views this as appropriate, as NuScale has conservatively calculated a pulse duration. Use of the TNT-equivalent methodology for calculating pressure in this context would require additional scoping and is used in this context for additional information only.

Based on the additional information provided in the RAI response (ADAMS Accession No. ML17275A530), the staff finds the parameters used in the final analysis calculated using the TNT-equivalent method in TR-0716-50424 to be representative of conditions that would realistically exist inside containment during a severe accident, and the RAI response is, therefore, acceptable.

To determine the limiting pressure applied to the containment during a detonation pulse, the applicant calculated a deflagration-to-detonation transition pressure pulse resulting at 72 hours into a severe accident. The staff conducted a confirmatory analysis of the final calculated pressure and determined that, while there were some inputs chosen in the calculation that could result in higher calculated pressures, the calculation was performed appropriately, and the pressure calculated by the applicant bounded that calculated by the staff for realistic severe accident conditions.

Based on the above discussion, the staff finds the applicant calculated a realistic value for a limiting pressure pulse inside the containment resulting from a hydrogen detonation event at 72 hours. Evaluation of the effects of that pressure pulse on the containment is located in Section 3.3 of this report.

Based on the above evaluation, the staff finds that PDC 41, proposed by the applicant in lieu of GDC 41, is suitable for the NuScale design; the intent of GDC 41 is to provide, as necessary, systems to control the concentration of hydrogen or oxygen in the containment atmosphere following postulated accidents to assure that containment integrity is maintained. As discussed above, NuScale has provided analyses to demonstrate containment integrity is maintained and the associated systems required to maintain safe shutdown in the event of a postulated accident involving a combustion event.

Evaluation of Exemption Request #2, 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Reactors"

The staff evaluated NuScale's request not to provide a means of controlling hydrogen concentration in containment to comply with 10 CFR 50.44(c)(2). This regulation requires

limiting hydrogen concentrations following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 per cent fuel clad-coolant reaction to less than 10 per cent (by volume).

In part, 10 CFR 50.12(a)(1) states the Commission may grant exemptions from the requirements of the regulations, which are the following:

> *Authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security.*

The NRC is empowered to establish by rule or order, and to enforce, such standards to govern these uses as the Commission may deem necessary or desirable to protect the health and safety of the public. The NRC staff determined that granting of the licensee's proposed exemption will not result in a violation of the AEA.

The exemption requested by NuScale would not impact any design function, would not change any plant systems or the response of systems to postulated accident conditions. There would be no change to the predicted radioactive releases from postulated accident conditions. The plant response to previously evaluated accidents or external events would not be adversely affected, and no new accident precursors would be created. Therefore, the staff finds that granting this exemption would pose no undue risk to public health and safety.

The exemption requested would not alter the design, function, or operation of any structures or plant equipment that is necessary to maintain a safe and secure plant status. In addition, the exemption would have no impact on plant security or safeguards. Therefore, the staff finds that the common defense and security would not be impacted by this exemption.

Six special circumstances are listed in 10 CFR 50.12(a)(2), meeting any of which an exemption may be granted. Specifically, NuScale stated that the requested exemption met the underlying purpose of 10 CFR 50.44, to prevent a loss of containment structural integrity, safe shutdown functions, or accident mitigation features caused by a hydrogen combustion event. Denying this exemption would not improve safety or the ability of any SSCs to perform their functions. Therefore, the staff finds application of the regulation would not be necessary to serve the underlying purpose of the rule.

6.2.5.5 Combined License Information Items

There are no COL information items related to Section 6.2.5.

6.2.5.6 Conclusion

The staff finds that NuScale has addressed the required information related to combustible gas control for the design. However, because of the open item related to the post accident sampling system (PASS) exemption request (exemption #16), the staff was unable to finalize its conclusions as to acceptability.

6.2.6 Containment Leakage Testing

6.2.6.1 Introduction

The purpose of containment leakage testing design is to verify the leaktight integrity of the CNV to not exceed the allowable leakage rate in the TS that protect against uncontrolled releases to the environment. Appendix J specifies that this includes Types A, B and C testing.

NuScale has requested an exemption from the containment leak rate testing requirements of GDC 52, which states that the reactor containment and other equipment that may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing (ILRT) (Type A) can be conducted at containment design pressure. NuScale will still perform LLRTs, Types B and C.

NuScale has also requested an exemption from Appendix J to not perform the Type A leak rate tests required by Appendix J. Type A tests are intended to measure the primary reactor containment overall integrated leakage rate (1) after the containment has been completed and is ready for operation, and (2) at periodic intervals thereafter.

This exemption request is found in DCA Part 7, Revision 2, Section 7 (ADAMS Accession No ML18310A289).

6.2.6.2 Summary of Application

DCA Part 2, Tier 1: The CNV serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment. The containment leakage testing program performs the following safety-related functions that are verified ITAAC: the leakage rate for LLRTs (Type B and Type C) for the CNV to meet the requirements of Appendix J, and the preservice design pressure leakage test to ensure that the integrated leakage of the CNV meets the design criteria. Type B tests are intended to detect and measure local leaks for reactor containment flanged penetrations. Type C tests are intended to detect and measure local leakage rates for CIVs. In addition, the preservice design pressure test requires the CNV to be pressurized with water to design pressure, pressure held for 30 minutes, and no observed leakage visible from any joint.

DCA Part 2, Tier 2: The CLRT system is designed to verify the leak-tight integrity of the CNV by not exceeding the allowable leakage rate specified in the TS. The preoperational and periodic containment leakage testing requirements for CNV flanged openings (Type B) and the CNV piping penetrations for CIVs (Type C) are designed to meet the leakage acceptance criteria of Appendix J. An exemption has been requested to not meet the ILRT requirements of the GDC 52 requirements for the CNV (Type A) test.

The NuScale maximum allowable CNV leak rate, La, is 0.20 wt% of the containment air mass per day at the calculated Pa. The CNV leak rate testing is designed to verify that leakage from the CNV remains below the TS limit for CNV operability. The combined leakage rate for all penetrations and valves subject to Type B and Type C tests is limited to less than 0.60 La.

NuScale Exemption Request #7 proposes to not include Type A testing capability. Instead, NuScale credits the CNV design, inspections, preservice design pressure test, and, inservice testing to address potential leakage typically identified through Type A LRT.

Known leakage pathways will be tested preoperationally and periodically by Type B and Type C LLRTs. All CNV penetrations are either ASME Section III, Class 1 flanged joints, or ASME Section III, Class 1 welded nozzles with CIVs. All CIVs will be Type C tested. There are 41 total penetrations in the CNV, including the CNV main closure flange, all of which will be either Type B or Type C tested. The Type C tests for CIVs and the Type B tests for bolted flange penetrations are identified in the DCA Part 2, Tier 2, Table 6.2-4. Each of these Type B or C penetrations will be leak tested at each refueling.

All CNV flanged openings have bolted connections that are designed and constructed to ASME Section III, Class 1 criteria. All but one of these openings have identical concentric double O-ring metallic seal design, including the main CNV flange. The remaining seal is a double C-ring metallic seal design. All seals have a test port to facilitate Type B testing by pressurizing the space between the seals to Pa. The flanged openings, which contain EPA modules are provided with a test port for LLRT of the EPA module and are Type B tested periodically in accordance with the requirements of the owner's Appendix J testing program.

Conduct of the Appendix J, Types B and C testing, will be in accordance with ANSI/ANS 56.8, 1994. Schedules for the Types B and C testing will be in accordance with Appendix J, Option A, as committed to in TS SR 5.5.9. DCA Part 2, Tier 2, Section 6.2.6, specifies Types B and C testing in accordance with TS SR 5.5.9, which commits to Appendix J, Option A, frequencies. TS SR 3.6.1.1 requires visual exams and LRT, in accordance with the CLRT program. Satisfactory LLRT and ISI examination are required for containment OPERABILITY, per TS Bases 3.6.1, LCO.

ITAAC: NuScale DCA Part 2, Tier 1, Table 2.1-4, NPM ITAAC, lists the following:

- A leakage test will be performed of the pressure containing or leakage limiting boundaries, and CIVs. (Item #7)
- The leakage rate for LLRTs (Type B and Type C) for pressure containing or leakage-limiting boundaries and CIVs meets the requirements of Appendix J. (Item #7)
- A preservice design pressure leakage test of the CNV will be performed. No water leakage is observed at CNV bolted flange connections. (Item #23)

Technical Specifications: LCO 3.6.1 states that the containment shall be OPERABLE in Modes 1 and 2, and Mode 3 with RCS temperature hot greater than or equal to 200 degrees F:

> ***Technical Specification Bases:*** *3.6.1," Containment," LCO: The containment is designed to maintain leakage integrity less than or equal to 1.0 La. Leakage integrity is assured by performing local LLRT and containment ISI. Total LLRT leakage is maintained less than or equal to 0.60 (La) in accordance with Appendix J. Satisfactory LLRT and ISI examinations are required for containment OPERABILITY*.

Requested Exemptions: NuScale requests an exemption, in DCA Part 7, Exemption Request #7, from GDC 52, which requires that the containment be designed so that periodic ILRT can be conducted at containment design pressure.

Appendix J "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50, specifies Type A testing directly related to GDC 52. NuScale requests that, with approval of the GDC 52 exemption, the NPP DC rule include an exemption from the requirements of Appendix J, Type A, tests for plants referencing the NuScale certified design.

Technical Reports: TR-1116-51962-P, Revision 0 (ADAMS Accession No. ML17005A134), and incorporated by reference in the DCA.

6.2.6.3 Regulatory Basis

The following NRC regulations contain the relevant requirements for this review:

- GDC 52, "Capability for Containment Leakage Rate Testing," as it relates to the reactor containment and exposed equipment being designed to accommodate the test conditions for the CILRT (up to the containment design pressure).

- GDC 53, "Provisions for Containment Testing and Inspection," as it relates to the reactor containment being designed to permit appropriate inspection of important areas (such as penetrations), an appropriate surveillance program, and leakage rate testing at the containment design pressure of penetrations having resilient seals and expansion bellows.

- GDC 54, "Piping System Penetrating Containment," as it relates to piping systems penetrating primary reactor containment being designed with a capability to determine if valve leakage rate is within acceptable limits.

- 10 CFR Part 50, Appendix J, specifies requirements and acceptance criteria for preoperational and periodic testing of the leaktightness of the reactor containment and penetrations.

- 10 CFR 50.12 discusses Specific Exemptions from the regulations, and the criteria for required by the Commission for granting them.

- 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a facility that incorporates the design certification has been constructed and will be operated in conformity with the design certification, the provisions of the AEA, and the NRC's regulations.

The staff reviewed NuScale DCA Part 2, Tier 2, Section 6.2.6, using guidance provided in NuScale DSRS Section 6.2.6, which lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections. Exemption Request No. 7 is evaluated below.

6.2.6.4 Technical Evaluation for Exemption Request No. 7

GDC 52 requires that the containment be designed so that periodic ILRT can be conducted at containment design pressure, the underlying purpose of which is to provide design capability for testing that assures that containment leakage integrity is maintained and that CNV leakage does not exceed allowable leakage rate values.

Appendix J requires that primary reactor containments meet the containment leakage test requirements to provide for preoperational and periodic verification by tests of the leaktight integrity of the primary reactor containment (Type A), and systems and components that penetrate containment (Type B and Type C).

NuScale bases its exemption request to not meet the above regulations for Type A testing on meeting the underlying purpose of the regulation. NuScale provides analysis, CNV design specifications, design capability for inspection and examination, and design capability for testing, to provide assurance that the leakage integrity of containment is maintained and that CNV leakage does not exceed allowable leakage rate values.

NuScale has selected La, at Pa, to be 0.20 wt% of the containment air mass per day. For the NuScale design, La is established as a safety analysis operational limit and the TS limit for CNV operability. This maximum allowed leakage rate is the basis for the accident radiological leakage to the environment. Typically, this leakage would be demonstrated through a combination of preoperational and periodic Type A (containment integrated leak rate test (CILRT)) and Types B and C (LLRT) testing. NuScale has requested an exemption from performing any Type A testing.

NuScale asserts that its CNV design, in combination with the results from Types B and C testing, is sufficiently representative of accident conditions to provide confidence that the TS leak rate, La, would not be exceeded. In accordance with Appendix J, the sum of Types B and C test leakages must remain less than 0.60 La, an acceptance criterion that will be demonstrated at each refueling.

NuScale asserts that all contributors to potential CNV leakage could be identified and detected by LLRT and other means and that Type A testing is not necessary.

- NuScale states that, for the CNV, which is designed as an ASME Section III, Class 1 pressure vessel, leakage because of vessel design or fabrication flaws would be identified during the structural integrity test required by the ASME Code. This is a hydrostatic leakage test at CNV design pressure, with no visible leakage as its acceptance criterion.

- NuScale states that the CNV is 100 percent inspectable, both inside and outside, whereby aging-related flaws leading to potential leakage could be observed. ASME Section XI requires 100 percent vessel inspection every 10 years.

- At each refueling, the CNV and several of its bolted flange penetrations are opened and reclosed. NuScale has committed to inspecting the flanges, flange bolts, and flange gaskets for each opened penetration, including the main closure flange.

- NuScale proposes a preservice design pressure test to confirm the low-leakage design of the CNV.

NuScale asserts that, with the above-listed items and LLRT, it will demonstrate that La would not be exceeded during an accident.

- Type C piping penetrations are welded to the CNV, and the welds would be inspected, in accordance with ASME Section XI.

- CIVs, outside the CNV but in series in the piping penetrations, would be Type C tested in accordance with Appendix J. This testing would be performed at each refueling.

The CNV bolted flange connections (Type B) are designed with metallic seals, one seal in each of two concentric flange grooves.

- For each bolted flange connection, a test port is provided between the two seals to enable Type B testing of the flanges at Pa.

- Each bolted flange is Type B tested at each refueling, all flanges are as-found tested, and if the flange is opened, as-left tested.

- There are 21 bolted flange connections, including the CNV main closure flange.

- If a flange is opened, the bolts are inspected for flaws or indications of leakage and replaced, if necessary.

- If a flange is opened, the flange seals are also inspected for wear and replaced, if necessary.

- If a flange includes an EPA module, the EPA module will also have test ports and be Type B tested for module leakage at each refueling.

In accordance with Appendix J, the sum of Types B and C test leakages must remain less than or equal to 0.60 La. NuScale maintains that its design, which includes specifying low leakage for the CIVs, no leakage at piping penetration welds, and bolted flange design with low expected leakage, will demonstrate that LLRT expected leakage is very low in comparison with 0.60 La.

To support the assertion of low expected leakage regarding the CNV flange design, NuScale analyzed the CNV flanged opening bolts, in accordance with ASME III, NB-3231, for the design conditions. NuScale analyzed the contact at the sealing surface for all CNV bolted flange connections during two different accident conditions. NuScale also analyzed the contact at the sealing surface for the preservice design pressure test and for the CNV lift conditions. This analysis is found in NuScale calculation EC-A013-1691, Revision 2, September 24, 2018, “Containment Vessel Flange Bolting Calculation,” whose input, assumptions, and results will be described in NuScale TR-1116-51962-P, Revision 1. **This is Confirmatory Item 6.2.6-1**.

NuScale calculation, EC-A013-1691, Revision 2, analyzed the bolt design for the CNV flanged openings, using ANSYS and based in part on the flange seal design and seal specification as will be described in TR-1116-51962, Section 3, Revision 1. **This is Confirmatory Item 6.2.6-1**. This calculation also analyzed the contact pressure at the sealing surfaces for all CNV bolted flange connections during accident conditions, calculating flange contact pressures and corresponding flange gaps, based on CNV internal accident pressure of 1,100 psia, and temperatures representative of the maximum temperatures for each of the accident scenarios. These results were reviewed during the staff audit.

As described in TR-1116-51962, Draft Revision 1, three different CNV conditions were analyzed—the preservice design pressure conditions (1,100 psia and 140 degrees F), the RRV Inadvertent opening scenario, and the CVCS injection line break scenario. For the two accident scenarios, the flange gaps calculated at the CNV inner surface were very small and, at the outer surface, there were no or extremely small gaps, indicating no expected leakage at these accident conditions at all the flanges analyzed. For the preservice design pressure case, the flange gaps calculated were slightly larger (E-04 in.).

The staff agrees with these analyzed results—very small flange gaps for both accident scenarios and slightly larger gaps for the preservice design pressure case—in conjunction with the preservice design pressure test that demonstrates no leakage at the bolted flanges, that little or no Type B leakage would be expected during accident conditions. The leakage from

each bolted flange as measured by the Type B tests would be a reasonable representation of the bolted flange leakage under accident conditions. The preservice design pressure test is intended as a performance-based confirmation of the leak tightness of the CNV design at accident pressure. See ITAAC Item 23 in DCA Part 2, Tier 1, Table 2.1-4; DCA Part 2, Tier 2, Section 6.2.6.5.2; and TR-1116-51962-P, for the details of conducting the preservice design pressure test.

Calculation EC-A013-1691, Revision 2, also produced the CNV flange bolt preloads necessary to ensure that contact pressure be maintained at all the bolted flanges under accident conditions. The results of this analysis demonstrated that the specification of the flange seals, sizing of the flange bolts, and application of the calculated bolt preloads resulted in maintaining contact pressure from the inboard seal to the CNV outer surface, thereby making leakage at Pa at the seals unlikely. These preloads will be provided in TR-1116-51962, Revision 1, Table 3-1, “CNV Bolted Flange Calculation Applied Preloads.” **This is Confirmatory Item 6.2.6-2.**

For the CNV lift case, when the reactor containment module is lifted in the reactor pool, the deadweight load is used to evaluate the effect for joint tightness during the CNV lift. The contact load on the flange surface for all bolted flange connections are reduced from the preload contact load, but the contact load remains high and all of the joints remain in contact. This shows a tight joint is maintained during lifting. The tight joint keeps the flanges in contact and no separation occurs; the seal position attained during preload application does not change during lift. This analysis is included in calculation EC-A013-1691, Revision 2, and discussed in the response to **RAI 9474, Question 6.2.6-22** (ADAMS Accession No. ML18267A403) and in TR-1116-51962-P, Draft Revision 1.

The NRC based its review on the following:

- The NuScale CNV is designed as an ASME Section III, Class 1, pressure vessel, whose leakage because of vessel design or fabrication flaws would be identified during the structural integrity test required by the ASME Code. This is a hydrostatic leakage test at CNV design pressure, with no visible leakage as an acceptance criterion.

- The NuScale CNV is 100 percent inspectable, both inside and outside, whereby aging-related flaws leading to potential leakage could be observed. ASME Section XI requires 100-percent vessel inspection every 10 years.

- At each refueling, the CNV and several of its bolted flange penetrations are opened and reclosed. NuScale has committed to inspecting the flanges, flange bolts, and flange gaskets for each opened penetration, including the main closure flange.

- NuScale proposes a preservice design pressure test to confirm the low-leakage design of the CNV.

- The containment leakage test program includes successful Types B and C LRT at each refueling.

The staff approves the exemption from the requirement to perform Type A testing and the requirement to provide design capability for ILRT as stipulated in GDC 52. The staff has reasonable assurance that the NuScale CNV design, with its ISI, ASME design, and preservice design pressure test, would be shown to not exceed its TS allowable leakage rate.

Evaluation for Meeting the Exemption Criteria of 10 CFR 50.12, Specific Exemptions

Pursuant to 10 CFR 52.7, the Commission may, upon application by any interested person or upon its own initiative, consider exemptions from the requirements of 10 CFR Part 52. As 10 CFR 52.7 further states, the Commission's consideration will be governed by 10 CFR 50.12, which states that an exemption may be granted when (1) the exemptions are authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security, and (2) special circumstances are present. Specifically, 10 CFR 50.12(a)(2) lists six special circumstances for which an exemption may be granted. It is necessary for one of these special circumstances to be present in order for the NRC to consider granting an exemption request.

Authorized by Law

The NRC staff has determined that granting of the licensee's proposed exemption will not result in a violation of the AEA or the Commission's regulations because, as stated above, 10 CFR Part 52 allows the NRC to grant exemptions. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption is authorized by law.

No Undue Risk to Public Health and Safety

The proposed exemption will not impact any design function. There is no change to plant systems or the response of systems to postulated accident conditions. There is no change to the predicted radioactive releases because of postulated accident conditions. Furthermore, the plant response to previously evaluated accidents or external events is not adversely affected, and the change described does not create any new accident precursors. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the exemption poses no undue risk to public health and safety.

Consistent with Common Defense and Security

The proposed exemption will not alter the design, function, or operation of any structures or plant equipment necessary to maintain a safe and secure plant status. In addition, the changes have no impact on plant security or safeguards. Therefore, as required by 10 CFR 50.12(a)(1), the staff finds that the common defense and security is not impacted by this exemption.

Special Circumstances

<u>Underlying Purpose of the Rule</u>

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule. The staff finds that NuScale meets the underlying purpose of the regulation by providing CNV design specifications, design capability for inspection and examination, and design capability and commitment for leakage testing, both preoperationally and during the plant lifetime, to provide assurance that the leaktight integrity of containment would be maintained and that CNV leakage would not exceed allowable leakage rate values.

<u>Undue Hardship</u>

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever compliance would result in undue hardship incurred by others similarly situated. The staff recognizes that the NuScale containment design presents challenges to performing ILRT at

containment design pressure. Type A testing requirements were written for large, much lower pressure containments. The NuScale containment design would be subject to leakage testing acceptance criteria established for much lower pressures and subject to fewer temperature variations during Type A testing. The challenge for NuScale in meeting the acceptance criteria would be more challenging than that envisioned when the regulation was adopted. Granting this exemption for no Type A testing and design required by GDC 52 to provide for Type A testing would not prevent NuScale from demonstrating that the CNV leakage would not exceed allowable leakage rate values by other means.

Benefit to Public Health and Safety

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever an exemption would result in benefit to public health and safety. NuScale has proposed a design relying on passive safety systems and a design much simpler than conventional PWRs. NuScale has achieved improvement in safety over existing plants through simplicity of design and passive safety systems. The availability of passive safety systems for decay heat removal and emergency core cooling, as well as other features of the NuScale design, eliminates the need for external power under accident conditions. This is expected to result in the likelihood of a lower core damage frequency. In addition, an exemption from GDC 52 and Appendix J Type A tests would also result in benefit to public health and safety by maintaining occupational radiation doses as low as reasonably achievable. Granting this exemption would credit the simplicity of the design, the reliance on passive safety systems, and the benefit derived by avoiding operational radiation doses that would be incurred by Type A testing.

Circumstances Not Considered when the Regulation Was Adopted

The requirements for GDC 52 and the test criteria described in Appendix J were established for containment designs that were large, permanent welded steel plate structures with many internal subcompartments. Such designs resulted in areas that were difficult or impossible to inspect. Additionally, the extensive number of welds for the steel plate structure provided the potential for leakage that would be identified only through an ILRT, which does not exist for the NuScale pressure vessel CNV design. Type A testing is intended to indicate leakage from these inaccessible areas. The NuScale design allows for visual inspection of the entire inner and outer surface, which would lead to identifying unknown leakage pathways or degradation that could develop. Granting this exemption for no Type A testing and design required by GDC 52 to provide for Type A testing would recognize that the NuScale design primarily relies on the preservice pressure test, successful Appendix J Types B and C testing at each refueling, periodic ISI, and direct observation of the entire vessel to identify potential degradation or unknown leakage pathways for the remainder of the service life for wall leakage. Based on these design, inspection, and testing features, staff recommends granting this exemption request.

6.2.6.5 Evaluation of Compliance with Containment Leak Rate Testing Regulations and Guidance

Appendix J requires that primary reactor containments meet the containment leakage test requirements for preoperational and periodic verification by tests of the leak-tight integrity of the containment, and systems and components which penetrate containment of water-cooled power reactors. In addition, Appendix J establishes the acceptance criteria for these tests. The purposes of the tests are to assure that (a) leakage through the containment and systems and components penetrating the primary containment do not exceed allowable leakage rate values as specified in the TS or associated bases, and (b) periodic surveillance of reactor containment

penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment. The staff finds that the design, successful Types B and C LRT, in combination with meeting the acceptance criteria for the preservice design pressure tests and other inspections and actions discussed above, provides reasonable assurance of demonstrating that the NuScale CNV design will not exceed its TS leakage rate and therefore meets the underlying purpose of Appendix J.

Regulations in 10 CFR 52.47(b)(1) require that a DCA contain the ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the design certification, the provisions of the AEA, and the NRC's regulations. NuScale has proposed ITAAC for the Type B and Type C tests. NuScale has also proposed ITAAC for a preservice design pressure test to confirm that the bolted flange penetration design would be confirmed. The staff finds that successful Types B and C LRT, in combination with meeting the acceptance criteria for the preservice design pressure tests, provides reasonable assurance of demonstrating that the NuScale CNV design will not exceed its leakage rate TS requirement.

The staff reviewed the ITAAC as it relates to CLRT, as presented in NuScale DCA Part 2, Tier 1. The staff finds that the applicant has adequately identified ITAAC consistent with the requirements for Type B and Type C testing in DCA Part 2, Tier 1, Table 2.1-4, Item No. 7. The staff also finds that the applicant, as part of its exemption request, has proposed an ITAAC for a preservice CNV design pressure leakage test. The acceptance criteria for this test is no visible leakage at the bolted flange connections, after the CNV hydrostatic pressure has been held for 30 minutes. This is described in DCA Part 2, Tier 1, Table 2.1-4, Item No. 23. The staff finds the no visible leakage criteria an acceptable demonstration of the low leakage design of the containment. NuScale has committed to these acceptance criteria in **RAI 9474** supplemental response (ADAMS Accession No. ML19045A654). This will be incorporated into DCA, Tier 2, section 6.2.6, revision 3. This is **Confirmatory Item 6.2.6-3.** The staff's evaluation of the ITAAC described above is contained in Chapter 14 of this SER.

GDC 53 relates to the reactor containment being designed to permit appropriate inspection of important areas (such as penetrations), an appropriate surveillance program, and LRT at the containment design pressure of penetrations having resilient seals and expansion bellows. NuScale has committed to Type B testing in DCA Part 2, Tier 2, Section 6.2.6.2, "Containment Penetration Leakage Rate Test." The staff finds that the NuScale design complies with 10 CFR Part 50, Appendix J, and associated guidance for Type B testing. Therefore, the design meets the criteria of GDC 53 for inspection, surveillance and leakage testing of containment penetrations.

GDC 54 relates to piping systems penetrating primary reactor containment being designed with a capability to determine if the valve leakage rate is within acceptable limits. NuScale has committed to Type C testing in DCA Part 2, Tier 2, Section 6.2.6.3, "Containment Isolation Valve Leakage Rate Test." The staff finds that the NuScale design complies with 10 CFR Part 50, Appendix J, and associated guidance for Type C testing. Therefore, the staff finds that the NuScale design meets the criteria of GDC 54 for periodic operability testing of isolation valves in the containment penetrations.

The staff reviewed TS LCO 3.6.1, which states that the containment shall be OPERABLE in Modes 1 and 2 and Mode 3 with RCS temperature-hot greater than or equal to 200 degrees F. Containment operability is demonstrated by meeting the TS leakage limit of 0.20 percent in

24 hours. Based on the discussion above in the technical evaluation, the staff finds that successful Types B and C LRT, in combination with meeting the acceptance criteria for the preservice design pressure tests, would provide reasonable assurance of demonstrating that the NuScale CNV design would not exceed its leakage rate TS requirement. The staff's evaluation of the TS described above is contained in Chapter 16 of this SER.

6.2.6.6 Combined License Information Items

Table 6.2.6-1 NuScale Combined License Information Items for Section 6.2.6

Item No.	Description	DCA Part 2, Tier 2 Section
6.2-1	A COL applicant that references the NPP DC will develop a "Containment Leakage Rate Testing Program" that will identify which option is to be implemented under Appendix J. Option A defines a prescriptive-based testing approach whereas Option B defines a performance- based testing program. Appendix J requires that a program consisting of a schedule for conducting Type A, B, and C tests be developed for leak testing the primary reactor containment and related systems and components penetrating primary containment pressure boundary. Since the staff has recommended granting Exemption Request #7, this LRT program will address Types B and C testing to meet Appendix J for this COL item.	6.2
6.2-2	A COL applicant that references the NPP DC will verify that the final design of the CNV meets the design-basis requirement to maintain flange contact pressure at accident temperature, concurrent with Pa.	6.2

Programs and Manuals: 5.5.9 CLRT Program

A program shall implement the LRT of the containment as required by 10 CFR 50.54(o) and Appendix J, Option A, as modified by approved exemptions.

The maximum allowable La, at Pa, shall be 0.20 percent of containment air weight per day.

6.2.6.7 Conclusion

The staff approves the exemption request to not require Type A testing nor to require design capability for ILRT as stipulated in GDC 52. The staff has reasonable assurance that the NuScale CNV design, with its ISI, ASME design, and preservice design pressure test would be shown to not exceed its TS allowable leakage rate.

Based on its review of the information provided by NuScale and subject to the closure of Confirmatory Items 6.2.6-1, 6.2.6-2 and 6.2.6-3, the staff concludes that the NuScale DCA

Part 2, for the CLRT design, is acceptable and meets the relevant requirements of GDC 53, GDC 54, and Appendix J.

6.2.7 Fracture Prevention of Containment Pressure Boundary

6.2.7.1 Introduction

This section of the DCA, Part 2, Tier 2 describes fracture prevention of the reactor containment pressure boundary materials.

6.2.7.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.1.1, provides the DCA Part 2, Tier 1, information associated with this section.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 6.2.7, "Fracture Prevention of Containment Vessel," describes the design, as summarized, in part, below.

The NuScale CNV is an evacuated pressure vessel fabricated from low alloy and austenitic stainless steel. This vessel is maintained partially submerged in a below-grade pool. The CNV is an ASME Code Class MC containment designed, analyzed, fabricated, inspected, tested, and stamped as an ASME Code Class 1 pressure vessel. The applicant states that the design, fabrication, and materials of construction for the CNV system include margin to provide reasonable assurance that the CNV pressure boundary will not undergo brittle fracture and that the probability of rapidly propagating fracture will be minimized. The applicant cites adherence to the fracture toughness criteria of ASME Code, Section III, Subsection NB, "Class 1 Components," and ASME Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure," as the basis for this statement. The applicant cites the information presented in DCA, Part 2, Tier 2, Section 3.13, pertaining to bolting and threaded fasteners for discussion of those components.

6.2.7.3 Regulatory Basis

The staff reviewed NuScale DCA Part 2, Tier 2, Section 6.2.7, in accordance with SRP Section 6.2.7, "Fracture Prevention of Containment Pressure Boundary," Revision 1, issued March 2007.

The reactor containment system (CNTS) includes the functional capability of enclosing the reactor system and of providing a final barrier against the release of radioactive fission products. Fracture of the containment pressure boundary should be prevented for it to fulfill its design function. The NuScale design should address the following regulations:

- GDC 1 requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Section 6.1.1 addresses the applicant's discussion and the staff's evaluation.

- GDC 16 requires that the reactor containment and associated systems establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. Section 6.2.4 addresses the applicant's discussion and the staff's evaluation.

- GDC 51 requires that the reactor containment boundary be designed with sufficient margins to assure that, under operating, maintenance, testing, and postulated accident conditions, (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of a rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

The staff reviewed the DCA Part 2 to ascertain whether the containment pressure boundary materials meet the requirements of GDC 51. The staff's evaluation of GDC 1 is provided in Section 6.1.1.4, and the evaluation of GDC 16 is provided in Section 6.2.4.4 of this SER.

6.2.7.4 Technical Evaluation

The staff reviewed the information included in DCA Part 2, Tier 2, Section 6.2.7, in accordance with the guidance provided in SRP Section 6.2.7. Within the ASME Code, detailed fracture toughness requirements are placed on ferritic materials, as nonferritic materials exhibit sufficient inherent fracture toughness that additional requirements are deemed unnecessary. For example, the austenitic stainless steel used for the CNV lower shell, SA-965, FXM-19, was explicitly chosen specifically for its superior fracture toughness and resistance to neutron embrittlement. The staff noted that the NuScale CNV is to be designed, analyzed, fabricated, inspected, tested, and stamped as an ASME Code, Class 1 pressure vessel. This requirement ensures a higher degree of margin and hence assurance than the ASME Code Class MC designation traditionally used in large light-water designs.

The subject ferritic materials are acceptable if they meet the requirements of GDC 51, as it relates to the RCPB being designed with sufficient margins to ensure that under operating, maintenance, testing, and postulated accident conditions, the ferritic materials will behave in a nonbrittle manner and the probability of rapidly propagating fracture is minimized.

The applicant stated that the ferritic containment pressure boundary will conform to ASME Code, Section II, material specifications and meet the fracture toughness criteria and testing requirements of ASME Code, Section III, Subarticle NB-2300, "Fracture Toughness Requirements for Materials." The applicant further stated that ASME Code, Section XI, Appendix G, shall apply to all CNV ferritic materials classified as pressure-retaining components of the RCPB. Based on its review, the staff finds that the ferritic materials of the RCPB will be acceptably tested and demonstrated to meet the fracture toughness requirements for Class NB components as specified in Article NB-2300 of ASME Code Section III, Division 1. The staff finds that the application of ASME Code, Section XI, Appendix G, is appropriately addressed. A full evaluation of the use of NB is provided in DCA Part 2, Tier 2, Section 3.8.2. Consequently, the staff finds that NuScale has adequately met the requirements of GDC 51 through application of the above-noted ASME Code, Sections II, III, and XI, requirements and Appendix G.

6.2.7.5 Conclusion

Based on the review of the information included in the NuScale DCA Part 2, the staff finds that the fracture toughness of the materials used in the RCPB meets the fracture toughness requirements specified in GDC 51.

The staff concludes that, under operating, maintenance, testing, and postulated accident conditions, the NuScale DCA Part 2 provides reasonable assurance that the materials used in

the RCPB will not undergo brittle fracture and that the probability of a rapidly propagating fracture will be minimized, thereby meeting the requirements of GDC 51.

6.3 Emergency Core Cooling System

6.3.1 Introduction

The NuScale ECCS functions to provide core cooling during and after AOOs and postulated accidents including LOCAs. The ECCS consists of three RVVs mounted on the RPV upper head, which is directly connected to the pressurizer steam space, inside the CNV; two RRVs mounted on the side of the RPV inside the CNV; and associated RVV and RRV actuators located outside the upper CNV. All five valves are closed during normal plant operation and open following the receipt of an actuation signal resulting from applicable accident conditions such as low RPV water level or high containment water level. The ECCS also provides low temperature overpressure protection (LTOP) for the RPV. The ECCS valves are designed to actuate by stored energy and do not rely on power or a system that is not safety-related for actuation. The Tier 1 information associated with the ECCS is found in DCA Part 2, Tier 1, Section 2.1, and Section 2.5, “Module Protection System and Safety Display Indication System.”

After actuation, the ECCS is a passive system that vents steam from the RPV through the RVVs to the containment and returns the condensate back to the RPV downcomer region through the RRVs. The location of the RRV penetrations on the side of the RPV ensures that the RPV coolant level is maintained above the core and the fuel remains covered. The vented steam condenses on the internal surface of the CNV wall, which transfers heat through the CNV wall to the reactor cooling pool located outside the CNV. At least one RRV and two RVVs are required to open to provide adequate core cooling. The ECCS does not provide replacement or addition of inventory from an external source and does not provide a reactivity control function. The ECCS was designed to retain sufficient coolant inventory in the RPV to maintain the core covered and cooled without periods of refill and reflood.

6.3.2 Summary of Application

DCA Part 2, Tier 1: Tier 1 information associated with the ECCS is found in DCA Part 2, Tier 1, Section 2.1 and Section 2.5.

DCA Part 2, Tier 2: Information provided by the applicant in DCA Part 2, Tier 2, Section 6.3, is summarized here, in part, as follows:

DCA Part 2, Tier 2, provides information regarding the ECCS design as a whole and on a component basis. The DCA details the functional, reliability, protection, and environmental design bases, including design requirements; power source requirements; single-failure requirements; design-basis environment requirements; missile protection, seismic design, testing and inspection requirements; and instrumentation and system actuation signal requirements, as well as requirements for minimizing contamination.

The staff reviewed the application in accordance with the SRP Section 6.3 and DSRS Section 6.3 for “Emergency Core Cooling System,” the guidance provided in applicable RGs, and the NRC’s regulations.

ITAAC: The ITAAC associated with DCA Part 2, Tier 2, Section 6.3, are given in DCA Part 2, Tier 1, Section 2.1.2, Table 2.1-2.

Technical Specifications: The following generic TS is applicable to this area of review:

Generic TS 3.5.1, "Emergency Core Cooling System"

Technical Reports: DCA Part 2, Tier 2, Table 1.6-2, identifies TR-0916-51299, "Long-Term Cooling Methodology," Revision 0, January 2017 (ADAMS Accession No. ML17009A490), as being incorporated by reference into DCA Part 2, Tier 2, Section 6.3.

6.3.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in SRP Section 6.3 and are summarized below:

- GDC 2, as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the ECCS to perform its safety function in consideration of the most severe of the natural phenomena

- GDC 4, as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer), which are required to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents

- GDC 5, as it relates to SSCs important to safety not being shared among nuclear power units (NuScale reactor modules) unless it can be demonstrated that such sharing will not impair their ability to perform their safety function

- GDC 17, as it relates to electrical power system which shall permit functioning of structures, systems, and components important to safety.

- GDC 27, "Combined Reactivity Control Systems Capability," as it relates to controlling the rate of reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained

- GDC 35, as it relates to demonstrating that the ECCS would provide abundant emergency core cooling to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts

- GDC 36 and 37 as they relate to the ECCS being designed to permit appropriate periodic inspection of important components to assure the integrity and capability of the system and to permit appropriate periodic pressure and functional testing

- 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," in regard to the ECCS being designed so that its cooling performance is in accordance with acceptable evaluation models, which identifies and accounts for uncertainties in the analysis method and inputs. Alternatively, an ECCS evaluation model that may be developed in conformance with Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50

- 10 CFR 50.34(f)(1)(vii) for applicants subject to 10 CFR 50.34(f), with respect to eliminating the need for manual actuation of the RVV to assure adequate core cooling (TMI Action Plan item II.K.3.18 of NUREG-0737)

- 10 CFR 50.34(f)(1)(x) for applicants subject to 10 CFR 50.34(f), with respect to the RVV-associated equipment and instrumentation being capable of performing their intended functions during and following an accident, while taking no credit for equipment or instrumentation that is not safety-related, and accounting for normal expected air (or nitrogen) leakage through valves (TMI Action Plan item II.K.3.28 of NUREG-0737)

- 10 CFR 20.1406, "Minimization of Contamination," as it applies to the ECCS and its subsystems for both normal operations and recovery from accident conditions with respect to how the facility design and procedures will minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste

- 10 CFR 50.34(f)(2)(xi) for applicants subject to 10 CFR 50.34(f), with respect to the requirement that RCS relief and safety valves be provided with a positive indication in the control room of flow in the discharge pipe (TMI Action Plan item II.D.3 of NUREG-0737)

- 10 CFR 50.34(f)(2)(xviii) for applicants subject to 10 CFR 50.34(f), with respect to the requirement that instrumentation or controls provide an unambiguous, easy-to-interpret indication of inadequate core cooling (TMI Action Plan item II.F.2 of NUREG-0737)

- 10 CFR 52.47(b)(1), which requires that a DCA contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the AEA, and the NRC's regulations

- 10 CFR 50.46(b)(5), as it relates to requirements for long-term cooling to maintain the core temperature at an acceptably low value and decay heat removed for the extended period of time required by the long-lived radioactivity remaining in the core

The staff notes that the applicant has provided PDC for GDC 35. The PDC 35 proposed by NuScale is functionally identical to GDC 35 with the exception of the discussion related to electric power. A discussion of NuScale's reliance on electric power and the related exemption to GDC 17 can be found in Chapter 8 of this report, as well as the staff's evaluation of TR-0815-16497 (ADAMS Accession No. ML17340A524).

The staff notes that the applicant has provided PDC for GDC 27. The PDC 27 proposed by NuScale does not rely on poison addition via the ECCS. A discussion of PDC 27 and the related exemption to GDC 27 is in Section 15.0.6 of this SER.

Review interfaces with other SRP sections can also be found in SRP Section 6.3. Acceptance criteria adequate to meet the above requirements can be found in DSRS 6.3, and include, but are not limited to, the following:

- RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Safety Guide 1)," November 1970

- RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” March 2012

- RG 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems,” February 2010

- RG 1.29, March 2007

- RG 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” May 1989

- Design Certification/Combined License-Interim Staff Guidance (DCA/COL-ISG)-019, “Review of Evaluation To Address Gas Accumulation Issues in Safety Related Systems,” July 2010

6.3.4 Technical Evaluation

6.3.4.1 Functional Design Basis

The staff reviewed the ECCS design relative to the following functions stated in DCA Part 2, Tier 2:

- RCPB

- LTOP

- Core cooling in situations when cooling cannot be provided by other means (such as a LOCA)

Reactor Coolant Pressure Boundary

The ECCS components are also part of the RCS pressure boundary. GDC 14 states that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The detailed evaluation regarding how NuScale complies with these requirements is summarized in Chapter 3 of this document. The staff issued **RAI 8820, Question 03.09.06-1** (ADAMS Accession No. ML17213A540), related to the capability of the first-of-a-kind ECCS valve system to perform its safety function. NuScale is in the process of conducting ECCS valve design demonstration testing to show compliance with 10 CFR 50.43(e) for this new safety feature of the NuScale reactor design. This is being tracked as **Open Item 03.09.06-1**.

Low Temperature Overpressure Protection

The ECCS is also designed to ensure that RPV pressure-temperature limits are not exceeded. When RPV temperatures are less than 325 degrees F, the ECCS provides LTOP. The analytical limit for this LTOP function is a variable pressure value determined by the calculated saturation pressure for the RPV cold leg temperature plus 250 degrees F. When the wide-range RCS cold temperature is below 162 degrees F, the limit is fixed at a minimum pressure of 350 psia. The module protection system (MPS) logic provides the actuation signal that opens the RVVs. Once the wide range RCS cold temperature exceeds 32 degrees °F during startup, the LTOP function is disabled. This part of the ECCS operation is reviewed in detail as part of DCA Part 2, Chapter 5. Section 5.4 of this document describes the staff’s review of the

intended LTOP function of ECCS. The NuScale ECCS is designed to provide LTOP of the RPV so that GDC 30, “Quality of Reactor Coolant Pressure Boundary,” and GDC 31 requirements are met. GDC 30 and GDC 31 address the design, fabrication and testing of RCPB components to ensure that sufficient margin is provided to ensure they behave in a nonbrittle manner and also to ensure that rapid propagation failure is minimized. As mentioned above, that part of the operation and functional design are considered acceptable, as documented in DCA Part 2, Chapter 5.

GDC 27 Reactivity Control

The NuScale reactor is designed to not rely on the ECCS to maintain and control reactivity changes. The ECCS does not provide replacement or addition of inventory from an external source and does not provide a reactivity control function. After ECCS is actuated, the NPM is not designed to inject additional neutron poison into the RPV. The ECCS is designed to retain sufficient coolant inventory in the RPV to maintain the core covered and cooled without periods of refill and reflood. NuScale proposed PDC 27 and submitted an exemption request to the requirements of GDC 27. The staff review of this exemption request is incomplete and is tracked as an Open Item 15.0.6-6 in Section 15.0.6 of this Report.

Core Cooling Performance Evaluation

While the design of the ECCS is reviewed in this section of the report, the ECCS performance evaluation under accident conditions is provided in SER Chapter 15. The limiting cases for ECCS performance evaluation are the LOCAs and the inadvertent opening of ECCS valves.

6.3.4.1.1.1 Inadvertent Emergency Core Cooling System Valve Actuation

The inadvertent ECCS valve opening AOO analysis is provided in DCA Part 2, Tier 2, Section 15.6.6, and the staff’s evaluation is contained in Section 15.6.6 of this SER. The staff’s review of this evaluation remains incomplete. Therefore, the staff is tracking this as **Open Item 6.3-1**.

6.3.4.1.1.2 Secondary System Pipe Break Inside Containment

The behavior of the ECCS system following a secondary system pipe break inside containment is discussed in DCA Part 2, Tier 2, Section 6.3. Specifically, text in DCA Part 2, Tier 2, Section 6.3.2.2, provides the minimum and maximum flow coefficients and opening times for the RRVs and RVVs, which are consistent with the values assumed in the safety analyses. Additionally, TS LCO 3.5.1 justifies that the RVVs and RRVs operate as assumed in safety analyses. The staff evaluation of this event is incomplete and is described in Section 15.2.8 of this SER.

6.3.4.1.1.3 Loss-of-Coolant Accidents

The ECCS design provides fuel protection during postulated LOCAs. The system provides core cooling following the LOCA at a rate such that clad-metal water reactions are limited to negligible amounts, and fuel and cladding damage that could interfere with long-term effective core cooling is prevented.

DCA Part 2, Tier 2, Section 15.6.5, “Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary,” addresses the LOCA accident analysis and the corresponding ECCS performance evaluation. A spectrum of breaks

was analyzed. The evaluations are performed using TR-0516-49422, “Loss-of-Coolant Accident Evaluation Model,” Revision 0 (ADAMS Accession No. ML17004A138). The staff’s evaluation of TR-0516-49422 is ongoing and is being tracked as **Open Item 15.0.2-2**.

As part of the LOCA break location and size evaluation, the staff issued RAI 9358, Question 31191 (ADAMS Accession No. ML18026A519). This RAI requests that NuScale justify why the bolted ECCS valve-to-vessel connection provides confidence that the probability of gross rupture is extremely low if NuScale desires to treat the bolted connection as a break exclusion area. The staff’s evaluation of this RAI response is provided in Section 3.6.2 of this evaluation and staff has identified this as an open item. The staff is tracking this as **Open Item 03.06.02-1.** The relevant detailed evaluation of the LOCA analyses is provided in Section 15.6.5 of this document.

6.3.4.1.1.4 Long-Term Cooling

DCA Part 2, Tier 2, Section 6.3.3, states that containment heat removal in conjunction with ECCS operation provides sufficient capacity to reduce pressure to less than 50 percent of design pressure within 24 hours after a DBE. Additionally, DCA Part 2, Tier 2, Section 6.3.3, references TR-0916-51299 for the long-term cooling analysis which shows that the UHS provides sufficient cooling in the event of an accident in one NPM and permits the simultaneous safe shutdown and cooldown of the remaining NPMs, then maintains them in a safe shutdown condition. The reactor pool volume provides adequate heat transfer for more than 72 hours of postaccident cooling.

The applicant performed calculations to determine that maximum boron concentrations in the core region following actuation of the ECCS do not result in boron precipitation that interrupts that natural circulation flow in the core and long-term cooling capability. The calculation established a maximum boron concentration in the core region and an associated boric acid solubility temperature. DCA Part 2, Tier 2, Section 6.3.3, states that analyses show that, under bounding conditions of core mixing volume for long-term core cooling conditions, boron precipitation does not occur. DCA Part 2, Tier 2, Section 6.3.3, references TR-0916-51299, (ADAMS Accession No. ML17009A490), which is incorporated by reference in DCA Part 2, Tier 2, Table 1.6-2, for the details on the long-term cooling analysis. TR-0916-51299 is evaluated in SER Section 15.6.5.2, where the staff finds the boron precipitation analysis acceptable.

Although the boron precipitation analysis is considered acceptable, the staff identified that the long-term cooling evaluation does not address the potential for boron dilution in the core region caused by boron volatility and local transport. The NRC staff was concerned that boron dilution during long-term cooling could result in a more severe return to power scenario than what is currently identified in DCA Part 2, Tier 2, Section 15.0.6. Accordingly, the NRC staff issued **RAI 8930, Question 15-27,** requesting an update to the DCA to address this concern. Sections 15.6.5 and 15.0.6 of this SER document the staff review of long-term core cooling and the potential for a return to power after ECCS actuation. The concern related to boron volatility is discussed in Section 15.0.6 of this SER. The staff is tracking this as **Open Item 15.0.6-5**.

GDC 35 (through PDC 35) requires an ECCS to be designed to provide sufficient core cooling to transfer heat from the core at a rate such that fuel and cladding damage does not interfere with or prevent long-term core cooling. In DCA Part 2, Tier 2, Sections 15.6.5 and 15.6.6, NuScale provided its ECCS analysis results; however, the staff evaluations in Sections 15.6.5 and 15.6.6 are incomplete. Therefore, the staff cannot establish a finding regarding the exemption to GDC 35. This issue is being tracked as **Open Item 6.3-2**.

6.3.4.1.1.5 In-Vessel Debris Downstream Effects Evaluation

NuScale has assessed the ability of the ECCS to withstand the effects of debris generation and transport to ensure adequate short- and long-term core cooling following an accident. Evaluations were conducted to assess the debris impact on ECCS components, the fuel, and the core. The NuScale design minimizes the effects of postaccident debris accumulation in the containment volume (CNV) by eliminating or minimizing the sources of debris. The debris minimization is accomplished by the following:

- not permitting metallic or nonmetallic, fiber or mineral thermal insulation inside containment except for reflective metallic insulation and fibrous or organic insulation materials when encapsulated in a manner that prevents debris generation (i.e., conduit or sheathed)
- employing cleanliness controls for fabrication and preoperational and operational phases that meet and satisfy the applicable requirements of ASME NQA-1
- eliminating significant corrosion by insuring the purity of the reactor pool water
- eliminating materials, paint, or coatings within the NuScale CNV that contribute to corrosion-related hydrogen production or alters post-LOCA coolant chemistry to enhance SCC of austenitic stainless steel
- not permitting protective coatings on cabling
- eliminating protective coatings within the CNV
- requiring operational cleanliness requirements to minimize latent debris
- minimizing boron concentration in the core following ECCS actuation to eliminate large amounts of boron precipitation that could affect long-term cooling capabilities

Consequently, latent debris, defined in DCA Part 2, Tier 2, Section 6.3.2.5, as, for example, unintended dirt, dust, paint chips, fibers, pieces of paper, plastic, or tape, is expected to be the source of debris inside the CNV. A conservative estimate of latent debris present in the NuScale plant is provided by examining the latent debris present in current operating plants. In DCA Part 2, Tier 2, Section 6.3, NuScale followed the approach used for operating PWRs and determined the latent fiber debris amount as 5.6 grams (g) per assembly. The staff reviewed, in SER Section 6.2.2, the assumptions regarding latent debris and found them acceptable.

To maximize the debris loading in the core region, NuScale assumed all debris passes through the RRVs. It is estimated that 5.6 g/assembly fiber, 33.4 g/assembly particulate and a total amount of 27.1 lbm of chemical precipitates could reach the core. Typical PWR chemical precipitates are not expected to form in the NuScale containment as boron is used to control reactivity and buffering agents are not included in containment. The containment and components within the containment are either fabricated from or clad with stainless steel to preclude the production of corrosion components. In addition, the maintenance of a rigorous cleanliness program minimizes the collection of material that could react with boric acid to form other chemical species. Nevertheless, the NuScale calculations show that the indicated amount of chemical species can be tolerated in the core and not affect core heat transfer. The staff considers this approach to be conservative.

NuScale applied the fuel bundle head loss testing results obtained by AREVA for the same type of fuel bundle spacer grids and debris filter. The fuel bundle head loss testing conducted by AREVA applied 7.5 g/assembly fiber loading with higher fuel bundle inlet velocity and showed a bundle pressure drop increase by only 0.1 psid, which is much smaller than the estimated gravity driving head across the NuScale reactor core during the long-term cooling phase. Therefore, complete debris blockage at the core inlet is considered impossible. NuScale also performed calculations to show that any increased flow resistance from latent debris is minor and that clad temperatures remain below 800 degrees F.

In summary, NuScale uses information from operating plants to estimate the latent debris amount for the NuScale plant. Based on this information, the referenced analysis, and referenced testing, the staff reached the conclusion that the core inlet is not blocked by debris following a LOCA. In addition, NuScale described the analysis used as the basis for concluding that an acceptable post-LOCA peak cladding temperature (PCT) will be maintained in the reactor core at a postulated localized debris blockage of a fluid subchannel around the limiting fuel rod at the peak power location. Conservative assumptions used in the calculation further justify the conclusion that an acceptable PCT will be achieved following a reactor trip. Therefore, staff concludes that the in-vessel debris downstream effects will not result in a post-LOCA PCT greater than 800 degrees F.

6.3.4.2 System Design Features

Shared Systems

The NuScale ECCS is designed to be independent among all the NPM modules and allows each individual NPM to discharge heat through containment to the commonly shared pool. The UHS is the only commonly shared component for all NPMs. Therefore, the staff finds that the ECCS design meets the regulatory requirements of GDC 5 because ECCS components are not shared among NPMs.

Power Requirements

The NuScale ECCS is designed to passively cool the core for up to 30 days without an ac or dc power supply. Therefore, all the requirements of GDC 17 on electrical power are considered not applicable to the ECCS system design. NuScale also requested an exemption to have a dc power system that is not safety grade. The staff’s evaluation of the requested exemption to GDC 17 is evaluated in Chapter 8 of this report.

Instrumentation

Regulations in 10 CFR 50.34(f)(2)(xi) (TMI Action Plan II.D.3 of NUREG-0737) require proper indications in the control room of the status of ECCS. The NuScale reactor is designed for all five ECCS valves to have valve position indications available in the control room. In addition, solenoid power indication for the ECCS trip and reset valves is provided in the control room. The review of these valve position signals and solenoid power indication in the control room is provided in Chapter 7 of this document.

On April 15, 2019, NuScale informed the NRC staff (ADAMS Accession No. ML19105B292) that there are MPS design changes in progress related to the removal of the ECCS actuation on the reactor vessel (riser) level. This is also noted in the staff’s evaluation in Section 15.6.5 of this SER and is being tracked as **Open Item 15.6.5-1**.

System Boundary

DCA Part 2, Tier 2, Section 6.3.1, states that NuScale ECCS components do not extend beyond the CNV boundary, and all the pipe lines and ECCS valve bodies are enclosed in the containment. Therefore, the ECCS does not have any direct fluid mass exchange with the reactor pool or the air space outside of the CNV. Based on the information provided in DCA Part 2, Tier 2, Section 6.3.1, the staff finds that the ECCS design for the NPM satisfies 10 CFR 20.1406, because there is not direct fluid mass exchange with the reactor pool or the space outside the CNV.

Testing, Inspection, and Qualification

NuScale designed the MPS to provide the capability to perform periodic pressure and functional testing of the ECCS that ensures operability and performance of system components and the operability and performance of the system as a whole. The general installation and design of the ECCS provides accessibility for testing and inspection. The ECCS valves are designed to accommodate the preservice and inservice testing and inspection requirements of IWC-2500 and ISTC-3100 of the ASME Code with the valves in place. No maintenance, inspection, or testing of ECCS components is conducted during normal operations. Inspection and maintenance of the ECCS main valves are conducted only during NPM refueling outages. Because the CNV interior is inaccessible during normal operation, the required maintenance and inspections are performed in the NPM inspection bay during reactor outages. Therefore, the ECCS design satisfies GDC 36 and GDC 37, which require that an ECCS be designed to permit appropriate periodic inspection of important component and functional testing to assure the structural and leaktight integrity, the operability of the key components, and the system as a whole. DCA Part 2, Tier 2, Section 6.3 refers to the preservice and inservice testing and inspection description in Section 3.9.6. The initial testing of the ECCS verifies that the as-designed and as-constructed system functions as credited in the safety analysis. The staff evaluation is documented in Section 3.9.6 of this document. Preoperational testing is covered by DCA Part 2, Tier 2, Sections 3.9.6, 6.6, and 14.2. These sections address the structural and leaktight integrity of components and assurance of the operability of ECCS valves. The staff's review of structural and leaktight integrity of components and assurance of the operability of ECCS valves is contained in Section 3.9.6 of this report. The staff review of this area is currently ongoing and includes an open item associated with **RAI 8820, Question 03.09.06-1** (ADAMS Accession No. ML17213A540). The staff is tracking this issue as **Open Item 03.09.06-1**.

Environmental Requirements

Some active components of the NuScale ECCS are located inside the CNV. These components will be exposed to a harsh environment (high-temperature steam at high pressure) in the event of an accident such as a RCS break, a steamline break within containment, and inadvertent ECCS valve actuation. The environmental qualification of these valves is addressed in DCA Part 2, Tier 2, Section 3.11, and DCA Part 2, Tier 2, Section 7.2.2, and is reviewed under the corresponding sections of this report.

The components of the ECCS (valves, hydraulic lines, and actuator assemblies) are designed to be Quality Group A, seismic Category I components and meet the requirements of ASME Code, Section III, Subsection NB, 2013 Edition. Additional information addressing compliance with the applicable codes and classification of ECCS components is provided in DCA Part 2, Tier 2, Section 1.9 and Section 3.2.

The staff confirmed that seismic Category I is specified for the ECCS. The system and its components are designed to seismic Category I requirements. Equipment seismic qualification is addressed in DCA Part 2, Tier 2, Section 3.10, “Seismic and Dynamic Qualification of Mechanical and Electrical Equipment.” These areas are evaluated by the staff in Section 3.10 of this report.

NuScale described the missile protection in DCA Part 2, Tier 2, Section 3.5. The effects of pipe rupture are discussed in DCA Part 2, Tier 2, Sections 3.6.1 and 3.6.2. DCA Part 2, Tier 2, Section 3.3, addresses wind and tornado loadings. Fire protection is covered in DCA Part 2, Tier 2, Section 9.5. The staff evaluation of these effects is documented in Section 3.6.1 and Section 3.6.2 of this report.

System Reliability

ECCS reliability is provided by redundant valves remotely and separately actuated by two divisions of the ESFAS function of the MPS. The ECCS valves and actuators are assigned to separate divisions for instrumentation and electric power. One RVV is designed for actuation by either division of ESFAS by redundant trip valves controlled by separate divisions of instrumentation and electric power.

Redundancy of ECCS components, features, and capabilities is provided to ensure the system safety function can be accomplished assuming the single failure criteria. The capability to provide periodic pressure and functional testing of the ECCS, as discussed in Section 6.3.3.2.5 of this SER, ensures operability and performance of system components and the operability and performance of the system as a whole. The staff issued **RAI 8820, Question 03.09.06-1, 2, 3, and 4** (ADAMS Accession No. ML17153A377), to request more detailed information about the failure mode and effects analysis of the ECCS valve design and testing. The staff evaluation of the failure mode and effects is documented in Section 3.9.6 of this report. This issue is being tracked as **Open Item 03.09.06-1**.

In addition, the staff is investigating the single failure of an IAB and the subsequent impact on the analysis of inadvertent opening of the ECCS. Whether the IAB is subject to single failure is still unresolved, and the staff provided SECY-19-0036 (ADAMS Accession No. ML19060A162) to the Commission requesting guidance on how the staff should treat the IAB. The staff is tracking this as **Open Item 15.0.0.5-1**.

The ECCS main valves are not susceptible to water hammer. However, DCA Part 2, Tier 2, Section 6.3.2.5, indicates that the ECCS trip valve hydraulic lines and trip reset valves have some susceptibility to water hammer because of the possibility for flow in the hydraulic lines to transition to two-phase choked flow as the hot fluid in the lines is discharged to the CNV during ECCS valve actuation. Accordingly, the ECCS valve system is designed to accommodate the effects of water hammer. Therefore, the staff issued **RAI 9469, Question 31517** (ADAMS Accession No. ML18162A351), to request the addition of text to the DCA that describes how the ECCS actuator hydraulic line design, analysis, and tests demonstrate the ability to withstand water hammer effects. NuScale responded to this RAI with a description of a generic water hammer evaluation process. No specific analysis or evaluation was provided. During a followup public meeting, NuScale indicated it intended to provide a supplemental RAI response to address the specific evaluation about the ECCS valve hydraulic lines. In addition, the upcoming ECCS valve design demonstration testing has the intent to measure the pressure and temperature along the hydraulic lines, which can be used to determine the water hammer assumption validity. Therefore, this is **Open Item 03.09.06-1** associated with **RAI 8820, Question 03.09.06-1** (ADAMS Accession No. ML17213A540).

The ECCS actuator hydraulic lines also contain borated water. During normal operation, the pressure in these lines is equalized to RCS pressure. After each operating cycle, these lines are submerged in the reactor pool environment during refueling. As part of the valve component testing, NuScale will perform a hydraulic line flashing evaluation and propose relevant regular flushing maintenance requirements. Therefore, this is **Open Item 03.09.06-1** associated with **RAI 8820, Question 03.09.06-1** (ADAMS Accession No. ML17213A540).

Technical Specifications

TS relating to the ECCS are presented in DCA Chapter 16. The required actions and surveillance requirements were reviewed, together with the completion times allotted for corrective action and surveillance frequencies. The staff evaluation is documented in Chapter 16 of this document.

6.3.4.3 Combined License Information Items

DCA Part 2, Tier 2, Section 6.3.2.5, provided **COL Item 6.3-1**, directing a COL applicant that references the NPP DC to describe a containment cleanliness program that limits debris within containment. SER Section 6.2.2.4 references COL Item 6.3-1.

Item No.	Description	DCA Part 2, Tier 2 Section
COL (6.3-1)	A COL applicant that references the NuScale Power Plant design certification will describe a containment cleanliness program that limits debris within containment. The program should contain the following elements: • Foreign material exclusion controls to limit the introduction of foreign material and debris sources into containment. • Maintenance activity controls, including temporary changes, that confirm the emergency core cooling system function is not reduced by changes to analytical inputs or assumptions or other activities that could introduce debris or potential debris sources into containment. • Controls that limit the introduction of coating materials into containment. • An inspection program to confirm containment vessel cleanliness prior to closing for normal power operation.	6.3

6.3.5 Conclusion

The staff's conclusion regarding the overall acceptability of the applicant's DCA Part 2, Tier 2, Section 6.3, is contingent upon the applicant's satisfactory completion of the issues identified above as open items.

6.4 Control Room Habitability

6.4.1 Introduction

Control room ventilation is normally provided by the control room HVAC system (CRVS), described in DCA Part 2, Tier 2, Section 9.4.1, "Control Room Area Ventilation System." Should there be a loss of the CRVS, the control room habitability system (CRHS) provides breathable air to the control room during the first 72 hours. After 72 hours, CRVS, if restored, will continue providing HVAC service to the control building for the remainder of the event recovery period.

The NuScale design does not consider control room habitability a safety function. Consequently, CRHS is not safety-related, and is therefore not credited for mitigation of DBAs.

The Technical Support Center (TSC) is served by CRVS, not by CRHS.

6.4.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 3.1, "Control Room Habitability," contains Section 3.1.1, "Design Description." The CRHS supplies air to the control room envelope (CRE) during high airborne radiation or loss of the CRVS.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 6.4.1, "Design Basis," describes the CRHS as a system designed to provide clean breathing air to the CRE and to maintain a positive control room pressure for habitability and control of radioactivity should the CRVS become inoperable that is not safety-related.

ITAAC: The CRHS ITAAC are shown in DCA Part 2, Tier 1, Table 3.1-2, "Control Room Habitability System Inspection, Tests, Analysis, and Acceptance."

Technical Specifications: The CRHS has no TS or SRs.

Technical Reports: None.

6.4.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in SRP Section 6.4, "Control Room Habitability System," and are summarized below:

- GDC 4, as it relates to SSCs important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with postulated accidents

- GDC 5, as it relates to ensuring that sharing among nuclear power units of SSCs important to safety will not significantly impair the ability to perform safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining unit(s)

- GDC 19, "Control Room," as it relates to providing a control room from which actions can be taken to maintain the nuclear power unit in a safe condition under normal and accident conditions and providing radiation protection adequate to permit access and occupancy of the control room

- 10 CFR 50.34(f)(2)(xxviii), as it relates to evaluations of potential pathways for radioactivity and radiation and associated design provisions to preclude certain control room habitability problems

- 10 CFR 52.47(b)(1), which requires that a DCA contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the AEA, and NRC regulations.

Review interfaces with other SRP sections can also be found in SRP Section 6.4. Acceptance criteria adequate to meet the above requirements include the following:

- RG 1.196, "Control Room Habitability at Light-Water Nuclear Power Reactors," and RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," that provide acceptable guidance for meeting control room habitability requirements

- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

- RG 1.52, "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 Plant."

- ASME Code AG-1, "Code on Nuclear Air and Gas Treatment"

- RG 1.78, "Evaluating the Habitability of a Nuclear Power Plant Control Room during a Postulated Hazardous Chemical Release"

- TMI Action Plan Item III.D.3.4 (NUREG-0737), regarding protection against the effects of toxic substance releases, either on site or off site

- GSI, Item B-36, "Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems"

- GSI, Item B-66, "Control Room Infiltration Measurements"

- GSI-83, "Control Room Habitability (Revision 3)"

6.4.4 Technical Evaluation

Review of the CRHS in the DCA was performed in accordance with SRP Section 6.4, Section III, "Review Procedures." The system is composed of the eductor, silencers, valves, and piping in the CRE, along with the air compressor, air bottles, and air bottle racks in the control building.

On March 14, 2019, the applicant submitted a request for exemption from the requirements of GDC 19 (ADAMS Accession No. ML19073A331). The proposed exemption provides NuScale-specific PDC 19, "Control Room," to clarify the intent for meeting the required capability for remote safe shutdown. NuScale states that the proposed PDC 19 clarifies but does not, in effect, change the control room habitability requirements from those in GDC 19. If proposed PDC 19 is found acceptable and the exemption is granted, then NuScale's control room

habitability requirements would instead be given in PDC 19 in place of GDC 19. A conforming change would be made to this safety evaluation to reference the appropriate requirement. Because of the timing of the submittal relative to the writing of this safety evaluation section, the staff has not completed its review of the proposed exemption request; therefore, the following evaluation refers to the requirements in GDC 19. The effect of the staff's review of the GDC 19 exemption request on the evaluation of control room radiological habitability is **Open Item 6.4-1**.

6.4.4.1 Reliability

According to DCA Section 3.2, CRHS meets the NuScale QA program AQ-S. AQ indicates that pertinent augmented quality assurance requirements for SSCs that are not safety related are applied to ensure that the function is accomplished when needed based on that functionality's regulatory requirements. The CRHS follows industry codes: Compressed Gas Association Inc. G-7.1, Commodity Specification for Air, American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section VIII, Division 1, and ASME Code for Pressure Piping, B31.1. Design, procurement, and material selection are regulated by these codes. AQ-S indicates that the pertinent requirements of 10 CFR 50 Appendix B are applicable to SSCs that are not safety related and are classified as Seismic Category I or Seismic Category II in accordance with the quality assurance program. Section 3.2 of this SER provides detailed review on quality group and seismic category.

In the event of a loss of alternating current (AC) power, the DCA states that the CRE will receive air from the emergency air storage bottles automatically. These design features can maintain control room habitability during seismic and loss of AC event.

The air storage bottles are sized with up to 25 percent of the bottles out of service. The capacity of the emergency air has plenty of margin. There is no regulatory requirement mandating what percentage of the out of service bottles shall be assumed. The staff concludes that 25 percent is acceptable because it is conservative.

According to the DCA, the CRHS has built-in redundancy as described below. There are two supply lines, feeding the CRE with bottled air. One supply line has isolation valves that open automatically. Another supply line has a manual valve. There are two differential pressure instruments, measuring CRE differential pressure with respect to the surroundings. Also, there are two pressure relief lines connected to the vestibule.

The DCA states that instrumental indications are available to operators to show bottled air pressure, bottled air flow rate, CRE recirculation flow rate, CRE differential pressure with respect to the surroundings, air supply isolation solenoid valve position, CRE pressure relief isolation valve position, and CRE pressure, which controls pressure reducing valve. These instrumental indications provide the operators the status of system performance.

The CRHS is initiated automatically by plant protection system. The isolation dampers in the CRVS ducts that penetrate the CRE close automatically. A constant air flow rate is maintained by the orifice downstream of the pressure regulating valves. Operation of the CRHS can also be initiated by manual actuation.

The DCA states that the air bottles are located in a separate room outside of CRE, resulting in enough space for maintenance activities. Bottles are arranged in groups to allow isolation by group. There is an external air connection point that will allow the connection of a post 72 hour air supply from off-site air bottles to supply air and pressurization to the CRE for extended

accident conditions, if needed. The manual valve in the feeding line connecting air bottles and CRE is accessible in the CRE. The CRHS system layout provides ready access for maintenance and operation.

The CRHS is to be tested through the following programs: construction test, start-up testing, and ITAAC. NuScale maintains CRHS system reliability using Owner-Controlled Requirements Manual. Because the reliability of CRHS is consistent with a non-safety system as designed, the staff concludes that it is acceptable.

6.4.4.2 Control Room Envelope

The CRE was reviewed to determine if it included those facilities discussed in the acceptance criteria of the Control Room Emergency Zone in SRP Section 6.4. As stated in DCA Part 2, Tier 2, Table 1.9-2, aspects of RG 1.196 related to control room habitability design are applicable to the NuScale design. As stated in DCA Part 2, Tier 2, Section 6.4.5, the CRE periodic testing conforms to RG 1.197.

COL Item 6.4-5 directs a COL applicant to specify testing and inspection requirements for the CRHS, including CRE integrity testing.

Based on the above discussion and consistency with RG 1.197, the staff concludes that the CRE design is acceptable.

6.4.4.3 Ventilation System Criteria

The supply, return, smoke purge, and general exhaust ductwork serving the CRE are penetrations through the CRE boundary. Redundant isolation dampers are used to isolate the CRE from adjacent zones.

The CRE unfiltered inleakage with the CRHS pressurizing the CRE is assumed to be 10 cubic feet per minute (cfm) plus 5 cfm assumed for air lock operation. Because the CRE is pressurized, the unfiltered inleakage is expected to be 0 cfm. However, the dose analysis conservatively assumes 10 cfm of unfiltered inleakage.

Both RG 1.196 and 1.197 include guidance for meeting the requirements of GDC 19. Specifically, these RGs discuss having a CRE in which actions can be taken to operate the plant safely under normal conditions and to maintain the reactor in a safe shutdown condition during accident situations.
SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” dated March 28, 1994, states “The NRC regulations have several definitions for safe shutdown…” Appendix R, “Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979,” to 10 CFR Part 50, states that the phrase "safe shutdown" is used throughout the appendix as applying to both hot and cold shutdown.

The EPRI Utility Requirements Document (URD) for Advanced Light Water Reactors defined a safe stable shutdown condition as 215.6 degrees C (420 degrees F) and stated that passive safety systems need not be capable of achieving cold shutdown. EPRI based this contention on the belief that the passive DHRSs have an inherently high long-term reliability. EPRI contended that the passive advanced light-water reactor designs meet the GDC 34 requirements because they use a redundant safety-grade passive system that can operate at full RCS pressure and place the reactor in the long-term cooling modes immediately after shutdown, and because

conditions maintained by the systems are safe and fully consistent with the GDC 34 requirement to maintain fuel and RCPB within acceptable limits.

RAI 8845 was issued to get clarification regarding the need for NuScale to go to cold shutdown. NuScale responded to RAI 8845 (ADAMS Accession No. ML17304A666) stating that residual heat is removed from the reactor coolant system (RCS) via the DHRS, a passive system that reduces RCS temperature to the safe shutdown temperature of 420°F within 36 hours of reactor trip. NuScale explained that the DHRS transfers heat from the RCS to the ultimate heat sink (UHS), which has sufficient inventory to maintain the reactor at safe shutdown conditions beyond 72 hours. NuScale also explained that DCA Section 9.2.5.2 and DCA Table 9.2.5-2 demonstrate that even without makeup the safety-related UHS inventory remains above the top of the DHRS for 30 days.

Based on the information discussed above, the staff concludes that the operability of CRHS does not affect DHRS operation because no operator actions are required to remove decay heat. Therefore, staff's concerns regarding NuScale's need to go to cold shutdown are resolved and RAI 8845 has been closed.

Control Room Habitability Is Not a Safety Function

GDC 19 requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs.

In DCA Part 2 Section 6.4.1, NuScale states that the CRHS is not a safety-related system designed to provide clean breathing air to the CRE and maintain a positive control room pressure for habitability and control of radioactivity when conditions prohibit the CRVS from fulfilling these functions. Furthermore, in DCA Part 2, Tier 2, Section 6.4.4, NuScale states that the classification of CRHS as not safety-related is consistent with the definition in 10 CFR 50.2, "Definitions."

According to 10 CFR 50.2, "safety-related structures, systems and components" means those SSCs that are relied upon to remain functional during and following DBEs to assure the following:

> (1) The integrity of the RCPB
>
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition;
>
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures….

The NuScale design does not credit any control room operator actions to mitigate DBEs. Specifically, in DCA Part 2, Tier 2, Section 15.0.0.6.4, the applicant states that the plant is placed in a safe state automatically upon event initiation for at least 72 hours. As such, the operators are not required to perform any safety-related functions within that time period. The NuScale plant is maintained in a safe condition with natural circulation of the reactor coolant and natural circulation of the DHRS.

Based on the above review, the staff finds that NuScale's CRHS system can be categorized as a system that is not safety-related.

Isolation Dampers

The design and construction of isolation dampers are described in DCA Part 2, Tier 2, Section 9.4.1, "Control Room Area Ventilation System," and are reviewed in Section 9.4.1 of this SER.

Occupancy Limitations

The CRHS can provide breathable air to 20 personnel in the CRE for 72 hours with up to 25 percent of the bottles out of service during accident-operating conditions.

The CRHS has an isolation mode that isolates the outside air. The degree of leaktightness is 15 cfm allowable air inleakage. According to DCA Part 2, Tier 2, Table 6.4-1, the carbon dioxide level in the main control room (MCR) is maintained below 5,000 parts per million (ppm). According to RG 1.78, Table 1, the toxicity limit for CO^2 is 40,000 ppm. Therefore, the staff concludes that the CRE design is consistent with RG 1.78 and sufficient to remove concern for the buildup of CO^2.

Based on the above discussion, the staff concludes that the ventilation system criteria are acceptable.

Pressurization Rate

During normal and emergency modes of operation, one of two control room HVAC systems, CRHS or CRVS, maintain the CRE at a minimum 3.175 millimeters (mm) (0.125 in.) water gauge of positive pressure with respect to the surrounding areas. Since CRE is pressurized all the time, the staff concludes that the system pressurization rate is not an issue when normal operation mode is changed to emergency mode.

Atmosphere Filtration

The CRHS does not have an atmosphere filtration unit. The atmospheric filtration system is described in DCA Part 2, Tier 2, Section 9.4.1. The staff's evaluation is contained in Section 9.4.1 of this SER.

6.4.4.4 Control Room Radiological Habitability

GDC 19 requires that a control room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including LOCAs. It also requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions, without personnel receiving radiation exposures in excess of 0.05 sievert (Sv) (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

The NuScale ventilation and emergency habitability systems used to provide the control room occupants with protection from airborne radiological releases from postulated accidents are classified as not safety-related. These systems, the CRHS and the CRVS, are designed to ensure that the GDC 19 control room habitability dose criterion is met. In order for the system radionuclide exclusion and removal capabilities to be credited in the control room dose

analyses, the staff's expectation, as described in SRP 6.4 and RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," is that such systems would be classified as safety-related ESF systems to ensure that the system function to protect control room occupants can be relied upon under accident conditions for the duration of the accident. Because of the passive safety features and the designed response of the NuScale reactor, NuScale states that no operator actions are required or credited to mitigate the consequences of DBEs to maintain control or bring the reactor to safe shutdown. NuScale further states that no long-term monitoring from the control room is necessary postaccident. Monitoring of the reactor pool depth as the UHS to ensure adequate heat removal over the duration of the event can be achieved from outside of the control room by UHS water level instrumentation displays at the remote shutdown station, as described in DCA Part 2, Tier 2, Section 9.2.5. As described in DCA Part 2, Tier 2, Chapter 20, there are four reliable independent UHS and spent fuel pool level instruments which send information to the control room and the remote shutdown station that provide mitigation of beyond-DBE coping capability to maintain the decay heat removal key safety function over the long term.

To support the evaluation of NuScale's position that control room habitability is not a safety-related function (because there are no operator actions taken from the control room postaccident), the staff determined that it needed more information on the role of the control room operator in postaccident monitoring of combustible gas in containment. The DCA and related technical reports on the topic of combustible gas do not clearly describe if there is a need for operator actions within the control room to un-isolate the containment and set up use of the process sampling system (PSS) inline monitors during accident conditions to accomplish continuous monitoring of hydrogen and oxygen concentrations in the containment atmosphere in accordance with 10 CFR 50.44(c)(4). By letter dated January 31, 2019 (ADAMS Accession No. ML19031C975), the applicant requested an exemption from the TMI-related requirements related to postaccident sampling given in 10 CFR 50.34(f)(2)(viii). In this letter, the functional capability for combustible gas monitoring in the NuScale plant is described by NuScale as not being affected by the postaccident sampling exemption request. The staff has not completed its review of the proposed exemption request. However, in conjunction with its review of the postaccident sampling exemption request, the staff plans to ask additional questions about the operator actions and other clarifications to determine how NuScale plans to accomplish combustible gas monitoring following a DBE or beyond-DBE. The staff will evaluate the applicant's responses to questions on operator actions for combustible gas monitoring on the control room habitability assessment on the basis of demonstrating compliance with the dose requirements in GDC 19 as **Open Item 6.4-2**.

Additionally, because there are no required operator actions to be taken from within the control room to control or mitigate the event, NuScale has described that, if the control room becomes uninhabitable, the operators could evacuate. NuScale has stated that there is no impact on the accident management if operators are not available in the control room, therefore, postaccident control room habitability is not a safety function and the dose criteria in GDC 19 can be met with systems that are not safety-related. Given this difference in design and the accident response role of control room operators, the staff has evaluated the CRHS design considering the appropriate control room postaccident habitability and related system functions that are not safety-related.

NuScale states that its design provides two diverse, reliable, systems that are not safety-related to provide breathable air and exclusion or reduction of radionuclide intake during a DBE. DCA Part 2, Tier 2, Sections 6.4 and 9.4.1, provide descriptions of the CRHS and CRVS, including operation in accident conditions. DCA Part 2, Tier 2, COL Items 6.4-5 and 9.4-1, direct the COL

applicant to specify testing and inspection requirements for both systems and for CRE integrity testing. Controls over availability and reliability of both CRHS and CRVS will be in the Owner-Controlled Requirements Manual. DCA Part 2, Tier 2, Section 15.0.3, provides a description of the DBA control room dose analyses. The staff's evaluation of the DBA radiological consequences analyses, including methodologies and inputs related to the calculation of control room dose, is described in Section 15.0.3 of this SER. By letter dated January 31, 2019 (ADAMS Accession No. ML19032A146), NuScale described its proposal to submit a revision to the accident source term methodology as described in TR-0915-17565 and associated changes to portions of the DCA. NuScale's submittal of these proposed changes were received on April 19, 2019, (ADAMS Accession No. ML19112A220) and have not been reviewed for their impact on the staff's evaluation of control room radiological habitability. The staff's review of the revisions to the accident source term methodology as it affects the evaluation of control room radiological habitability is captured in **Open Item 6.4-3**.

As described in DCA Part 2, Tier 2, Sections 6.4 and 9.4.1, the CRVS is the normal ventilation system for the CRE and TSC. The CRVS has a single outside air intake and includes a filtration unit which is not in the flowpath during normal operation. There are two sets of radiation monitors in the CRVS, one in the outside air intake and the other after the filtration train. Upon detection of a high-radiation level in the outside air intake, an initiation signal automatically routes the outside air through the CRVS filtration unit to supply filtered outside air to the CRE and TSC. If high radiation levels are detected downstream of the CRVS filtration unit, an initiation signal automatically isolates the CRE from the outside air intake and surrounding areas (e.g., the TSC), and initiates the CRHS, which is not safety-related, to provide clean bottled air to pressurize the CRE and provide breathable air to maintain a habitable environment in the CRE for 72 hours. The CRHS is also automatically initiated if (1) normal ac power is not available to both the CRVS air handling units for 10 minutes, (2) normal ac power is not available to all the highly reliable dc power system—common (highly reliable dc power system-common (EDSS-C)) battery changers, or (3) there is a loss of dc power from EDSS-C to either division of the plant protection system. After 72 hours, if available, the CRVS will be used for the remaining duration of the accident.

The staff evaluated NuScale's design proposal that the CRVS and CRHS can be backup systems for each other, considering that both systems were designed to be reliable and capable of operation during accident conditions to maintain control room habitability. The staff's evaluation of the CRHS and CRVS system design and reliability is described previously in this section of the SER and in Section 9.4.1 of the SER. Although the CRVS and CRHS are not safety-related, the staff finds that the diverse, reliable design of the systems, coupled with lack of need for control room operators to take action in the control room to control and mitigate DBEs considering the NuScale design passive safety features and functions, assures sufficient capability to respond (as needed) to fulfill the control room habitability function, which is not safety-related.

To assess the capabilities of the CRVS and CRHS to maintain control room radiological habitability, NuScale modeled two separate control room analysis cases for each of the DBAs evaluated in DCA Part 2, Tier 2, Chapter 15. One case modeled the operation of the CRVS in supplemental filtration mode for the duration of the accident. The other case modeled the operation of the CRHS for 72 hours, followed by operation of the CRVS in supplemental filtration mode for the remainder of the accident. The CRVS filters were assumed to remove iodine with 99 percent efficiency. CRE unfiltered inleakage during operation of the CRVS was assumed to be 147 cfm through the envelope, plus 5 cfm for ingress and egress through the control room air lock. CRE unfiltered inleakage was assumed to be 10 cfm through the

envelope, plus 5 cfm for ingress and egress. TSC unfiltered inleakage during CRVS operation was assumed to be 56 cfm through the TSC ventilation envelope, plus 10 cfm for ingress and egress. The staff evaluated this proposal by NuScale and finds that these assumptions are consistent with guidance in RG 1.183, considering that systems not related to safety are credited for a function that is not safety-related.

An item of note is that, although the CRHS includes an external supply connection to be able to replenish the air bottles from an offsite source, this capability is not modeled in the dose analyses. Because the TSC is not served by the CRHS, for the case where the CRVS is not available or capable of providing acceptable air and pressurization, the TSC is assumed to be uninhabitable and the TSC function is transferred to another location in accordance with the emergency plan. Because the CRHS does not serve the TSC, the TSC habitability dose analyses only model the operation of the CRVS in supplemental filtration mode, unlike the control room habitability dose analyses, which model two cases.

Because one case of the control room dose analyses takes credit for operation of the CRVS supplemental filtration mode after the CRHS is exhausted at 72 hours, and, therefore, models rely on both systems to provide sufficient control room habitability, the staff requested in RAI 9534, Question 06.04-4, dated June 15, 2018, that NuScale provide an assessment of the sensitivity of the control room dose results to the operation of the systems. The staff requested this information to determine if the capability of the CRVS to provide pressurization and filtered outside air after 72 hours after the CRHS has been exhausted is necessary to meet the GDC 19 dose requirement. In responses dated August 30, 2018 (ADAMS Accession No. ML18242A685), as supplemented by letters dated October 2, 2018 (ADAMS Accession No. ML18275A426), and January 11, 2019 (ADAMS Accession No. ML19011A114), the applicant provided information to show that the dose to the operator in the control room, although higher than that reported in the DCA, is less than 0.05 Sv (5 rem) TEDE for all DBAs for the case where the CRVS supplemental filtration mode is not available after the CRHS is exhausted at 72 hours. In addition, the applicant's sensitivity analyses showed that the doses are also less than 0.0 Sv (5 rem) TEDE for all DBAs in the more extreme case where neither the CRHS nor the CRVS operate for the duration of the accident. The staff performed its own independent sensitivity analyses and found similar results. Based on the staff's review of the applicant's description of its sensitivity analysis and results, and on the staff's own confirmatory sensitivity analyses, the staff finds that RAI 9534 is closed and resolved and that the modeling and credit for the CRHS and CRVS system capabilities in the control room dose analyses are acceptable.

The DBA radiological consequence analyses are described in DCA Part 2, Tier 2, Section 15.0.3, and the results include the projected dose to control room operators from DBAs. The applicant's analyses show that the resulting dose to control room operators for the duration of the accident is less than 0.05 Sv (5 rem) TEDE for each of the DBAs evaluated for radiological consequences in DCA Part 2, Tier 2, Chapter 15. The NRC staff's review of the NuScale DBA radiological consequence analyses is discussed in Section 15.0.3 of this SER.

By reviewing the information provided in DCA Part 2, Tier 2, Sections 6.4 and 9.4.1, on CRHS and CRVS system capabilities and operation, and in DCA Part 2, Tier 2, Section 15.0.3, on the control room dose analyses, the staff has evaluated the applicant's dose analysis modeling assumptions related to the operation of the control room habitability and ventilation systems during a DBA and cannot yet reach a conclusion whether the dose analyses and the dose results are acceptable or whether the radiological habitability requirements of GDC 19 are met until resolution of **Open Items 6.4-1 through 6.4-3**.

6.4.4.5 Relative Location of Source and Control Room

Radiation Sources

The staff reviewed the NuScale design against SRP Section 6.4, Acceptance Criterion 5, “Relative Location of Source and Control Room, Radiation Sources,” as discussed below. In addition, review of the design against TMI Action Plan, Item III.D.3.4 and 10 CFR 50.34(f)(2)(xxviii), is discussed in this section.

SRP Section 6.4, Acceptance Criterion 5A, “Radiation Sources,” states that the control room ventilation inlets should be separated from the major potential release points by at least 31 m (100 ft) laterally and by 16 m (50 ft) vertically or be based on dose analyses. As it applies to radiation sources, 10 CFR 50.34(f)(2)(xxviii) requires that the applicant evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source-term release and to make the necessary design provisions to preclude such problems. This requirement is identified as TMI Action Plan Item III.D.3.4.

The staff reviewed the CRVS air intake location relative to radioactivity release points during an accident. The staff determined that the intake is physically separated and located far enough away from steam generator safety valves, relief valves, reactor building HVAC system exhaust, and any reactor building or turbine generator building wall penetration or door to prevent direct pathways of radioactivity. The short-term (accident) radiological atmospheric dispersion factors (χ/Qs) for the MCR and TSC are given in DCA Part 2, Tier 2, Table 2.0-1, “Site Parameters.”

The analysis of the radiological consequences of DBAs in the control room and TSC is discussed in DCA Part 2, Tier 2, Section 15.0.3. DCA Part 2, Tier 2, Section 6.4.4, describes the control room habitability design evaluation, including control room HVAC system operation for both the CRVS and CRHS during accidents. The impact on control room habitability is evaluated for each of the DBAs analyzed in DCA Part 2, Tier 2, Chapter 15, and the dose results are listed in DCA Part 2, Tier 2, Table 15.0-12, “Radiological Dose Consequences for Design Basis Analyses.”

In DCA, Revision 2, the limiting DBA for dose in the MCR and TSC was determined by NuScale to be the maximum hypothetical accident, an accident with significant core damage, with a resulting control room TEDE of 0.0214 Sv (2.14 rem). DCA Part 2, Tier 2, Section 12.3, provides information on the accident source and radiation shielding design for the control room dose analyses, and the DBA control room dose calculations in DCA Part 2, Tier 2, Chapter 15, include the contribution from direct radiation shine from the radioactive material in the MCR charcoal filters, the containment atmosphere, the radioactive plume outside the facility, and sky-shine. The staff has evaluated the control room air intake and the assumed unfiltered inleakage locations relative to the distance to the DBA release locations. The staff confirmed that the control room dose analyses in DCA Part 2, Tier 2, Chapter 15, used short-term (χ/Qs) for the control room that appropriately accounted for the relative locations of DBA releases to the control room intake and assumed inleakage location. More detail on the staff’s review of the control room (χ/Qs) can be found in Section 2.3.4 of this SER. The staff has determined that the NuScale design conforms to RG 1.183 guidance for control room habitability analyses because the release and control room receptor locations for the NuScale plant have been appropriately and specifically modeled in the analyses. More detail on the staff’s review of the control room DBA radiological consequences analyses is given in Section 15.0.3 of this SER.

Based on the above, the staff cannot yet reach a conclusion whether the NuScale design is satisfactory and complies with 10 CFR 50.34(f)(2)(xxviii) as it applies to radiation sources or meets Criterion 5A in SRP Section 6.4 until resolution of **Open Items 6.4-1 through 6.4-3**.

Toxic Sources

The staff reviewed the NuScale design against SRP Section 6.4, Acceptance Criterion 5B, "Toxic Gases." This criterion states that the minimum distance between the toxic gas source and the control room will depend on the amount and the type of the gas in question, and other site-specific parameters. SRP Section 6.4, Acceptance Criterion 5B, is met by demonstrating conformance with the guidance of RG 1.78, as described below.

As it relates to toxic gases, TMI Action Plan Item III.D.3.4 recommends that control room operators be adequately protected against the effects of the accidental release of toxic and radioactive gases such that the nuclear power plant can be safely operated or shut down under DBA conditions.

DCA Part 2, Tier 2, Section 6.4.4, includes COL item 6.4-1, which states a COL applicant that references the NuScale design will comply with RG 1.78, Revision 1.

6.4.4.6 Radiation Hazards

The staff reviewed the NuScale design against SRP Section 6.4, Acceptance Criterion 6, "Radiation Hazards." SRP Section 6.4, Acceptance Criterion 6B, states that applicants for DCs shall meet the requirements of GDC 19. Compliance with GDC 19 is discussed below. Radiological protection of the TSC, which is included in the CRVS envelope, is also addressed in this section. Control room operators, as well as TSC occupants, are protected from radiation sources by a combination of shielding and distance.

Radiation Shielding

The NuScale plant structures, as described in DCA Part 2, Tier 2, Section 6.4, provide radiation shielding for the MCR and TSC. DCA Part 2, Tier 2, Sections 12.3.1, "Facility Design Features," and 12.3.2, "Shielding," describe the radiation sources and shielding for the NuScale design. According to the NuScale design description, the MCR and TSC are protected from radioactivity inside the module containments and from releases into the space under the bioshield by the concrete walls in the control building that surround the MCR and TSC, as well as the concrete in the reactor building walls and water in the reactor pool that lies in between the radioactivity source and the MCR and TSC. Additionally, as described in the DCA, the occupants inside the MCR and TSC are also protected from the shine from the radioactive cloud that could pass over the control building by the concrete walls and the roof, and the MCR and TSC are shielded from the shine from radioactive material accumulating on the CRVS filters by the floor between the CRVS filters and the MCR/TSC. In the DBA radiological consequences analyses discussed in DCA Part 2, Tier 2, Chapter 15, the shielding around the MCR and TSC is modeled to estimate doses in the MCR and TSC from containment shine and direct radiation from the release cloud external to the building. The dose from containment shine and the external cloud are added to other dose pathways to give a total dose in the MCR and TSC that meets the dose criterion in GDC 19, as shown by the applicant in DCA Part 2, Tier 2, Section 15.0.3. Accordingly, the staff has determined that this design addresses the radiation shielding concerns expressed in SRP Section 6.4.

NuScale states that, since the CRE is well shielded with its enclosure inside the control building, the principal sources that affect operator dose in the control room are as follows:

- the radiation that bypasses the filter, because of filter inefficiency in the CRVS supply air
- the unfiltered inleakage from all other sources

Control Room Envelope Unfiltered Inleakage

The design value for CRE unfiltered inleakage during operation of the CRHS is 25.5 cubic meters per hour (cmh) (15 cfm). This value includes 8.5 cmh (5 cfm) for ingress/egress through the control room envelope 2-door airlock, which the staff agrees is in accordance with guidance in SRP Section 6.4 and RG 1.197 for assumed unfiltered inleakage through ingress/egress through an airlock. The total amount assumed for unfiltered inleakage includes 17 cmh (10 cfm) inleakage through the CRE boundary. Although maintaining the CRE at a positive pressure with respect to adjacent spaces helps minimize unfiltered inleakage, the staff has stated that it does not provide assurance that there will be no sources of unfiltered inleakage other than from ingress and egress, as discussed in RG 1.197. The assumption of 25.5 cmh (15 cfm) unfiltered inleakage into the CRE also does not include the measurement uncertainty for CRE integrity testing. However, the staff's guidance is that the measurement uncertainties only need to be added to measured inleakage values when they are above 170 cmh (100 cfm), as stated in RG 1.197, Section 1.4, "Test Results and Uncertainty."

As described in DCA Part 2, Tier 2, Section 6.4, after the CRHS is exhausted at 72 hours after initiation of the system, the CRVS will actuate to provide filtered outside air to the CRE. During operation of the CRVS in filtration mode, the design value for CRE unfiltered inleakage is 258.5 cmh (152 cfm). This value includes 8.5 cmh (5 cfm) for ingress/egress through the control room envelope 2-door airlock and 250 cmh (14 cfm) inleakage through the CRE boundary, including CRVS filter bypass. The 25.5 cmh (15 cfm) unfiltered CRE inleakage assumption for operation of the CRHS for 72 hours after initiation and the 258.8 cmh (152 cfm) unfiltered CRE inleakage assumption for operation of the CRHS in filtration mode are used as input assumptions in the radiological evaluations in DCA Part 2, Tier 2, Chapter 15, that show compliance with GDC 19.

Based on the above discussions on "Radiation Shielding" and "CRE Unfiltered Leakage," the staff determined that Section 6.4, Acceptance Criterion 6, is met, as it pertains to shielding for the NuScale design because the applicant has demonstrated protection from radiation hazards by distance and shielding using the above assumptions in its evaluation of the radiological consequences of DBAs in the control room.

Input Parameters to the Radiological Dose Analysis

The values presented in DCA Part 2, Tier 2, Section 6.4, and Chapter 15, that pertain to the modeling of the control room HVAC system response and unfiltered inleakage were reviewed by the staff. The staff compared the input to the Chapter 15 dose analysis to the characteristics of the design and determined that the values of CRE volume, carbon filter efficiency, high-efficiency particulate air (HEPA) filter efficiency, outside recirculation, and total air flow, clean bottled air injection rate and duration, roof slab thickness, and the MCR charcoal filter shine to the CRE below were satisfactorily modeled in the control room dose analyses. More detail on the staff's review of the control room DBA radiological consequences analyses is given in Section 15.0.3 of this SER.

Radiation Protection

As listed in DCA Part 2, Tier 2, Table 15.0-12, the CRE and control room HVAC system design ensures that the control room dose for all DBAs is less than the 0.05 Sv (5 rem) TEDE dose criterion specified in GDC 19, and the SRP Section 6.4 dose acceptance criteria. The staff determined that this adequately protects CRE occupants during postulated accident conditions and is therefore acceptable.

Because the NuScale design includes the TSC within the CRE served by the CRVS, the radiological consequences of DBAs in the TSC are also evaluated and shown by the DCA Part 2, Tier 2, Chapter 15, DBA radiological consequence analyses to result in doses less than 0.05 Sv (5 rem) TEDE. The staff's review of the DBA radiological consequences analyses, including the control room and TSC radiological habitability, is discussed in Section 15.0.3 of this SER.

Technical Support Center Habitability

NUREG-0696, "Functional Criteria for Emergency Response Facilities," issued February 1981, Section 2.6, "Habitability," provides guidance on TSC habitability, stating that the TSC should have the same radiological habitability as the control room under accident conditions and that TSC personnel should be protected from radiological hazards, including direct radiation and airborne radioactivity from in-plant sources under accident conditions, to the same degree as control room personnel. NUREG-0696, Section 2.6, also states that applicable criteria are specified in GDC 19 and SRP Section 6.4.

Guidance on the TSC ventilation system in NUREG-0696, Section 2.6, states that the TSC ventilation system should function in a manner comparable to the CRVS and that a TSC ventilation system that includes HEPA and charcoal filters is needed, at a minimum.

During operation of the CRVS in filtration mode, the design value for unfiltered inleakage to the TSC is 112 cmh (66 cfm). This value includes 17 cmh (10 cfm) for ingress/egress through the TSC door and 95 cmh (56 cfm) inleakage through the CRE boundary, including CRVS filter bypass.

Since the proposed TSC is incorporated into the CRE, it shares the same ventilation systems and is subject to the same radiological protection as the MCR during the operation of the CRVS. Because the TSC is not served by the CRHS, for the case where the CRVS is not available or capable of providing acceptable air and pressurization, the TSC is uninhabitable and the TSC function is transferred to another location in accordance with the emergency plan, as discussed in DCA Part 2, Tier 2, Section 13.3. The staff's evaluation of emergency planning is discussed in Section 13.3 of this SER. Based on the TSC being within the CRE and served by the CRVS, the above conclusions with respect to the radiological consequences in the control room also apply to the TSC, and the TSC design complies with SRP Section 6.4 and satisfies the dose criterion in GDC 19, as they apply to the proposed TSC location. Therefore, the staff determined that the TSC meets the guidance of NUREG-0696, Section 2.6.

Based on the above discussions, the staff cannot yet reach a conclusion whether the NuScale design conforms to the guidelines of NUREG-0696, Sections 2.4, 2.5, and 2.6, as they apply to TSC ventilation and habitability until resolution of **Open Items 6.4-1 through 6.4-3**.

6.4.4.7 Toxic Gas Hazards

As discussed in "Toxic Sources" under 6.4.4, issues on toxic gas hazards will be addressed by the COL applicant.

6.4.4.8 Equipment Qualification

The CRHS components are located below grade on the seismic Category I portions of the control building, which provides protection from potential adverse environmental conditions.

The SSCs required to provide breathing air inventory to the CRE for at least 72 hours are specified to be designed to seismic Category I criteria. These SSCs are the air storage bottles and the supply piping and components (including the regulating valves and actuation valves) to the CRE. The CRE pressure relief piping and components are also specified to be seismic Category I criteria.

CRHS components are not subject to pipe whipping or fluids discharging from nearby systems that could degrade their performance.

The staff concludes that CRHS materials are compatible with the expected environmental conditions encountered during all phases of plant operation. Therefore, CRHS meets the requirements of GDC 4.

6.4.4.9 ITAAC

The staff reviewed the following ITAAC requirements in DCA Part 2, Tier 1, Table 3.1-2:

- CRE air exfiltration test
- CRHS valves operation
- CRHS solenoid-operated valves function
- CRE heat sink temperature
- CRHS positive pressure

Staff's evaluation of the ITAAC listed above is contained in Chapter 14 of this SER.

6.4.4.10 Technical Specifications

RG 1.196, Section 2.3, states that "(t)echnical specifications require licensees to periodically perform measurements of several parameters important to maintaining CRH."

The NuScale design considers control room habitability a function that is not safety-related. Therefore, the measurements of control room habitability parameters are not included in its TS.

RAI 9321 (ADAMs Accession No. ML18079B138) requested that NuScale provide TS limiting conditions where appropriate. In its response to RAI 9321, NuScale provided an acceptable basis as to why TS limiting conditions were not required and further stated that, although CRE operability does not meet the criteria for inclusion within the TS, NuScale is including availability and reliability controls, including periodic testing, in the Owner-Controlled Requirements Manual. This is consistent with the policy for relocation of requirements that were formerly in TS at existing plants. These controls provide assurance, commensurate with the importance to safety of control room availability for the NuScale design, that the control room radiological

protection features will be maintained available in a manner similar to all non-TS design-basis features.

Based on the above discussion, the staff found the NuScale response acceptable and, therefore, RAI 9321 was closed.

Both RG 1.78 and RG 1.197 identify ASTM E741 as an effective method for testing for CRE inleakage. NRC Technical Specification Task Force Traveler TSTF-448 also proposes to include ASTM E741 into TS SRs. The NuScale design considers control room habitability a function that is not safety-related. Therefore, the CRE inleakage test is not included in its TS but is included on the Initial Test Program.

6.4.4.11 Sharing with Multiple Modules

According to GDC 5, SSCs important to safety shall not be shared among nuclear power plants unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

NuScale is one plant with up to 12 NPMs. The control room is shared for all 12 modules. NuScale states in Section 1.10, COL Item 1.10-1 that a COL applicant that references the NPP DC will evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk-significant SSCs of existing operating unit(s) and newly constructed operating unit(s) at the co-located site in accordance with 10 CFR 52.79(a)(31). The evaluation will include identification of management and administrative controls necessary to eliminate or mitigate the consequences of potential hazards and demonstration that the LCOs of an operating unit would not be exceeded. This COL item is not applicable to construction activities (build-out of the facility) at an individual NuScale power plant with operating NPMs.

Since NuScale's control room habitability is not a safety function, the staff concludes that sharing of CRHS SSCs in a single control room with multiple NPM units does not significantly impair their ability to perform their safety functions, including, in the event of an accident in one NPM, preventing an orderly shutdown and cooldown of the remaining NPMs. Therefore, GDC 5 is satisfied.

6.4.5 Combined License Information Items

Table 6.4-1 provides a list of habitability systems-related COL information item numbers and descriptions from DCA Part 2, Tier 2, Table 1.8-2.

Table 6.4-1 NuScale COL Information Items

Item No.	Description	DCA Part 2, Tier 2 Section
6.4-1	A COL Applicant that references the NuScale Power Plant design certification will comply with Regulatory Guide 1.78 Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."	6.4.4
6.4-5	A COL Applicant that references the NuScale Power Plant design certification will specify testing and inspection requirements for the control	6.4.4

	room habitability system and control room envelope integrity testing as specified in Table 6.4-4.	

The staff finds the above listing to adequately describe actions identified for the COL applicant or holder.

6.4.6 Conclusion

The staff's review was based on SRP Section 6.4, and it addresses plant-specific control room habitability data as discussed in DCA Part 2, Tier 2, Section 6.4, and the control room air conditioning system, as discussed in DCA Part 2, Tier 1, Section 3.1.1.

Because of the open items identified above, the staff is unable to make a finding.

6.5 Fission Product Removal and Control Systems

6.5.1 Engineered Safety Features Filter Systems

This section is not applicable to the NPP design because it does not use ESF filter systems or ESF ventilation systems to mitigate the consequences of a DBA. Although, the NPP design consists of a reactor building HVAC system that is not safety-related and includes filtering, it is not credited in the dose analysis. The staff's evaluation of the CRVS, which is not safety-related, and its filtration capabilities with respect to control room habitability is discussed in Section 6.4 of this SER.

6.5.2 Containment Spray System

This section is not applicable to the NPP design.

6.5.3 Fission Product Control Systems

The NPP design has no active system to control fission products in the containment following a DBA. The only ESF fission product control system credited to mitigate the consequences of a DBA in the NPP design is the CNV, in conjunction with the CIS. The CNV passively removes fission products by its inherent natural aerosol removal mechanisms, which include thermophoresis, diffusiophoresis, hygroscopicity, and sedimentation. The staff's evaluation of fission product removal by the CNV and CIS are discussed in Sections 15.0.3 and 6.2.4 of this SER.

6.5.4 Ice Condenser as a Fission Product Cleanup System

This section is not applicable to the NPP design.

6.5.5 Pressure Suppression Pool as a Fission Product Cleanup System

This section is not applicable to the NPP design.

6.6 Inservice Inspection and Testing of Class 2 and 3 Systems and Components

6.6.1 Introduction

ISI and testing are periodically implemented at nuclear power plants to assess the structural and leaktight integrity of ASME Code Class 2 and 3 systems throughout the operating lifetime of the facility. As required by 10 CFR 50.55a(g)(3), reactor designs certified on or after July 1, 1974, are required to be designed to provide access to enable the performance of ISI of ASME Code Class 2 and 3 systems. Typically, a design should be developed that permits the implementation of an ISI program consistent with the provisions of ASME Code, Section XI, as supplemented by augmented ISI requirements in 10 CFR 50.55a. However, based on the specific attributes of a reactor design, additional augmented ISIs may need to be proposed, and designed for, to support the design's compliance with, for example, GDC 36; 39; 45, "Inspection of Cooling Water System; and 55, "Reactor Coolant Pressure Boundary Penetrating Containment," as applicable to the Class 2 or 3 system.

6.6.2 Summary of Application

DCA Part 2, Tier 1: There are no Tier 1 entries for this area of review. The system-based descriptions of DCA Part 2, Tier 2, Chapter 2, "Unit Specific Structures, Systems, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria," address ASME design-related Code requirements for system components.

DCA Part 2, Tier 2: The applicant has provided a DCA Part 2, Tier 2, description of its ISI program for Class 2 and 3 components in Section 6.6, summarized here, in part, as follows:

Section 6.6 of the DCA Part 2 states that the preservice inspections (PSIs) and ISIs are to be conducted in accordance with the ASME Code, Section XI. Aside from describing the use of ASME Code, Section XI, Section 6.6 addresses the ISI requirements for the components and configurations unique to the NuScale design.

The application specifically addresses the following eight areas:

(1) components subject to examination
(2) accessibility
(3) examination techniques and procedures
(4) inspection intervals
(5) examination categories and requirements
(6) evaluation of examination results
(7) system pressure tests
(8) augmented inservice protection programs

In each of these areas, the application references the applicable ASME Code requirements.

6.6.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review and the associated acceptance criteria are given in SRP Section 6.6 and are summarized below:

- 10 CFR 50.55a, as it relates to the requirements for inspecting and testing ASME Code Class 2 and 3 as specified in ASME Code, Section XI

- GDC 36, as it pertains to designing the ECCS to permit appropriate periodic inspection of important components, such as spray rings in the RPV

- GDC 37, as it pertains to designing the ECCS to permit appropriate testing to assure structural integrity, leaktightness, and the operability of the system

- GDC 39, as it pertains to designing the CHRS to permit appropriate periodic inspection of important components to assure the integrity and capability of the system

- GDC 45, as it pertains to designing the cooling water system to permit appropriate periodic inspection of important components to assure the integrity and capability of the system

- GDC 55, as it pertains to the application of appropriate requirements, such as higher quality design, fabrication, and testing, additional provisions for ISI, to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them to be provided as necessary to assure adequate safety

Review interfaces with other SRP sections can be found in SRP Section 6.6.

6.6.4 Technical Evaluation

The staff reviewed DCA Part 2, Tier 2, Section 6.6, "Inservice Inspection of ASME Code Class 2 and 3 Components," in accordance with SRP Section 6.6, "Inservice Inspection and Testing of Class 2 and 3 Components." Section 6.6 of the DCA Part 2 details the proposed requirements for the ISI and testing of the Class 2 and 3 components. The PSIs and ISIs are to be conducted in accordance with ASME Code, Section XI. The proposed initial ISI program will incorporate the latest edition and addenda of the ASME Code, Section XI, approved in 10 CFR 50.55a(b) on the date 18 months before initial fuel load, subject to the conditions listed in 10 CFR 50.55a(b). The proposed ISI of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 18 months before the start of the 120-month inspection interval (or the optional ASME Code cases listed in RG 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 18, that are incorporated by reference in 10 CFR 50.55a(b)), subject to the conditions listed in 10 CFR 50.55a(b).

6.6.4.1 Components Subject to Inspection

PSIs and ISIs are performed on ASME Code Class 2 and 3 components in accordance with ASME Code, Section XI. These components are considered RG 1.26, Quality Group B and C components.

The ASME Code Class 2 boundaries, based on RG 1.26 (Reference 6.6-3), for Quality Group B, are as follows:

- portions of the RCPB but which are excluded from the Class 1 boundary, pursuant to Section 5.2.4

- systems or portions of systems that are designed for reactor shutdown or residual heat removal

- portions of the steam systems extending from the outermost CIV up to but not including the turbine stop and bypass valves and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during normal reactor operation

- systems or portions of systems that are connected to the RCPB and are not capable of being isolated from the boundary during normal reactor operation by two valves, each of which is normally closed or capable of automatic closure

- systems or portions of systems that are designed for the following:

  – emergency core cooling
  – postaccident containment heat removal
  – postaccident fission product removal

The ASME Code Class 3 boundaries, based on RG 1.26 (Reference 6.6-3), for Quality Group C, are not part of the RCPB but include the following:

(1) cooling water systems or portions of cooling water systems that are designed for emergency core cooling, postaccident containment heat removal, postaccident containment atmosphere cleanup, or residual heat removal from the reactor and from the spent fuel storage pool (including primary and secondary cooling systems), including portions of these systems that are required for their safety functions and that do not operate during normal operation and cannot be tested adequately but are, however, included in Class 2

(2) cooling water and seal water systems or portions of these systems that are designed for the functioning of other Class 2 or 3 components and systems

(3) systems or portions of systems that are connected to the RCPB and are capable of being isolated from that boundary during normal reactor operation by two valves, each of which is normally closed or capable of automatic closure

(4) systems other than radioactive waste management systems, not covered by items (1), (2), and (3) above, that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (RG 1.3, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors, and RG 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors), that exceed 0.05 Sv (0.5 rem) to the whole body or its equivalent to any part of the body

DCA Part 2 Section 6.6 covers the process for inspecting all components defined as Class 2 and 3 by the ASME Code. The description of pipes that are considered Class 1, 2, and 3 are described in the DCA Part 2 “Certified Design Descriptions and Inspections, Tests, Analyses, & Acceptance Criteria (ITAAC) PART 2—TIER 1,” Table 2.1 1, “NuScale Power Module Piping Systems.”

6.6.4.2 Accessibility

Design for accessibility is required to meet the GDC, ASME Code, and 10 CFR 50.55a. The design and arrangement of system components are acceptable if an adequate clearance is

provided in accordance with ASME Code, Section XI, Subarticle IWA-1500, “Accessibility.” Regulations in 10 CFR 50.55a(g)(3)(ii) require Class 2 and 3 components, including supports, to be designed and be provided with access to enable the performance of inservice examination of these components, in addition to meeting the PSI requirements set forth in the editions and addenda of Section III or XI of the ASME Code of record.

In accordance with 10 CFR 50.55a(g)(3), Class 2 and Class 3 systems and components (including supports) are designed and provided with access to enable the performance of ISI.

6.6.4.3 Examination Techniques and Procedures

The examination techniques used for ISI include visual, surface, and volumetric examination methods. The examination procedures describe the examination equipment, inspection techniques, operator qualifications, calibration standards, flaw evaluation methods, and records. The techniques and procedures meet the requirements of ASME Code, Section XI, Articles IWA-2000, IWC-2000, and IWD-2000. PSI and subsequent ISI are conducted with equivalent equipment and techniques. Ultrasonic examination, equipment, and procedures are qualified in accordance with ASME Code, Section XI, Appendices VII and VIII.

Table 6.6-1 in the DCA Part 2 describes the Class 2 and 3 components, including the vessel welds, nozzle welds, and piping welds, their examination category, and the examination methods. The DCA Part 2 also describes the required examination volume, the acceptance criteria, and the details for the inspections.

6.6.4.4 Inspection Intervals

The examination program for the 120-month inspection interval is described in the Reactor Module Test Inspection Elements report and are fully developed in the Owners ISI Program to be developed as part of the COL (see COL Item 6.6-2). The initial ISI program incorporates the latest edition and addenda of the ASME Code approved in 10 CFR 50.55a(b) on the date 18 months before fuel load. ISIs of components and system pressure tests conducted during successive 120-month inspection intervals conform with the requirements of the latest edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 18 months before the start of the 120-month interval, subject to the conditions listed in 10 CFR 50.55a(b).

6.6.4.5 Evaluation of Examination Results

The evaluation of Class 2 and 3 examination results will follow the requirements of ASME Code, Section XI, Article IWA-3000. Evaluations of Class 2 and 3 examination results are also performed in accordance with Articles IWC-3000 and IWD-3000, respectively. The results of the examinations and evaluations are documented in accordance with Article IWA-6000 and the procedures for repair and replacement of Class 2 and 3 components are in accordance with Section XI, Article IWA-4000. This is acceptable because it meets the applicable requirements of ASME Code, Section XI.

6.6.4.6 System Pressure Tests

DCA Part 2 Section 6.6.7, “System Pressure Tests,” states that they will be conducted in accordance with the requirements of ASME Code, Section XI, and the TS requirements for operating limits during heatup, cooldown, and system hydrostatic testing. Since the applicant’s methodology for performing pressure testing of the Class 1 boundary and components meets

the requirements of the ASME Code, the methodology for performing system pressure testing is, therefore, acceptable.

6.6.4.7 Code Exemptions

No code exemptions are requested in this section.

6.6.4.8 Relief Requests

No relief requests are described in this section.

6.6.4.9 Code Cases

The ASME Code cases referenced by the COL application are reviewed for acceptability and compliance with RG 1.147. Code cases not specifically referenced in RG 1.147 will be reviewed and accepted on a case-by-case basis.

No specific code cases are described in this section, although the DCA Part 2 does refer to the optional ASME code cases listed in RG 1.147.

6.6.4.10 Inspections, Tests, Analyses, and Acceptance Criteria

The ITAAC associated with DCA Part 2, Tier 2, Section 6.6, are given in several sections of DCA Part 2, Tier 1. These ITAAC indicate that inspections will be performed on as-built components and piping and that reports exist that conclude the following:

- The NPM ASME Code Class 1, 2, and 3 piping systems comply with ASME Code, Section III requirements.
- The NPM ASME Code Class 1 and 2 components conform to the rules of construction of ASME Code, Section III.
- The NPM ASME Code Class CS components conform to the rules of construction of ASME Code, Section III.

The staff's evaluation of the ITAAC described in the DCA Part 2 is contained in Chapter 14 of this SER.

6.6.4.11 COL Action Items and Certification Requirements and Restrictions

Table 6.6-1 NuScale Combined License Information Items for Section 6.6

COL Item	Description	DCA Part 2, Tier 2 Section
6.6-1	A COL applicant that references the NuScale Power Plant design certification will implement an Inservice Testing Program in accordance with 10 CFR 50.55a(f).	6.6
6.6-2	A COL applicant that references the NuScale Power Plant design certification will develop a preservice inspection and Inservice Inspection Program plans in accordance with Section XI of the	6.6

	ASME Code and will establish the implementation milestones for the program. The COL applicant will identify the applicable edition of the ASME Code utilized in the program plan consistent with the requirements of 10 CFR 50.55a. The COL applicant will, if needed, address the use of a single ISI Program for multiple NPMs, including any Alternative to the Code that may be necessary to implement such an ISI Program	

The staff determines the above listing to be adequate and appropriate.

6.6.4.12 Operational Programs

The examination categories and methods specified in the DCA Part 2, Tier 2, are acceptable if they meet the requirements in ASME Code, Section XI, Articles IWC-2000 and IWD-2000, "Examination and Inspection." Every area subject to examination falling within one or more of the examination categories in Articles IWC-2000 and IWD-2000 must be examined, at least to the extent specified. The requirements of Article IWB-2000 also list the methods of examination for the components and parts of the pressure-retaining boundary.

6.6.5 Conclusions

The design of the Class 2 and 3 systems incorporate provisions for access to enable the performance of ISI examinations in accordance with 10 CFR 50.55a(g)(3) and ASME Code, Section XI. The final ISI program is required to meet the latest ASME Code, Section XI, Edition/Addenda incorporated by reference 18 months before the date scheduled for initial loading of fuel. The final ISI program will consist of a PSI and ISI plan. The periodic inspections and pressure testing of pressure-retaining components of the Class 2 and 3 systems are performed in accordance with the requirements in applicable subsections of Section XI of the ASME Code and provide reasonable assurance that evidence of structural degradation or loss of leaktight integrity occurring during service will be detected in time to permit corrective action before the safety function of a component is compromised. Compliance with the PSI and ISI program required by the ASME Code constitutes an acceptable basis for satisfying, in part, the requirements of GDC 36, 37, 39, 45, and 55 for the Class 2 or 3 systems.

The staff concludes the description of the PSI and ISI program is acceptable and meets the inspection and testing requirements of GDC 36, 37, 39, 45, and 55 for the Class 2 or 3 systems and 10 CFR 50.55a. This conclusion is based on the applicant meeting the requirements of the ASME Code, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," as reviewed by the staff and determined to be appropriate for this application.