6 ENGINEERED SAFETY FEATURES

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (hereinafter referred to as the staff) review of Chapter 6, "Engineered Safety Features," of the NuScale Power, LLC (hereinafter referred to as the applicant), Standard Design Approval Application (SDAA), Part 2, "Final Safety Analysis Report [FSAR]." The staff's regulatory findings documented in this report are based on Revision 2 of the SDAA, dated April 9, 2025 (Agencywide Document Access and Management System (ADAMS) Accession No. ML25099A251).

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

6.1 Engineered Safety Feature Materials

6.1.1 Engineered Safety Feature 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, the applicant submitted information in SDAA Part 8, "License Conditions; Inspections, Tests, Analyses, and Acceptance Citeria (ITAAC)," SDAA Part 2, Section 6.1.1, "Engineered Safety Feature Materials," and TR-123952, Revision 1 (ADAMS Accession Nos. ML24274A153 – non-prop, ML24274A154 - prop), "NuScale Containment Leakage Integrity Assurance."

The information that the applicant provided can be found in Revision 2 to the SDAA, and in letters dated August 2, 2024 (ADAMS Accession No. ML24215A000); and December 11, 2024 (ADAMS Accession No. ML24346A130).

The staff's 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

FSAR: The applicant provided a description of the ESF materials in FSAR, Chapter 6, Section 6.1.1, which is summarized in the following discussion.

As described in FSAR, Chapter 1, Section 1.9, "Conformance with Regulatory Criteria," and Section 6.1.1 of this chapter, 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, issued June 2010

- RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” Revision 4, issued October 2013

- RG 1.44, “Control of the Processing and Use of Stainless Steel,” Revision 1, issued March 2011

- RG 1.71, “Welder Qualification for Areas of Limited Accessibility,” Revision 1, issued March 2007

- RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” Revision 39, issued December 2021

SDAA Part 2, Chapter 1, Table 1.9-3, “Conformance with NUREG-0800, Standard Review Plan (SRP) and Design Specific Review Standard (DSRS),” summarizes the differences between the SDAA and NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), Section 6.1.1, “Engineered Safety Features Materials,” Revision 2, March 2007.

6.1.1.2.1 Metallic Materials

FSAR, Section 6.1.1 states 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 FSAR, Section 6.3, “Emergency Core Cooling System,” and Section 6.2, “Containment System,” respectively. The DHRS is described in FSAR, Section 5.4.3, “Decay Heat Removal System.”

FSAR, Table 6.1-2, “Material Specifications for ESF Components,” lists, with the exception of valves, the material grade and material type for the ESF pressure boundary materials; weld materials, including materials used for buttering; and associated supports. In a letter dated December 11, 2024 (ML24346A219), the applicant provided information concerning dissimilar metal welds (DMWs). FSAR, Table 6.1-4, “Pressure Retaining Materials for Reactor Coolant Pressure Boundary and Engineered Safety Feature Valves,” lists the allowable materials for the components of reactor coolant pressure boundary (RCPB) and ESF valves, which include the reactor vent valves (RVVs), reactor recirculation valves (RRVs), reactor safety valves (RSVs), DHRS actuator valves, reactor coolant system (RCS) injection and discharge line isolation valves, pressurizer spray line isolation valves, reactor pressure vessel (RPV) high point degasification isolation valves, thermal relief valves, chemical and volume control system (CVCS) reverse flow check valves, and containment isolation valves (CIVs). The CIVs are listed in FSAR, 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.

FSAR, 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). SDA Part 2, Figure 6.2-3, “Containment System Piping and Instrumentation Diagram,” shows the CNTS piping and the system classification breaks. FSAR, Figure 6.6-1, “ASME Class Boundaries for NuScale Power Module Piping Systems,” shows the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) Class boundaries. FSAR, Table 6.1-3, “Material Specifications for Containment Vessel Related non-Engineered Safety Feature 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 Safety Evaluation Report (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,” Revision 3, March 2007, 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 FSAR, Table 6.1-3 that are within the CNV are reviewed in 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 this SER. The materials of the portion of the RCS within the CNV are reviewed in Section 5.2.3 of this SER. The CIVs of these systems are part of the CNTS and are reviewed in this section of the SER.

6.1.1.2.2 Material Selection and Fabrication

FSAR, Section 6.1.1.1, “Material Selection and Fabrication,” states 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, as applicable. 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, “Material Properties.” The design, fabrication, and materials of construction of the CNV includes sufficient margin to provide reasonable assurance that the CNV pressure boundary avoids brittle fracture. The design life of the CNV is 60 years. ASME 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 CNV (Section 5.2). 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 lower CNV shell above the RPV flange elevation, the lower flange, upper flange, upper shell, and top head are fabricated from martensitic stainless steel (F6NM). The lower shell, bottom head and associated supports are fabricated from FXM-19 austenitic stainless steel forgings. The applicant stated that the FXM-19 austenitic stainless steel, which has a 60-year design life peak neutron fluence of less than 1E19 neutrons per square centimeter (n/cm^2) (E>1MeV), comprises the core region of the CNV. The applicant also stated that FXM-19 austenitic stainless steel demonstrates good resistance to neutron embrittlement when exposed to neutron fluence levels below 1E19 n/cm^2 (E> 1MeV). The region of the CNV that is fabricated from martensitic stainless steel has a peak neutron fluence of less than 1E17 n/cm^2 (E>1MeV). SDAA, Part 2, Section 6.2.1 provides additional CNV design detail.

The use of F6NM is in accordance with ASME Code Case N-774, “Use of 13Cr-4Ni (Alloy UNS S41500) Grade F6NM Forgings Weighing in Excess of 10,000 lb (4540 kg) and Otherwise

Conforming to the Requirements of SA-336/SA-336M for Class 1, 2, 3 Construction Section III, Division 1." Code Case N-774 is listed in RG 1.84, Rev. 39, "Design, Fabrication, and Material Code Case Acceptability, ASME Section III, Division 1," as permitted for use without conditions.

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 steels are subjected to sensitizing temperatures for greater than 60 minutes during postweld heat treatment, nonsensitization of the materials is verified by testing in accordance with ASTM A262, "Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels," Practice A or E. Furnace-sensitized austenitic stainless steel is not used.

The applicant stated that the delta ferrite content of austenitic stainless steel weld filler material conforms to the guidelines stipulated in ASME Code, Section III, Paragraphs NB-2433, NC-2433, or NF-2433, as well as RG 1.31. Alloy 52/152/52M filler metals are used for welding Alloy 690 to provide a high level of corrosion resistance.

Information regarding pressure-retaining bolting material is discussed in FSAR, 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 this SER.

The NuScale design does not use thermal insulation (metallic or nonmetallic) inside the CNV. Any insulation on the CNV is above the reactor pool level and uses reflective metallic insulation. There are no fibrous insulation materials. FSAR, 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 this SER.

The applicant stated that the design generally avoids the use of cold-worked austenitic stainless steel, and if it is used, then the yield strength (as determined by 0.2 percent offset method) is limited to 90 kilopounds 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 said they establish controls 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 said they will also establish controls for special processes such as welding, heat treating and non-destructive testing of the CNV and ESF materials satisfy the applicable requirements of 10 CFR Part 50, Appendix B, Criterion IX.

The applicant stated that, "[t]here are no socket welds on lines larger than 3/4-inch NPS for the Class 1 lines in Table 6.1-2. Socket welds used on piping less than 3/4-inch NPS conform to 10 CFR 50.55a(b)(1)(ii) and ASME B16.11. There are no socket welds on piping in Table 6.1-3, including piping of NPS 2 or less in size."

Licensing Technical Report (LTR)-123952, "NuScale Containment Leakage Integrity Assurance," Revision 1, dated September 30, 2024, (ML24274A153-NP, ML24274A154-P) Section 8.2, "Aging Degradation Management," assesses potential degradation mechanisms for the CNV pressure boundary materials.

FSAR, Section 6.2.1.1.1, “Design Bases,” states that the CNV is an ASME Code Class MC (steel) containment that is designed, analyzed, fabricated, inspected, tested, and stamped as a ASME Code Class 1 pressure vessel.

6.1.1.2.3 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 conditions, 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 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 post-accident coolant chemistry. FSAR, Section 5.2.3 contains additional information related to the RCS chemistry and is reviewed in Section 5.2.3 of this SER. The secondary water chemistry control program is described and reviewed in Section 10.4.6 of this SER. The UHS cleanup system is described in FSAR Section 9.1.3, “Spent Fuel Pool Cooling and Cleanup System,” and reviewed in Section 9.1.3 of this SER. Since the selected materials are compatible with the various water chemistries, none of the components is designed with a corrosion allowance.

ITAAC: The inspections, tests, analyses, and acceptance criteria (ITAAC) associated with FSAR, Section 6.1.1 are in SDAA Part 8, “License Conditions; Inspections, Tests, Analyses, and Acceptance Citeria (ITAAC) and FSAR, Chapter 14, “Initial Test Program and Inspections, Tests, Analysis, and Acceptance Criteria.” These ITAAC are evaluated in Section 14.3 of this SER.

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

Technical Reports: The staff reviewed the following NuScale Technical Reports:

- TR-123952, “NuScale Containment Leakage Integrity Assurance,” Revision 1, dated September 30, 2024, (ML24274A153-NP, ML24274A154-P)
- TR-121516-NP, “CNV Ultimate Pressure Integrity,” Revision 1, dated June 2023 (ML23304A331)

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 Criterion (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.

- GDC 35, “Emergency Core Cooling” requires that the system safety function shall be 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 amount

- GDC 41, “Containment Atmosphere Cleanup” requires that systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

- 10 CFR Part 50, Appendix B, Criteria IX, “Control of Special Processes,” requires establishing measures to assure that special processes are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

- 10 CFR Part 50, Appendix B, Criteria XIII, “Handling, Storage and Shipping,” requires establishing measures to control the handling, storage, shipping, cleaning, and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration.

The guidance in SRP Section 6.1.1, “Engineered Safety Features Materials,” Revision 2, March 2007, lists the acceptance criteria adequate to meet the above requirements as well as review interfaces with other SRP sections.

SRP Branch Technical Position (BTP) 6-1, “pH For Emergency Coolant Water for Pressurized Water Reactors,” March 2007, lists the controls on the post-accident coolant water pH and chemistry to meet the requirements of GDC 14.

6.1.1.4 Technical Evaluation

6.1.1.4.1 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, “Rules for Inservice Inspection of Nuclear Power Plant Components,” 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 ASME Code, Section II, Parts A, B, and C and ASME Code, Section III, Appendix I. 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 FSAR, Table 6.1-2, Table 6.1-3, and Table 6.1-4 and verified that the materials are acceptable for use in accordance with ASME Code, Section II, Section III and RG 1.84.

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. FSAR, Table 6.1-4 states that cast austenitic stainless steel components may be used for RCPB and ESF valves. Based on the location of these valves, the staff finds that the cast austenitic stainless steel components in the RCPB and the ESF valves are not susceptible to radiation embrittlement because of the distance from the reactor vessel.

NUREG/CR-4513, “Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems,” Revision 2 with errata, specifies a screening temperature of above 250 degrees Celsius (°C) (482 degrees Fahrenheit (°F)) for cast components. Note 4 of FSAR, Table 6.2-5 states that all CIVs, with the exception of containment evacuation (CE) valves, have a design temperature of 650 degrees F and CE valves have a design temperature of 600 degrees F. Therefore, cast RCPB and ESF valve components are subject to the screening criteria.

Note 2 of FSAR, Table 6.1-4 states that carbon is limited to 0.03 percent maximum and delta ferrite is limited to 20 percent maximum for CF3 and CF8 cast austenitic stainless steel valves. Based on the criteria set forth in NUREG/CR-4513, “Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems,” Revision 2 with errata, the staff finds that the delta ferrite limits are within the bounds to ensure that the cast austenitic stainless steel components in the RCPB and ESF valves 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, and are, therefore, acceptable.

6.1.1.4.2 Austenitic Stainless Steel

The NuScale design must meet the requirements of GDC 4, GDC 14, and the quality assurance requirements of Appendix B to 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.

Components that interact with the primary, secondary, or UHS water chemistries are either fabricated out of nickel alloys or austenitic stainless steel. The portion of the CNV fabricated from F6NM is discussed below._NuScale committed 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 unstabilized austenitic stainless steel American Iron and Steel Institute (AISI) Type 3XX and FXM19 (also known as Nitronic 50), respectively. The applicant also stated that cold working of austenitic stainless steel from abrasive work is minimized, and when abrasive work is used, ferritic carbon steel contaminants are avoided. _In addition, the applicant stated that if_cold-worked austenitic stainless steel is used, then the yield strength (as determined by 0.2 percent offset method) is limited to 90 ksi maximum. This limit on cold worked austenitic stainless steel is consistent with SRP Section 5.2.3, “Reactor Coolant Pressure Boundary Materials,” Revision 3 March 2007,and is, therefore, acceptable.

NuScale stated that it is complying with RG 1.44, which provides guidance on the control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion cracking. Notes in FSAR, Section 6.1.1, Tables 6.1-2, 6.1-3, and 6.1-4 indicate that a maximum carbon content of 0.03 percent is applied to all unstabilized Type 3XX stainless steels when they will be welded or exposed to a temperature range of 800 °F to 1500 °F subsequent to final solution anneal. This requirement is consistent with the regulatory position in RG 1.44 and is, therefore, acceptable.

The maximum carbon content of SA965 Grade FXM--19 and Type 2XX weld filler metals is restricted to 0.04 percent, in accordance with- TR-123952-NP, Revision 0. The NRC staff notes that FXM-19 is more resistant to SCC than Type 3XX austenitic stainless steels.

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.

NuScale stated that the lower portion of the CNV is solution-annealed austenitic stainless steel, SA965, Grade FXM-19. Weld filler metals E209, ER209, ER240, or ER240 (Type 2XX) will be used for welds between the SA965, Grade FXM-19 components. Weld metals E209, ER209, ER240, and ER240 are compatible with SA965, Grade FXM-19 base metal and are, therefore, acceptable. The lower portion of the CNV is located closer to the reactor core compared to a traditional light-water reactor (LWR) containment. Because of the proximity to the reactor core, NuScale considered the impact of radiation embrittlement on the CNV. NuScale used the same methodology to calculate the neutron fluence on the CNV as the RPV. NuScale stated that the peak neutron fluence in the lower CNV is less than 1E19 n/cm^2 (E > 1.0 MeV). The staff reviewed NuScale’s methodology. The staff finds the use of SA965, 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 SA965, Grade FXM-19, an austenitic stainless steel, is resistant to radiation embrittlement. Additional detail related to the fracture prevention of the CNV and the impact of radiation embrittlement on the CNV is located in FSAR, Section 6.2.7, "Fracture Prevention of Containment Vessel," and reviewed in Section 6.2.7 of this SER.

6.1.1.4.3 F6NM Martensitic Stainless Steel

The NuScale design must meet the requirements of GDC 1; 10 CFR Part 50, Appendix B, Criterion IX; and 10 CFR 50.55a. Designers may meet these requirements by meeting ASME Code, Section III, Division 1 Subsection NB and Non-Mandatory Appendix D, Article D-1000. In addition, moisture control on low hydrogen welding materials must conform to the requirements of ASME Code, Section III. A designer may comply with RG 1.71 for welding in areas of limited accessibility to meet the requirements. RG 1.50, *Control of Preheat Temperature for Welding of Low-Alloy Steel*, Revision 1, March 2011applies to P Nos. 3, 4, and 5 materials, however, F6NM is classified as a P Number 6, Grade 4 material. Therefore, RG 1.50 does not apply to the NuScale CNV. F6NM is not included in RG 1.50 because F6NM has not been used for pressure vessel fabrication in previous light water reactor designs. However, preheat must be carefully controlled to successfully weld F6NM and avoid excessive repairs.

The applicant stated that stated that "The qualification of welders for making welds in areas with limited access, and the methods for monitoring and certifying such welds, are in accordance with RG 1.71." Supplemental performance qualification requirements for welding caused by restricted physical and visual access provides additional assurance that welds that have restricted access will be of sufficient quality and weld repairs will be minimized. The NRC staff notes that excessive weld repairs in operating US nuclear plants have been the source of degradation initiation, sometimes decades after a plant begins operation. The NRC staff notes that the applicant will apply RG 1.71 to all ESF materials welds, where applicable.

F6NM is considered to have better weldability than other grades of martensitic stainless steels due to its lower carbon content. For ASME Code purposes, F6NM is considered a ferritic steel. F6NM (UNS S41500) has good notch toughness and is used in several industries where harsh environments necessitate the need for a material with good corrosion resistance and high strength. Large F6NM forgings have been used in German-designed reactor coolant pumps. F6NM has also been used in German-designed CRDM latch housings, which is also the basis for the French EPR CRDM latch housing design. The NRC staff is unaware of any degradation issues in any foreign nuclear plant that uses or has used F6NM. Preheat, welding and post weld heat treatment of F6NM is different from typical vessel materials used in nuclear power plants such as low alloy steels (LAS) SA-533 and SA-508 (P No.-3 materials). Welding F6NM necessitates the need for special considerations in addition to the requirements listed in ASME Code, Section III and ASME Code, Section IX due to its susceptibility to hydrogen-induced cracking (HIC) and potential issues related to Charpy impact values and mils lateral expansion values for E410NiMo and ER410NiMo weld metal that utilize fluxed welding processes.

ASME Code, Section III, Non-Mandatory Appendix D, Article D-1000 recommends a minimum preheat temperature of 400°F for P Number 6 Grade 4 materials, such as F6NM. The NRC staff notes that ASME Code, Section III, Division 1, Subsection NB-4611 states that, "[i]t is cautioned that the preheating suggested in Section III Appendices, Nonmandatory Appendix D does not necessarily ensure satisfactory completion of the welded joint and that the preheating requirements for individual materials within the P-Number may be more or less restrictive."

A high minimum preheat temperature, such as 400°F, can be disadvantageous when welding F6NM due to the relatively low martensite start (M_s) and martensite finish (M_f) temperatures of F6NM base material and 410NiMo type weld metal. It is advantageous to use as low of a preheat temperature as possible to promote the formation of martensite. However, sufficient preheat to mitigate the potential for HIC in necessary. In a letter dated December 11, 2024 ML24346A214) the applicant stated that, "[f]or F6NM welds, in lieu of the ASME Nonmandatory Appendix D minimum preheat guidance, the optimal minimum preheat temperature is qualified through the procedure qualification process which includes ASME Code Section IX and Section III procedure qualification requirement, as well as the additional requirements specified in the design specification for the prevention of hydrogen assisted cracking. The preheat temperature is selected to prevent hydrogen cracking and promote the transformation of martensite in the heat affected zone of the F6NM base material." The NRC staff finds that the applicant's approach is acceptable because it will take appropriate steps to prevent HIC, while at the same time promote the formation of martensite during welding.

SDAA Section 6.1.1, Table 6.1-2 specifies the use E410NiMo and ER410NiMo weld metal. The lower critical (Ac1) temperature for 410NiMo type weld metals and F6NM base material can be as low as 1150°F or slightly lower. ASME Code, Section III, Table NB-4622.1-1 specifies that P-No. 6, Grade 4 materials (F6NM) receive a mandatory Post Weld Heat Treatment (PWHT) of welds at a temperature of 1,050°F-1,150° F. The staff notes that somewhat recent nuclear industry experience has shown that PWHT temperatures, when using locally applied heaters, can produce varying PWHT temperatures outside of their intended values. Heating above the Ac1 temperature during PWHT can result in the formation of fresh martensite in lieu of the preferred final microstructure of tempered martensite. In a letter dated August 2, 2024, (ML24215A139) the applicant modified SDAA Section 6.1.1.1 to state, "[p]ost weld heat treatment of SA-336 Gr F6NM for the CNV and supports shall be 1075°F +/- 25°F." The NRC staff finds that the applicant has provided sufficient margin in its maximum specified PWHT temperature to provide reasonable assurance that it will not exceed the Ac1 temperature during PWHT, and is, therefore, acceptable.

When qualifying weld procedures for the welding of vessels, it is typical for fabricators to qualify the procedure for long PWHT times, 20 hours or more, that will encompass the PWHT time needed during initial fabrication and any subsequent repairs required during the fabrication process. Mandatory minimum PWHT times are listed in ASME Code, Section III, Table NB-4622.1-1. For typical vessel materials used in light water reactors, such as P-No 1 and P-No. 3 materials, this approach has proven adequate. However, minimum PWHT times listed in Table NB-4622.1-1 may not be adequate for some welding processes that use flux when welding F6NM with 410NiMo type weld metal. Information provided by the applicant as well as data acquired by the NRC independently show that while weld metal used in welds made with non-fluxed welding processes generally exhibit sufficient Charpy impact values as well as mils lateral expansion values, some welds made with fluxed processes exhibited poor or marginal values before PWHT and after PWHT at times significantly longer than the minimum times listed in Table NB-4622.1-1. In a letter dated December 11, 2024 (ML24346A214), the applicant stated that, "[for flux shielded welding processes involving 410NiMo martensitic stainless steel, an additional group of test specimens receive a simulated post-weld heat treatment at, or below, the minimum hold time prescribed in ASME Code, Section III, Table NB-4622.1-1 for the nominal thickness of the joint to be welded, assuring the minimum post-weld heat treatment applied during production welding results in acceptable material properties." During the audit (ML24211A0890), the NRC staff noted that in addition to the above statement, the applicant modified EQ-146988, Revision 2, ASME Design Specification for Containment Vessel, Section 5.4.2.6 to require that an additional group of weld metal test specimens receive a simulated post-weld heat treatment at, or below, the minimum hold time prescribed in ASME Code, Section III, Table NB-4622.1-1 for the nominal thickness of the joint to be welded. Based on the above, the NRC staff finds that the applicant has appropriately addressed simulated PWHT of weld metal that utilize a fluxed welding process to ensure that production welds will exhibit acceptable Charpy impact and mils lateral expansion properties.

There are several dissimilar metal welds (DMWs) in the CNV. The largest DMW in the containment is the weld that joins the CNV lower shell (F6NM) and the CNV lower transition shell (FXM-19 also known as Nitronic 50). During the audit (ML24211A089), the NRC staff reviewed NuScale document ER-149435, Revision 0, "Dissimilar Welding of FXM-19 to F6NM." The applicant has selected E/ER/EC240 or E/ER/EC209 to perform the aforementioned DMW due to its matching strength to FXM-19. These two weld metals are compatible with FXM-19 and F6NM. When determining the acceptability of the use of these two weld metals, the applicant considered solidification cracking, liquation cracking, cold cracking in the F6NM base material, intergranular stress corrosion cracking, and high temperature embrittlement susceptibility potentially caused by PWHT of the weld buttering on the F6NM lower shell. The NRC finds that the applicant has appropriately considered the potential for fabrication flaws and inservice degradation of the above F6NM to FXM-19 DMW.

There are several stainless steel-to-F6NM safe end DMW welds on the CNV head. The weld materials selected are listed in Table 6.1-2. These weld metals are compatible with both base materials and have extensive use in currently operating nuclear power plants. Therefore, the NRC staff finds that the materials, including weld materials for the remainder of the CNV DMWs, are acceptable.

The NRC staff notes that the applicant is still in the process of qualifying welding procedures. The applicant has done extensive testing with various weld processes. In addition, the applicant has investigated the Ac1, upper critical (Ac3), M_s and M_f temperatures by calculation and by experimentation on F6NM base material and 410NiMo weld metal samples. During the audit (ML24211A0890), the NRC staff has reviewed documentation made accessible to the staff by the applicant related to its welding procedure qualification process. The NRC staff has determined that the applicant considering the special requirements beyond the requirements in ASME Code, Section III that are necessary to successfully weld F6NM.

The staff finds the applicant's compliance with RG 1.71 acceptable, as it ensures the quality of welds with limited accessibility. Furthermore, the licensee will follow the requirements in ASME Code, which requires, in part, that the applicant take the appropriate steps to prevent the uptake of moisture in welding electrodes and apply a minimum preheat temperature to prevent HIC. The staff finds that these actions, along with the steps that the licensee will take, as described in the paragraphs above, will meet the requirements of GDC 1, 10 CFR Part 50, Appendix B, Criterion IX, and 10 CFR 50.55a.

6.1.1.4.4 Composition and Compatibility of Engineered Safety Feature Fluids

To meet the requirements of GDC 4, GDC 14, GDC 35, and GDC 41, SSCs important to safety must 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 the 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 Sections 5.2.3, 10.4.6, and 9.1.3 of this SER, respectively. NuScale did not incorporate a corrosion allowance because the materials of construction are not susceptible to any appreciable corrosion, given the operating environment Since NuScale 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 to be acceptable.

6.1.1.4.5 Component and System Cleaning

To meet the requirements of 10 CFR Part 50, Appendix B, Criteria IX, and XIII, the applicant must 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 deterioration.

The applicant stated that controls will be established to meet 10 CFR Part 50, Appendix B, Criterion XIII, as well as the applicable provisions 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 ASME 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.

6.1.1.4.6 Thermal Insulation

To meet the requirements of GDC 1, 14, and 31, ESF systems must 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.

6.1.1.4.7 Branch Technical Position 6-1

The staff's guidance includes BTP 6-1, which describes the minimum pH range for the post-LOCA recirculating fluid. BTP 6-1 recommends a minimum fluid pH of 7.0 to prevent 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 PWR containments have a pH buffer chemical that dissolves in the post-LOCA fluid and maintains the pH above 7. Another basis for pH7 is iodine retention in the post-LOCA fluid, but this is addressed in SRP Section 6.5.2, “Containment Spray as a Fission Product Cleanup System,” and Section 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors,” 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. FSAR, 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 FSAR, Section 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors,” 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 61 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, Sections 6.1.1 and 6.1.2.

6.1.1.4.8 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, Section 6.1.1.

6.1.1.6 Conclusion

Based on its review of the information provided by NuScale, the staff concludes that the NuScale SDAA 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; GDC 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 FSAR Section 6.1.2, according to SRP Section 6.1.2, and there are no conclusions for this section of the application

DCA Part 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. DCA Part 2 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
- free of organic material in both the insulation and jacketing

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

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

6.2 <u>Containment Systems</u>

6.2.1 Introduction

This section describes the NRC staff’s review for conformance to RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” Revision 4, issued March 2012, regarding component performance in the ECCS flow path 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

6.2.1.1.1 Introduction

The primary functions of the reactor containment building are to protect the safety related SSCs located within it and to establish an essentially leaktight 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.

6.2.1.1.2 Summary of Application

FSAR: The information associated with this evaluation is in FSAR, Section 6.2.1.1, “Containment Structure.” 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 the passive ultimate heat sink (UHS) 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 lowalloy 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 Boiler and Pressure Vessel Code (BPVC) Class MC (metal containment) and is rated as an ASME BPVC 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 of the pressures and temperatures associated with combustible gas deflagration described in FSAR 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 and cooled in the CNV, passively returns to the reactor vessel by natural circulation through the ECCS described in Section 6.3 of the DCA. The methodology addresses LOCAs, valve-opening non-LOCA DBEs, and secondary pipe breaks; potential single failures have been considered in the methodology. The supporting M&E release analyses for the primary and secondary systems are presented in FSAR Section 6.2.1.3, “Mass and Energy Release Analyses for Primary System Release Events,” and Section 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 presented in FSAR, Table 6.2-3, “Containment Response Analysis Results.” The peak containment pressure as well as the peak containment temperature occur as a result of the LOCA caused by the RCS (CVCS) discharge line double guillotine break from the downcomer, with the loss of normal AC power and no single failure. The peak pressure and CNV wall temperature results for the secondary system main steamline and feedwater line break events are bounded by the limiting CNV DBE 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 FSAR, Section 6.2.2, “Containment Heat Removal.”

ITAAC: The ITAAC associated with FSAR, Section 6.2.1.1, are provided in SDAA Part 8, Rev. 2, Table 2.1-1, “NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC 02.01.xx),” and Table 2.1-2, “NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria Additional Information.” These ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: The related US460 Generic Technical Specifications (TS) associated with FSAR, Section 6.2.1.1, are provided in SDAA Part 4, Revision 1, Volume 1, Section 3.6.1, “Containment,” and Section 3.5.3, “Ultimate Heat Sink.” The corresponding sections in SDAA Part 4, Revision 1, Volume 2 provide the TS bases.

Topical Reports:

NuScale TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P), “Loss-of-Coolant Accident Evaluation Model [LOCA EM]”

NuScale TR-0516-49416-A, Revision 5 (ML25136A339 NP, ML25136A340 P), “Non-Loss-of-Coolant Accident Evaluation Model [non-LOCA EM]”

The unique nature of the NuScale Power Module (NPM-20) containment design and heat removal systems necessitates development of a specific containment response analysis methodology (CRAM). The NPM-20 containment design is presented in LOCA EM TR, Revision 3 for the NuScale’s 462-MWe US460 Plant, where each of the six NPM-20 modules are rated as 77 MWe and 250 MWth. TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P), "Loss-of-Coolant Accident Evaluation Model," describes the supporting containment thermal hydraulic analysis methodology for primary and secondary system M&E releases into the CNV of the NPM-20 as well as the resulting pressure and temperature response of the CNV to support the FSAR Chapter 6 analyses and meets the applicable regulatory guidance, including “Design Specific Review Standard for NuScale SMR Design,” Section 6.2.1, “Containment Functional Design,” of the design-specific review standard (DSRS) for the NuScale design, issued in June 2015 (ADAMS Accession No. ML15118A922). The CRAM, as presented in LOCA EM TR, Revision 3, is used to determine CNV peak pressure and peak temperature. A spectrum of design basis M&E release events is analyzed to bound all of the LOCA events and valve opening transients in the primary and secondary systems pipe break accidents. The CRAM uses conservative initial conditions and boundary conditions. The spectrum analyses results are presented in EC A013 7725, “NPM-20 CNV Pressure and Temperature Response Analysis,” Revision 0 (ADAMS Accession No. ML23011A012), to show the calculation of the CNV internal design pressure and temperature. Limiting analysis results are summarized in FSAR Section 6.2.1.1.3. FSAR Table 6.2-3 presents the results of the base case and limiting CNV pressure and wall temperature analyses for primary release (LOCA and valve opening events), and limiting secondary system break scenarios. The table shows that the limiting peak containment pressures and temperatures are shown to be less than the NPM-20 CNV design pressure (8274 kilopascals (kPa) (absolute) (1200 pounds-force per square inch, absolute (psia))) and the design temperature (315.6 °C (600 °F)). The qualification of the LOCA and non-LOCA methodologies are presented in the respective topical reports LOCA EM TR, Revision 3,and non-LOCA EM TR, Revision 4, which detail 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 the CRAM described in LOCA TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P). 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 the CRAM part of the LOCA EM TR.

6.2.1.1.3 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 LOCA

- 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 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 16, “Containment Design,” 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 38, “Containment Heat Removal,” 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 50, “Containment Design Basis,” 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 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(f)(3)(v)(A)(1), 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 guidance in NuScale DSRS 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.

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 Section 8.1 of this SER.

6.2.1.1.4 Technical Evaluation

NuScale needed to demonstrate that the NPM-20 CRAM 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 sections incorporate the graded staff review narrative applicable to this area of review.

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

To satisfy the requirements of GDC 16 and 50 on sufficient design margin, the applicable NuScale DSRS guidance 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. FSAR, Section 3.1.4.9, “Criterion 38-Containment Heat Removal,” 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 to 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, GDC 50, and PDC 38 relevant to the containment design basis and guided by 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 was a major part of the staff review for the DCA.

6.2.1.1.4.1.1 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 the 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. The containment heat removal function is described in Section 6.2.2 of this SER.

The CNV evaluation model divides the NPM-20 M&E release into two phases; (1) the blowdown phase that starts at the break initiation or valve opening, and (2) ECCS actuation that occurs as a result of a module protection system (MPS) signal or a loss of augmented direct current power system (EDAS). NPM-20 has two RRVs and two RVVs. The RVVs do not have an inadvertent actuation block (IAB), and open immediately upon the loss of EDAS power to their actuator solenoid valves. Opening of the RVVs releases the reactor coolant into the containment volume, which pressurizes the CNV and depressurizes the RPV that results in reactor trip and closure of the containment isolation valves (CIVs).

The RRVs are equipped with the IAB that prevents their spurious opening at full operating pressure and, thus, leads to a delayed opening. The IAB prevents the RRVs from opening when the differential pressure between the RPV and CNV is greater than the IAB threshold pressure setpoint (900 psid) and allows them to open only after the differential pressure between the RPV and CNV decreases below the IAB release pressure setpoint (400 to 500 psid). For the RRVs, after loss of DC power, the IAB arming valves close because the valve differential pressure is greater than the threshold setpoint, thereby preventing the immediate opening of RRVs. As the RPV pressure decreases and the CNV pressure increases, the RRVs open as soon as the differential pressure drops below the IAB release pressure setpoint. RRVs open under the following conditions.

a) If the pressure differential across the RRVs is greater than the IAB threshold (900 psid) when the ECCS signal actuates, then the RRVs stay closed until the pressure differential decreases to below the IAB release pressure (400-500 psid).
b) If the pressure differential across the RRVs decreases to below the IAB threshold pressure when the ECCS signal actuates, then the RRVs open at that time.
c) If the pressure differential across the valves is less than the valve opening spring force (approximately 15 psid), then the valves open even without an ECCS actuation signal.

As the pressures equalize, the break and valve flow decreases. After the RPV and CNV pressures equalize, further increase in the CNV pool level starts the liquid RCS flow from the CNV into the reactor vessel through the RRVs to provide long-term cooling (LTC) via recirculation. Pressure equalization terminates the reactor vessel level decrease before core uncovery. Heat transfer to the CNV wall and to the reactor pool eventually exceeds the energy addition from the break flow and the RVV flow. When this occurs, it completes the period of peak containment pressure and temperature, and a gradual depressurization and cooling phase begins.

6.2.1.1.4.1.2 Break Spectrum and Single Failures

NPM-20 is an advanced, light-water, integral pressurized water reactor (PWR) that uses a compact, high-pressure steel CNV partially immersed in a reactor pool coupled with passive safety-related systems. The secondary system includes a traditional steam-power conversion system including a steam turbine generator, condenser, and feedwater system. Piping in an NPM containment that can potentially break is limited to the RCS injection line, RCS discharge line, pressurizer spray supply line, and pressurizer high point vent line. The RCS injection line is supplied by the chemical and volume control system (CVCS) and the discharge line returns to the CVCS. Pipe breaks inside the CNV are evaluated as LOCA, while the ones outside the CNV are not defined as LOCA. Inadvertent opening of RVVs and RRVs leading to a decrease in RCS inventory inside the RPV are not included in the LOCA definition. However, the staff found it appropriate that the LOCA EM has been extended to model inadvertent RRV and RVV opening transients, as the thermal-hydraulic phenomena and the Phenomena Identification and Ranking Table (PIRT) remain the same for LOCA and non-LOCA events.

The spectrum of postulated M&E release events analyzed in EC-A0137725 (ML23011A012) for the NPM-20 using the CRAM- is the same break spectrum as considered by the LOCA TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P). All the input initial and boundary conditions, and the tabulated and graphical results for all DBE spectra analyzed for AC and EDAS power availability and ECCS actuation sensitivities documented in both documents are identical. While the LOCA TR content is only valid for the CRAM demonstration, the results documented in FSAR Chapter 6 as supported by EC-A0137725 constitute the licensing basis.

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 AC and EDAS power availability, ECCS actuation level, initial and boundary conditions, and single failures of RVV, RRV, or RRV-RVV train. With respect to the assumptions for power availability and initial and boundary conditions, the staff found they were determined consistent with DSRS Section 6.2.1.1.A and therefore appropriate. With respect to single failures, the staff's review is documented in Section 15.0.0.5 of this SER.

The primary side events analyzed by NuScale for the NPM-20 design include the CVCS discharge and injection line breaks, pressurizer high point vent (HPV) line break, inadvertent RRV and RVV opening, and inadvertent actuation of ECCS signal. M&E release into the CNV through the break or opening pressurizes the CNV, while RPV depressurizes. Depending on the DBE, the high containment pressure signal or loss of EDAS power causes reactor trip, containment isolation, and DHRS actuation. Reactor trip, containment isolation, and DHRS actuation occur immediately when all electric power is assumed lost. As the transient progresses, coolant is continuously lost to the CNV through the break or the valve opening until all ECCS valves open. When EDAS power is available, the ECCS valve opening occurs once an ECCS actuation signal is generated. However, for the loss of both normal AC and EDAS power, the ECCS valve opening occurs immediately following the loss of power. While the RVVs open once an ECCS actuation signal is generated, IABs prevent the RRVs from opening until the IAB release pressure is reached. After the ECCS valves open, coolant lost to CNV is free to flow back into the RPV downcomer through the RRVs. This way, a two-phase natural circulation cooling loop is established and the module transitions to long-term cooling. Inadvertent opening of RVVs and RRVs, leading to a decrease in RCS inventory inside the RPV, are not included in the LOCA definition. However, the LOCA EM has been extended to model inadvertent RRV and RVV opening transients. The staff found it appropriate that the LOCA EM has been extended to model inadvertent RRV and RVV opening transients, as the PIRT remains the same for LOCA and non-LOCA events.

For postulated secondary system pipe ruptures, limiting single failure are also considered for main steam isolation valve (MSIV) or feedwater isolation valve (FWIV) failure, in addition to the RVV, RRV, and RRV-RVV train failures considered for the primary system release events. For the NuScale design, full reactor 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 results for both primary and secondary release events. The maximum break opening area is modeled such that M&E releases to containment are maximized. The RVV nozzles and reactor recirculation valve nozzles are not considered in the spectrum of possible break locations. As part of the LOCA break location and size evaluation, the staff has evaluated NuScale's justification that the bolted ECCS valve-to-vessel connection provides confidence that the probability of gross rupture is extremely low such that it can be treated as a break exclusion area. The staff's evaluation of this issue is in Section 3.6.2 of this document, and the relevant detailed evaluation of the LOCA analyses is provided in Section 15.6.5 of this document.

EC-A013-7725 (NPM-20 CNV Pressure and Temperature Response Analysis) has presented the detailed descriptions and assumptions for the containment pressure and temperature results summarized in FSAR Table 6.2-3. This includes a total of eight design basis M&E release scenarios that include the following six primary system, and two secondary system (MSLB/FWLB) M&E release initiating events, as follows.

1. LOCA caused by RCS (CVCS) discharge line break from the downcomer (limiting CNV event) (DL)
2. LOCA caused by RCS (CVCS) charging or injection line break from riser (limiting LOCA event) (CL)
3. LOCA caused by RPV high point vent line break near the top of the vessel (HPV)
4. AOO due to inadvertent opening of an RVV caused by a mechanical failure (RVV)
5. AOO due to inadvertent opening of an RRV caused by a mechanical failure (RRV)
6. AOO due to inadvertent opening of both RVVs caused by an inadvertent ECCS actuation signal (ECC)
7. Main Steamline Break (MSLB)
8. Feedwater Line Break (FWLB)

Each of the above six primary containment M&E release scenario was analyzed for a spectrum of seven DBEs. The staff found the formulation of the seven DBEs for each spectrum to be appropriate for meeting the regulatory guidance for the sensitivity of the peak containment pressure and temperature to loss of AC power, loss of EDAS, ECCS actuation level, and single failure (RVV, RRV, or RRV-RVV train). Both secondary containment M&E release scenarios (MSLB and FWLB) were analyzed for a spectrum of nine DBEs each that also covered the additional single failures for MSIV and FWIV. LOCA LTR, Rev. 3, Table 9-9 summarized the initial conditions for the M&E release analysis as a part of the CRAM methodology. The staff determined that the initial and boundary conditions and assumptions for the M&E release analysis are chosen to maximize peak CNV pressure and temperature. By a letter dated

August 2, 2024 (ML24215A142), the applicant added these parameters in the FSAR Chapter 6, as a new Table 6.2-2, "Containment Pressure, Temperature Response Analysis Initial Conditions."

6.2.1.1.4.1.3 NPM-20 Containment Vessel and Reactor Pool Models and Nodalization Studies

LOCA EM LTR, Revision 4, Section 9.6.1 presents the impact of three CNV-reactor pressure vessel (RPV) nodalization schemes, summarized in Table 9-7, on the NPM-20 LOCA figures of merit (FOMs) for the reactor coolant system (RCS) injection line and high point vent line breaks, with no DHRS operation credited. No nodalization sensitivity results are presented for the limiting containment DBA, i.e., RCS discharge line (DL) break. Even though Section 9.7.1 recognizes analyzing the effect of coarse versus fine nodalization for both CNV and reactor pool as a part of the CRAM, no reactor pool nodalization sensitivity results are reported in LOCA EM LTR, Revision 4 for NPM-20. Figures 5-1, 5-2, and Table 9-7 describe the nodal volumes for the RPV and CNV, but do not include any reactor pool nodalization.

New FSAR Table 6.2-2 (ML24215A142) lists 500 °F as the initial CNV wall temperature for the entire CNV wall above the pool water level. The staff conducted an audit of the CNV wall inner surface initial temperature distribution that was calculated by NuScale using a three-dimensional finite element model (FEM) and determined it duly accounted for the radiation heat transfer. The 3-D FEM analysis results showed the maximum CNV head temperature to be around 500 °F under the initial pool and ambient air conditions specified for the DBA. The staff audit showed that the actual axial CNV wall temperatures initialized in the NRELAP5 model above the pool level are typically 500 °F. The audit also showed the CNV inner wall temperatures below the pool level to be slightly higher than 140°F that is the initial pool temperature documented in the new FSAR Table 6.2-2 (ML24215A142). The staff reviewed the various assumptions for developing the initial CNV wall temperature distribution used in NRELAP5 and found them to be conservative.

The staff looked into the impact of {{ }} in the upper part of the NPM-20 containment above the RPV (LOCA EM TR, Revision 3, Figure 5-1 vs. Figure 5-2) that would have maximum thermal stratification and temperatures during the DBE transient due to the lowering of pool level from 65 feet in the DCA to 52 feet in SDAA and assuming an adiabatic boundary condition for the CNV heat structure outer surface above the pool water level. The staff observed that for the NPM-160 design, the peak containment pressure was not sensitive to containment nodalization. In addition, the total number of CNV nodes between NPM-160 and NPM-20 is kept the same, and the initial wall temperature is kept uniform for all the containment nodes above the pool water level for both NPM designs. This addresses the staff's safety significance concerns about the reduction in the number of nodes in the upper containment region above the RPV.

By letter dated January 9, 2025 (ML25014A012), NuScale provided additional LOCA spectrum analysis results going down from 100% LBLOCA to 2% SBLOCA regime for the US460 containment and pool nodalization sensitivity. The additional information addressed the staff concerns about the heat removal performance degradation of the DHRS and the containment due to the reactor cooling pool heat up and thermal stratification, and its impact on the containment pressurization accompanied by delayed ECCS actuation, especially toward the smaller break end of the spectrum. The results demonstrated that that the peak containment pressure and temperature are not very sensitive to the DHRS performance degradation caused by the pool heat-up, and the largest breaks remain bounding for the smaller breaks with respect to the peak containment pressure and temperature. The DHRS can perform its safety function to remove the decay heat even if the reactor pool heats up and becomes thermally stratified during a small LOCA event progression. Even for the conservative sensitivity study of assuming an unrealistically high initial pool temperature around DHRS, the peak containment pressure and temperature are still maintained within the design limits. The sensitivity cases demonstrated that the natural convection modeling approach is sufficient to account for the DHRS and containment performance during LOCA events, considering the effects of reactor pool heat up and thermal stratification. NuScale also provided justifications for modeling the pool heat-up around the containment, thermal stratification, and the natural convection heat transfer. The staff concludes that NuScale has addressed all of the staff concerns in this regard.

6.2.1.1.4.1.4 Decay Heat Removal System (DHRS) Availability in NPM-20 Containment Events

The decay heat removal system (DHRS) is a passive safety-related system that relies on a closed-loop, two-phase natural circulation to remove heat from the RCS through the steam generator (SG) and rejects it to the reactor pool through the DHRS condenser. It provides additional capacity to remove decay heat for both primary and secondary system release and is more effective during the initial blowdown period before the ECCS actuation. DHRS was not credited to the containment DBEs for NPM-160, but it is credited to NPM-20 containment safety analyses for both primary and secondary systems M&E release events.

In the NPM-20 design, one DHRS train is available for each of the two steam generators (SGs), capable of independently removing 100 percent of the decay heat load to cool the reactor's primary side RCS inventory. With the individual passive DHRS condenser submerged in the reactor pool, the DHRS piping connects to the main steam and feedwater lines specific to the associated SG. During normal operation, the DHRS condensers are maintained with sufficient water inventory but are isolated by valves on the steam side of the SG, and do not remove heat. After an event initiates, the M&E release into the CNV through the break or opening causes CNV pressurization and RPV depressurization. Depending upon the evaluated scenario, the high containment pressure signal or loss of EDAS power causes reactor trip, containment isolation, and DHRS actuation. Upon the actuation of DHRS, the feedwater/steam lines isolation valves close and the DHRS actuation valves open, creating a closed loop between the SG and DHRS condenser, which establishes long-term decay heat removal using the unaffected SG and the associated DHRS. In the event of a postulated MSLB or FWLB inside the CNV, the M&E release into the CNV also includes the RCS inventory present in the DHRS train associated with the SG loop with the break.

With DHRS operation, the primary system begins a gradual cooldown and depressurization. Basically, the DHRS provides secondary side reactor cooling when normal feedwater is not available. A staff audit of the submitted NRELAP5 containment decks showed appropriately modeled passive safety-related DHRS and ECCS systems. Both DHRS trains are included in the NRELAP5 model. The model considered the water in the affected/unaffected helical coil SGs and feedwater lines, feedwater transfer to the affected/unaffected helical coil SG before the closure of the isolation valves in the feedwater lines and upon flooding with the DHRS heat exchanger inventory in the affected loop, and steam in the helical coil SG. Because the DHRS is a closed system, the total water mass in the unaffected SG loop remains constant in NRELAP5 calculations for the DHRS system operation.

Starting from the base case, the staff conducted some sensitivity studies using the submitted NRELAP5 decks for NPM-20 for the limiting containment DBE and M&E release. The study reproduced the graphical results as well as the peak CNV pressure and temperature for the limiting 100 percent RCS discharge line break LOCA. The staff performed a sensitivity study of the limiting 100 percent RCS discharge line break LOCA as well as a 5 percent RCS discharge line break SBLOCA to study the impact of DHRS performance degradation by employing a fouling factor on the CNV pressure and temperature transients. The study concluded that a deterioration in the DHRS performance does lead to a rise in peak CNV pressure and temperature. As discussed above, by letter dated January 9, 2025 (ML25014A012), NuScale provided additional information that demonstrated that the largest breaks remain bounding for the smaller breaks with respect to the peak containment pressure and temperature.

6.2.1.1.4.1.5 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 TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P), 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 in TR-0516-49416-A, Revision 5 (ML25136A339 NP, ML25136A340 P). 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 evaluations are performed using TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P), which the staff has reviewed and found acceptable, subject to the conditions and limitations listed in the staff's safety evaluation report (ADAMS Accession No. ML24312A004).

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

DSRS Section 6.2.1.1.A, paragraph III.4 requires the staff to review assumptions used in the containment response analysis to determine if the analyses are acceptably conservative. NPM-20 CRAM has been described and demonstrated in LOCA EM TR, Revision 3. Subject to the conditions and limitations listed in the staff's safety evaluation report for the LOCA EM TR, the application of the CRAM to NPM-20 design is acceptable.

In the CRAM, 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. {{

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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.

As shown by FSAR Table 6.2-3, the applicant determined that both the overall limiting peak calculated CNV pressure and the maximum CNV wall temperature result from the RCS discharge line break event with loss of normal alternating current (ac) was 6336 kPa (absolute) (919 psia). With the CNV design pressure of 8274 kPa (absolute) (1200 psia), the SDAA containment design has an approximately 23.4 percent pressure safety margin. The same RCS discharge line LOCA DBA led to the overall limiting maximum CNV wall temperature of 277.8 °C (532 °F) that has sufficient margin with respect to the NPM-20 containment design temperature of 315.6 °C (600 °F). {{

}}. Also, a loss of both normal AC and EDAS power is less limiting than all power available with respect to peak CNV pressure and wall temperature, since it results in an immediate ECCS actuation at which point no significant liquid level has accumulated within the CNV to occupy CNV free volume prior to ECCS valve opening. {{

}}. The staff found these observations to be consistent with the graphical results {{ }}.

New FSAR Table 6.2.2, "Containment Pressure, Temperature Response Analysis Initial Conditions" (ML24215A142) documents 3.0 psia as the initial CNV atmosphere pressure, and 52 ft as the initial reactor pool level above the pool floor. {{

}} is consistent with the containment response methodology described in the LOCA EM TR, Revision 3.

6.2.1.1.4.1.7 NuScale NPM-20 Containment ITAAC

The staff review established a need for ITAAC to verify the as-built plant configuration to provide reasonable assurance that, if ITAAC are performed and the acceptance criteria are met, the as-built plant will operate in accordance with the design under the NRC regulations. Section C.II.1.2.11 of Regulatory Guide 1.206 guides the applicant to develop ITAAC to verify key input parameters used in the containment safety analyses for the design, such as about containment heat removal during LOCA, main steam line break (MSLB), and main feedline break. SRP Section 14.3, "ITAAC," emphasizes the need to ensure that the key parameters and assumptions from the safety analyses for the design are verified by ITAAC. SRP Section 14.3 includes containment analyses among the fundamental analyses whose integrity needs to be preserved in the as-built plant referencing the design. SRP Section 14.3 also exemplifies "importance to be maintained for the life of the facility" as a key parameter attribute, and not meeting the numeric value commitment specified in the ITAAC acceptance criteria as an indication of failure to properly implement the design. NuScale SDAA, Part 8, Rev. 1 presents the ITAAC developed for the NuScale US460 Plant. The staff review determined a lack of verification of some key physical parameters of the as-built containment structure relied upon in the FSAR Section 6.2.1 safety analyses, through ITAAC, as discussed below.

Containment Free Volume ITAAC

Containment free volume is a key input parameter to be verified by ITAAC because it meets the various "key parameter" attributes underscored in several 14.3 SRP Sections cited in the above. It is used in the design basis transient and accident analyses and its validation through an ITAAC provides an overall indication that the containment has been built as designed. The containment design safety margins are not particular criteria for "key parameters," and are separately regulated by General Design Criteria (GDC) 16, 50, and 38 in Appendix A to CFR Part 50. Relative to the large LWR designs, the smaller NuScale containment volume would be more sensitive to construction and future changes, further justifying the need for a containment free volume ITAAC. By letter dated January 22, 2025 (ML25022A197), NuScale included a one-time ITAAC to verify the as-built CNV free volume for the first module ever built. For the first NPM-20, the applicant will show that the as-built CNV free volume conservatively bounds the minimum value of 6000 cubic feet documented in New FSAR Table 6.2.2 (ML24215A142) and used in the Chapter 6 containment design-basis analyses. NuScale has revised SDAA Part 8 Tables 2.1-1 and 2.1-2 to include the one-time ITAAC 02.01.23 to verify the CNV as-built free volume**.** With NPM-20 being a standard design module and free volume being a passive feature not subject to significant changes from module to module, the staff finds this response acceptable. The staff accepts that this ITAAC will not be performed for any subsequent adopters of the SDAA and the specified design control process will be used to maintain the CNV free volume in accordance with the design.

Containment Vessel Passive Heat Sink Parameters ITAAC

Containment heat sink dimensions, thicknesses, and materials constitute another set of key physical parameters that have been relied upon in the containment safety analyses and verified for the as-built configuration in an ITAAC by previous design centers. FSAR, Table 6.1-2 describes the materials used for fabrication of the NPM-20 CNV and associated components. NuScale has changed the upper CNV material from SA-508 for NPM-160 to SA-336 (F6NM) for NPM-20, while the lower CNV vessel material, SA-965 (FXM-19), remains the same. ITAAC 02.01.02 in SDAA Part 8, Table 2.1-1 about the CNV passive heat sinks is unchanged between the US 600 certified design and the US 460 standard design. The staff agrees with NuScale that this ITAAC will cover the ITAAC needed to verify the as-built passive heat sink parameters credited to the containment design basis analyses to conservatively bound the key input design assumptions about containment heat sinks dimensions and materials.

The DCA ITAAC 02.01.02 used the NPM-160 containment passive heat sink materials, dimensions, and properties information was provided in Tables 8-4, 8-5, and 8-6 of the "Containment Response Analysis Methodology Technical Report (CRAM)", TR-0516-49084-P, that was incorporated by reference in the NuScale DCA. The staff found that the three tables for the as-built containment heat sink configuration have been eliminated due to the merger of the CRAM Technical Report, TR-0516-49084-P, Revision 3 into the LOCA TR, Revision 3 and are no more available in the FSAR. By letter dated January 22, 2025 (ML 25022A198), NuScale provided the updated heat sink information for NPM-20 as three modified tables similar to the CRAM Tables 8-4, 8-5, and 8-6. As ITAAC 02.01.02 does not make an explicit reference to the validation of the containment heat sinks, NuScale provided a statement on the docket to confirm that the prescribed heat sink information in the three mark-up tables will be validated for the as-built containment structure as a part of ITAAC 02.01.02.

The staff concludes that the NuScale has provided sufficient information about the two ITAAC, and the FSAR, LOCA EM LTR, and SDAA Part 8 (ITAAC) mark-ups are acceptable. This has addressed the staff concerns.

6.2.1.1.4.1.8 Test Data Review

At the DCA stage, 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, GDC 50, PDC 38, 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 FSAR, 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 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 was an additional- step of evaluating the scaling and distortions—these were addressed primarily in the LOCA EM TR, Revision 2 (ADAMS Accession No. ML19331B585). The staff's review of the test data was performed under the LOCA EM TR, Revision 2 review. 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. The evaluations are performed using TR-0516-49422, which the staff has reviewed and found acceptable, subject to the conditions and limitations listed in the SER for TR-0516-49422 (ADAMS Accession No. ML20044E059). The staff concludes that the NIST-1 test data evaluation and the demonstration of the NRELAP5 applicability to the NPM-160 (US600) design is also applicable to the NPM-20 design (US460) as the containment response analysis PIRT has not changed and no new related physical phenomenon was identified.

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

LOCA EM TR, Revision 34, Section 8.3.2 describes the NIST facility scaling as it relates to NPM-20. NuScale conducted LOCA and IORV (Inadvertent Opening of an RPV Valve) NIST-2 tests along with a scaling analysis of the high-ranked phenomena identified in the LOCA PIRT. The staff's review focuses on the phenomena of 56% increase in core power, stored energy, break flow and energy, and ECCS flow and energy. The FOMs were the peak CNV pressure and the collapsed liquid level above the top of the core. NuScale demonstrated that the scaling distortions are either insignificant or do not impact the LOCA FOMs, making them acceptable to the staff, as summarized in SER Section 4.8.3. As in the DCA scaling, important phenomena related to peak CNV pressure and temperature are determined to be accurately scaled. As documented in SER Section 4.8.3, during the NIST-2 scaling review applicable to NPM-20, some significant scaling distortions were identified primarily between the NIST facility and the NPM-20 design
concerning the long-term cooling phase and DHRS. The staff accepts the NuScale NIST scaling applicability to the NPM-20 safety design FOMs for the containment pressure and temperature response.

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

TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P) 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 the CRAM. The LOCA and non--LOCA methodologies discussed in Section 15.0.2 of this SER were reviewed by the staff under LOCA TR and non--LOCA TR reviews, respectively. Section 15.0.2 of this SER also provides a summary of the staff's evaluation of the LOCA and non--LOCA methodologies. The staff conducted an audit of the applicant's LOCA and non---LOCA methodologies. The evaluations were performed using TR-0516-49422 and TR-0516-49416, which the staff has reviewed and found acceptable, subject to the conditions and limitations listed in the staff's SER for LOCA EM TR TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P).

There are no significant differences in physical phenomena between the LOCA and valve opening events for the NPM. Therefore, the PIRTs developed for the LOCA and non-LOCA methodologies are also applicable to the valve opening events and the CRAM. 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 LOCA EM TR, Revision 3.

Based on the information provided by the applicant, 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, LOCA EM TR, Revision 3 references these methodologies and identifies and justifies differences for the containment response analysis when compared to these methodologies.

6.2.1.1.4.1.9 Liquid Thermal Stratification inside the Containment Vessel

The NRELAP5 analysis shows that the pressurized RCS liquid flowing into the CNV flashes into steam. As the CNV pressure increases from approximately 21 kPa (absolute) (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. If 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 would not be 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, leading 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. At the DCA review stage, the staff requested that NuScale assess the impact of subcooled liquid water temperature stratification on the calculated CNV peak pressure.

NuScale provided an 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. Based on the results of the NPM and NIST-1 nodalization studies, the staff is able to confirm NRELAP5's capability to predict CNV liquid temperature stratification and, thus, peak CNV pressure accurately for the limiting DBE.

6.2.1.1.4.1.10 Containment Wall Condensation Heat Transfer

The CRAM models steam condensation on the CNV inner walls, as discussed in Section 6.8 of the LOCA EM TR, Revision 3 (ML23008A002). The LOCA TR 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. The staff's generic review of this correlation and its implementation in NRELAP5 is performed as part of the LOCA EM TR, Revision 3 review. LOCA EM TR, Revision 3 SER provides a summary of the staff's evaluation of the containment wall condensation heat transfer. The staff conducted an audit of the applicant's containment heat transfer modeling. The evaluations were performed using LOCA EM TR, Revision 3, which the staff has reviewed and found acceptable.

At the DCA stage, the staff was not able to assess the conservatism in the NRELAP5 condensation heat transfer model and the sensitivity of the predicted pressure and temperature margin to condensation modeling. Therefore, the staff sought justification from the applicant for the applicability of the steam correlation to the NuScale containment design and geometry. In response, NuScale conducted a sensitivity study to evaluate the impact of using a classical flat plate condensation correlation on CNV peak pressure for the limiting event. The staff audited the study and its results that showed that the condensation correlation used in the analysis of record is conservative with respect to CNV pressure for both NPM as well as NIST--1 HP-49 test CNV geometry. The related details of the staff evaluation of the licensing condensation correlation are documented in the audit summary report for the NuScale containment and ventilation systems (ADAMS Accession No. ML ML19308A041).

The staff evaluated the regulatory compliance of the NPM-20 CNV design with the applicable requirements of GDCs 16, 38, and 50 for the limiting containment DBA, as identified in the FSAR with respect to the containment design pressure and temperature. The staff performed an independent confirmatory analysis of the limiting CNV DBA, i.e., the RCS discharge line break LOCA, while accounting for the containment design changes from NPM-160 to NPM-20 as well as the resulting reduction in the M&E release load. In order to properly compare the applicant's NRELAP5 results with the staff confirmatory results using an independent code, the staff conducted an audit of the NPM-20 containment condensation convergence and energy balance, as follows.

a) {{

}}. The staff concluded that the NRELAP5 convergence issue encountered with the NPM-20 wall condensation is not safety significant.

b) The staff audited the safety analysis graphic results for the limiting RCS discharge line break for the NPM-20 CNV design CNV liquid and vapor temperatures; CNV wall temperature; reactor pool temperature; and energy balance. The NPM-20 results were similar to the ones presented for NPM-160 in the CRAM Technical Report, TR-0516-49084-P, Revision 3 and no non-physical behavior or inconsistency was observed.

c) The staff conducted an audit of the NRELAP5 code changes implemented between the versions used for the NPM-160 design in the approved DCA and the NPM-20 design for US460 that could impact the prediction of containment pressure and temperature response. The audited information covered the NRELAP5 applicability to NPM-20 analyses for the code changes from NRELAP5 Version 1.4 through Version 1.7. Based on the audited information, the staff concludes that the impact of the NRELAP5 code changes on the containment M&E release and calculated PCP is not significant.

The staff determined that the condensation of steam in NPM-20 occurs in a fairly comparable-to-slightly-lower pressure and temperature CNV environment and volume as in NPM-160. In addition, the PIRT ranking and the knowledge level of the condensation phenomenon have stayed the same and no new phenomenon is identified in NPM-20 to affect the condensation characteristics inside the CNV. Therefore, the staff concludes that extending the applicability and usage of the condensation correlation approved for NPM-160 to the NPM-20 design is appropriate. The staff concludes that the general condensation modeling approach used in NRELAP5 for NPM-20 is conservative for calculating CNV heat transfer and peak containment pressure.

6.2.1.1.4.1.10.1 Effect of Non-condensable Gases on NRELAP5 Model Prediction

In the NuScale design, the CNV atmosphere is maintained at a near vacuum initial condition during normal operation. However, FSAR, Section 6.2.1.3, describes a high initial CNV pressure that is used to maximize the initial NCG concentration in the CNV, and the condensation modeled on CNV inside diameter includes the effects of NCG. The LOCA TR, Revision 3 (ADAMS Accession No ML23008A002) 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 NCG.

LOCA LTR, Revision 4 presents the formulation and assumptions for modeling the adverse effect of NCG on condensation heat transfer inside the containment that would lead to a higher CNV peak pressure as compared to pure steam. {{

}}. NRELAP5 uses a two-fluid model for flow of a two-phase vapor/gas-liquid mixture that can contain NCG components in the vapor/gas phase as well. LOCA LTR, Revision 4, Section 8.2.8.2 describes {{ }} can be used to model film condensation with and without the presence of NCGs. The presence of NCGs suppresses condensation due to the reduced saturation temperature corresponding to the partial pressure of vapors at the condensate film interface. With NCG present, {{ }}. The staff reviewed the modeling details of NCG effects in LOCA LTR, Revision 4 and found them identical to what was reviewed at the DCA stage.

FSAR Section 6.2.1.3 conservatively assumes that {{ }} in order to maximize the effects of NCG on CNV heat removal. New FSAR Table 6.2-2 markup (ML24215A142) documents an initial total NCG mass of 137 lbm inside the CNV, which is the same value specified in the LOCA TR, Revision 4, Table 9-9. Table 9-9 that lists the initial conditions for M&E release events for the CRAM also documents {{ }}. The staff audited the NPM NCG mass calculations inside the NPM-20 under steady state operating conditions. The total NPM NCG source term is calculated {{ }}. The staff found that the NCG total initial amount in the CNV and RPV documented in the in new FSAR Table 6.2-2 markup (ML24215A142) is consistent with {{ }}. The staff determined that the {{ }} was conservative based on the bounding initial CNV pressure/temperature, pressurizer level/temperature, and control rod position applicable to the containment design basis analysis. The staff finds it reasonable to assume {{ }} as sensitivity studies of the NCG chemical composition performed during the US600 design certification had demonstrated that to be an appropriate assumption. The staff concludes that NuScale has conservatively accounted for the adverse effect of NCGs on the CNV heat removal in the FSAR containment safety analysis.

Both new FSAR Table 6.2-2 and LOCA TR, Revision 3, Table 9-9 document 3.0 psia as the initial CNV atmosphere pressure, while an audit of the NRELAP5 CNV biased decks provided by NuScale, showed {{ }} as the initial containment pressure. It was found that a higher {{ }}. The staff found it conservative to assume a higher CNV initial pressure for analysis to maximize the adverse effect of NCG on condensation on the CNV wall inner surface.

6.2.1.1.4.1.11 Initial and Boundary Conditions for the NPM-20 Input NRELAP5 Model

The NPM-20 containment analysis model is based on the LOCA EM that is developed using the evaluation model development and assessment process (EMDAP) of "Transient and Accident Analysis Methods," Regulatory Guide (RG) 1.203, and satisfies the applicable requirements of "ECCS Evaluation Models," 10 CFR Part 50, Appendix K and "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," 10 CFR 50.46. Multiple layers of conservatism are incorporated in the LOCA EM to ensure a conservative bounding approach to analyzing LOCA transients. These conservatisms stem from the application of the modeling requirements of 10 CFR Part 50, Appendix K and through a series of conservative modeling features implemented via input modeling assumptions.

LOCA EM TR, Revision 3 documents the CRAM and is referenced in the FSAR Chapter 6. LOCA EM TR, Revision 3 presents the updated NuScale LOCA EM that also integrates the CRAM updated for the design changes from NPM-160 to NPM-20 to evaluate the containment pressure and temperature response. The earlier CRAM version developed for the NPM-160 design was described in a separate technical report, TR-0516-49084-P, Rev.3 that was incorporated by reference in the NuScale DCA Chapter 6. TR-0516-49084-P, Rev.3 was also used in the present review for comparing the CRAM methodologies for NPM-160 and NPM-20 in the following, and is referred to as "CRAM TR" in the present SER.

The staff also reviewed EC-A013-7725 (NPM-20 CNV Pressure and Temperature Response Analysis) (ML23011A012) that was submitted with the NRELAP decks to ensure that the containment analysis results presented in the LOCA EM TR, Revision 3 also reflected the NPM-20 design parameters. {{

}}.

The staff ensured that EC-A013-7725, Table 3-1 is consistent with Table 9-9 of LOCA EM TR, Revision 3 that summarizes the initial conditions used in the CRAM. New FSAR Table 6.2-2 markup (ML24215A142) summarizes key initial conditions for the containment pressure and temperature response analysis for the NPM-20. The staff concludes that NuScale followed the same conservatism rationale for the NPM-20 CNV and reactor pool modeling assumptions that was used for NPM-160 to minimize the CNV heat removal and maximize the CNV pressure. The staff also found that the following NPM-20 design and analysis changes made in the US460 are appropriately incorporated in Table 3-1:

a) Loss of normal AC and DC power may not occur at time zero or on the turbine trip,

b) RVVs do not have IAB anymore and open immediately,

c) Only RRVs have IABs and open at the IAB design criteria based on the differential release pressure,

d) DHRS is credited to the CNV design basis scenarios.

The staff compared LOCA EM TR, Revision 3, Table 9-9 with CRAM TR Table 5-1 on the initial conditions for the primary system's M&E release event analyses. The tables present the conservative parameters values for the CRAM initial conditions for NPM-20 and NPM-160, respectively. Many parameter values were identical. The staff conducted an audit of the NRELAP5 decks and ensured that the modified parameter values were used in the NRELAP5 containment decks for NPM-20. The following are the staff's specific observations about some key input CNV design parameters.

a) The LOCA EM TR, Revision 3 explains that a {{ }} increase in RPV fluid volume (via thermal expansion) is applied to the nominal value {{ }} that is calculated from

RPV cold dimensions by increasing the initial PZR liquid level from 68 percent to 75.5 percent in the NPM-20 CNV model. The staff agrees that increasing the RPV fluid volume {{ }} for NPM-160 {{ }} for NPM-20 by increasing the PZR liquid level is more conservative with respect to the mass release upon an event initiation.

b) LOCA EM TR Table 9-9 lists CNV free volume as {{ }} nominal and 6000 cubic ft used in the CRAM, while CRAM TR Table 5-1 documented it as {{ }}. CRAM TR Table 3-5 also mentioned that the CNV free volume used is conservatively reduced to 6000 cubic ft. The staff concludes that the CNV free volume value of 6000 cubic ft is documented in LOCA EM TR, Revision 3 and is consistently used in the NPM-20 CRAM demonstration. Containment free volume is typically verified by ITAAC because it meets the various “key parameter” attributes underscored in several 14.3 SRP Sections. Therefore, NuScale has included a one-time ITAAC for the US460 design to verify the as-built CNV free volume for the first module ever built to conservatively bound the minimum value used in the NPM-20 CRAM.

c) {{ }}. The CNV atmosphere is maintained at a near-vacuum initial condition in the NPM design normal operation. The staff finds using a higher initial pressure (3.0 psia) in the CRAM to be conservative, as it bounds the CNV internal pressure as well as the initial NCG mass inside the CNV to maximize the adverse effect of NCG on CNV wall condensation.

d) {{ }}. The staff found the change to be conservative as it would reduce the heat transfer from the containment to the reactor pool, thereby, leading to a higher peak containment pressure and temperature.

e) {{ }}. The staff found the change to be conservative as it would reduce the heat transfer from the containment to the reactor pool, thereby, leading to a higher peak containment pressure and temperature.

LOCA EM TR, Revision 3, Table 9-9 added some additional CRAM parameters that were not included in CRAM TR Table 5-1. The following are the related staff observations.

a) {{ }}.

b) {{

}}.

c) {{

}}. The staff found that the new range is duly incorporated in the NRELAP5 containment analysis model.

d) T-ave is a parameter that is controlled at a lower nominal value of 540 °F for NPM-20 than for NPM-160. The staff investigated why, despite the 56 percent core power increase, the primary T-ave value of 545°F used for NPM-20 is 10°F lower than 555°F used for NPM-160. A cooler average initial RCS temperature appeared inconsistent with the FSAR Figure 4.4-2 that shows an increase in the analytical design operating limits in the FSAR with Thot increasing from 590°F to 605°F and the High [Thot] Temperature Analytical Limit increasing from 610°F to 620°F, when compared with DCA Figure 4.4-9. As Thot is a function of the core power level and the RCS flow rate, the higher Thot and higher Thot analytical limit are consistent with the design power increase. However, the staff found that the lower T-ave value for NPM-20 reflects its desired nominal operating condition maintained for the US460 design by utilizing other selected design parameters. In addition, the CRAM conservatively applies an additional 5 °F margin to the 540 °F T-ave value for using a 545 °F T-ave in the NPM-20 limiting M&E release and CNV response analyses that is consistent with the uncertainty and deadband used in the FSAR Chapter 15 analyses. Therefore, the staff concludes that a credible explanation exists for the apparent inverse trend between the RCS T-ave and the plant operating temperature envelope.

Hereunder are two additional conservatisms that staff investigated to ensure an NPM-20 model development with respect to a conservative M&E release into the containment.

a) LOCA EM TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P), uses the 1973 ANS decay heat standard with a 1.2 multiplier for the NuScale LOCA methodology for modeling the RPV FOMs to meet the Appendix K requirements. However, the CRAM, also documented in LOCA EM TR, Revision 3, is based on the ANS-5.1-1979 decay heat standard and a two-sigma uncertainty assumption, which is less conservative. The staff determined that using the less conservative 1979 ANS decay heat standard in the CRAM over the 1973 standard is justified as SRP Section 6.2.1.3 allows the ANS-5.1-1979 decay heat model specified in SRP Section 9.2.5 as an acceptable means for calculating M&E release into the containment. A staff review of ANS-5.1-1979 standard showed that it defines the uncertainty statistically based on standard deviation in a normal distribution and suggests the 2-sigma level as an appropriate decay heat uncertainty for comparison with the 1973 ANS 5.2 standard and its uncertainties. The staff also noted that even though the CRAM allows for a 2-sigma uncertainty, the FSAR Chapter 6 containment safety analyses have used a bounding 1.2 multiplier that was also approved for the US600 design for the DCA. The staff accepts the decay heat model and uncertainty used in the CRAM for NPM-20.

b) LOCA EM TR, Revision 3 documents that the effect of liquid droplet entrainment from the break/valve flow was evaluated {{

}}. LOCA LTR, Revision 4 concludes a conservative modeling approach for liquid entrainment, but no results are presented. The staff looked into an NPM-160 sensitivity study that showed {{ }} leads to a lower PCP and temperature due to an overall reduction in the M&E release rate because of {{

}}. The staff determined that the PCP and temperature were less sensitive {{
}} for the liquid break LOCAs than the steam breaks at the PZR top. The secondary line breaks that are essentially non-limiting, are less sensitive {{
}} than the primary side breaks due to the smaller volume and smaller pressure of secondary systems. Based on the consistency between the NPM-160 and NPM-20 design PIRTs with respect to PCP and temperature calculations and the similarity of the thermal-hydraulics of the liquid entrainment phenomenon inside the CNV, the staff accepts that a conservative modeling approach for liquid entrainment has been used.

The staff found the NPM-20 initial and boundary conditions, assumptions, and rationales presented in LOCA EM TR, Revision 3, Sections 5.5 and 5.6, to be conservative and consistent with the submitted NREALP5 containment model.

6.2.1.1.4.1.12 NRELAP5 Modeling Results

The staff reviewed the NRELAP5 containment analysis results and supporting details presented in EC-A013-7725 (NPM-20 CNV Pressure and Temperature Response Analysis) (ML23011A012). The results presented in EC-A013-7725 were obtained by using the CRAM that is documented in LOCA EM TR, Revision 3. The CRAM is used to evaluate the M&E release from a spectrum of primary system and secondary system design basis transients and accidents for the NPM-20 CNV pressure and temperature response. The CRAM describes the conservative containment analysis assumptions and outlines the bounding input parameters. The CRAM also supports analyses for non-LOCA events, as well as the extended passive cooling evaluation. New FSAR Table 6.2-2 markup (ML24215A142) summarizes the common initial conditions conservatively chosen for the M&E release analysis to maximize peak CNV temperature and pressure, while EC-A013-7725, Table 3-1 and LOCA EM TR, Revision 3, Table 9-9 provide more details. {{

}}. FSAR Table 6.2-3 summarizes the containment response analysis results for the NPM-20 primary system (LOCA and valve opening events) and secondary system (MSLB, FWLB) design basis M&E release scenarios.

The staff reviewed the containment response analysis results for each of the six primary containment M&E release scenarios for NPM-20 as summarized in {{
}}.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. The description of each M&E release scenario contains a table of seven DBEs analyzed as the scenario spectrum, based on the respective assumptions for loss of AC power, loss of EDAS, ECCS actuation level, and single failure (RVV, RRV, or RRV-RVV train). The seven DBEs for each scenario spectrum were identified to perform the CNV response sensitivity to test various combinations at the event initiation for being the limiting event, e.g., whether or not the normal AC power and EDAS power (DC) supply is available, a single failure is active, and ECCS

actuation on low RPV riser level is biased to the high end of range. The two RRVs in the NPM-20 use an IAB release pressure of 3447 kPa (differential) (500 pounds-force per square inch differential (psid)). The staff found the formulation of the seven DBEs for each release spectrum to be appropriate for meeting the regulatory guidance for the sensitivity of the peak containment pressure and temperature to loss of power and single failure. {{

}}.

The table for each scenario spectrum also includes the corresponding NRELAP5 output results for the seven sensitivity DBEs for the reactor trip signal time for scram (sec), ECCS valves opening time (sec), peak CNV pressure occurrence time (sec), as well as the corresponding peak CNV pressure and peak CNV temperature values. {{

}}

The staff compared the consolidated spectrum transients for the above quantities with the sequence of events for the limiting DBE laid out in FSAR Table 6.2-7 and found that the various temperature, pressure, and level trends and their abrupt changes physically correlated well with the resulting RVV and RRV opening. The staff also conducted some sensitivity studies using the submitted NRELAP5 decks for the limiting containment DBE response and the M&E release and found the peak CNV pressure and temperature results to be about the same as those documented in EC-A013-7725. The staff was also able to confirm the limiting M&E release data documented in FSAR Table 6.2-9. Summarily, the spectrum tables for the six primary system M&E release cases of {{

}} DBEs.

The staff evaluated the regulatory compliance of the NPM-20 CNV design with the applicable requirements of GDCs 16, 38, and 50 for the limiting containment DBA, as {{

}}. Based on the M&E release spectrum analyses, the overall limiting peak calculated containment pressure and temperature occurs as the result of the RCS discharge line break. The LOCA caused by the RCS (CVCS) discharge line break from the downcomer (DL-2) with the loss of normal AC power (but no EDAS loss and no active single-failure) gave the peak CNV pressure of 919 psia and the maximum CNV wall temperature of 532 ^{o}F, bounding all remaining 41 primary and 18 secondary release DBEs analyzed. The containment pressure was reduced by 50 percent from its peak value, well within 24 hours of the initiation of the DBA and stayed low to meet PDC 38. More detailed results and relevant discussions on the primary and secondary release DBEs are provided in the following two sections. The staff also reviewed NuScale's submittal, dated March 28, 2025 (ML25087A256, proprietary and ML25087A257, non-proprietary) that incorporated NRELAP code version changes from 1.6 to 1.7, and break flow model and NPM-20 basemodel updates that include the addition of DHRS heat structures. Despite higher M&E release rates, the resulting safety margins slightly increased for all eight limiting spectrum cases documented in Table 6.2-3, due to earlier actuation of ECCS. The staff determined the changes to be non-safety significant and, thus, acceptable.

For the NPM-20 CNV design, the staff has also performed confirmatory analyses of the limiting containment DBA using MELCOR and TRACE codes while accounting for the containment design changes from NPM-160 to NPM-20, conservative assumptions, resulting reduction in the M&E release load, and DHRS crediting. The staff's MELCOR and TRACE confirmatory calculations predicted lower peak CNV pressures and temperatures than the corresponding FSAR results for both the primary and secondary systems release limiting DBEs. The staff concludes that NuScale has demonstrated acceptable safety margins for the NPM-20 containment that meet the regulatory requirements of GDC 16, GDC 50, and PDC 38.

6.2.1.1.4.1.12.1 Postulated Primary-system Pipe Ruptures inside Containment

The M&E release and containment pressure and temperature response methodology as presented in LOCA TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P), analyzes a spectrum of possible break locations, sizes, and types of M&E releases. {{ }} (NPM-20 CNV Pressure and Temperature Response Analysis) provides the NRELAP5 containment analysis results and supporting details for the six primary system M&E release scenarios to determine the limiting results for the NPM-20 CNV design. The limiting NPM-20 CNV design basis M&E release event is a LOCA caused by the RCS (CVCS) discharge line break from the downcomer (DL-2) and is based on a loss of normal AC power, no DC power (EDAS) loss, no active single failure, {{ }}, and an IAB release pressure of 3447 kPa (differential) (500 psid). As described in {{ }} SDAA FSAR Section 6.2.1.1.3, Table 6.2-3, the peak containment pressure (PCP) calculated for the limiting event is 6336 kPa (absolute) (919 psia).

FSAR, Table 6.2-1 on containment design and operating parameters specifies the NPM-20 design pressure and temperature. With the NPM-20 CNV design pressure of 8274 kPa (absolute) (1200 psia), the SDAA containment design has an about 22 percent pressure safety margin with respect to the limiting containment DBE. This meets the NuScale DSRS Section 6.2.1.1.A guidance of the containment design providing at least a 10-percent margin above the PCP. The same RCS discharge line LOCA DBA also resulted in the overall maximum CNV wall temperature of 277.8 °C (532 °F), showing sufficient margin with respect to the NPM-20 containment design temperature of 315.6 °C (600 °F).

FSAR Chapter 6, Figures 6.2-7 through 6.2-14 show the graphical results for the limiting containment DBE, which include the CNV maximum pressure and maximum CNV wall temperature and instantaneous and integrated M&E release from the break and the ECCS valves. FSAR Table 6.2-9 presents the primary systems instantaneous M&E release rates for the limiting containment pressure and temperature event. The staff audited the submitted NRELAP5 run for the limiting events against the SDAA M&E release and pressure and temperature response results and found them to be identical. The staff also found that the design basis sequence of events, availability of AC/EDAS power, and ECCS actuation for the limiting CNV DBE documented in FSAR, Table 6.2-7 correlates with the changes and trends observed in the pressure and temperature transients, as well as the M&E release transients. In addition, NuScale included a revised Figure 6.2-7 in FSAR, Revision 1. The containment pressure results now show that the CNV pressure reduces to less than 50 percent of the peak calculated pressure in less than 24 hours and stays low, as required by PDC 38. The staff concludes that the results are within the design pressure and design temperature of the CNV.

6.2.1.1.4.1.12.2 Postulated Secondary-System Pipe Ruptures inside Containment

Section 6.2.1.4 of this SER evaluates the secondary system M&E releases following an MSLB and FWLB. The NuScale- LOCA EM TR TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P) and non-LOCA- TR-0516-49416-A, Revision 5 (ML25136A339 NP, ML25136A340 P) 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 NRELAP5 decks and found that 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 FSAR, 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's 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. {{ }}. The NuScale MSLB and FWLB DBEs have unique features compared to typical PWRs. The NuScale design uses helical coil steam generators (HCSGs) 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 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 EDAS power, the two RVVs open immediately but the two RRVs do not open until the differential pressure drops below the IAB setpoint. -

The staff also reviewed the NRELAP5 CNV pressure and temperature results of the two secondary system M&E release inside the CNV, i.e., MSLB and FWLB, as summarized in {{ }}. Both the secondary system analyses used the same input parameters that are used in the primary system release analysis. Both MSLB and FWLB M&E release scenarios were analyzed for a spectrum of {{ }} DBEs {{ }}. {{ }}. The staff also obtained M&E release rates from NuScale for the limiting secondary systems release containment pressure and temperature event.

The staff found the formulation of the {{ }} DBEs to be appropriate for both secondary release spectra for meeting the regulatory guidance for the sensitivity of the peak containment pressure and temperature to the loss of AC power, loss of EDAS, ECCS actuation level, and single failure. The staff found that the addition of the MSIV and FWIV single failures to both MSLB and FWLB M&E release scenarios in addition to the single failures of RVV, RRV, or RRV-RVV train, completed the required sensitivity study for the secondary system release events. The staff found it appropriate that the limiting events {{ }} for both MSLB and FWLB are mainly driven by an immediate loss of AC and EDAS power with no single failure that leads to an immediate opening of both RVVs and actuates both RRVs at the very start of the transient. The staff noted that the immediate RVVs opening due to the EDAS power loss adds the primary side M&E release into containment during the secondary system M&E release. Coupled with the AC power loss this would result in a delayed RRV opening and, thus, a delayed peak pressure timing, which leads to a higher peak containment pressure. As identified by the sensitivities, no single failure was found to be more limiting for the secondary system release events.

The peak calculated containment pressure resulting from a secondary side M&E release is postulated as the result of a double-ended main steamline break MSLB inside the containment. For postulated secondary system pipe ruptures, single failures are considered for MSIV and FWIV failure, in addition to the RVV, RRV, and RRV-RVV train failures that were considered for the primary system release events. The limiting secondary system release analysis {{ }} is found to be based on a loss of both normal AC and DC power (EDAS), no active single failure, {{ }}, and an IAB release pressure of 3447 kPa (differential) (500 psid). The loss of EDAS power actuates the ECCS and immediately opens the RVV at the reactor trip, which results in a peak CNV pressure of 6123 kPa (absolute) (888 psia). The event also led to the peak calculated temperature of 526 ^{o}F. The peak CNV pressure and CNV wall temperature results for secondary system line break event are bounded by the primary system M&E release events. The RCS LOCA peak CNV pressure and temperature bound the MSLB and FWLB peak CNV pressures and temperatures.

Based on the presented results, the staff concluded that the CVCS discharge line break LOCA and inadvertent ECCS valves openings are more limiting for NPM-20 than the MSLB and FWLB transients.

6.2.1.1.4.2 External Pressure

To satisfy the requirements of PDC 38 and GDC 50 with respect to the functional capability of the CHRS 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-20 containment CNV external design pressure is 345 kPa (absolute) (50 psia), which is based on an internal pressure of 0 kPa (absolute) (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 15.9 m (52 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 internal pressure is normally at near-vacuum conditions, and thus there is no minimum containment pressure analysis as the CNV is designed to the conditions that are more conservative than could exist in the plant design basis, as stated above. As described in the DCA audit report (ADAMS Accession No. ML20054A060), crediting the entire liquid pool hydrostatic head in the containment stress (or differential pressure) analysis as the external pressure boundary condition would essentially shift the point of peak CNV dP from the bottom of the containment to the top, and would result in an additional CNV pressure margin equal to the total containment liquid hydrostatic pressure head. This was confirmed during the audit and documented in the CRAM TR (ADAMS Accession No. ML19330F387). During the NuScale DCA review and audit, the staff had made a determination that atmospheric pressure was already credited in the CNV structure ASME stress analysis that had resulted in the NPM-160 containment design pressure rating. As NPM-20 does not have any related design changes and the module free volume has remained the same, the staff concludes that the same finding can also be extended to the NPM-20 design that has a 8274 kPa (absolute) (1200 psia) CNV ultimate design pressure, as accepted in Section 3.8.2.4.3 of this SER. Therefore, the staff concludes that the NPM-20 containment structure is designed for external pressure conditions that could not be exceeded and, thus, meets PDC 38 and GDC 50 under LOCA conditions with respect to the functional capability of the containment against possible damage from external pressure conditions.

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

NuScale FSAR, 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. FSAR, 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 FSAR, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment." The CNV instrumentation provided to monitor and record the required containment parameters and the capability to operate in post-accident environments are discussed in FSAR, Chapter 7, "Instrumentation and Controls," and Section 3.11. The portions of post-accident monitoring (PAM) equipment required to be environmentally qualified are discussed in FSAR, Section 3.11.2.1. FSAR, 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.

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

The staff evaluation of the Three Mile Island (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.

6.2.1.1.4.5 Environmental and Dynamic Effects Design Bases (GDC 4)

The staff's evaluation of the NuScale design relative to the requirements of GDC 4 is in Sections 3.6.3 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.

6.2.1.1.4.6 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 6 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 FSAR, 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 EDAS). The FSAR 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 FSAR 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 post-accident, 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. Section 9.2.5.4.4 of this SER concludes that GDC 5, as it relates to capability of shared UHS to perform its required safety function, is satisfied.

6.2.1.1.4.7 Technical Specifications

Safety analyses in FSAR, 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 SDAA Part 4, "Generic Technical Specifications—NuScale Nuclear Power Plants," Revision 1, Volume 1, "Specifications." The TS associated with this evaluation of NuScale FSAR, Section 6.2.1.1 are provided in SDAA Part 4, Revision 1, Volume 1, Section 3.6.1, "Containment," and Section 3.5.3, "Ultimate Heat Sink." Limiting Condition for Operation (LCO) 3.6.1 deals with the operability of the containment. LCO 3.5.3 (Ultimate Heat Sink) is included to maintain the UHS within the specified limits that ensure that the reactor pool level remains at least 52 feet and the bulk average temperature does not exceed 120 degrees F. The CNV atmosphere is normally maintained at a near-vacuum initial condition in the NPM design. {{

}}. The staff note that LCO 3.5.3, in conjunction with LCO 3.4.5 (RCS Operational Leakage), restricts the containment pressure during operation to the curve defined by FSAR, Figure 5.2-2, "Containment Leakage Detection Acceptability."

FSAR, Figure 5.2-2 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 FSAR, Figure 5.2-2, the staff notes that LCO 3.4.5 does, in fact, restrict operation to slightly less than 3 psia used in the limiting peak CNV pressure/temperature analyses. The NPM-20 containment response analyses are based on a conservative 140 ^{o}F initial pool temperature as compared to the 120 ^{o}F limit used in Technical Specification 3.5.3. Figure 5.2-2 still shows 2.8 psia as the upper operational limit of acceptable RCS leak detection range at 140 ^{o}F, which is still less than 3 psia used in the limiting peak CNV pressure/temperature analyses. Therefore, the staff has found the TS controls relative to initial containment pressure acceptable.

6.2.1.1.5 Conclusion

The staff reviewed the NPM standard design for containment structure using the relevant NRC regulations and associated acceptance criteria specified in DSRS Section 6.2.1.1.A. The NRC staff finds that the applicant has shown that the calculated containment peak pressure and temperature have sufficient margin to the containment design pressure and temperature for all limiting DBEs analyzed. The applicant has also shown that containment pressure is reduced to less than 50 percent of the peak calculated pressure for all DBEs within 24 hours of the postulated event initiation. Confirmatory analyses performed by the staff using two independent thermal-hydraulic codes, MELCOR and TRACE, agreed with the applicant’s conclusions regarding the acceptability of the containment design. Based on the above evaluation, the staff finds that the applicant has fully addressed the required information related to the NPM containment design basis calculations and meets the acceptance criteria that are based on the relevant regulatory requirements.

6.2.1.2 Containment Subcompartments

As stated in FSAR, Section 6.2.1.2, the NuScale CNV has no interior subcompartments. Therefore, the staff agrees that no containment subcompartment design basis M&E release analysis required for the NPM-20 containment design.

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

6.2.1.3.1 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 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 two RVVs, located at the top of the reactor vessel, and the two reactor RRVs, located in the lower part of the reactor vessel. Any breach of the RCS pressure boundary results in an 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 the 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.

6.2.1.3.2 Summary of Application

FSAR: FSAR, 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 develop the containment response analyses. The model used is a modification of the LOCA evaluation model used for transient analyses in Chapter 15. FSAR, 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 NCG concentrations inside containment, and the various conservatisms applied to the initial stored energy inside the primary system.

FSAR, 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 stated that if power is available, the reactor trips on high containment pressure, and when the ECCS setpoint is reached (low primary system inventory), the ECCS valves open and, shortly after that, the limiting containment conditions are reached. In general, Section 6.2.1.3 provides a high-level description of the containment system response. The detailed material the staff relied upon to make its finding is located in LOCA EM TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P) and is discussed in further detail below.

Technical Specifications: TS associated with the analysis assumptions described in this section are located in SDAA Part 4, Revision 1, Volume 1, TS Section 3.5, "Passive Core Cooling Systems (PCCS)," for the UHS, DHRS, and ECCS valves, and in TS Section 3.6, "Containment Systems" for the containment.

Topical Reports: LOCA EM TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P) provides a detailed description of the evaluation model used to produce the M&E release values. The model is a modification of the NRELAP model used for the FSAR, Chapter 15 transient analyses, as described in TR0516-49422-P. Qualification of the NRELAP5 model is demonstrated in TR0516-49422-P. CRAM as documented in LOCA EM TR, Revision 3, relies on this qualification as part of the demonstration of NRELAP5's adequacy to model the phenomena present during a LOCA or a valve-opening event within a NuScale module.

LOCA EM TR, Revision 3 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 conservative 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 FOM for each break.

6.2.1.3.3 Regulatory Basis

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

- 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 Section 6.2.1.3 and DSRS Section 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-Of-Coolant Accidents (LOCAs)”, 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.

6.2.1.3.4 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 FSAR, Section 6.2.1.3, and a more detailed discussion of the inputs, methodology, and associated modeling nuances in LOCA EM TR, Revision 3.

The approach chosen by NuScale for calculating containment conditions following a high -energy line break differs slightly from the traditional approach, in that NuScale used the same computer code (NRELAP5) 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 is evaluated separately as part of the NRC staff’s evaluation of LOCA EM TR, Revision 3 (ML23008A002). NuScale used both a base LOCA transient model and a 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 minimize the CNV heat removal and maximize the resulting containment pressure and temperature.

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 FSAR, 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 NCG.

The CRAM, as documented in LOCA EM TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P), is based on the ANS-5.1-1979 decay heat standard and a two-sigma uncertainty assumption. LOCA EM TR-0516-49422, Revision 4 also uses the more conservative 1973 ANS decay heat standard with a more conservative 1.2 multiplier, i.e., 20% uncertainty, for the NuScale LOCA methodology for modeling the RPV FOMs to meet the Appendix K requirements. The staff determined that using the less conservative 1979 ANS decay heat standard in the CRAM over the 1973 standard is in compliance with both the SRP and DSRS Sections 6.2.1.3 for the containment design that allow the ANS-5.1-1979 decay heat model specified in SRP Section 9.2.5 as an acceptable means for calculating M&E release into the containment. A staff review of ANS-5.1-1979 standard also showed that it defines the uncertainty statistically based on standard deviation in a normal distribution and suggests the 2-sigma level as an appropriate decay heat uncertainty for comparison with the 1973 ANS 5.2 standard and its uncertainties. The staff also noted that even though the CRAM allows for a 2-sigma uncertainty, the FSAR Chapter 6 containment safety analyses have used the 1979 ANS standard with a bounding 1.2 multiplier that was also approved for the US600 design for the DCA. The staff accepts the fission product decay heat model and uncertainty used for the NPM-20 containment safety analyses, as documented in FSAR Section 6.2.1.3.2.

Although the initial containment pressure is expected to be very low (less than 0.1 psia), there appeared to be no explicit technical specification to control on the containment pressure upper limit during operation, save for the high containment pressure reactor trip. In this regard, the applicant's response to the DCA RAI 8793 (ADAMS Accession No. ML17209B011) is equally relevant to SDAA FSAR. Following the applicant's explanation, a review by the staff has indicated that LCO 3.5.3 (Ultimate Heat Sink), in conjunction with LCO 3.4.5 (RCS Operational Leakage), restricts the containment pressure during operation to the curve defined by FSAR Figure 5.2-2, "Containment Leakage Detection Acceptability," but only to slightly less than 3 psia for a pool temperature of 140 degrees F, which is higher than the 120 ^{o}F limit restricted in Technical Specification 3.5.3. To evaluate the effect of this increase on initial conditions, the staff had requested during the DCA review that the applicant also provide a sensitivity study documenting the effect of increasing the initial pressure. In its response (ADAMS Accession No. ML17265A825), the applicant provided the results of a sensitivity study increasing the initial containment pressure to 3 psia, which resulted in a small peak containment pressure rise of 2 psia. The staff expects such a small effect of increasing the initial CNV pressure observed for NPM-160 to be also applicable to NPM-20 that has the same CNV free volume as NPM-160. 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 the conditions used acceptable for NPM-20.

Based on the above initial conditions, the staff finds that NuScale has adequately conformed with the guidance provided in DSRS Section 6.2.1.3 and used appropriate bounding initial conditions 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 and temperature results.

Break Spectrum

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. 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: 1) the traditional break size and location considerations and 2) the evaluation of the ECCS actuation condition and timing.

Break locations analyzed include the RCS (CVCS) discharge from the downcomer and (CVCS) injection lines from the riser, pressurizer spray line, and RPV high-point-vent (HPV) degasification line near the top of the vessel, as well as the inadvertent openings of a reactor safety valve, RVV, RRV by a mechanical failure, and inadvertent opening of both RVVs caused by an inadvertent ECCS actuation signal (ECC). The pressurizer spray line, RPV HPV degasification line, and reactor safety valve are all located near the top of the reactor. The RPV HPV 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 six M&E release locations. These included breaks of the RCS discharge and injection lines, the break of the RPV HPV degasification line, and inadvertent openings of an RRV, an RVV, and inadvertent actuation of ECCS valves. 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, the containment pressure and temperature are limiting for the RCS discharge line break when it is coincident with loss of AC power but still holding on to EDAS.

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. The limiting pressure for this break is produced from a similar event sequence as that causing the peak containment pressure and temperature in the RCS discharge line break. In both the RCS discharge and injection line breaks, the peak CNV pressure and temperature were caused by a loss of AC power but a continuation of EDAS that barred the two RVVs to open immediately after the transient initiated. In both cases, ECCS actuation was delayed for several minutes after the reactor trip, which maximized the peak containment pressure and temperature. Detailed system response curves are provided for this transient in EC-A013-7725 (NPM-20 CNV Pressure and Temperature Response Analysis).

The RPV HPV degasification line break and the inadvertent opening of an RVV present 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 inadvertent RRV opening peak containment pressure and temperature for the corresponding sensitivity cases are second only to the most bounding RCS discharge line break. The peak containment pressure and temperature resulting from an inadvertent actuation of the ECCS valves fell in the middle of the range of all six events analyzed.

The bounding M&E release event, the RCS discharge line break, 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 ECCS valves open by reaching the IAB differential pressure setpoint around 170 seconds into the transient. EC-A013-7725 provides the summarized results of all seven sensitivity cases on the same graph. The staff audited the corresponding NRELAP5 models for each of the sensitivity cases for RCS discharge line break examined by the applicant and agrees that the discharge line break case represented in EC-A013-7725 is the limiting M&E release scenario, based on the bounding design basis containment pressure and temperature response.

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 (an RVV, or an RRV, 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. All six primary system and two secondary system design basis release events {{

}} consistently showed that a secondary side failure of a valve to close has a negligible impact on pressure and a failure of an ECCS valve to open results 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

LOCA EM TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P), documents that CRAM maximizes the CNV peak pressure and temperature by using a bounding M&E release from the RPV to CNV. For this purpose, the CRAM uses the Moody critical flow model for two-phase and saturated water-steam mixture flow, and the Henry-Fauske critical flow model for single-phase flow, i.e., subcooled liquid and superheated steam conditions. The Moody model is consistent with the guidance associated with the model endorsed by the staff in DSRS Section 6.2.1.3; {{

}}, and the staff found the model acceptable in previous analyses reviewed by the NRC. This is an important conservatism in modeling the critical flow for M&E release into the containment consistent with Appendix K. 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. LOCA EM TR, Revision 3, Sections 8.2.2 and 8.2.3 demonstrate the adequacy of the evaluation model for two-phase and single-phase choked and un-choked flows for predicting the M&E release based on the assessment of the NRELAP5 mass flow predictions with experimental data.

LOCA EM TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P), which incorporates the CRAM, {{

}}. The staff concluded that the code changes did not make any significant changes in the M&E release or CNV T/H response that justifies the break-flow modeling approach and the fidelity of calculation of M&E release into the CNV.

FSAR, Table 6.2-9 presents the total break and ECCS M&E release rates for the combined, limiting peak containment pressure and temperature DBE, i.e., the RCS discharge line break LOCA. FSAR, Figures 6.2-13 and 6.2-14 provide the graphical representations of the primary system's M&E release rates for the limiting CNV DBE. However, no secondary M&E release tables were presented for NPM-20 in the FSAR, Chapter 6 or LOCA EM TR, Revision 3. Considering the NRELAP5 code changes, convergence issues, choked flow model implementation, and uncertainties associated with DHRS performance and ECCS actuation, the staff decided to further evaluate the primary and secondary systems M&E release modeling for NPM-20, as follows.

1. NuScale provided the M&E release tables for the limiting secondary systems release event with respect to peak containment pressure (PCP) and temperature. NuScale provided a {{ }}. NuScale emphasized that the resulting PCP for the limiting MSLB (888 psia) is less than the maximum PCP (919 psia) for the limiting primary system's M&E release event.

2. LOCA LTR, Revision 4 documents that the effect of liquid droplet entrainment from the break/valve flow was evaluated by {{ }}. LOCA LTR, Revision 4 concludes a conservative modeling approach for liquid entrainment, but no results are presented. The staff looked into the NPM-160 sensitivity study that showed {{ }} leads to lower PCP and temperature due to an overall reduction in the M&E release rates because {{ }}. The staff determined that the PCP and temperature were less sensitive to {{ }} for the liquid break LOCAs than the steam breaks at the PZR top. The liquid entrainment is precluded in the core region as the core remains uncovered and there is no reflooding phase. The secondary line breaks that are essentially non-limiting, are less sensitive to {{ }} than the primary side breaks due to the smaller volume and smaller pressure of secondary systems. Based on the consistency between the NPM-160 and NPM-20 design PIRTs with respect to PCP and temperature calculations, and the similarity of the thermal-hydraulics of the liquid entrainment phenomenon inside the CNV, the staff concludes a conservative modeling approach for liquid entrainment has been used.

3. New FSAR Table 6.2-2 markup (ML24215A142) documents T-ave that is a parameter that is controlled at a lower nominal value of 540 °F for NPM-20 than for NPM-160. The staff investigated why, despite the 56 percent core power increase, the primary T-ave value of 545°F used for NPM-20 is 10°F lower than 555°F used for NPM-160 in the DCA

containment safety analyses. A cooler average initial RCS temperature appeared inconsistent with the FSAR Figure 4.4-2 that shows an increase in the analytical design operating limits in the SDAA with Thot increasing from 590°F to 605°F and the High [Thot] Temperature Analytical Limit increasing from 610°F to 620°F, when compared with DCA Figure 4.4-9. As Thot is a function of the core power level and the RCS flow rate, the higher Thot and higher Thot analytical limit are consistent with the design power increase. However, the staff found that the lower T-ave value for NPM-20 reflects its desired nominal operating condition maintained for the US460 design by other selected design parameters. In addition, per new FSAR Table 6.2-2 markup, the NuScale containment safety analysis conservatively applies an additional 5 °F margin to the 540 °F T-ave value for using a 545 °F T-ave to model the NPM-20 limiting M&E release that is consistent with the uncertainty and deadband used in the FSAR Chapter 15 analyses. Therefore, the staff concludes that a credible explanation exists for the apparent inverse trend between the RCS T-ave and the plant operating temperature envelope.

6.2.1.3.5 Conclusion

The staff finds that NuScale has fully addressed the required information related to primary coolant system M&E release calculations for the design, and based on the above evaluation, meets GDC 50 with respect to the calculation of M&E release related to primary system breaks.

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

6.2.1.4.1 Introduction

Similar to a LOCA, the rupture of a secondary system pipe can result in an M&E release into the containment. 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 two 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.

6.2.1.4.2 Summary of Application

FSAR: FSAR, 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 FSAR, Chapter 15.

FSAR, Section 6.2.1.4, "Mass and Energy Release Analysis for Secondary System Pipe Ruptures Inside Containment," 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 described in FSAR, Section 6.2.1.4, 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 FSAR, Section 6.2.1.4.3, which provides high level information related to the initial and boundary conditions considered for the secondary system release analyses. Similar to the primary side M&E release conditions discussed in the previous section of this report, FSAR, Section 6.2.1.4, provides only a high level description of the system response. The applicant's information in EC-A013-7725 that the staff reviewed to make its findings about the secondary systems M&E release is discussed in further detail below.

ITAAC: ITAAC associated with the containment and the ECCS system are provided in SDAA Part 8, Rev. 1, Table 2.1-1, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria," and Table 2.1-2, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria Additional Information." These ITAAC are evaluated in Section 14.3 of this SER.

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

Topical Reports: LOCA EM TR, Revision 3 provides a detailed description of the evaluation model used to produce the M&E release values for the containment DBEs. The containment model as used in EC-A013-7725 as well as the FSAR Chapter 6 containment M&E release events and the CNV pressure and temperature response is a modification of the NRELAP model used for the Chapter 15 transient analyses as described in LOCA EM TR, Revision 3. Qualification of the NRELAP5 model is demonstrated in LOCA EM TR, Revision 3, which relies on NRELAP5's adequacy to model the phenomena present during a LOCA within a NuScale module. As part of the LOCA EM TR, Revision 3, the applicant has provided a full accounting of the modifications made to the model to incorporate the secondary side components, which include modeling choices made in non-LOCA TR, Revision 4.

Additionally, LOCA TR, Revision 3 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 conservative 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 the plots of the FOM for each break.

6.2.1.4.3 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, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," 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 Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," 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 Section 6.2.1.4 and NuScale DSRS Section 6.2.1.4.

6.2.1.4.4 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 FSAR, Section 6.2.1.4 and a more detailed discussion of the inputs, methodology, and associated modeling nuances in LOCA EM TR, Revision 3.

The approach chosen by NuScale for calculating containment conditions following a high -energy line break differs slightly from the traditional approach, in that NuScale used 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 M&E release 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 is evaluated separately as part of the NRC staff's LOCA topical report evaluation and non--LOCA topical report evaluation. NuScale used both a base LOCA transient model and base non--LOCA transient model to perform evaluations for the design -basis transient calculations. The CRAM 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.

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 LOCA EM TR, Revision 3, the applicant stated that the secondary system inventory and distribution are determined in the NRELAP5 steady-state model balance based on the primary side conditions. As a result, the applicant stated 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. As opposed to NPM-160, DHRS heat removal is credited in the NPM-20 CNV analysis, consistent with the NPM-20 CRAM described in the LOCA TR, Revision 3. {{

}}. 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. For the FWLB, the DHRS plays a substantially larger role, as the transient takes place over hours. 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 had performed sensitivity studies to examine the effect of initial pressurizer pressure on the transient. For the MSLB, reducing initial pressurizer pressure had 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 PZR 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 PZR 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 CNV (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. EC-A013-7725, (NPM-20 CNV Pressure and Temperature Response Analysis) shows that most limiting MSLB with maximum containment pressure (888 psia) and temperature (526 °F) case occurs when the MSLB occurs with the loss of AC and EDAS powers with no concurrent failure. This leads to the greatest inventory of secondary-system release caused by the immediate opening of the two RVV at the very start of the initiating event and actuating the two RRVs to open as soon as the pressure differential drops below the specified IAB setpoint. As the primary source of M&E for an FWLB is the RCS, other inputs were selected to minimize the time to ECCS actuation. To drive the limiting M&E release from the RCS for an FWLB, initial PZR pressure is minimized. However, because of the lower enthalpy of the feedwater, the FWLB limiting event also requires an ECCS actuation to yield the peak containment pressure (874 psia) and temperature (524 °F). This includes the assumption of a loss of AC power and loss of EDAS coincident with the event initiation.

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 LOCA caused by RCS (CVCS) discharge line break from downcomer with the loss of normal AC power is the limiting primary side release, 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 (RRV or RVV) or a single RRV and RVV at the same time to open. As discussed in Section 5.4.3 of this SER, 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. EC-A013-7725 also investigated the sensitivity of the MSLB and FWLB containment response to the MSIV and FWIV single failures and found them non-limiting. The staff therefore finds that single failures have been appropriately considered for the secondary -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

The applicant stated that it used Moody's critical flow model for two-phase flow, and {{ }}. 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. {{ }}, 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, at the DCA stage, the applicant discussed the effect of entrainment of droplets in the secondary systems release fluid. The applicant stated that a sensitivity study on entrainment in the break flow in the case of MSLB indicated 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 {{

}}. The containment wall temperature is modeled in NRELAP5 and remains below the design temperature of 600 degrees F. Likewise, as documented in the containment safety analysis section, {{

}} in NRELAP5 was also not safety significant. 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.

6.2.1.4.5 Conclusion

The staff finds that NuScale has fully addressed the required information related to secondary -side break M&E release calculations within the containment for the design, and based on the above evaluation, meets GDC 50 with respect to the calculation of M&E release related to secondary -side breaks.

6.2.2 Containment Heat Removal

6.2.2.1 Introduction

To maintain containment integrity and conform with the requirements of PDC 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 5 other NPM-20 modules for each US460 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

FSAR: FSAR, Section 6.2.2, “Containment Heat Removal,” 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 RVVs and subsequently RRVs 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 stated that there is sufficient inventory in the UHS to remove heat from the containment for at least 30 days. The UHS is discussed in FSAR, 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 (CES), which is discussed in FSAR, Section 9.3.6, “Containment Evacuation System and Containment Flooding and Drain System.

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

Technical Specifications: TS associated with containment heat removal are in SDAA Part 4, Revision 1, Volume 1, TS Section 3.5.3, “Ultimate Heat Sink,” and TS Section 3.6.1, “Containment.” TS 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 TS Section 3.6.1 requires containment be operable (ensuring inventory control within the containment).

Topical Reports: LOCA EM TR, Revision 3 describes the containment response behavior, including containment heat removal, for a selection of the NPM-20 design basis transients. This TR is referenced in FSAR Chapter 6 and discussed further in Sections 6.2.1.1.A, 6.2.1.3, and 6.2.1.4 of this report. However, EC-A013-7725 (NPM-20 CNV Pressure and Temperature Response Analysis) also provides further modeling details and descriptions on the NPM-20 design basis transients.

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,” Revision 5, and are summarized below:

- GDC 38, which requires the following:
  - the CHRS is capable of rapidly reducing the containment pressure and temperature following a LOCA and to maintain these parameters at acceptably low levels

– the CHRS performs in a manner consistent with the function of other systems

– the safety-grade design of the CHRS 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 leak tight 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 can be found in the staff’s evaluation of TR-0815-16497.

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 52.7, “Specific Exemptions,” which states the following:

> 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 public health and safety, and are consistent with the common defense and security

  (2) 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, “Containment Heat Removal Systems,” 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, the following documents provide guidance associated with acceptance criteria that confirm that the above requirements have been adequately addressed:

- Nuclear Energy Institute (NEI) 04-07, Volume 1, “Pressurized Water Reactor Sump Performance Evaluation Methodology,” Revision 0, issued December 2004 (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 (ML050550156)
- Rudland, W.H., NRC, “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 (ML100960495)
- NRC Staff, “NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation dated March 28, 2008 (ML080230462)
- 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 PDC 38 is discussed in Section 6.2.1.1.4.1 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. DCA Part 2, Tier 2, section 6.2.2.1 states the following:

> The passive cooling of a CNV does not include or require active components to perform the containment heat removal function. The CNV provides a large heat transfer surface with no active components needed for heat removal to the UHS water. Reference 6.2-1 and Reference 6.2-3 describe testing of the passive containment heat removal function for LOCA conditions. The design supports an exemption from GDC 40.

In its exemption request (SDAA Part 7, "Exemptions," section 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 ensured by the materials and physical configuration of the NPM being partially submerged in the reactor pool, which functions as the ultimate heat sink. The applicant stated that this configuration directly removes heat from containment without a CHRS and that containment heat removal is performed without reliance on electrical power, valve actuation, cooling water flow, or other active system or component operations.

6.2.2.4.1 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, "Licenses, Certifications, and Approvals for Nuclear Power Plants." 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.

The staff considered NuScale's exemption request and determined that this exemption, if shown to be applicable and properly supported in a request for exemption by a CP or COL applicant that references the SDA, would be justified and could be issued to the CP or COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

Authorized by Law

The NRC staff has determined that granting of the licensee's proposed exemption would not result in a violation of the Atomic Energy Act of 1954, as amended (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 would 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 the 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

Underlying Purpose of the Rule

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(ii), are present whenever application of the regulation in the circumstances would not serve or is not necessary to achieve the underlying purpose of the rule. For the NuScale design, containment heat removal for DBEs is achieved through heat transfer from the CNV to the reactor pool. 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 are in FSAR, Section 6.2.2. FSAR, 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."

As described in FSAR, 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 leak tight containment, which ensures sufficient inventory is available for cooling the post-transient heat load. These functions are addressed by other measures described and evaluated elsewhere in this report: GDC 36, "Inspection of Emergency Core Cooling System," and GDC 37, "Testing of Emergency Core Cooling System," for ECCS, in section 6.3; GDC 16, "Containment Design," and GDC 51, "Fracture Prevention of Containment Pressure Boundary," in section 6.2.7 of this report; and GDC 53, "Provisions for Containment Testing and Inspection," for containment integrity, in section 6.2.6.

Accordingly, the NRC staff determined that the proposed CHRS design meets the underlying purpose of the rule, in this case to verify that the performance characteristics of the CHRS remain with acceptable parameters and to ensure operability.

Undue Hardship

Special circumstances, in accordance with 10 CFR 50.12(a)(2)(iii), are present whenever compliance would result in undue hardship or other costs that are significantly more than those incurred by others similarly situated. The staff recognizes that the NuScale containment design presents challenges in testing the integral containment heat removal function. The primary parameters driving the value of the heat transfer are related to the condensation inside the CNV, which is impractical to test after the fuel is loaded using the as-built configuration, as doing so involves a transient, and the through-wall conduction and convection to the reactor *pool that are difficult to measure empirically. 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. As stated above, other parameters impacting the containment heat removal function, such as fouling, will be inspected. The testing requirements were written for large containments with active cooling systems, such as containment sprays, rather than passive conduction and convection driven heat transfer mechanisms; even for the spray systems, the convective efficacy of the spray system is confirmed through analysis, not periodic testing. Accordingly, the staff determined that testing of the containment heat removal function directly would present an undue burden as the design differences between the NuScale design and the traditional LWR CHRSs make such testing impractical, and the provisions taken by the applicant are sufficiently effective to ensure the functionality of the CHRS.*

Conclusion

The staff concludes that the requested exemption would not impact the consequences of a DBE, 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, would not present an undue risk to public health and safety, and is consistent with the common defense and security. The NRC staff has also determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) and (iii) are present, as the NuScale design meets the underlying purpose of GDC 40, and applying the existing testing requirement could represent an undue hardship for the NuScale design. 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 containment heat removal function without the need for testing.

The staff therefore concluded that an exemption from GDC 40, if shown to be applicable and properly supported in a request for exemption by a CP or COL applicant that references the SDA, would be justified and could be issued to the CP or COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the CP or COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

6.2.2.4.2 GSI-191 & Long-term Cooling Requirements

6.2.2.4.2.1 A Summary of the NPM-160 Compliance with GSI-191 & Long-term Cooling Requirements for the NuScale DCA (US600 Design)

This section documents the relevant DCA review summary of the NPM-160 compliance with GSI-191 & long-term cooling requirements for the NuScale US600 Design for the DCA for reference purposes, while the following section focuses on the GSI-191 related changes made in the NPM-20 design for US460 design that could potentially affect the DCA findings about the NPM compliance with the GSI-191 and long-term cooling requirements. The staff chose to preserve the following material in the present SER for the details that would be applicable to both the DCA (US600) and SDAA (US460).

For an efficient review, the staff compared the GSI-191 and long-term cooling information in the approved DCA FSAR and the SDAA FSAR. The staff focused on the deviations between the approved DCA and SDAA FSARs information about the GSI-191 and long-term cooling, with the assumption that the information that had been approved in the DCA but stayed the same in the SDAA would be still acceptable. So, just focusing on comparing the modified parts of the SDAA FSAR with the corresponding approved DCA parts greatly helped develop insights into the safety-significance of the SDAA changes. However, necessary analyses and meeting the acceptance criteria were still relied upon in making the safety findings for the SDAA.

Long-Term Cooling

To comply with the 10 CFR 50.46(b)(5) requirements for long term cooling, the applicant had performed an evaluation of the impact of debris on long term cooling events for NPM-160, summarized in DCA Part 2, Tier 2, Section 6.3.3.1, "Debris Generation and Impact Evaluation." Because there are no pumps, this evaluation takes a different form than is typical, as referenced in the NEI 0407 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.

The NRC staff had 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 and identified in the FSAR, provide assurance that debris generated material will remain within the bounds assessed in the analysis.

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}}. Further, in DCA Part 2, Tier 2, Section 6.3.3.1, the applicant stated limits of 7.5 gm/FA (0.61 lbm total for all FAs) for total fiber and 30 lbm for total particulate. The total particulate limit includes chemical precipitates, as explained below in Section 6.2.2.4.1. The applicant also stated that analyses demonstrate acceptable core cooling up to higher limits.

The NRC staff agreed that the method used to calculate in-containment debris is reasonable 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.

Chemical Effects Introduction

To determine the compliance of the NuScale design with the requirements of GDC 35, PDC 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, issued November 2003; NEI 0407; and the staff's SER of NEI 0407 for PWR sump debris evaluations. The NuScale design conforms to Revision 4 of RG 1.82, which contains the following guidance for PWRs:

Chemical Reaction Effects

a. 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).

b. 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).

c. 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, the applicant described 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, "Engineered Safety Feature Materials," 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, 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.

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, temperature). 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 NRC staff review guidance regarding Generic Letter 2004 02 Closure in the area of coatings evaluation (ML080230462) 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 March 2008 guidance).

DCA Part 2, Tier 2, Section 6.3.3.1 describes estimates and design limits for the post-LOCA latent debris. {{

}}. Given the design approach of preventing chemical effects, the staff finds this large, assumed quantity of chemical precipitate acceptable for the analysis.

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 precipitates beyond the design limit were included in the analysis.

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.

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 are 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 precipitates that may form are typically soft, nonabrasive, low shear, and readily stay in solution because of the flow conditions present within the system; they can be treated like particulates. The applicant also specified the size of quantity of the debris as follows. {{

}}. 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 (Revision 4).

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 PDC 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 described the quantity and type of material in the post-LOCA fluid as latent particulates with a limit of 30 lbm combined particulate and chemical 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 PDC 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 are 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 (Revision 4) 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 (Revision 4) and meets the regulatory requirements in PDC 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.

6.2.2.4.2.2 *NPM-20 Compliance with GSI-191 & Long-term Cooling Requirements for the US600 to US460 Design Changes*

FSAR, Section 6.2.2.2, “System Design,” states that FSAR, Section 6.3.2.4, “System Reliability,” describes conformance with RG 1.82 (Revision 4) and the approach used to address GSI-191 and assessment of debris accumulation on pressurized water reactor sump performance. FSAR, Section 6.2.2.2, “System Design”, also states that protective coatings are not used or allowed in the CNV, and therefore the effects of post-accident debris generated by coatings are precluded in the design. FSAR, Section 6.2.2.3, “Design Evaluation,” adds that Section 6.3.2.4 describes the generation of post-accident debris, debris transport, and downstream effects considered in the design.

FSAR, 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 two RVVs mounted on the upper head of the RPV and two RRVs that are mounted on the side of the RPV.

The staff review of the NPM-20 compliance with GSI-191 and long-term cooling requirements with respect to the containment functional design and heat removal system is described in the following paragraphs.

FSAR, Section 6.2.2 offers only a few qualitative statements about GSI-191 essentially referring to the Section 6.3 contents. Therefore, the staff reviewed Section 6.3.2.4, “System Reliability,” and Section 6.3.3.1, “Debris Generation and Impact Evaluation,” and identified several thermal-mechanical gaps b/w US460 and US600 designs but no chemical gap was identified. To determine the compliance of the NPM-20 design with the requirements of GDC 35, PDC 38, and 10 CFR 50.46(b)(5), the staff investigated the changes made for the US460 design with respect to the US600 design that could impact the DCA findings about the NPM compliance with GSI-191 and long-term cooling requirements.

The staff needed to establish how the debris limits address the GSI-191 and long term cooling for NPM-20 and whether the latent debris has any significant impact on core cooling during ECCS operation. FSAR, Section 6.3.2.4 reports the analysis to evaluate the impact on ECCS and long term core cooling operation of the generation of post-LOCA or high energy line break debris and the presence of latent debris. The staff audited the underlying analysis and assumptions to determine how the post-accident debris effect was evaluated for NPM-20 along with the debris transport and downstream effects in the containment system (CNTS) design, and whether the CNTS design complied with the regulatory positions of RG 1.82 and addresses the generic safety issue GSI-191.

In the following is summarized the evaluation and conclusions drawn by the staff from its audit of the GSI-191 assumptions and design parameters for NPM-20 (US460) vis-à-vis NPM-160 (US600).

Safety Significance and Justification of the GSI-191 & LTC Related Differences b/w the NPM-20 (US460) and NPM-160 (US600) Designs

a) FSAR, Section 6.3.2.2 describes that the venturi used in the inlet of each RVV and RRV between the RPV and the RRV/RVV in the NPM-20 design maintains margins for precluding the potential for flow blockage due to debris. The staff determined that large debris are not present in the CNV as thermal insulation and coatings are not used and foreign material exclusion and debris removal is performed during the CNV closeout. There is no credible mechanism for potential debris transport to the RVVs along with the vapor boiling off the reactor coolant inside the RPV. Any potential latent fiber, particulate, and chemical debris in the containment are not large enough to block the RRV venturis. In addition, the fluid velocity through the RRVs following the ECCS actuation is above the settling velocity for the latent debris. Even if some latent debris drops out as flow decreases with time, it will deposit in lower spaces of CNV and RPV leaving the RRV flow path unaffected during the long term cooling.

b) The staff identified that FSAR, Section 6.3.2.2 does not specify the maximum flow coefficient for the RVV-diffuser unit, while the DCA does specify a maximum value. Likewise, the maximum terminal pressure drop ratio is not specified in the FSAR,, while DCA specifies a maximum value. It was found that maximum flow coefficient values are not relied upon as US460 design credits the actual ECCS valve venturis to limit the flow and depressurization rate in early stages of inadvertent ECCS valve opening events. Venturis were not present in the US600 certified design. The staff audited a document with more details on the ECCS valve venturis and accepts that the actual physical geometry of the venturi will dictate the fluid flow characteristics. It was also found that the pressure drop across the debris bed for the NPM-20 core is small enough not to affect the system flow and, therefore, sufficient heat removal maintains the fuel clad temperature below 800 ^{o}F.

c) The staff noted that numerous analysis assumptions made in the FSAR Section 6.3.3.1 about the debris generation, mixing, transport, accumulation, and buildup were not included in the FSAR. In addition, the diameters, densities, and maximum latent debris fractions of fiber and particulate components specified in the DCA were not included the FSAR. {{

}}. The staff performed an audit of the NPM-20 GSI-191 debris loading evaluation at uprated conditions that lists the debris amounts for the US460 design. FSAR, Section 6.3.3.1 discusses debris generation and transport. The staff concluded that the audited analyses demonstrate adequate design margin with respect to the defined acceptance criteria for the debris estimates. Adequate core cooling is ensured up to the debris levels of 7.5 gm/FA (fiber), 30 lbm (particulate) and 194.5 lbm (aluminum), as established by the FSAR, as the debris limits.

d) FSAR Section 6.3.3.1 specifies a debris deposition limit on any fuel rod of 50 mils that would correspond to a pin-to-pin distance of 122 mils to preclude debris deposition on two adjacent fuel rods from touching. No such minimum clearance between adjacent fuel rods is specified in the FSAR,. In addition, DCA limits the core inlet pressure drop due to debris accumulation to less than 0.1 psid, while the FSAR, does not specify any pressure drop acceptance criteria for the debris bed. {{

}}. FSAR, Section 6.3.3.1 also documents that the pin-to-pin distance provides conservative margin to debris deposition. The same allowable pressure drop limit (< 0.1 psid) across the US600 reactor core also applies to the US460 SDAA evaluation of debris effects.

e) The DCA evaluation of the time to introduce all debris from CNV into the RPV conservatively assumes an initial 35 lbm debris mass, of which half is delivered to the core in about 45 minutes. The FSAR includes no comparable information for the NPM-20 design. A staff audit of the assumptions and calculations of the debris transport from the containment to the reactor core showed the same debris amount to be homogeneously mixed in the containment liquid volume, and a conservative constant flow rate from the containment to the RPV. NuScale provided a revised FSAR, markup to correct the time needed to deliver 99 percent debris to the core, from five hours to 10 hours.

f) The DCA states that, “the temperature at the center of the blockage is calculated to reach 592 degrees F which remains below the acceptance criteria of 800 degrees F,” while the FSAR, states that, “the peak cladding temperature is below the acceptance criteria of 800 degrees F.” The staff found the calculated temperature at the axial center of the blockage for the US460 design is more conservative than 592°F calculated for the DCA and still below the acceptance criterion of 800 °F.

g) The DCA specifies 30 percent debris in suspension, 0.085 percent fiber concentration, 944 lbm of particulate and chemical species, and 2.86 lbm (1297 g) total debris mass as conservative design limits to not affect the core heat transfer. The staff noted that the FSAR, prescribes none of these limits for the NPM-20 design. In this regard, the staff confirmed the following information for NPM-20 (US600).

    i. The particulate and/or chemical species loading that can be tolerated in the core and would not affect the core heat transfer is higher than the one used in the DCA and thus is more conservative.

    ii. The fiber concentration computed for the FSAR is fairly comparable to the one used in the DCA. Consistent use of the amounts of fibrous materials as well as combined particulate and chemical materials was ensured in various GSI-191 analyses audited. The resulting flows and pressures are conservative with respect to the accident analysis, and result in less than 0.1 psid pressure drop across the debris bed.

    iii. The total debris mass used in NPM-20 is more conservative than the one used in the DCA. {{

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Based on its evaluation, the staff concludes that the NPM-20 design possesses adequate design margins for core cooling and meets the defined acceptance criteria.

Potential Contribution of the NPM-20 Containment Component Materials to the Post-accident Debris Load

The staff audited information to establish whether the components located within the NPM-20 containment contribute to the post-accident debris load and whether they are permitted to use fibrous or organic insulation materials. The staff also looked into whether the materials selected for the components within containment take into consideration the anticipated water chemistry conditions, and whether the cables installed inside the containment are specifically designed not to contribute to debris loading under the anticipated accident conditions. The staff’s audit observations are:

a) Containment inner surfaces are made up of stainless steel. There is no insulation (metallic or non-metallic) or coatings used within the NPM-20 CNV. Latent debris (defined as unintended dirt, dust, paint chips, fibers, pieces of paper, plastic, tape, etc.) is the expected source of debris in the CNV and would have both fibrous and particulate constituents.

b) Materials in the CNV are selected to be compatible with the chemical conditions in which they exist. So, their selection takes into consideration the anticipated water chemistry conditions.

c) Cables installed in the CNV are designed to withstand the postulated accident conditions (e.g., environmental, seismic, dynamic loads). Cables that run through the CNV are unpainted, corrosion resistant, seamless, Type 304L stainless steel jacketed, mineral insulating cabling. As discussed in FSAR, Section 6.1.2, the cable material is free of organic material in the insulation and sheath. So the cables installed inside the containment are designed not to contribute to debris loading under the anticipated accident conditions.

d) No new debris sources or chemical products are identified in the NPM-20 debris analysis beyond the DCA that has already been reviewed and approved.

Summarily, the staff determines that the CNV and RPV space volumes and fluid and debris masses are fairly comparable b/w NPM-20 and NPM-160. Therefore, the overall debris loading as well as the resulting pressure drops across the debris bed are also comparable and have ample margin with respect to the acceptance criteria, as was the case in the DCA. The staff accepts the technical bases for precluding the potential for flow blockage of the RRVs, RVVs, and associated venturis with reduced areas caused by any latent debris and their adverse impact on the ECCS flow. The staff concludes that the NPM-20 containment system design complies with the generic safety issue GSI-191 and long term cooling regulatory requirements.

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.

Table 6.2-2 NuScale COL Information Items for Section 6.2.2

Item No.	Description	FSAR Section
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 PDC 38 and GDC 39 with regard to the functional design of the containment are met. The staff evaluation of the applicant’s demonstration of adequate containment heat removal to meet PDC 38 is discussed in Section 6.2.1.1.A of this report. Further, the applicant has provided a rationale that would justify an exemption to GDC 40 based on the nature of the design and associated demonstrated compliance with GDC 39, if a COL applicant references the NuScale US460 SDA, shows that the exemption is applicable and properly supported, that the design remains unchanged in all aspects material to the exemption, and that there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

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 post-accident 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

The CIS protects against the release of radioactive material to the environment as a result of an accident by ensuring the leakage through containment isolation valves (CIVs) and passive containment isolation barriers are within the acceptance criteria. In addition, there is automatic actuation and closure of the CIVs when specific defined limits for process variables are exceeded. Periodic in-service inspection and testing of these containment isolation components maintains both of these capabilities.

FSAR Table 6.2-3 provides a list of the containment penetrations and FSAR Table 6.2-4 provides a list of the CIVs along with its open or closed position for normal and accident conditions.

The CIVs consist of primary system containment isolation valves (PSCIVs) and secondary system containment isolation valves (SSCIVs). The PSCIVs have two valves in series on each line through containment, and the pair of valves meet the intent of GDC 55 and GDC 56. The SSCIVs have a single valve on each line (and bypass line) and meet GDC 57.

NuScale has requested an exemption from specific requirements of GDCs 55, 56, and 57 to several containment penetrations. This exemption request is in SDAA Part 7, “Exemptions,” Revision 0, “10 CFR 50, App. A, GDC 55, 56, and 57 Containment Isolation,” issued December 31, 2022 (ADAMS Accession No. ML22365A018).

ITAAC: In SDAA Part 8, Table 2.1-1, the applicant listed the ITAAC associated with containment isolation equipment. SDAA Part 8, Table 2.1-1, includes ITAAC # 8 for CIV closure time and ITAAC # 9 for the length of piping between each penetration and its associated outboard CIV. These ITAAC are evaluated in Section 14.3 of this SER.

Initial Test Program: In FSAR Section 14.2, "Initial Plant Test Program," Table 14.2-38, "Containment System Test # 38," the applicant described the testing related to the containment system, to include the CIS.

Technical Specifications: In NuScale SDAA Part 4, Section 3.6.2, "Containment Isolation Valves," the applicant described the technical specifications for the CIVs. These TS provide LCOs and surveillance requirements (SRs) 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 TS and associated bases are located in Chapter 16 of this SER.

Technical Reports: None.

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 to 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, "Sharing of Structures, Systems, and Components," requires SSCs important to safety not to 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, "Containment Design," 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:

- one locked-closed isolation valve inside and one outside containment
- one automatic isolation valve inside and one locked-closed isolation valve outside containment
- one locked-closed isolation valve inside and one automatic isolation valve outside containment
- 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.

- 10 CFR 52.47(a)(8) relates to demonstrating compliance with any technically relevant portions of the 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).

- 10 CFR 52.139, “Standards for Review of Applications,” requires, in part, that applications filed under this subpart will be reviewed for compliance with the standards set out in 10 CFR Part 50 and its appendices.

  - 10 CFR 50.63(a)(2) relates to ensuring that appropriate containment integrity is maintained in the event of an SBO for a specified duration.

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 SRP sections. The following NRC regulatory guides are also applicable for this review:

- RG 1.141, “Containment Isolation Provisions for Fluid Systems,” Revision 1
- RG 1.155, “Station Blackout,” Revision 0

6.2.4.4 Technical Evaluation

The staff reviewed the CIS described in FSAR Section 6.2.4, using the staff’s review guidance provided in NuScale DSRS Section 6.2.4. 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, 56, and 57, as described in FSAR Section 6.2.4. The design also departs from 10 CFR 50.34(f)(2)(xiv)(E), as described in FSAR Section 9.3.6. For each departure discussed above, the applicant seeks an exemption. The application’s exemption requests reside in SDAA Part 7.

In general, GDC 55, 56, and 57 require each line that penetrates primary reactor containment to be provided with one or more CIVs. In GDC 55 and 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., DHRS) as the isolation barrier outside containment instead of the specified CIV.

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

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

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

6.2.4.4.1 GDC 1 and GDC 2

The provisions in GDC 1 and 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.

The staff based its review of CIS compliance with GDC 1 and 2 requirements, on adherence to 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."

In FSAR Section 6.2.4, the applicant stated 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 FSAR Table 3.2-1, "Classification of Structures, Systems, and Components," and FSAR Section 6.2.4, CIVs are designed to meet seismic Category 1 requirements to satisfy the guidance specified in RG 1.29. Furthermore, FSAR 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 SDAA conforms to regulatory guidance, the staff finds the CIVs meet the requirements of GDC 1 and 2.

6.2.4.4.2 GDC 4

The provisions in GDC 4 are applicable to the review of the CIS 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).

In FSAR Section 6.2.4, the applicant stated that the CIS meets the requirements of GDC 4. Specifically, the CIVs and barriers are designed to accommodate the effects of and are compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (including LOCAs). In addition, these containment isolation components are protected against dynamic effects of missiles and pipe whip and discharging fluids that result from in-plant equipment failure or from events and conditions external to the facility. The applicant described missile protection in FSAR Section 3.5, “Missile Protection”; protection against dynamic effects in FSAR Section 3.6, “Protection against Dynamic Effects Associated with Postulated Rupture of Piping”; and environmental conditions in FSAR 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 FSAR 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 SER).

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

6.2.4.4.3 GDC 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.

To satisfy GDC 5, the staff’s review of the CIS is to make sure that the CIVs are essentially independent from the other units.

In FSAR Section 6.2.4, the applicant stated that the CIVs and other containment barriers are independent to one NPM and do not function for the other NPMs at NuScale Power Plant. The staff reviewed the FSAR information described above and confirmed that the CIVs and 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.

6.2.4.4.4 GDC 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 CIS. Specifically, the CIS 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 FSAR Section 6.2.4, the applicant stated that the containment isolation components (valves and barriers) 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 (i.e., GDC 1, 2, and 4) in this section and function to prevent the release of radioactive materials to the environment. In addition, FSAR 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 sections of this report) that are designed to support providing an essentially leak tight barrier. The CIS is also designed to support containment leakage testing in order to verify the leak tight integrity of the CIS. Furthermore, the design, qualification, and testing of the isolation valves is addressed in FSAR 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 SER.

In FSAR Section 6.2.4, the applicant stated that the isolation function is accomplished given the occurrence of a single active failure in the isolation provisions. Specifically, the applicant provided FSAR Table 6.2-5, “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 FSAR 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 SER. The design of the steel containment, to include a discussion of these passive containment isolation barriers, is described in FSAR Section 3.8.2, “Steel Containment,” and is evaluated in Section 3.8.2 of this SER.

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

6.2.4.4.5 GDC 54

The provisions in GDC 54 require piping systems that penetrate the primary reactor containment to have leak detection, isolation, and containment capabilities with 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 CIS 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 below (i.e., GDC 55, 56, and 57 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 FSAR Section 6.2.4, the applicant described the parameters sensed for the initiation of a containment isolation signal from the engineered safety feature actuation system (ESFAS) (see FSAR Table 7.1-4, “Engineered Safety Feature Actuation System Functions”). The staff finds that the diversity of parameters that are sensed for isolating containment (e.g., high containment pressure, low pressurizer level, low-low pressurizer level, low AC voltage to the battery chargers, and high under-the-bioshield temperature) is sufficient to ensure timely isolation of all nonessential piping penetrations 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 FSAR 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 that the design meets these GDC 54 requirements.

In FSAR 5.2.5.1, "Leakage Detection and Monitoring," the applicant described how leakage is monitored using pressure, level, and radioactivity instrumentation. Coupled with the methods available to detect intersystem leakage described in FSAR Section 5.2.5.4, "Chemical and Volume Control System Intersystem Leakage Monitoring," this 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 that 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 FSAR Table 6.2-5. As identified in the table, the normally open CIVs close in less than or equal to 10 seconds, assuming signal delay and valve stroke times. The staff finds that 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 CES (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., CES isolation valves) is evaluated in Section 15.0.3 of this report. In addition, the applicant's evaluation model for the ECCS performance accounts for containment isolation closure times, consistent with the isolation times presented in the FSAR Table 6.25. ECCS performance is evaluated in Section 6.3 and Chapter 15 of this report. Overall, the staff finds the valve performance capabilities reflect the safety importance of isolating these piping systems.

Discussions below related to (1) TMI requirements and (2) TS also address reliable isolation capability and the safety importance of isolating these piping systems.

Based on the discussion above, the staff finds that the CIS meets the requirements of GDC 54.

6.2.4.4.6 GDC 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 FSAR Section 6.2.4, the applicant stated that the NuScale design contains four containment piping penetrations that are subject to GDC 55. These four piping penetrations are associated with RCS injection, RCS pressurizer spray, RCS discharge, and RCS high-point degasification lines. While the applicant provided 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 in SDAA Part 7.

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, “Class 1 Components,” and seismic Category 1 criteria), as described in FSAR Section 6.2.4 and Section 3.1.5.6, “Criterion 55-Reactor Coolant Pressure Boundary Penetrating Containment.” In addition, in FSAR Section 6.2.4, the applicant described 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 SDAA 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 CNV 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, “Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping,” including associated BTP 34, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment” (this characterization is discussed further in Section 3.6.2 of this report). Therefore, a break between the CNV and the isolation valves need not be considered. Although these lines are not part of an ESF system or required for safe shutdown (i.e., considered nonessential), 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 FSAR Section 6.2.4, the applicant described that all GDC 55--related CIVs are welded directly to a containment isolation test fixture valve, which is welded 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 CNV 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 FSAR 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 52.139, 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.

The staff considered NuScale's exemption request and determined that this exemption, if shown to be applicable and properly supported in a request for exemption by a CP or COL applicant that references the SDA, would be justified and could be issued to the CP or COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

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 would 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 staff finds that NuScale's containment isolation provisions meet the underlying purpose of the rule (see special circumstances discussion below), and this exemption does not impact the consequences of any DBE and does not create new accident precursors. As a result, the staff concludes that the proposed exemption request is acceptable in terms of public health and safety. 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 would 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 would 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

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. Therefore, because special circumstances are present, the staff finds that the exemption meets the requirements of 10 CFR 50.12(a)(2).

Conclusion

As discussed above, in accordance with 10 CFR 50.12(a)(1), the staff finds that the requested exemption to GDC 55 is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security. The NRC staff has also determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) are present, as the NuScale design meets the underlying purpose of GDC 55.

Accordingly, an exemption from GDC 55, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale’s SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the CP or COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

6.2.4.4.7 GDC 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 FSAR Section 6.2.4, the applicant stated that the NuScale design contains four containment piping penetrations that are subject to GDC 56. These four piping penetrations are associated with the CES, the CFDS, and cooling lines (i.e., supply and return) for the 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 (FSAR Table 6.2-3) 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 provided 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 in SDAA Part 7.

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 1 criteria), as described in FSAR Section 6.2.4 and Section 3.1.5.7, "Criterion 56-Primary Containment Isolation." In addition, in FSAR Section 6.2.4, the applicant described 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 stated that containment isolation provisions different from the explicit requirements of GDC 56 are acceptable if the differences are justified. The applicant's justification (see SDAA 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 CNV 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 34 (this characterization is discussed further in Section 3.6.2 of this report). Therefore, a break between the CNV and the isolation valves need not be considered. Although these lines are not part of an ESF system or required for safe shutdown (i.e., considered nonessential), 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 FSAR Section 6.2.4, the applicant described that all GDC 56--related CIVs are welded directly to a containment isolation text fixture valve, which is welded to the nozzle via a safe end and, upon receipt of a closure signal or loss of actuating power reposition, are 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 CNV nozzle. The staff also finds that 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 FSAR 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 52.139, 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.

The staff considered NuScale's exemption request and determined that this exemption, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

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 staff finds that NuScale's containment isolation provisions meet the underlying purpose of the rule (see special circumstances discussion below), and this exemption does not impact the consequences of any DBE and does not create new accident precursors. As a result, the staff concludes that the proposed exemption request is acceptable in terms of public health and safety. 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

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. Therefore, because special circumstances are present the staff finds the exemption meets the requirements of 10 CFR 50.12(a)(2).

<u>Conclusion</u>

As discussed above, in accordance with 10 CFR 50.12(a)(1), the staff finds that the requested exemption to GDC 56 is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security. The NRC staff has also determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) are present, as the NuScale design meets the underlying purpose of GDC 56. Accordingly, the staff concluded that an exemption from GDC 56, if shown to be applicable and properly supported in a request for exemption by a CP or COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the CP or COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

6.2.4.4.8   GDC 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 FSAR Section 6.2.4, the applicant stated that the NuScale design contains eight containment piping penetrations that are subject to GDC 57. These eight piping penetrations are associated with the main steam system (two lines), the feedwater system (two lines), and the DHRS (four lines). The applicant provided 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 proposed 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 in SDAA Part 7.

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.

(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 SDAA Part 7, the applicant stated that 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 1, 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 FSAR Chapter 3, "Design of Structures, Systems, Components and Equipment," 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 FWIVs, in FSAR Section 6.2.4, the applicant stated 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 the GDC 1 discussion above), protection against natural phenomena (see the GDC 2 discussion above), and environmental and dynamic effects (see the GDC 4 discussion above). In addition, in FSAR Section 6.2.4, the applicant described 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 SDAA Part 7, the applicant stated 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 1 and Quality Group B standards, and has a design temperature and pressure rating at least equal to that of containment. As described in FSAR Section 5.4.3, "Decay Heat Removal System," the DHRS is a welded design with a design pressure equal to that of the RPV, which greatly exceeds the design pressure for containment. Specifically, the DHRS design pressure is 2,200 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 34 (this characterization is discussed further in Section 3.6.2 of this report)-. Therefore, the applicant proposed 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, will 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 FSAR Section 6.2.4, the applicant stated that, except for the MSIVs, all GDC 57-related CIVs are welded directly to the containment isolation test fixture valve, which is welded to the nozzle safe-end. The MSIVs, as described in FSAR Section 6.2.4, are 4 feet from the CNV. A distance of 4 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 that the isolation valve location information provided in the FSAR 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 52.139, 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 57 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 57.

The staff considered NuScale's exemption request and determined that this exemption, if shown to be applicable and properly supported in a request for exemption by a CP or COL applicant that references the SDA, would be justified and could be issued to the CP or COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the CP or COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

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 staff finds that NuScale's containment isolation provisions meet the underlying purpose of the rule (see special circumstances discussion below), and this exemption does not impact the consequences of any DBE and does not create new accident precursors. As a result, the staff concludes that the proposed exemption request is acceptable in terms of public health and safety. 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

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 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 1, ASME Code, Section III, Class 2, with a design temperature and pressure equal to that of the RPV). 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, because special circumstances are present, the staff finds that the exemption meets the requirements of 10 CFR 50.12(a)(2).

Conclusion

As discussed above, in accordance with 10 CFR 50.12(a)(1), the staff finds that the requested exemption to GDC 57 is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security. The NRC staff has also determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) are present, as the NuScale design meets the underlying purpose of GDC 57. Accordingly, the staff concluded that an exemption from GDC 57, if shown to be applicable and properly supported in a request for exemption by a CP or COL applicant that references the SDA, would be justified and could be issued to the CP or COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the CP or COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

6.2.4.4.9 10 CFR 50.34(f)(2)(xiv)

The regulation in 10 CFR 52.47(a) states, in part, that the SDAA must contain an FSAR that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the SSCs and of the facility, 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)....

The regulation in 10 CFR 50.34(f) states, in part, the following:

> 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…

The regulation in 10 CFR 50.34(f)(2)(xiv) requires the CIS to do the following (II.E.4.2):

(A) Ensure all non-essential systems are isolated automatically by the CIS,

(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 FSAR Table 1.9-5, "Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933)," the applicant indicated 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 nonessential 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 FSAR Table 6.2-4 and concludes that they are consistent with their classifications.

In FSAR Section 6.2.4, the applicant described that CIVs serving in nonessential 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 nonessential systems are isolated automatically by the CIS is satisfied. Additionally, in FSAR 6.2.4, the applicant provided information to conclude that each nonessential penetration has two isolation barriers in series, thereby meeting the requirements of 10 CFR 50.34(f)(2)(xiv)(B). Furthermore, in FSAR 6.2.4, the applicant stated 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 a CIS that utilizes a containment setpoint 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," issued November 1980). As shown in FSAR Table 7.1-4, a containment system isolation actuation signal is initiated on containment pressure. In FSAR Table 1.9-5, the applicant stated that the pressure setpoint is compatible with normal operating pressure (e.g., above the highest allowable containment pressure for leak detection operability given by FSAR Figure 5.2-2), 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 nonessential piping penetrations are isolated before significant containment pressurization occurs or before there is a significant release of radionuclides into containment (see additional discussion related to 10 CFR 50.34(f)(2)(xiv)(E) below), 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 CIS meets the requirements of 10 CFR 50.34(f)(2)(xiv) items (A) through (D).

The remaining discussion in this section 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 TMI, the NRC added requirements to its power reactor safety regulations. In 10 CFR 50.34, "Contents of Applications; Technical

Information," and the *Federal Register* notice 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," issued May 1980, 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 DBEs 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," issued July 1979, the containment is to provide a final barrier to the release of radioactivity in the event of an accident and isolation of nonessential systems penetrating the containment boundary before significant releases from the building is imperative.

In FSAR Section 9.3.6, the applicant indicated that the CES connects the containment atmosphere to the environment and stated that the NuScale design supports an exemption from 10 CFR 50.34(f)(2)(xiv)(E) as applied to the CES.

In SDAA Part 7, Exemption # 13, "10 CFR 50.34(f)(2)(xiv)(E) Containment Evacuation System Isolation," the applicant requested an exemption from 10 CFR 50.34(f)(2)(xiv) item (E) as applied to the CES. The applicant described that 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 stated 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 CNV. Therefore, the applicant concluded 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 SDAA Part 7, Exemption # 13, the applicant described that the NuScale design meets the underlying purpose of the rule by isolating CES using two automatic containment isolation signals: (1) high CNV pressure signal and (2) low-low pressurizer level. The applicant explained that the NuScale design differs from the traditional large LWR 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 stated 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 described 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 SDAA 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 52.139, 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. As described above, the applicant seeks an exemption from the requirements of 10 CFR 50.34(f)(2)(xiv)(E).

The staff considered NuScale's exemption request and determined that this exemption, if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

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 would 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 staff finds that NuScale's containment isolation provisions meet the underlying purpose of the rule (see special circumstances discussion below), and this exemption does not impact the consequences of any DBE and does not create new accident precursors. As a result, the staff concludes that the proposed exemption request is acceptable in terms of public health and safety. 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 would 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 would 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

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. Therefore, because special circumstances are present, the staff finds the exemption meets the requirements of 10 CFR 50.12(a)(2).

<u>Conclusion</u>

As discussed above, in accordance with 10 CFR 50.12(a)(1), the staff finds that the requested exemption to 10 CFR 50.34(f)(2)(xiv)(E) is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security. The NRC staff has also determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) are present, as the NuScale design meets the underlying purpose of 10 CFR 50.34(f)(2)(xiv)(E).

Accordingly, the staff concluded that an exemption from 10 CFR 50.34(f)(2)(xiv)(E), if shown to be applicable and properly supported in a request for exemption by a COL applicant that references the SDA, would be justified and could be issued to the COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

6.2.4.4.10 10 CFR 50.34(f)(2)(xv)

The regulation in 10 CFR 52.137(a) states, in part, that the SDAA must contain an FSAR that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the SSCs 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)....

The regulation in 10 CFR 50.34(f) states, in part, the following:

> 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….

The regulation in 10 CFR 50.34(f)(2)(xv) states the following:

> 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)

TS 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 (BTP 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 FSAR Table 1.9-5, the applicant provided a conformance status associated with TMI requirements to include 10 CFR 50.34(f)(2)(xv). The applicant stated that its 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, does not contain subcompartments 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 greater than or equal to 200 degrees F).

Based on the TS 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).

6.2.4.4.11 10 CFR 50.34(f)(3)(iv)

The regulation in 10 CFR 52.137(a) states, in part, that the SDAA must contain an FSAR that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the SSCs 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)....

The regulation in 10 CFR 50.34(f) states, in part, the following:

> 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….

The regulation in 10 CFR 50.34(f)(3)(iv) states the following:

> 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 FSAR Table 1.9-5, the applicant described conformance with TMI requirement 10 CFR 50.34(f)(3)(iv) as not technically relevant. In addition, in FSAR 1.9-5, the applicant stated that a 3-foot diameter containment opening is not necessary since the NuScale design already addresses severe accident scenarios that could lead to containment failure. In FSAR Section 6.2, the applicant discussed 10 CFR 50.34(f)(3)(iv) as follows:

> The NuScale CNV 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 FSAR 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.

As demonstrated in 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

(2) generation of NCG because of in-vessel cladding oxidation and ex--vessel corium--concrete interaction

Containment venting through an external filter or through a boiling-water reactor 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 NCG 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 FSAR Section 19.2, "Severe Accident Evaluation," show that long -term containment pressures in a severe accident stay below those that could fail the containment. 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).

6.2.4.4.12 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 FSAR Section 6.2.4, position indication for CIVs is provided in the main control room (MCR). In addition, as described in FSAR Chapter 7, position indication for containment isolation is a post-accident monitoring variable (see also FSAR Table 7.1-7, "Summary of Post-accident Monitoring Variables").

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.

6.2.4.4.13 Station Blackout (10 CFR 50.63) and Appropriate Containment Integrity

An SBO means the complete loss of ac electric power to the essential and non-essential 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, 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 an SBO. As described in FSAR 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 an EDAS) is available in the control room for operators to verify valve closure. This arrangement satisfies the guidance provided in RG 1.155, Regulatory 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.

6.2.4.4.14 Reliance on Electrical Power

As described in FSAR 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 LOCA. Because the containment isolation safety function does not rely upon electrical power, this supports findings made in Chapter 8 of this report regarding overall reliance on electrical power for the NuScale plant.

6.2.4.4.15 Overpressure Protection

In FSAR Section 6.2.4, the applicant addressed overpressure protection in the region between two closed primary CIVs and between the FWIV and its associated feedwater check valve because of heat up of fluid between the closed barriers. In particular, in response to increasing pressure between the two primary valves, a thermal relief device integral to the inboard valve relieves fluid pressure back to the CNV. For feedwater, as the fluid heats up and expands, the pressure increase is relieved passively through a small port in the check valve disc. The staff finds the applicant's 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.

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

Based on its review of the information provided by NuScale, the staff concludes that the NuScale FSAR Section 6.2.4 for the CIS design conforms to the acceptance criteria of NuScale DSRS Section 6.2.4, except with respect to the exemptions described above and their associated rationales. Conformance with the criteria in DSRS Section 6.2.4 and justification for exemption requests, as described in this section of the SER, 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, provided that a CP or COL applicant that references the SDA would be justified and could be issued to the CP or COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the CP or COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

6.2.5 Combustible Gas Control in Containment

6.2.5.1 Introduction

Control of combustible gases in containment is described in FSAR section 6.2.5. Following a design basis event, 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 combustible gases are generated by radiolytic decomposition of water during the course of an accident. In some accident scenarios, significant amounts of combustible gases can be generated. The NuScale plant is designed to limit scenarios that could produce a hydrogen combustion event.

6.2.5.2 Summary of Application

The design of the NPM maintains the CNV atmosphere inert to prevent combustion of hydrogen and oxygen. Specifically, the NuScale US460 design credits a single safety related passive autocatalytic recombiner (PAR) for preventing a combustible mixture in the CNV for DBEs, including anticipated operational occurrences (AOOs), that are expected to occur at least once in the lifetime of the plant for non-core damage events. A primary post-accident safety function is to maintain the integrity of the CNV. The PAR is a new design component not included in the US600 DCA design.

FSAR section 6.2.5, details how the applicant addressed the potential buildup of combustible gases in the containment resulting from a fuel clad-coolant interaction up to 100 percent. The design basis condition for the CNV is to maintain its atmosphere inert for at least 72 hours following a design basis accident or a severe accident.

The applicant’s determination that the design controls combustible gas concentrations is based on an analysis of a spectrum of accidents assuming a previously intact containment boundary. In FSAR, section 6.2.5.1, “Combustible Gas Control,” the applicant stated that the containment is adequately mixed for these events and, as described in FSAR Section 19.2, evaluation of a bounding BDBE case that produces more hydrogen than the 100% clad water reaction and determines that the CNV does not exceed its design pressure.

The CNV itself is an ASME Code Class MC pressure vessel that is stamped and constructed in accordance with ASME Code Class 1 MC and is designed to control fission product releases from the reactor coolant pressure boundary, contain the inventory released from a LOCA, and support ECCS operation by acting as a heat transfer conduit to the ultimate heat sink. These design functions include accommodation of the hydrogen generated by a 100 percent fuel clad metal-water reaction.

In SDAA Part 7, Exemption Request Number 2, NuScale requested an exemption from the requirements of 10 CFR 50.34(f)(2)(xvii)(C), which requires hydrogen monitoring in containment.

ITAAC: NuScale SDAA Part 8 includes Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for the PAR: to inspect its physical arrangement and installation; to analyze and test, or a combination of both, to verify it performs its function of recombining hydrogen and oxygen at the minimum recombination rate; and to include the PAR in the scope of equipment qualification ITAAC. These ITAAC are evaluated in section 14.3 of this SER.

Technical Specifications: NuScale SDAA Part 4 includes technical specifications (TS) and associated bases for the PAR. Limiting Condition for Operation (LCO) 3.6.4 ensures that the PAR is available to preclude formation of a combustible atmosphere during either a DBA or significant beyond-DBAs by passively limiting oxygen concentration in the containment. LCO 3.6.4 requires the PAR be operable in Modes 1 and 2. Surveillance Requirements (SR) 3.6.4.1 and 3.6.4.2 require visual inspection and catalyst performance respectively at each refueling.

Technical Reports: There are no technical reports associated with this area of review.

6.2.5.3 Regulatory Basis

The applicant proposed the regulatory applicability for combustible gas control in containment to 10 CFR 50.44(d). The evaluation of combustible gas control against this regulatory basis is discussed below in the technical evaluation section.

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in SRP section 6.2.5, “Combustible Gas Control,” 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.

- 10 CFR 52.137(a)(2), (a)(4), (a)(8), (a)(9), (a)(12), (a)(19), (a)(23), and (a)(25); Part 52 – Standard Design Approvals for Nuclear Power Plants, Contents of Applications; technical information.

- 10 CFR 52.137(a)(12), The application must include an analysis and description of the equipment and systems for combustible gas control as required by § 50.44 of this chapter.

- 10 CFR 50.44(d), Applications subject to this paragraph must include:

  (1) Information addressing whether accidents involving combustible gases are technically relevant for their design, and

  (2) If accidents involving combustible gases are found to be technically relevant, information (including a design-specific probabilistic risk assessment) demonstrating that the safety impacts of combustible gases during design basis and significant beyond-DBAs have been addressed to ensure adequate protection of public health and safety and common defense and security.

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

    (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 SRP section 6.2.5 for a water-cooled plant that does not fall within the description for 10 CFR 50.44(c) is found in RG 1.7, Revision 3. Review interfaces with other SRP sections can also be found in SRP section 6.2.5. The staff notes that NuScale proposed PDC 41, “Containment Atmosphere Cleanup,” to satisfy the requirements associated with GDC 41, which is discussed further below.

6.2.5.4 Technical Evaluation

The applicant proposed the regulatory applicability for combustible gas control in containment as 10 CFR 50.44(d). Although the potential for production of combustible gases from fuel clad oxidation is comparable to traditional LWRs, NuScale emphasized other aspects. The CNV is normally evacuated with insignificant quantities of oxygen. The traditional source of generation of combustible gases, fuel clad oxidation, does not produce oxygen. Thus, the NPM is an oxygen limited design. The sources which produce additional oxygen in containment are releases from a non-core damage LOCA and radiolysis. Due to the small size of the CNV net free volume, oxygen production from radiolysis can lead to a combustible mixture within the time frame of a DBA if no credit is taken for the PAR. The potential for DBAs and for accumulation of combustible gases is different from traditional LWRs. See description of a non-core damage LOCA below.

Staff concurs with the applicability of regulations and guidance to the SDAA US460, as included in:

10 CFR 50.44(d)(2)–If accidents involving combustible gases are found to be technically relevant, information (including a design-specific probabilistic risk assessment) demonstrating that the safety impacts of combustible gases during design basis and significant beyond-DBAs have been addressed to ensure adequate protection of public health and safety and common defense and security.

The guidance in RG 1.7, Revision 3 includes the following regulatory positions with regard to combustible gas control of nuclear power plants::

Regulatory Position C.1, Combustible Gas Control Systems

SSCs installed to mitigate the hazard of combustible gas in containment should be designed to provide reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed. The required system performance criteria will be based on the results of design-specific reviews.

Equipment survivability expectations under severe accident conditions should consider the circumstances of applicable initiating events (such as station blackout or earthquakes) and the environment (including pressure, temperature, and radiation) in which the equipment is relied upon to function.

Regulatory Position C.2.1, Hydrogen Monitors and C.2.2 Oxygen Monitors

The equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen and oxygen in the containment atmosphere following a beyond-DBA for accident management, including emergency planning.

Regulatory Position C.3, Atmosphere Mixing Systems

10 CFR 50.44 requires that all containments have a capability for ensuring a mixed atmosphere. All containment types should have an analysis of the effectiveness of the method used for providing a mixed atmosphere. This analysis should demonstrate that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity.

All containment types should have an analysis of the effectiveness of the method used for providing a mixed atmosphere. This analysis should demonstrate that combustible gases will not accumulate within a compartment or cubicle to form a combustible or detonable mixture that could cause loss of containment integrity.

Regulatory Position C.4, Hydrogen Production

Materials within the containment that would yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable.

Regulatory Position C.5, Containment Structural Integrity

Steel containments meet the requirements of the ASME Boiler and Pressure Vessel Code (edition and addenda as incorporated by reference in 10 CFR 50.55a(b)(1)), Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, considering pressure and dead load alone.

NuScale proposed a PDC (PDC 41) in place of GDC 41 in FSAR section 1.9.2, table 1.9-3, “Conformance with NUREG-0800, Standard Review Plan (SRP) and Design-Specific Review Standard (DSRS).” The proposed PDC is functionally similar to GDC 41, with only the provisions associated with electric power eliminated from the PDC. 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 BDBE which generates hydrogen and oxygen in the CNV for at least 72 hours. The staff finds the proposed PDC 41 meets the regulatory intent of GDC 41 and is acceptable.

Combustible Gas Control

1) Combustible gas control systems. The CNV must have an inerted atmosphere or oxygen concentrations must be limited to less than 4 per cent to maintain inert conditions.

NuScale quantitatively evaluated the combustible gas mixtures produced radiolytically within the RCS during a LOCA with an intact core. The combustible gas mixture was vented to the CNV by timed ECCS actuation. NuScale concluded the containment composition with no credit for the PAR in operation is flammable for all cases within 24 hours. With credit for the PAR, the CNV atmosphere is maintained inert indefinitely. During the regulatory audit (ML24211A089), staff reviewed NuScale’s evaluation and agrees with the conclusions. Staff finds that for crediting the PAR, the NuScale design meets the criterion for an inert CNV. The PAR was conservatively sized based upon the above accident to limit oxygen concentration in containment to less than 4 percent. The minimum oxygen recombination rate of the PAR is 15 moles/hr at a partial pressure of 1.69 kilopascals. To perform its function unimpaired during DBEs and SAs, the PAR is designed to withstand the normal operating environment of 60 years and the post-accident environment. For normal operation the PAR needs to withstand the high radiation environment of both neutrons and gamma radiation to which it is exposed.

The NuScale US460 standard design credits a single PAR for preventing a combustible mixture in the CNV for DBEs, including AOOs, and during a severe accident (SA). NuScale’s analysis demonstrates that the containment cannot be maintained inert without a PAR during DBEs, including AOOs (NuScale’s analysis was for a cooldown transient). NuScale’s analysis of the presence and treatment of combustible gas in the RCS, including an evaluation of the RCS venting into the CNV scenario described above, concluded the containment composition with no credit for the PAR in operation is flammable. The staff has reviewed a confirmatory calculation for the above DBA with more conservative initial assumptions and concluded that the PAR would maintain an inert CNV for 24 hours.

The NPM design controls combustible gases to prevent hydrogen combustion inside containment following a severe accident (SA). In FSAR section 19.2.3.2, NuScale describes modeling potential SA scenarios and identifies the limiting challenges to the RPV and the CNV.

The selected SA scenarios are identified In FSAR table 19.2-1 and represent the spectrum of conditions of potential severe accident phenomena, such as hydrogen generation, that may challenge containment integrity. NuScale evaluated the identified SA scenarios and demonstrated that in all cases for an intact CNV, the oxygen concentration remained below 4% for 72 hours, including consideration for oxygen generation from radiolysis, without crediting the PAR. Based on the NRC staff's review, supported by the information reviewed during the regulatory audit, the NRC staff finds that the NuScale design for controlling combustible gases generated during severe accidents has been met by maintaining an inert environment in the CNV.

The above results were calculated for the SA scenarios identified in FSAR section 19.2, table 19.2-1, "Core Damage Simulations for Severe Accident Scenarios." Each SA scenario was analyzed twice; first with hydrogen generated from clad oxidation but no inclusion from radiolysis, and second with hydrogen from clad oxidation and oxygen from radiolysis. In all cases there was insufficient oxygen, less than 4%, to support combustion. The PAR was not credited with functioning in any of the SA scenarios. Staff finds that the CNV atmosphere is maintained inert for a non-core damage LOCA by crediting the PAR. Staff also finds that following an SA the oxygen is limited and the CNV is maintained inert for 72 hrs.

2) Equipment Survivability

Containments that do not rely upon an inerted atmosphere to control combustible gases must be able to establish and maintain safe shutdown and containment structural integrity with systems and components capable of performing their functions during and after exposure to the environmental conditions created by the burning of hydrogen. As described above in Combustible Gas Control, the staff has found that the PAR maintains an inert atmosphere in the CNV during a DBA for 24 hours, and beyond 72 hours the PAR maintains the CNV inert indefinitely. During a SA for 72 hours post-accident, an inert atmosphere is maintained without crediting the PAR.

A discussion of equipment survivability is located in FSAR section 19.2.3.3.8, "Equipment Survivability." The list of SSCs required to establish safe shutdown and maintain containment integrity was provided in FSAR table 19.2-8, "Equipment Survivability List." The equipment survivability list includes the PAR, which must be shown to survive and function following accident conditions other than combustion.

3) Containment Mixed Atmosphere

The NuScale CNV represents the final barrier to a radionuclide release by serving as the means of inventory retention and provides a structure for the primary heat transfer to the ultimate heat sink, the reactor pool. As such, it is important to maintain containment integrity, both during design basis and significant beyond design basis scenarios where the containment is initially intact.

In an atmosphere of combustible gases of hydrogen and oxygen, one criterion for not supporting combustion of the hydrogen is an oxygen concentration of less than or equal to 4 per cent without crediting the PAR. This is an inert atmosphere. The NuScale CNV is effectively inerted for many accident sequences. Oxygen is generally the limiting reactant for producing a bounding combustion loading in the scenarios examined by the applicant.

The provisions set forth in RG 1.7 require that containment should be adequately mixed to support the analysis. Mixing is 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 CNV is relatively straightforward based on the single volume that makes up 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. This time 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). NuScale explained that it evaluated the Ra for both the annular region and the upper volume and determined that the Ra would exceed this transition regime to turbulence by at least one order of magnitude. Based on its review, that was available to the staff during the regulatory audit, the NRC staff agrees that bulk turbulent mixing is likely to occur for at least 72 hours in the annular region and the upper region. Therefore, the NRC staff finds that the design meets the criteria for a mixed atmosphere in the CNV following a DBA and an SA.

4) Hydrogen Production

Materials within the containment that would yield hydrogen gas by corrosion from the emergency cooling or containment spray solutions should be identified, and their use should be limited as much as practicable.

Staff notes that FSAR section 6.3 states that ECCS is fabricated by stainless steel or stainless steel clad for the CNV and components inside the CNV, precluding corrosion.

5) Structural Analysis of Containment Integrity

Steel containments must meet the requirements of the ASME Boiler and Pressure Vessel Code (edition and addenda as incorporated by reference in 10 CFR 50.55a(b)(1)), Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, considering pressure and dead load alone.

As discussed above, in the event of an accident resulting in combustible gas concentrations inside the containment, the applicant evaluated the peak post containment pressure following a burning event of hydrogen generated from 100 per cent fuel clad reaction to determine the structural loads on the containment at 72 hours post-accident. In FSAR Section 19, Table 19.1-29, AICC analysis describes these results. This analysis does not credit the performance of the PAR. The bounding analysis for the NuScale NPM-160 (US600 design) results were utilized as the two designs are similar, with the NPM-20 (US460 standard design) at a higher core power, 250 MW versus 160 MW, and the bounding hydrogen amounts produced by oxidation during a severe accident at 72 hours. Assuming the same initial thermal hydraulic conditions, the NPM-160 calculation was repeated for the NPM-20, resulting in a containment pressure for the NPM-20 less than its design pressure of 1200 psia. During the regulatory audit, the staff reviewed NuScale's evaluation and agrees with the conclusions. The staff agrees with this approach and concludes that the NPM-20 design meets the criteria of RG 1.7, C.5.

Evaluation of SDAA Part 7, Exemption Request No. 2, 10 CFR 50.34(f)(2)(xvii)(C) Combustible Gas Monitoring

NuScale requested an exemption from 10 CFR 50.34(f)(2)(xvii)(C), which requires the capability for monitoring hydrogen concentration in containment during an accident.

10 CFR 50.34(f)(2)(xvii) requires equipment for monitoring hydrogen in containments that use an inerted atmosphere for combustible gas control.

The US460 standard design precludes combustion in containment during a DBEs and SAs by passively controlling the oxygen concentration by relying on the PAR to maintain an inert atmosphere.

As discussed above the NPM relies on the PAR to maintain the containment inert through the continuous recombination of oxygen and hydrogen that may initially be present in the NPM, may be released at the beginning of and during beyond-DBAs, and any generated post accident. The design does not rely on hydrogen monitoring to assess core damage. As described in FSAR section 7.1, the radiation monitors under the bioshield and the core exit thermocouples provide the ability to detect and assess core damage.

Specific Exemption

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 would 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 would not impact any design function. 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. This exemption concerns only the capability to monitor combustible gases during a beyond-DBA; the design relying on the PAR precludes combustion. Therefore, the exemption does not present an undue risk to the public health and safety. 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 would 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 would 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

Underlying Purpose of the Rule

In accordance with 10 CFR 50.12(a)(2)(ii), special circumstances 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 design with the addition of the PAR precludes combustion that could challenge containment structural integrity, safe-shutdown functions, or accident mitigation features. Combustible gas monitoring is not necessary to support severe accident management and emergency planning. 10 CFR 50.34(f)(2)(xvii)(C), which also requires containment hydrogen monitoring capability, is a Three Mile Island requirement that is not applicable to the NuScale design as described above. Therefore, the design also meets the underlying purpose of 10 CFR 50.34(f)(2)(xvii)(C). The staff concludes that special circumstances are present in accordance with 10 CFR 50.12(a)(2)(ii), as the design achieves the underlying purpose of the rule.

10 CFR 50.12(a)(2)(vi)– Special circumstances are present whenever--There is present any other material circumstance not considered when the regulation was adopted for which it would be in the public interest to grant an exemption. The design has a very low likelihood of core damage that would lead to significant amounts of combustible gases accumulating within containment, and the design relying on the PAR passively controls oxygen levels to preclude combustion. Combustible gas monitoring could require unisolating the containment during the response to an accident, where containment isolation is essential to both severe accident prevention and mitigation and in the public interest to avoid. Therefore, the staff finds that the difference in risk tradeoff is a material circumstance not considered when the regulation was adopted.

Exemption Request No. 2 Conclusion

In accordance with 10 CFR 50.12(a)(1), the staff finds that the requested exemption to 10 CFR 50.34(f)(24)(xvii)(C) is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security. The NRC has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) are present, and the underlying purpose of the rule is met. Staff has also determined that other material circumstances exist that were not considered when the regulation was adopted, as described above, for which it would be in the public interest to grant an exemption.

The staff considered NuScale's exemption request and determined that an exemption from 10 CFR 50.34(f)(2)(xvii)(C), if shown to be applicable and properly supported in a request for exemption by a CP or COL applicant that references the SDA, would be justified and could be issued to the CP or COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the CP or COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

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.

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 concludes that, for DBA and beyond-DBA scenarios where combustible gas concentrations may be relevant, preventing a loss of containment structural integrity, a loss of safe-shutdown functions, or a loss of accident mitigation features does not require providing hydrogen and oxygen monitoring in containment. The containment design relies on a PAR to maintain an inert environment post-accident. Therefore, the staff concludes that an exemption from the requirements of 10 CFR 50.34(fc)(2)(xvii)(C), if shown to be applicable and properly supported in a request for exemption by a CP or COL applicant that references the SDA, would be justified and could be issued to the CP or COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

The applicant demonstrated that the containment atmosphere will be adequately mixed, the necessary inventory of equipment to maintain safe shutdown and containment integrity is qualified to the conditions expected in a postulated accident involving hydrogen and oxygen generation. Hydrogen monitoring capability is not required, and containment integrity will be maintained during a postulated accident. Further, the staff finds NuScale has satisfied PDC-41 with the addition of the PAR to control the concentration of oxygen in the CNV following postulated accidents to assure that the CNV integrity is maintained.

6.2.6 Containment Leakage Testing

6.2.6.1 Introduction

The purpose of containment leakage rate testing (CLRT) design is to verify the leak tight integrity of the CNV does not exceed the allowable leakage rate in the Technical Specifications (TS) that protect against uncontrolled releases to the environment. Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” to 10 CFR Part 50 specifies that this includes Types A, B, and C testing.

NuScale has requested an exemption from the CLRT 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 local leakage rate tests (LLRTs) (Type B and Type C).

As part of this exemption request, NuScale has requested an exemption from Appendix J to not perform the Type A integrated leakage rate tests (ILRTs) 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 in SDAA Part 7, "Exemptions," Revision 0, "10 CFR Part 50, App. A, GDC 52 Containment Leakage Rate Testing," issued December 31, 2022 (ADAMS Accession No. ML22365A018).

6.2.6.2 Summary of Application

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 by ITAAC: the leakage rate for LLRTs (Type B and Type C) for the pressure containing or leakage-limiting boundaries and containment isolation valves (CIVs) to meet the requirements of 10 CFR Part 50, 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.

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 to 10 CFR Part 50. The design, in conformance with 10 CFR 50.54(o), accommodates the 10 CFR 50 Appendix J, test frequencies of Option A or Option B, with the exception of a containment ILRT. 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, L_a, is 0.20 weight percent (wt%) of the containment air mass per day at the calculated P_a, Peak Accident Pressure. The CLRTs are 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 L_a.

NuScale Exemption Request No. 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 ILRT.

Known leakage pathways will be tested pre-operationally and periodically by Type B and Type C LLRTs. All CNV penetrations are either ASME Code, Section III, Class 1 flanged joints capable of Type B testing, or ASME Code, Section III, Class 1 welded nozzles with isolation valves capable of Type C testing, or part of a closed system inside containment (i.e., steam generator system piping). An additional Type B test occurs at the main CNV flange. All pipe penetrations with CIVs will be Type C tested, except for the GDC 57 main steam and feed water lines. There are 46 total penetrations in the CNV, including the CNV main closure flange. The Type B tests for bolted flange penetrations and the Type C tests for CIVs are identified in the FSAR Table 6.2-3 and 6.2-4, respectively. Each of these Type B or C penetrations will be leak tested before initial entry into Mode 4. Subsequent periodic Type B and C tests occur at a baseline frequency of at least once per 30 months until establishment of acceptable performance in accordance with NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR 50, Appendix J." NEI 94-01, Revision 3-A, provides methods for complying with the provisions of 10 CFR Part 50, Appendix J, Option B, and delineates a performance-based approach for determining containment leakage rate testing frequencies. Under Option B of 10 CFR Part 50 Appendix J, extended test intervals are possible for Type B and C penetrations beyond the baseline frequency following the completion of two consecutive periodic as-found tests if the results of each test are within allowable administrative limits.

All CNV flanged openings have bolted connections that are designed and constructed to ASME Code, Section III, Class 1 criteria. All of these openings, including the main CNV flange have identical double O-ring seals with a test port to facilitate Type B testing by pressurizing the space between the seals to P_a. The flanged openings, which contain Electrical Penetration Assembly (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, Containment System Leakage Testing Requirements, issued 1994. Schedules for the Types B and C testing will be in accordance with Appendix J, Option A or Option B, as committed to in TS Programs and Manuals, Section 5.5.9, Containment Leakage Rate Testing Program. FSAR Section 6.2.6, "Containment Leakage Testing," specifies Types B and C testing in accordance with TS 5.5.9, which accommodates both Appendix J, Option A and Option B frequencies. TS Surveillance Requirement 3.6.1.1 requires visual exams and LRT in accordance with the CLRT program. Satisfactory LLRT and inservice inspection (ISI) examination are required for containment OPERABILITY, per TS Bases 3.6.1, LCO.

ITAAC: NuScale SDAA Part 8, Table 2.1-1, 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 #12)

These ITAAC are evaluated in Section 14.3 of this SER.

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

Technical Specification Bases: LCO 3.6.1 states that the containment is designed to maintain leakage integrity less than or equal to 1.0 L_a. Leakage integrity is assured by performing local LLRT and containment ISI. Total LLRT leakage is maintained less than or equal to 0.60 L_a in accordance with Appendix J. Satisfactory LLRT and ISI examinations are required for containment OPERABILITY.

Programs and Manuals: 5.5.9 Containment Leakage Rate Testing Program

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

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

Technical Reports: TR-123952-P, Revision 0, “NuScale Containment Leakage Integrity Assurance,” and incorporated by reference in the SDAA.

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 containment integrated leakage rate test (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 LRT 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, as it relates to specific requirements and acceptance criteria for preoperational and periodic testing of the leak tightness of the reactor containment and penetrations.

The staff reviewed NuScale FSAR Section 6.2.6, using guidance provided in Standard Review Plan (SRP) Section 6.2.6, “Containment Leakage Rate Testing,” which lists the acceptance criteria adequate to meet the above requirements, as well as review interfaces with other SRP sections.

6.2.6.4 Technical Evaluation

The staff reviewed the containment leakage testing described in FSAR Section 6.2.6, using the staff’s review guidance provided in SRP Section 6.2.4. SRP Section 6.2.4 identifies the staff’s review methodology and acceptance criteria for evaluating compliance with requirements related to containment leakage rate testing.

The staff’s review of the applicant’s compliance with requirements and departures from requirements is in the following sections.

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 to 10 CFR Part 50 requires that primary reactor containments meet the containment leakage test requirements to provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment (Type A) and systems and components that penetrate containment (Type B and Type C).

NuScale based its exemption request to not meet the above regulations for Type A testing on meeting the underlying purpose of the regulation. NuScale provided 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 selected L_a, at P_a, to be 0.20 wt% of the containment air mass per day. For the NuScale design, L_a 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 containment leakage to the environment. Typically, this leakage would be demonstrated through a combination of preoperational and periodic Type A (containment integrated leak rate test) and Types B and C (LLRT) testing. NuScale has requested an exemption from performing any Type A testing.

NuScale asserted 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, L_a, would not be exceeded. In accordance with Appendix J, the sum of Types B and C test leakages must remain less than 0.60 L_a, an acceptance criterion that will be demonstrated to reflect the most recent tests results.

NuScale asserted 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 stated that, for the CNV, which is designed as an ASME Code, 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 stated that the CNV is 100 percent inspectable, both inside and outside, whereby aging-related flaws leading to potential leakage could be observed. ASME Code, Section XI, “In-Service Inspection of Nuclear Power Plant Components,” 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 committed to inspecting the flanges, flange bolts, and flange gaskets for each opened penetration, including the main closure flange.
- NuScale proposed a preservice design pressure test to confirm the low-leakage design of the CNV.

NuScale asserted that, with the above-listed items and LLRT, it will demonstrate that L_a 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 Code, 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 a frequency in accordance with the CLRT Program.

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 P_a.
- Each bolted flange is Type B tested at a frequency in accordance with the CLRT program; all flanges are as-found tested, and if the flange is opened, as-left tested.
- 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 a frequency in accordance with the CLRT program.

In accordance with 10 CFR Part 50 Appendix J, the sum of Type B and C test leakages must remain less than or equal to 0.60 L_a. 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 L_a.

To support the assertion of low expected leakage regarding the CNV flange design, NuScale confirmed that the CNV bolting design for the closure flange and EPAs, instrument seal assemblies (IPA), and access ports, will be in accordance with ASME Code, Section III, Paragraph NB-3231 for the design conditions. NuScale performed a containment flange bolting calculation that demonstrates that the bolt preload for the design pressure is sufficient to resist the hydrostatic end force and maintain a compression load on the gasket contact surface to ensure a tight joint when the design pressure is applied to the internal surfaces. The staff agrees that the CNV bolted closure design and preload design requirements ensure Type B flange seals, including EPAs and ISAs, remain in contact at accident temperature concurrent with peak accident pressure, therefore 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. ITAAC Item #12 in SDAA Part 8, Table 2.1-1; FSAR Section 6.2.6.5.2; and TR-123952-P, Revision 0, provides the details for conducting the preservice design pressure test.

The NRC based its review on the following:

- The NuScale CNV is designed as an ASME Code, 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 Code, 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 a frequency in accordance with the CLRT program.

The staff agrees that the applicant’s rationale would justify an exemption from the requirement to perform Type A testing and the requirement to provide design capability for ILRT as stipulated in GDC 52, provided that the exemption is shown to be applicable and properly supported in a request for exemption by a CP or COL applicant that references the SDA, and that there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the CP or COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

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 would 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 would 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 would 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 would 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

Underlying Purpose of the Rule

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 reasonable assurance that the leak-tight integrity of containment would be maintained and that CNV leakage would not exceed allowable leakage rate values.

Undue Hardship

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 for not providing design features for Type A testing as required by GDC 52 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 a 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 a benefit to public health and safety by maintaining occupational radiation doses as low as is 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 Type 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.

Conclusion

In accordance with 10 CFR 50.12(a)(1), the staff finds that the requested exemption from 10 CFR Part 50, Appendix J, and GDC 52 is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security. The NRC has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) are present; therefore, the underlying purpose of the rule is met, and the existing requirements would present an undue hardship for the NuScale design. In addition, an exemption from GDC 52 and Appendix J Type A tests would also result in a benefit to public health and safety by maintaining occupational radiation doses as low as is reasonably achievable, and the NuScale design presents circumstances not considered when the regulation was adopted. The staff therefore concluded that this exemption, if shown to be applicable and properly supported in a request for exemption by a CP or COL applicant that references the SDA, would be justified and could be issued to the CP or COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the CP or COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

6.2.6.5 Compliance with Containment Leakage-Rate Testing Regulations and Guidance

Appendix J to 10 CFR Part 50 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 that 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 (1) 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 (2) 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 Type 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.

The staff reviewed the ITAAC, as it relates to CLRT, as presented in NuScale SDAA Part 8. The staff finds that the applicant has adequately identified ITAAC consistent with the requirements for Type B and Type C testing in SDAA Part 8, Table 2.1-1, 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 criterion 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 SDAA Part 8, Table 2.1-1, Item No. 12. The staff finds the no visible leakage criteria an acceptable demonstration of the low leakage design of the containment. This has been incorporated into the NuScale FSAR Section 6.2.6. 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 FSAR 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 FSAR 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 in 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 wt% in 24 hours. Based on the discussion above in the technical evaluation, the staff finds that successful Type 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 in Chapter 16 of this SER.

6.2.6.6 Combined License Information Items

Table 6.2.6-1 NuScale COL Information Items for Section 6.2.6

Item No.	Description	FSAR Section
6.2-1	An applicant that references the NuScale Power Plant US460 standard design will verify that the final design of the containment vessel meets the design basis requirement to maintain flange contact pressure at accident temperature, concurrent with peak accident pressure.	6.2

6.2.6.7 Conclusion

The staff considered NuScale's exemption request related to Type A testing and ILRT and determined that this exemption, if shown to be applicable and properly supported in a request for exemption by a CP or COL applicant that references the SDA, would be justified and could be issued to the CP or COL applicant for the reasons provided in NuScale's SDAA, provided there are no changes to the design that are material to the bases for the exemption. Where there are changes to the design material to the bases for the exemption, the CP or COL applicant that references the SDA would be required to provide an adequate basis for the exemption.

The staff has reasonable assurance that the NuScale CNV design, with its ISI, ASME Code 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, the staff concludes that the NuScale FSAR Section 6.2.6 for the CLRT design is acceptable and meets the relevant requirements of GDC 53, GDC 54, and 10 CFR Part 50, Appendix J (except the provisions related to Type A testing covered by the exemption request as discussed above).

6.2.7 Fracture Prevention of Containment Pressure Boundary

6.2.7.1 Introduction

This section of the SDAA Part 2 describes fracture prevention of the reactor containment pressure boundary materials.

6.2.7.2 Summary of Application

SDAA Part 2: FSAR Section 6.2.7 describes the design, as summarized, in part, below.

The NuScale CNV is an evacuated pressure vessel fabricated from martensitic (ferritic) 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 stated 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 cited adherence to the fracture toughness criteria of ASME Code, Section III, Subsection NB and ASME Code, Section XI, Appendix G, “Fracture Toughness Criteria for Protection Against Failure,” as the basis for this statement.

6.2.7.3 Regulatory Basis

The staff reviewed NuScale SDAA Part 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 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 of this SER 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 of this SER 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 SDAA 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 of this SER, and the evaluation of GDC 16 is provided in Section 6.2 of this SER.

6.2.7.4 Technical Evaluation

The staff reviewed the information included in FSAR, 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 typically exhibit sufficient inherent fracture toughness that additional requirements are deemed unnecessary. For example, the austenitic stainless steel used for the CNV lower shell, SA965, FXM19, was 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 staff focused its review on the novel selection of a martensitic steel for a portion of the CNV. Typically, martensitic steels exhibit higher hardness and lower toughness than milder forms of ferritic steels. 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. 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.

The staff verified that as the SA-336, F6NM material (and associated welding materials, noted in Table 6.1-1, etc.) falls under the ferritic requirements of ASME Code, Section III and Section XI, as applied to the CNV, the selection of a martensitic steel does not affect the NRC staff review in a novel way. Specifically, such material must still meet the fracture toughness requirements of ASME Code, Section III, NB-2331 and through reference in Section III, ASME Code, Section II, SA-336, 8.3.4 fracture toughness minimum values. These requirements ensure that the material will be tested to confirm specified minimum toughness properties at a temperature no warmer than 0 °F, thereby conservatively bounding typical operating conditions (e.g., room temperature and above).

The staff audited several engineering documents including ones concerning the CNV. During the audit, the staff noted that the audited documents conform to the fracture toughness requirements of ASME Code, Section III and Section XI. In particular, the staff noted during the audit that the fracture toughness properties of the subject martensitic materials were conservatively specified in excess of the ASME Code requirements. This provides additional assurance that the NuScale CNV will have adequate fracture toughness properties.

Finally, the staff reviewed whether the martensitic materials would receive sufficient neutron fluence to credibly affect the fracture toughness of the subject materials. The applicant discusses the potential for neutron embrittlement of the CNV in several locations in the SDAA, for example Section 8.2.6 of TR-123952-NP, "NuScale Containment Leakage Integrity Assurance." The staff confirmed that based on the configuration of the CNV, the martensitic materials are estimated to receive less than 1×10^{17} n/cm^2 (E > 1 MeV) and therefore are unlikely to have measurable reductions in fracture toughness due to neutron embrittlement. Consequently, the initial fracture toughness properties of the subject martensitic materials can be relied on to represent the design life fracture toughness properties of the CNV materials.

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 ASME Code, Section III, Division 1, Subarticle NB-2300. Compliance with the requirements of ASME Code, Section III, NB-2331 will ensure adequate fracture toughness is verified for the subject materials. 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 in FSAR, Section 3.8.2. Consequently, the staff finds that NuScale has adequately met the requirements of GDC 51 through application of the above noted requirements in ASME Code, Sections II, III, and XI and Appendix G.

6.2.7.5 Conclusion

Based on the review of the information included in NuScale SDAA 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, NuScale SDAA 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 primary function is to provide core cooling during and after AOOs and postulated accidents, including LOCAs. The ECCS consists of two RVVs mounted on the RPV upper head, which are 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 four valves are closed during normal plant operation and open following the receipt of an actuation signal resulting from applicable event conditions such as low RPV water level, or upon the loss of the augmented DC power system. The ECCS actuation values are identified in FSAR Table 3.9-16 and Table 6.3-1. The associated interlocks are identified in FSAR Table 7.1-5. The ECCS valves can also open on low differential pressure without an ECCS actuation signal; however, this is not a safety-related function. In addition, the ECCS valves provide a safety-related function to remain closed and function as part of the RCPB during normal operation and provide 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 (i.e., open).

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 one RVV are required to open to provide adequate core cooling. The ECCS does not provide replacement or addition of inventory from an external source. 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. In addition, the ECCS includes a supplemental boron feature that provides a long term reactivity control function.

6.3.2 Summary of Application

FSAR Section 6.3 provides information regarding the ECCS design as a whole and on a component basis. The FSAR 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, and 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 SRP Section 6.3, "Emergency Core Cooling System," Revision 3 and DSRS Section 6.3, "Emergency Core Cooling System," Revision 3 the guidance provided in applicable RGs, and the NRC's regulations.

ITAAC: SDAA Part 8, Section 2.1, describes the NPM and associated systems, one of which is the ECCS. SDAA Part 8, Table 2.1-4, provides an inventory of the mechanical equipment for the ECCS; specifically, the RVVs, RRVs, associated trip and reset valves, and flow restricting venturis. SDAA Part 8, Table 2.1-5, similarly inventories the electrical equipment, which includes the ECCS trip and rest valves; these valves are also considered for equipment qualification in SDAA Part 8, Table 2.4-1, which also lists the instrumentation required to monitor the ECCS. SDAA Part 8, Table 2.1-1 items 5, 11, 14, and 19 contain the specific ITAAC associated with ECCS. These ITAAC are evaluated in Section 14.3 of this SER.

Technical Specifications: Generic TS 3.5.1, "Emergency Core Cooling System," and TS 3.5.4, "Emergency Core Cooling System Supplemental Boron (ESB)," are applicable to this area of review:

Technical Reports: There are no technical reports associated with FSAR 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 and DSRS 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 LOCAs

- GDC 5, as it relates to SSCs important to safety not being shared among nuclear power units (NuScale power 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 systems that shall permit functioning of SSCs important to safety

- GDC 17, "Electric Power Systems," as it relates to electrical power systems that shall permit functioning of SSCs 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 33, “Reactor Coolant Makeup,” as it relates to demonstrating that the ECCS provides specified acceptable fuel design limit (SAFDL) protection from leaks and small breaks from the RCPB

- 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, “Inspection of Emergency Core Cooling System,” and GDC 37, “Testing of Emergency Core Cooling System,” 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 may be developed in conformance with Appendix K 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 (NUREG-0737, TMI Action Plan Item II.K.3.18)

- 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 (NUREG-0737, TMI Action Plan Item II.K.3.28)

- 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 (NUREG-0737, TMI Action Plan Item II.D.3)

- 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 (NUREG-0737, TMI Action Plan Item II.F.2)

- 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 remove decay heat for the extended period of time required by the long-lived radioactivity remaining in the core

The staff notes that while GDC 33 identified above is not listed in SRP or DSRS Section 6.3, the applicant provided a description in FSAR Section 6.3 regarding how the ECCS design satisfies the underlying purpose of GDC 33. This, in part, supports the staff's review of the exemption request to GDC 33 evaluated in Section 9.3.4.4.7 of this document. The staff's evaluation of the ECCS design and how it satisfies the underlying purpose of GDC 33 is included in the technical evaluation below.

The staff also 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.

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 Section 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),” Revision 0, issued November 1970
- RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss of Coolant Accident,” Revision 4, issued March 2012
- RG 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems,” Revision 1, issued February 2010
- RG 1.29, “Seismic Design Classification for Nuclear Power Plants,” Revision 4, issued March 2007
- RG 1.157, “Best Estimate Calculations of Emergency Core Cooling System Performance,” Revision 0 issued May 1989
- Design Certification/Combined License-Interim Staff Guidance (DC/COL) ISG-019, “Review of Evaluation to Address Gas Accumulation Issues in Safety Related Systems,” issued July 2010

6.3.4 Technical Evaluation

6.3.4.1 Functional Design Basis

The staff reviewed the ECCS design relative to the following safety related functions stated in the FSAR:

- RCPB
- LTOP
- core cooling in situations when cooling cannot be provided by other means (such as a LOCA or loss of access to non-safety-related secondary side SSCs)

The staff notes that the ECCS for the NPM-20 design also includes an ECCS supplemental boron (ESB) feature to perform the safety-related function of maintaining a safe shutdown condition following a DBE. The ESB provides additional boron to ensure the reactor remains subcritical and is evaluated in more detail below.

6.3.4.1.1 Reactor Coolant Pressure Boundary

FSAR Section 6.3.1 states that during normal operation the RVVs and RRVs are closed and function as part of the RCS pressure boundary. During normal operation the main valve disks are held closed against spring force by a pressurized control chamber initially reset by the CVCS. FSAR Table 3.9-16, “Active Valve List,” identifies that the RVVs, RRVs, and associated trip and reset valves perform the function to maintain the RCPB. FSAR Table 6.3-4, “Classification of Structures, Systems, and Components,” classifies the ECCS valves (RVVs, RRVs, and associated trip and reset valves) as safety-related. The staff finds the classification of the ECCS valves as safety-related for this function acceptable because they meet the definition of safety-related in 10 CFR 50.2 and the definition of a basic component in 10 CFR Part 21. Specifically, the ECCS valves are relied upon to remain closed during normal operation (LTOP function during shutdown conditions is discussed below) when there is not an actuation signal or loss of the augmented DC power system to assure the integrity of the RCPB, and are relied on to open when there is a valid ECCS actuation signal. Therefore, with respect to this safety-related function, the RVVs and RRVs are required to be designed and manufactured under a quality assurance program complying with 10 CFR Part 50, Appendix B and any defect or noncompliance will be subject to 10 CFR Part 21 given their role to protect one of the fission product barriers. GDC 14 states that the RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and of gross rupture. The detailed evaluation regarding how NuScale complies with these requirements is summarized in Chapter 3 of this SER. The staff reviewed the capability of the ECCS valve system to perform its safety function, including auditing NuScale’s ECCS valve design demonstration testing to show compliance with 10 CFR 50.43(e) for this safety feature of the NuScale US460 design (ML23304A429). The details of the staff’s review and audit in this area are described in Section 3.9.6 of this document.

6.3.4.1.2 Low-Temperature Overpressure Protection

The ECCS is also designed to ensure that RPV pressure-temperature limits are not exceeded. When wide range RCS cold temperatures are less than 290 °F, the ECCS provides LTOP. The pressure setpoint for the LTOP function is a variable based on the calculated saturation temperature for the RCS as depicted in FSAR Figure 5.2-3. The MPS logic provides the actuation signal that opens the RVVs. Once the wide range RCS cold temperature exceeds 290 °F during startup, the LTOP function is bypassed. The NuScale ECCS is designed to provide LTOP of the RPV to satisfy GDC 15, “Reactor Coolant System Design,” GDC 30, “Quality of Reactor Coolant Pressure Boundary,” and GDC 31 requirements. GDC 15 requires overpressure protection for the RCS to ensure that design pressure conditions are not exceeded during the normal range of operations, including AOOs. GDC 30 addresses the design, fabrication, and testing of RCPB components and GDC 31 ensures that sufficient margin is provided to ensure that the components behave in a nonbrittle manner and also to ensure that rapid propagation failure is minimized. The LTOP function of ECCS is reviewed in more detail in Section 5.2.2 and Chapter 7 of this SER.

6.3.4.1.3 Core Cooling Performance Evaluation

While the design of the ECCS is reviewed in this section of the report, the ECCS performance evaluation under AOO and 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. The behavior of the ECCS system following a signal from the MPS, operator action, or by loss of power from the augment DC power system is discussed in FSAR Section 6.3. FSAR Section 6.3.2.2 provides the minimum flow coefficients and maximum opening times for the RRVs and RVVs, which are consistent with the values assumed in the safety analyses. Additionally, TS LCO 3.5.1 assures that the RVVs and RRVs operate as assumed in the safety analyses.

6.3.4.1.4 Inadvertent Emergency Core Cooling System Valve Actuation

FSAR Section 6.3.2.2 states that a venturi is provided in between the RPV and the inlet of each RVV and RRV, and its purpose is to slow the depressurization rate by limiting blowdown flow during inadvertent valve opening events. The RVV venturi is stated to have a throat diameter of 3.5 inches and the RRV venturi throat diameter is one inch. This is a key design feature relied on to prevent the exceedance of minimum critical heat flux ratio (MCHFR) limits. The inadvertent ECCS valve opening AOO analysis is provided in FSAR Section 15.6.6, “Inadvertent Operation of the Emergency Core Cooling System,” and the staff’s evaluation is contained in Section 15.6.6 of this SER.

6.3.4.1.5 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.

FSAR 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. The applicant analyzed a spectrum of breaks and performed evaluations using TR-0516-49422-A, Revision 5 (ML25136A217 NP, ML25136A220 P), which the staff has reviewed and found acceptable, subject to the conditions and limitations listed in the staff’s SER for TR-0516-49422 (ML24312A004).

As part of the LOCA break location and size evaluation, the staff identified that the design-basis LOCA break spectrum did not include consideration for losses of coolant at the bolted ECCS valve to vessel connection or for the CVCS lines between the CNV and CIVs. The staff’s evaluation of this issue is documented in Section 15.6.5 of this SER.

6.3.4.1.6 Long-Term Cooling

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. FSAR Section 6.3.1 states the ECCS is designed as a passive system and is able to perform its safety function for at least seven days after a DBE without support from a non-safety system. FSAR Section 6.3.3 states that containment heat removal in conjunction with ECCS operation provides sufficient capacity to rapidly reduce containment pressure (defined in FSAR Section 6.2.1.1.1 as less than 50 percent of peak calculated pressure within 24 hours after a DBE). Additionally, FSAR Section 6.3.3 references the extended passive cooling evaluation model (TR-124587) and the results documented in FSAR Section 15.0, 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 for 72 hours.

During the long-term cooling period following an anticipated operational occurrence (AOO) or postulated design basis accident, the DHRS system is actuated to remove core decay heat and other sensible heat by condensing steam in the isolated DHRS system loops. The primary side boil off vapor in the upper riser region can contain combustible gases, hydrogen, and oxygen, which would be generated via radiolysis of water. Chapter 5 specifies that the COL applicant will conform with the EPRI PWR Primary Water Chemistry Guidelines, and COL item 5.2-2 specifies that the COL applicant will develop and implement a water chemistry plan. The primary side radiochemical design is meant to include a minimum quantity of dissolved hydrogen gas at and above which radiolytic species and ions are unable to follow a chemical reaction pathway that would lead to formation of oxygen gas. If water chemistry is unable to be maintained above this specified critical hydrogen concentration, for example during prolonged CVCS isolation, combustible gases will accumulate in the pressurizer vapor space and will reach the combustion threshold concentration and could cause loss of general mitigating design features including the ECCS. Therefore, the 8-hour ECCS actuation timer proposed by NuScale is needed to perform the function of venting the accumulated combustible gas. During future licensing and construction stages, when the plant water chemistry program is defined, and operational procedures are developed consistent with COL items 13.5-3 and 13.5-6, the applicant's basis for the procedural allowances to bypass the timer in certain situations will need to consider radiolytic gas accumulation (see FSAR Section6.3.2.2). This would typically include quantified predictions for, as one example, dissolved hydrogen depletion rate, or the criteria for bypass would be conservative in nature (note that the considerations in bypassing the timer are only in part based on combustible gas control – criticality control and boron distribution is another consideration). Additionally, the difference between the NPM integral-PWR design and the conventional PWRs to which the EPRI PWR Primary Water Chemistry Guidelines directly apply would need to be considered by the applicant and licensee. For instance, the NPM pressurizer liquid-vapor interface surface area is very large relative to the volume of primary coolant, compared to the same quantity in conventional PWR designs; also, an NPM pressurizer is inside/part of the RPV, unlike in conventional PWR designs.

The staff audited NuScale's calculation of energy deposition to the primary coolant from the shutdown core and determined that NuScale has conservatively computed the time, post-CVCS-isolation, at which radiolytic gases would reach the concentration necessary for combustion and determined that this time is significantly later than the designed ECCS actuation at 8 hours after CNV/RPV/CVCS isolation. See Section 6.3.4.1.9 below for discussion of the 8-hour timer built into the NPM-20 design. {{

}}. The two conservatisms just stated are in addition to typical forms of conservative biasing that the applicant used, such as assumptions of limiting temperatures, pressures, and fluid volumes. After the ECCS is actuated, the combustible gases are vented to the CNV and the PAR is relied on to recombine the gases. See Section 6.2.5 of this report for the staff's review and evaluation of the PAR.

Additionally, the staff assessed whether radiolytic gases could come out of solution and accumulate in the SG region during ECCS operation. In a letter dated January 16, 2025 (ML25016A300), the applicant determined that the buoyancy force would transport the radiolytic gas to the containment through reactor vent valves. Because the low-pressure region in the steam generator region and downward condensate droplets could impede this transport process out of the RPV and to the PAR in containment, the staff performed a confirmatory analysis and concluded that the total amount of radiolytic gas available to accumulate in the downcomer region during ECCS operation will not be sufficient to exceed the combustible limit within 72 hours.

FSAR Section 15.0.5, "Extended Passive Cooling for Decay Heat and Residual Heat Removal"; Section 15.6.5; and Section 15.6.6, established the basis to justify that the ECCS is designed to provide sufficient core cooling once it is activated. The staff reviewed the NuScale analyses and confirmed that they are acceptable, as documented in Sections 15.0.5, 15.6.5, and 15.6.6 of this report. Therefore, on the basis of the evaluations performed by the staff in Chapter 15 of this SER, which demonstrate sufficient core cooling, the NRC staff concludes that the NuScale ECCS design meets the requirements of PDC 35.

FSAR Section 19.3.2.2 states safety analyses, PRA insights, and sensitivity studies provide reasonable assurance that core cooling and containment integrity is maintained during the time period beginning 72 hours after a DBE and lasting the following 4 days, with only safety related SSC. See Section 19.3 of this SER for the staff's evaluation of the 7-day assessment of the US460 design.

6.3.4.1.7 Reactivity Control

The ECCS includes a passive ECCS supplemental boron (ESB) feature. FSAR Section 6.3.2.2.1 states the ESB feature ensures the reactor remains subcritical for at least 72 hours following a DBE. In addition, FSAR Section 19.3.2.2 concludes the reactor will remain subcritical within the first 7 days following a DBE.

FSAR Section 6.3.2.2.1 states two condensate channels extend outwards and upwards along the CNV inner wall (one main channel and one auxiliary channel). The purpose of the main channel is to direct condensate to the dissolver basket for dissolution while the purpose of the auxiliary channel is to direct water to the inside annulus space of the dissolver to provide additional condensate flow. Each ESB dissolver basket contains solid boron oxide that dissolves into condensate during ECCS operation for recirculation into the core to maintain subcriticality. Mixing tube components in the lower CNV transport condensate to mix with liquid in the bottom of containment.

FSAR Section 6.3.2.2.1 states that during refueling outages the ESB hoppers are loaded with solid boron oxide pellets and, once the CNV is drained during the startup process, the hoppers are remotely actuated to release the pellets into the associated dissolver basket. Each dissolver contains sensors that indicate the quantity of boron oxide in order to confirm the pellets have been delivered from the hoppers to the dissolvers.

Boron oxide pellet size and quantity is varied on a cycle-specific basis. TS LCO 3.5.4 requires the form and quantity of boron in the ESB dissolvers to be within the limits specified in the cycle-specific Core Operating Limits Report (COLR). TS SR 3.5.4.1 requires the form and quantity of boron to be verified prior to entering Mode 1 after operation that could have previously affected the form or quantity of boron in the ESB dissolver. While the pellet size may change on a cycle-specific basis, the extended passive cooling analysis assumes upper and lower limits for diameter, and the pellet form does not change. FSAR Section 15.0.5.3.1 states the boron oxide pellets are modeled as equilateral cylinders with 3/8-inch diameters. In a letter dated November 20, 2024 (ML24325A593), the applicant clarifies {{ }} and the maximum pellet diameter is the conservative value applied to the slow-biased boron dissolution analysis.

The applicant performed boron transport analysis to ensure adequate mixing within the RCS and sufficient dissolved boron is provided by the ESB such that the core remains subcritical during long term cooling. In addition, the applicant assessed the maximum boron concentrations in the core region following actuation of the ECCS determined that they do not result in boron precipitation that interrupts that natural circulation flow in the core and long term cooling capability. The boron precipitation analysis biased important parameters, such as initial boron concentration, dissolution rate, and mass, and demonstrates margin to boron solubility limits. This staff's review of this analysis is presented in Section 15.0.5 of this SER.

The applicant's calculation supporting the subcriticality evaluation presented in FSAR Section 15.0.5 is limited to the first 72 hours following a DBE. Consistent with SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," (ML003708224), the staff evaluated whether the applicant assessed subcriticality for a period up to 7-days following a DBE. FSAR Section 19.3.2.2 states that the analyses documented in FSAR Section 15.0.5 demonstrate that the limiting time for subcriticality occurs in the first 72 hours. In a letter dated December 11, 2024 (ML24346A237), the applicant further supported this conclusion by evaluating the trends for critical boron concentration documented in the underlying engineering calculation to demonstrate that boron concentration beyond 72 hours would remain above the critical concentration. The staff disagrees with the applicant's statement that the restrictive Chapter 15 analysis assumptions are not necessary for consideration of Regulatory Treatment of Non-Safety Systems (RTNSS) criterion B for the period from 72 hours to 7 days. The conservatisms referred to in Chapter 15 analysis, such as the assumption of a stuck control rod, are considered within the design basis and are consistent with SECY-96-128 to assume that such a design basis condition exists when performing the analysis. Accordingly, the staff bases its conclusion on the applicant's assessment described above, which includes a stuck control rod, and finds the 7-day subcriticality evaluation acceptable because the boron concentration stays above the critical concentration. See Section 19.3 of this SER for additional staff evaluation of the 7-day assessment of the US460 design.

In addition, during its review the staff observed that several AOO transients documented in SDAA Chapter 15 cause an increase of RCS pressure and result in a reactor safety valve to lift in order to maintain RCS pressure within design limits. This discharged steam from the RCS could condense on the CNV walls, flow into the ESB condensate channels, and partially dissolve the boron oxide pellets. This sequence could impact the initial conditions of boron oxide pellet quantity, size, and form assumed in the boron transport analysis. In letter dated February 4, 2025 (ML25035A089), the applicant stated TS LCO 3.5.4 addresses the operability requirements for the ESB and includes verification of the ESB after operation that could affect the form or quantity of boron in the ESB dissolvers. The applicant further clarified that a reactor safety valve lift would be an example of such an operation. Based on the clarification by the applicant the staff agrees operability of the ESB will be ensure by the surveillance requirement to verify quantity and form of boron which is required to be performed prior to startup following a transient that resulted in the discharge of steam to the containment. See Chapter 16 of this report for the staff's detailed review of the technical specifications.

6.3.4.1.8 8-hour timer

The NPM-20 design includes an automatic 8-hour timer that actuates ECCS after an automatic or manual reactor trip to 1) allow the ECCS supplemental boron to recirculate into the reactor core region to maintain subcriticality and 2) vent any combustible gases that have accumulated in the RCS due to the radiolytic decomposition of water. The 8-hour timer and resulting ECCS actuation ensures the above functions are performed without requiring operator actions. FSAR Section 6.3.2.2 states operators may manually block the automatic actuation of ECCS if subcriticality at cold conditions is confirmed and if it's confirmed that sufficient hydrogen concentration will be maintained in the RCS throughout DHRS cooldown to preclude radiolytic generation of combustible gases. While post-event recovery actions are outside the scope of the standard design review, these potential actions are important to consider and capture in the development of operating procedures. In a letter dated November 20, 2024 (ML24325A595), the applicant states Technical Specification Limiting Condition for Operation 5.4 and FSAR Section 13.5 requires operating procedures in accordance with Regulatory Guide 1.33, Revision 3. This includes operating procedures for plant shutdown to hot standby, hot standby to cold shutdown, and preparation for refueling and refueling equipment operations. In letter dated November 20, 2024 (ML24325A593), the applicant adds that if operators desire to bypass the 8-hour ECCS actuation they must perform a reactivity balance to verify subcriticality and will account for several factors, such as control rod position, RCS boron concentration, and xenon content. These activities will be governed by site-specific operational procedures. (See Section 13.5 of this SER for the staff's review of COL information items associated with plant procedures). Consistent with the safety analyses presented in FSAR Chapter 15, the staff confirmed that the 8-hour ECCS timer does not rely on operator actions and finds that manual operator action to bypass the 8-hour timer and prevent the RCS from rapidly depressurizing into the CNV will appropriately account for the intended safety function of the 8-hour timer to control reactivity and vent combustible gases.

6.3.4.1.9 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, including an assessment of debris impact on ECCS components, the fuel, and the core. The NuScale design minimizes the effects of post-accident debris accumulation in the CNV by eliminating or minimizing the sources of debris. The debris minimization is accomplished by the following:

- not permitting thermal insulation (metallic or nonmetallic) inside containment
- 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 on the inside or outside surface of the CNV, or on any other ESF or non-ESF system components located 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 FSAR Section 6.3.2.4, as, unintended dirt, dust, paint chips, fibers, pieces of paper, plastic, or tape, etc. is expected to be the source of debris inside the CNV. A conservative estimate of latent debris present in the NuScale plant assessed by the applicant by examining the latent debris present in current operating plants.

NuScale prescribes debris limits in FSAR Section 6.3.3.1. Based on analysis of the debris limits, and referenced testing, NuScale concludes that the core inlet will not be blocked by debris following a LOCA and that an acceptable post-LOCA peak cladding temperature 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. The staff verified conservative assumptions were used in the calculation to justify the applicant's conclusion that an acceptable peak cladding temperature will be achieved following a reactor trip. Therefore, the staff concludes that the in-vessel debris downstream effects will not result in a post-LOCA peak cladding temperature greater than 800 degrees F (see Section 6.2.2.4.2 of this report for the staff's detailed review and audit of the NuScale analysis of debris generation and its impact on long-term core cooling related to GSI-191).

6.3.4.1.10 GDC 33 Exemption Evaluation

FSAR Section 6.3.1 describes how the ECCS design does not require a reactor makeup system and satisfies the underlying purpose of GDC 33. The request for exemption from GDC 33 is provided in SDAA Part 7, Section 5. The staff discussion of this exemption is provided in Section 9.3.4.4.7 in this document. This section provides the staff's evaluation of the ECCS design and how it satisfies the underlying purpose of GDC 33.

GDC 33 requires, in part, a system to supply reactor coolant makeup to assure that SAFDL acceptance criteria are met with onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available). FSAR Section 6.3.1 states that the ECCS setpoints ensure automatic actuation of ECCS valves in responses to design basis LOCA events or 24 hours after loss of AC power. The staff's evaluation of the ECCS performance in response to a design basis LOCA as discussed above is in Section 15.6.5 of this document and the staff determined that SAFDL acceptance criteria are not exceeded.

FSAR Section 15.0.5.3 provides analysis for ECCS performance during extended passive cooling conditions. This analysis considered break area percentages within the design basis LOCA evaluation and very small breaks, such as those due to leakage and concludes that the ECCS maintains core cooling for 72 hours. The staff's evaluation of the methodology is in Section 15.0.5 of this report.

In addition, as provided in the evaluation above, the ECCS is designed as a passive system and is assessed to perform its safety function for core cooling and reactivity control for at least seven days after a DBE without support from a non-safety system.

6.3.4.2 System Design Features

6.3.4.2.1 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, with respect to core cooling. Therefore, the staff finds that the ECCS design meets the regulatory requirements of GDC 5 because ECCS components are not shared among NPMs. The staff's evaluation of the UHS is provided in Section 9.2.5 of this SER.

6.3.4.2.2 Power Requirements

The safety function of the ECCS valves to remain closed during normal operation and maintain the integrity of the RCPB requires operability of the augmented DC power system to provide power to maintain the trip solenoid valves in the closed position. A loss of the augmented DC power system results in the opening of all eight trip solenoid valves (de-energize-to-open) causing a blowdown of the RCS via immediate opening of the two RVVs, and eventual opening of the two RRVs after the IABs release. The reliability of the ECCS as a whole is discussed further below, and the staff's evaluation of capability, capacity, and reliability of the augmented DC power system is provided in Section 8.3 of this SER. The loss of the augmented DC power system is evaluated as an initiating event in Chapter 15 of this SER.

The safety function of the ECCS to passively cool the core following a valid actuation of the MPS sensing a loss of coolant from the RCS, or actuation based on the MPS 8-hour timer following a non-LOCA event, is accomplished without an AC or DC power supply. This satisfies, in part, the underlying purpose of GDC 17 for electrical power requirements. NuScale requested an exemption to GDC 17 and the staff's evaluation of this requested exemption to GDC 17 is in Section 8.1.4 of this report. The loss of the AC power system is evaluated in Chapter 15 of this SER.

6.3.4.2.3 Instrumentation

During a LOCA IORV event, the ECCS is actuated based on RPV riser "level." The RPV riser level is determined by a thermal dispersion switch assembly that is stated to be capable of measuring the presence or absence of local liquid water at discreet locations axially along the probe assembly. In NRELAP5 models of NPM-20, ECCS actuation is discerned by a surrogate based on the computed, local, cell-homogenous void fraction at the elevation of the sensor, relative to the reference height for measuring RPV level.

Regulations in 10 CFR 50.34(f)(2)(xi) (NUREG-0737, TMI Action Plan Item II.D.3) require proper indications in the control room of the status of ECCS. The NuScale reactor is designed for all four ECCS valves to have valve position indications and solenoid power indication for the ECCS trip and reset valves available in the control room. In addition, the ESB contains weight element loading sensors (shown in FSAR Figure 6.3-1) that provide indication in the main control room that the proper quantity of boron oxide pellets is initially positioned in the ESB dissolver.

The staff's review and evaluation of the ECCS instrumentation, including digital-based common cause sensor failures and defense-in-depth assessment is contained in Chapter 7 of this report. The staff's review and evaluation of RPV riser level sensor modeling is documented in Chapter 15 of this report..

6.3.4.2.4   System Boundary

FSAR Section 6.3.1, states that NuScale ECCS components do not extend beyond the CNV boundary. All ECCS components such as tubing, pipes, and main valve bodies are enclosed in the containment. The staff notes that the ECCS trip and reset valves assemblies are mounted directly on the CNV wall outside of containment; however, they do not have any direct fluid mass exchange with the reactor pool or the air space outside of the CNV. Based on the above, the staff finds that the ECCS design for the NPM US460 satisfies 10 CFR 20.1406 because there is not direct fluid mass exchange with the reactor pool or the space outside the CNV.

6.3.4.2.5   Testing, Inspection, and Qualification

NuScale designed the module protection system 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 in-service testing and inspection requirements of ASME Code, Section XI, Subarticles IWC-2500 and ISTC-3100 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.

Preoperational tests of the ECCS valves included in the NuScale initial test program verify that the as-constructed system functions as credited in the safety analysis. Specifically, FSAR Table 14.2-40, describes a first of a kind performance test of the integral ECCS for the first NPM, and FSAR Table 14.2-56, describes ECCS valve, including the IAB, testing for all NPMs.

The staff also confirmed that the ESB will be tested. During an audit NuScale stated that proper ESB function is assured by boron dissolution separate effects testing; technical specification operability requirements; existing ITAAC; and conservatisms applied within the evaluation of ESB performance. The staff views these elements, and other system information provided in the FSAR, as pertinent to the establishment of performance requirements for the system and necessary to perform the safety analysis of the design, all of which are necessary to support a final safety determination of the standard design. In addition, FSAR Table 14.2-40 includes a first-of-a-kind test for the ESB feature to ensure proper functioning of the system, including acceptable boron dissolution and mixing.

The staff considers these tests as essential to demonstrating initial operability at a component level and of the system as a whole. The first-of-a-kind ECCS and ESB tests provide assurance that the system operates in accordance with the design, validates the analytical models, and verifies the correctness and conservatism of assumptions used to predict plant responses to anticipated transients and postulated accidents as described in the applicant’s response (ML24325A593).

Based on the above, the staff finds that the ECCS design satisfies GDC 36 and 37, which require that an ECCS be designed to permit appropriate periodic inspection of important component and functional testing to assure the structural and leak tight integrity, the operability of the key components, and the system as a whole. The staff’s evaluation of ECCS preservice and in-service testing and inspection is in Sections 3.9.6 and 6.6 of this report. The staff’s evaluation of the initial test program is in Section 14.2 of this report. The staff’s review of structural and leak tight integrity of components and assurance of the operability of ECCS valves is in Section 3.9.6 of this report.

6.3.4.2.6 Environmental and Dynamic 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 an RCS break, a steam line break within containment, and inadvertent ECCS valve actuation. The environmental qualification of these valves is addressed in FSAR Section 3.11 and FSAR Section 7.2.2 and is reviewed under the corresponding sections of this report.

FSAR Table 6.3-5 provides the SSC classification, quality group, and seismic classification of the ECCS. The components of the ECCS (valves, hydraulic lines, and actuator assemblies) are designed to be Quality Group A, seismic Category 1 and meet the requirements of ASME Code, Section III, Subsection NB, 2017 Edition. Regarding the ESB, FSAR Section 3.9.3.5 states that the ESB components are non-pressure retaining; however, they are designed to withstand applicable loads for the DBEs and the stress limits specific to the material specification mechanical property requirements to maintain their structural integrity. In letter dated November 20, 2024 (ML24325A593), the applicant adds that while the ESB is non-pressure retaining it is a safety related system subject to commensurate treatment, which includes application of 10 CFR Part 50, Appendix B quality assurance requirements, material selection, welding practices and nondestructive examination to that of a safety related SSC. Additional information addressing compliance with the applicable codes and classification of ECCS components is provided in FSAR Sections 1.9, 3.2, and 3.9.

The staff confirmed that seismic Category 1 is specified for the ECCS, including the ESB. The system and its components are designed to seismic Category 1 requirements. Equipment seismic qualification is addressed in FSAR 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 FSAR Section 3.5. The effects of pipe ruptures and valve discharge are discussed in FSAR Sections 3.6.1 and 3.6.2. Fire protection is covered in FSAR Section 9.5. The staff evaluation of these effects is documented in the corresponding sections of this report.

During its review of the NPM-160 DCA the staff focused on two phenomena in evaluating the valves controlling operation of the ECCS: water hammer and flashing of borated water. As discussed in the associated audit documentation (ADAMS Accession No. ML19340A019), the staff audited the design 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. NuScale performed ECCS valve testing and showed that, during the IAB actuation and valve opening operation, the water hammer phenomenon was not observed, which NuScale attributed to the small diameter of the hydraulic lines. The staff found that the design and operation of each ECCS valve system, including the small hydraulic tubing, precludes the susceptibility to water hammer mechanisms. Therefore, based on the above test information and the described ECCS valve design, the staff concluded that the ECCS valve systems in each NuScale reactor module are not susceptible to water hammer. In addition, the ECCS actuator hydraulic lines also contain borated water, which may flash during a depressurization. The staff reviewed information from NuScale (ADAMS Accession Nos. ML18043A162 and ML19233A203) that stated during normal operation, the pressure in these lines is equalized to

RCS pressure. The staff notes that after each operating cycle, these lines are submerged in the reactor pool environment during refueling. These lines will be flushed regularly during each refueling cycle to ensure no accumulation of plated-out boric acid. Therefore, based on the testing described above and the flushing procedures, the staff concluded that the NuScale ECCS valve system design addressed the issue for water hammer and flashing adequately. The staff confirmed that the changes made to the ECCS valves for the SDAA NPM-20 design do not impact the staff's previous review and conclusions associated with the potential for water hammer and flashing of borated water, and therefore, the aforementioned conclusions remain applicable to the NPM-20 design. See Section 3.9.6 of this report for the staff's detailed evaluation of the NuScale ECCS valve system.

6.3.4.2.7 System Reliability

ECCS reliability of the core cooling safety function is provided by redundant valves remotely and separately actuated by two divisions of the ESFAS function of the module protection system. The ECCS valves and actuators are assigned to separate divisions for instrumentation and electric power.

Redundancy of ECCS components, features, and capabilities is provided to ensure that the system core cooling safety function of RVV and RRV operations can be accomplished assuming a single failure of an ECCS main valve to open. As discussed above, the capability to provide periodic pressure and functional testing of the ECCS ensures operability and performance of system components, and preoperational performance tests ensure the operability and performance of the system as a whole. The staff evaluation of the failure mode and effects analysis of the ECCS valves, as well as additional ECCS valve testing, is documented in Section 3.9.6 of this report.

The RRVs include an IAB valve which is a first-of-a-kind, safety significant, active component. In order to meet the requirements for the ECCS in 10 CFR Part 50, an applicant must show that it has addressed the single failure criterion (SFC). The SFC is defined in 10 CFR Part 50, Appendix K and derived from the definition of single failure in 10 CFR Part 50, Appendix A. During its review of the NMP-160 DCA design, the staff noted that, although the applicant assumed a single failure of a main ECCS valve to open, the applicant did not apply the SFC to the IAB valve in regard to the valve's function to close. NuScale disagreed with the staff's application of the SFC to the IAB valve, which led the staff to request Commission direction to resolve this issue in SECY-19-0036, "Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves," dated April 11, 2019 (ADAMS Accession No. ML19060A081). In SECY-19-0036, the staff summarized the NRC's historical practice for applying the SFC. Specifically, the staff summarized SECY-77-439, "Single Failure Criterion," dated August 17, 1977 (ADAMS Accession No. ML060260236), in which it informed the Commission of how the staff then generally applied the SFC, and SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994 (ADAMS Accession No. ML003708068), and the associated staff requirements memorandum (SRM), dated June 30, 1994 (ADAMS Accession No. ML003708098), which related to Commission direction on application of the SFC in specified fact or application-specific circumstances. In view of this historical practice, the staff, in SECY190036, requested Commission direction on application of the SFC to the IAB valve's function to close.

In response to the paper, the Commission directed the staff in SRM-SECY-190036, “Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves,” dated July 2, 2019 (ADAMS Accession No. ML19183A408), to “review Chapter 15 of the NuScale Design Certification Application without assuming a single active failure of the inadvertent actuation block valve to close.” The Commission further stated the following:

> This approach is consistent with the Commission’s safety goal policy and associated core damage and large release frequency goals and existing Commission direction on the use of risk informed decision making, as articulated in the 1995 Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities and the White Paper on Risk Informed and Performance Based Regulation (in SRM SECY-98-0144 and Yellow Announcement 99-019).

Based on the staff’s historic application of the SFC and Commission direction on the subject, as described in SECY-77-439, SRM-SECY-94-084, and SRM-SECY-19-0036, the NRC has retained some discretion, in application-specific circumstances, to decide when to apply the SFC. The Commission’s decision in SRM-SECY-19-0036 provides direction regarding the appropriate application and interpretation of the regulatory requirements in 10 CFR Part 50 to the NuScale IAB valve’s function to close. This decision is similar to those documented in previous Commission documents that addressed the use of the SFC and provided clarification on when to apply the SFC in other specific instances.

NuScale requested an exemption to GDC 17 which also cascades into an exemption request to GDC 35 requirements for electric power. This exemption request is evaluated in Chapter 8 of this SER. NuScale’s PDC 35 is functionally identical to GDC 35, except with respect to the discussion regarding electric power. Based on the above discussion, including redundancy of the ECCS to perform its design function assuming a single failure of a main ECCS valve to open, and the Commission determination that the SFC does not need to be assumed for the NuScale IAB valve’s function to close, the staff concludes that the ECCS system satisfies PDC 35 with respect to demonstrating that, “the safety system function can be accomplished, assuming a single failure.” The staff’s evaluation of NuScale’s PDC 35 with regard to sufficient core cooling in the ECCS system is discussed above. On the same basis, the staff finds that the requirements of 10 CFR Part 50, Appendix K to assume, “the most damaging single failure of ECCS equipment has taken place” for accident evaluations under the associated requirements of 10 CFR 50.46 are met for NuScale’s ECCS, as described in Section 15.0.0.5 of this SER. Therefore, evaluation of a single failure of the NuScale IAB valve’s function to close is not necessary for the evaluation of DBEs in FSAR Chapter 6 or Chapter 15.

The General Design Criteria provided in 10 CFR Part 50, Appendix A were developed with the consideration that the loss of any fission product barrier shall not occur within the lifetime of the facility. ECCS reliability of the safety related function to remain closed and protect RCPB integrity is provided by two redundant trip solenoid valves in series. As discussed above, the power to the trip solenoid valves is provided by the augmented DC power system. Because the RVVs do not include IABs a loss of the augmented DC power system results in the immediate opening of the RVVs and blowdown of the RCS at full pressure and temperature conditions (loss of the RCS fission product barrier). In addition, the staff performed a regulatory audit of the augmented dc power system SSC classification report (ML24211A09). This report specifies that {{

}}. While the loss of the augmented DC power system is analyzed as an anticipated operational occurrence initiating event in FSAR Chapter 15, loss of the RCS fission product barrier on a frequent basis would not be considered acceptable and is inconsistent with the GDCs. The staff did not assess the acceptability of the safety classification of the augmented DC power system within this Chapter of the staff's SER. See SER Section 8.3 for the staff's evaluation of the augmented dc power system. See Chapter 15 of this SER for the treatment and reliance on the augmented dc power system in the safety analysis. The RRVs include an IAB valve that prevents the main valve from opening and losing the RCS fission product barrier under full RCS pressure and temperature conditions due to the loss of DC power until after the reactor has been shutdown and decay heat is being removed.

Technical Specifications

TS relating to the ECCS are presented in SDAA Part 4. The staff reviewed required actions and SRs, together with the completion times allotted for corrective action and surveillance frequencies. The staff's evaluation is documented in Chapter 16 of this document.

6.3.5 Combined License Information Items

FSAR Section 6.3.2.4 provides COL Item 6.31 directing a COL applicant that references the NuScale Power Plant US460 standard design to describe a containment cleanliness program that limits debris within containment. SER Section 6.2.2.5 also references COL Item 6.31.

Table 6.3-1 NuScale COL Information Items for Section 6.3

Item No.	Description	FSAR Section
6.3-1	An applicant that references the NuScale Power Plant US460 standard design 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 prohibit the introduction of coating materials into containment. • An inspection program to confirm containment vessel cleanliness before closing for normal power operation.	6.3

6.3.6 Conclusion

The staff finds the applicant has satisfied the requirements in PDC 17 and 35, as well as the portions of GDC 2, 4, 5, 27, 36, and 37 pertinent to the ECCS, and the other requirements listed in Section 6.3.3 of this report. An exemption from GDC 33 as described in Section 6.3.4 of this report, is discussed further in Chapter 9 of this report. As detailed in the technical evaluation above, the ECCS acts as a robust system to maintain the integrity of the RCPB, provide core cooling during and after AOOs and postulated accidents, including LOCAs, and ensure the reactor remains subcritical following DBEs. The components, and their associated performance, in the arrangement described in this section provide the ECCS's capability to preserve sufficient RPV inventory such that the core remains covered, ensuring adequate core cooling and the design conditions of the CNV are not expected to be exceeded. Confirmation that the ECCS described here successfully accomplishes those design basis functions is provided in FSAR Chapter 15, Sections 15.0.5, 15.6.5, and 15.6.6 and Chapter 6, Section 6.2.1. The staff also finds that the ECCS provides effective LTOP for the RPV such that RPV pressure-temperature limits are not exceeded. Based on these considerations, the staff concludes that the design of the ECCS is acceptable and meets the applicable regulations specified in Section 6.3.3 of this SER. The staff further concludes that the applicant has included the appropriate COL items to assure site-specific features of the ECCS will be addressed and appropriately implemented at the COL stage.

6.4 Control Room Habitability

6.4.1 Introduction

Control room habitability refers to the conditions required for life support and safe, effective operation of the plant during normal conditions and following an accident. These conditions include adequate lighting, food, water, air, and climate control.

The control room habitability system (CRHS) provides breathable air to the control room envelope (CRE) for 72 hours without reliance on electrical power if the normal control room HVAC system (CRVS) is unavailable. The CRHS equipment provides habitability functions to protect the operators against postulated releases such as radioactive materials, toxic gas, and smoke.

After 72 hours, the CRVS, if restored, provides filtered heating, ventilation, and air conditioning (HVAC) service to the control building (CRB) for the remainder of an event recovery period. The CRVS serves the entire CRB. The technical support center (TSC) is served by the CRVS, but not the CRHS. The CRVS is evaluated separately in Section 9.4.1 of this safety evaluation.

The plant protection system (PPS) isolates the CRE and breathing air is provided by the CRHS under conditions of loss of all alternating current power or high radiation levels. If smoke is detected, CRVS is automatically isolated from outside air, and CRVS begins recirculation.

CRE lighting during CRHS activation and/or normal plant operation is evaluated per SRP Section 9.5.3 "Lighting Systems," Revision 3, March 2007(ADAMS Accession Number ML070550036) in section 9.5.3 of this Safety Evaluation.

6.4.2 Summary of Application

FSAR Section 6.4, "Control Room Habitability," describes the CRHS system. The CRHS is a non-safety- -related system that provides emergency breathing air to the CRE to maintain habitability and control radioactivity when conditions prohibit the CRVS from fulfilling these functions. The CRHS maintains a positive control room envelope pressure of at least 1/8-inch water gauge with respect to adjacent areas. The CRHS maintains the environment in the CRE habitable for personnel during abnormal and station blackout conditions when CRVS is unavailable. The CRHS supports control room habitability for the shared main control room. The CRE is comprised of the main control room (MCR); a reference room; a shift manager's office; a shift turnover room; office space; and other areas to support MCR operation. The CRE includes air locks for ingress and egress.

The supply, return, and general exhaust CRVS ductwork serving the CRE are the only heating, ventilation, and air conditioning penetrations through the CRE. These penetrations include redundant isolation dampers that are located within the CRE to protect CRE occupants from hazardous conditions. These dampers can be closed to isolate the CRE, allowing the CRHS to pressurize and provide breathable air to the CRE. The CRE isolation dampers are qualified to shut tight against CRE pressure in support of the CRHS for maintaining MCR habitability. There are no single active failures that would prevent isolation of the CRE.

On a loss of power to both CRVS air handler units or loss of power to the common augmented direct current power system battery chargers, after a ten-minute delay the CRVS isolates the CRE, and the PPS actuates the CRHS. System operation following loss of normal AC power does not affect the safety of MCR personnel or performance of equipment needed to safely operate the plant.

The CRHS performs a function verified by ITAAC that is not safety related; namely, the CRHS supports the CRB by providing breathable air to the CRE for 72 hours without reliance on electrical power if the CRVS is unavailable. After 72 hours, the CRVS, if restored, provides filtered heating, ventilation, and air conditioning (HVAC) service to the CRB for the remainder of an event recovery period.

SDAA Part 8: The CRHS ITAAC are shown in SDAA Part 8, Table 3.1-1, "Control Room Habitability System Inspection, Tests, Analysis, and Acceptance (ITAAC)." These ITAAC are evaluated in Section 14.3 of this SER.

The CRHS performs the following non-safety related system functions that are verified by ITAAC. The CRHS supports the CRB by providing clean breathing air to the CRE including the MCR and maintains a positive control room pressure during high radiation or loss of normal AC power conditions.

SDAA Part 8 lists the following five Design Commitments to be verified via ITAAC:

- The air exfiltration out of the CRE is less than or equal to the assumptions used to size the CRHS inventory and the supply flow rate. – ITAAC 03.01.01 (i.e., Table 3.1-1)
- CRHS valves 00-CRH-SV-0007A; 00-CRH-SV-0007B; 00-CRH-SV-0028A and 00-CRH-SV-0028B change position under design basis temperature, differential pressure, and flow conditions. – ITAAC 03.01.02
- CRHS air-supply isolation solenoid-operated valves 00-CRH-SV-0007A and 00-CRH-SV-0007B and solenoid-operated CRE pressure relief isolation valves 00-CRH-SV-0028A and 00-CRH-SV-0028B perform their function to fail open on loss of motive power under design basis temperature, differential pressure, and flow conditions. – ITAAC 03.01.04
- The CRE heat sink passively maintains the temperature of the CRE within an acceptable range for the first 72 hours following a DBA. – ITAAC 03.01.04
- The CRHS maintains a positive pressure in the MCR relative to adjacent areas. – ITAAC 03.01.05

Technical Specifications: There are no TS requirements associated with the CRHS.

6.4.3 Regulatory Basis

SRP Section 6.4, “Control Room Habitability System,” Revision 3, March 2007(ADAMS Accession Number ML070550069) and SRP Sections 12.3–12.4, “Radiation Protection Design Features,” Revision 5, September 2013 (ADAMS Accession Number ML13151A475) provide staff review guidance and acceptance for the CRHS. The following are the relevant requirements of NRC regulations for this area of review:

- GDC 4, as it relates to the CRHS being appropriately protected against dynamic effects and being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal operation, maintenance, testing, and postulated accidents. The GDC 4 evaluation includes the adequacy of environmental support for safety related SSCs within areas served by the CRHS.
- GDC 5, as it relates to shared SSCs among nuclear power units.
- GDC 19, as it relates to maintaining the nuclear power unit in a safe condition under accident conditions and providing adequate radiation protection as it relates to providing adequate protection to permit access to an occupancy of the control room under accident conditions.
- 10 CFR 50.34(f)(2)(xxviii), in that it mandates that the licensee “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 make necessary design provisions to preclude such problems.”
- 10 CFR 50.63(a)(2) mandates that “The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Licensees are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.”
- 10 CFR 20.1406, as it relates to the design features that will facilitate eventual decommissioning and minimize, to the extent practicable, the contamination of the facility and the environment and the generation of radioactive waste.

The following NRC regulatory guides are also applicable for this review:

- RG 1.178, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release" Revision 2 December 2021 (ADAMS Accession Number ML21253A071)
- RG 1.155, “Station Blackout” August 1988 (ADAMS Accession Number ML003740034)

6.4.4 Technical Evaluation

The CRHS is a non-safety- -related system that provides emergency breathing air to the CRE to maintain positive CRE pressure for habitability and to control radioactivity in the CRE when conditions prohibit the CRVS from fulfilling these functions. The CRHS maintains a positive control room envelope pressure of at least 1/8-inch water gauge with respect to adjacent areas.

The major components of the CRHS include:

- high pressure air compressor;
- high pressure air storage bottles;
- air bottle racks;
- eductor;
- silencers; and
- piping, valves, and instrumentation.

The CRHS design includes an external air supply connection so that the air bottles can be replenished from an offsite source if the compressor is unavailable or local air is contaminated.

The CRVS is a non-safety related system that serves the entire CRB, which includes the CRE, the TSC, and other areas. Except in recirculation mode, the CRVS maintains the areas served at a positive pressure [at least 31.1 pascals (gauge) (1/8inch water, gauge)] with respect to the outside environment to limit infiltration of dust and radioactive materials.

The staff reviewed the CRHS in accordance with the review procedure in SRP 6.4. The results of the staff's review are provided below.

6.4.4.1 GDC 4, "Environmental and Dynamic Effects Design Bases"

General Design Criterion 4 is considered in the design of the CRHS. The CRHS is designed to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The CRHS is designed to maintain a suitable ambient temperature and humidity for personnel and equipment in the MCR and other areas within the CRE during normal operation.

The CRB itself is a mild environment with no credible high energy sources as the result of equipment failure. FSAR Section 9.4.1.3 states that there is no credible source of a high energy pipe failure within the CRB that could cause loss of function of the CRE isolation dampers. These CRE isolation dampers are located within the CRE of the CRB. The CRE is located within a portion of the CRB classified as seismic Category I. As mentioned above, the CRE isolation dampers are also designed to seismic Category I and are protected from external events to the extent that the CRB is protected from such events.

The radiation monitors and smoke detectors located in the CRVS outside air intake and downstream ductwork allow the PPS or plant control system to isolate the CRE and the outside air intake as needed in the event of fires, failures, malfunctions, or high radiation.

When normal HVAC service from the CRVS is not available, as displayed in FSAR Table 6.4-2, the thermal mass of the CRB and its contents limit the temperature increase to a maximum dry bulb temperature of 103ºF after 72 hours loss of normal cooling. During this same 72-hour period, with the absence of normal cooling the relative humidity drops from 60% at “0” hours to 22.2% at 72 hours. The analysis is bounded by the climatology site parameters of SDAA Table 2.0-1 and uses the conservative assumption that control room equipment powered by the normal DC power system remains powered for 3 hours. After 72 hours, the CRVS, if available, provides cooling to the CRE.

The CRHS components are in the CRB, which provides protection from potential adverse environmental conditions. The CRHS air bottle racks are designed to ensure that the air bottles do not become missiles within the CRB. The air bottles and racks, supply piping and associated valves, and pressure relief solenoid valves are designated as Seismic Category I per RG 1.29, Revision 6 July 2021 (ADAMS Accession Number ML21155A003). The air compressor and piping up to the isolation valves between the compressor and the air bottles are designed to Seismic Category III standards.

The CRHS is not designed to serve safety related functions. The staff finds that the CRHS SSCs, including CRE isolation dampers, are compatible with the CRE environmental conditions created by the CRHS, including the effects of missiles that may result from equipment failures or tornadoes, and therefore concludes that the design of the CRHS complies with the requirements of GDC 4.

6.4.4.2 GDC 5, “Sharing of Structures, Systems, and Components”

General Design Criterion 5 is considered in the design of the CRHS. Both the CRHS and the CRVS serve multiple reactor modules within the CRE. This includes the MCR which contains the controls for up to six NuScale Power Modules. The CRHS is designed such that a failure of one portion of the system does not impair the ability to perform its regulatory required functions including, in the event of an accident in one module, an orderly shutdown and cooldown of the remaining module(s).

Both the CRHS and the CRVS serve similar non-safety related functions: (a) maintain positive CRE pressure for habitability; and (b) control CRE radioactivity. However, CRHS is the backup system to CRVS. FSAR Section 9.4.1.3, states, “the CRVS does not have a function relative to shutting down an NPM or maintaining it in a safe shutdown condition. Operation of the CRVS does not interfere with the ability to operate or shut down a module.”

FSAR Section 15.0 (Transient and Accident Analyses), indicates that no operator actions are required or credited to mitigate the consequences of DBEs. As such, the operators perform no safety related functions, as defined in 10 CFR 50.2.

The applicant stated that GDC 5 is satisfied because control room operators can safely shut down all reactors should they have to evacuate the control room and the reactor modules will remain in a safe condition when control room habitability is lost during a DBA.

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.

To satisfy GDC 5, the staff’s review of the CRHS is to make sure that sharing of CRHS SSCs in multiple-NPM plants does not significantly impair their ability to perform their safety functions, including, in the event of an accident in one NPM, the ability of NuScale control room operators to shut down the remaining NPMs.

Control room habitability is not credited to be operational during DBAs because neither the CRVS nor the CRHS is safety related. The analyses summarized in Chapter 15 demonstrate that no DBAs require the evacuation of the MCR. In the event of a beyond-DBA that requires the evacuation of the MCR, very little time (on the order of minutes) is required to trip the unaffected reactors from the control room.

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The NuScale design does not credit any operator actions to mitigate DBEs. Specifically, in FSAR Section 15.0.0.6.4, the applicant stated the following:

> There are no operator actions credited in the evaluation of NuScale DBEs. After a DBE, automated actions place the NPM in a safe-state and it remains in the safe-state condition for at least 72 hours without operator action, even with assumed failures.

The NuScale design does not need to credit any operator actions to mitigate DBEs. The staff determined that sharing of CRHS SSCs in multiple NPM plants does not significantly impair their ability to perform their safety functions, including, in the event of an accident in one NPM, an orderly shutdown and cooldown of the remaining NPM(s), and therefore concludes that the design of the CRHS complies with the requirements of GDC 5.

6.4.4.3 GDC 19, "Control Room"

6.4.4.3.1 Evaluation of Request for Exemption from GDC 19

In SDAA Part 7, Exemption #17, NuScale requested an exemption from GDC 19, "Control Room," and proposed to implement a design-specific Principal Design Criterion (PDC) 19 that meets the underlying purpose of the GDC 19 requirement for means to maintain the reactor in a safe condition in the event of a control room evacuation (ML ML20204A986). NuScale states that, the NPM design, as reflected in the Final Safety Analysis Report (FSAR) (SDAA Part 2), conforms to proposed PDC 19, assuring the design capability for safe shutdown from equipment outside the control room, in lieu of the requirements for "design capability for prompt hot shutdown" and "potential capability for subsequent cold shutdown" as specified in GDC 19.

GDC 19 states the following:

> *Criterion 19 - Control Room.* A control room shall 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 loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

> Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.

NuScale's proposed PDC 19 is stated in SDAA Part 2, Section 3.1, "Conformance with U.S. Nuclear Regulatory Commission General Design Criteria," and summarized below:

> A control room shall 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 loss-of-coolant accidents.
>
> Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent any part of the body for the duration of the accident.
>
> Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Requirements associated with the review of this exemption request not specified above include the following:

- 10 CFR 52.137(a) which requires, in part, the following:
    - 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:

    (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 (GDC), establishes minimum requirements for the principal design criteria for water cooled nuclear power plants similar in design and location to plants for which construction permits have 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 following:

  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 following two conditions that must be met for granting an exemption:

  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)).

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

This exemption is not inconsistent with the AEA of 1954, as amended 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

This exemption does not affect the performance or reliability of power operations, does not impact the consequences of any DBE, and does not create 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

Underlying Purpose of the Rule

Special circumstances are present (10 CFR 50.12(a)(2)(ii)) in that application of the regulation in the particular circumstances is not necessary to achieve the underlying purpose of the rule. The underlying intent of the remote shutdown portion of GDC 19 is to provide means for operators to place and maintain the reactor in a safe condition in the event of a control room evacuation and the requirement of “cold shutdown” in GDC 19 is not necessary to achieve this purpose. For NuScale's passive advanced light water reactor design, the establishment of PDC 19 to require remote "safe shutdown" capability instead of "cold shutdown" specifically, is supported and consistent with NRC guidance, such as SECY-94-084, “Policy and Technical issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” dated March 28, 1994, which applies to passive residual heat removal systems and RG 1.189, “Fire Protection for Nuclear Power Plants,” regarding fire in the main control room. In the event of a MCR evacuation, all reactors are tripped, and decay heat removal and containment isolation are initiated prior to operators evacuating the MCR. These actions result in passive cooling that achieves and maintains safe shutdown (i.e., Mode 3 where keff < 0.99 and all NPM reactor coolant temperatures are < 420°F). Operators can also place the reactors in safe shutdown from outside the MCR in the module protection system equipment rooms within the reactor building. Accordingly, the NRC staff determined that the applicant has met the underlying purpose of the remote shutdown portion of GDC 19 by providing means for operators to maintain the reactor in a safe condition in the event of a control room evacuation.

In SDAA Part 7, Section 17, “10 CFR 50, Appendix A, Criterion 19, Control Room,” the applicant stated that special circumstances described in 10 CFR 50.12(a)(2)(iv) associated with a benefit to public health and safety are present. However, as described in 10 CFR 50.12(a)(2), where the staff finds that special circumstances are present in accordance with 10 CFR 50.12(a)(2)(ii), a staff finding on whether special circumstances exist in accordance with 10 CFR 50.12(a)(2)(iv) is not necessary for the exemption to be granted. Because the staff finds that special circumstances are present in accordance with 10 CFR 50.12(a)(2)(ii), the staff makes no finding regarding the presence of special circumstances described in 10 CFR 50.12(a)(2)(iv).

Conclusion

The staff concludes that PDC 19 maintains the required control room and remote shutdown capabilities, but clarifies that safe shutdown is the necessary reactor condition to achieve and maintain from outside the MCR. In accordance with 10 CFR 50.12(a)(1), the staff finds that the requested Exemption #17 to GDC 19 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. The NRC has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) are present because application of the regulation is not necessary to achieve the underlying purpose of the rule. The staff approves granting NuScale’s proposed Exemption #17 from the requirements of GDC 19.

6.4.4.3.2 Evaluation PDC 19, "Control Room"

The CRHS, in conjunction with the CRVS, provides compliance with PDC 19, as it relates to maintaining the CRE in a safe condition under accident conditions and providing adequate radiation protection.

The CRVS has radiation monitors and smoke detectors located in the outside air intake. Upon detection of smoke in the outside air duct, the outside air isolation dampers are closed by the plant control system to isolate the CRB from the environment. The CRVS is then operated in recirculation mode to provide conditioned air to the occupied areas of the CRB, with no outside air being introduced into the building. The CRHS is not automatically actuated upon smoke detection. The CRB is not pressurized under these conditions.

When gaseous or particulate radioactivity in the outside air duct exceeds the high setpoint, the normal outside air flow path is isolated, and 100 percent of the outside air is bypassed through the air filtration unit (AFU). If high levels of radiation are detected downstream of the AFU, or if normal AC power is lost for 10 minutes, or if power is lost to all highly reliable dc power system-common (EDSSC) battery chargers, the CRE is isolated and breathable air is supplied by the CRHS. FSAR Section 9.4.1.1 states that, "[t]he CRVS serves no safety-related functions, is not risk-significant, is not credited for mitigation of DBAs, and has no safe-shutdown functions." The performance of the CRVS is evaluated in Section 9.4.1 of this SER.

FSAR Section 6.4.4.2 contains COL Item 6.4-1 which reads:

> An applicant that references the NuScale Power Plant US460 standard design will comply with Regulatory Guide 1.78 Revision 2, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

During the audit of FSAR Section 6.4, the staff evaluated how requirement COL Item 6.4-1 interfaced with FSAR Chapter 14 Initial Test Program and Inspections, Tests, Analyses, and Acceptance Criteria. The staff questioned the ability of a COL applicant to complete the Test Method of System Level Test 16.02.03, Test Objective 1. The Test Method pertained to simulating a toxic gas signal and verification that the "Outside air dampers close to isolate makeup air." The staff posited that given the United States' power reactor fleet's history with toxic gas detection (e.g. chlorine detection), that safety related toxic gas detectors for an array of toxic gas nuclear plant scenarios is not a realistic/sustainable expectation. By letter dated December 11, 2024 (ML24346A243) the applicant amended the subject Test Method to indicate that toxic gas detection may not be available per regulatory guidance RG 1.78, Revision 2. The staff found this resolution acceptable as it provides greater clarity to the intent of System Level Test 16.02.03.

As noted, CRVS includes redundant isolation dampers that close to isolate the CRE. Upon CRE isolation, the supply of breathing air from the CRHS limits the concentration of carbon dioxide in the CRE to 5000 ppm maximum for 72 hours. This CO_2 concentration assumes a maximum of 20 people present in the CRE. According to RG 1.78, Table 1, the toxicity limit for carbon dioxide is 40,000 parts per million. Therefore, the staff concludes that the CRE design is consistent with RG 1.78 and is sufficient to remove concern for the buildup of carbon dioxide.

The staff finds that the system design can protect control room personnel during normal operation and that the guidance of RG 1.78, "Evaluating the Habitability of a Nuclear Power

Plant Control Room during a Postulated Hazardous Chemical Release," will be followed. Therefore, the staff concludes that the CRVS complies with the requirements of PDC 19.

6.4.4.4 10 CFR 50.34(f)(2)(xxviii) Additional TMI-related requirements.

The evaluation of potential pathways for radioactivity and radiation that may lead to control room habitability problems required by 10 CFR 50.34(f)(2)(xxviii) is performed in Section 12.3 of this report.

6.4.4.5 10 CFR 20.1406, "Minimization of Contamination"

FSAR Section 12.3.3.5, states the following:

> During normal operations, the normal control room HVAC system (CRVS) supplies conditioned air to the CRB, including the control room envelope (CRE), the technical support center, and the other areas of the CRB, with outside air that has been filtered (low and high efficiency) to maintain a suitable environment for personnel and equipment. The CRVS is designed to maintain a positive pressure inside the CRB with respect to adjacent spaces. See Section 9.4.1 for additional details.
>
> If a high radiation indication is received from an outside air intake radiation monitor, the supply air is routed through the CRVS filter unit which provides additional HEPA and charcoal filtration. The CRVS is designed to maintain operator doses in the MCR and technical support center within PDC 19 limits.
>
> If power is not available, or if a high radiation indication is received from the radiation monitors in the CRE supply duct, the CRE isolation dampers close and the control room habitability system is initiated.

The CRHS is also designed to maintain a positive pressure inside the CRE with respect to adjacent spaces.

The staff finds that both the CRVS AFUs and the isolation of ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts and within the CRE.

Therefore, based on the above CRVS and CRHS design considerations, the staff concludes that both the CRVS and the CRHS in concert, constitute compliance with 10 CFR 20.1406.

6.4.4.6 10 CFR 50.63, "Loss of All Alternating Current Power"

In FSAR Section 9.4.1.3, the applicant stated the following:

> In a station blackout event, the CRE isolation dampers close to form part of the CRE boundary.

In FSAR Section 8.4, "Station Blackout," the applicant stated the following:

> The SBO duration for passive plant designs is 72 hours pursuant to Nuclear Regulatory Commission policy provided by SECY-94-084 and SECY-95-132 and the associated staff requirements memoranda. Passive plants are required to demonstrate that safety related functions can be performed without reliance on

AC power for 72 hours after the initiating event. The relevant guidelines of Regulatory Guide (RG) 1.155 are applied as they pertain to compliance with 10 CFR 50.63 for the passive NuScale design.

The SBO does not pose a significant challenge to the plant, which does not rely on AC power for performing safety functions. A safe and stable shutdown is automatically achieved and maintained for 72 hours without operator actions.

The control room remains habitable for the duration of the SBO event using the control room habitability system. The control room instrumentation to monitor the event mitigation and confirm the status of reactor cooling, reactor integrity, and containment integrity also remains available.

The staff finds that the design of the CRHS complies with 10 CFR 50.63 regarding the capability for responding to a station blackout (SBO), specifically maintaining acceptable environmental conditions to support operator access and egress and equipment functionality during the SBO and recovery period because the CRHS is consistent with the guidance of RG 1.155, Regulatory Position C.3.2.4, and remains operational. Therefore, the CRE room temperature would be expected to be maintained and would not challenge equipment operability or operator performance. After 72 hours, backup power is expected to be available, and the CRVS will then be utilized to provide air conditioning and building pressurization.

6.4.4.7 Initial Test Program

The staff evaluates the ITP in Section 14.2 of this SER.

6.4.4.8 Technical Specifications

There are no TS associated with the CRHS.

6.4.5 Combined License Information Items

The staff evaluates COL Item 6.4-1 in Section 6.4.4.3.2 of this SER.

6.4.6 Conclusion

The staff evaluated the CRHS for the NuScale design using the guidance of SRP Section 6.4. Based on the above evaluation the staff finds that the CRHS design meets GDC 4, GDC 5, PDC 19 with Exemption 17, 10 CFR 20.1406, 10 CFR 50.34(f)(2)(xxviii) and 10 CFR 50.63.

6.5 Fission Product Removal and Control Systems

6.5.1 Engineered Safety Features Filter Systems

This section is not applicable to the NPM-20 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 NPM-20 design.

6.5.3 Fission Product Control Systems

The NPM-20 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 is 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 NPM-20 design.

6.5.5 Pressure Suppression Pool as a Fission Product Cleanup System

This section is not applicable to the NPM-20 design.

6.6 Inservice Inspection and Testing of Class 2 and 3 Systems and Components

6.6.1 Introduction

ISI and in-service testing are periodically implemented at nuclear power plants to assess the structural and leak tight 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 NRC regulations, for example, GDC 36; GDC 39; GDC 45, "Inspection of Cooling Water System"; and GDC 55, as applicable to the Class 2 or 3 system.

6.6.2 Summary of Application

FSAR, Section 6.6 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 in-service 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 in NUREG-0800 - Section 6.6, Revision 2, "Inservice Inspection and Testing of Class 2 and 3 Components," dated March 2007 (ML070550071) 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, leak tightness, 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, and additional provisions for ISI, to minimize the probability or consequences of an accidental rupture of each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment 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 SDAA, Section 6.6 in accordance with SRP Section 6.6. SDAA, Section 6.6 details the proposed requirements for the ISI and in-service testing of the ASME Code 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(a) on the date 18 months before initial fuel load, subject to the conditions listed in 10 CFR 50.55a(b). Additionally, 10CFR50.55a(g)(4)(ii) requires ISI of components and system pressure tests conducted during successive 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, subject to the conditions listed in 10 CFR 50.55a(b). In addition, the optional ASME BPVC cases listed in the latest approved version of RG 1.147 "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1" may be used.

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 Revision 6 “Quality Group Classifications and Standards for Water-, Steam-, And Radioactive-Waste-Containing Components of Nuclear Power Plants”, Quality Group B and C, components.

The ASME Code Class 2 boundaries, based on RG 1.26, Revision 6, for Quality Group B, are as follows:

- (1) part of the reactor coolant pressure boundary, as defined in 10 CFR 50.2, “Definitions,” but excluded from the requirements of 10 CFR 50.55a(c)(1) for reactor coolant pressure boundary components pursuant to 10 CFR 50.55a(c)(2) (as mentioned above in the section on Quality Group A)

- (2) not part of the reactor coolant pressure boundary but part of the following:

    a. systems or portions of systems important to safety that are designed for (1) emergency core cooling, (2) post-accident containment heat removal, or (3) post-accident fission product removal.

    b. systems or portions of systems important to safety that are designed for (1) reactor shutdown or (2) residual heat removal.

    c. hose portions of the steam systems of boiling-water reactors extending from the outermost containment isolation valve 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 all modes of normal reactor operation; alternatively, for boiling-water reactors containing a shutoff valve (in addition to the two containment isolation valves) in the main steam line and the main feedwater line, those portions of the steam and feedwater systems extending from the outermost containment isolation valves up to and including the shutoff valve or the first valve that is either normally closed or capable of automatic closure during all modes of normal reactor operation;

    d. those portions of the steam and feedwater systems of pressurized-water reactors extending from and including the secondary side of steam generators up to and including the outermost containment isolation valves, and connected piping up to and including the first valve (including a safety or relief valve) that is either normally closed or capable of automatic closure during all modes of normal reactor operation;

    e. systems or portions of systems that are connected to the reactor coolant pressure boundary and are not capable of being isolated from the boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure.

The ASME Code Class 3 boundaries, based on RG 1.26, Revision 6 for Quality Group C, are not part of the RCPB but include the following:

(1) cooling water and auxiliary feedwater systems or portions of those systems important to safety that are designed for (1) emergency core cooling, (2) post-accident containment heat removal, (3) post-accident containment atmosphere cleanup, or (4) residual heat removal from the reactor and from the spent fuel storage pool (including primary and secondary cooling systems), although Quality Group B includes portions of those systems that are required for their safety functions and that (1) do not operate during any mode of normal reactor operation and (2) cannot be tested adequately;

(2) cooling water and seal water systems or portions of those systems important to safety that are designed for the functioning of components and systems important to safety, such as reactor coolant pumps, diesels, and the control room;

(3) systems or portions of systems that are connected to the reactor coolant pressure boundary and are capable of being isolated from that boundary during all modes of normal reactor operation by two valves, each of which is either normally closed or capable of automatic closure

(4) systems, other than radioactive waste management systems, not covered by Regulatory Positions 2.a through 2.c (above) that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses that exceed 0.1 rem total effective dose equivalent; only single component failures need be assumed for those systems located in Seismic Category I structures, and no credit should be taken for automatic isolation from other components in the system or for treatment of released material, unless the isolation or treatment capability is designed to the appropriate seismic and quality group standards and can withstand loss of offsite power and a single failure of an active component.

The SDAA Section 6.6 covers the process for inspecting all components defined as Class 2 and 3 by the ASME Code. ASME Class boundaries for piping penetrating the CNV are depicted by Figure 6.6-1.

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 the applicable edition of 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 provided with access to enable the performance of in-service 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), ASME Code Class 2 and Class 3 systems and components (including supports) are designed and provided with access to enable the performance of the in-service examinations.

6.6.4.3 Examination Techniques and Procedures

The ISI examination techniques 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 BPVC Section XI, Articles IWA-2000, IWC-2000 and IWD-2000.

Preservice inspection and subsequent ISI use equivalent equipment and techniques. Preservice inspections occur once, in accordance with ASME BPVC, Section XI, Article IWC-2000 and Article IWD-2000.

Qualification of ultrasonic examination equipment, and procedures is in accordance with ASME BPVC, Section XI.

The ASME BPVC Case N-849, “In Situ VT-3 Examination of Removable Core Support Structure Without Removal” meets the conditions in NRC RG 1.147.

Alternate examination methods are not used.

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 owner’s ISI program. The initial ISI program incorporates the latest edition and addenda of the ASME Code approved in 10 CFR 50.55a(a), subject to the conditions listed 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(a) 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 ASME Code Class 2 and 3 examination results will follow the requirements of ASME Code, Section XI, Article IWA-3000. Evaluations of ASME Code Class 2 and 3 examination results are also performed in accordance with ASME Code, Section XI, Articles IWC-3000 and IWD-3000, respectively. The results of the examinations and evaluations are documented in accordance with ASME Code, Section XI, Article IWA-6000, and the procedures for repair and replacement of Class 2 and 3 components are in accordance with ASME Code, 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

Performance of system pressure tests of Class 2 systems are in accordance with ASME BPVC Section XI, Articles IWA-5000 and IWC-5000 and Table IWC-2500-1.

Performance of system pressure tests of Class 3 systems are in accordance with ASME BPVC Section XI, Articles IWA-5000 and IWD-5000 and Table IWD-2500-1.

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 acceptable to the staff.

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, Revision 21. 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 FSAR does refer to the optional ASME Code cases listed in RG 1.147.

6.6.4.10 COL Action Items and Certification Requirements and Restrictions

Table 6.6-1 NuScale COL Information Items for Section 6.6

Item No.	Description	FSAR Section
6.6-1	An applicant that references the NuScale Power Plant US460 standard design will develop Preservice Inspection and Inservice Inspection Program plans in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, and will establish the implementation milestones for the program. The applicant will identify the applicable edition of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code used in the program plan consistent with the requirements of 10 CFR 50.55a. The applicant will, if needed, address the use of a single Inservice Inspection Program for multiple NuScale Power Modules, including any alternative to the code that may be necessary to implement such an Inservice Inspection Program.	6.6

The staff determines the above COL item to be adequate and appropriate because ASME Code Section XI is the approved method for light water reactors to meet the requirements of 10CFR50.55a.

6.6.4.11 Operational Programs

The examination categories and methods specified in the SDAA are acceptable if they meet the requirements in ASME Code, Section XI, Articles IWC--2000 and IWD-2000-. 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.

Augmented Inservice Inspection to Protect Against Postulated Piping Failures

The Class 2 and Class 3 ISI program includes augmented ISI to protect against postulated piping failures. These inspections provide reasonable assurance of the structural integrity of cold-worked austenitic stainless steel components.

SDAA Section 3.6.2 defines high energy piping systems as fluid systems that, during normal plant conditions, are either in operation or maintain pressurization under conditions where either the maximum operating temperature exceeds 200°F or the maximum operating pressure exceeds 275 psig. The examination areas include the high energy fluid piping systems described in the SDAA Section 3.6.1 and Section 3.6.2.

6.6.5 Conclusions

The design of the ASME Code 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 and 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 ASME Code Class 2 and 3 systems are performed in accordance with the requirements in applicable subsections of ASME Code, Section XI, and provide reasonable assurance that evidence of structural degradation or loss of leak tight 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 ASME Code Class 2 or 3 systems.

The staff concludes that 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 ASME Code Class 2 or 3 systems and 10 CFR 50.55a. This conclusion is based on the applicant meeting the requirements of ASME Code, Section XI, Division 1, as reviewed by the staff and determined to be appropriate for this application.