4 REACTOR

This chapter of the safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review of Chapter 4, "Reactor," of the NuScale Power, LLC (NuScale or the applicant), Design Certification Application (DCA), Part 2, "Final Safety Analysis Report," Revision 1, issued March 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18086A172).

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 reactor coolant pressure boundary (RCPB) or impair the capability to cool the core.

4.2.1.1 Summary of Application

DCA Part 2, Tier 1: DCA Part 2, Revision 0, issued December 2016, did not contain DCA Part 2, Tier 1, entries for this area of review. In its response to **Request for Additional Information (RAI) 9406** (ADAMS Accession No. ML18115A484), the applicant provided a markup of DCA Tier 1, Chapter 2, "Unit Specific Systems, Structures, and Components Design Descriptions and Inspections, Tests, Analyses, and Acceptance Criteria," that describes the fuel system. The staff reviewed the information in the response, which included a physical description of the fuel system design and design goals, and concludes that DCA Tier 1 will capture the necessary level of fuel system design information to conduct its safety review. 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. The NRC staff is tracking this commitment as **Confirmatory Item 4.2-01** until the applicant incorporates the markups into the design control document.

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

4.2.1.1.1 Fuel Assembly Description

NuScale's fuel assembly contains 264 fuel rods and burnable absorber rods, 24 guide tubes, and one 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.

4.2.1.1.2 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 fuel cladding is M5™ tubing with a nominal wall thickness of 0.6096 millimeters (0.024 inch). The applicant stated that the M5™ cladding material significantly improves corrosion resistance.

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.

The plenum spring at the top of the fuel pellet column keeps the column in its proper position during handling and shipping. 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.

4.2.1.1.3 Burnable Absorber Rod Description

To reduce the beginning-of-life moderator 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.

4.2.1.1.4 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.

4.2.1.1.5 Design Evaluation

The applicant summarized the design evaluations of the fuel rod, fuel assembly, and in-core control components. The fuel rod design considers all events that are expected during normal operations, AOOs, and postulated accidents. The design of the in-core control components considers events during normal operations, AOOs, and postulated accidents. DCA Part 2, Tier 2, Chapter 15, “Transient and Accident Analyses,” evaluates some postulated accident events such as reactivity-initiated accident events, loss-of-cooling accident (LOCA) events, and anticipated transients without scram.

The applicant summarized the design evaluations for each component and event and concluded that the appropriate specified acceptable fuel design limits (SAFDLs) are met. The methodologies used in the analyses were determined to apply to the applicant’s fuel design in the referenced Topical Report TR-0116-20825-P-A, “Applicability of AREVA Fuel Methodology for the NuScale Design,” Revision 1, dated November 24, 2017 (ADAMS Accession No. ML18040B306). Technical Report TR-0816-51127-P, “Fuel and Control Rod Assembly Designs,” Revision 1, dated January 2017 (ADAMS Accession No. ML17007A002) 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: TR-0816-51127-P, Revision 1.

4.2.2 Regulatory Basis

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

- Title 10 of the *Code of Federal Regulations* (10 CFR), Section 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 mode and establishing acceptance criteria for light-water nuclear power reactor ECCSs.

- 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, “Domestic Licensing of Production and Utilization Facilities,” as it relates to assuring that structures, systems, and components 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,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” as it relates to assuring 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 control rod insertability 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.

4.2.3 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 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 review 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-P 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. The SER for Chapter 15, Section 15.0.6, evaluates this exemption and the proposed PDC and identifies concerns with the evaluation of thermal margin and probability of occurrence. The NRC staff has issued **RAI 8771, Question 15-1,** and **RAI 9505, Question 15-18,** to address these items. The NRC staff is tracking the thermal-margin evaluation and probability of occurrence associated with the exemption to GDC 27 as **Open Item 4.2-02** and **Open Item 4.2-03**, respectively.

The NRC staff notes that NuScale has requested an exemption from GDC 35 and proposed PDC 35 in lieu of GDC 35. The PDC proposed by NuScale is functionally identical to the GDC, with the exception of the discussion related to electric power. The modification to the electric power discussion in PDC 35 is tied to the exemption request to GDC 17 and proposed PDC 17. SER Chapter 8 discusses NuScale’s reliance on electric power and the related exemption to GDC 17, “Electric Power Systems,” as well as the staff’s evaluation, dated December 13, 2017, of TR-0815-16497, “Safety Classification of Passive Nuclear Power Plant Electrical Systems” (ADAMS Accession No. ML17340A524). The applicability of TR-0815-16497 to the NuScale DCA is discussed in Section 8.3.1.4 of this SER and is being evaluated as **Open Item 8.3-1**. The exemption from GDC 35 is being evaluated as part of **Open Item 8.3-1**.

The SER for Chapter 6, Section 6.3, and Chapter 15, evaluates the ECCS against the proposed PDC 35. This evaluation raised concerns and issued RAI 8930, Question 15-7, regarding boron dilution, RAI 9506, Question 15-8, regarding margin for stuck rods, and RAI 9496, Question 15-19, regarding return to power, to address these items. The staff is tracking the resolution of boron dilution and margin for a stuck rod as **Open Item 4.2-04**.

4.2.3.1 Design Bases

4.2.3.1.1 Fuel System Damage

DCA Part 2, Tier 2, Section 4.2, summarizes the analyses that cover fuel system damage, fuel rod damage, and core coolability. The 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. 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
- reactivity control assembly insertability

Stress/Strain Limits. Section 4.1.1 of TR-08016-51127-P, Revision 1, provides a sample 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, February 1999 (ADAMS Accession No. ML003681168), 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, June 2003 (ADAMS Accession No. ML15162B043). These sample analyses demonstrated that the calculated stresses and loadings are all within the design criteria for all the required conditions.

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

Fuel Assembly Fatigue. Section 4.1.2 of TR-08016-51127-P, Revision 1, provides a sample cladding fatigue analysis performed in accordance with BAW-10227P-A, Revision 1 (ADAMS Accession No. ML15162B043). This sample 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 can meet the regulatory requirements on cladding fatigue in accordance with the guidance in SRP Section 4.2.

Fuel Fretting. Section 4.1.3 of TR-08016-51127-P, Revision 1, provides a sample fretting analysis performed in accordance with EMF-92-116(P)A, Revision 0. Using fretting tests, this sample analysis demonstrates that 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 can meet 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 of TR-08016-51127-P, Revision 1, provides a sample oxidation, hydriding, and crud buildup analysis performed in accordance with BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code", January 2004, (ADAMS Accession No. ML042930236). Using the COPERNIC code and a representative power history envelope, this sample analysis demonstrates that the calculated oxide thickness is below the limit of 100 micrometers. The corrosion limit limits the hydrogen pickup, and crud buildup is built into the corrosion thickness.

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

Dimensional Changes. Section 4.1.5 of TR-08016-51127-P, Revision 1, provides a sample fuel rod bow analysis performed in accordance with XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," Supplements 1 through 4, February 1983 (ADAMS Accession No. ML081710709). This sample analysis demonstrates that 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.

The staff concludes that the applicant demonstrated that the NuScale fuel design can meet the regulatory requirements on fuel rod bow in accordance with the guidance in SRP Section 4.2.

Rod Internal Pressure. Section 4.1.8 of TR-08016-51127-P, Revision 1, provides a sample fuel rod internal pressure analysis performed in accordance with BAW-10231P-A, Revision 1. Using a bounding analysis with the COPERNIC code, this sample analysis demonstrates that significant margin exists between the rod internal pressure limit of 1,850 pounds per square inch (coolant system pressure) and the calculated maximum internal pressure.

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

Fuel Assembly Liftoff. Section 4.1.9 of TR-08016-51127-P, Revision 1, provides a sample fuel assembly liftoff analysis performed in accordance with EMF-92-116(P)A, Revision 0. Using bounding flow rates for the limiting AOO, this sample 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 the fuel assembly liftoff in accordance with the guidance in SRP Section 4.2.

Reactor Control Assembly Insertability. Section 4.3.5 of TR-0816-51127-P, Revision 1, addresses fuel assembly structural damage, which could prevent control assembly insertability. SER Section 4.2.3.5 addresses the evaluation of control assembly insertability under seismic and LOCA loads.

4.2.3.1.2 Fuel Rod Failure

Hydriding. Section 4.2.1 of TR-08016-51127-P, Revision 1, provides a sample internal hydriding analysis performed in accordance with EMF-92-116(P)A, Revision 0. Using fabrication limits for fuel pellet moisture, this sample analysis demonstrates that 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 of TR-08016-51127-P, Revision 1, provides a sample cladding collapse analysis performed in accordance with BAW-10084P-A, Revision 3, "Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse," July 1995 (ADAMS Accession No. ML14191B170) and that used 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 that are at each time step as proposed by the revision to the creep collapse methodology. Using the CROV code initiated with COPERNIC, this sample 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. Violation of the DNBR limits is not allowed for normal operation and AOOs. SER Section 4.4 reviews the DNBR margin analysis. The various design-basis accident evaluations, as detailed in SER Chapter 15, review the cladding temperature under postulated accident conditions.

Overheating of the Fuel Pellets. Section 4.2.4 of TR-08016-51127-P, Revision 1, provides a sample analysis on the overheating of fuel pellets performed in accordance with BAW-10231P-A, Revision 1. Using a bounding analysis with the COPERNIC code, this sample analysis demonstrates that significant margin exists between the NuScale power limits and the 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 fuel melting.

Pellet/Cladding Interaction. No generic criterion for fuel failure resulting from pellet cladding interaction or pellet cladding mechanical interaction exists. Cladding strain and fuel melting limits were assumed to act as a surrogate.

Section 4.2.6 of TR-08016-51127-P, Revision 1, provides a sample transient clad strain analysis performed in accordance with BAW-10231P-A, Revision 1. Using a bounding analysis with the COPERNIC code, this sample analysis demonstrates that significant margin exists between the NuScale power limits and the power level required to obtain a 1-percent cladding strain limit.

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

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

Mechanical Fracturing. SER Section 4.2.3.5 presents the staff's review of fuel rod mechanical fracturing.

4.2.3.1.3 Fuel Coolability

Some of these damage mechanisms, 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 4.2.3.5 presents 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.

4.2.3.2 Description and Design Drawings

The staff reviewed the fuel system description and design drawings given in DCA Part 2, Tier 2, Section 4.2. TR-0816-51127-P, Revision 1, 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 this acceptable.

4.2.3.3 Design Evaluation

4.2.3.3.1 Operating Experience

Section 3 of TR-0816-51127-P, Revision 1, 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 that technical report to justify the models used to analyze the NuFuel-HTP2 fuel assembly for use in the NuScale plant design. The staff reviewed the applicability of the models to the NuScale plant design as part of its SER for TR-0116-20825-NP-A, Revision 1, February 2018 (ADAMS Accession No. ML18040B306).

4.2.3.3.2 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, discusses AREVA's domestic and international operating experience in support of the NuScale fuel design. The staff compared the 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 are not significantly different in terms of the parameters that are important to fuel behavior; therefore, the staff finds that the AREVA operating experience applies to NuScale fuel assemblies.

DCA Part 2, Tier 2, Section 4.2.4.2, 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 of TR-0816-51127, Revision 1. 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 TR ANP-10337NP-A, Revision 0, April 2018 (ADAMS Accession No. ML18144A821).

4.2.3.4 Testing, Inspection, and Surveillance Plans

DCA Part 2, Tier 2, Section 4.2.4, 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.

The applicant manufactures and inspects its fuel assemblies and CRAs 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, as presented in DCA Part 2, Tier 2, Section 4.2.4.3. The component testing under this program includes nondestructive examinations (NDEs) and destructive examinations to support qualifications.

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 applicant's 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.

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

TR-0816-51127-P, Revision 1, presents the fuel assembly structural response to externally applied forces. The analysis is based on the methodology in ANP-10337NP-A, Revision 0. The applicant is currently modifying its analysis on the fuel assembly's structural response to externally applied forces, as detailed in its letter dated February 12, 2019 (ADAMS Accession No. ML19045A334). The staff will review the analysis when it becomes available. Therefore, the staff is tracking the applicant's analysis as **Open Item 4.2-01**.

4.2.3.6 10 CFR 50.46 Exemption Request for M5™

The NuScale fuel design consists of low-enriched uranium oxide fuel within an M5™ zirconium-based alloy cladding. NuScale has requested an exemption from 10 CFR 50.46 and Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 to permit the use M5™ alloy fuel rod cladding in its design. In BAW-10227P-A, Revision 1 (ADAMS Accession No. ML15162B043), the NRC 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 (ADAMS Accession No. ML15162B043) 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, (3) steam oxidation kinetics and applicability of Baker-Just weight gain correlation. The staff-approved BAW-10240P-A, "Incorporation of M5 Properties in Framatome ANP Approved Methods," August 17, 2004, (ADAMS Accession No. ML042800312) 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 post-quench 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 post-quench ductility tests on unirradiated and irradiated M5™ cladding segments demonstrate that the 10 CFR 50.46(b) acceptance criteria (i.e., 1,204 Celsius (2,200 degrees Fahrenheit (F)) and 17-percent equivalent cladding reacted) remain conservative up to current burnup limits. Information in the previously approved M5™ TRs and recent ANL LOCA research demonstrate that the acceptance criteria in 10 CFR 50.46 remain valid for the M5™ alloy and meet the underlying purpose of the rule to maintain a degree of post-quench ductility in the fuel cladding material.

In addition, using LOCA models and analysis methods, the NuScale analysis in DCA Part 2, Tier 2, Section 15.6.5, 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 TRs 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, granting the exemption request will ensure that the NuScale design achieves the underlying purpose of the rule. Based on results of metal-water reaction testing and mechanical testing, which ensure the applicability of 10 CFR 50.46 acceptance criteria and 10 CFR Part 50, Appendix K, methods, the staff finds granting an exemption from the requirements of

10 CFR 50.46 and 10 CFR Part 50, Appendix K, for the use of fuel rods with M5™ zirconium-based alloy for NuScale to be acceptable.

The staff has reviewed the applicant's request for an exemption to permit the use of AREVA's advanced zirconium-based M5™ alloy for fuel design. Based on its evaluation, as presented 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 underlying purposes of the rules. 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 NuScale's use of the M5™ alloy fuel rod cladding in its fuel design.

4.2.4 Combined License Information Items

DCA Part 2, Tier 2, Section 4.2 does not list any combined license (COL) information items.

4.2.5 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, "Reactor design"; and 10 CFR 50.34. The staff notes that several of the accident scenarios are evaluated in the appropriate Chapter 15 section within this SER and therefore the conclusions regarding regulatory compliance in terms of fuel under those specific postulated accidents will be presented in the respective staff safety evaluations at the completion of those reviews. The staff based its conclusion on the following:

- The applicant has provided sufficient evidence that these design objectives will be 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, "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces," 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 has provided for the testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The applicant has committed to perform online fuel failure monitoring and postirradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

The staff concludes that the applicant has 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.

Because of the open items identified in SER Section 4.2.3, the NRC staff cannot make a conclusion that the nuclear design of the NPM meets GDC 27 or GDC 35.

As detailed in the staff's technical evaluation, the applicant is currently revising the fuel assembly structural response to external forces analysis to address **RAI 9225** and other related analysis issues, and the staff is tracking this revision as **Open Item 4.2-01**. Therefore, the staff is unable to make a safety finding with regard to GDC 2, "Design bases for protection against natural phenomena," at this time.

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, 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, 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, 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 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 reactor protection system (RPS) 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. NuScale 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 215.56 degrees Celsius (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. NuScale 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, no decay heat or voiding is present, and equilibriums of samarium taken into account.

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

DCA Part 2 Tier 2, Section 4.3.2, describes the nuclear core design and provides 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 electric 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, 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 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 0, issued December 2016 (ADAMS Accession No. ML17005A116), 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 “Reactor design,” 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 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, “Combined reactivity control systems,” 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, steam line 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, 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, further describes the design basis. DCA Part 2, Tier 2, Section 4.3.2.2.6, 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. 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, "Subchannel Analysis Methodology," Revision 2, issued October 2018 (ADAMS Accession No. ML18305B218), 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 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, states that the Doppler coefficient and the MTC are the two primary reactivity feedback mechanisms that compensate for a rapid reactivity increase, provide inherent reactivity control, and satisfy GDC 11. DCA Part 2, Tier 2, Figure 4.3-13 and Figure 4.3-14, 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, and boron worth in DCA Part 2, Tier 2, Figure 4.3-21. 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 (ADAMS Accession No. ML18234A295) for TR-0616-48793, "Nuclear Analysis Codes and Methods Qualification," Revision 0, issued August 2016 (ADAMS Accession No. ML16243A517), 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, Tier2, 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, states that the NuScale design uses two independent means for reactivity control: (1) CRAs and 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 215.56 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. 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, 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, that the core is free of xenon, that no decay heat or voiding is present, and equilibrium concentration of samarium are taken into account. The NRC 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 NRC 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 NRC 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, further states that the boron addition system is managed to maintain sufficient quantity of 4,000 parts per million (ppm) boron to ensure the ability to support a 12-NPM shutdown. 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 design-basis events (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 for Chapter 15, Section 15.0.6, evaluates this exemption and the proposed PDC and identifies concerns with the evaluation of thermal margin and probability of occurrence. The NRC staff has issued **RAI 8771, Question 15-1,** and **RAI 9505, Question 15-18,** to address these items. The NRC staff is tracking the thermal-margin evaluation and probability of occurrence associated with the exemption to GDC 27 as **Open Item 4.3-02** and **Open Item 4.3-03**, respectively. Additionally, the NRC staff has identified that long-term reactivity control following an AOO or postulated accident depends upon the distribution of soluble boron throughout the RCS. Accordingly, the NRC staff has issued **RAI 8930, Question 15-27,** asking the applicant to describe and justify its methodology for evaluating the boron distribution during long-term cooling following ECCS actuation. The NRC staff is tracking this as **Open Item 4.3-04**.

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, 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, 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, shows the PDILs, and DCA Part 2, Tier 2, Figure 4.3-18, 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 has 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, 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 of this SER, 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, states that the transient analyses assume a maximum allowed CRA withdrawal rate of 15 inches per minute, and DCA Part 2, Tier 2, Table 14.2-80, states that Test No. 80 will verify that the rod insertion and withdrawal speeds are within design limits. The NRC staff did not identify design information on the limit of the withdrawal speed. Accordingly, the NRC staff issued **RAI-9449, Question 4.6-2**, asking the applicant to update DCA Part 2, Tier 2, Section 4.3, to specify a design limit on the CRA withdrawal rate. By letter dated April 9, 2018 (ADAMS Accession No. ML18099A161), the applicant updated DCA Part 2, Tier 2, Section 4.3.1.4, to clarify the design maximum rod withdrawal rate. The NRC staff finds the proposed update acceptable because it provided the requested maximum design-basis CRA withdrawal rate. The NRC staff has confirmed that the applicant incorporated the proposed update into Revision 2 of DCA Part 2, Tier 2, Section 4.3.1.4. Additionally, DCA Part 2, Tier 2, Table 15.0-1, 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. However, as discussed in SER Section 4.3.4.4, the NRC staff is still evaluating the exemption to GDC 27 and is tracking this as **Open Item 4.3-05**.

4.3.4.5 Criticality during Refueling

DCA Part 2, Tier 2, Section 4.3.2.6, 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 (pcm) margin from criticality. Additionally, Appendix B to TR-1116-52011, "Technical Specifications Regulatory Conformance and Development," Revision 1, October 2018 (ML18305A964) 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, 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, 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 and Table 4.3-11, provide the results of the applicant's analyses, which show that the reactor was stable over this configuration. Additionally, the NRC 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 NRC 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-0616-48793 and TR-0716-50350, "Rod Ejection Accident Methodology," Revision 0, issued December 2016 (ADAMS Accession No. ML16365A242), describe the applicant's use of these methods in detail. The NRC staff has reviewed and approved TR-0616-48793 for the design and analysis of the NuScale reactor core (ADAMS Accession No. ML18234A295). The NRC staff is currently reviewing TR-0716-50350 and is tracking this as **Open Item 4.3-06**.

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 NRC 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, discusses the vessel fluence analysis with the results in DCA Part 2, Tier 2, Table 4.3-12. TR-0116-20781, "Fluence Calculation Methodology and Results," Revision 0, December 2016 (ADAMS Accession No. ML17005A116) 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, 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, obtains samples as early as 3.5 electric 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
- GTS 3.1.3
- GTS 3.1.4
- GTS 3.1.5
- GTS 3.1.6
- GTS 3.1.8, “Physics Tests Exceptions”
- GTS 3.1.9
- GTS 3.2.1
- GTS 3.2.2
- GTS 3.4.1, “RCS Pressure and Temperature, and Flow Resistance Critical Heat Flux (CHF) Limits”
- GTS 3.5.3

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. However, the LCOs that reference the COLR were not supported by an established methodology. Accordingly, the NRC staff issued **RAI 8772, Question 4.03-1**, and followup **RAI 9445, Question 16-43**, asking the applicant to update DCA Part 2 and the GTS to provide the methodology and technical basis for the core operating limits in the COLR. By letters dated July 13, 2017 (ADAMS Accession No. ML17194B384), and June 12, 2018 (ADAMS Accession No. ML18163A417), the applicant provided responses to **RAI 8772, Question 4.03-1**, and **RAI 9445, Question 16-43**, that updated the GTS to describe the specific methodologies used to establish each core operating limit in the COLR. The NRC staff found the responses acceptable because each LCO that references the COLR has a specified methodology that is used to establish its limit. The NRC staff has confirmed that the applicant incorporated the proposed update into GTS 5.6.3, Revision 2.

Under 10 CFR 50.36(c)(2)(ii)(B), the NRC requires the licensee to establish a TS 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.” DCA Part 2, Tier 2, Section 4.3.2.2.1, states that the heat flux hot channel factor (F_Q) is used to ensure that the SAFDL for fuel centerline melting is not exceeded. NuScale GTS do not include an LCO for F_Q. Additionally, 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 Section 4.3.4.1 of this SER, and (2) the power distribution used in the LOCA evaluation model, TR-0516-49422, “Loss-of-Coolant Accident Evaluation Model,” Rev. 0, dated December 2016 (ADAMS Accession No. ML17004A138.), does not specify F_Q as an input parameter. The NRC staff has confirmed that the applicant incorporated the proposed update into Revision 2 to DCA Part 2, Tier 2, Section 4.3.2.2.1.

4.3.4.10 Testing and Verification

DCA Part 2, Tier 2, Section 4.3.2.2.7, 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 Monitoring

DCA Part 2, Tier 2, Section 4.3.2.2.9, 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).

Because of the open items identified in SER Section 4.3.4, the NRC staff cannot make a conclusion that the nuclear design of the NPM meets GDC 27 or GDC 28. Additionally, because of the open item associated with stability (discussed in Section 4.4.4.8 of this SER) NRC staff cannot make a conclusion that the nuclear design of the NPM meets GDC 12.

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 that CHF does not occur with a 95-percent probability at a 95-percent confidence level 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, 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, 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, describes the thermal-hydraulic evaluation and includes information on analytical models and inputs.

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

DCA Part 2, Tier 2, Section 4.4.6, 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, 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, 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
- GTS 5.5.10, “Setpoint Program (SP)”
- GTS 5.6.3

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 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 (BWRs).

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, states that the design basis for the thermal-hydraulic design of the NPM is to have a NuScale-specific CHF correlation to ensure that CHF does not occur with a 95-percent probability at a 95-percent confidence level during normal operation and abnormal operating occurrences. DCA Part 2, Tier 2, Section 4.4.2.2, further discusses the NSP2 and NSP4 CHF correlations. TR-0116-21012, "NuScale Power Critical Heat Flux Correlations," Revision 1, issued November 2017 (ADAMS Accession No. ML17335A089), which the NRC staff has reviewed and approved (ADAMS Accession No. ML18214A480), describes the NSP2 and NSP4 CHF correlations and their development.

DCA Part 2 Tier 2, Section 4.4.2.9 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. Section 3.12 of TR-0915-17564, which 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. TR-0915-17564, 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. TR-0915-17564, Section 1.1, states that the analysis results presented in the report are for demonstration of the analytical methodology and that the applicant is not seeking approval of the results as part of the report. DCA Part 2, Tier 2, Section 4.4, does not provide the penalties that are used to establish the design limits for the NSP2 and NSP4 CHF correlations that support the NuScale DCA. The NRC staff needs to establish a finding that the penalties applied to the NSP2 and NSP4 CHF correlations provide suitably conservative safety limits for use in transient and accident analyses. Accordingly, the NRC staff issued **RAI 9462, Question 04.04-3**, asking the applicant to update DCA Part 2, Tier 2, Section 4.4.2.9.2, to provide the penalties and their bases used to set the CHF ratio limits for the NSP2 and NSP4 CHF correlations, respectively. By letter dated June 8, 2018, (ADAMS Accession No. ML18159A325), the applicant updated DCA Part 2, Tier 2, Section 4.4.2.9.2, to clarify that the minimum CHF ratio design limit includes a heat flux engineering uncertainty factor and a rod bow penalty that are based on the subchannel analysis methodology. The NRC staff conducted an audit as part of the review, which included the CHFR penalties (ADAMS Accession No. ML19092A361). During this audit, the NRC 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 by the applicant and the information obtained by the NRC staff during the audit, the staff finds the applicant's response acceptable because the updates clarify the 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 has confirmed that the applicant incorporated the proposed update into Revision 2 to DCA Part 2, Tier 2, Section 4.4.2.9.2.

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. 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. TR-0915-17564, Revision 2, Section 3.10.4, also discusses flux tilt. The NRC staff needs to establish a finding that the methodology for calculating T_q is suitably conservative. Accordingly, the NRC staff issued **RAI 9462**, **Question 04.04-4**, asking the applicant to update DCA Part 2, Tier 2, to describe the methodology used to determine T_q. By letter dated June 8, 2018, (ADAMS Accession No. ML18159A325), the applicant updated DCA Part 2, Tier 2, Section 4.4.2.10, to clarify 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. Based on the information discussed in this paragraph, the NRC staff finds the markup to DCA Part 2, Tier 2, Section 4.3.2.7, acceptable because it describes the methodology used to determine T_q, which the NRC staff found acceptable in SER Section 4.3.4.6. The NRC staff has confirmed that the applicant incorporated the proposed update into Revision 2 to DCA Part 2, Tier 2, Section 4.4.2.10.

DCA Part 2, Tier 2, Section 4.4.2.8, states that a thermal-margin trip (e.g., the overtemperature ΔT (OTΔT) trip in typical Westinghouse 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, 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, 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, describes the core bypass flowpaths as the reflector block cooling channels, the guide tubes, and instrument tube bypass flowpaths. DCA Part 2, Tier 2, Section 4.4.3.1.1.2, 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 NRC 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 NRC 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, TR-0176-50439, "NuScale Comprehensive Vibration Assessment Program Technical Report," Revision 0, issued December 2016 (ADAMS Accession No. ML17005A122), provides maximum design flow velocities based on the computational fluid dynamics analysis for components that affect core bypass flow. TR-0716-50439, Section 3.1.2.1, further states that "(thermal-hydraulic) TH analysis provides validated maximum design flow rate results based on testing." The NRC staff needs to establish a finding that the design-basis flow rate used in the subchannel analyses provides adequate margin to account for analysis uncertainties (e.g., manufacturing tolerances, boundary conditions, modeling simplifications). Accordingly, the NRC staff issued **RAI 9645, Question 04.04-6,** asking the applicant to further justify the basis for the bypass flow rate. The NRC staff is tracking this issue as **Open Item 4.4-07**.

4.4.4.3 Evaluation Methods

DCA Part 2, Tier 2, Section 4.4.2.7, 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, "Non-Loss-of-Coolant Accident Analysis Methodology," Revision 1, issued August 2017 (ADAMS Accession No. ML17222A827), describes the process for identifying the cases for subchannel analysis and extraction of boundary condition data. TR-0915-17564, which NRC staff has reviewed and approved (ADAMS Accession No. ML18338A031), details the application of VIPRE-01 to the NPM. The NRC staff is currently reviewing TR-0516-49416 and is tracking this as an **Open Item 4.4-08**.

4.4.4.4 Technical Specifications

DCA Part 2, Tier 2, Section 4.4.4.5.1, states that the primary contributors to pressure loss in the system are the fuel assembly and steam generator regions and that pressures 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 NRC staff compared the maximum and minimum design flow values in DCA Part 2 Tier 2, Table 4.4-2, with the RCS flow rates assumed in the transient and accident analyses in DCA Part 2, Tier 2, Table 15.0-6, 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.

Under 10 CFR 50.36(c)(2)(ii)(B), the NRC requires establishment of a TS 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 applicant did not identify RCS flow as an LCO to the NuScale GTS. Accordingly, the NRC staff issued **RAI 8773, Question No. 04.04-2; RAI 9174, Question 16-37**; and **RAI 9643, Question 04.04-5**, asking the applicant to provide sufficient justification that the RCS flow rate during normal operation will be maintained within the bounds of the transient and accident analyses. By letters dated April 25, 2017 (ADAMS Accession No. ML17214A896), and November 13, 2017 (ADAMS Accession No. ML17317A585), the applicant updated DCA Part 2, Tier 2, Section 4.4.5.2, and GTS 3.4.1 to provide a surveillance of the RCS flow during power ascension following refueling outages 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. However, in its response, the applicant did not address the potential for secondary-side perturbations or changes in the axial power shape to affect the overall RCS flow. The staff is still awaiting a response to **RAI 9643** and is tracking this as **Open Item 4.4-09**.

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, 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-included 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 outage 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 BWRs (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, issued May 1981 (ADAMS Accession No. ML003740137), 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 NRC staff compared the primary system components and fluid velocities of the subject reactors (i.e., the reactor designs that were approved for elimination of the LPMS) against the NPM. The NRC 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 NRC 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 NRC staff's comparison of the NPM to operating reactors, the NRC 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, 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. 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 NRC staff finds the RCS flow monitoring acceptable because it is more restrictive than the 24-hour monitoring criteria stated in NuScale DSRS, Section 4.4.

4.4.4.7 Instrumentation

DCA Part 2, Tier 2, Section 4.4.6.1, 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, NRC staff established a finding that the parameters available on the safety display and indication system (SDIS), which is located in the control room, that provide indication of ICC are a suitable combination of information that indicate 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 Section 18.7.4.5 of this SER, the NRC staff finds that the NuScale design provides adequate instrumentation that provide in the control room an unambiguous indication of ICC.

4.4.4.8 Stability

DCA Part 2, Tier 2, Section 4.4.1.4, 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, states that the NuScale flow stability protection solution uses a regional exclusion solution as described in TR-0516-49417, “Evaluation Methodology for Stability Analysis of the NuScale Power Module,” Revision 0, issued July 2016 (ADAMS Accession No. ML16250A851). DCA Part 2, Tier 2, Section 4.4.7, further discusses the flow stability evaluation for the NPM and states that TR-0516-49417 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 evaluates the stability analysis. The NRC staff is currently reviewing TR-0516-49417. The staff is tracking this as **Open Item 4.4-10**.

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

Because of the open items identified in SER Section 4.4.4, the NRC staff cannot conclude that the thermal-hydraulic design of the NPM meets the requirements of GDC 10, or GDC 12.

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 letter dated June 12, 2017, (ADAMS Accession No. ML17163A436) and for the response to **RAI 9057**, (ADAMS Accession No. ML17249A662) by letter dated September 6, 2017.

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 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.

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 Use of Sensitized 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.

COL Information or Action Items: There are no COL information items or action items 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 structures, systems, and components (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
- 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, 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, are austenitic stainless steel (SA-965, Type 304LN). The fabrication of the CRDM pressure housing will use Types 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 material 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.

Austenitic Stainless Steel Components. 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," 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, 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 megapascals (90 kilopounds per square inch) 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.

<u>Other Materials</u>. 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 and Table 4.5-1, 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 565 degrees Celsius (1,050 degrees Fahrenheit), 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 1,093 degrees Celsius (2,000 degrees Fahrenheit) 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.

Inspections, Tests, Analyses, and Acceptance Criteria

The applicant delineated the ITAAC associated with DCA Part 2, Tier 2, Section 4.5.1, in DCA Part 2, Tier 1, Section 2.1, supplemented by letter dated June 12, 2017, (ADAMS Accession No. ML17163A436). The Tier 1 information and the ITAAC items include the following:

- The system description in DCA Part 2, Tier 1, Section 2.1.1, specifies that the CRDM pressure housings (latch housing, rod travel housing, and rod travel housing plug) form the pressure boundary between the environment inside the RVP and the containment vessel.

- ITAAC Item 2 specifies that the as-built ASME Code components identified in Table 2.1-2, including the CRDM, meet the requirements of ASME Code, Section III, as documented in the ASME Code data reports.

The staff finds the system description of the CRDM components and ITAAC Item 2 acceptable because the ASME Code data reports will provide the required information that verifies that the as-built CRDM assemblies will be designed, constructed, inspected, and tested in accordance with ASME Code, Section III.

4.5.1.5 Combined License Information Items

There are no COL information items from DCA Part 2, Tier 2, Table 1.8-2 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 the 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.

DCA Part 2, Tier 2, Revision 0, 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 describes 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 90,000 pounds per square inch.

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 describes acid pickling as "avoided on stainless steel" and "not used on sensitized austenitic stainless steel."

<u>Other Materials</u>

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 precipitation-hardened stainless steel 17-4, Stellite 3, and Alloy 718. Note that, with Stellite 3, which is a cobalt-based alloy, low-cobalt or cobalt-free alloys may be substituted if testing qualifies their resistance to wear and corrosion.

4.5.2.3 Regulatory Basis

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

- GDC 1 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, Article 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. 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 NuScale evaluation and inclusion of inspections was acceptable as they pertain to the consideration of the above-listed degradation mechanisms.

By letter August 2, 2017 (ADAMS Accession No. ML17214A895), the applicant clarified that crevice corrosion was not a potential degradation mechanism for the NuScale design. The staff reviewed the applicant's response and, based on large light-water reactor operating experience, 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, 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, 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, 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; and the water chemistry requirements for oxygen content in DCA Part 2, Tier 2, Section 5.2, Table 5.2-5. 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, lists several materials as “Other Materials.” Precipitation-hardened stainless steel 17-4 PH, Grade 630, is identified with a corresponding heat treatment requirement in accordance with ASME Code, Section II, Material Specification H1100. DCA Part 2, Tier 2, Section 4.5.2.5, identifies cobalt-containing alloys (i.e., either Stellite 3 casting or a “qualified low-cobalt or cobalt-free alloy) for use on wear surfaces of the core support locking assemblies. Finally, DCA Part 2, Tier 2, Section 4.5.2.5, cites Alloy 718 for use in threaded fasteners with a reference to DCA Part 2, Tier 2, Section 3.13.1.

The staff evaluated the identified materials and associated heat treatments and found the discussion of “Other Materials” acceptable because they are 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 control rod drive system (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, describes the mechanical design of the CRDM. DCA Part 2 Tier 2, Section 7.0.4, 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, Sections 9.3.4, discusses the CVCS in more detail.

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 shall 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 shall be provided and shall 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 shall 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 shall 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 shall 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, 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, 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, describes the separation between the safety-related MPS and module control system that is not safety-related. The staff’s Chapter 7 SER 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 control rod assembly 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, 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 RCBP 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 343.3 degrees Celsius (650 degrees Fahrenheit) and an RPV design pressure of 2,100 pounds per square inch, 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. 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 provides the cooling requirements for the CRDMs. In its response to **RAI 9242, Question 04.06-1,** the applicant stated that the CRDM coils are designed using a Class N insulation system, which is rated to 200 degrees Celsius (392 degrees Fahrenheit), and that, to account for some margin, DCA Part 2 specifies that the maximum temperature design criterion for the CRDM is 356 degrees Fahrenheit. 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).

A single failure in the CRDS should not prevent the system from performing its safety-related function. DCA Part 2, Tier 2, Section 4.6.2, states that the effectiveness of the CRDS, despite possible single failures, is demonstrated in DCA Part 2, Tier 2, Chapter 15, and shows that the CRDS performs a reactor trip when plant parameters exceed the reactor trip setpoint. 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 of this SER. 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 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 evaluated 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 24, 2018, describes the staff's criteria for evaluating the GDC 27 exemption request. SER Section 15.0.6 evaluates this exemption and the proposed PDC and identifies concerns with the evaluation of thermal-margin and probability of occurrence of a potential post-trip return to power. The NRC staff has issued **RAI 8771, Question 15-1,** and **RAI 9505, Question 15-18,** to address these items. The NRC staff is tracking the thermal-margin evaluation and probability of occurrence associated with a return to power described in the exemption to GDC 27 as **Open Item 4.3-02** and **Open Item 4.3-03**, respectively.

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 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 AMSE Class 1 requirements.

DCA Part 2, Tier 2, Section 3.1.3.9, 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, provides the reactivity requirements for control rods; DCA Part 2, Tier 2, Section 15.4.1, defines the maximum allowed withdrawal rate of a CRA to be 38.1 cm (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, 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 NRC staff has identified that long-term reactivity control following an AOO or postulated accident depends upon the distribution of soluble boron throughout the RCS. Accordingly, the NRC staff has issued **RAI 8930, Question 15-27,** asking the applicant to describe and justify its methodology for evaluating the boron distribution during long-term cooling following ECCS actuation. The NRC staff is tracking this as **Open Item 4.3-04**.

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 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 1.5.1.7, 1.5.1.11, and 1.5.1.12, 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 NuScale 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 conditional 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 GDC 29 because the tests and design of the CRDS ensure an extremely high probability that the CRDS will accomplish its safety function in the event of an AOO.

Because of the open items identified in SER Section 4.6.4, the NRC staff cannot conclude that the functional design of the CRDS meets GDC 27 or GDC 28.