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 (the applicant), US460 Standard Design Approval Application (SDAA), Part 2, "Final Safety Analysis Report (FSAR)." The staff's regulatory findings documented in this report are based on Revision 1 of the SDAA, dated October 31, 2023 (Agencywide Documents Access and Management System Accession No. ML23304A335) as supplemented by subsequent docketed information described in each subsection, which includes FSAR changes to Revision 1. The precise parameter values, as reviewed by the staff in this SER, are provided by the applicant in the SDAA using the English system of measure. Where appropriate, the NRC staff converted these values for presentation in this SER to the International System (SI) units of measure based on the NRC's standard convention. In these cases, the SI converted value is approximate and is presented first, followed by the applicant-provided parameter value in English units within parentheses. If only one value appears in either SI or English units, it is directly quoted from the SDAA and not converted.

4.1 Summary Description

The design of the NuScale Power Module 20 (NPM-20) is a self-contained nuclear steam supply system comprising a reactor core, a pressurizer, and two steam generators (SGs) integrated within the reactor pressure vessel (RPV) and housed in a compact steel containment vessel CNV).

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

4.2 Fuel System Design

4.2.1 Introduction

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

4.2.2 Summary of Application

FSAR Section 4.2, "Fuel System Design," describes the fuel system design, as summarized, in part, below for the NPM-20 reactor.

4.2.2.1 Fuel Assembly Description

The NuScale fuel assembly contains 264 fuel rods, 24 guide tubes, and 1 instrumentation guide tube in the center of a 17 by 17 square array that is held together by a bottom and top nozzle and guide tubes welded to four spacer grids. A fifth lower grid is captured by rings welded to the guide tubes. The guide thimble tubes serve as channels to guide control rod assemblies (CRAs) over their entire length of travel. In-core instrumentation is inserted in the central guide tube of selected fuel assemblies.

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

4.2.2.2 Fuel Rod Description

The applicant stated that the fuel rods consist of uranium dioxide (UO_2) cylindrical ceramic pellets with a maximum enrichment up to 4.95 weight percent uranium (U)-235 and a round wire nickel-chromium-based alloy X-750 compression spring located in the plenum, encapsulated within an M5^{TM} tube that serves as the fuel cladding. The fuel rods are internally pressurized with helium during assembly.

The applicant stated that the UO_2 pellets are concave at both ends to better accommodate thermal expansion and fuel swelling. The nominal density of the UO_2 in the pellets is 96.5-percent theoretical density.

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

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

4.2.2.3 Burnable Absorber Poisoned Fuel Rod Description

The applicant stated that gadolinia burnable poison is added to some of the fuel rods to reduce the excessive reactivity in fuel that has higher enrichment in order to control power peaking and reduce the required soluble boron concentration. In the poisoned fuel rods, gadolinium oxide (Gd_2O_3) is mixed with the UO_2. The poisoned rod is mechanically similar to nonpoisoned fuel rods. The NPM-20 design does not use fuel with different U-235 enrichments in the axial direction, and all safety analyses in the current application assume uniform axial U-235 enrichment in the fuel rods, poisoned or nonpoisoned as stated in FSAR Section 4.3.2.1, “Nuclear Design Description.” The total column length and other geometric dimensions are the same as the regular fuel rods. However, the density of the poisoned fuel varies depending on the quantity of Gd_2O_3 that is added to the fuel. There is no significant change in the mechanical or chemical properties of the poisoned fuel rods because Gd_2O_3 takes only a small fraction of the fuel mass.

4.2.2.4 Control Rod Assembly Description

The CRAs consist of 24 neutron absorber elements connected with a stainless-steel spider hub that couples to the control rod drive mechanism (CRDM) shaft extension. The neutron absorber elements contain silver-indium-cadmium and boron carbide neutron (B_4C) absorbers in a stainless steel clad.

The CRA design analysis considered potential failure mechanisms, including stress and loads, strain, creep collapse, fatigue, wear, internal pressure, and component melting. The applicant presented the results of its updated analyses in Technical Report TR-117605-P, Revision 0, “NuFuel-HTP2™ Fuel and Control Rod Assembly Designs”, issued December 2022 (ML23304A336 (nonproprietary), ML23304A337 (proprietary)), which is incorporated by reference into FSAR Section 4.2 as discussed in Section 4.2.2.5 of this SER. FSAR Section 4.3.2.5, “Control Rod Patterns and Reactivity Worth,” supplemented by ML24346A167, provides the maximum 1.43 percent element-wise boron (B)-10 depletion for a 20 effective full power year (EFPY) lifetime. The applicant concluded that the control rod will meet the performance requirement over 20 EFPY design life for base load operation.

4.2.2.5 Design Evaluation

The applicant stated that the design evaluations of the fuel rod, fuel assembly, and in-core control components consider events during normal operations, AOOs, and postulated accidents (including infrequent events). FSAR Chapter 15, “Transient and Accident Analyses,” evaluates AOOs and postulated accidents.

The applicant summarized the design evaluations for each fuel component and concluded that the appropriate specified acceptable fuel design limits (SAFDLs) are acceptable. The staff determined that the methodologies used in the analyses apply to the applicant’s fuel design in the referenced topical reports (TR)-0116-20825-P-A, Revision 1, “Applicability of AREVA Fuel Methodology for the NuScale Design,” issued February 2018 (ML18040B307 (proprietary), ML18040B306 (nonproprietary)), and TR-0716-50351-P-A, Revision 1, “NuScale Applicability of AREVA Method for the Evaluation of Fuel Assembly Structural Response to Externally Applied Forces,” issued May 2020 (ML20122A248 (nonproprietary), ML20122A249 (proprietary)), as supplemented by TR-108553-P-A, Revision 0, “Framatome Fuel and Structural Response Methodologies Applicability to NuScale,”, issued October 2022 (ML22292A312 (nonproprietary), ML22292A313 (proprietary)). TR-117605-P, Revision 0, supplemented by ML24346A193, provides detailed analyses of the NuScale fuel assembly using the approved methods as mentioned above.

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

Technical Specifications: FSAR Chapter 16, “Technical Specifications,” does not provide technical specifications (TS) associated with FSAR Section 4.2.

Technical Reports: FSAR Table 1.6-2, “NuScale Referenced Technical Reports,” identifies TR-117605-P, Revision 0, supplemented by ML24346A132, as incorporated by reference into FSAR Section 4.2.

4.2.3 Regulatory Basis and Relevant Guidance

The following NRC regulations are the relevant requirements for the NuScale NPM-20 design:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” and 10 CFR 50.34, “Contents of applications; technical information,” as they relate to the cooling performance analysis of the emergency core cooling system (ECCS), using an acceptable evaluation model, and establishing acceptance criteria for light-water nuclear power reactor ECCSs

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

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

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

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

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

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

4.2.4 Technical Evaluation

The staff reviewed the fuel design for the NuScale US460. 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.

FSAR Section 4.2 references TR-0116-20825-P-A, Revision 1, as supplemented by TR-108553-P-A, Revision 0, to justify the applicability of various codes and methods for the analyses of the applicant’s fuel designs. The SERs for TR-0116-20825-P-A, Revision 1 and TR-108553-P-A, Revision 0, give the staff’s evaluation of the applicability of these codes and methods. The staff reviewed the technical contents together with the SER for TR-0116-20825-P-A, Revision 1 and TR-108553-P-A, Revision 0, and finds the methodologies described in the TRs remain applicable to the fuel for the NPM-20 design.

To assist its review of the NPM-20 reactor design, the staff developed independent analyses for the fuel rod performance using the FAST (Fuel Analysis under Steady-state and Transients) code (PNNL-35701, “FAST-1.2.1: A Computer Code for Thermal-Mechanical Nuclear Fuel Analysis under Steady-State and Transients,” issued March 2024 (ML24177A227). The staff analyses performed using FAST calculated the cladding corrosion, rod internal pressure, power to 1 percent cladding strain, power to melt, and fatigue of the fuel rods. In general, the FAST predictions were in reasonable agreement with the applicant’s calculations, which were performed using COPERNIC, a fuel performance analysis computer code, which the staff approved in TR-0116-20825-P-A, Revision 1 and TR-108553-P-A, Revision 0. This outcome was expected since the predecessor to FAST, FRAPCON, was used to assess the COPERNIC code for light-water reactor (LWR) fuel designs that are used in currently operating pressurized-water reactors (PWRs). The sections below summarize the staff's review and conclusions.

The applicant requested an exemption from GDC 35 and proposed PDC 35 in lieu of GDC 35. The applicant's PDC 35 is functionally identical to GDC 35, except for the discussion related to electric power. The modification to the electric power discussion in PDC 35 is tied to the exemption request for GDC 17, "Electric Power Systems," and proposed PDC 17. Sections 8.1.2 through 8.1.4 of this SER provide the staff's evaluation of the exemption to GDC 17 and, by extension, the electric power provision of GDC 35. Sections 6.3, 15.6.5, and 15.6.6 of this SER evaluate the ECCS against the proposed PDC 35.

4.2.4.1 Design Bases

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

4.2.4.1.1 Fuel System Damage

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

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

<u>Stress/Strain Limits</u>. Section 4.1.1, "Stress and Loading Limits," of TR-117605-P, Revision 0, supplemented by ML24346A197, provides a stress and loading analysis. The shipping and handling stress analysis and the fuel assembly/component stress analysis were performed in accordance with EMF-92-116(P)(A), Revision 0, "Generic Mechanical Design Criteria for PWR Fuel Designs," issued February 1999 (ML003681173). The clad stress analysis and the cladding buckling analysis were performed in accordance with BAW-10227P-A, Revision 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," issued June 2003 (ML15162B043). The staff finds that these analyses demonstrate that the calculated stresses and that loadings are all within the design criteria for all required conditions. The staff

noted that the bounding calculation results provided showed small margin to the limit in several areas, however, the applicant confirmed that the appropriate uncertainties specified by EMF-92-116(P)(A), Revision 0 and BAW-10227P-A Revision 1 were applied). On these bases, the staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements on stress and loading in accordance with the guidance in SRP Section 4.2.

Fuel Assembly Component Fatigue. Section 4.1.1 of TR-117605-P, Revision 0, provides fatigue strength calculations for fuel assembly components and structural connections performed in accordance with the O'Donnell-Langer curve, published by W.J. O'Donnell and B.F. Langer, "Fatigue Design Basis for Zircaloy Components," Nuclear Science and Engineering, Volume 20, pp. 1-12, September 1964. The fatigue analysis evaluates cyclic loading due to normal operation and AOOs combined with the operating-basis earthquake (OBE), for a total of 137 transients over the life of the fuel. The results show that the combined OBE is less than one-third of the SSE ground motion and is enveloped by the SSE analysis. The staff finds also that the curve is an appropriate limit for the fuel assembly component fatigue calculation. On this basis, the staff concludes that analyses demonstrate that the calculated fatigue usage factor is significantly less than the design fatigue limit and acceptable. Section 4.1.2, "Cladding Fatigue," of TR-117605-P, Revision 0, provides a sample analysis of the fuel design using the COPERNIC fuel rod analysis code, performed in accordance with BAW-10227P-A, Revision 1 that showed a low fatigue usage factor. The staff performed an independent analysis using the FAST computer code with the same transients and the result shows a similar low fatigue usage factor. The staff finds that these analyses demonstrate a calculated cladding fatigue usage factor that is significantly below the limit of 0.9 for UO_2 and UO_2-Gd_2O_3 fuel with a representatively large number of operating transients.

Based on its review and confirmatory analyses, the staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements on fuel assembly component fatigue in accordance with the guidance in SRP Section 4.2.

Fuel Fretting. Section 4.1.3, "Fretting," of TR-117605-P, Revision 0, summarizes fretting test results and evaluation performed in accordance with EMF-92-116(P)(A), Revision 0. The staff notes that these tests were performed to support review of the previous NuScale NPM-160 design, but the flow conditions of the tests are bounding relative to both the current (NPM-20) and previous cores (NPM-160). The water momentum flux for the tests were appreciably greater than the nominal momentum flux for the NPM-20 core. The staff finds that, by using fretting tests, this evaluation 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 meets the regulatory requirements on fretting and flow-induced vibrations in accordance with the guidance in SRP Section 4.2.

Oxidation and Hydriding. Section 4.1.4, "Oxidation, Hydriding, and Crud Buildup," of TR-117605-P, Revision 0, provides an oxidation, hydriding, and crud buildup analysis performed in accordance with BAW-10231P-A, Revision 1, "COPERNIC Fuel Rod Design Computer Code," issued January 2004 (ML042930233). Using the COPERNIC code and a bounding power history envelope, this analysis demonstrates that the calculated oxide thickness is below the design limit of 100 micrometers (0.00394 inch). The results from COPERNIC were in reasonable agreement with the results of the confirmatory analysis performed using the FAST computer code (ML24177A227) and the same design inputs. The staff finds that the corrosion limit restrains the hydrogen pickup, and crud buildup is built into the corrosion thickness.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements of GDC 10 with considerations of cladding oxidation, hydriding, and crud buildup as discussed in SRP Section 4.2.

Dimensional Changes. Section 4.1.5, "Fuel Rod Bow," of TR-117605-P, Revision 0, provides a fuel rod bow analysis using the methodology in XN-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing," Supplements 1 through 4, issued October 1983 (ML081710709). Section 4.1.5 of TR-117605-P, Revision 0, states that rod bow penalties are derived for both linear heat generation rate (LHGR) and critical heat flux (CHF) based on these calculations. FSAR Section 4.4 discusses application of these penalties. The staff audited (ML24211A089) the rod bowing calculation package and confirmed that the calculation used the methodology provided in TR-117605-P, Revision 0. The staff performed a hand calculation of rod bowing and was able to duplicate the applicant's results. On these bases, the staff finds that this analysis demonstrates that the NuScale fuel is within the current experience base. The staff's evaluation of the impact of rod bow on thermal limit and SAFDLs are discussed in Section 4.4.4 of this SER.

Section 4.1.6, "Axial Growth," of TR-117605-P, Revision 0, supplemented by ML24346A195, provides a rod growth analysis for the life of the fuel in accordance with EMF-92-116(P)(A), Revision 0. The staff confirmed that the analysis follows the methodology in EMF-92-116(P)(A), Revision 0 for calculating a maximum fluence value that would result in unacceptable rod growth. This calculated fluence value is significantly greater than the fluence expected based on the fuel burnup limit. The staff finds that the analysis demonstrates, using the worst-case tolerances and growth models, that sufficient clearance is maintained between the fuel assembly and top nozzle and between the fuel assembly and core plate for the burnup range requested.

Section 4.1.7, "Fuel Assembly Distortion Evaluation," of TR-117605-P, Revision 0, discusses the potential for fuel rod distortion to affect control rod insertion. The applicant stated that applicable Framatome operating experience shows little in-reactor fuel distortion. Furthermore, NuScale fuel has a greater lateral stiffness than Framatome 17 by 17 fuel, which, the staff notes, suggests an improved resistance to fuel assembly distortion compared to recent Framatome fuel used in operating PWRs.

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

Rod Internal Pressure. Section 4.1.8, "Fuel Rod Internal Pressure," of TR-117605-P, Revision 0, provides a fuel rod internal pressure analysis performed in accordance with BAW-10231P-A Revision 1. The applicant used the COPERNIC code and a bounding condition in fuel rod internal pressure analysis. The staff audited (ML24211A089) the fuel rod internal pressure calculation package. The staff confirmed that the results of the applicant's analysis demonstrate significant margin exists between the rod internal pressure limit of 13.79 megapascals (MPa) (2,000 pounds force per square inch (psi)) (equal to reactor coolant system (RCS) pressure) and the calculated maximum internal pressure as given in Section 4.1.8 of TR-117605-P, Revision 0.

The staff performed confirmatory analyses using the FAST computer code with the same power history and bounding conditions. The results of the confirmatory calculations showed good agreement with the rod internal pressure calculated by the applicant.

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

Fuel Assembly Liftoff. Section 4.1.9, “Assembly Liftoff,” of TR-117605-P, Revision 0, provides a bounding fuel assembly liftoff analysis performed in accordance with EMF-92-116(P)(A), Revision 0. The staff finds that, by using bounding flow rates for the limiting AOO, this analysis demonstrates that significant margin to fuel assembly liftoff exists.

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

Control Rod Insertability. Section 4.3.5, “Fuel Assembly Structural Damage from External Forces,” of TR-117605-P, Revision 0, addresses fuel assembly structural damage, which could prevent control rod insertability. Section 4.2.4.5 of this SER provides the staff’s evaluation of control rod insertability under seismic and loss-of-coolant accident (LOCA) loads.

4.2.4.1.2 Fuel Rod Failure

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

Hydriding. Section 4.2.1, “Internal Hydriding,” of TR-117605-P, Revision 0, provides an internal hydriding analysis performed in accordance with EMF-92-116(P)(A), Revision 0. The staff finds that, by using fabrication limits for fuel pellet moisture, this analysis demonstrates failure caused by internal hydriding will be insignificant.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements of GDC 10 with respect to internal hydriding in accordance with the guidance in SRP Section 4.2.

Cladding Collapse. Section 4.2.2, “Cladding Collapse,” of TR-117605-P, Revision 0, provides a cladding collapse analysis performed in accordance with the methodology defined in BAW-10084P-A, Revision 3, “Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse” (CROV computer code), issued October 1980 (ML19260G472 (proprietary), ML20002D564 (non-proprietary)), using the creep model from BAW-10227P-A, Revision 1. This analysis used the initial conditions specified by the methodology in BAW-10231P-A, Revision 1, and relied on maximum calculated fast flux and cladding temperatures at each time step as proposed by the revision to the creep collapse methodology. The staff finds that, by using the CROV code initiated with COPERNIC, this analysis demonstrates that significant margin to creep collapse exists.

Based on its review of the fuel performance analyses, the staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements of GDC 10 for cladding collapse in accordance with the guidance in SRP Section 4.2.

Overheating of the Cladding. Based on the guidance in SRP Section 4.2, failures are assumed to be precluded if the thermal-margin criteria (departure from nucleate boiling ratio (DNBR)) are satisfied. SRP Section 4.2 also states that violation of the DNBR limits is not allowed for normal operation and AOOs. As shown in FSAR Table 15.0-2, "Acceptance Criteria —Thermal-Hydraulic and Fuel," the applicant self-imposed stricter acceptance criteria; namely, that SAFDLs will also be met for postulated accidents. Section 4.4 of this SER discusses the DNBR margin analysis. The various design-basis event (DBE) evaluations, as detailed in Chapter 15 of this SER, document the cladding temperature under postulated accident conditions.

Overheating of Fuel Pellets. Section 4.2.4, "Overheating of Fuel Pellets," of TR-117605-P, Revision 0, provides an analysis on the overheating of fuel pellets performed in accordance with BAW-10231P-A, Revision 1. Confirmatory analyses performed using the FAST computer code resulted in similar power-to-melt values at various burnup levels. The staff finds that, by using a bounding analysis with the COPERNIC code, this analysis demonstrates significant margin between the NuScale power limits and the fuel melting limits.

The staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements of GDC 10 on overheating of the fuel pellets in accordance with the guidance in SRP Section 4.2.

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

Pellet/Cladding Interaction. No generic criterion for fuel failure resulting from pellet/cladding interaction or pellet/cladding mechanical interaction exists. SRP Section 4.2 states that cladding strain and fuel melting limits can be used as surrogate criteria for fuel failure resulting from pellet/cladding interaction or pellet/cladding mechanical interaction.

Section 4.2.6, "Pellet-Cladding Interaction," of TR-117605-P, Revision 0, provides a transient clad strain analysis performed in accordance with BAW-10231P-A, Revision 1. Confirmatory analyses performed with FAST computer code using the same transients at various burnup levels resulted in a similar transient power necessary to induce 1 percent cladding strain. The staff finds that by using a bounding analysis with the COPERNIC code, this analysis demonstrates significant margin between the NPM-20 power limits and the power level required to reach the 1 percent cladding strain limit specified in SRP Section 4.2.

Based on review of the information in the FSAR, responses to audit questions, and independent calculations performed using FAST, the staff concludes that the applicant demonstrated that the NuScale fuel design meets the regulatory requirements for transient cladding strain and fuel melting limits and, therefore, has reasonable assurance that the fuel will not fail as a result of pellet-cladding interaction or pellet-cladding mechanical interaction.

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

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

4.2.4.1.3 Fuel Coolability

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

4.2.4.2 Description and Design Drawings

The staff reviewed the fuel system description and design drawings in FSAR Section 4.2. TR-117605, Revision 0, gives additional fuel assembly design information. The staff found that the applicant followed the guidance in SRP Section 4.2 by providing an accurate representation of the fuel system; therefore, the staff finds the fuel system description and design drawings acceptable.

4.2.4.3 Design Evaluation

4.2.4.3.1 Operating Experience

Chapter 3, of TR-117605-P, Revision 0, notes that Framatome's 17 by 17 High Temperature Performance (HTP™) fuel assemblies are similar in material and design to the NuFuel-HTP2™ fuel assemblies. The applicant used the operating experience described in TR-117605-P, Revision 0, to justify the models used to analyze the NuFuel-HTP2™ fuel assembly in the NPM-20 design. The staff evaluated the applicability of the models to the NuScale NPM-160 module design in its SER for TR-0116-20825-P-A, Revision 1 and in its SER for TR-108553-P-A, Revision 0, which is a supplement for the application to the NPM-20 module design.

FSAR Section 4.2.4.1, "Operating Experience," discusses Framatome's domestic and international operating experience in support of the NuScale fuel design. The staff compared the NuScale fuel assembly components with Framatome's operating fleet database and notes that significant experience has been developed for the same components. The staff further notes that the NuScale plant operational parameters important to fuel behavior are not significantly different from those in the Framatome operating fleet; therefore, the staff finds that the Framatome operating experience applies to NuScale fuel assemblies.

4.2.4.3.2 Testing, Inspection, and Surveillance Plans

SRP Section 4.2 provides review guidance on testing, inspection, and surveillance plans. FSAR Section 4.2.4.2, "Prototype Testing," presents the prototype testing of the NuScale fuel assemblies, CRAs, and fuel assembly components. The testing covers areas related to fuel assembly structural response, which can differ from the full-sized Framatome operating fleet database. The staff reviewed the prototype testing discussed in FSAR Section 4.2.4.2, and in Chapter 5, "Fuel Assembly Testing," of TR-117605-P, Revision 0. Based on its review of the information provided by the applicant, the staff finds that the testing follows the methodology in referenced and approved ANP-10337P-A, Revision 0, "PWR Fuel Assembly Structural Response to Externally Applied Dynamic Excitations," issued April 2018 (ML18144A816), and is, therefore, acceptable.

4.2.4.4 Testing, Inspection, and Surveillance Plans

FSAR, Section 4.2.4, “Testing and Inspection Plan,” contains the testing and inspection plan for the fuel design. Because the NuScale fuel design is similar to existing Framatome 17 by 17 fuel assembly designs, the staff finds that the related operating and testing experience with Framatome’s fuel designs is applicable to the NPM-20 design.

FSAR Section 4.2.4.2, states that the tests on fuel assemblies and CRAs are used to determine the performance characteristics of the fuel assemblies, CRAs, and fuel assembly components.

FSAR Section 4.2.4.3, “Manufacturing Testing and Inspection,” states that the fuel and CRA will be manufactured and inspected under a quality assurance program in accordance with Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50 as described in FSAR Chapter 17, “Quality Assurance and Reliability Assurance”. The component testing under this program includes nondestructive examinations (NDEs) and destructive examinations to support qualifications.

In FSAR Section 4.2.4, the applicant stated that additional inspections, including onsite receipt inspections, online fuel system monitoring, and post-irradiation monitoring, are planned for the fuel assembly and CRAs from the first licensed module.

Based on the description in FSAR Section 4.2.4.3, the staff finds that the applicant’s testing, inspection, and surveillance plans are sufficient to ensure that (1) the fuel is manufactured to the design specifications and (2) fuel performance outside of the predictions made by the fuel analysis will be detected. The applicant’s methods are consistent with the guidance in SRP Section 4.2 and, therefore, are acceptable. The staff notes that Combined License (COL), COL Item 14.3-1 states that the COL applicant would be responsible for implementing testing, inspection, and surveillance plans, and the staff would verify such plans at the COL stage as indicated in Section 13.3 of this SER.

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

4.2.4.5.1 Design Requirements and Acceptance Criteria

FSAR Section 4.2.1.5, “Fuel Assembly Structural Design,” defines the bases for the fuel assembly structural design. FSAR Section 4.2.1.5.10, “Loss-of-Coolant Accident and Seismic Loading,” states the following:

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

FSAR Section 4.2.2, “Description and Design Drawings,” and Section 4.2.3, “Design Evaluation,” define assembly-component-specific design requirements that fulfill the above high-level design requirement. The FSAR refers to TR-117605-P, Revision 0, for further details regarding these design requirements. Section 4.3.5 of TR-117605-P, Revision 0, states that specific acceptance criteria for fuel assembly components are identified in ANP-10337P-A, Revision 0, except for fuel cladding acceptance criteria. Section 4 of ANP-10337P-A, Revision 0, defines component acceptance criteria that satisfy the underlying regulatory requirements in GDC 2 and Appendix S to 10 CFR Part 50, except for cladding acceptance criteria. The staff previously reviewed and accepted the analytical models, testing protocols, and acceptance criteria in ANP-10337P-A, Revision 0, which sufficiently covers most components except cladding. The applicability of ANP-10337P-A, Revision 0, to the fuel design and operating parameters of the NuScale NPM-160 design is documented in TR-0716-50351-P-A, Revision 1, and in TR-108553-P-A, Revision 0 for the fuel design and operating parameters of the NuScale NPM-20 design. TR-0716-50351-P-A, Revision 1, includes several modifications to the ANP-10337P-A, Revision 0, methods to make it applicable to the fuel design and operating parameters of the NuScale NPM-160 design.

The FSAR defines fuel cladding acceptance criteria in Section 4.2.1, "Design Bases," that were defined in BAW-10227P-A, Revision 2, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel)" (ML23037A888 (nonproprietary), ML23037A926 (proprietary)). The staff previously reviewed and accepted the stress limits of BAW-10227P-A, Revision 2. However, it is important to note that the analysis methodology (ANP-10337P-A, Revision 0) and the cladding stress limits and acceptance criteria (BAW-10227P-A, Revision 2) are separate documents that were not reviewed together. BAW-10227P-A, Revision 2 permits a certain amount of fuel cladding plastic deformation that was not considered when the staff approved ANP-10337P-A, Revision 0. When the staff reviewed ANP-10337P-A, Revision 0, BAW-10227P-A, Revision 1, was the most current M5 material document. It restricted primary membrane plus bending stress to the material yield limit, so plastic deformation of the fuel rods was not a concern when ANP-10337P-A, Revision 0, was approved. BAW-10227P-A, Revision 2, raises the stress limits and acceptance criteria to permit a certain amount of permanent deformation of the fuel rods in faulted conditions. TR-117605-P, Revision 0, Section 4.3.5.2.3, "Stress Analysis," supplemented by ML24346A165, states that, if predicted stress intensity exceeds the elastic limit, analysis and testing will be conducted to evaluate fuel rod deformation in accordance with Appendix E to ANP-10337P-A, Revision 0. Permanent fuel rod deformation must remain below the level that would incur a penalty on either minimum critical heat flux ratio or local power peaking as specified in XN-75-32(P)(A). The NRC staff found XN-75-32(P)(A) applicable to fuel systems such as that used in the NPM-20 in the NRC staff SERs for TR-0116-20825-P-A, Revision 1 and TR-108553-P-A, Revision 0.

In addition, the staff finds that BAW-10227P-A, Revision 2, provides reasonable assurance that adherence to the BAW-10227P-A, Revision 2 limits will ensure that fuel rods will not fragment. Specifically, irradiated fuel cladding data is presented to demonstrate that the stress limits are sufficient to preclude cladding structural failure. The stress limit criteria are based on American Society of Mechanical Engineers (ASME) code stress analysis methods. In these acceptance criteria, the primary membrane plus bending stress limit allows the material to exceed its yield strength, and a relatively small amount of plastic deformation is permitted.

The ductility of irradiated zirconium alloys is known to decrease from its unirradiated state. Although the stress limits permit some amount of plastic deformation, fuel rod fragmentation as a result of loading should not occur. The staff notes that the test data in BAW-10227P-A, Revision 2, show that sufficient ductility (for this specific cladding material) remains to support the stress limit. The stress criteria are sufficient to demonstrate that the fuel will not structurally fail.

In addition, the staff notes that, in the analyses, the applicant used the ASME Service Level C stress criteria that permit an unspecified amount of plastic strain and permanent deformation. The stress limit does not permit what the ASME Code calls "gross deformation," but the amount of deformation the stress limits do permit has not been assessed. To address the concern that the ASME Service Level C stress criteria permit an unspecified amount of plastic strain and permanent deformation, NuScale added an acceptance criterion to limit fuel rod permanent deformation to a value related to fuel rod bowing. The fuel rod deformation limit does not need to be evaluated when cladding stress remains below the yield strength, which is the case for the NPM-20.

The staff audited the supporting documents (ML24211A089) for the applicant's fuel rod stress analysis of the NPM-20. The audited analysis did not calculate fuel rod permanent deformation or compare deformation to bowing criteria because the predicted stress remains below the elastic limit. The fuel rod deformation limit acceptance criterion is only applicable when the calculated primary membrane plus bending stress exceeds the yield strength of the cladding material, so it is not necessary to calculate fuel rod permanent deformation unless the yield stress is exceeded. Section 4.3.5.2.3 of TR-117605-P, Revision 0, supplemented by ML24346A165, commits to additional analyses for fuel rod permanent deformation if it is necessary for future design changes in reactor or fuel.

The staff notes that the fuel rod permanent deformation that is permitted by the BAW10227PA, Revision 2, stress limits is expected to be relatively small and have a negligible impact on safety. The fuel rod deformation limit criterion is expected to ensure that fuel rod permanent deformation remains negligible to reactor operations, but it was not necessary to confirm this expectation because the fuel rod deformation limit acceptance criterion is only applicable when the calculated primary membrane plus bending stress exceeds the yield strength of the cladding material, and that was not the case in this review. The staff concludes that permanent fuel rod deformation is not a concern for the NPM-20 as presented in the FSAR, supplemented by ML24346A165, based on (1) the current analysis results and (2) the fact that the ANP-10337-P-A, Revision 0, Appendix E methodology was reviewed and approved for a different purpose (i.e., assessing permanent deformation of guide tubes). On these bases, the staff finds it acceptable for the fuel stress analyses to reference BAW10227PA, Revision 2 stress limits in conjunction with the proposed method and criterion for ensuring that permanent fuel rod deformation would not incur a rod bow penalty.

It important to note that this fuel rod permanent deformation limit is used in conjunction with the stress limits; both deformation and stress limits must be satisfied. As such, the staff's conclusion on the acceptability of this criterion is limited to this application.

4.2.4.5.2 Design Evaluation

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

FSAR Section 4.2.3.5.2, "Analysis of Combined Loss-of-Coolant Accident and Seismic Loading," supplemented by ML23304A446, describes the fuel assembly structural design evaluation for external loads associated with combined LOCA and SSE. TR-117605-P, Revision 0 gives further details regarding the design evaluation. Section 4.3.5, "Fuel Assembly Structural Damage from External Forces" of TR-117605-P, Revision 0 details the model development, testing, and analytical results for the NuFuel-HTP2™ design evaluation. The design analyses cover fuel in the operating bay location. The applicant performed the design evaluations in accordance with the approved AREVA TR ANP-10337P-A, Revision 0 methodology, as modified by TR-0716-50351-P-A, Revision 1. From these evaluations, the applicant concluded that the fuel assembly meets structural integrity, control rod insertability, and coolable geometry criteria during and following a LOCA and an SSE. This includes the fuel mechanical fracturing and fuel system distortion criteria discussed in Section 4.2.4.1 of this SER. The structural design and analysis of the RFT and spent fuel cooling racks, including structural analysis of the fuel in response to external forces when fuel is in these locations, will be provided by applicants referencing the US460 SDAA as indicated by COL Items 3.8-1 and 9.1-2.

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

Even though LS-DYNA is more limited than CASAC in prescribing modal damping ratios, the first and third mode natural frequencies and first mode damping using LS-DYNA matched the applicant’s results reasonably well. The third mode damping in LS-DYNA was slightly lower than the applicant’s but this would not be expected to invalidate the confirmatory results. The comparison of bending moments induced by horizontal fuel deflection resulted in good agreement between the independent confirmatory analyses and the analyses of record. The impact force calculations resulted in larger disagreement, but both analyses demonstrated margin to the grid load limits. The staff attributes the differences to the known limitation with the LS-DYNA third mode behavior. The staff considers that the closed-form calculation of the fuel rod bending moment induced by lateral impact loads is reasonable and acceptable. The staff also finds that the justification for the assumed internal grid stiffness is appropriate.

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

The NPM-20 uses B_4C as absorption material in the control rods. The $^{10}B(n, \alpha)^{7}Li$ reaction will produce helium and the buildup of helium will cause the internal pressure to increase over time. The increase in control rod internal pressure could potentially cause the control rod cladding to deform and impact the insertability of the control rods. The staff audited the applicant’s calculation for potential control rod deformation caused by the increase in internal pressure due to the buildup of helium and finds that the control rod internal pressure increase will not cause a concern with control rod insertability as long as the B-10 loss does not exceed the limit specified in Section 6.2.6, “Control Rod Internal Pressure,” of TR-117605-P, Revision 0, supplemented by ML24346A167.

Based on the staff’s review of the material presented in TR-117605-P, Revision 0, supplemented by ML24346A165, the staff concludes that the applicant analyzed the loads, determined the strength, and presented acceptance criteria consistent with the guidance in SRP Section 4.2, Appendix A, “Evaluation of Fuel Assembly Structural Response to Externally Applied Forces.” The applicant accomplished this, in part, by using the NRC-approved methodology in ANP-10337P-A, Revision 0 for analyzing the NuScale fuel assembly response and demonstrating margin to the limits. Furthermore, the staff concludes that the applicant's analysis adequately addressed fuel mechanical fracturing, fuel system distortion, control rod insertability, and coolable geometry under seismic and LOCA loads. Therefore, the staff finds that the NuScale fuel design meets the regulatory requirements in GDC 2 and Appendix S to 10 CFR Part 50.

4.2.4.6 10 CFR 50.46 Exemption Request for M5™

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

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, that are authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security. The Commission will not consider granting an exemption unless special circumstances are present. In accordance with 10 CFR 50.12(a)(2)(ii), special circumstances are present whenever the application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule.

The underlying purpose of 10 CFR 50.46 is to establish acceptance criteria for ECCS performance. The staff's review and approval of BAW-10227P-A, Revision 1, addressed all the important mechanical and material behavior aspects of M5™ with regard to ECCS performance requirements, including (1) the applicability of 10 CFR 50.46(b) fuel acceptance criteria, (2) M5™ material properties such as fuel rod ballooning and rupture strains, and (3) steam oxidation kinetics and applicability of the Baker-Just weight gain correlation. The staff -approved BAW-10240(P)-A, Revision 0, "Incorporation of M5™ Properties in Framatome ANP Approved Methods," issued May 2004 (ML042800314 (nonproprietary), ML042800316 (proprietary)), further addresses M5™ material properties with regard to LOCA applications.

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

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

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

Paragraph I.A.5 of Appendix K to 10 CFR Part 50 states that the rates of energy release, hydrogen generation, and cladding oxidation from the metal-water reaction shall be calculated using the Baker-Just equation. Because the Baker-Just equation presumes the use of Zircaloy-clad fuel, strict application of the rule would not permit the use of the equation for the advanced zirconium-based M5™ alloy for determining acceptable fuel performance. However, the underlying intent of this portion of Appendix K is to ensure that the analysis of fuel response to LOCAs is conservatively calculated. The approved AREVA 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, the staff concludes that application of the requirements in 10 CFR 50.46 and 10 CFR Part 50, Appendix K, is not necessary for the applicant to achieve the underlying purposes of the rules. Granting the exemption request will ensure that the NuScale design achieves the underlying purpose of the rule. In addition, the staff has determined that, under 10 CFR 50.12(a), the exemption is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security. Thus, the staff approves the exemption from the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, to permit the applicant's use of the M5™ alloy fuel rod cladding in its fuel design.

4.2.4.7 Fuel System Design Change Process

In Chapter 7, "Design Change Process," of TR-117605-P, Revision 0, the applicant presented a fuel mechanical design change process by which desired fuel design changes can be evaluated and potentially would not require NRC review and approval before implementation of the new fuel design. The staff reviewed the process to determine its acceptability.

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

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

- The fuel design criteria described in TR-117605-P, Revision 0, Chapter 4, "Design Evaluation," are clearly identified and no changes to the design criteria are allowed without NRC review and approval. In addition, the design criteria must be demonstrated to be valid for the new fuel design.

- The approved methodologies described in TR-117605-P, Revision 0, Chapter 4, used to evaluate the fuel against the fuel design criteria are identified and all conditions and limitations (e.g., fuel burnup limit) to the methodologies must be met. In addition, the methodologies must be demonstrated to be valid for the new fuel design.

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

The staff also concludes that the additional aspects of the design change process in Chapter 7 of TR-117605-P, Revision 0, provide documentation, quality assurance adherence, testing, and surveillance requirements (as applicable) that ensure new fuel designs are properly tracked and validated.

Based on the staff's evaluation as noted above, the staff finds that, if properly followed, the fuel design change process as presented in Chapter 7 of TR-117605-P, Revision 0, as supplemented, meets all regulatory requirements related to fuel design.

4.2.5 Combined License Information Items

Table 4.2-1 lists the relevant COL information item and description from FSAR.

Table 4.2-1 NuScale COL Information Item for FSAR Section 4.2

COL Item No.	Description	FSAR Section
4.2-1	An applicant that references the NuScale Power Plant US460 standard design and wishes to utilize non-baseload operations will provide justification for the fuel performance codes and methods corresponding to the desired operation.	4.2

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

4.2.6 Conclusion

The staff concludes that the fuel system for the NPM-20 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 2, 10, 27, and 35; and 10 CFR 50.34. The staff notes that several of the DBEs are evaluated in the appropriate Chapter 15 section within this SER; therefore, the conclusions regarding regulatory compliance for fuel under those specific postulated accidents are presented in the respective staff SEs. The staff based its conclusion on the following:

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

- The applicant established plans for the testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. The applicant included design features that permit online fuel failure monitoring and defined a post-irradiation surveillance program to detect anomalies or confirm that the fuel has performed as expected. Future COL applicants will be responsible for implementation of these testing and inspection plans, and the staff would verify these plans at the COL stage.

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

In addition, based on its review described above regarding fuel assembly structural response to external forces analysis in TR-117605-P, Revision 0, the staff finds that the NPM-20 fuel system design meets the requirements of GDC 2 and Appendix S to 10 CFR Part 50.

4.3 Nuclear Design

4.3.1 Introduction

The staff reviewed FSAR Section 4.3, "Nuclear Design," using the guidance in SRP Section 4.3, Revision 3 "Nuclear Design," issued March 2007 (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

FSAR Section 4.3 describes the nuclear design of the NPM-20, as summarized below.

FSAR Section 4.3.1, "Design Basis," describes the NPM-20 approach to addressing the regulatory criteria in 10 CFR Part 50 and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." 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-P-A, Revision 1. Section 4.2.1, "Safety Evaluation Report," of TR-0116-20825-P-A, Revision 1 specifies that the applicability of this report is limited to fuel rods with burnups below 62 gigawatt-days per metric ton of uranium. Section 4.2.1, "Safety Evaluation Report," of TR-108553-P-A, Revision 0, confirms that Section 4.2.1, "Safety Evaluation Report," of TR-0116-20825-P-A, Revision 1 is still applicable to the rated thermal power of the NPM-20 design.

- The moderator temperature coefficient (MTC) and Doppler coefficient together provide inherent reactivity control to satisfy GDC 11, “Reactor inherent protection.”

- The power distribution and the reactor protection system are designed to ensure that specified 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 NPM-20 design uses soluble boron through the chemical and volume control system (CVCS) and control rods as the two independent means for reactivity control. The applicant defined shutdown margin (SDM) as the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition, assuming that the moderator temperature is 174 °C (345 °F) and that all CRAs are fully inserted except for the single assembly of highest reactivity worth, which is assumed to be fully withdrawn. Subcriticality assuming the highest worth CRA stuck out in the 72 hours following DBEs is evaluated following the methodology described in TR-124587-P, Revision 0, “Extended Passive Cooling and Reactivity Control Methodology,” issued January 5, 2023 (ML23005A308 (nonproprietary), ML23005A309 (proprietary)), based on criticality analysis of limiting core states identified through transient analysis.

- The NPM-20 design uses the combined capabilities of the CRAs and CVCS, in conjunction with the ECCS supplemental boron (ESB) function, to control reactivity changes under postulated accident conditions with appropriate margin for stuck rods, ensuring the capability to cool the core is maintained and meets the regulatory requirement of GDC 27**.**

- The design of the NPM-20 reactor and associated systems and the administrative controls on the CRA position provide an inherently stable core with respect to axial and radial power stability.

FSAR Section 4.3.2, “Nuclear Design Description,” describes the nuclear core design and provides the following additional details that address the design’s compliance with the design-basis:

- The NuScale core design comprises 37 fuel assemblies with 16 fuel assembly locations that contain CRAs. The 16 CRAs are divided into two shutdown groups and two regulating groups, with each group containing four CRAs. The fuel rods consist of ceramic pellets of UO_2 with up to 4.95 percent enriched U-235 with Gd_2O_3 as a burnable absorber and a zirconium-based cladding.

- The fuel cycles are nominally 18 months and equivalent to a minimum 520 effective fullpower days. The NPM is designed with a heavy reflector to improve neutron economy. The reflector is made of stainless steel, which reflects neutrons back into the core and flattens the power distribution to improve fuel performance. The reflector is located between the core periphery and the core barrel; it provides the core envelope and directs flow through the core. The reflector also includes holes to allow water to flow to prevent overheating.

- 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 reperformed.

- 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 distributed throughout the vertical height of the reactor core. The bases to TS 3.2.2 also note that the detectors are fixed in evenly spaced axial locations. The signals from the SPNDs are synthesized into three -dimensional assembly and peak rod power distributions. FSAR Chapter 7.0 provides details on the in-core power distribution monitoring system.

- The loss of CRA worth resulting from the depletion of the absorber material is negligible. A calculation of B-10 loss for a CRA lifetime of 20 EFPYs demonstrates that less than 1.43 percent of the boron in the bottom portion of the CRA is lost because of B-10 depletion.

- The maximum CRA internal pressure, including the initial helium backfill and additional pressure added by the helium produced by B-10 (n, α) reactions, meets the limit set by the CRA vendor Framatome.

The description of the analytical methods in FSAR Section 4.3.3, “Analytical Methods,” states that (1) Studsvik Scandpower Core Management Software simulation tools with Evaluated Nuclear Data File (ENDF)/B-VII cross section data are used to perform the nuclear analysis and (2) the Monte Carlo N -Particle Transport Code, Version 6.1 (MCNP6), with ENDF/B-VII.1 cross section data (Los Alamos National Laboratory Technical Report LA-CP-13-00634, “MCNP6 User's Manual,” issued May 2013) is used to perform reactor vessel fluence calculations which are described in TR-118976-P Revision 1, “Fluence Calculation Methodology and Results,” issued August 26, 2024 (ML24239A845 (nonproprietary), ML24239A846 (proprietary)). FSAR Section 4.3.2.8, “Neutron Fluence,” specifies that MCNP6 version 1.0 is used to perform vessel fluence calculations. During the regulatory audit, NRC staff noted that MCNP6, version 1.0, is also used to perform CRA boron-10 depletion calculations.

ITAAC: FSAR Part 8, Table 2.5-1 provides ITAAC number 02.05.01, which includes a design commitment that the module protection system (MPS) automatically actuates a reactor trip. The staff evaluates this ITAAC in Chapter 14 of this SER.

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 Bank Insertion Limits”
- GTS 3.1.6, “Regulating Bank 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 3.5.4, “Emergency Core Cooling Supplemental Boron”
- GTS 5.6.3, “Core Operating Limits Report (COLR)”

Technical Reports: TR-118976-P, Revision 1, is incorporated into the application by reference as noted in FSAR Table 1.6-2, as supplemented by ML24346A132.

4.3.3 Regulatory Basis

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

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

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

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

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

- GDC 25, “Protection system requirements for reactivity control malfunctions,” requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

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

- GDC 27 requires that the reactivity control systems be designed to have a combined capability of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.
- GDC 28, “Reactivity limits,” requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than the limited local yielding nor sufficiently disturb the core, its support structures, or other RPV internals to significantly impair the capability to cool the core. These postulated reactivity accidents shall consider rod ejection (unless prevented by positive means), rod dropout, 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

FSAR Section 4.3.1.3, “Power Distribution,” supplemented by ML24346A341, 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. FSAR Section 4.3.2.2, “Power Distribution,” further describes the design-basis. FSAR Section 4.3.2.2.6, “Limiting Power Distributions,” clarifies that the applicant used limiting power distributions in the steady state and transient analyses to ensure that SAFDLs are not exceeded during normal operations and AOOs. FSAR Figures 4.3-5 through 4.3-14, supplemented by ML24346A341, show power peaking factors, assembly power distributions, relative pin power distributions for selected fuel assemblies, and axial power distributions calculated with the US460 core analysis model. FSAR Figure 4.3-4,” Axial Offset Window,” shows the analytical axial offset window, which was developed to encompass axial offsets achievable during normal operation by considering depletion over various durations. The applicant stated that, for each cycle core design, a limit is imposed on the $F_{\Delta H}$, which is conservatively applied in the safety analysis. Additionally, the applicant stated that an analysis of axial power shapes that it considers possible is performed to identify the bounding axial power shapes for use in the CHF and transient analyses.

The US460 GTS establish LCOs controlling the enthalpy rise hot channel factor ($F_{\Delta H}$) and axial offset (AO). However, it does not establish an LCO that limits the local power peaking or peak linear heat generation rate (LHGR), such as an LCO on heat flux hot channel factor (F_Q). While the enthalpy rise hot channel factor ($F_{\Delta H}$) limit and axial offset (AO) window provide some constraints on the core power distribution, local power peaking and the peak LHGR still have an impact on the evaluations of fuel thermal design limits for various design basis events.

Therefore, the staff issued RAI-10269 R1, Question 4.3-28 to request that an LCO on heat flux hot channel factor (F_Q) be provided and request specific comparisons of analyses and their impacts on the MCHFR figure of merit to evaluate the sensitivity to local power peaking.

In its response to the RAI question, (ML25058A345 (nonprop), ML25058A347 (prop)) NuScale provided the requested comparisons. The comparisons requested by the staff show the effect of pronounced local peaking relative to axial power shapes used as input to subchannel analysis. The same hot pin power is used in compared cases. The results show that pronounced local peaking can reduce MCHFR compared to the normally used power shapes. For the direct comparison case where the hot pin powers are set to be equal to each other and have the same general power shape, Case 2B in Table 2 of the RAI response shows MCHFR results which can be compared to MCHFR results from Case 3B in the RAI response. The same results seen in the comparison between Case 2B and Case 3B would be expected in the limiting MCHFR Case presented in the RAI response if the same evaluation were performed with a pronounced local peaking.

Additionally, the applicant's response to RAI-10269 R1, Question 4.3-28, stated that power shapes used in subchannel analysis are developed based on unphysical axial xenon distributions. It stated these unphysical xenon distributions produce power shapes with axial offset values that exceed what NuScale considers possible during operation in order to increase the size of the axial offset window, and that this "results in a bounding treatment of operationally possible local axial power peaking values." The RAI response shows that the nominal cases result in a maximum nodal LHGR value of 7.18 kW/ft, and the cases with artificial xenon distributions result in a maximum nodal LHGR of 8.36 kW/ft. While the expanded AO window and unphysical axial xenon distributions do provide some conservatism with respect to local peaking, they do not consider all possible axial power shapes that may occur during operation. Additionally, the AO window can be reduced if margins are challenged. Axial power shapes and local peaking can be impacted by effects such as, but not limited to, assembly cross-flow and rodded depletion.

In the RAI response, NuScale compared the NPM-20 reactor with operating reactors and stated that the nominal average linear heat rate in the NPM-20 design is 3.9 kW/ft, with an approximate nominal peak value of 7.5 kW/ft. In operating reactors, the nominal average linear heat rate is approximately 5-7 kW/ft, and the nominal peak linear heat rate may exceed 14 kW/ft. Additionally, NuScale states that the NPM-20 core is smaller, axially and radially, than a typical PWR and the assemblies have an active fuel height of 6.5 feet, leading to a height-to-diameter ratio of approximately 1. NRC staff understands that a smaller core may lead to a more tightly coupled power distribution and a local power peaking that is more likely to propagate to the rest of the core during certain types of spatial power oscillations and may help reduce local power peaking excursions.

In its response to the RAI, the applicant further states that the purpose of the $F_{\Delta H}$ limiting condition for operation is to set limits on core power density to ensure fuel design criteria are met and accident analysis assumptions remain valid. The staff disagrees with this determination because the definition of $F_{\Delta H}$, as stated in the response to the RAI, is the ratio of the maximum integrated rod power to the average rod power of the core and therefore it cannot be used as a reliable parameter for detecting or limiting local power peaking or maximum core power density.

Although NRC staff does not consider the AO window (with conservative xenon treatment) and the $F_{\Delta H}$ limit to be sufficient to prevent the local peaking assumed in Chapter 15 analyses from being exceeded, and notes that local power peaking does impact figures of merit, the staff finds that GTS without an F_Q LCO is acceptable for the NPM-20 design because it has (1) low power density compared to operating reactors, (2) smaller, more tightly coupled core neutronic characteristics and power distribution, and (3) mitigating assumptions that conservatively increase local peaking in the treatment of the AO window. The staff finds that the combination of these factors provides reasonable assurance that unexpected changes in local peaking will not be significantly higher than those assumed in the analysis.

The axial power distribution will be affected by axial blankets that typically use lower U-235 enrichment. FSAR Section 4.3.2.1, states that fuel may include axial enrichment variation, but the section clarifies that the NPM-20 design that is analyzed in FSAR Chapter 4 does not use fuel with variable axial enrichment. The staff finds this clarification consistent with the nuclear analyses for power distribution and therefore acceptable. Future core designs will be analyzed under the appropriate change control processes using methodologies in NRC-approved TRs that are applicable to the NPM-20 design. These change processes require licensees or vendors, in part, to ensure that applicability of nuclear analysis codes to cycle-specific core designs is supported by the evaluation model assessment.

The staff audited the reactor design and associated core analyses, modeling of the reactor design (including the fuel-loading patterns for the equilibrium core), and the reflector design, which has a significant impact on power distribution. In a letter dated December 11, 2024 (ML24346A169), the applicant stated that the US460 core analysis model incorporates the NPM-20 reflector design. The NRC staff confirmed through an audit that the applicant based its lattice physics calculations on the NPM-20 reflector as designed.

The NRC staff audited the calculations supporting FSAR Chapter 4 and examined the AO window (ML24211A089). During this audit, the NRC staff observed that the applicant used its nuclear design methodology to perform evaluations based on possible operating conditions (e.g., power, time in cycle, CRA insertion, perturbed conditions), which show that the AO window is maintained within the bounds assumed in the safety analysis.

TR-0915-17564-P-A, Revision 2, “Subchannel Analysis Methodology,” issued February 2019 (ML19067A256 (nonproprietary), ML19067A257 (proprietary)), which the NRC staff has reviewed and approved (ML18338A031), describes in detail the method for applying the power distribution in the safety analysis. In particular, the NRC staff’s SE for TR-0915-17564-P-A, Revision 2, found the applicant’s approach for using bounding radial and axial power distributions acceptable. In addition, the applicant developed TR-108601-P-A, Revision 4, “Statistical Subchannel Analysis Methodology Supplement 1 to TR-0915-17564-P-A, Revision 2, Subchannel Analysis Methodology,” issued November 2023 (ML24106A160 (nonproprietary), ML24106A161 (proprietary)). The staff documented its evaluation of TR-108601-P-A, Revision 4 in an SE (ML24058A019). 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.

The applicant provided the core-wise and typical assembly pin-wise power distributions for an equilibrium core in FSAR Figure 4.3-8 to Figure 4.3-12. The staff audited (ML24211A089) the applicant’s calculation for power distributions in both the radial and axial directions. The staff also performed confirmatory calculations for the power distributions. The results of the staff’s confirmatory calculations show good agreement with the applicant’s power distributions. Based on the information discussed in this section and the analytical methods discussed in Section 4.3.4.7 of this SER, 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

FSAR Section 4.3.1.2, "Negative Reactivity Feedback," 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. The combination of the Doppler coefficient and the MTC should ensure that the overall reactivity coefficient associated with an increase in core power is negative. In FSAR Section 4.3.2.3, "Reactivity Coefficients," the applicant gives more detail on the calculations of the Doppler coefficient and the MTC for the NPM-20 design. Because the NPM-20 design uses natural circulation in the primary side for removing the heat generated by the reactor, flow rate and moderator density will fluctuate and cause some reactivity feedback to the core. The applicant stated that the flow reactivity is incorporated in determining the MTC. FSAR Figure 4.3-15, "Moderator Temperature Coefficient of Reactivity from Zero to Full Power" and Table 4.3-2, "Nuclear Design Parameters (for Equilibrium Cycle)" give values for the MTC. The NRC staff's confirmatory analyses that predicted values for the MTC that show good agreement with the applicant's results.

Additionally, the applicant presented values for the power coefficient in FSAR Figure 4.3-17, "Maximum and Minimum Power Coefficient," and boron reactivity worth in FSAR Figure 4.3-18, "Differential Boron Worth Coefficient." The applicant's analysis shows that the power coefficient is negative for all power levels. The applicant obtained the results using the analytical methods discussed in Section 4.3.4.7 of this SER. Additionally, as discussed in Section 3.5.2 of the NRC staff's SE for TR-0616-48793-P-A, Revision 1, "Nuclear Analysis Codes and Methods Qualification," issued December 14, 2018 (ML18348B036 (nonproprietary), ML18348B037 (proprietary)), 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 in FSAR Section 4.3.1.2 and in Section 3.5.2 of the staff's SE for TR-0616-48793-P-A, Revision 1, 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. Section 4.3.4.7 of this SER gives the staff's evaluation of the analytical methods used by the applicant for calculations of these various important reactivity feedback coefficients.

4.3.4.3 Reactivity Control

FSAR Section 4.3.1.5, "Shutdown Margin and Subcriticality During Long-Shutdown,"," states that the NuScale design uses two independent means for reactivity control: (1) CRAs, and (2) soluble boron. FSAR Section 3.1.3.7, "Criterion 26-Reactivity Control System Redundancy and Capability," clarifies that the CVCS in conjunction with the boron addition system fulfills the requirement for the second reactivity control system specified in GDC 26. 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 the CVCS uses forced flow to deliver soluble boron.

FSAR Section 4.3.1.5, supplemented by ML24346A174, defines SDM as the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition, assuming all CRAs are fully inserted except for the single assembly of highest reactivity worth, which is assumed to be fully withdrawn. GTS 1.1, "Definitions", specifies that SDM assumes moderator temperature is 174 °C (345 °F). The staff finds the temperature threshold of 174 °C (345 °F) for defining SDM to be consistent with the safe-shutdown requirements for passive designs specified in SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," dated March 28, 1994 (ML003708068). FSAR 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 (ML24211A089) the calculations supporting FSAR Chapter 4, which included a SDM calculation**.**

During this audit, the NRC staff observed that (1) the applicant performed the calculation consistent with the definition of SDM and (2) the results of the calculation showed that the equilibrium cycle produced margin with respect to the SDM acceptance criterion. The staff's confirmatory analyses of the SDM calculations support the conservatism of the values presented in the FSAR. The NRC staff recognizes that SDM is verified in accordance with GTS 3.1.1, GTS 3.1.5, and GTS 3.1.6, in MODE 1 with the effective multiplication factor, k-effective, $k_{eff} > 1$. Based on the information described in this paragraph, the NRC staff finds that the control rods are capable of achieving subcriticality 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. In Chapter 15 of this SER, the staff 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).

Some design features of the NPM-20, such as the use of augmented quality direct current power to hold reactor vent valves (RVVs) shut, mean that the reactor coolant temperature may fall below the reactor coolant temperature assumed in the SDM definition during some AOOs. NuScale addressed these scenarios through application of the extended passive cooling methodology, which staff evaluates in Section 15.0.5 of this SER. This methodology does not credit the use of CVCS to adjust boron concentration following accident initiation, and application of the methodology demonstrates that CRAs in conjunction with the ESB are able to maintain subcriticality in the 72-hour period following the initiation of DBEs, assuming the highest worth CRA is stuck out. The quantity and form of boron in the ESB dissolvers is maintained in accordance with GTS 3.5.4. This specification gives assurance that, despite potential operational occurrences such as the introduction of steam into containment through a reactor safety valves (RSV) actuation, the performance of the ESB dissolvers will be within the assumptions of the extended passive cooling analysis. Section 6.3 of this SER contains further staff evaluation of the ESB feature of ECCS. Based on this evaluation, the NRC staff finds that CRAs in conjunction with ESB are capable of assuring that during AOOs, with appropriate margin for stuck rods, SAFDLs are not exceeded, consistent with GDC 26.

FSAR Section 4.3.1.5 states that both the CRAs and the CVCS are capable of controlling reactivity changes resulting from planned normal operation. Additionally, FSAR 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). FSAR Section 4.3.1.4, "Maximum Controlled Reactivity Insertion," clarifies that the maximum controlled reactivity addition rate is limited such that the SAFDLs are not violated during normal operation, AOOs, or postulated accidents. (In

Section 15.4 of this SER, the staff evaluates reactivity and power distribution anomalies from both inadvertent CRA and CVCS operation. Based on the information described in these sections of the FSAR, the NRC staff finds that the CVCS is capable of reliably controlling the rate of reactivity changes resulting from planned normal power changes to ensure 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.

The applicant further stated that, when transitioning to cold conditions, the CVCS provides the necessary boron concentration to ensure the reactor remains subcritical. FSAR Section 9.3.4.2.1, "Chemical and Volume Control System," states that the boron addition system is managed to maintain a sufficient quantity of boron to ensure the ability to support a shutdown of all NPMs to an RCS pressure and temperature of 250 pounds per square inch absolute (psia) and 100 °F. The calculation includes one NPM with the highest reactivity worth CRA held in the fully withdrawn position and assumes all the remaining NPMs have all rods inserted and do not require an increase in boron concentration. Although the evaluation was performed to ensure the CVCS has the capability to support multiple units on the same site, this analysis indicates that the CVCS has sufficient boration capacity to fulfill the GDC 26 requirement to hold a reactor core subcritical under cold shutdown conditions. Based on the information described in this paragraph, the NRC staff finds that during normal operation when alternating current and direct current power are available, the CVCS is capable of holding the reactor subcritical under cold conditions. FSAR Section 9.3.4, "Chemical and Volume Control System," provides more detailed information on the design and operation of the CVCS system and requirements for reactivity insertion capacity in the context of the US460 plant. Chapter 9 of this SER documents the staff's evaluation of the CVCS performance If the CVCS is unavailable, automatic actuation of the ECCS provides additional soluble boron through the ESB function to ensure the reactor remains subcritical for at least 72 hours following the event. Section 6.3.4 of this SER provides the staff's evaluation of the ESB system.

Based on the above information and the following discussion concerning diverse flowpaths in the RCS, the NRC staff finds that the reactivity control systems in the NPM-20 design are consistent with GDC 26.

FSAR Sections 4.3.1 and 4.3.1.5, as supplemented by ML24346A174, discuss SDM subcriticality during a long-term cooldown following a postulated accident. The insertion of CRAs together with passive boron addition from the ESB feature of ECCS provide the safety-related means to shut down the reactor and maintain it in a shutdown condition. During power operation, the CVCS is used to adjust soluble boron concentration to maintain CRAs within PDILs to preserve the capability of the CRAs to rapidly reduce power and protect fuel design limits upon a reactor trip. The ability of these systems to achieve and maintain subcriticality during and following a postulated accident is evaluated through application of the extended passive cooling methodology as reviewed in Section 15.0.5 of this SER. Based on the results of this analysis, the NRC staff finds that reactivity control systems in the NPM-20 are designed to reliably control reactivity changes such that under postulated accident conditions and with appropriate margin for stuck rods the ability to bring the reactor subcritical and cool the core is maintained consistent with GDC 27.

The RCS includes design features to ensure that the soluble boron delivered by the CVCS or ESB remains well mixed following reactor scram. These features support the capability of these reactivity control systems and are needed for the NRC staff to reach the above findings relative to GDC 26 and GDC 27. FSAR Section 4.3.1.5 states that if the CVCS is unavailable, automatic actuation of the ECCS provides additional boron concentration through the ESB function to ensure the reactor remains subcritical for at least 72 hours following the event. Section 9.3.4 of the FSAR discussed in detail the operation of the CVCS, and Section 6.3 of the FSAR provides a detailed discussion of the ECCS, including the ESB feature. Additionally, FSAR Section 4.3.1.5 states that diverse flowpaths in the upper and lower riser barrel are provided to ensure that RCS boron remains mixed during extended decay heat removal system (DHRS) or ECCS operation. FSAR Section 15.0.5, “Extended Passive Cooling for Decay and Residual Heat Removal,” discusses these scenarios in detail. For some non-LOCA scenarios, condensation of steam could reduce the downcomer boron concentration after the DHRS cools the RCS sufficiently to cause the RCS level to drop below the top of the riser. ECCS actuation may not occur in these scenarios if the 8-hour ECCS timer following reactor scram is blocked. Reduced boron concentration in the downcomer could, under certain conditions, cause a positive reactivity insertion when natural circulation is restored. As addressed as part of the methodology in TR-124587-P, Revision 0, Section 3.2.2.1, “Upper Riser Flowpaths,” and described in the FSAR, the NPM design includes holes in the upper portion of the riser to promote mixing in the downcomer and mitigate a core dilution event under riser uncovery conditions. FSAR Section 15.0.5.1 also discusses that boron dilution due to condensation in the downcomer may also occur in some loss-of-coolant events once level drops below the top of the riser. It states that flowpaths in the upper riser promote mixing in these scenarios to preclude unacceptable reactivity insertion when the ECCS is actuated. It also states that in some LOCAs or inadvertent ECCS operation events, holes in the lower riser provide a flowpath for boron mixing during extended ECCS cooling events. The staff evaluates the effectiveness of the riser hole to prevent unacceptable levels of downcomer dilution in Section 15.0.5 of this SER.

Postevent recovery actions with respect to boron distribution, from both LOCA and non-LOCA events, are important to ensure that a core dilution event is prevented. FSAR Section 15.0.4, “Safe, Stabilized Condition”, supplemented by ML24346A257, states that the fluid boron concentration and boron distribution in the NPM are important when exiting passive ECCS and DHRS cooling modes and need to be accounted for to ensure SDM limits are preserved. The staff notes that these postevent recovery actions are outside the scope of the SDAA review but are important to capture in the development of operating procedures. The applicant included COL Item 13.5-3 in FSAR, Section 13.5.2, “Operating and Maintenance Procedures,” for development of operating procedures at a future licensing stage.

4.3.4.4 Control Rod Patterns and Reactivity Worths

FSAR Section 4.3.1.4, states that the NuScale design limits the worth of the CRAs, CRA insertion depth, and maximum CRA withdrawal rate. FSAR Section 4.3.2.1 states that the 16 CRAs are divided into two shutdown groups and two regulating groups and that each group contains four CRAs. FSAR Section 4.3.2.1, further clarifies that the shutdown groups are fully withdrawn during operation and that the PDILs restrict the amount by which the regulating groups can be inserted at power. FSAR Section 4.3.2.4.2, “Control Rod Assemblies,” further states 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. FSAR Figure 4.3-1, “Power Dependent Insertion Limits,” shows the PDILs. FSAR Figure 4.3-14, “Control Rod and Incore Instrument Locations,” shows the CRA locations and group structures.

The NRC staff audited (ML24211A089) the calculations supporting FSAR Chapter 4, which included the calculation for control rod worths, the axial offset window, and the process used to set the PDILs. The NRC staff noted during the audit 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 FSAR Chapter 15. In Section 15.4.8 of this SER, the staff evaluates the rod ejection accident, which can limit CRA reactivity insertion. Based on the description in FSAR Section 4.3.2.1, supplemented by ML24346A175, and the analyses that set the PDILs, the NRC staff finds that the applicant established adequate PDILs for use in accident and transient analyses. Additionally, the NRC staff has determined that GTS 3.1.5 and GTS 3.1.6 verify the position of the CRAs.

FSAR Figure 4.3-25, “Integral Rod Worth for Regulating Bank from Power Dependent Insertion Limits,” provides the integral bank worths for the regulating banks. The NRC staff performed confirmatory analyses as part of its review and obtained values for individual stuck rod worths at the beginning of cycle, middle of cycle, and end of cycle as well as integral and differential regulating bank worths at beginning 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 Section 4.3.4.7 of this SER, which the NRC staff has previously reviewed and approved (ML18234A295). The staff audited the applicant’s calculation for the integral and differential control rod worths (ML24211A089) and finds that the calculations used conservative assumptions with respect to the neutron flux distribution and uncertainties associated with CRA worth. Additionally, the NRC staff audited the calculations supporting CRA depletion analysis (see ML24346A167 (nonproprietary), ML24346A168 (proprietary)). During the audit, the NRC staff noted that the total loss of B-10 in each control rod (i.e., in each of the 24 individual control rods that are part of the CRA) is restricted, as explained by NuScale in a letter dated December 11, 2024 (ML24346A167 (nonproprietary), ML24346A168 (proprietary)), to a percent of initial B-10 over the CRA lifetime in order to meet the control rod internal pressure limit set by Framatome. This limit also provides an assurance that the loss of control rod worth is insignificant over the designed lifetime of the CRA. The staff audited the applicant’s engineering calculation for CRA B-10 loss and the resultant CRA internal pressure increase caused by the $^{10}B(n, \alpha)^{7}Li$ reaction and determined that the result is reasonable in its engineering judgement for baseload operation. In addition, FSAR Section 4.3.2.2.8, “Testing,” discusses startup testing. It states that control rod worth measurements confirm the capability of the core to be shut down, and the shutdown requirement is confirmed by measuring the power defect (the reactivity difference between zero power and full power). This required startup test on control rod worth gives further assurance that significant control rod worth loss will be detected. On these bases, the staff finds that there is a reasonable assurance that the CRA internal pressure will meet the limit set by Framatome and meet the regulatory requirements of GDCs 26 and 27 as discussed in Section 4.3.4.4 of this SER.

FSAR Section 4.3.1.4, supplemented by ML24346A177, states that the control rod design places limits on the worth of the CRAs, CRA insertion depth, and maximum CRA withdrawal rate. The reactivity addition rate limit is used to ensure that the SAFDLs are not violated during normal operation, AOOs, or postulated accidents. FSAR Section 15.4.1.2, Section 15.4.2.2, and Section 15.4.3.2, all titled “Sequence of Events and Systems Operation,” state that the expected normal travel rate of the CRAs is 15.24 centimeters (6 inches) per minute and the assumed maximum allowed withdrawal rate is 38 centimeters (15 inches) per minute. The NRC staff documented its review of transient analysis assumptions in Section 15.4 of this SER. FSAR Table 14.2-73, Test # 73 – “Control Rod Drive System—Manual Operation, Rod Speed, and Rod Position Indication,” states that Test No. 73 will verify that the rod insertion and withdrawal speeds are within design limits. The NRC staff finds this design information acceptable because it provides the maximum design -basis CRA withdrawal rate consistent with the value used in FSAR Chapter 15 transient analyses. Additionally, FSAR Table 15.0-1, “Design-Basis Events,” categorizes the uncontrolled CRA withdrawal events resulting from a malfunction of the reactivity control system as AOOs. The NRC staff recognizes that, as required by GDC 10, AOO acceptance criteria prohibit the violation of SAFDLs.

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

4.3.4.5 Criticality during Refueling

FSAR Section 4.3.2.6, "Criticality of the Reactor During Refueling," states that maintaining an effective neutron multiplication factor of 0.95 or less at all times prevents criticality during refueling. Refueling is performed with CRAs inserted in the fuel assemblies, however, 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 accounting for the CRA in a fuel assembly (for the case in which a fuel assembly that contains a CRA is being moved) and an additional 0.05 in k_{eff} margin from criticality. Additionally, GTS 3.5.3 establishes limits on the bulk ultimate heat sink boron concentration to ensure that SDM is maintained during refueling. Furthermore, GTS 5.6.3 requires that the bulk average boron concentration limit is established using the methods described in FSAR 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

FSAR Section 4.3.2.7, "Stability," states that the design-basis for the reactor and associated systems is to provide an inherently stable core with respect to axial and radial power distributions. FSAR Section 4.3.2.7, as supplemented by ML24346A176, evaluates the potential xenon--induced power distribution oscillation. The applicant calculated the potential radial and axial oscillations using the SIMULATE5 code (see Section 4.3.4.7 of this SER) at various times (beginning and end of cycle) in the cycle and at various power levels, at 0, 25, 75 and 100 -percent power. In the calculation models, the applicant induced xenon oscillations through the insertion of control rods. FSAR Table 4.3-6, "Limiting Cycle-Specific Xenon Stability Indices" gives the results of the applicant's analyses, which show that the reactor was stable over this configuration. Additionally, the staff audited the calculations (ML24211A089) supporting FSAR Chapter 4 and the xenon stability analyses. During this audit, the NRC staff observed that the applicant performed the xenon stability analyses using the regulating CRAs in one quadrant to induce xenon oscillations at different power levels and cycle times (beginning and end of cycle) and that the analyses produced results that are consistent with the information presented in FSAR Table 4.3-6. Based on the information discussed in this section, the staff finds that the NuScale design is inherently stable with respect to axial and radial power stability because (1) the applicant performed conservative stability analyses using an approved analytical method, and (2) the analyses showed that the reactor stabilizes for all perturbations. The staff evaluates additional stability considerations in Section 4.4.4.8 of this SER.

4.3.4.7 Analytical Methods

FSAR Section 4.3.3 discusses the analytical methods used by the applicant to analyze the nuclear design. The applicant used the Studsvik Scandpower Core Management Software simulation tools, including CASMO5, CMSLINK5, SIMULATE5, and S3K, to perform steady-state and transient neutronic analysis. TR-0716-50350-P, Revision 3, "Rod Ejection

Accident Methodology," dated October 20, 2023 (ML23293A292), describes the applicant's use of these methods in detail. The NRC staff has reviewed and approved TR-0616-48793-P-A, Revision 1, for the design and analysis of the NuScale reactor core. The staff has reviewed and approved TR-0716-50350-P-A, subject to the limitations and conditions in the SER (ML23310A166).

Additionally, the applicant stated that MCNP6, Version 1.0, with cross-sections based on ENDF/B-VII.1, is used to perform vessel fluence calculations. The staff recognizes that MCNP is frequently used in the analysis of neutron particle transport and has been previously approved for use in performing vessel fluence analyses. Based on the previous approval of MCNP6 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

FSAR Section 4.3.2.8 discusses the RPV fluence calculations. The licensing technical report TR-118976-P Revision 1 gives the details of the fluence calculations. The NRC staff compared the fluence calculation in TR-118976-P, Revision 1, against the guidance in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," issued March 2001 (ML010890301), and determined that the applicant performed the analysis consistent with RG 1.190.

In TR-118976-P, Revision 1, the applicant calculated the neutron source distribution using the nuclear design codes CASMO5 and SIMULATE5 and the NRC-approved core design methodology described in TR-0616-48793, Revision 1. The applicant used the MCNP 6.1 code to calculate the neutron fluence at the various radial, axial and azimuthal layers of RPV and CNV. The ENDF/B-VII.1 cross section library is used in all of these calculations.

The applicant added the uncertainties associated with the neutron fluence calculations (supplemented by ML24215A098). The applicant conservatively did not apply the negative estimated bias to the final calculation result**.** The applicant presents the results in Table 5-1 of TR-118976-P, Revision 1. With 60 years of operation at a 95 percent capacity factor, the estimated maximum neutron fluence with energy with greater than 1 mega-electron volt (MeV) at the inner surface of the RPV is provided in TR-118976-P, Revision 1. The applicant also provided the maximum fluence at the top of the lower RPV flange to support a partial exemption from the requirements of 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for light-water nuclear power reactors for normal operation," and 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events." The maximum fluence with $E \geq 1$ MeV at the top of the lower RPV flange is less than $1X10^{17}$ n/cm^2 such that requirements of 10 CFR 50.60 and Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50 do not apply and the upper RPV shell material is not screened for pressurized thermal shock under the requirements of 10 CFR 50.61. Section 5.3 of this SER contains further discussion of the staff's evaluation of these exemptions.

The staff reviewed the calculations for the neutron source term distribution in the reactor, the core, and the RPV geometries and finds that the applicant has appropriately modeled the core, baffle plate, reactor reflector, and RPV for the neutron fluence. The staff also audited (ML24211A089) the MCNP6 output files. During the audit, the applicant presented fluence calculations that passed the 10 statistical checks performed by MCNP6. The staff finds that relative errors of the mean values are consistent with those represented in the applicant's uncertainty analysis. In a letter dated August 2, 2024 (ML24215A001/ML24215A095), the applicant (1) stated that the convergence of the fluence calculations is assured by examining the global convergence of the mesh tallies and (2) presented qualitative evaluations of plotted results. The staff determined that the calculated neutron fluence on the RPV and the flange of the upper reactor closure head are acceptable. The staff evaluates RPV neutron embrittlement in Section 4.5 of this SER.

4.3.4.9 Technical Specifications

The NRC staff reviewed the applicable TS identified in Section 4.3.2 of this SER 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 core operating limits report (COLR) for the following:

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

The regulation at 10 CFR 50.36(c)(2)(ii)(B) 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 1.1and is specified in GTS 5.6.3. The NRC staff finds this acceptable because GTS 5.6.3.b requires each LCO that references the COLR to use NRC-approved methods when establishing its limit.

The staff reviewed the scope of applicable GTS identified for the nuclear design and noted that FSAR Section 4.3.2.2.1, “Definitions,” states that maximizing the heat flux hot channel factor (F_Q) through use of F_Z and $F_{\Delta H}$ ensures that the SAFDLs are not exceeded. The maximum power peaking is controlled by the radial power peaking and axial power shape. To achieve this goal, the application introduced a parameter named axial offset (AO) and AO window. The AO is defined as the ratio of the total core power, P, and the difference between power at the top half (PT) of the bottom half (PB), i.e., AO = (PT-PB)/P. Section 4.3.4.1 of this SER contains the staff's evaluation on the need for an LCO for the heat flux hot channel factor. The staff concluded that the applicant identified an appropriate scope of GTS for the nuclear design, and therefore the requirements of 10 CFR 50.36(c)(2)(ii)(B) are met.

4.3.4.10 Testing and Verification

FSAR Section 4.3.2.2.7, "Verification of Power Distribution Analysis," discusses the benchmarking process used to develop nuclear reliability factors that are used to account for uncertainties in reactor physics parameters and power distributions. TR-0616-48793-P-A, Revision 1 details the methodology for development and update of the nuclear reliability factor that the NRC staff has reviewed and approved (ML18234A295).

4.3.4.11 In-Core Neutron Flux and Temperature Monitoring

FSAR Section 4.3.2.2.9, "Monitoring," discusses the ICIS. In Section 3.5.3.7 of the SER for TR-0616-48793-P-A, Revision 1, 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. The staff notes that FSAR Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position," supplemented by ML24346A289, requires the neutron monitoring system to reconstruct assembly power with a maximum uncertainty of 20 percent of the calculated normalized power. The NRC staff expects that this level of detector performance is achievable based on comparison to individual sources of uncertainty available in the open literature. Based on its review of TR-0616-48793-P-A, Revision 1, the NRC staff finds the ICIS design acceptable, because the nuclear design quantifies and accommodates uncertainties associated with ICIS measurements. Based on its review of the information in the FSAR, the staff determined with a reasonable assurance that the in-core neutron flux and temperature monitoring systems will perform their designed functions adequately and meet the regulatory requirements of GDC 13. Chapter 7 of this SER documents the review of the ICIS system.

4.3.5 Combined License Information Items

No COL information items are associated with FSAR 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 Section 4.3.4 of this SER, the NRC staff concludes the following:

- The nuclear design for the NPM-20 satisfies GDC 10 for the following reasons:
  - The applicant used an approved analytical method to calculate the power distributions, reactivity coefficients, and SDM (see Section 4.3.4.7 of this SER).
  - The safety analyses use bounding power distributions (see Section 4.3.4.1 of this SER).
  - 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 Section 4.3.4.3 of this SER).

- The nuclear design for the NPM-20 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 Section 4.3.4.2 of this SER).

- The ICIS for the NPM-20 satisfies GDC 13 because the nuclear design quantifies and accommodates uncertainties associated with ICIS measurements (see Section 4.3.4.11 of this SER).

- The nuclear design of the NPM-20 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 Section 4.3.4.4 of this SER).

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

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

  - The CRAs with the ESB 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 Section 4.3.4.3 of this SER).

  - 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 Section 4.3.4.3 of this SER).

  - The CVCS is capable of holding the reactor subcritical under cold conditions (see Section 4.3.4.3 of this SER).

- The nuclear design of the NPM-20 satisfies GDC 27 because, under postulated accident conditions and with appropriate margin for a stuck rod, the capability to cool the core is maintained, and with the highest worth control rod withdrawn (stuck out) and all other control rods inserted, the reactor remains subcritical under cold conditions.

- The nuclear design of the NPM-20 satisfies GDC 28 because appropriate limits are established for the potential amount and rate of reactivity increase.

4.4 Thermal-Hydraulic Design

4.4.1 Introduction

The staff reviewed FSAR 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 (ML15355A468) (DSRS). The objective of the staff’s review was to establish reasonable assurance that (1) the applicant used acceptable analytical methods to conduct the thermal-hydraulic design, (2) the design provides acceptable margins of safety from conditions that would lead to fuel damage during normal operation and AOOs, and (3) the design is not susceptible to thermal-hydraulic instability.

4.4.2 Summary of Application

The thermal-hydraulic design-basis in FSAR Section 4.4.1, “Design Bases,” supplemented by ML24346A181, 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, NSPN-1, NSP4, and the Extended Hench-Levy, are used to ensure, with a 95-percent probability at a 95 percent confidence level, that CHF does not occur during normal operation and AOOs.

- The fuel melting temperature is not exceeded in any part of the core during normal operation and AOOs.

- The design-basis core bypass flow of 7.5 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.

FSAR Section 4.4.2, “Thermal and Hydraulic Design of the Reactor Core,” describes the thermal-hydraulic design of the reactor core and provides the following details:

- the CHF and linear heat generation rate
- the core flow distribution, core pressure drops, and hydraulic loads
- correlations and physical data
- thermal effects of operational transients
- uncertainties in estimates and flux tilt considerations

FSAR Section 4.4.3, “Thermal and Hydraulic Design of the Reactor Coolant System,” describes the thermal-hydraulic design of the RCS and provides details on core bypass flow, operating restrictions, thermal-margin limits, and power maneuvering characteristics.

FSAR Section 4.4.4, “Evaluation,” describes the thermal-hydraulic evaluation and includes information on analytical models and inputs.

FSAR Section 4.4.5, “Testing and Verification,” briefly discusses testing and verification.

FSAR Section 4.4.6, “Instrumentation Requirements,” states that temperature is continuously monitored at the inlet and outlet of the 12 fuel assemblies identified in FSAR Figure 4.3-14, as in-core instrumentation locations. Additionally, FSAR Section 4.4.6, explains why the design does not provide a loose parts monitoring system (LPMS) for the NPM.

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

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

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

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

Technical Reports: TR-169856-P, Revision 0, "NuScale US460 Statistical Subchannel Critical Heat Flux Analysis Probabilistic Uncertainties," issued July 31, 2024 (ML24213A316 (nonproprietary), ML24213A317 (proprietary)) is incorporated into the application by reference as noted in FSAR Table 1.6-2, supplemented by ML24346A132.

4.4.3 Regulatory Basis

The following NRC regulations give 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.
- The regulation at 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.

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 ensure 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

FSAR Section 4.4.1.1, "Critical Heat Flux," supplemented by ML24346A181, states that the design-basis for the thermal-hydraulic design of the NPM is to have a NuScale--specific CHF correlation to ensure, with a 95 percent probability at a 95 percent confidence level, that CHF does not occur during normal operation and abnormal operating occurrences. FSAR Section 4.4.2.1, "Critical Heat Flux," further discusses the NSP4 CHF correlation, which is used to evaluate thermal margin for normal operation, AOOs, infrequent events, and accidents, with the exception of those characterized by rapid depressurization. TR-0116-21012-P-A, Revision 1, "NuScale Power Critical Heat Flux Correlations," issued December 2018 (ML18360A632), and TR-107522-P-A, Revision 1, "Applicability Range Extension of NSP4 CHF Correlation, Supplement 1 to TR-0116-21012-P-A, Revision 1," issued April 2023 (ML23118A377), which the NRC staff has reviewed and approved, describes the NSP4 CHF correlation and its development. Additionally, FSAR Section 4.4.2.1, supplemented by

ML24346A181, discusses the NSPN-1 and the Extended Hench-Levy CHF correlations, which are used for events exhibiting a rapid depressurization**.** The applicant stated that TR-0516-49422-P, Revision 3, “Loss-of-Coolant Accident Evaluation Model,” dated January 8, 2023 (ML23008A002 (nonproprietary), ML23008A003 (proprietary)) provides the Extended Hench-Levy and NSPN-1 CHF correlation development details, correlation limit, and range of applicability.

FSAR Section 4.4.2.7, “Uncertainties in Estimates,” supplemented by ML24346A181, states that uncertainties or biases are incorporated into the subchannel methodology to provide conservatism and these uncertainties establish the CHF analysis limit for assessing thermal margin for the NSP4 CHF correlation. TR-108601-P-A Revision 4) and TR-0915 17564-P-A, Revision 2), which the NRC staff has reviewed and approved, describe the subchannel methodology for events without rapid depressurization and the methodology used to combine the penalties and biases, along with the specifics regarding the CHF correlation performance, uncertainty, and 95/95 safety limit determination.

The referenced methodology in TR-108601-P-A, Revision 4, describes the methodology used to combine the penalties in FSAR Table 4.4-2, “Subchannel Methodology Uncertainty and Bias Application,” to determine the statistical critical heat flux analysis limit (SCHFAL), which is used as the FSAR Chapter 15 statistical analysis limit of 1.43 (critical heat flux ratio (CHFR) limit) for events without rapid depressurization. The conditions used in the TR-108601-P-A, Revision 4, methodology to create the SCHFAL specifically for the US460 Chapter 15 analysis are provided as a range of applicability for the SCHFAL in FSAR Table 4.4-8, supplemented by ML24346A181. Additionally, FSAR Section 4.4.2.7.3, “Uncertainties in Physical Data Inputs,” provides the penalties and their bases used to set the CHFR limits for the NSP4 CHF correlation. FSAR Section 4.4.2.7.3, states that the minimum CHFR design limit includes a heat flux engineering uncertainty factor and a rod bow penalty that are based on the subchannel analysis methodology. The staff conducted an audit as part of the review, which included some of the CHFR penalties that are applicable to the NPM-20 (ML24211A089). During this audit, the staff observed that the calculated values for the heat flux engineering uncertainty factor and rod bow penalty were adequately used to determine the minimum CHFR limits in the FSAR.

TR-169856-P, Revision 0, provides the values and distributions for the uncertainties and penalties used to calculate the statistical critical heat flux analysis limit using the methodology approved in TR-108601-P-A, Revision 4. Most of the uncertainties and penalties in TR-169856-P, Revision 0, are treated deterministically and are adequate because the treatment is conservative. The enthalpy rise engineering, heat flux engineering, and rod bow uncertainties are determined probabilistically. These uncertainties are determined and justified using the methodology in approved TR-0915-17564-P-A, Revision 2, Sections 3.12.4, 3.12.5, and 3.12.8, respectively, and therefore are acceptable. The distributions used for the enthalpy rise engineering, heat flux engineering, and rod bow uncertainties are treated conservatively relative to how the specific uncertainty was determined and are adequate. The CHF correlation uncertainty and distribution treatment provided in TR-169856-P is consistent with the 95/95 NSP4 correlation uncertainty determined in TR-0915-17564-P-A, Revision 2, therefore the staff finds it acceptable.

Based on the information in the FSAR, supplemented by ML24346A181, and the information obtained by the NRC staff during the audit, the staff finds that the NSP4 CHF correlation and SCHFAL provide suitably conservative safety limits for use in transient and accident analyses because the FSAR gives an adequate basis for the minimum CHFR penalties and because the applicant applied adequate penalties in the calculation of the minimum CHFR design limits, using the SCHFAL, for events that do not have rapid depressurization. The NRC staff reviewed and accepted the NSPN-1 and Hench-Levy CHF correlations in TR-0516-49422-P, Revision 3, subject to the limitations and conditions in the associated SER for events that result in rapid depressurization (ML25007A192).

In addition to the uncertainties discussed in FSAR Section 4.4.2.7, the applicant considered flux tilt in FSAR Section 4.4.2.8, “Flux Tilt Considerations.” FSAR Section 4.4.2.8 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.

FSAR Section 4.4.2.8, supplemented by ML24346A176, states that the radial tilt was determined as part of the xenon transients, as discussed in FSAR Section 4.3.2. The NRC staff reviewed the applicant’s evaluation of xenon transients in Section 4.3.4.6 of this SER and found it acceptable. Accordingly, the NRC staff finds the applicant’s treatment of flux tilt acceptable because it is based on an acceptable xenon transient methodology.

FSAR Section 4.4.2.6, “Thermal Effects of Operational Transients,” states that a thermal -margin trip (e.g., the overtemperature ΔT (OTΔT) trip in typical Westinghouse Electric Corporation designs or the DNBR trip in typical Combustion Engineering designs) is not necessary to mitigate AOOs for the NPM. The NRC staff reviewed FSAR Table 15.0-7, “Analytical Limits and Time Delays,” and found that a thermal -margin trip is not credited to mitigate DBEs. Chapter 15 of this SER evaluates these events.

FSAR Sections 4.4.4.5 and 4.4.4.6 discuss the impacts and effects of crud on the CHF calculations in the core and in the pressure drop throughout the RCS. The FSAR states that accumulation of crud has a negligible impact on flow resistances through the core and crud buildup is bounded by the flow area reduction uncertainty included in the enthalpy rise engineering uncertainty. Additionally, the FSAR states that the fuel temperatures and heat transfer inputs bound the effects of crud with respect to CHF. The staff reviewed the treatment of uncertainties related to flow area reduction uncertainty included in the enthalpy rise engineering uncertainty and the treatment of the heat transfer inputs associated with the CHF calculations as described in the FSAR and in a letter dated December 11, 2024 (ML24346A131/ ML24346A185 (nonproprietary), ML24346A186 (proprietary)). Accordingly, the NRC staff finds the applicant’s treatment of the impacts of crud acceptable because it is conservatively accounted for in the thermal hydraulic analyses as described in the FSAR and ML24346A185/ ML24346A186.

4.4.4.2 Bypass Flow

FSAR Section 4.4.1.3, “Core Flow,” states that the design basis for the NPM-20 core flow is that uncertainties in core flow are considered on a 95/95 basis, and do not credit the core bypass flowpaths, including the reflector block cooling channels, guide tubes, and instrument tube bypass flowpaths. The staff audited the calculations that examined the bypass flow calculations with explicit modeling of all flowpath constituents including uncertainties, used to demonstrate the core flow design-basis (ML24211A089). During its audit, the NRC staff observed that the calculated values for the bypass flow, with explicit modeling of all flowpath constituents including uncertainties, provide margin to the core flow design basis. In addition, the staff considered the impact on core flow of the riser holes discussed in Section 4.3.4.3, “Reactivity Control,” and Section 15.0.5, “Extended Passive Cooling for Decay and Residual Heat Removal,” of this SER. Since the holes are located above the core, the staff concludes that the riser holes have an insignificant impact on the core bypass flow percentage. Based on the information provided by the applicant and the information obtained by the staff during the audit, the staff finds the thermal-hydraulic design of the reactor core for the NPM-20 provides adequate margin to the design-basis bypass flow because the applicant performed suitably conservative analyses that demonstrated margin to the design-basis bypass flow limit.

4.4.4.3 Evaluation Methods

FSAR Section 4.4.2.5, “Correlations and Physical Data,” states that non-LOCA analyses are performed using the NRELAP5 code and that, once the limiting cases for each transient are identified, the determination of the thermal -margin is performed using the VIPRE-01 subchannel methodology. Section 4.3.5 of TR-0516-49416-P, Revision 4, “Non-Loss-of-Coolant Accident Analysis Methodology,” issued January 5, 2023 (ML23005A305 (nonproprietary), ML23005A306 (proprietary)), describes the process for identifying the cases for subchannel analysis and extraction of boundary condition data. TR-0915-17564-P-A, Revision 2, which the NRC staff has reviewed and approved, as supplemented by TR-108601-P-A, Revision 4, details the application of VIPRE-01 to the NPM. The staff has reviewed and approved TR-0516-49416-P, Revision 4, subject to the limitations and conditions in the SE (ML24334A049). Additionally, FSAR Section 4.4.2.5, states that rapid depressurization analyses are performed using the NRELAP code and that determination of thermal margin is also performed using the NRELAP5 code. The staff has reviewed TR-0516-49422-P, Revision 3, subject to the limitations and conditions in the staff’s SER, which details the application of NRELAP to the NPM-20.

4.4.4.4 Technical Specifications

FSAR Section 4.4.4.5.1, “Reactor Coolant System Flow Determination,” supplemented by ML24346A185, states that the primary contributors to pressure loss in the system are the fuel assembly and SG regions and that pressure losses in these regions are confirmed by testing. The staff compared the maximum and minimum design flow values in FSAR Table 4.4-1, “Plant Reactor Design Comparison,” with the RCS flow rates assumed in the transient and accident analyses in FSAR Table 15.0-6, “Module Initial Conditions Ranges for Design Basis Event Evaluation,” and found the flow range assumed in the transient and accident analyses bounds the maximum and minimum design flow values as high and low, respectively.

Additionally, the RCS flow is surveilled during power ascension following refueling outages, in accordance with GTS 3.4.1, to confirm that the RCS loop resistance used in the thermal-hydraulic design and the FSAR Chapter 15 transient and accident analyses remains bounding. In FSAR Section 4.4.5, “Testing and Verification,” the applicant stated that “Before achieving criticality and during initial power ascension, testing is performed to confirm thermal and hydraulic design parameters, such as RCS flow rate and core peaking factors, are consistent with the analyses.” The staff also notes that GTS 3.4.1 requires RCS flow resistance to be determined to be within the limits specified in the COLR following each refueling outage. The bases for GTS 3.4.1 further describe that the flow rates assumed in the safety analyses are based on a conservative value of flow resistance through the RCS, and that the resistance must be verified to ensure that the assumptions in the safety analyses remain valid. Based on GTS 3.4.1, the NRC staff finds that operation of the NPM within the RCS flow bounds assumed in the safety analyses is ensured because the flow is confirmed following refueling outages.

The analytical limits used in the transient and accident analyses are provided in FSAR Table 15.0-7 and are verified in accordance with GTS 3.3.1 and GTS 5.5.10. In Chapter 15 of this SER, the staff evaluates the transient and accident analyses. Chapter 16 of this SER contains the staff’s evaluation of the surveillance requirements associated with GTS 3.3.1 and GTS 5.5.10.

4.4.4.5 Loose Parts Monitoring

FSAR Section 4.4.6.2, “Module Protection System,” states that the NPM does not include a 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, or being generated in, the RPV, (2) the NPM uses corrosion-resistant materials and has a flow-induced vibration program that further minimizes the potential for loose parts being generated in the RPV, (3) a foreign materials exclusion program minimizes the potential for loose parts entering the RPV, (4) underwater vessel inspections during outages verify that there are no loose parts in the RPV, and (5) the NuScale fuel assembly has a mesh filter at the bottom of each fuel assembly. The NRC staff has previously reviewed and approved a regulatory relaxation that eliminated the requirement of the LPMS in operating boiling-water reactors (ML010310355). The NRC staff approved the elimination of the LPMS 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, and (2) the safety benefits of the LPMS were not commensurate with the cost of maintenance and the associated radiation exposure for plant personnel.

The staff compared the primary system components and fluid velocities of the reactor designs that were approved for elimination of the LPMS against the NPM-20 conditions. The staff determined that the NPM has lower flow rates and a relatively simplified design compared to the subject reactors. The CVCS is the only system that provides fluid flow directly into the RPV in the NPM design (i.e., the NPM design does not have forced circulation). Accordingly, the staff finds that the NPM is no more susceptible to issues associated with loose parts than the reactors for which the LPMS requirement has been eliminated. Based on the information in FSAR Section 4.4.6.2, the prior NRC staff approval for the elimination of the LPMS requirement, and the staff’s comparison of the NPM to operating reactors, the staff finds that the absence of an LPMS for the NPM is acceptable.

4.4.4.6 Reactor Coolant System Flow Monitoring

FSAR Section 4.4.5, states that initial testing is performed in accordance with the plant test program which is included in FSAR Section 14.2, “Initial Plant Test Program.” Testing is performed before achieving criticality and during initial power ascension to confirm that thermal and hydraulic design parameters including RCS flow rate and core peaking factors are consistent with the analyses. FSAR Section 4.2.4 details the fuel assembly component surveillance which is performed during refueling outages. GTS 3.3.1 states that the calorimetric heat balance is performed in accordance with Surveillance Requirement 3.3.1.2. Based on the continuous monitoring of RCS flow and on GTS 3.3.1, the staff finds the RCS flow monitoring acceptable because it is more restrictive than the 24 -hour monitoring criteria in DSRS Section 4.4.

4.4.4.7 Instrumentation

FSAR Section 4.4.6.1, “In-core Instrumentation System,” states that neutron flux measurements are used to determine a three-dimensional power distribution in the core. FSAR Section 4.3.2.2.9 shares further details, including 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 Section 18.7.4 of this SER, the staff concludes that the safety display and indication system displays parameters that indicate ICC and that visual and audible indications of containment abnormal conditions are provided to operators. Based on the description in FSAR Section 4.4.6.1 and the information in Section 18.7.4 of this SER, the staff finds that the NuScale design has adequate instrumentation that provides, in the control room, an unambiguous indication of ICC.

4.4.4.8 Stability

FSAR Section 4.4.1.4, “Stability,” states that the design-basis for the hydrodynamic stability of the NPM is that normal operation and AOOs do not lead to hydrodynamic instability. FSAR Section 4.4.7 states that the NuScale flow stability protection solution uses a regional exclusion solution, as described in TR-0516-49417-P-A, Revision 1, “Evaluation Methodology for Stability Analysis of the NuScale Power Module,” issued March 2020 (ML20086Q664 (nonproprietary), ML20086Q668 (proprietary)). FSAR Section 4.4.7, further discusses the flow stability evaluation for the NPM and states that TR-0516-49417-P-A, Revision 1, documents the evaluation methodology and that FSAR 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. Section 15.9 of this SER gives the staff’s evaluation and acceptance of the exclusion zone and the flow stability analysis. The NRC staff documented its review and approval of the flow stability protection solution and flow stability evaluation methodology in the SER for TR-0516-49417-P-A, Revision 1.

4.4.5 Combined License Information Items

FSAR Section 4.4 has no COL information items.

4.4.6 Conclusion

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

- The thermal-hydraulic design for the NPM satisfies GDC 10 for the following reasons:
  - The applicant evaluated CHF using an acceptable correlation (see Section 4.4.4.1 of this SER).
  - The applicant evaluated CHF margin during normal operation and AOOs using an acceptable evaluation model (see Section 4.4.4.3 of this SER).
  - Adequate TS are provided to ensure operation of the NPM is contained within the bounds of the safety analyses (see Section 4.4.4.4 of this SER).
  - RCS flow is continuously monitored (see Section 4.4.4.6 of this SER).
- The thermal-hydraulic design for the NPM satisfies GDC 12 because it uses an acceptable pressure temperature exclusion zone (see Section 4.4.4.8 of this SER).
- The thermal-hydraulic design of the NPM satisfies 10 CFR 50.34(f)(2)(xviii) because the design has adequate instrumentation that provides, in the control room, an unambiguous indication of ICC (see Section 4.4.4.7 of this SER).

4.5 <u>Reactor Materials</u>

4.5.1 Control Rod Drive Structural Materials

4.5.1.1 Introduction

This section of the FSAR describes the materials used in the CRDM for both the RCPB portion of the CRDM and non-pressure-boundary CRDM components.

4.5.1.2 Summary of Application

The FSAR 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 ASME Boiler and Pressure Vessel Code for Class 1 components.

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

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

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

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

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

4.5.1.3 Regulatory Basis

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

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

- GDC 14, “Reactor coolant Pressure boundary,” requires that the RCPB be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The application of GDC 14 assures that control rod drive materials are selected, fabricated, installed, and tested to provide assurance of an extremely low probability of significant degradation and, in the extreme, to minimize the potential for a gross RCPB failure that could substantially reduce the capability to contain reactor coolant inventory or to confine fission products.

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

The following guidance is used to meet the above requirements:

- RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal"
- RG 1.44, "Control of the Processing and Use of Stainless Steel"
- ASME NQA-1, 2015 Addenda

4.5.1.4 Technical Evaluation

The staff reviewed and evaluated the information included in FSAR 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.

4.5.1.4.1 Materials Specifications

The staff reviewed FSAR Section 4.5.1, to determine the suitability for service of the materials selected for CRDM structural components FSAR Section 4.5.1, gives information on the types, grades, heat treatments, and properties used for the materials of the CRDM components. FSAR Section 3.9.4.1.1, "Control Rod Drive Mechanism," states that the pressure housing consists of the single-piece pressure housing (bolted to the reactor vessel head), and the top plug assembly. The removable top plug assembly is threaded onto the top of the pressure housing to complete the RCPB. The bolted connection is part of the RCPB and also provides structural support for the CRDM in order to perform its safety function. Degradation of the bolted connection could lead to shifting of the CRDM or total release of the CRDM which would affect the safety function of the CRDM. In accordance with FSAR Section 5.3.1.7, "Reactor Vessel Fasteners" and Table 5.2-3, "Reactor Coolant Pressure Boundary Component and Support Materials Including Reactor Vessel, Attachments, and Appurtenances," threaded inserts are used in the bolted connection.

FSAR Section 3.13.2.1, supplemented by ML24313A066, includes augmented examination requirements to perform a VT-1 examination for surface defects and corrosion on threaded inserts and its seal welds whenever an ASME Class 1 component is disassembled. When defects or corrosion are detected in these areas of routinely disassembled ASME Class 1 components, the examinations will be expanded to include a VT-1 examination of the threaded inserts and seal welds for the CRDM, reactor recirculation valve (RRV) and RVV connections. The NRC staff finds that the augmented inspection is acceptable because areas such as the SG feedwater plenum access covers, the SG main steam plenum access covers, the pressurizer heater bundles, and the instrument seal assemblies would be routinely disassembled and inspected and will be used as a basis to provide indication of deterioration that could also be affecting the integrity of the threaded inserts and the associated seal weld for the RRV valves, RVV valves and CRDM connections. The augmented inspection plan ensures that these defects will be detected as indication that the underlying reactor vessel head alloy steel is degrading which could compromise the bolted connection. The inspection of these areas would give a statistically significant number of threaded inserts and seals welds to provide adequate assurance of the integrity of the threaded inserts and seal welds. If defects or corrosion are found in the threaded insert or seal welds for areas such as the SG feedwater plenum access covers, the SG main steam plenum access covers, the pressurizer heater bundles, and the instrument seal assemblies, the inspection would be expanded to include the CRDM, RRV valves and RVV valves connections to verify the integrity of these threaded inserts and seal welds.

The materials used for the pressure housing components identified in FSAR Table 5.2-3, supplemented by ML24215A089, are austenitic stainless steel (SA182, Type F304 or F304LN and SA-479, Type 304 or Type 410) with additional requirements of 0.03 percent maximum carbon. The staff reviewed the specifications and grades of the CRDM pressure housing materials and verified that the materials listed meet the requirements of ASME Code, Section III, Paragraph NB-2121, which requires the use of materials listed in ASME Code, Section II, Part D, Subpart 1, Tables 2A and 2B. The pressure boundary materials are low-carbon austenitic stainless steels, 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 ensure that they function reliably to meet the requirements of GDC 26. Non-pressure-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, nickel-based Alloy 625, and cobalt-based alloys (Haynes 25 and Stellite 6). Filler metals that are used for non-RCPB items that may be external to or within the RCPB 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 non-pressure boundary CRDM components in contact with reactor coolant that meet the requirements of ASME Code, Section III, Paragraphs NB2160, NC2160, NB3120, and NC3120. These materials have satisfactory operating experience, are compatible with the reactor coolant, and are procured in the solution annealed condition. In addition, FSAR Section 4.5.1.2, “Austenitic Stainless Steel Components,” supplemented by ML24215A089, 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 non-pressure 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 non-pressure 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 non-pressure-retaining CRDM components are acceptable and meet the requirements in GDC 1, GDC 14, GDC 26, and 10 CFR 50.55a.

4.5.1.4.2 Austenitic Stainless Steel Components.

FSAR Section 4.5.1.2, supplemented by ML24215A089, states that the processing and welding of austenitic stainless steel base materials, which are procured in the solution-annealed condition for CRDM applications, are consistent with the recommendations of RG 1.44 to prevent sensitization. The staff notes that the solution annealed condition ensures a homogenous and nonsensitized material. In addition, austenitic stainless steels that are subjected to sensitization temperatures are procured with a maximum carbon content of 0.03 percent and are verified to be nonsensitized by testing in accordance with American Society for Testing and Materials A262, "Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels," issued September 2015. The controls specified in FSAR 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 to 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 to SCC in austenitic stainless steels. FSAR Section 4.5.1.1, "Materials Specifications," states that cold-worked austenitic stainless steel materials are avoided and that austenitic and martensitic stainless steels with a 0.2 percent offset yield strength greater than 620 MPa (90,000 psi) are not used in CRDM components to reduce the probability of SCC. This practice is consistent with SRP Section 4.5.1 when strain-hardened stainless steels are used and, therefore, is acceptable.

FSAR Section 4.5.1.2 states that the recommendations of RG 1.31 are used for the filler metal material in the CRDM components and are analyzed for delta ferrite content and limited to a ferrite number (FN) between 5 FN and 20 FN except for Types 316 and 316L, which are limited to the range of 5 FN to 16 FN. 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.

4.5.1.4.3 Other Materials

Materials other than austenitic stainless steels that are used to fabricate pressure boundary and non-pressure boundary CRDM components are described below. These materials include Type 410 martensitic stainless steel, nickel-based Alloy X-750, and cobalt-based material (Stellite 6 and Haynes 25) for the components in FSAR Figures 4.6-1 through 4.6-5, and Figure 1 of a letter dated August 2, 2024 (ML24215A001/ ML24215A090 (nonproprietary), ML24215A091 (proprietary)).

FSAR Section 4.5.1.3, "Other Materials," and Table 4.5-1, "Acceptable Control Rod Drive Mechanism Materials," supplemented by ML24215A089 and ML24215A090, state that the magnetic part of the latch mechanism; magnetic parts including the poles and supports of CRDM shaft lobed section, taper key, connecting rod, retaining ring, extension tubes, and protective sleeves; the magnetic parts for the rod holdout (RHO) ball grip assembly contained within the RCPB; and the coil housings and drive coils for the remote disconnect mechanism and RHO that are external to the RCPB, are fabricated from Type 410 martensitic stainless steel. Type 410 components used in the CRDMs are quenched and tempered with a minimum tempering temperature of 566 °C (1,050 °F), which is consistent with SRP Section 4.5.1, paragraph II.4, to ensure that these materials will not deteriorate from SCC in service. The staff finds this acceptable because the heat treatment is in accordance with the guidance in SRP Section 4.5.1, paragraph II.4, to provide assurance that these martensitic stainless steels will not deteriorate from SCC in service.

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

Nickel-based Alloy X-750 spring material and heat treatment conform to the requirements of AMS 5698 or AMS 5699, which include solution heat treatment above 1,100 °C (2,000 °F), based on operating experience for minimizing SCC in this alloy. In addition, the CRDM coil springs are not designed to be stressed beyond their elastic limit or creep limit to maintain spring functionality and minimize the potential for SCC. Finally, there have been no operating experience reports of SCC of nickel-based Alloy X-750 CRDM springs fabricated to the requirements of AMS 5698 and AMS 5699. Therefore, the staff finds this material and the heat treatment of this precipitation-hardenable alloy acceptable because it is based on industry experience and will ensure that the material properties of the component are capable of maintaining its structural integrity and performing its intended function.

FSAR Section 4.5.1.3 and Table 4.5-1, supplemented by ML24215A089 and ML24215A090, state that Haynes 25 and Stellite 6 material are used for wear-resistant parts such as pins, grip arms and links for the latch mechanisms within the RCPB. These materials are commonly used in operating plants and have satisfactory operating experience; therefore, they are acceptable to the staff.

Alloy 625, Alloy 718, Type 440C, and their associated weld filler metals, E410 and E430, are used for improved strength in select CRDM components listed in FSAR Table 4.5-1, such as for CRDM shaft couplings, anti-rotation keys, steel balls, spring bushings, and magnetic jack assembly support items. Grades 8, B8 and B8M, Type 630 H1100 are used for fasteners both within and external to the RCPB. Nickel-based Alloy 625 and Alloy 718, along with stainless steel Type 440C, and Grades 8, B8 and B8M have been used in operating plants and have satisfactory operation experience in pressurized-water environments; therefore, they are acceptable to the staff.

4.5.1.4.4 Cleaning and Cleanliness Controls

FSAR Section 4.5.1.4, “Material Cleaning and Cleanliness Control,” discusses the cleaning and cleanliness controls for the CRDM during manufacture and assembly. FSAR 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 FSAR Section 4.5.1.4 states that controls for the handling and cleaning of austenitic stainless steel surfaces are used to control contamination as specified in ASME NQA-1, the staff finds this acceptable. Therefore, the staff finds the applicant's cleaning and cleanliness controls for CRDM components acceptable and consistent with SRP Section 4.5.1.

4.5.1.5 Combined License Information Items

No COL information items are associated with FSAR Section 4.5.1.

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 FSAR describes the reactor vessel internals (RVIs) and core support materials.

4.5.2.2 Summary of Application

FSAR Section 4.5.2, "Reactor Internals and Core Support Materials," describes the design, as summarized, in part, below.

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

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

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

FSAR Section 4.5.2 describes the materials used to fabricate RVIs and core support structures. The application provides information about the controls on welding, NDE, fabrication and processing of austenitic stainless steel components, and materials other than austenitic stainless steel. Each topic is discussed below.

4.5.2.2.1 Materials Specifications

FSAR Section 4.5.2, supplemented by ML24346A187 and a letter dated August 2, 2023 (ML24215A001/ ML24215A092), includes Table 4.5-2, "Reactor Vessel Internals Materials," which lists the components in the RVIs and the type and specification of the alloy of each component. The FSAR states that all portions of the RVI that perform a core support function are designed and fabricated as Class CS in accordance with ASME Code, 2017 Edition, Section III, Subsection NG. The materials for core support structures and threaded structural fasteners conform to the requirements of ASME Code, 2017 Edition, Section III, Subsubarticle NG-2120, and the applicable requirements of ASME Code, 2017 Edition, Section II, Part D, Tables 2A and 2B and Code Case N-60-6. The remaining portions of the RVI are designated as internal structures and are designed to conform to ASME Code, 2017 Edition, Section III, Paragraph NG-1122.

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

4.5.2.2.2 Controls on Welding

The FSAR requires all welding of RVI materials to conform to the applicable requirements of ASME Code, 2017 Edition, Section III, Articles NG-2000, NG-4000, and NG-5000. Welding is conducted using procedures qualified according to the rules of ASME Code, 2017 Edition, Section III, Subarticle NG-4300, and Section IX of the latest ASME Code edition. Welders and welding operators are qualified in accordance with ASME Code, Section IX, of the latest edition and RG 1.71, Revision 1.

Electroslag welding is not permitted on RVI and core structural supports. Nickel-base Alloy 600 and associated weld filler materials are not used in the RVI.

4.5.2.2.3 Nondestructive Examination

The FSAR 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.

4.5.2.2.4 Fabrication and Processing of Austenitic Stainless Steel Components

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

The FSAR requires implementation of the guidance in RG 1.44 to control the use of sensitized austenitic stainless steel.

The FSAR 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 also applies to weld filler metals, as well.

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

FSAR Section 5.2.3 describes further controls to minimize harmful contaminants. The applicant described acid pickling as “avoided on stainless steel” and “not used on sensitized austenitic stainless steel.”

4.5.2.2.5 Other Materials

Every component in FSAR Table 4.5-2 is an austenitic stainless steel.

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 FSAR 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

The FSAR specifies that core support materials will satisfy the requirements of ASME Code, Section III, Subarticle NG-2120 and the applicable requirements of ASME Code, Section II, Part D, Tables 2A and 2B, and Code Case N-60-6. The remaining portions of the RVIs are designed to conform to ASME Code, Section III, Article NG-1122. The staff finds this to be acceptable because it complies with the ASME Code and 10 CFR 50.55a.

In FSAR Section 4.2.3.1.4 and TR-117605-P, Revision 0, Section 4.1.4, the applicant stated that crevice corrosion was not a potential degradation mechanism for the NuScale design. Based on large LWR operating experience and examination of the applicant’s engineering drawings on the docket, the staff accepted that the potential for crevice corrosion was low enough to not merit further consideration.

4.5.2.4.2 Controls on Welding

The staff reviewed the controls on welding in FSAR Section 4.5.2.2, “Control on Welding,” specifically, the citations of ASME Code sections, RG 1.71 guidance, and FSAR Section 5.2.3 information. The staff found the information 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 FSAR Section 4.5.2.3, “Nondestructive Examination,” specifically, the citation of ASME Code sections. The staff found the information presented acceptable because it complies with the SRP criteria for this topic.

4.5.2.4.4 Fabrication and Processing of Austenitic Stainless Steel Components

The staff reviewed FSAR Section 4.5.2.4, “Fabrication and Processing of Austenitic Stainless Steel Components,” with emphasis on heat treatment, controls on sensitization, compatibility with reactor coolant, abrasive work, and minimization of contamination. The staff confirmed that the applicant noted appropriate controls on heat treatments. The staff confirmed that environmental conditions are controlled and that welding procedures are developed to minimize the probability of sensitization and microfissuring. This is achieved by following the guidance of RG 1.44 and RG 1.31, respectively. The staff confirmed the RVI and core support material compatibility with coolant through a review of the selection of materials for each component; a commitment to RGs and ASME Code requirements; the topics detailed in FSAR Section 4.5.2.4 and Section 5.2.3.4, “Fabrication and Processing of Austenitic Stainless Steels”; and the water chemistry requirement for oxygen content in FSAR Section 5.2, “Integrity of Reactor Coolant Boundary,” Table 5.2-4, “Reactor Coolant Water Chemistry Controls.” The oxygen concentration requirements of less than or equal to 0.10 parts per million (ppm) meets 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. FSAR 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

No materials other than austenitic stainless steels are used in the reactor internals.

4.5.2.5 Combined License Information Items

No COL information items are associated with FSAR, Section 4.5.2.

4.5.2.5 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 FSAR, Section 4.6, “Functional Design of Control Rod Drive System,” to confirm that the CRDS can reliably control reactivity, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of a postulated accident. The NMP-20 design also uses the CVCS to control reactivity. The staff’s review in this section focused on the functional performance of the CRDS, including the consideration of single and common-cause failures.

4.6.2 Summary of Application

FSAR Section 4.6 describes the system, as summarized, in part, below.

The NMP-20 design includes two reactivity control systems: (1) the CRDS and (2) the CVCS. The NMP-20 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. FSAR Section 3.9.4, “Control Rod Drive System,” describes the mechanical design of the CRDM. FSAR Section 7.0.4, “System Descriptions,” provides the instrumentation and controls (I&C) for the CRDS. Finally, FSAR Section 14.2 addresses the initial startup testing of the CRDS.

FSAR Chapter 15 demonstrates that, for all DBEs, the CRDS and ECCS (through the supplemental boron system) are capable of maintaining the reactor within acceptable limits under the assumption that the most reactive control rod is stuck out.

FSAR Section 9.3.4, discusses the CVCS in more detail.

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

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

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

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

4.6.3 Regulatory Basis

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

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

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

- GDC 25, as it relates to the protection system’s capability to assure that the SAFDLs are not exceeded for any single malfunction of the reactivity control systems

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

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

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

- GDC 29, "Protection against anticipated operational occurrences," as it relates to the requirement that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs

4.6.4 Technical Evaluation

The staff reviewed FSAR 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.

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

FSAR Figure 4.6-1, "Representative Overview of Control Rod Drive Mechanism Locations in Relation to the Reactor Pressure Vessel and Containment Vessel," depicts the CRDS and its relationship to the reactor and CNVs. FSAR Figures 4.6-2 through 4.6-6 give details on the CRDMs. FSAR Section 3.9.4 further describes the CRDS.

In Section 3.9.4 of this SER, the NRC staff 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. Section 4.2 of this SER contains the staff evaluation of the CRA design.

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

FSAR Table 3.9-19, "Classification of Structures, Systems, and Components," states that the control rod drive shaft, latch mechanism, and CRA are safety-related, are designed to be seismic Category I, and are required to meet the quality assurance requirements of 10 CFR Part 50, Appendix B. FSAR Table 3.11-1, "List of Environmentally Qualified Equipment

Located in Harsh Environments,” shows that CRDS is part of the equipment qualification program and must function to mitigate design-basis accidents.

In addition, the staff reviewed whether CRDS components not included Table 3.11-1, such as the control rod drive and position coils and associated CRDM control cabinets, will be designed to perform their functions in the environmental conditions expected during normal operation. In a letter dated August 2, 2024 (ML24215A001/ML24215A093), the applicant stated the CRDM drive and position coils and associated control cabinets are not subject to 10 CFR 50.49 “Environmental qualification of electric equipment important to safety for nuclear power plants.” Thus, they are not subject to the GDC 4 harsh environmental qualification in Table 3.11-1, because these components do not perform safety-related functions post-DBE. While the applicant has demonstrated that the components are not subject to a harsh environment, the scope of relevant GDC applicable to the CRDS (e.g., GDC 4 and GDC 26) includes both normal operation and transient considerations. Specifically, GDC 4 requires the CRDS, which includes the control rod drive and position coils and associated CRDM control cabinets, to “be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing.” Similarly, GDC 26 requires the CRDS to be capable of performing its safety function of providing a positive means for inserting control rods and reliably controlling reactivity changes during normal operation. Accordingly, the applicant further clarified that the CRDM drive and position coils are water cooled to maintain temperature below the design temperature of the CRDM coil windings (discussed in more detail below), and the control and position indication cabinets are located in separate rooms in the reactor building that are held at a normal temperature of 105°F, ambient pressure, and relative humidity of less than 85 percent. The applicant also added that the CRDS is subject to an operability assurance program as described in FSAR Section 3.9.4.4, “Control Rod Drive System Operability Assurance Program.”. Therefore, notwithstanding the staff clarification of applicable GDC regarding normal operation, the staff finds acceptable the exclusion of the control rod drive and position coils and associated CRDM control cabinets from FSAR Table 3.11-1.

FSAR Section 4.6.2, “Design Bases,” states that jet impingement loads generated from high -energy lines inside the CNV are analyzed in FSAR Section 3.6, and supporting methodology described in TR-121507-P, Revision 1 “Pipe Rupture Hazards Analysis,” issued November 7, 2024 (ML24312A401 (nonproprietary), ML24312402 (proprietary)). 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 RVVs 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. In Section 3.6 of this SER, staff evaluates high -energy line breaks inside the CNV.

In accordance with GDC 4, 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. FSAR Section 4.6.1 states that the CRDS components internal to the RCPB are designed to function in borated primary coolant with up to 2,000-ppm boron at primary coolant pressures and temperatures ranging from ambient up to the RPV design pressure and temperature above normal operating conditions. FSAR Section 3.9.4.3 species the RCS design pressure as 2,200 psia, and the design temperature as 650 °F.

FSAR 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 FSAR Figure 4.6-3, “Representative Control Rod Drive Mechanism Drive Coil and Cooling Jacket Assembly.” FSAR Section 4.6.1, adds that the reactor component cooling water system (RCCWS) discussed in FSAR Section 9.2.2, “Reactor Component Cooling Water System,” provides the cooling requirements for the CRDMs. FSAR Section 4.6.1 states that the cooling requirements for the CRDMs are provided by the reactor component cooling water system, which maintains the CRDM winding temperature below the maximum design temperature of 392 °F.

Section 9.2.2 of this SER 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.

Due to the unique orientation of the CRDMs above the pressurizer in a borated steam environment, the staff assessed the potential for chemical buildup to impact CRDM function and documented it in Section 3.9.4.4.5 of this SER.

In accordance with GDC 25, a single failure in the CRDS should not prevent the system from performing its safety -related function. The applicant evaluated failures of the CRDM in a failure modes and effects analysis (FMEA). However, the applicant did not provide the FMEA as part of its application. Therefore, the staff audited the FMEA (ML24211A089). The FMEA demonstrated that (1) no single failure in the CRDS could prevent a reactor trip and (2) the ability to rod drop on command was retained. Additionally, the FSAR Chapter 15 safety analysis accounts for the highest-worth controls rod assembly stuck out of the core. 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 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 the appropriate margin to consider for stuck rods. Chapter 7 of this SER evaluates the CRDS requirements of GDC 23 and GDC 25 with respect to I&C aspects of the protection system. Chapter 15 of this SER 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 notes that, in the long term following a reactor trip, additional boron may be needed from the CVCS or ESB in order to maintain the reactor subcritical. Section 4.3.4.3 of this SER contains the staff's evaluation associated with GDC 26 related to independent reactivity control systems.

The analyses in FSAR Chapter 15, supplemented by ML25030A349 and ML24353A218, show that the CRDS is capable of bringing the core to a shutdown condition and maintaining fuel integrity, consistent with the design information in FSAR Section 4.3. The applicant stated that the NPM-20 provides reactivity control systems, in conjunction with absorber addition by the ECCS, to reliably control reactivity changes under postulated accident conditions. The NPM-20 design includes both a movable rod reactivity control system and a passive poison addition function as part of the ECCS. The applicant has performed calculations to demonstrate that the core has margin sufficient to shut down the reactor, while crediting the ESB function, and assuming the highest-worth control rod is stuck out.

As discussed in Sections 15.0.5.4 and 4.3.4 of this SER, the staff confirmed during its review that the capability of the CRAs, in conjunction with the ESB, to control reactivity changes under postulated accident conditions with appropriate margin for stuck rods, ensures the reactor is shut down and the capability to cool the core is maintained. Furthermore, the CRA with the highest worth is assumed to remain fully withdrawn from the core and, together with added boron from the ESB recirculated into the core during ECCS operation, is capable of holding the reactor subcritical under DBE conditions, including postulated accidents, for a minimum of 72 hours following the event. Accordingly, the staff concludes that the applicant has satisfied the requirements of GDC 27.

GDC 28 requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to prevent the adverse effects of postulated reactivity accidents. A postulated failure of the CRDS that causes a rod ejection has the potential to result in a relatively high rate of positive reactivity insertion, which could challenge fuel design limits, the RCPB, and the capability to cool the core. FSAR Section 4.6.2, states that the CRDM is made as a single-piece housing, with a top plug. In Section 3.9.4 of this SER, staff evaluates the mechanical aspects of the CRDM housing. Section 15.4.8 of this SER covers the rod ejection analysis.

FSAR Section 3.1.3.9, “Criterion 28-Reactivity Limits,” states that the NMP-20 design places limits on the worth of CRAs, the maximum CRA withdrawal rate, and CRA insertion (i.e., PDILs). FSAR Table 4.3-3, “Reactivity Requirements for Control Rods,” provides the reactivity requirements for control rods. FSAR Section 15.4.1, “Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition,” defines the maximum allowed withdrawal rate of a CRA to be 38 centimeters (15 inches) per minute. 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. Sections 15.4.1 through 15.4.3 of this SER contain the NRC staff’s evaluation of rod withdrawal events. In Section 15.4.8 of this SER, staff evaluates a rod ejection accident.

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

Section 15.0.5 of this SER gives the staff’s review associated with reactivity control during long-term cooling and boron mixing.

FSAR Section 4.6.3, “Testing and Verification of the Control Rod Drive System,” refers to FSAR Section 3.9.4.4, and Section 4.2.4 for the testing and verification of the CRDS. FSAR 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 included performance testing, stability testing, endurance testing, and production testing. In addition, FSAR

Section 1.5.1, “NuScale Testing Programs,” describes testing programs associated with the design features of the CRDS.

FSAR Section 4.6.3 refers to the preoperational and initial startup test program for the CRDS in FSAR Section 14.2. The following tests from the FSAR apply to the CRDS:

- Table 14.2-65: Test #65, “Steam Generator Flow-Induced Vibration”
- Table 14.2-73: Test #73, “Control Rod Drive System—Manual Operation, Rod Speed, and Rod Position Indication”
- Table 14.2-74: Test #74, “Control Rod Assembly Full-Height Drop Time”
- Table 14.2-75: Test #75 “Control Rod Assembly Ambient Temperature Full-Height Drop Time”
- Table 14.2-92: Test #92 “Control Rod Assembly Misalignment”
- Table 14.2-98: Test #98, “Reactor Trip from 100 Percent Power”

Section 14.2 of this SER gives the staff’s review of the applicant’s initial test program.

The staff concludes that the CRDS meets the requirements of GDC 29 because the tests described 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 FSAR 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 Section 4.6.4 of this SER, the NRC staff concludes as follows:

- The functional design of the CRDS satisfies GDC 4 because the CRDMs are designed to accommodate the effects of and are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.
- The functional design of the CRDS satisfies GDC 23 because the CRDMs fail into a safe state.
- The functional design of the CRDS satisfies GDC 25 because the evaluation of a single malfunction of the reactivity control system (1) uses conservative values for the rate of reactivity insertion and (2) accommodates a stuck control rod.
- The functional design of the CRDS satisfies GDC 26 because the AOOs are evaluated with appropriate margin for stuck control rods.
- The functional design of the CRDS satisfies GDC 27 because the CRDS, with the control rod of the highest worth stuck out and in conjunction with the ECCS supplemental boron system, is capable of reliably controlling reactivity changes by

holding the reactor core subcritical and the capability to cool the core is maintained following a postulated accident.

- The functional design of the CRDS satisfies GDC 28 because the CRDS is designed with appropriate limits on the potential amount and rate of reactivity increase.

- The functional design of the CRDS satisfies GDC 29 because the tests and design of the CRDS ensure, with an extremely high probability, that the CRDS will accomplish its safety function in the event of an AOO.