4 REACTOR

This chapter of the final safety evaluation report (FSER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's (hereinafter referred to as the staff) review of Chapter 4, "Reactor," of the NuScale Power, LLC (hereinafter referred to as the applicant), Design Certification Application (DCA), Part 2, "Final Safety Analysis Report." The staff's regulatory findings documented in this report are based on Revision 5 of the DCA, dated July 29, 2020 (Agencywide Document Access and Management System (ADAMS), Accession No. ML20225A071). The precise parameter values, as reviewed by the staff in this safety evaluation, are provided by the applicant in the DCA 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 DCA and not converted.

4.1 Summary Description

The design of the NuScale Power Module (NPM) is a self-contained nuclear steam supply system comprising a reactor core, a pressurizer, and two steam generators integrated within the reactor pressure vessel (RPV) and housed in a compact steel containment vessel.

This chapter describes the review of the reactor and the reactor core designs, the fuel rod and fuel assembly design, the core control and monitoring components, and the nuclear and thermal-hydraulic design.

4.2 Fuel System Design

4.2.1 Introduction

The design and safety objectives of the fuel system are to ensure that fuel design limits will not be exceeded during normal operations or anticipated operational occurrences (AOOs) and that the effects of postulated accidents will not cause significant damage to the fuel and reactor coolant pressure boundary (RCPB) or impair the capability to cool the core.

4.2.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.9, "Fuel Assembly Design," describes the fuel system design and design goals. In addition, the information in DCA Part 2, Tier 1, ensures that only fuel assemblies approved by the staff or developed under the approved change process can be loaded into a NuScale reactor.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 4.2, "Fuel System Design," describes the fuel system, as summarized, in part, below.

4.2.2.1 Fuel Assembly Description

The NuScale fuel assembly design contains 264 fuel rods and burnable absorber rods, 24 guide tubes, and 1 instrument tube in a 17 by 17 array that is held together by a bottom and top nozzle and guide tubes welded to four spacer grids. A lower grid is captured by rings welded to the guide tubes. The guide thimble tubes serve as channels to guide control rod assemblies

(CRAs) over their entire length of travel. In-core instrumentation is inserted in the central guide tube of selected fuel assemblies.

The fuel assembly analysis demonstrates that the fuel is not damaged during normal operations, AOOs, and postulated accidents.

4.2.2.2 Fuel Rod Description

The applicant stated that the fuel rods consist of enriched uranium dioxide (UO_2) cylindrical ceramic pellets and a round wire Type 302 stainless steel compression spring located in the plenum, encapsulated within an M5™ tube that serves as the fuel cladding. The fuel rods are internally pressurized with helium during assembly.

The applicant stated that the UO_2 pellets are concave at both ends to better accommodate thermal expansion and fuel swelling. The nominal density of the UO_2 in the pellets is 96-percent theoretical density.

In addition, the applicant stated that the fuel rod plenum, which is located above the pellet column, allows space for axial thermal differential expansion of the fuel column and accommodates the initial helium loading and evolved fission gases. The plenum spring at the top of the fuel pellet column keeps the column in its proper position during handling and shipping.

The M5™ fuel cladding has a nominal wall thickness of 0.61 millimeter (0.024 inch). The applicant stated that the M5™ cladding material significantly improves corrosion resistance compared to earlier zirconium alloys.

4.2.2.3 Burnable Absorber Rod Description

To reduce the beginning-of-life moderator temperature coefficient, the applicant included fixed burnable neutron absorber rods in selected fuel assemblies, replacing fuel rods at selected locations. The burnable absorber rod is mechanically similar to fuel rods but consists of gadolinium oxide (Gd_2O_3) mixed in enriched UO_2 in the central rod portion (axially) and enriched UO_2 at the top and bottom. The total column length is the same as the column length of the fuel rods.

4.2.2.4 Control Rod Assembly Description

The CRAs consist of 24 neutron absorber elements connected with a stainless steel spider hub that couples to the control rod drive mechanism (CRDM) drive shaft extension. The neutron absorber elements contain silver-indium-cadmium and boron carbide neutron absorbers in a stainless steel clad.

The CRA design analysis covers potential failure mechanisms, including stress and loads, strain, creep collapse, fatigue, wear, internal pressure, and component melting.

4.2.2.5 Design Evaluation

The applicant stated that the design evaluations of the fuel rod, fuel assembly, and in-core control components consider events during normal operations, AOOs, and postulated accidents (including infrequent events). DCA Part 2, Tier 2, Chapter 15, “Transient and Accident Analyses,” evaluates AOOs and postulated accidents. It also evaluates anticipated transients without scram, which are beyond-design-basis events.

The applicant summarized the design evaluations for each fuel component and Chapter 15-analyzed event and concluded that the appropriate specified acceptable fuel design

limits (SAFDLs) are met. The staff determined that the methodologies used in the analyses apply to the applicant's fuel design in the referenced topical reports (TR)-0116-20825-P-A, "Applicability of AREVA Fuel Methodology for the NuScale Design," Revision 1, issued February 2018 (ADAMS Accession No. ML18040B306), and in TR-0716-50351-P-A, "NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," Revision 1, dated May 2020 (ADAMS Accession No. ML20122A248). TR-0816-51127, "NuFuel-HTP2™ Fuel and Control Rod Assembly Designs," Revision 3, issued December 2019 (ADAMS Accession No. ML19353A719), provides the detailed analyses of the NuScale fuel assembly using the approved methods.

Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC): No ITAAC items are associated with this area of the review.

Technical Specifications: DCA Part 2, Tier 2, Chapter 16, "Technical Specifications," does not provide technical specifications (TS) associated with DCA Part 2, Tier 2, Section 4.2.

Technical Reports: DCA Part 2, Tier 2, Table 1.6-2, "NuScale Referenced Technical Reports," identifies TR-0816-51127, Revision 3, as incorporated by reference into DCA Part 2, Tier 2, Section 4.2.

4.2.3 Regulatory Basis

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

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," and 10 CFR 50.34, "Contents of Applications; Technical Information," as they relate to the cooling performance analysis of the emergency core cooling system (ECCS), using an acceptable evaluation model, and establishing acceptance criteria for light-water nuclear power reactor ECCSs

- Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," as it relates to certain structures, systems, and components (SSCs) being designed to withstand the effects of safe-shutdown earthquakes (SSEs)

- General Design Criterion (GDC) 2, "Design Bases for Protection against Natural Phenomena," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, as it relates to ensuring that SSCs important to safety are designed to withstand the effects of natural phenomena without the loss of capability to perform their safety functions

- GDC 10, "Reactor Design," as it relates to ensuring that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs

- GDC 27, "Combined Reactivity Control Systems Capability," as it relates to the reactivity control systems being designed with appropriate margin and, in conjunction with the ECCS, being capable of controlling reactivity to maintain the capability of cooling the core under postulated accident conditions

- GDC 35, “Emergency Core Cooling,” as it relates to designing the reactor fuel system such that the performance of the ECCS will not be compromised following a postulated accident

The guidance in Section 4.2, “Fuel System Design,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP), lists the acceptance criteria that are adequate to meet the above requirements and review interfaces with other SRP sections.

4.2.4 Technical Evaluation

The staff followed the guidance in SRP Section 4.2 to ensure that (1) the fuel system is not damaged during normal operations and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) core coolability is always maintained.

DCA Part 2, Tier 2, Section 4.2, references TR-0116-20825-P-A, Revision 1, to justify the applicability of various codes and methods for the analysis of the applicant’s fuel designs. The SER for TR-0116-20825-P-A, Revision 1 (ADAMS Accession No. ML18040B306), provides the staff’s evaluation of code and method applicability.

The staff used the information in TR-0116-20825-P-A, Revision 1; DCA Part 2, Tier 2, Section 4.2; and TR-0816-51127, Revision 3, to develop its confirmatory analyses using an independent code, FRAPCON, to assist it in identifying any potential areas for additional review. The sections below summarize the staff’s review, following the guidance in SRP Section 4.2.

The applicant requested an exemption to GDC 27 and proposed Principal Design Criterion (PDC) 27 in lieu of GDC 27. The applicant’s PDC 27 does not rely on poison addition through the ECCS. SER Section 15.0.6 provides the staff’s evaluation of this exemption and proposed PDC.

In addition, the applicant requested an exemption from GDC 35 and proposed PDC 35 in lieu of GDC 35. The applicant’s PDC 35 is functionally identical to the GDC, except for the discussion related to electric power. The modification to the electric power discussion in PDC 35 is tied to the exemption request for GDC 17, “Electric Power Systems,” and proposed PDC 17. SER Section 8.1.5 provides the staff’s evaluation of the exemption to GDC 17 and, by extension, the electric power provision of GDC 35. SER Sections 6.3, 15.6.5, and 15.6.6 evaluate the ECCS against the proposed PDC 35.

4.2.4.1 Design Bases

DCA Part 2, Tier 2, Section 4.2, summarizes the analyses that cover fuel system damage, fuel rod damage, and core coolability. Fuel system damage mechanisms encompass all components within the fuel assembly and are applicable to normal operation, including the effects of AOOs. Fuel rod failure mechanisms are specific to the fuel rod and cladding and are associated with normal operation, AOOs, and postulated accidents. Finally, fuel coolability applies to the fuel assembly retaining its rod-bundle geometry during postulated accidents. The applicant’s analyses cover each failure mechanism and provide the applicable SAFDLs and a concluding summary of the ability of the DCA Part 2, Tier 2, fuel system design, based on the NuFuel-HTP2™ fuel assembly, to meet these limits. This SER section describes the staff evaluation of fuel system damage, fuel rod failure, and fuel coolability.

4.2.4.1.1 Fuel System Damage

Fuel system damage criteria should ensure that fuel system dimensions remain within their tolerances and that the fuel will function as assumed in the safety analyses. The sections below address the following fuel system damage criteria:

- stress/strain limits
- fuel assembly component fatigue
- fuel fretting
- oxidation and hydriding
- dimensional changes (bowing/growth)
- rod internal pressure
- fuel assembly liftoff
- control rod insertability

Stress/Strain Limits. Section 4.1.1, "Stress and Loading Limits," of TR-0816-51127, Revision 3, provides a stress and loading analysis. The shipping and handling stress analysis and the fuel assembly/component stress analysis were performed in accordance with EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," Revision 0, issued February 1999 (ADAMS Accession No. ML003681173), and the clad stress analysis and the cladding buckling analysis were performed in accordance with BAW-10227P-A, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," Revision 1, issued June 2003 (ADAMS Accession No. ML15162B043). The staff finds that these analyses demonstrate the calculated stresses and that loadings are all within the design criteria for all required conditions.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements on stress and loading in accordance with the guidance in SRP Section 4.2.

Fuel Assembly Component Fatigue. Section 4.1.1 of TR-0816-51127, Revision 3, provides fatigue strength calculations for fuel assembly components and structural connections performed in accordance with EMF-92-116(P)(A), Revision 0. The staff finds that these sample analyses demonstrate the calculated fatigue usage factor is significantly less than the design fatigue lifetime when considering normal operation, AOOs, and the operating-basis earthquake (OBE).

Section 4.1.2, "Cladding Fatigue," of TR-0816-51127, Revision 3, provides a sample cladding fatigue analysis performed in accordance with BAW-10227P-A, Revision 1 (ADAMS Accession No. ML15162B043). The staff finds that this analysis demonstrates a calculated fatigue usage factor that is far below the limit of 0.9 for UO_2 and UO_2-Gd_2O_3 fuel with a representatively large number of operating transients.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements on fuel assembly component fatigue in accordance with the guidance in SRP Section 4.2.

Fuel Fretting. Section 4.1.3, "Fretting," of TR-0816-51127, Revision 3, provides a fretting analysis performed in accordance with EMF-92-116(P)(A), Revision 0. The staff finds that, by using fretting tests, this analysis demonstrates the NuScale fuel design is not expected to experience flow-induced vibration or fretting wear issues.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements on fretting and flow-induced vibrations in accordance with the guidance in SRP Section 4.2.

Oxidation and Hydriding. Section 4.1.4, "Oxidation, Hydriding, and Crud Buildup," of TR-0816-51127, Revision 3, provides an oxidation, hydriding, and crud buildup analysis performed in accordance with BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," Revision 1, issued January 2004 (ADAMS Accession No. ML042930233). Using the COPERNIC code and a representative power history envelope, this analysis demonstrates that the calculated oxide thickness is below the design limit of 100 micrometers (0.00394 inch). The staff finds that the corrosion limit restrains the hydrogen pickup, and crud buildup is built into the corrosion thickness.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements on oxidation, hydriding, and crud buildup as defined in SRP Section 4.2.

Dimensional Changes. Section 4.1.5, "Fuel Rod Bow," of TR-0816-51127, Revision 3, provides a fuel rod bow analysis performed in accordance with XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," Supplements 1 through 4, issued October 1983 (ADAMS Accession No. ML081710709). The staff finds that this analysis demonstrates the NuScale fuel is within the current experience base and states that rod bow penalties would be applied to linear heat generation rate and critical heat flux (CHF) based on any calculated bow.

Section 4.1.6, "Axial Growth," of TR-0816-51127, Revision 3, provides a growth analysis performed for the life of the fuel in accordance with EMF-92-116(P)(A), Revision 0. The staff finds that the analysis demonstrates, using the worst case tolerances and growth models, that sufficient clearance is maintained between the fuel assembly and top nozzle and between the fuel assembly and core plate.

Section 4.1.7, "Fuel Assembly Distortion Evaluation," of TR-0816-51127, Revision 3, discusses the potential for fuel rod distortion to affect control rod insertion. The applicant stated that applicable AREVA operating experience shows little in-reactor fuel distortion. Furthermore, NuScale fuel has a greater lateral stiffness than AREVA 17 by 17 fuel, which the staff notes suggests an improved resistance to fuel assembly distortion compared to current AREVA fuel.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements on dimensional changes in accordance with the guidance in SRP Section 4.2.

Rod Internal Pressure. Section 4.1.8, "Fuel Rod Internal Pressure," of TR-0816-51127, Revision 3, provides a fuel rod internal pressure analysis performed in accordance with BAW-10231P-A, Revision 1. The staff finds that, by using a bounding analysis with the COPERNIC code, this analysis demonstrates significant margin exists between the rod internal pressure limit of 12.76 megapascals (MPa) (1,850 pounds-force per square inch (psi)) (equal to reactor coolant system (RCS) pressure) and the calculated maximum internal pressure.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements on rod internal pressure in accordance with the guidance in SRP Section 4.2.

Fuel Assembly Liftoff. Section 4.1.9, "Assembly Liftoff," of TR-0816-51127, Revision 3, provides a fuel assembly liftoff analysis performed in accordance with EMF-92-116(P)(A), Revision 0. The staff finds that, by using bounding flow rates for the limiting AOO, this analysis demonstrates that significant margin to fuel assembly liftoff exists.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements for fuel assembly liftoff in accordance with the guidance in SRP Section 4.2.

Control Rod Insertability. Section 4.3.5, "Fuel Assembly Structural Damage from External Forces," of TR-0816-51127, Revision 3, addresses fuel assembly structural damage, which could prevent control rod insertability. SER Section 4.2.4.5 addresses the evaluation of control rod insertability under seismic and loss-of-coolant-accident (LOCA) loads.

4.2.4.1.2 Fuel Rod Failure

Fuel rod failure should not occur as a result of specific causes during normal operation and AOOs, but it is permitted as a result of postulated accidents. Fuel rod failures can be caused by hydriding, cladding collapse, overheating of the cladding, overheating of the fuel pellet, excessive fuel enthalpy, pellet/cladding interaction, bursting, or mechanical fracturing. The sections below evaluate each of these failure mechanisms.

Hydriding. Section 4.2.1, "Internal Hydriding," of TR-0816-51127, Revision 3, provides an internal hydriding analysis performed in accordance with EMF-92-116(P)(A), Revision 0. The staff finds that, by using fabrication limits for fuel pellet moisture, this analysis demonstrates failure caused by internal hydriding will be precluded.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements for internal hydriding in accordance with the guidance provided in SRP Section 4.2.

Cladding Collapse. Section 4.2.2, "Cladding Collapse," of TR-0816-51127, Revision 3, provides a cladding collapse analysis performed in accordance with BAW-10084P-A, Revision 3, "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," issued July 1995 (ADAMS Accession No. ML14191B170), using the creep model from BAW-10227P-A, Revision 1 (ADAMS Accession No. ML15162B043). This analysis used the initial conditions specified by the methodology in BAW-10231P-A, Revision 1, and relied on maximum calculated fast flux and cladding temperatures at each time step as proposed by the revision to the creep collapse methodology. The staff finds that, by using the CROV code initiated with COPERNIC, this analysis demonstrates that significant margin to creep collapse exists.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements for cladding collapse in accordance with the guidance in SRP Section 4.2.

Overheating of the Cladding. Based on the guidance in SRP Section 4.2, failures are assumed to be precluded if the thermal-margin criteria (departure from nucleate boiling ratio (DNBR)) are satisfied. SRP Section 4.2 also states that violation of the DNBR limits is not allowed for normal operation and AOOs. As shown in DCA Part 2, Tier 2, Table 15.0-2, "Acceptance Criteria-Thermal Hydraulic and Fuel," the applicant self-imposed stricter acceptance criteria; namely, that SAFDLs will also be met for postulated accidents. SER Section 4.4 reviews the

DNBR margin analysis. The various design-basis event (DBE) evaluations, as detailed in SER Chapter 15, review the cladding temperature under postulated accident conditions.

Overheating of the Fuel Pellets. Section 4.2.4, "Overheating of Fuel Pellets," of TR-0816-51127, Revision 3, provides an analysis on the overheating of fuel pellets performed in accordance with BAW-10231P-A, Revision 1. The staff finds that, by using a bounding analysis with the COPERNIC code, this analysis demonstrates significant margin between the NuScale power limits and the fuel melting limits.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements on overheating of the fuel pellets in accordance with the guidance in SRP Section 4.2.

Excessive Fuel Enthalpy. The evaluation in SER Section 15.4.8 covers the review of a sudden increase in fuel enthalpy from a reactivity-initiated accident below the fuel melting temperature.

Pellet/Cladding Interaction. No generic criterion for fuel failure resulting from pellet/cladding interaction or pellet/cladding mechanical interaction exists. SRP Section 4.2 specifies cladding strain and fuel melting limits as a surrogate.

Section 4.2.6, "Pellet/Cladding Interaction," of TR-0816-51127, Revision 3, provides a transient clad strain analysis performed in accordance with BAW-10231P-A, Revision 1. The staff finds that by using a bounding analysis with the COPERNIC code, this analysis demonstrates significant margin between the NuScale power limits and the power level required to reach the 1-percent cladding strain limit specified in SRP Section 4.2.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements for transient cladding strain and fuel melting limits and, therefore, has reasonable assurance that the fuel will not fail as a result of pellet/cladding interaction or pellet/cladding mechanical interaction.

Bursting. SER Section 15.6.5 presents the staff's evaluation of fuel rod bursting.

Mechanical Fracturing. SER Section 4.2.4.5 presents the staff's evaluation of fuel rod mechanical fracturing.

4.2.4.1.3 Fuel Coolability

Some of the damage mechanisms that could result in reduction of fuel coolability, including the overheating of the cladding, excessive fuel enthalpy, bursting, cladding embrittlement, violent expulsion of fuel, generalized cladding melting, and fuel rod ballooning, are addressed in the other DCA chapters or are bounded by other analyses. SER Section 4.2.4.5 evaluates other damage mechanisms, including fuel assembly distortion, mechanical fracturing, and fuel assembly structural damage from external forces, related to the fuel assembly's structural response to externally applied loads.

4.2.4.2 Description and Design Drawings

The staff reviewed the fuel system description and design drawings in DCA Part 2, Tier 2, Section 4.2. TR-0816-51127, Revision 3, contains additional fuel assembly design information. The staff found that the applicant followed the guidance in SRP Section 4.2 by providing an accurate representation of the fuel system; therefore, the staff finds the fuel system description and design drawings acceptable.

4.2.4.3 Design Evaluation

4.2.4.3.1 Operating Experience

Chapter 3, "NuFuel-HTP2™ Fuel Assembly Description," of TR-0816-51127, Revision 3, notes that AREVA's 17 by 17 High Temperature Performance (HTP™) fuel assemblies are similar in material and design to the NuFuel-HTP2™ fuel assemblies. The applicant used the operating experience described in TR-0816-51127, Revision 3, to justify the models used to analyze the NuFuel-HTP2™ fuel assembly for use in the NuScale plant design. The staff evaluated the applicability of the models to the NuScale plant design in its SER for TR-0116-20825-P-A, Revision 1, issued February 2018 (ADAMS Accession No. ML18040B306).

4.2.4.3.2 Testing, Inspection, and Surveillance Plans

SRP Section 4.2 provides review guidance on testing, inspection, and surveillance plans. DCA Part 2, Tier 2, Section 4.2.4.1, "Operating Experience," discusses AREVA's domestic and international operating experience in support of the NuScale fuel design. The staff compared the NuScale fuel assembly components with the AREVA operational fleet database and notes that significant experience has been developed for the same components. The staff further notes that the NuScale plant operational parameters important to fuel behavior are not significantly different from those in the AREVA operating fleet; therefore, the staff finds that the AREVA operating experience applies to NuScale fuel assemblies.

DCA Part 2, Tier 2, Section 4.2.4.2, "Prototype Testing," presents the prototype testing of the NuScale fuel assemblies, CRAs, and fuel assembly components. The testing covers areas related to fuel assembly structural response, which can differ from the full-sized AREVA operational fleet database. The staff reviewed the prototype testing discussed in DCA Part 2, Tier 2, Section 4.2.4.2, and in Chapter 5, "Fuel Assembly Testing," of TR-0816-51127, Revision 3. Based on its review of the information provided by the applicant, the staff finds that the testing follows the methodology provided in referenced and approved topical report ANP-10337P-A, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," Revision 0, issued April 2018 (ADAMS Accession No. ML18144A821), and is therefore acceptable.

4.2.4.4 Testing, Inspection, and Surveillance Plans

DCA Part 2, Tier 2, Section 4.2.4, "Testing and Inspection Plan," contains the testing and inspection plan for the fuel design. Because the NuScale fuel design is similar to existing AREVA 17 by 17 fuel assembly designs, the staff agrees with the applicant that related AREVA operating and testing experience is applicable to NuScale.

DCA Part 2, Tier 2, Section 4.2.4.3, "Manufacturing Testing and Inspection," states that fuel assemblies and CRAs will be manufactured and inspected under a quality assurance program in accordance with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. The component testing under this program includes nondestructive examinations (NDEs) and destructive examinations to support qualifications.

In DCA Part 2, Tier 2, Section 4.2.4, the applicant stated that additional inspections, including onsite receipt inspections, online fuel system monitoring, and postirradiation monitoring, are planned for the fuel assembly and CRAs from the first licensed module.

Based on the description in DCA Part 2, Tier 2, Section 4.2.4.3, the staff finds that the applicant's testing, inspection, and surveillance plans are sufficient to ensure that the fuel is manufactured to the design specifications and that fuel performance outside of the predictions made by the fuel analysis will be detected. The applicant's methods are consistent with the guidance in SRP Section 4.2 and, therefore, are acceptable. The staff notes that a combined license (COL) applicant would be responsible for implementing testing, inspection, and surveillance plans, and the staff would verify such plans at the COL stage.

4.2.4.5 Evaluation of the Fuel Assembly's Structural Response to Externally Applied Forces

Design Requirements and Acceptance Criteria

DCA Part 2, Tier 2, Section 4.2.1.5, "Fuel Assembly Structural Design," defines the bases for the fuel assembly structural design. In DCA Part 2, Tier 2, Section 4.2.1.5.11, "Loss-of-Coolant Accident and Seismic Loading," the DCA states the following:

> The fuel assembly is designed to remain operable during and after an operating basis earthquake (OBE) and to maintain structural integrity, a coolable geometry, and CRA insertion capability during and after a safe shutdown earthquake (SSE) and LOCA.

DCA Part 2, Tier 2, Section 4.2.2, "Description and Design Drawings," and Section 4.2.3, "Design Evaluation," define assembly-component-specific design requirements that fulfill the above high-level design requirement. The DCA refers to TR-0816-51127, Revision 3, for further details regarding these design requirements. Section 4.3.5 of TR-0816-51127 states that specific acceptance criteria for fuel assembly components are identified in AREVA topical report ANP-10337P-A, Revision 0. Section 4 of ANP-10337P-A defines component acceptance criteria that satisfy the underlying regulatory requirements in GDC 2 and Appendix S to 10 CFR Part 50. The staff previously reviewed and accepted the analytical models, testing protocols, and acceptance criteria within ANP-10337P-A. The applicability of ANP-10337P-A to NuScale is documented in TR-0716-50351, Revision 1. TR-0716-50351-P-A includes several modifications to the ANP-10337P-A methods to make it applicable to the NuScale design. The staff's evaluation and acceptance of TR-0716-50351-P-A is provided in the safety evaluation (ADAMS Accession No. ML20122A248).

Design Evaluation

DCA Part 2, Tier 2, Section 4.2.3.4, "Spacer Grids Evaluation," states that the severity of the OBE for the NuScale Power Plant design is one-third of the severity of the SSE. In accordance with 10 CFR Part 50, Appendix S, which states that an OBE does not need to be evaluated if its ground motion is less than or equal to one-third of the SSE ground motion design response spectra, the applicant did not perform a separate OBE evaluation of the fuel assembly. The staff concludes that, with the specification of the OBE as one-third of the SSE, exclusion of an explicit response or design analyses for the OBE is acceptable. SER Section 3.7.1 contains for the staff's safety evaluation of the NuScale design ground motions and seismic analysis.

DCA Part 2, Tier 2, Section 4.2.3.5.2, "Analysis of Combined Loss-of-Coolant Accident and Seismic Loading," describes the fuel assembly structural design evaluation for external loads associated with combined LOCA and SSE. The DCA refers to TR-0816-51127 for further details regarding the design evaluation. Section 4.3.5 of TR-0816-51127 details the model development, testing, and analytical results for the NuFuel-HTP2™ design evaluation. The design analyses cover fuel in both the operating bay and reactor flange tool (RFT) locations.

While generally similar, the operating bay and RFT locations have different operating and physical conditions (e.g., temperature, reduced gap size, stiffness), which are modeled accordingly in their respective analyses. The applicant performed the design evaluations in accordance with the approved AREVA topical report ANP-10337P-A methodology, as modified by TR-0716-50351. From these evaluations, the applicant concluded that the fuel assembly meets structural integrity, control rod insertability, and coolable geometry criteria during and following a LOCA and an SSE. This includes the fuel mechanical fracturing and fuel system distortion criteria discussed in SER Section 4.2.4.1.

The staff used independent confirmatory and sensitivity analyses to assist the review of the NuFuel-HTP2™ design analysis. The independent analyses used the commercially available explicit dynamic finite element code LS-DYNA to recreate the CASAC models used by NuScale to calculate the fuel seismic/LOCA response. The confirmatory analyses included both horizontal and vertical analyses for both the operating bay location and horizontal analyses for the RFT location.

LS-DYNA is more limited than CASAC in prescribing modal damping ratios, but the first and third mode natural frequencies and first mode damping matched the applicant's reasonably well. The horizontal fuel deflection comparison resulted in good agreement between the independent confirmatory analyses and the analyses of record. The impact force calculations resulted in larger disagreement, but both analyses demonstrated margin to the limits. The staff attributes the differences to the known limitation with the LS-DYNA third mode behavior.

The operating bay vertical independent confirmatory analysis demonstrated good agreement. The LS-DYNA and CASAC analyses predicted brief separations between the bottom nozzle and the core plate, but these separations are far too small to dislodge the fuel assembly from its locating features, thereby demonstrating compliance with the fuel assembly liftoff design criterion.

Based on the staff's review of the material presented in TR-0816-51127, the staff concludes that the applicant analyzed the loads, determined the strength, and presented acceptance criteria consistent with the guidance provided in SRP Section 4.2, Appendix A, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces." This was accomplished, in part, by the applicant's use of the NRC-approved methodology described in topical report ANP-10337P-A in analyzing the NuScale fuel assembly response and demonstrating margin to the limits. Furthermore, the staff concludes that the applicant's analysis adequately addressed fuel mechanical fracturing, fuel system distortion, control rod insertability, and coolable geometry under seismic and LOCA loads. Therefore, the staff finds that the NuScale fuel design meets the regulatory requirements in GDC 2 and Appendix S to 10 CFR Part 50.

4.2.4.6 10 CFR 50.46 Exemption Request for M5™

The NuScale fuel design consists of low-enriched UO_2 fuel within an M5™ zirconium-based alloy cladding. The applicant requested an exemption from 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 to permit the use of M5™ alloy fuel rod cladding in its design. In BAW-10227P-A, Revision 1, the staff reviewed and approved the use of M5™ alloy fuel rod cladding and assembly components for batch application. However, exemption from 10 CFR 50.46 and Appendix K to 10 CFR Part 50 is still required because the M5™ alloy fuel rod cladding is not specified in the regulations.

The regulations at 10 CFR 50.12, "Specific Exemptions," state, in part, that 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, which are authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security. The Commission will not consider granting an exemption unless special circumstances are present. In accordance with 10 CFR 50.12(a)(2)(ii), special circumstances are present whenever the 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 10 CFR 50.46 is to establish acceptance criteria for ECCS performance. The staff's review and approval of BAW-10227P-A, Revision 1, addressed all of the important mechanical and material behavior aspects of M5™ with regard to ECCS performance requirements, including (1) the applicability of 10 CFR 50.46(b) fuel acceptance criteria, (2) M5™ material properties such as fuel rod ballooning and rupture strains, and (3) steam oxidation kinetics and applicability of the Baker-Just weight gain correlation. The staff-approved BAW-10240(P)-A, "Incorporation of M5™ Properties in Framatome ANP Approved Methods," issued May 2004 (ADAMS Accession No. ML042800314), further addresses M5™ material properties with regard to LOCA applications.

The results of a recently completed LOCA research program at Argonne National Laboratory (ANL) show that cladding corrosion and associated hydrogen pickup significantly affected postquench ductility. The research identified a new embrittlement mechanism referred to as hydrogen-enhanced beta layer embrittlement. Pretest characterization of irradiated M5™ fuel cladding segments at ANL provides further evidence of favorable corrosion and hydrogen pickup characteristics of M5™ as compared with standard zircaloy. Because of its favorable hydrogen pickup, fuel rods with M5™ zirconium-based alloy cladding are less susceptible to this new embrittlement mechanism.

Furthermore, ANL postquench ductility tests on unirradiated and irradiated M5™ cladding segments demonstrate that the 10 CFR 50.46(b) acceptance criteria (i.e., 1,200 degrees Celsius (C) (2,200 degrees Fahrenheit (F)) and 17-percent equivalent cladding reacted) remain conservative up to current burnup limits. Information in the previously approved M5™ topical reports and recent ANL LOCA research demonstrate that the acceptance criteria in 10 CFR 50.46(b) remain valid for the M5™ alloy and meet the underlying purpose of the rule to maintain a degree of postquench ductility in the fuel cladding material.

In addition, using LOCA models and analysis methods, the applicant's analysis in DCA Part 2, Tier 2, Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," demonstrates that the M5™ fuel rods continue to satisfy 10 CFR 50.46 acceptance criteria. For the reasons above, granting the exemption request will ensure that the NuScale design achieves the underlying purpose of the rule.

Paragraph I.A.5 of Appendix K to 10 CFR Part 50 states that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just equation. Because the Baker-Just equation presumes the use of zircaloy-clad fuel, strict application of the rule would not permit the use of the equation for the advanced zirconium-based M5™ alloy for determining acceptable fuel performance. However, the underlying intent of this portion of Appendix K is to ensure that the analysis of fuel response to LOCAs is conservatively calculated. The approved AREVA topical reports show that, because of the similarities in the chemical composition of the advanced zirconium-based M5™

alloy and zircaloy, the application of the Baker-Just equation in the analysis of the M5™-clad fuel rods will continue to conservatively bound all post-LOCA scenarios.

For the reasons stated above, the staff concludes that application of the requirements in 10 CFR 50.46 and 10 CFR Part 50, Appendix K, is not necessary for the applicant to achieve the underlying purposes of the rules. Granting the exemption request will ensure that the NuScale design achieves the underlying purpose of the rule. In addition, the staff has determined that, under 10 CFR 50.12(a), the exemption is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security. Thus, the staff approves the exemption from the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, to permit the applicant's use of the M5™ alloy fuel rod cladding in its fuel design.

4.2.4.7 Fuel System Design Change Process

In Chapter 7, "Design Change Process," of TR-0816-51127 the applicant presented a fuel mechanical design change process that provides a process by which desired fuel design changes can be evaluated and potentially would not require NRC review and approval before implementation of the new fuel design. The staff reviewed the process to determine its acceptability.

The applicant identified the applicable locations within the DCA and referenced topical and technical reports to find the fuel design criteria for the NuFuel HTP2™ design and the approved methodologies by which compliance with these criteria is evaluated. The applicant provided conditions that must be met for any design change to be made without requiring NRC review and approval.

The staff reviewed the change process and the associated conditions and concludes that if followed as stated, the change process will require any safety-significant fuel design changes to receive NRC review and approval. The staff reaches this conclusion due to the following aspects of the change process:

- The fuel design criteria described in TR-0816-51127-P, Chapter 4, "Design Evaluation," are clearly identified and no changes to the design criteria are allowed without NRC review and approval. In addition, the design criteria must be demonstrated to be valid for the new fuel design.

- The approved methodologies described in TR-0816-51127-P, Chapter 4, used to evaluate the fuel against the fuel design criteria are identified and all conditions/limitations (e.g., fuel burnup limit) to the methodologies must be met. In addition, the methodologies must be demonstrated to be valid for the new fuel design.

- Changes to the HTP2 grid design are limited to changes that do not alter the functional mixing behavior or rod support mechanism.

The staff also concludes that the additional aspects of the design change process provided in Chapter 7 of TR-0816-51127-P provides documentation, quality assurance adherence, testing, and surveillance requirements (as applicable) that ensure new fuel designs are properly tracked and validated.

Based on the staff's evaluation as noted above, the staff finds that, if properly followed, the fuel design change process as presented in Chapter 7 of TR-0816-51127-P meets all regulatory requirements related to fuel design.

4.2.5 Combined License Information Items

Table 4.2-1 lists the relevant COL information item and description from DCA Part 2, Tier 2.

Table 4.2-1 NuScale COL Information Item for DCA Part 2, Tier 2, Section 4.2

COL Item No.	Description	DCA Part 2, Tier 2 Section
4.2-1	A COL applicant that references the NuScale Power Plant design certification and wishes to utilize non-baseload operations will provide justification for the fuel performance codes and methods corresponding to the desired operation.	4.2

The staff evaluated the proposed COL information item and determined that it is acceptable and necessary because all expected operational modes should be considered in fuel design, nuclear, and transient and accident analyses. The applicant did not provide justification that its analysis methodologies are applicable to nonbaseload operations. If a COL applicant proposes to operate in a nonbaseload manner, COL Item 4.2-1 will ensure that effects of such operations are captured in the analysis methodologies.

4.2.6 Conclusion

The staff concludes that the fuel system for the NPM has been designed so that (1) the fuel system will not be damaged as a result of normal operation and AOOs, (2) fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when it is required, and (3) core coolability will always be maintained, even after severe postulated accidents, thereby meeting the related requirements of 10 CFR 50.46; GDC 10, 27, and 35; and 10 CFR 50.34. The staff notes that several of the DBEs are evaluated in the appropriate Chapter 15 section within this SER; therefore, the conclusions regarding regulatory compliance in terms of fuel under those specific postulated accidents are presented in the respective staff safety evaluations. The staff based its conclusion on the following:

- The applicant provided sufficient evidence that these design objectives are met, based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with structural response and fuel densification have been performed in accordance with (1) methods that the staff has reviewed and found to be acceptable and (2) the guidelines in Appendix A to SRP Section 4.2. Those analytical predictions dealing with control rod ejection have been performed in accordance with the interim criteria for reactivity-initiated accidents in Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," to SRP Section 4.2.

- The applicant established plans for the testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The applicant included design features that permit online fuel failure monitoring and defined a postirradiation surveillance program to detect anomalies or confirm that the fuel has performed as

expected. Implementation of these testing and inspection plans will be the responsibility of future COL applicants, and the staff would verify these plans at the COL stage.

The staff concludes that the applicant described methods for adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and thereby meets the related requirements of 10 CFR 50.34.

In addition, based on its review described above regarding fuel assembly structural response to external forces analysis in TR-0816-51127, Revision 3, the staff finds that the NPM fuel system design meets GDC 2 and Appendix S to 10 CFR Part 50.

4.3 Nuclear Design

4.3.1 Introduction

The staff reviewed DCA Part 2, Tier 2, Section 4.3, "Nuclear Design," using the guidance in SRP Section 4.3, "Nuclear Design," issued March 2007 (ADAMS Accession No. ML070740003). The objective of the staff's review is to establish reasonable assurance that fuel design limits will not be exceeded during conditions of normal operation, including AOOs, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair its capability to cool the core.

4.3.2 Summary of Application

DCA Part 2, Tier 1: The following DCA Part 2, Tier 1, information applies to this area of review:

- DCA Part 2, Tier 1, Section 2.1.1, "Design Description," provides a design commitment that the RPV has surveillance capsule holders to hold a capsule that contains RPV material surveillance specimens.

- DCA Part 2, Tier 1, Section 2.2.1, states that each NPM contains a chemical and volume control system (CVCS) that is not safety related and that manages reactor coolant chemistry.

- DCA Part 2, Tier 1, Section 2.5.1, "Design Description," states that the module protection system (MPS) supports the normal direct current power system by removing electrical power to the control rod drive system (CRDS) for a reactor trip. DCA Part 2, Tier 1, Section 2.5.1, also includes a design commitment for the MPS to automatically initiate a reactor trip signal.

- DCA Part 2, Tier 1, Section 2.6.1, "Design Description," states each NPM has its own safety-related neutron monitoring system that monitors the neutron flux level of the reactor core by detecting neutron leakage from the core. DCA Part 2, Tier 1, Section 2.6.1, further states that the safety-related system function of the neutron monitoring system is to support the MPS by providing neutron flux data for various reactor trips.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 4.3, describes the nuclear design of the NPM, as summarized below.

The nuclear design basis in DCA Part 2, Tier 2, Section 4.3.1, "Design Basis," describes the NPM approach to addressing the regulatory criteria identified in SRP Section 4.3. The applicant addressed the following aspects of the nuclear design basis:

- Core average cycle burnup is designed such that the peak rod exposure is less than the approved value in TR-0116-20825. Section 4.2.1, "Safety Evaluation Report," of TR-0116-20825 specifies that the report is limited in application to fuel rod burnups below 62 gigawatt-days per metric ton of uranium.

- The moderator temperature coefficient and Doppler coefficient together provide inherent reactivity control to satisfy GDC 11, "Reactor Inherent Protection."

- The power distribution and the reactor protection system are designed to ensure that SAFDLs are not exceeded at a 95-percent probability at a 95-percent confidence level.

- The maximum CRA withdrawal rate is established such that the CHF limits are not exceeded for an accidental CRA withdrawal. The maximum CRA worth and CRA insertion limits preclude rupture of the RCPB from a rod withdrawal or rod ejection accident.

- The NuScale design uses soluble boron through the CVCS and control rods as the two independent means for reactivity control. The applicant defined shutdown margin (SDM) as the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition, assuming that the moderator temperature is 216 degrees C (420 degrees F) and that all CRAs are fully inserted except for the single assembly of highest reactivity worth, which is assumed to be fully withdrawn. The applicant defined long-term shutdown (LTSD) capability as the amount of reactivity by which the reactor is subcritical or would be subcritical under the assumptions that the core is free of xenon, with no decay heat or voiding present, and with equilibriums of samarium taken into account.

- The design of the NuScale reactor and associated systems and the administrative controls on the CRA position provide an inherently stable core with respect to axial and radial power stability.

DCA Part 2 Tier 2, Section 4.3.2, "Nuclear Design Description," describes the nuclear core design and provides the following additional details that address the design's compliance with the design basis:

- The NuScale core design comprises 37 fuel assemblies with 16 fuel assembly locations that contain CRAs. The 16 CRAs are broken up into two shutdown groups and two regulating groups, with each group containing four CRAs. The fuel rods consist of ceramic pellets of up to 4.95-percent enriched UO_2 with Gd_2O_3 as a burnable absorber and a zirconium-based cladding.

- For each cycle design, a limit is imposed on the maximum-allowed enthalpy rise hot channel factor ($F_{\Delta H}$), which is conservatively applied in the subchannel analysis. Power distributions are conservatively selected for use in transient and accident analyses and are expected to be bounding for all cycles. If the calculated power distributions for a given cycle are not bounded by the values assumed in the transient and accident

analyses, the core design is revised to bring the calculated power distribution within the bounding value, or the affected transient and accident analyses are performed again.

- A startup test program is implemented for the initial startup to confirm that the nuclear design analyses agree with the predictions. Additionally, tests are performed at the beginning of each reload cycle to verify the selected safety-related parameters of the reload design. Five characteristics (i.e., reactivity balance, reactivity control, power distribution, shutdown capability, and shutdown requirement) are confirmed for each newly loaded cycle.

- The in-core instrumentation system (ICIS) consists of 48 self-powered neutron detectors (SPNDs) arranged into 12 instrument strings. Each string of SPNDs is fixed in evenly spaced axial locations. The signals from the SPNDs are synthesized into three-dimensional assembly and peak rod power distributions through the use of prefit coefficient data from detailed SIMULATE5 code calculations.

- The loss of CRA worth resulting from the depletion of the absorber material is negligible. A conservative calculation of a CRA lifetime of over 20 effective full-power years demonstrates that less than 2 percent of the boron in the upper portion of the CRA is lost because of depletion.

The description of the analytical methods in DCA Part 2, Tier 2, Section 4.3.3, "Analytical Methods," states that Studsvik Scandpower Core Management Software simulation tools are used to perform the nuclear analysis and that the Monte Carlo N-Particle Transport Code, Version 6 (MCNP6), is used to perform fluence calculations.

ITAAC: The following ITAAC are evaluated in SER Chapter 14:

- DCA Part 2, Tier 1, Table 2.1-7, ITAAC No. 12, includes a design commitment that the RPV is provided with surveillance capsule holders to hold a capsule that contains RPV material surveillance specimens.

- DCA Part 2, Tier 1, Table 2.5-7, ITAAC No. 10, provides a design commitment that the MPS automatically actuates reactor trip.

Technical Specifications: The following NuScale Generic Technical Specifications (GTS) apply to this area of review:

- GTS 3.1.1, "Shutdown Margin (SDM)"
- GTS 3.1.2, "Core Reactivity"
- GTS 3.1.3, "Moderator Temperature Coefficient (MTC)"
- GTS 3.1.4, "Rod Group Alignment Limits"
- GTS 3.1.5, "Shutdown Group Insertion Limits"
- GTS 3.1.6, "Regulating Group Insertion Limits"
- GTS 3.1.7, "Rod Position Indication (RPI)"
- GTS 3.1.9, "Boron Dilution Control"
- GTS 3.2.1, "Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)"
- GTS 3.2.2, "Axial Offset (AO)"
- GTS 3.4.2, "Reactor Coolant System (RCS) Minimum Temperature for Criticality"
- GTS 3.5.3, "Ultimate Heat Sink"
- GTS 5.6.3, "Core Operating Limits Report (COLR)"

Technical Reports: DCA Part 2, Tier 2, Table 1.6-2, identifies TR-0116-20781, "Fluence Calculation Methodology and Results," Revision 1, issued July 2019 (ADAMS Accession No. ML19183A485), as incorporated by reference into DCA Part 2, Tier 2, Section 4.3.

4.3.3 Regulatory Basis

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

- GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.

- GDC 11 requires that the reactor core and associated coolant systems be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

- GDC 12, "Suppression of Reactor Power Oscillations," requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations that can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.

- GDC 13, "Instrumentation and Control," requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the RCPB, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within the prescribed operating ranges.

- GDC 25, "Protection System Requirements for Reactivity Control Malfunctions," requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

- GDC 26, "Reactivity Control System Redundancy and Capability," requires that two independent reactivity control systems of different design principles be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that, under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, SAFDLs are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure that the acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

- GDC 27 requires that the reactivity control systems be designed to have a combined capability of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

- GDC 28, “Reactivity Limits,” requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than the limited local yielding nor sufficiently disturb the core, its support structures, or other RPV internals to significantly impair the capability to cool the core. These postulated reactivity accidents shall consider rod ejection (unless prevented by positive means), rod dropout, steamline rupture, changes in reactor coolant temperature and pressure, and the addition of cold water.

4.3.4 Technical Evaluation

4.3.4.1 Power Distributions

DCA Part 2, Tier 2, Section 4.3.1.3, “Power Distribution,” states that the design basis for the nuclear design of the NPM is that the power distribution and the reactor protection system are designed to ensure that SAFDLs are met at a 95-percent probability at a 95-percent confidence level. DCA Part 2, Tier 2, Section 4.3.2.2, “Power Distribution,” further describes the design basis. DCA Part 2, Tier 2, Section 4.3.2.2.6, “Limiting Power Distributions,” clarifies that the applicant used limiting power distributions in the steady-state and transient analyses to ensure that SAFDLs are not exceeded during normal operations and AOOs. DCD Part 2, Tier 2, Figure 4.3-3,” Axial Offset Window,” shows the analytical axial offset window, which was developed to encompass axial offsets achievable during normal operation by considering depletion over various durations. The applicant stated that, for each cycle core design, a limit is imposed on the $F_{\Delta H}$, which is conservatively applied in the safety analysis. Additionally, the applicant stated that an analysis of possible axial powers is performed to identify the bounding axial power shapes for use in the CHF and transient analyses. The NRC staff audited the calculations supporting DCA Part 2, Tier 2, Chapter 4, and examined the AO window (ADAMS Accession No. ML18025B026). During this audit, the NRC staff observed that the applicant used its nuclear design methodology to perform evaluations over a large range of possible operating conditions (e.g., power, time in cycle, CRA insertion, perturbed conditions), which show that the AO window is maintained within the bounds assumed in the safety analysis.

TR-0915-17564-P-A, “Subchannel Analysis Methodology,” Revision 2, issued February 2019 (ADAMS Accession No. ML19067A256), which the NRC staff has reviewed and approved (ADAMS Accession No. ML18338A031), describes the method for applying the power distribution in the safety analysis in detail. In particular, the NRC staff’s safety evaluation for TR-0915-17564-P-A found the applicant’s approach for using bounding radial and axial power distributions acceptable. Additionally, the NRC staff recognizes that verification of the power distribution during operation is performed in accordance with GTS 3.2.1 and GTS 3.2.2. Based on the information discussed in this section and the analytical methods discussed in SER Section 4.3.4.7, the NRC staff finds the power distributions acceptable because (1) the safety analyses apply a conservatively bounding power distribution when evaluating thermal-margin, (2) the applicant used an approved core design methodology to perform analyses that demonstrate operation within the bounding power distributions used in the safety analyses, and (3) operation within the bounding power distributions used in the safety analyses is verified in accordance with GTS 3.2.1 and GTS 3.2.2.

4.3.4.2 Reactivity Coefficients

DCA Part 2, Tier 2, Section 4.3.1.2, “Negative Reactivity Feedback,” states that the Doppler coefficient and the moderator temperature coefficient (MTC) are the two primary reactivity feedback mechanisms that compensate for a rapid reactivity increase, provide inherent reactivity control, and satisfy GDC 11. The combination of the Doppler coefficient and the MTC

should ensure that the overall reactivity coefficient associated with an increase in core power is negative. DCA Part 2, Tier 2, Figure 4.3-13, "Moderator Temperature Coefficient of Reactivity at Full Power" and Figure 4.3-14, "Moderator Temperature Coefficient of Reactivity at Zero Power," provide values for the MTC. The NRC staff has conducted confirmatory analyses that predicted values for the MTC that were bounded by the applicant's results. Additionally, the applicant presented values for the power coefficient in DCA Part 2, Tier 2, Figure 4.3-16, "Maximum and Minimum Power Coefficient," and boron reactivity worth in DCA Part 2, Tier 2, Figure 4.3-21, "Boron Worth Coefficient." The applicant's analysis shows that the power coefficient is negative for all power levels. The applicant obtained the results using the analytical methods discussed in SER Section 4.3.4.7 that the NRC staff has reviewed and approved (ADAMS Accession No. ML18234A295). Additionally, as discussed in Section 3.5.2 of the NRC staff's safety evaluation for TR-0616-48793-P-A, "Nuclear Analysis Codes and Methods Qualification," Revision 1, issued December 2018 (ADAMS Accession No. ML18348B036), the MTC, power coefficient, fuel coefficient (i.e., the Doppler coefficient), and kinetics parameters are adequately verified during startup testing and GTS surveillance (i.e., GTS 3.1.3). Based on the information provided DCA Part 2, Tier 2, Section 4.3.1.2 and Section 4.3.4.7, and the information provided in Section 3.5.2 of the staff's SER for TR-0616-48793, the NRC staff finds that the NuScale reactor core and associated coolant system are designed such that prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity because (1) the applicant's analyses show that the power coefficient is negative for all power levels, (2) the applicant performed the analyses using an approved methodology, and (3) reactivity coefficients are adequately verified through startup testing and GTS surveillance.

4.3.4.3 Reactivity Control

DCA Part 2, Tier 2, Section 4.3.1.5, "Shutdown Margin and Long Term Shutdown Capability," states that the NuScale design uses two independent means for reactivity control: (1) CRAs and (2) soluble boron through the CVCS. Based on the description of the CRAs and CVCS, the NRC staff finds that the NPM design provides for two independent reactivity control systems of different design principles because the CRAs are control rods and because the CVCS uses soluble boron.

DCA Part 2, Tier 2, Section 4.3.1.5, defines SDM as the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition, assuming that the moderator temperature is 216 degrees C (420 degrees F) and that all CRAs are fully inserted except for the single assembly of highest reactivity worth, which is assumed to be fully withdrawn. The staff finds the temperature threshold of 216 degrees C (420 degrees F) for defining SDM to be consistent with the safe-shutdown requirements for passive designs specified in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994. DCA Part 2, Tier 2, Section 4.3.1.5, further states that, for AOOs, rapid CRA insertion following a reactor trip protects the SAFDLs. (SER Chapter 15 evaluates the transient and accident analyses.) The NRC staff audited the calculations supporting DCA Part 2, Tier 2, Chapter 4, which included an SDM calculation (ADAMS Accession No. ML18025B026). During this audit, the NRC staff observed that the applicant performed the calculation consistent with the definition of SDM and that the results of the calculation showed that the nuclear design produced margin with respect to the SDM acceptance criteria. The NRC staff recognizes that SDM is verified in accordance with GTS 3.1.1. Based on the information described in this paragraph, the NRC staff finds that the control rods are capable of reliably controlling reactivity changes under conditions of normal operation, including AOOs, and with appropriate margin for stuck rods, such that SAFDLs are not exceeded, in part, because SDM ensures that the reactor

can be brought to a shutdown state. SER Chapter 15 evaluates additional considerations with regard to the integrated NPM design margin to ensure that SAFDLs are not exceeded during AOOs (e.g., CRD insertion time, heat removal capabilities, and margin to thermal limits).

DCA Part 2, Tier 2, Section 4.3.1.5, states that both the CRAs and CVCS are capable of controlling reactivity changes resulting from planned, normal operation. Additionally, DCA Part 2, Tier 2, Section 4.3.1.5, states that the CVCS is used to adjust soluble boron concentration to account for reactivity changes resulting from core burnup and power maneuvering to maintain the CRAs within the power-dependent insertion limits (PDILs). DCA Part 2, Tier 2, Section 4.3.1.4, "Maximum Controlled Reactivity Insertion," clarifies that the maximum controlled reactivity addition rate is limited, such that the SAFDLs are not violated during normal operation, AOOs, or postulated accidents. (SER Section 15.4 evaluates reactivity and power distribution anomalies.) Based on the information described in this paragraph, the NRC staff finds that the CVCS is capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes to assure that acceptable fuel design limits are not exceeded because the maximum rate of reactivity insertion within the capacity of the CVCS does not result in SAFDLs being exceeded.

In DCA Part 2, Tier 2, Section 4.3.1.5, the applicant defined LTSD as the amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition, assuming that all CRAs are fully inserted, the core is free of xenon, no decay heat or voiding is present, and equilibrium concentrations of samarium are taken into account. The staff audited the calculations supporting DCA Part 2, Tier 2, Chapter 4, which included an LTSD calculation (ADAMS Accession No. ML18025B026). During this audit, the staff observed that the applicant performed the calculation consistent with the definition of LTSD and that the results of the calculation showed that the nuclear design produced margin with respect to the LTSD acceptance criteria. Additionally, the staff performed confirmatory analyses that produced results that were consistent with the applicant's results. DCA Part 2, Tier 2, Section 9.3.4.2.1, "Chemical and Volume Control System," further states that the boron addition system is managed to maintain a sufficient quantity of 2,000 parts per million (ppm) boron to ensure the ability to support a shutdown of 12 NPMs. Based on the information described in this paragraph, the NRC staff finds that the CVCS and CRAs are each capable of holding the reactor subcritical under cold conditions because (1) the definition of LTSD ensures the ability of the CRAs to hold the reactor subcritical under cold conditions with all CRAs fully inserted, (2) the applicant's analysis shows that the reactor core is designed to satisfy the LTSD acceptance criteria, and (3) the CVCS is capable of providing highly concentrated boron sufficient to ensure subcriticality under cold conditions.

DCA Part 2, Tier 2, Section 4.3.1.5, states that, for DBEs, the insertion of all CRAs provides the safety-related means to shut down the reactor and maintain it in a shutdown condition. DCA Part 2, Tier 2, Section 4.3.1.5, further states that the conservative analysis, which accounts for a stuck CRA and does not credit the CVCS, indicates that a return to power could occur following a reactor trip. The applicant requested an exemption to GDC 27 and proposed PDC 27. SER Section 15.0.6 provides the staff's evaluation of this exemption and proposed PDC.

DCA Part 2, Tier 2, Section 4.3.1.5, states that LOCA events could result in condensation of unborated water in the containment vessel and RPV downcomer once the steam generator tubes become uncovered. Under certain conditions, reduced boron concentration in the downcomer could cause a positive reactivity insertion when ECCS actuates. DCA Part 2, Tier 2, Section 4.3.1.5, states that ECCS actuation signals on high containment water level and low RCS pressure are sufficient to preclude a core dilution event following the ECCS actuation. The

staff evaluation of the ECCS signal effectiveness to prevent a core dilution event following a LOCA is in Section 15.6.5 of this SER.

DCA Part 2, Tier 2, Section 4.3.1.5, states that, for some non-LOCA scenarios, condensation of steam could reduce the downcomer boron concentration after the DHRS cools the RCS sufficiently to cause RCS level to drop below the top of the riser. Reduced boron concentration in the downcomer could, under certain conditions, cause a positive reactivity insertion when natural circulation is restored. As discussed in SER Section 5.4.1.2, the NPM design includes four small holes in the riser to promote mixing in the downcomer and mitigate a core dilution event under riser uncovery conditions. The staff evaluation of the riser hole effectiveness to prevent unacceptable levels of downcomer dilution is in Section 15.0.5 of this SER.

The staff notes that breaks smaller than that analyzed in the LOCA spectrum (i.e., RCS leakage cases that are smaller than the total CVCS makeup capacity) can result in riser uncovery and boron dilution similar to the scenarios described above (in circumstances where CVCS is not available). The applicant evaluated RCS leakage in DCA Part 2, Tier 2, Section 6.3.1 to support the exemption to GDC 33. The staff evaluation of the ECCS and riser hole effectiveness to prevent a core dilution event, as it relates to the applicant's exemption to GDC 33, is in Section 6.3 of this SER.

Post-event recovery actions with respect to boron distribution, from both LOCA and non-LOCA events, are important to ensure that a core dilution event is prevented. DCA Part 2, Tier 2, Section 15.0.4, states that the fluid boron concentration and boron distribution in the NPM are important when exiting passive ECCS and DHRS cooling modes and need to be accounted for to ensure shutdown margin limits are preserved. The staff notes that these post-event recovery actions are outside the scope of the design certification review but are important to capture in the development of operating procedures. The applicant included COL item 13.5-2 in DCA Part 2, Tier 2, Section 13.5.2, "Operating and Maintenance Procedures," for development of operating procedures at a future licensing stage.

4.3.4.4 Control Rod Patterns and Reactivity Worths

DCA Part 2, Tier 2, Section 4.3.1.4, states that the NuScale design places limits on the worth of the CRAs, CRA insertion depth, and maximum CRA withdrawal rate. DCA Part 2, Tier 2, Section 4.3.2.1, "Nuclear Design Description," states that 16 CRAs are broken up into two shutdown groups and two regulating groups and that each group contains four CRAs. DCA Part 2, Tier 2, Section 4.3.2.1, further clarifies that the shutdown groups are fully withdrawn during operation and that both regulating groups move together until the Group 2 PDIL is reached; once both groups reach the Group 2 PDIL, Group 1 can be inserted further up to the Group 1 PDIL. DCA Part 2, Tier 2, Section 4.3.2.4.12, "Control Rod Assemblies," further discusses this by stating that CRA insertion is restricted to ensure that sufficient negative reactivity is available to maintain shutdown capability and to limit the amount of reactivity insertion possible during a rod ejection event. DCA Part 2, Tier 2, Figure 4.3-2, "Power Dependent Insertion Limits," shows the PDILs, and DCA Part 2, Tier 2, Figure 4.3-18, "Control Rod and Incore Instrument Locations," shows the CRA locations and group structures.

The NRC staff audited the calculations supporting DCA Part 2, Tier 2, Chapter 4, which included the process used to set the PDILs (ADAMS Accession No. ML18025B026). In its audit report, the NRC staff states that the applicant set and verified the AO window and PDILs using several calculations to ensure that acceptance criteria are satisfied for the SDM analysis and the accident analyses in DCA Part 2, Tier 2, Chapter 15. SER Section 15.4.8 evaluates the rod ejection accident, which can limit CRA insertion. Based on the description in DCA Part 2,

Tier 2, Section 4.3.2.1, and the analyses that set the PDILs, the NRC staff finds that the applicant established adequate PDILs for use in accident and transient analyses. Additionally, the NRC staff has determined that GTS 3.1.5 and GTS 3.1.6 verify the position of the CRAs.

DCA Part 2, Tier 2, Figure 4.3-20, "Integral Regulating Bank Worth for Withdrawal from the Power Dependent Insertion Limits," provides the integral bank worths for the regulating banks. The NRC staff performed confirmatory analyses as part of its review and obtained values for individual rod worths at the beginning of cycle, middle of cycle, and end of cycle that were consistent with the applicant's analyses. The applicant obtained the results using the analytical methods discussed in SER Section 4.3.4.7, which the NRC staff has reviewed and approved (ADAMS Accession No. ML18234A295). Additionally, the NRC staff audited the calculations supporting DCA Part 2, Tier 2, Chapter 4, which included a CRA depletion analysis (ADAMS Accession No. ML19092A361). During the audit, the NRC staff noted that (1) the CRA depletion analysis showed little loss of burnable absorber over a lifetime of 20 effective full-power years and (2) uncertainties associated with CRA worth are accommodated in accordance with the core design methodology (ADAMS Accession No. ML16243A517).

DCA Part 2, Tier 2, Section 15.4.1.2, "Sequence of Events and Systems Operation," states that the transient analyses assume a maximum allowed CRA withdrawal rate of 38 centimeters (15 inches) per minute, and DCA Part 2, Tier 2, Table 14.2-80, "Control Rod Drive System—Manual Operation, Rod Speed, and Rod Position Indication Test # 80," states that Test No. 80 will verify that the rod insertion and withdrawal speeds are within design limits. The staff confirmed that DCA Part 2, Tier 2, Section 4.3.1.4, also specifies that the maximum rod withdrawal rate is 38 centimeters (15 inches) per minute. The NRC staff finds this design information acceptable because it provides the maximum design-basis CRA withdrawal rate consistent with the Chapter 15 transient analyses. Additionally, DCA Part 2, Tier 2, Table 15.0-1, "Design Basis Events," categorizes the uncontrolled CRA withdrawal events resulting from a malfunction of the reactivity control system as AOOs. The NRC staff recognizes that, as required by GDC 10, AOO acceptance criteria prohibit the violation of SAFDLs.

Based on the information discussed in this section, the NRC staff finds that the control rod patterns and reactivity worths are sufficient to ensure adequate SDM and LTSD and to provide conservative inputs to the safety analyses in DCA Part 2, Tier 2, Chapter 15.

4.3.4.5 Criticality during Refueling

DCA Part 2, Tier 2, Section 4.3.2.6, "Criticality of the Reactor During Refueling," states that maintaining an effective neutron multiplication factor of 0.95 or less at all times prevents criticality during refueling, that refueling is performed with CRAs inserted in the fuel assemblies, and that the calculated required boron concentration for refueling assumes that the two highest worth CRAs are not inserted. The NRC staff finds that this describes a conservative approach for preventing criticality during refueling because it establishes additional margin by not inserting a CRA (for the case in which a fuel assembly that contains a CRA is being moved) and an additional 5,000-percent-mil margin from criticality. Additionally, Appendix B to TR-1116-52011, "Technical Specifications Regulatory Conformance and Development," Revision 4, issued May 2020 (ADAMS Accession No. ML20141L804), states that GTS 3.5.3 establishes limits on boron concentration in the refueling area as well as the rest of the pool. Furthermore, GTS 5.6.5 clarifies that the bulk average boron concentration limit is established using the methods described in DCA Part 2, Tier 2, Section 4.3. Based on the conservative approach for preventing criticality and GTS 3.5.3, the NRC staff finds reasonable assurance that the NuScale design prevents criticality during refueling.

4.3.4.6 Stability

DCA Part 2, Tier 2, Section 4.3.1.6, "Stability," states that the design basis for the reactor and associated systems is to provide an inherently stable core with respect to axial and radial power stability. DCA Part 2, Tier 2, Section 4.3.2.7, "Stability," evaluates xenon-induced power distribution oscillations. The applicant performed evaluations using the SIMULATE5 code (see SER Section 4.3.4.7) at various times in the core's life and from 25- to 100-percent power. The applicant induced xenon oscillations through the insertion of control rods. DCA Part 2, Tier 2, Table 4.3-10, "Stability Indices for Radial Oscillation due to Radial Perturbation" and Table 4.3-11, "Stability Indices for Axial Oscillation due to Radial Perturbation," provide the results of the applicant's analyses, which show that the reactor was stable over this configuration. Additionally, the staff audited the calculations supporting DCA Part 2, Tier 2, Chapter 4, which included the xenon stability analyses (ADAMS Accession No. ML19092A361). During this audit, the NRC staff observed that the applicant performed the xenon stability analyses using bounding CRA insertions and that the analyses produced results that are consistent with the information presented in DCA Part 2, Tier 2, Table 4.3-10 and Table 4.3-11. Based on the information discussed in this section, the staff finds that the NuScale design is inherently stable with respect to axial and radial power stability because (1) the applicant performed conservative stability analyses using an approved analytical method, and (2) the analyses showed that the reactor stabilizes for all perturbations. SER Section 4.4.4.8 evaluates additional stability considerations.

4.3.4.7 Analytical Methods

DCA Part 2, Tier 2, Section 4.3.3, discusses the analytical methods used by the applicant to analyze the nuclear design. The applicant used the Studsvik Scandpower Core Management Software simulation tools, including CASMO5, CMSLINK5, SIMULATE5, and S3K, to perform steady-state and transient neutronic analysis. TR-0716-50350-P-A, "Rod Ejection Accident Methodology," Revision 1, dated June 16, 2020 (ADAMS Accession No. ML20168B203), describes the applicant's use of these methods in detail. The NRC staff has reviewed and approved TR-0616-48793-P-A for the design and analysis of the NuScale reactor core (ADAMS Accession No. ML18348B036). The staff has reviewed and approved TR-0716-50350-P-A, subject to the limitations and conditions in the safety evaluation (ADAMS Accession No. ML20168B203).

Additionally, the applicant stated that MCNP6, Version 1.0, with cross-sections based on Evaluated Nuclear Data File (ENDF)/B-VII, is used to perform vessel fluence calculations. The staff recognizes that MCNP is a tool that is frequently used in the analysis of particle transport and has been previously approved for use in performing vessel fluence analyses. Based on the previous approval of MCNP for use in similar analyses, the NRC staff finds the use of MCNP6 acceptable for use in performing vessel fluence analyses.

4.3.4.8 Vessel Fluence

DCA Part 2, Tier 2, Section 4.3.2.8, "Vessel Irradiation," discusses the vessel fluence analysis with the results in DCA Part 2, Tier 2, Table 4.3-12, "Typical Fast Neutron Flux Levels (n/cm2-sec) in the Reactor Core and Reactor Pressure Vessel at Full Power." TR-0116-20781 provides the details of the fluence calculations. The NRC staff compared TR-0116-20781 against the guidance in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," issued March 2001 (ADAMS Accession No. ML010890301), and determined that the applicant performed the analysis consistent with RG 1.190 using several alternative approaches identified in Appendix C to TR-0116-20781. The NRC staff reviewed the alternatives that the applicant used and determined that they were reasonable because the applicant used sensitivity analyses to quantify the impact of each

alternative and accounted for the impact of these alternatives in the evaluation of fluence biases and uncertainties. Additionally, the applicant has a material surveillance program, as discussed in DCA Part 2, Tier 2, Section 5.3.1.6, "Material Surveillance," to monitor changes in fracture toughness properties. The NRC staff recognizes that the surveillance capsule withdrawal schedule in DCA Part 2, Tier 2, Table 5.3-5, "Surveillance Capsule Withdrawal Schedule," obtains samples as early as 3.5 effective full-power years from the start of NPM operation and serves as a means of validation for fluence analyses. Based on conformance to the guidance in RG 1.190 with justified alternatives and on the material surveillance program, the NRC staff finds the vessel fluence analysis acceptable.

4.3.4.9 Technical Specifications

The NRC staff reviewed the applicable TS identified in SER Section 4.3.2 to ensure that the plant will be operated within the bounds of the safety analyses. NuScale GTS 5.6.3, paragraph a, states that the core operating limits shall be established before each reload cycle or before any remaining portion of a reload cycle and shall be documented in the COLR for the following:

- GTS 3.1.1, "Shutdown Margin (SDM)"
- GTS 3.1.3, "Moderator Temperature Coefficient (MTC)"
- GTS 3.1.4, "Rod Group Alignment Limits"
- GTS 3.1.5, "Shutdown Group Insertion Limits"
- GTS 3.1.6, "Regulating Group Insertion Limits"
- GTS 3.1.8, "Physics Tests Exceptions"
- GTS 3.1.9, "Boron Dilution Control"
- GTS 3.2.1, "Enthalpy Rise Hot Channel Factor ($F_{\Delta H}$)"
- GTS 3.2.2, "AXIAL OFFSET (AO)"
- GTS 3.4.1, "RCS Pressure and Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits"
- GTS 3.5.3, "Ultimate Heat Sink"

Under 10 CFR 50.36(c)(2)(ii)(B), the NRC requires establishment of a TS limiting condition for operation (LCO) for a "process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier." The NuScale GTS contain several LCOs that reference limits specified in the COLR, which is a defined term in GTS 1.1, "Definitions," and which is specified in GTS 5.6.3. The NRC staff finds this acceptable because each LCO that references the COLR is required by GTS 5.6.3.b to use NRC-reviewed and -approved methods when establishing its limit.

The staff reviewed the scope of applicable GTS identified for the nuclear design and noted that DCA Part 2, Tier 2, Section 4.3.2.2.1, "Definitions," states that the heat flux hot channel factor (F_Q) is used to ensure that the SAFDL for fuel centerline melting is not exceeded. However, the

NuScale GTS do not include an LCO for F_Q. By letter dated June 12, 2018 (ADAMS Accession No. ML18163A417), the applicant updated DCA Part 2, Tier 2, Section 4.3.2.2.1, to clarify that F_Q is not used as an initial condition for any transient or design-basis accident and that an LCO for F_Q is therefore not needed. The NRC staff agrees that an LCO for F_Q is not needed because (1) the verification of the power distribution does not rely on F_Q as described in SER Section 4.3.4.1, and (2) the power distribution used in the LOCA evaluation model, TR-0516-49422-A, "Loss-of-Coolant Accident Evaluation Model," Revision 2, issued July 2020 (ADAMS Accession No. ML20189A644), does not specify F_Q as an input parameter. The staff concluded that the applicant identified an appropriate scope of GTS for the nuclear design, and therefore the requirements of 10 CFR 50.36(c)(2)(ii)(B) are met.

4.3.4.10 Testing and Verification

DCA Part 2, Tier 2, Section 4.3.2.2.7, "Verification of Power Distribution Analysis," discusses the benchmarking process used to develop nuclear reliability factors that are used to account for uncertainties in reactor physics parameters and power distributions. TR-0616-48793 details the nuclear reliability factor development and update methodology that the NRC staff has reviewed and approved (ADAMS Accession No. ML18234A295).

4.3.4.11 In-Core Neutron Flux and Temperature Monitoring

DCA Part 2, Tier 2, Section 4.3.2.2.9, "Monitoring," discusses the ICIS. In SER Section 3.5.3.7 for TR-0616-48793, the NRC staff considered the design of the ICIS on the reactor core design and, in particular, on the uncertainty associated with evaluating pin peaking factors. Based on its previous review as part of TR-0616-48793, the NRC staff finds the ICIS design acceptable, because the nuclear design quantifies and accommodates uncertainties associated with ICIS measurements.

4.3.5 Combined License Information Items

No COL information items are associated with DCA Part 2, Tier 2, Section 4.3.

4.3.6 Conclusion

Based on the NRC staff's technical review for the nuclear design of the NPM, as documented in SER Section 4.3.4, the NRC staff makes the following conclusions:

- The nuclear design for the NPM satisfies GDC 10 for the following reasons:
  - The applicant used an approved analytical method to calculate the power distributions, reactivity coefficients, and SDM (see SER Section 4.3.4.7).
  - The safety analyses use bounding power distributions (see SER Section 4.3.4.1).
  - The control rods are capable of reliably controlling reactivity changes under conditions of normal operation, including AOOs, and with appropriate margin for stuck rods, such that SAFDLs are not exceeded (see SER Section 4.3.4.3).
- The nuclear design for the NPM satisfies GDC 11 because the reactor core and associated coolant system are designed such that prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity (see SER Section 4.3.4.2).

- The ICIS for the NPM satisfies GDC 13 because the nuclear design quantifies and accommodates uncertainties associated with ICIS measurements (see SER Section 4.3.4.11).

- The nuclear design of the NPM satisfies GDC 25 because the maximum design-basis CRA withdrawal rate is specified, tested, and evaluated in the reactivity malfunction AOOs. This AOO evaluation uses SAFDLs as acceptance criteria (see SER Section 4.3.4.4).

- The nuclear design of the NPM satisfies GDC 26 for the following reasons:

  – The NPM design provides for two independent reactivity control systems of different design principles in the CRAs and the CVCS (see SER Section 4.3.4.3).

  – The CRAs are capable of reliably controlling reactivity changes under conditions of normal operation, including AOOs, and with appropriate margin for stuck rods, such that SAFDLs are not exceeded (see SER Section 4.3.4.3).

  – The CVCS is capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded (see SER Section 4.3.4.3).

The CVCS and CRAs are each capable of holding the reactor subcritical under cold conditions (see SER Section 4.3.4.3).

- The nuclear design of the NPM satisfies PDC 27 because, under postulated accident conditions and with appropriate margin for a stuck rod, the capability to cool the core is maintained, and with all control rods inserted, the reactor remains subcritical under cold conditions.

- The nuclear design of the NPM satisfies GDC 28 because appropriate limits are established for the potential amount and rate of reactivity increase.

4.4 Thermal-Hydraulic Design

4.4.1 Introduction

The staff reviewed DCA Part 2, Tier 2, Section 4.4, “Thermal and Hydraulic Design,” using the guidance in Section 4.4, “Thermal and Hydraulic Design,” of the “Design-Specific Review Standard for NuScale SMR Design,” Revision 0, issued June 2016 (ADAMS Accession No. ML15355A468) (DSRS). The objective of the staff’s review is to establish reasonable assurance that the applicant used acceptable analytical methods to conduct the thermal-hydraulic design, that the design provides acceptable margins of safety from conditions that would lead to fuel damage during normal operation and AOOs, and that the design is not susceptible to thermal-hydraulic instability.

4.4.2 Summary of Application

DCA Part 2, Tier 1: No DCA Part 2, Tier 1, information is associated with this area of review.

DCA Part 2, Tier 2: The thermal-hydraulic design basis in DCA Part 2, Tier 2, Section 4.4.1, describes the NPM approach for addressing the regulatory criteria identified in DSRS

Section 4.4. The applicant addressed the following aspects of the thermal-hydraulic design basis:

- NuScale-specific CHF correlations, NSP2 and NSP4, are used to ensure, with a 95-percent probability at a 95-percent confidence level, that CHF does not occur during normal operation and AOOs.
- The fuel melting temperature is not exceeded in any part of the core during normal operation and AOOs.
- The design-basis core bypass flow of 8.5 percent accounts for flow through the fuel assembly guide tubes, the reflector block, and the gap between the reflector block and core barrel.
- The hydrodynamic stability design basis is that normal operation and AOOs do not lead to hydrodynamic instability.

DCA Part 2, Tier 2, Section 4.4.2, "Description of Thermal and Hydraulic Design of the Reactor Core," describes the thermal-hydraulic design of the reactor core and provides the following details:

- the CHF and linear heat generation rate
- the core flow distribution, core pressure drops, and hydraulic loads
- correlation and physical data
- the basis for no thermal-margin trip in the NPM design
- uncertainties in estimates and flux tilt considerations

DCA Part 2, Tier 2, Section 4.4.3, "Description of the Thermal and Hydraulic Design of the Reactor Coolant System," describes the thermal-hydraulic design of the RCS and provides details on core bypass flow, operating restrictions, and thermal-margin limits.

DCA Part 2, Tier 2, Section 4.4.4, "Evaluation," describes the thermal-hydraulic evaluation and includes information on analytical models and inputs.

DCA Part 2, Tier 2, Section 4.4.5, "Testing and Verification," briefly discusses testing and verification.

DCA Part 2, Tier 2, Section 4.4.6, "Instrumentations Requirements," states that temperature is continuously monitored at the inlet and outlet of the 12 fuel assemblies identified in DCA Part 2, Tier 2, Figure 4.3-18, as in-core instrumentation locations. Additionally, DCA Part 2, Tier 2, Section 4.4.6, explains why the design does not provide a loose parts monitoring system (LPMS) for the NPM.

DCA Part 2, Tier 2, Section 4.4.7, "Flow Stability," describes the flow stability evaluation for the NPM, including instability mode classification, analysis methodologies, and stability protection. DCA Part 2, Tier 2, Section 4.4.7, states that DCA Part 2, Tier 2, Section 15.9, "Stability," demonstrates that the NPM-specific design is protected from unstable flow oscillations when operation is limited to a defined pressure-temperature exclusion zone.

ITAAC: No ITAAC are associated with this area of review.

Technical Specifications: The following GTS are applicable to this area of review:

- GTS 2.0, “Safety Limits.”
- GTS 3.3.1, “Module Protection System (MPS) Instrumentation.”
- GTS 3.4.1, “RCS Pressure and Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits”
- GTS 5.5.10, “Setpoint Program (SP).”
- GTS 5.6.3, “Core Operating Limits Report (COLR)”

Technical Reports: No TRs are incorporated by reference into DCA Part 2, Tier 2, Section 4.4.

4.4.3 Regulatory Basis

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

- GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.
- GDC 12 requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations that can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.
- 10 CFR 50.34(f)(2)(xviii) requires that instruments be provided in the control room that give an unambiguous indication of inadequate core cooling (ICC), such as primary coolant saturation meters in pressurized-water reactors (PWRs), and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and boiling-water reactors.

The guidance in DSRS Section 4.4 lists the following acceptance criteria that are adequate to meet the above requirements, as well as review interfaces with other SRP sections:

- There should be a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a boiling crisis during normal operation or AOOs.
- Problems affecting CHF, such as fuel densification or rod bowing, are accounted for by an appropriate design penalty, which is determined experimentally or analytically.
- Analysis codes that are appropriate for the NuScale design should be used to calculate local fluid conditions within fuel assemblies for use in CHF correlations.
- The design should address core oscillations and thermal-hydraulic instabilities.
- Methods for calculating single-phase and two-phase fluid flow in the reactor vessel and other components should include classical fluid mechanics relationships and appropriate empirical correlations.
- The proposed TS should ensure that the plant can be safely operated at steady-state conditions under all expected combinations of system parameters. The safety limits and

limiting safety settings must be established for each parameter or combination of parameters to provide a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a boiling crisis during normal operation or AOOs.

- Preoperational and initial startup test programs should follow the recommendations of RG 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," as it relates to measurements and the confirmation of thermal-hydraulic design aspects.

- The design description and proposed procedures for use of the LPMS should be consistent with the requirements of RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

- The thermal-hydraulic design should account for the effects of crud in the CHF calculations in the core or in the pressure drop throughout the RCS. Process monitoring provisions should assure the capability to detect degradation in the reactor coolant flow. The flow should be monitored every 24 hours.

- Instrumentation should be provided for an unambiguous indication of ICC. Procedures for the detection of, and the recovery from, ICC conditions must be consistent with technical guidelines, including applicable generic technical guidelines.

- Thermal-hydraulic stability performance of the core during an anticipated transient without scram event should not exceed acceptable fuel design limits.

4.4.4 Technical Evaluation

4.4.4.1 Critical Heat Flux

DCA Part 2, Tier 2, Section 4.4.1.1, "Critical Heat Flux," states that the design basis for the thermal-hydraulic design of the NPM is to have a NuScale-specific CHF correlation to ensure, with a 95-percent probability at a 95-percent confidence level, that CHF does not occur during normal operation and abnormal operating occurrences. DCA Part 2, Tier 2, Section 4.4.2.2, "Critical Heat Flux," further discusses the NSP2 and NSP4 CHF correlations, which are used to evaluate thermal margin for normal operation, AOOs, infrequent events, and accidents, with the exception of those characterized by rapid depressurization. TR-0116-21012-P-A, "NuScale Power Critical Heat Flux Correlations," Revision 1, issued December 2018 (ADAMS Accession No. ML18360A632), which the NRC staff has reviewed and approved per the enclosed SER, describes the NSP2 and NSP4 CHF correlations and their development. Additionally, DCA Part 2, Tier 2, Section 4.4.2.2, discusses the extended Hench-Levy CHF correlation, which is used for events exhibiting a rapid depressurization. The applicant stated that TR-0516-49422-A provides the extended Hench-Levy CHF correlation development details, correlation limit, and range of applicability.

DCA Part 2, Tier 2, Section 4.4.2.9, "Uncertainties in Estimates," states that, in addition to the 95-percent probability at a 95-percent confidence level limit of the CHF correlation, the applicant applied additional penalties or conservative biases to obtain the CHF SAFDL for the NSP2 and NSP4 correlations. Section 3.12 of TR-0915-17564, which the NRC staff has reviewed and approved (ADAMS Accession No. ML18338A031), describes the methodology used to calculate the penalties and biases.

DCA Part 2, Tier 2, Section 4.4.2.9, states that uncertainties or biases are incorporated into the subchannel methodology to provide conservatism and that these uncertainties establish the

design limit for the CHF correlation. The referenced methodology in TR-0915-17564-P-A, Revision 2, Section 3.12 and Section 3.4, describe the methodology used to obtain the penalties and the methodology used to combine the penalties, respectively. Additionally, DCA Part 2, Tier 2, Section 4.4.2.9.2, "Uncertainties in Physical Data Inputs," provides the penalties and their bases used to set the CHF ratio (CHFR) limits for the NSP2 and NSP4 CHF correlations, respectively. DCA Part 2, Tier 2, Section 4.4.2.9.2, states that the minimum CHFR design limit includes a heat flux engineering uncertainty factor and a rod bow penalty that are based on the subchannel analysis methodology. The staff conducted an audit as part of the review, which included the CHFR penalties (ADAMS Accession No. ML19092A361). During this audit, the staff observed that the calculated values for the heat flux engineering uncertainty factor and rod bow penalty were bounded by the values used to determine the minimum CHF ratio limits in DCA Part 2, Tier 2. Based on the information provided in the DCA and the information obtained by the NRC staff during the audit, the staff finds that the NSP2 and NSP4 CHF correlations provide suitably conservative safety limits for use in transient and accident analyses because the DCA provides an adequate basis for the minimum CHF ratio penalties and because the applicant applied conservative penalties in the calculation of the minimum CHF ratio design limits. The NRC staff reviewed and accepted the Hench-Levy CHF correlation in TR-0516-49422-A, subject to the limitations and conditions in the associated SER (ADAMS Accession No. ML20189A644).

In addition to the uncertainties discussed in DCA Part 2, Tier 2, Section 4.4.2.9, the applicant considered flux tilt in DCA Part 2, Tier 2, Section 4.4.2.10, "Flux Tilt Considerations." DCA Part 2, Tier 2, Section 4.4.2.10, states that the enthalpy rise peaking factor specified in the TS includes an additional term, T_q, to accommodate azimuthal tilt that could increase the enthalpy rise peaking factor above the design limit for core design calculations.

DCA Part 2, Tier 2, Section 4.4.2.10, states that the radial tilt was determined as part of the xenon transients, as discussed in DCA Part 2, Tier 2, Section 4.3.2.7. The NRC staff has reviewed the applicant's evaluation of xenon transients in SER Section 4.3.4.6 and found it acceptable. Accordingly, the NRC staff finds the applicant's treatment of flux tilt acceptable because it is based on an acceptable xenon transient methodology.

DCA Part 2, Tier 2, Section 4.4.2.8, "Thermal Effects of Operational Transients," states that a thermal-margin trip (e.g., the overtemperature ΔT (OTΔT) trip in typical Westinghouse Electric Corporation designs or the DNBR trip in typical Combustion Engineering designs) is not necessary to mitigate AOOs for the NPM. The NRC staff reviewed DCA Part 2, Tier 2, Table 15.0-7, "Analytical Limits and Time Delays," and found that a thermal-margin trip is not credited to mitigate DBEs. SER Chapter 15 evaluates these events.

4.4.4.2 Bypass Flow

DCA Part 2, Tier 2, Section 4.4.1.3, "Core Flow," states that the design basis for the NPM core flow is that 91.5 percent of the minimum design flow passes through the core. DCA Part 2, Tier 2, Section 4.4.3.1.1, "Core Bypass Flow," describes the core bypass flowpaths as the reflector block cooling channels, guide tubes, and instrument tube bypass flowpaths. DCA Part 2, Tier 2, Section 4.4.3.1.1.2, "Guide Tube and Instrument Tube Bypass," states that the total bypass flow assumed in the subchannel analysis is 8.5 percent and that, below 20-percent power, the bypass flow is assumed to be 9 percent. The staff audited the calculations supporting DCA Part 2, Tier 2, Chapter 4, that examined the bypass flowpath analyses (ADAMS Accession No. ML19092A361). During this audit, the staff observed that the applicant used the computational fluid dynamics software, ANSYS CFX, for the evaluations; that it performed analyses over a range of operating conditions and powers; and that the results of the analyses

showed that the coolant flow through the core was greater than the values assumed in the safety analyses, as stated in DCA Part 2, Tier 2, Section 4.4.3.1.1.2. Additionally, by letter dated June 3, 2019 (ADAMS Accession No. ML19154A605), the applicant stated that an independent calculation of bypass flow, with explicit modeling of all flowpath constituents including uncertainties, was performed to demonstrate the core flow design basis. The NRC staff conducted an audit as part of the review, which included the independent bypass flow calculation (ADAMS Accession No. ML19004A098). During this audit, the NRC staff observed that the calculated values for the bypass flow, with explicit modeling of all flowpath constituents including uncertainties, provide margin to the core flow design basis. In addition, the staff considered the impact of the riser holes discussed in Section 4.3.4.3, "Reactivity Control," of this SER, on core flow. Since the holes are located above the core, the staff concludes that the riser holes have no impact on the core bypass flow percentage. Based on the information provided by the applicant and the information obtained by the staff during the audit, the staff finds the thermal-hydraulic design of the reactor core for the NPM provides adequate margin to the design basis bypass flow because the applicant performed suitably conservative analyses that demonstrated margin to the design-basis bypass flow limit.

4.4.4.3 Evaluation Methods

DCA Part 2, Tier 2, Section 4.4.2.7, "Correlations and Physical Data," states that non-LOCA analyses are performed using the NRELAP5 code and that, once the limiting cases for each transient are identified, the determination of the thermal-margin is performed using the VIPRE-01 subchannel methodology. Section 4.3.5 of TR-0516-49416-A, "Non-Loss-of-Coolant Accident Analysis Methodology," Revision 3, issued July 2020 (ADAMS Accession No. ML20191A281), describes the process for identifying the cases for subchannel analysis and extraction of boundary condition data. TR-0915-17564-P-A, which the NRC staff has reviewed and approved (ADAMS Accession No. ML19067A256), details the application of VIPRE-01 to the NPM. The staff has no unresolved issues associated with the identification of limiting cases or extraction of boundary conditions as described in TR-0516-49416-A. The staff has reviewed and approved TR-0516-49416-A, subject to the limitations and conditions in the safety evaluation (ADAMS Accession No. ML20191A281). Additionally, DCA Part 2, Tier 2, Section 4.4.2.7, states that rapid depressurization analyses are performed using the NRELAP code and determination of thermal margin is also performed using the NRELAP5 code. The staff has reviewed TR-0516-49422-A, subject to the limitations and conditions in the staff's SER, which details the application of NRELAP to the NPM.

4.4.4.4 Technical Specifications

DCA Part 2, Tier 2, Section 4.4.4.5.1, "Reactor Coolant System Flow Determination," states that the primary contributors to pressure loss in the system are the fuel assembly and steam generator regions and that pressure losses in these regions are confirmed by testing. Additionally, DCA Part 2, Tier 2, Section 4.4.4.5.1, states that, at full power, the maximum design flow is 12.5 percent greater than the best estimate flow and that the minimum design flow is 8.3 percent less than the best estimate flow. The staff compared the maximum and minimum design flow values in DCA Part 2, Tier 2, Table 4.4-2, "Plant Reactor Design Comparison," with the RCS flow rates assumed in the transient and accident analyses in DCA Part 2, Tier 2, Table 15.0-6, "Module Initial Conditions Ranges for Design Basis Event Evaluation," and found the flow range assumed in the transient and accident analyses bounds the maximum and minimum design flow values as high and low, respectively. The staff considered the impact of the riser holes discussed in Section 4.3.4.3, "Reactivity Control," of this SER, on RCS flow. The staff confirmed during an audit (ADAMS Accession No. ML20160A224) that the riser holes have a minimal impact on steady-state RCS flow, so the transient and accident analysis range for RCS flow in Table 15.0-6 remains bounding.

Additionally, the RCS flow is surveilled during power ascension following refueling outages, in accordance with GTS 3.4.1, to confirm that the RCS loop resistance used in the thermal-hydraulic design and DCA Part 2, Tier 2, Chapter 15, transient and accident analyses remains bounding. The applicant stated, in DCA Part 2, Tier 2, Section 4.4.5.2, "Initial Power and During Operation," that the RCS flow measurement includes analytical biases applied to account for the potential for secondary-side perturbations and changes in the core axial flux offset ranges. Based on GTS 3.4.1, the NRC staff finds that operation of the NPM within the RCS flow bounds assumed in the safety analyses is ensured because the flow is confirmed following refueling outages.

The analytical limits used in the transient and accident analyses are provided in DCA Part 2, Tier 2, Table 15.0-7, and are verified in accordance with GTS 3.3.1 and GTS 5.5.10. SER Chapter 15 evaluates the transient and accident analyses. SER Chapter 16 evaluates the surveillance requirements associated with GTS 3.3.1 and GTS 5.5.10.

4.4.4.5 Loose Parts Monitoring

DCA Part 2, Tier 2, Section 4.4.6.2, "Module Protection System," states that the NPM does not include an LPMS because (1) low fluid velocities, from natural circulation combined with a design that has only small lines entering the RPV, minimize the potential for loose parts entering into, or being generated in, the RPV, (2) the NPM uses corrosion-resistant materials and has a flow-induced vibration program that further minimizes the potential for loose parts being generated in the RPV, (3) a foreign materials exclusion program minimizes the potential for loose parts entering the RPV, (4) underwater vessel inspections during outages verify that there are no loose parts in the RPV, and (5) the NuScale fuel assembly has a mesh filter at the bottom of each fuel assembly. The NRC staff has previously reviewed and approved a regulatory relaxation that eliminated the requirement of the LPMS in operating boiling-water reactors (ADAMS Accession No. ML010310355). The NRC staff approved the elimination of the LMPS requirement because (1) the operating history indicated that the LPMS did not provide the safety benefits originally envisioned in RG 1.133, Revision 1, and (2) the safety benefits of the LPMS were not commensurate with the cost of maintenance and the associated radiation exposure for plant personnel.

The staff compared the primary system components and fluid velocities of the reactor designs that were approved for elimination of the LPMS against the NuScale NPM conditions. The staff determined that the NPM has lower flow rates and a relatively simplified design as compared to the subject reactors. The CVCS is the only system that provides fluid flow directly into the RPV in the NPM design (i.e., the NPM design does not have forced circulation). Accordingly, the staff finds that the NPM is no more susceptible to issues associated with loose parts than the reactors for which the LPMS requirement has been eliminated. Based on the information in DCA Part 2, Tier 2, Section 4.4.6.2, the prior NRC staff approval for the elimination of the LPMS requirement, and the staff's comparison of the NPM to operating reactors, the staff finds that the absence of an LPMS for the NPM is acceptable.

4.4.4.6 Reactor Coolant System Flow Monitoring

DCA Part 2, Tier 2, Section 4.4.5.1, "Testing Prior to Startup," states that RCS flow is continuously measured using four sets of ultrasonic transducers, as further described in DCA Part 2, Tier 2, Section 7.1.1, "Design Bases and Additional Design Considerations." DCA 2, Tier 2, Section 4.4.5.1, also states that the ultrasonic flow is calibrated against a heat balance calculated flow. GTS 3.3.1 states that the calorimetric is performed in accordance with Surveillance Requirement 3.3.1.2. Based on the continuous surveillance and on GTS 3.3.1, the

staff finds the RCS flow monitoring acceptable because it is more restrictive than the 24-hour monitoring criteria stated in DSRS Section 4.4.

4.4.4.7 Instrumentation

DCA Part 2, Tier 2, Section 4.4.6.1, "Incore Instrumentation System," states that the ICIS uses neutron flux instruments in 12 fuel assemblies to determine a three-dimensional power distribution in the core and that temperature is continuously monitored at the inlet and outlet of the 12 fuel assemblies using thermocouples. Additionally, in SER Section 18.7.4.5, the staff established a finding that the parameters available on the safety display and indication system located in the control room provide indication of ICC and are a suitable combination of information that indicates primary coolant saturation and coolant levels in both the reactor vessel and containment vessel. Based on the description in DCA Part 2, Tier 2, Section 4.4.6.1, and the information in SER Section 18.7.4.5, the staff finds that the NuScale design has adequate instrumentation that provides, in the control room, an unambiguous indication of ICC.

4.4.4.8 Stability

DCA Part 2, Tier 2, Section 4.4.1.4, "Hydrodynamic Stability," states that the design basis for the hydrodynamic stability of the NPM is that normal operation and AOOs do not lead to hydrodynamic instability. DCA Part 2, Tier 2, Section 4.4.3.3.1, "Flow Stability Exclusion Regions," states that the NuScale flow stability protection solution uses a regional exclusion solution, as described in TR-0516-49417-P-A, "Evaluation Methodology for the Stability of the NuScale Power Module," Revision 1, issued March 2020 (ADAMS Accession No. ML20086Q664). DCA Part 2, Tier 2, Section 4.4.7, further discusses the flow stability evaluation for the NPM and states that TR-0516-49417-P-A documents the evaluation methodology and that DCA Part 2, Tier 2, Section 15.9, demonstrates that the NPM-specific design is protected from unstable flow oscillations when operation is limited to a defined pressure-temperature exclusion zone. SER Section 15.9 provides the NRC staff evaluation and acceptance of the exclusion zone and the flow stability analysis. The NRC staff's review and approval of the flow stability protection solution and flow stability evaluation methodology is documented in SER for TR-0516-49417-P-A, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," Revision 1.

4.4.5 Combined License Information Items

DCA Part 2, Tier 2, Section 4.4, contains no COL information items.

4.4.6 Conclusion

Based on the NRC staff's technical review for the thermal-hydraulic design of the NPM, as documented in SER Section 4.4.4, the NRC staff draws the following conclusions:

- The thermal-hydraulic design for the NPM satisfies GDC 10 for the following reasons:
  - The applicant evaluated CHF using an acceptable correlation (see SER Section 4.4.4.1).
  - The applicant evaluated CHF margin during normal operation and AOOs using an acceptable evaluation model (see SER Section 4.4.4.3).
  - Adequate TS are provided to ensure operation of the NPM is contained within the bounds of the safety analyses (see SER Section 4.4.4.4).

– RCS flow is continuously monitored (see SER Section 4.4.4.6).

- The thermal-hydraulic design for the NPM satisfies GDC 12 because it uses an acceptable pressure-temperature exclusion zone (see SER Section 4.4.4.8).

- The thermal-hydraulic design of the NPM satisfies 10 CFR 50.34(f)(2)(xviii) because the design has adequate instrumentation that provides, in the control room, an unambiguous indication of ICC (see SER Section 4.4.4.7).

4.5 Reactor Materials

4.5.1 Control Rod Drive Structural Materials

4.5.1.1 Introduction
This section of the DCA Part 2, Tier 2, describes the materials used in the CRDM for both the RCPB portion of the CRDM and nonpressure boundary CRDM components.

4.5.1.2 Summary of Application
DCA Part 2, Tier 1: The applicant provided the Tier 1 information associated with this section in DCA Part 2, Tier 1, Section 2.1, "NuScale Power Module," as supplemented by letters dated June 12, 2017 (ADAMS Accession No. ML17163A436), and September 6, 2017 (ADAMS Accession No. ML17249A662).

DCA Part 2, Tier 2: The applicant provided a Tier 2 design description in DCA Part 2, Tier 2, Section 4.5.1, "Control Rod Drive System Structural Materials," as supplemented by a letter dated June 12, 2017 (ADAMS Accession No. ML17163A436), and summarized here in part, as follows.

The application describes the materials specifications, fabrication and processing of stainless steel components, materials other than austenitic stainless steels, and cleanliness control.

The materials used to fabricate the CRDM pressure housing are austenitic stainless steel and martensitic stainless steel. CRDM pressure housing materials meet the requirements in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) for Class 1 components.

The CRDM nonpressure boundary components that are in contact with reactor coolant include quenched and tempered martensitic stainless steel, nickel-based alloy X-750, and cobalt-based alloys. The proposed materials have been successfully used in operating plants. The manufacturing and process controls for preventing intergranular corrosion of stainless steel components follow the guidance in RG 1.44, "Control of the Processing and Use of Stainless Steel."

Cleaning and cleanliness controls comply with the requirements of ASME NQA-1, "Quality Assurance Requirements for Nuclear Facility Applications."

ITAAC: DCA Part 2, Tier 1, Section 2.1, Table 2.1-4, "NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria," Items 2 and 3, provide the ITAAC associated with DCA Part 2, Tier 2, Section 4.5.1.

Technical Specifications: There are no TS for this area of review.

Technical Reports: There are no TRs for this area of review.

4.5.1.3 Regulatory Basis

SRP Section 4.5.1, "Control Rod Drive Structural Materials," provides the relevant NRC regulatory requirements and the associated acceptance criteria for this area of review, as summarized below, along with the review interfaces with other SRP sections:

- GDC 1, "Quality Standards and Records," and 10 CFR 50.55a, "Codes and Standards," require that SSCs important to safety be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions performed. The regulations at 10 CFR 50.55a also incorporate by reference applicable editions and addenda of the ASME Code. The application of requirements in 10 CFR 50.55a and GDC 1 to the control rod drive structural materials provides assurance that the CRDS will perform as designed.

- 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, rapidly propagating failure, and gross rupture. The application of GDC 14 assures that control rod drive materials are selected, fabricated, installed, and tested to provide assurance of an extremely low probability of significant degradation and, in the extreme, to minimize the potential for a gross RCPB failure that could substantially reduce the capability to contain reactor coolant inventory or to confine fission products.

- GDC 26 requires, in part, that one reactivity control system use control rods and that this system be capable of reliably controlling reactivity changes.

The following guidance is used to meet the above requirements:

- RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal"
- RG 1.44, "Control of the Processing and Use of Stainless Steel"
- ASME NQA-1, 2008 Edition with 2009 Addenda

4.5.1.4 Technical Evaluation

The staff reviewed and evaluated the information included in DCA Part 2, Tier 2, Section 4.5.1, to ensure that the materials specifications, fabrication, processing, and cleanliness controls are in accordance with the criteria of SRP Section 4.5.1.

<u>Materials Specifications</u>. The staff reviewed DCA Part 2, Tier 2, Section 4.5.1, to determine the suitability for service of the materials selected for CRDM structural components. DCA Part 2, Tier 2, Section 4.5.1, provides information on the types, grades, heat treatments, and properties used for the materials of the CRDM components. DCA Part 2, Tier 2, Section 3.9, "Mechanical Systems and Components," states that the pressure housing consists of the latch housing (welded to the reactor vessel nozzle), the rod travel housing, and the rod travel housing plug. The materials used for the pressure housing components identified in DCA Part 2, Tier 2, Table 5.2-4, "Reactor Coolant Pressure Boundary Component and Support Materials Including Reactor Vessel, Attachments, and Appurtenances," are austenitic stainless steel (SA-965, Type 304LN). The fabrication of the CRDM pressure housing will use Type 308, 309, and 316 austenitic stainless steel welding filler materials with a maximum carbon content of 0.03 percent. The staff reviewed the specifications and grades of the CRDM pressure housing materials and verified that the materials listed meet the requirements of ASME Code, Section III, paragraph NB-2121, which requires the use of materials listed in ASME Code, Section II,

Part D, Subpart 1, Tables 2A and 2B. The pressure boundary materials are low-carbon austenitic stainless steels with corresponding low-carbon stainless steel welding filler materials, which are more resistant to stress-corrosion cracking (SCC). Therefore, the staff finds the materials acceptable because the materials have low carbon content to resist SCC; are acceptable for use in ASME Code, Section III, Class 1 systems; and are included in ASME Code, Section II.

Compliance with the requirements of GDC 26 as they relate to the CRDM materials ensures that the material selection and fabrication support reliable rod movement for reactivity control that preserves fuel and cladding integrity. Accordingly, components of the CRDM that do not perform a pressure-retaining function must also be fabricated from materials that will assure that they function reliably to meet the requirements of GDC 26. Nonpressure-retaining CRDM component materials exposed to reactor coolant include austenitic stainless steels (Types 304 and 316), martensitic stainless steels (Type 410), nickel-based alloy X-750, and cobalt-based alloys (Haynes 25 and Stellite 6). Filler metals are Types 308/308L, 309/309L, and 316/316L, with a specified maximum carbon content of 0.03 percent.

Austenitic stainless steel materials (Types 304 and 316) are used for nonpressure boundary CRDM components in contact with reactor coolant that meet the requirements of ASME Code, Section III, paragraphs NB-2160, NC-2160, NB-3120, and NC-3120. These materials have satisfactory operating experience, are compatible with the reactor coolant, and are procured in the solution-annealed condition. In addition, DCA Part 2, Tier 2, Section 4.5.1.2, “Austenitic Stainless Steel Components,” specifies that these austenitic stainless steel materials will have a maximum carbon content of 0.03 percent if they are subjected to sensitizing temperatures after solution heat treatment. This low carbon content is consistent with the practices in RG 1.44 and reduces the occurrence of sensitization of the stainless steel that could lead to SCC. Therefore, the staff finds these materials acceptable for use in nonpressure boundary CRDM components, based on the material’s solution-annealed condition, which provides a homogeneous microstructure that minimizes SCC, and on the materials’ satisfactory operating experience.

ASME Code, Section II, Part D, Subpart 1, Table 2A and Table 2B, list the material specifications and types of materials mentioned above; therefore, these materials are acceptable for use in nonpressure boundary applications. In addition, these materials are commonly used in currently operating plants and have a successful operating history. In view of the foregoing, the staff determined that the materials and material specifications for the materials used in the nonpressure-retaining CRDM components are acceptable and meet the requirements in GDC 1, GDC 14, GDC 26, and 10 CFR 50.55a.

<u>Austenitic Stainless Steel Components</u>. DCA Part 2, Tier 2, Section 4.5.1.2, states that the processing and welding of austenitic stainless steel base materials, which are procured in the solution-annealed condition for CRDM applications, are consistent with the recommendations of RG 1.44 to prevent sensitization. The staff notes that the solution-annealed condition ensures a homogenous and nonsensitized material. In addition, austenitic stainless steels that are subjected to sensitization temperatures are procured with a maximum carbon content of 0.03 percent and are verified to be nonsensitized by testing in accordance with American Society for Testing and Materials A262, “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels,” issued September 2015. The controls specified in DCA Part 2, Tier 2, Section 5.2.3, “Reactor Coolant Pressure Boundary Materials,” are used to minimize the introduction of harmful contaminants, including chlorides, fluorides, and low-melting-point alloys on the surface of austenitic stainless steel. The guidance in RG 1.44 relates to the fabrication and processing of unstabilized austenitic stainless steels to

avoid sensitization, which can increase the susceptibility of SCC. Therefore, the staff notes that, in accordance with the guidance in RG 1.44, furnace-sensitized material is not used and that methods described in RG 1.44 are followed for cleaning and protecting austenitic stainless steels from contamination during handling, storage, testing, and fabrication and for determining the degree of sensitization during welding. The staff finds this acceptable because the applicant will follow the guidance in RG 1.44 to reduce the susceptibility of components to SCC.

Cold working can increase the susceptibility of SCC in austenitic stainless steels. DCA Part 2, Tier 2, Section 4.5.1.1, "Materials Specifications," states that cold-worked austenitic stainless steel materials are avoided and that austenitic and martensitic stainless steels with a 0.2-percent offset yield strength greater than 620 MPa (90,000 psi) are not used in CRDM components to reduce the probability of SCC. This practice is consistent with SRP Section 4.5.1 when strain-hardened stainless steels are used and, therefore, is acceptable.

DCA Part 2, Tier 2, Section 4.5.1.2, states that the recommendations of RG 1.31 are used for the filler metal material used in the CRDM components and are analyzed for delta ferrite content and limited to a ferrite number (FN) between 5FN and 20FN. The guidance in RG 1.31 pertains to the delta ferrite content in austenitic stainless steel welds to minimize the presence of microfissures, which could have an adverse effect on the integrity of components. The staff finds this acceptable because the applicant will follow the guidance in RG 1.31 to minimize the presence of microfissures in austenitic stainless steel welds.

Other Materials. Materials other than austenitic stainless steels that are used to fabricate pressure boundary and nonpressure boundary CRDM components are listed below. These materials include Type 410 martensitic stainless steel, nickel-based alloy X-750, and cobalt-based material (Stellite 6 and Haynes 25).

DCA Part 2, Tier 2, Section 4.5.1.3, "Other Materials," and Table 4.5-1, "Control Rod Drive Mechanism Materials," state that the magnetic part of the latch assembly, the water-cooled coil stacks, and lath housing assembly shield rings are fabricated from Type 410 martensitic stainless steel. Type 410 components used in the CRDMs are quenched and tempered with a minimum tempering temperature of 566 degrees C (1,050 degrees F), which is consistent with SRP Section 4.5.1, paragraph II.4, to ensure that these materials will not deteriorate from SCC in service. The staff finds this acceptable because the heat treatment is in accordance with the guidance in SRP Section 4.5.1, paragraph II.4, to provide assurance that these martensitic stainless steels will not deteriorate from SCC in service.

Nickel-based Alloy X-750 (Aerospace Material Specification (AMS) 5698, "Nickel Alloy, Corrosion and Heat-Resistant, Wire 72Ni - 15.5Cr - 0.95Cb - 2.5Ti - 0.70Al - 7.0Fe No. 1 Temper, Precipitation Hardenable," and AMS 5699, "Nickel Alloy, Corrosion and Heat-Resistant, Wire, 72Ni - 15.5Cr - 0.95Cb - 2.5Ti - 0.70Al - 7.0Fe, Spring Temper, Precipitation Hardenable") is used for the latch mechanism assembly springs and the remote disconnect lower and upper springs. The staff notes that the resistance of nickel-based Alloy X-750 to SCC depends on adequate processing and heat treatment.

Nickel-based Alloy X-750 spring material and heat treatment conform to the requirements of AMS 5698 or AMS 5699, which include solution heat treatment above 1,100 degrees C (2,000 degrees F), based on operating experience for minimizing SCC in this alloy. In addition, the CRDM coil springs are not designed to be stressed beyond their elastic limit or creep limit to maintain spring functionality and minimize the potential for SCC. Finally, there have been no operating experience reports of SCC of nickel-based Alloy X-750 CRDM springs fabricated to the requirements of AMS 5698 and AMS 5699. Therefore, the staff finds this material and the

heat treatment of this precipitation hardenable alloy acceptable because it is based on industry experience and will ensure that the material properties of the component are capable of maintaining its structural integrity and performing its intended function.

DCA Part 2, Tier 2, Section 4.5.1.3, states that Haynes 25 and Stellite 6 material are used for wear-resistant parts. These materials are commonly used in operating plants and have satisfactory operating experience; therefore, they are acceptable to the staff.

Cleaning and Cleanliness Controls. DCA Part 2, Tier 2, Section 4.5.1.4, "Cleaning and Cleanliness Control," discusses the cleaning and cleanliness controls for the CRDM during manufacture and assembly. DCA Part 2, Tier 2, Section 4.5.1.4, states that cleaning and cleanliness controls will be implemented in accordance with ASME NQA-1. SRP Section 4.5.1 recommends that cleaning and cleanliness controls for CRDMs should be implemented in accordance with ASME NQA-1, which has strict process controls for cleaning and protection against contamination of materials during all stages of component manufacture and installation. For example, tools used in abrasive work on austenitic stainless steel, such as grinding, should not contain and should not have been contaminated with ferritic carbon steel or other materials that could contribute to intergranular cracking or SCC. Because DCA Part 2, Tier 2, Section 4.5.1.4, states that controls for the handling and cleaning of austenitic stainless steel surfaces are used to control contamination as specified in ASME NQA-1, the staff finds this acceptable. Therefore, the staff finds the applicant's cleaning and cleanliness controls for CRDM components acceptable and consistent with SRP Section 4.5.1.

4.5.1.5 Combined License Information Items
There are no COL information items from DCA Part 2, Tier 2, Table 1.8-2, "Combined License Information Items," that affect this section.

4.5.1.6 Conclusions
The staff concludes that the selection of materials, fabrication processes, compatibility of materials, and cleaning and cleanliness controls are acceptable because they satisfy the NRC regulatory requirements (i.e., 10 CFR 50.55a and GDC 1, 14, and 26) and regulatory positions described above for RCPB materials, including the acceptable demonstration of appropriate materials selections and acceptable operating experience (for non-RCPB materials). Therefore, the staff concludes that the design of the CRDM materials is acceptable and meets the requirements of GDC 1, 14, and 26 and 10 CFR 50.55a.

4.5.2 Reactor Internal Core and Support Structure Materials

4.5.2.1 Introduction
This section of DCA Part 2, Tier 2, describes the reactor vessel internals (RVIs) and core support materials.

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

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 4.5.2, "Reactor Internals and Core Support Materials," describes the design, as summarized, in part, below.

ITAAC: No ITAAC items are associated with this area of the review.

Technical Specifications: There are no TS for this area of review.

Technical Reports: There are no TRs for this area of review.

DCA Part 2, Tier 2, Section 4.5.2, as supplemented by letters dated August 2, 2017, and August 28, 2017 (ADAMS Accession Nos. ML17214A895 and ML17240A423, respectively), describes the materials used to fabricate RVIs and core support structures. DCA Part 2, Tier 2, provides information about the materials specifications, controls on welding, NDE, fabrication and processing of austenitic stainless steel components, and items on materials other than austenitic stainless steel. Each topic is discussed below.

Materials Specifications

DCA Part 2, Tier 2, Table 4.5.2, lists all RVI materials and associated specifications. DCA Part 2, Tier 2, states that all portions of the RVI that perform a core support function are classified as Quality Group B and are designed and fabricated as Class CS in accordance with ASME Code, Section III, Subsection NG. These components and the associated threaded fasteners conform to the requirements of ASME Code, Section III, Subarticle NG-2120, and the applicable requirements of ASME Code, Section II, Part D, Tables 2A, 2B, and 4. The remaining portions of the RVI are designated as internal structures and conform to the requirements in ASME Code, Section III, Article NG-3000, paragraph NG-1122(c).

The sections below describe the design considerations necessary to account for degradation caused by neutron flux received by the RVIs.

Controls on Welding

DCA Part 2, Tier 2, requires all welding of RVI materials to conform to the applicable requirements of ASME Code, Section III, Articles NG-2000, NG-4000, and NG-5000. Welders and welding operators are qualified in accordance with ASME Code, Section IX, and RG 1.71, "Welder Qualification for Areas of Limited Accessibility," Revision 1, issued March 2007. No electroslag welding is permitted on RVI and core structural supports. DCA Part 2, Tier 2, cites further information that pertains to the welding of austenitic stainless steel in DCA Part 2, Tier 2, Section 5.2.3, as applicable to the welding of RVI and core support components.

Nondestructive Examination

DCA Part 2, Tier 2, requires NDE of core support structure materials to be in accordance with ASME Code, Section III, Subsection NG, and to use the NDE methods in ASME Code, Section V.

Fabrication and Processing of Austenitic Stainless Steel Components

DCA Part 2, Tier 2, describes the RVI components that contain austenitic stainless steel and notes that austenitic stainless steel parts are fabricated from materials procured in the solution-annealed state. Additionally, the applicant described cold-worked austenitic stainless steel as something "avoided to the extent practicable" during fabrication. Austenitic stainless steel used in RVI and core support components is not to exceed a yield strength of 620 MPa (90,000 psi).

DCA Part 2, Tier 2, requires implementation of the guidance in RG 1.44 to control the use of sensitized austenitic stainless steel.

DCA Part 2, Tier 2, further states that American Iron and Steel Institute Type 3XX series austenitic stainless steel subjected to sensitizing temperatures after undergoing solution heat treatment must be limited to a carbon content of no more than 0.03 weight percent. This applies to weld filler metals, as well. DCA Part 2, Tier 2, Table 4.5.2, lists weld materials that are in accordance with ASME Code, Section II, Part C. This is to be accomplished in accordance with RG 1.31.

Tools for abrasive work must not be contaminated by their previous usage on ferritic materials.

DCA Part 2, Tier 2, Section 5.2.3, describes further controls to minimize harmful contaminants. The applicant described acid pickling as "avoided on stainless steel" and "not used on sensitized austenitic stainless steel."

Other Materials

DCA Part 2, Tier 2, states that the materials exposed to primary reactor coolant are corrosion-resistant stainless steels; nickel-based alloys; and, "to a limited extent," cobalt-based alloys. The materials were selected for their proven light-water usage, as specified in ASME Code, Section III, paragraph NG-2160 and Subarticle NG-3120.

DCA Part 2, Tier 2, provides further details on the use of Alloy 718.

4.5.2.3 Regulatory Basis

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

- GDC1 and 10 CFR 50.55a require that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

SRP Section 4.5.2, "Reactor Internal and Core Support Structure Materials," lists the acceptance criteria adequate to meet the above requirements and review interfaces with other SRP sections.

4.5.2.4 Technical Evaluation

The staff divided its evaluation of the discussion on RVI and core support materials in DCA Part 2, Tier 2, Section 4.5.2, into five topics mapped to those described in SRP Section 4.5.2: (1) materials specifications, (2) controls on welding, (3) NDE, (4) fabrication and processing of austenitic stainless steel components, and (5) other materials.

4.5.2.4.1 Materials Specifications

DCA Part 2, Tier 2, specifies that core support materials will satisfy the requirements of ASME Code, Section III, Subarticle NG-2120, and the applicable requirements of ASME Code, Section II, Part D, Tables 2A, 2B, and 4. The remaining portions of the RVIs are designed to conform to ASME Code, Section III, Article NG-3000. The staff finds this to be acceptable because it complies with the ASME Code and 10 CFR 50.55a.

DCA Part 2, Tier 2, further states that the design of the RVIs considered degradation induced by peak neutron fluence and neutron irradiation, such as irradiation-assisted SCC, void swelling, stress relaxation, and irradiation embrittlement. DCA Part 2, Tier 2, references criteria from the Electric Power Research Institute's Materials Reliability Program (MRP). As addressed in MRP-227, "Safety Evaluation by the Office of Nuclear Reactor Regulation: Materials Reliability

Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," Revision 0, dated March 28, 2011 (ADAMS Accession No. ML110820773), this program forms part of the basis of staff-approved applications for the operating fleet that are used to manage the effects of the degradation mechanisms listed above. The applicant listed components screened in by these criteria. Components screened in for neutron degradation were noted as "included for augmented visual inspections" that consisted of Visual Testing (VT)-1 and VT-3 inspections, as detailed in DCA Part 2, Tier 2, Table 5.2-7, "Reactor Vessel Internals Inspection Elements." The staff confirmed that the applicant's evaluation was consistent with the Electric Power Research Institute's MRP as it pertains to the NuScale design and consequently that the applicant's evaluation and inclusion of inspections was acceptable as they pertain to the consideration of the above-listed degradation mechanisms.

The applicant stated that crevice corrosion was not a potential degradation mechanism for the NuScale design. Based on large light-water reactor operating experience, the staff accepted that the potential for crevice corrosion was low enough to not merit further consideration.

4.5.2.4.2 Controls on Welding

The staff reviewed the controls on welding in DCA Part 2, Tier 2, Section 4.5.2.2, "Control on Welding"; specifically, the citations of ASME Code sections; RG 1.71 guidance; and DCA Part 2, Tier 2, Section 5.2.3, information. The staff found the information presented acceptable because it complies with the SRP criteria for this topic.

4.5.2.4.3 Nondestructive Examination

The staff reviewed the NDE information in DCA Part 2, Tier 2, Section 4.5.2.3, "Nondestructive Examination"; specifically, the citation of ASME Code sections. The staff found the information presented acceptable because it complies with the SRP criteria for this topic.

4.5.2.4.4 Fabrication and Processing of Austenitic Stainless Steel Components

The staff reviewed DCA Part 2, Tier 2, Section 4.5.2.4, "Fabrication and Processing of Austenitic Stainless Steel Components," with emphasis on heat treatment, controls on sensitization, compatibility with reactor coolant, abrasive work, and minimization of contamination. The staff confirmed that the applicant noted appropriate controls on heat treatments. The staff confirmed that environmental conditions are controlled and that welding procedures are developed to minimize the probability of sensitization and microfissuring. This is achieved by following the guidance of RG 1.44 and RG 1.31, respectively. The staff confirmed the RVI and core support material compatibility with coolant through a review of the selection of materials for each component; a commitment to RGs and ASME Code requirements; the topics detailed in DCA Part 2, Tier 2, Section 4.5.2.4 and Section 5.2.3.4, "Fabrication and Processing of Austenitic Stainless Steels"; and the water chemistry requirements for oxygen content in DCA Part 2, Tier 2, Section 5.2, "Integrity of Reactor Coolant Boundary," Table 5.2-5, "Reactor Coolant Water Chemistry Controls." The oxygen concentration requirement of less than 0.005 ppm is below the limit noted in RG 1.44 known to inhibit SCC. The staff reviewed the fabrication and cleaning controls imposed on stainless steel components and found them acceptable because they allow no contamination with ferritic or other troublesome materials and subsequent usage on austenitic materials. DCA Part 2, Tier 2, Section 5.2.3, discusses cleaning chemicals, cleaning water chemistry, and halides in detail and references ASME NQA-1 requirements, in particular. Because the fabrication, processing, and cleaning controls conform to the recommendations and requirements of the ASME Code, RG 1.31, RG 1.44, and ASME NQA-1, the staff concludes that they are acceptable.

4.5.2.4.5 Other Materials

DCA Part 2, Tier 2, Section 4.5.2.5, "Other Materials," lists several materials as "Other Materials." Of these, DCA Part 2, Tier 2, Section 4.5.2.5, only identifies Alloy 718 for use in threaded fasteners with a reference to DCA Part 2, Tier 2, Section 3.13.1, "Threaded Fasteners (ASME Code Class 1, 2, and 3)."

The staff evaluated the identified material and found the discussion of "Other Materials" acceptable because it is consistent with other approved designs and operating reactors and regulatory requirements, specifically those in 10 CFR 50.55a.

4.5.2.5 Combined License Information Items
There are no COL information items from DCA Part 2, Tier 2, that affect this section.

4.5.2.6 Conclusion
Based on the staff's technical review of the information submitted by the applicant, the staff concludes that the NuScale design of the RVI and core support materials satisfies the relevant requirements of 10 CFR 50.55a and GDC 1 and, therefore, is acceptable because the NuScale RVI and core support structure materials satisfy ASME Code, Section III; RG 1.31; RG 1.44; and RG 1.71 and conform to the guidance in SRP Section 4.5.2.

4.6 Functional Design of Control Rod Drive System

4.6.1 Introduction

The staff reviewed DCA Part 2, Tier 2, Section 4.6, "Functional Design of Control Rod Drive System," to confirm that the CRDS can reliably control reactivity, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of a postulated accident. The NuScale design also uses CVCS to control reactivity. The staff's review in this section focused on the functional performance of the CRDS, including the consideration of single failure and common-cause failures.

4.6.2 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Tier 1, Section 2.1, and Section 2.8, "Equipment Qualification," provides the DCA Part 2, Tier 1, information associated with this section. DCA Part 2, Tier 1, Section 2.2, "Chemical and Volume Control System," provides the DCA Part 2, Tier 1, information associated with the CVCS.

DCA Part 2, Tier 2: DCA Part 2, Tier 2, Section 4.6, describes the system, as summarized, in part, below.

The NuScale design includes two reactivity control systems: (1) the CRDS and (2) the CVCS. The NuScale design relies on the CRDS to prevent and mitigate DBEs. The CVCS is designed to control reactivity changes resulting from planned, normal operation and is not required for DBE mitigation.

The CRDS safety-related functions release the CRAs during a reactor trip and maintain the pressure boundary of the RPV. DCA Part 2, Tier 2, Section 3.9.4, "Control Rod Drive System," describes the mechanical design of the CRDM. DCA Part 2, Tier 2, Section 7.0.4, "System Descriptions," provides the instrumentation and controls (I&C) for the CRDS. Finally, DCA

Part 2, Tier 2, Section 14.2, “Initial Plant Test Program,” addresses the initial startup testing of the CRDS.

DCA Part 2, Tier 2, Chapter 15, demonstrates that, for all DBEs, the CRDS is capable of maintaining the reactor within acceptable limits under the assumption that the most reactive control rod is stuck out.

DCA Part 2 Tier 2, Section 9.3.4, discusses the CVCS in more detail.

ITAAC: No ITAAC items are associated with this area of the review.

Technical Specifications: The following NuScale GTS apply to this area of review:

- GTS 3.1.5, “Shutdown Bank Insertion Limits”
- GTS 3.1.6, “Regulating Bank Insertion Limits”
- GTS 3.1.9, “Boron Dilution Control”

Technical Reports: There are no TRs for this area of review.

4.6.3 Regulatory Basis

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

- GDC 4, “Environmental and Dynamic Effects Design Bases,” as it relates to the requirement that SSCs important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions during normal plant operation, maintenance, testing, and postulated accidents

- GDC 23, “Protection System Failure Modes,” as it relates to the protection system failing into a safe state or into a state that is demonstrated to be acceptable for some other defined basis

- GDC 25, as it relates to the protection system’s capability to ensure that the specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems

- GDC 26, as it relates to the requirement that two independent reactivity control systems of different design principles be provided and be capable of reliably controlling reactivity changes under conditions of normal operation, including AOOs, to assure that specified acceptable fuel design limits are not exceeded; in addition, one of the systems must be capable of holding the reactor core subcritical under cold conditions

- GDC 27, as it relates to the requirement that the reactivity control systems be designed to have a combined capability of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained

- GDC 28, as it relates to the requirement that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant boundary nor disturb the core and its support structures to significantly impair the capability to cool the core

- GDC 29, “Protection against Anticipated Operational Occurrences,” as it relates to the requirement that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs

4.6.4 Technical Evaluation

The staff reviewed DCA Part 2, Tier 2, Section 4.6, in accordance with SRP Section 4.6, “Functional Design of Control Rod Drive System.” The staff evaluated the functional performance of the CRDS to confirm that it can provide a safe-shutdown response within acceptable limits during AOOs and prevent or mitigate the consequences of postulated accidents. The review covered the CRDS and its combined performance with other reactivity control systems to ensure conformance with the requirements of GDC 4, 23, 25, 26, 27, 28, and 29.

DCA Part 2, Tier 2, Section 4.6.1, “Description of the Control Rod Drive System,” describes the CRDS. The system consists of the CRDMs, including rod position indicators, and couples with the CRAs. During reactor operations, the CRDS supports the CRAs by latching, holding, and maneuvering the CRAs. The CRDS also includes rod position indication cabinets and cables, CRDM power cables, and cooling water supply and return piping inside containment. The CRDS safety-related functions release the CRA into the core during a reactor trip and maintain the pressure boundary of the RPV.

DCA Part 2, Tier 2, Figure 4.6-1, “Overview of Control Rod Drive Mechanism Locations in Relation to the Reactor Pressure Vessel and Containment Vessel,” depicts the CRDS and its relationship to the reactor and containment vessels. DCA Part 2, Tier 2, Figures 4.6-2 through 4.6-6, provide details on the CRDMs. DCA Part 2, Tier 2, Section 3.9.4, further describes the CRDS.

SER Section 3.9.4 evaluates the adequacy of the CRDS to perform its mechanical functions, including the testing program and consideration of design loads, stress limits, and allowable deformations. SER Section 4.2 evaluates the CRA design.

DCA Part 2, Tier 2, Section 7.0.4, discusses the information on I&C for the CRDS, and DCA Part 2, Tier 2, Section 7.0, “Instrumentation and Controls - Introduction and Overview,” describes the separation between the safety-related MPS and module control system that is not safety related. SER Chapter 7 evaluates the adequacy of these respective DCA Part 2 sections.

DCA Part 2, Tier 2, Table 3.2-1, “Classification of Structures, Systems, and Components,” states that the control rod drive shaft, latch mechanism, and CRA are safety related, are designed to be seismic Category I, and are required to meet the quality assurance requirements of 10 CFR Part 50, Appendix B. DCA Part 2, Tier 2, Table 3.11-1, “List of Environmentally Qualified Electrical/I&C and Mechanical Equipment Located in Harsh Environments,” shows that the control rod drive coil and CDRM control cabinet are part of the equipment qualification program and must function to mitigate design-basis accidents.

DCA Part 2, Tier 2, Section 4.6.2, “Evaluations of the Control Rod Drive System,” states that jet impingement loads generated from high-energy lines inside the containment vessel are analyzed in combination with leak-before-break analyses. The applicant stated that, based on the low jet pressure load and heavy walled construction of the CRDMs, jet impingement does not adversely affect CRDM scram functionality. In addition, jet impingement loads from the opening of the reactor safety valves and reactor vent valves are designed with a fluid jet diffuser

at the outlet of the valves to dissipate the energy of the fluid jet and protect safety-related SSCs in containment near the RPV head. SER Section 3.6 evaluates high-energy line breaks inside the containment vessel.

The CRDS should remain functional under adverse environmental conditions and after postulated accidents. The CRDMs are mounted on the RPV head and are ASME Code Class 1 pressure boundaries. DCA Part 2, Tier 2, Section 4.6.1, states that the CRDS components internal to the RCPB are designed to function in borated primary coolant with up to 2,000-ppm boron at primary coolant pressures and temperatures ranging from ambient conditions to a design temperature of 343 degrees C (650 degrees F) and an RPV design pressure of 14.5 MPa (absolute) (2,100 psi, absolute). In addition, DCA Part 2, Tier 2, Table 3.11-1, identifies the portions of the CRDS outside the RCPB, such as the rod position indication coils and cooling water piping, as environmentally qualified for harsh environments.

DCA Part 2, Tier 2, Section 4.6.1, states that the electric coil operating conditions of the CRDS require active cooling by water through a CRDS cooling water distribution header to cooling tubes in the drive coils of each CRDM, as is shown in DCA Part 2, Tier 2, Figure 4.6-3, "Control Rod Drive Mechanism Drive Coil and Cooling Detail." DCA Part 2, Tier 2, Section 4.6.1, adds that the reactor component cooling water system (RCCWS) in DCA Part 2, Tier 2, Section 9.2.2, "Reactor Component Cooling Water System," provides the cooling requirements for the CRDMs. In its response to Request for Additional Information 9242, Question 04.06-1 (ADAMS Accession No. ML18101B407), the applicant stated that the CRDM coils are designed using a Class N insulation system, which is rated to 200 degrees C (392 degrees F), and that, to account for some margin, DCA Part 2 specifies that the maximum temperature design criterion for the CRDM is 180 degrees C (356 degrees F). Therefore, the staff finds that the RCCWS is capable of maintaining the CRDMs below design requirements during normal operation. SER Section 9.2.2 presents the staff's detailed review of the RCCWS.

In accordance with the guidance in SRP Section 4.6, the staff confirmed that the CRDM cooling system meets the design requirements by performing a design review and auditing design specifications, as described in the staff's audit report (ADAMS Accession No. ML17331A357). Due to the unique orientation of the CRDMs above the pressurizer in a borated steam environment, the staff assessed the potential for chemical buildup to impact CRDM function and documented it in SER Section 3.9.4.4.6.

A single failure in the CRDS should not prevent the system from performing its safety-related function. The applicant evaluated failures of the CRDM in a failure modes and effects analysis (FMEA). However, the applicant did not provide the FMEA as part of its application. Therefore, the staff audited the FMEA, as documented in the audit report (ADAMS Accession No. ML17331A357). The FMEA demonstrated that no single failure in the CRDS could prevent a reactor trip and that the ability to rod drop on command was retained. The staff concluded that the applicant completed an FMEA and determined that the CRDS is capable of performing its safety-related function following the loss of any active component.

The staff notes that the failure of a single CRDM would not prevent other CRDMs from inserting CRAs into the core because the CRDMs operate independently. Therefore, the staff concludes that the safety-related reactor trip function is available in the event of a single failure in the CRDS. In addition, the staff notes that sufficient SDM exists if a CRA fails to insert, as described in SER Section 4.3.4.3. For these reasons, the staff concludes that the CRDS meets the requirements of GDC 23 with respect to the CRDS failing safe, GDC 25 with respect to stuck rod considerations during a single malfunction of the CRDS, and GDC 26 with respect to

one stuck CRA being the appropriate margin to consider for stuck rods. SER Chapter 7 evaluates the CRDS requirements of GDC 23 and GDC 25 with respect to I&C aspects of the protection system. SER Chapter 15 evaluates additional considerations with regard to the integrated NPM design margin to ensure that SAFDLs are not exceeded during AOOs (e.g., CRD insertion time, heat removal capabilities, and margin to thermal limits).

The staff's evaluation associated with GDC 26 related to independent reactivity control systems is in Section 4.3.4.3 of this SER.

The analyses in DCA Part 2, Tier 2, Chapter 15, show that the CRDS is capable of bringing the core to a shutdown condition and maintaining fuel integrity, consistent with the design information in DCA Part 2, Tier 2, Section 4.3. The applicant requested an exemption to GDC 27 and proposed PDC 27. SECY-18-0099, "NuScale Power Exemption Request from 10 CFR Part 50, Appendix A, General Design Criterion 27, 'Combined Reactivity Control Systems Capability,'" dated October 9, 2018, describes the staff's criteria for evaluating the GDC 27 exemption request. SER Section 15.0.6 evaluates this exemption with the evaluation of thermal-margin and probability of occurrence of a potential post-trip return to power.

GDC 28 requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to prevent the adverse effects of postulated reactivity accidents. A postulated failure of the CRDS that causes a rod ejection has the potential to result in a relatively high rate of positive reactivity insertion, which could challenge fuel design limits, the RCPB, and the capability to cool the core. DCA Part 2, Tier 2, Section 4.6.2, states that, to prevent a mechanical failure of the CRDM pressure housings, the housings are designed to be an integral part of the RPV, and the welds are inspected to ASME Class 1 requirements.

DCA Part 2, Tier 2, Section 3.1.3.9, "Criterion 28-Reactivity Limits," states that the NuScale design places limits on the worth of CRAs, the maximum CRA withdrawal rate, and CRA insertion (i.e., PDILs). DCA Part 2, Tier 2, Table 4.3-3, "Reactivity Requirements for Control Rods," provides the reactivity requirements for control rods; DCA Part 2, Tier 2, Section 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition," defines the maximum allowed withdrawal rate of a CRA to be 38 centimeters (15 inches) per minute; and TS LCO 3.1.5 and TS LCO 3.1.6 prescribe CRA insertion limits for the regulating and shutdown groups, respectively. The maximum worth of the CRAs and the limits on CRA insertion preclude rupture of the RCPB caused by a rod withdrawal or rod ejection accident. SER Section 15.4.8 evaluates a rod ejection accident.

DCA Part 2, Tier 2, Section 15.4.6, "Inadvertent Decrease in Boron Concentration in the Reactor Coolant System," evaluates the potential for a boron dilution event from a demineralized water supply through the CVCS. The CVCS dilution event is limited based on the closure of the safety-related demineralized water supply isolation valves. This design feature ensures that no damage occurs to the RCPB or disturbs the RVIs to the extent that it affects the ability to cool the core. TS 3.1.9 provides an LCO for the demineralized water isolation valves. SER Section 15.4.6 evaluates the dilution event.

The staff review associated with reactivity control during long-term cooling due to the potential for boron redistribution is evaluated in SER Section 15.0.6.

DCA Part 2, Tier 2, Section 4.6.3, "Testing and Verification of the Control Rod Drive System," refers to DCA Part 2, Tier 2, Section 3.9.4.4, "Control Rod Drive System Operability Assurance

Program," and Section 4.2.4, for the testing and verification of the CRDS. DCA Part 2, Tier 2, Section 3.9.4.4, states that a prototype testing program that integrates the CRDM, the control rod drive shaft, the CRA, and the fuel assembly was created to demonstrate the acceptable mechanical functioning of a prototype CRDS. The testing of the prototype includes performance testing, stability testing, endurance testing, and production testing. In addition, DCA Part 2, Tier 2, Sections 1.5.1, "NuScale Testing Programs," describe testing programs associated with the design features of the CRDS.

In addition, DCA Part 2, Tier 2, Section 4.6.3, refers to the preoperational and initial startup test program for the CRDS in DCA Part 2, Tier 2, Section 14.2. The following tests from DCA Part 2, Tier 2, apply to the CRDS:

- Table 14.2-44, "Control Rod Drive System Flow-Induced Vibration Test # 44"
- Table 14.2-80, "Control Rod Drive System—Manual Operation, Rod Speed, and Rod Position Indication Test # 80"
- Table 14.2-81, "Control Rod Assembly Full-Height Drop Time Test # 81"
- Table 14.2-81a, "Control Rod Assembly Ambient Temperature Full-Height Drop Time Test # 81A"
- Table 14.2-98, "Control Rod Assembly Misalignment # 98"
- Table 14.2-104, "Reactor Trip from 100 Percent Power Test # 104"

SER Section 14.2 provides the staff's review of the applicant's initial test program.

The staff concludes that the CRDS meets the requirements of GDC 29 because the tests addressed above, along with the design of the CRDS previously discussed, ensure an extremely high probability that the CRDS will accomplish its safety function in the event of an AOO.

4.6.5 Combined License Information Items

No COL information items are associated with DCA Part 2, Tier 2, Section 4.6.

4.6.6 Conclusion

Based on the NRC staff's technical review for the functional design of the CRDS, as documented in SER Section 4.6.4, the NRC staff makes the following conclusions:

- The functional design of the CRDS satisfies GDC 4 because the CRDMs are designed to accommodate the effects of and are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
- The functional design of the CRDS satisfies GDC 23 because the CRDMs fail into a safe state.
- The functional design of the CRDS satisfies GDC 25 because the evaluation of a single malfunction of the reactivity control system (1) uses conservative values for the rate of reactivity insertion and (2) accommodates a stuck control rod.

- The functional design of the CRDS satisfies GDC 26 because the AOOs are evaluated with appropriate margin for stuck control rods.
- The functional design of the CRDS satisfies PDC 27 because, with all control rods in the CRDS capable of holding the reactor core subcritical during cold conditions and with one rod out, the capability to cool the core is maintained, and with all control rods inserted, the reactor remains subcritical under cold conditions.
- The functional design of the CRDS satisfies GDC 28 because the CRDS is designed with appropriate limits on the potential amount and rate of reactivity increase.
- The functional design of the CRDS satisfies GDC 29 because the tests and design of the CRDS ensure, with an extremely high probability, that the CRDS will accomplish its safety function in the event of an AOO.