U.S. NUCLEAR REGULATORY COMMISSION

SAFETY EVALUATION BY THE OFFICE OF NEW REACTORS

TOPICAL REPORT TR-0616-48793, REVISION 1,

"NUCLEAR ANALYSIS CODES AND METHODS QUALIFICATION,"

NUSCALE POWER, LLC

PROJECT NO. 0769

1.0 INTRODUCTION AND BACKGROUND

By letter dated August 30, 2016 (Reference 1), as supplemented by letter dated July 6, 2017 (Reference 2), NuScale Power, LLC (NuScale), requested that the U.S. Nuclear Regulatory Commission (NRC) review and approve Topical Report (TR) TR-0616-48793, "Nuclear Analysis Codes and Methods Qualification." This TR describes the NuScale methodology for the design and steady-state analysis of the NuScale reactor core (RXC). This methodology includes the use of the Studsvik Scandpower, Inc. (Studsvik), Core Management Software, Version 5 (CMS5) Suite.

By letter dated March 7, 2017, NuScale provided supplemental information in support of the NuScale TR (Reference 3). In addition to the information provided by NuScale, Studsvik submitted a generic CMS5 TR for NRC review and approval (References 4 and 5). The NRC staff considered the information provided by Studsvik to support the review of TR-0616-48793.

By letter dated July 30, 2018, NuScale submitted Revision 1, TR-0616-48793, "Nuclear Analysis Codes and Methods Qualification," which incorporates all changes made to the TR during NRC staff's review (Reference 37).

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47, "Contents of Applications; Technical Information," and 10 CFR 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report," require a final safety analysis report (FSAR) to analyze the design and performance of structures, systems, and components. Safety evaluations performed to support the FSAR require reactor physics parameters to determine RXC performance under normal operations, including anticipated operational occurrences, and accident conditions. An applicant employs an approved nuclear analysis methodology to provide reactor physics parameters for use in safety evaluations. Additionally, an applicant uses the nuclear analysis methodology to establish a partial basis for demonstrating compliance with the following general design criteria (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities":

Enclosure 1

- GDC 11, “Reactor Inherent Protection,” 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 specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

- GDC 26, “Reactivity Control System Redundancy and Capability,” requires, in part, that the control rods be capable of reliably controlling reactivity changes to assure that, under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for stuck rods, specified acceptable fuel design limits are not exceeded.

- GDC 27, “Combined Reactivity Control Systems Capability,” requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, 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.

The scope of the NRC staff’s review includes:

1. the applicability of the CMS5 code suite to the NuScale RXC design;

2. the suitability of the base nuclear reliability factors (NRF); and

3. the applicability of the NRF update methodology.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed TR-0616-48793 as described in the following sections of this safety evaluation (SE):

- Section 3.2 of this SE assesses the fidelity to which the CMS5 code suite captures the geometry, material properties, and physics applicable to the NuScale design.

- Section 3.3 of this SE assesses the code-to-code benchmarking of the CMS5 code suite.

- Section 3.4 of this SE assesses code validation for the CMS5 code suite.

- Section 3.5 of this SE assesses the application uncertainty considerations of the CMS5 code suite for the figures of merit as applicable to the NuScale design.

3.1 Introduction and Outline

The TR-0616-48793 is divided into 10 sections that NRC staff evaluated as follows:

- Section 1.0 of the TR provides an introduction to the report and clarifies the specific items for which NRC approval is requested. The NRC has no regulatory findings associated with Section 1.0 of the TR.

- Section 2.0 of the TR describes the NuScale physical reactor design for which the TR methodology is applied. The NRC staff considers this information in its evaluations in Sections 3.2, "Geometric, Material, and Physics Modeling Capabilities," 3.4, "Code Validation," and 3.5, "Application Uncertainty," of this SE.

- Section 3.0 of the TR describes the CMS5 code suite and provides details on its application to the NuScale design. The NRC staff evaluates this information in Section 3.2 of this SE.

- Section 4.0 of the TR provides a background on the statistical techniques used to assess the code against benchmarks and to develop the NRFs. The NRC staff evaluates this information in Section 3.5.1 "Statistical Methodology," of this SE.

- Section 5.0 of the TR describes the benchmarking of the CMS5 code suite against a higher order code in Monte Carlo N-Particle 6 (MCNP6) and against experimental data. The NRC staff evaluates the benchmarking of the CMS5 code suite against MCNP6 in Section 3.3, "Code-to-Code Benchmarking" of this SE and the benchmarking against experimental data in Section 3.4 of this SE. In addition, the NRC staff evaluates the impact of this benchmarking on the development of the NRFs in Section 3.5 of this SE.

- Section 6.0 of the TR describes the benchmarking of the CMS5 code suite against operational data from the Three Mile Island Unit 1 (TMI-1) reactor. In addition, the TR presents the base NRFs for the CMS5 code suite as applied to current operating reactors. The NRC staff evaluates this benchmarking in Section 3.4 of this SE and evaluates its impact on the development of the NRFs in Section 3.5 of this SE.

- Section 7.0 of the TR describes the methodology for developing and updating the NRFs. The NRC staff evaluates this information in Section 3.5 of this SE.

- Section 8.0 of the TR describes the intended applications for the CMS5 code suite. The NRC staff considers this information in Section 3.5 of this SE.

- Section 9.0 of the TR provides a brief summary and conclusion statement. The NRC has no regulatory findings associated with Section 9.0 of the TR.

- Section 10.0 contains the references list. The NRC has no regulatory findings associated with Section 10.0 of the TR.

3.2 Geometric, Material, and Physics Modeling Capabilities

3.2.1 Geometric Modeling

Section 3.1 of the TR describes the CASMO5 code as a two-dimensional (2D) transport theory code based on the Method of Characteristics (MOC) that is used to generate pin cell or

assembly lattice physics parameters for light-water reactors. As described in Section 3.2 of the TR, CMSLINK5 processes these lattice physics parameters to develop a neutronic data library for use in SIMULATE5. Section 3.3 of the TR describes SIMULATE5 as a three-dimensional (3D), steady-state, nodal diffusion code that solves the multigroup nodal diffusion equation.

The CASMO5 theory manual (Reference 6) describes the CASMO5 neutron data library as [

] The 2D MOC calculation explicitly captures the heterogeneity in the fuel lattice, [ ]. The applicant applied this calculation to develop multigroup cross sections and discontinuity factors for use in SIMULATE5 (Reference 6). For the regions of the fuel containing spacing grids, the 2D MOC calculation [ ] (References 3 and 7). The analyses performed in SIMULATE5 capture the full 3D arrangement of the reactor core but with homogenized nodes (Reference 8). NuScale used the default four energy group structure in SIMULATE5 (References 3 and 9). Based on the description of the overall CMS5 analysis, which begins with a simplified geometry and detailed energy group structure and then progresses to a detailed geometry with a simplified group structure, the NRC staff finds that the process is consistent with established precedent (Reference 10) and reflects the current state of the art (Reference 11).

Section 2.0 of the TR describes the NuScale physical reactor design for which the TR methodology is applied. The NRC staff acknowledges that the fuel assembly design is consistent with existing fuel designs for pressurized-water reactors (PWR) but with reduced height. Additional geometrical attributes of the NuScale Power Module (NPM) considered by the NRC staff include the potential for multimodule effects, axial zoning of the control rod assemblies (CRA) in CMS5, the effect of fixed core instrumentation on the nuclear design, and the basis for the modeling of the radial reflectors.

The design description and analyses presented in the TR are for a single NuScale RXC. Because of the presence of several NPMs relatively close to each other and the location of ex-core detectors for a single NPM, the NRC staff questioned whether multimodule effects need to be considered. Accordingly, on May 8, 2017, the NRC staff issued Request for Additional Information (RAI) 8807, Question 29739, asking NuScale to provide evidence to show that multimodule effects can be neglected in the nuclear design and analysis of the NPM (Reference 12). In a letter dated July 6, 2017 (Reference 2), the applicant stated that, for an NPM operating at 100-percent power, a conservative analysis showed that the attenuated neutron flux seen at the exterior of the containment vessel of a neighboring module was approximately five orders of magnitude below the neutron flux level associated with the startup of a module. During a regulatory audit, the NRC staff examined the calculation supporting the applicant's RAI response and noted that NuScale's analysis showed that the total flux at the containment nuclear vessel that has been attenuated from an adjacent nuclear power module (that is operating at 100-percent power) is negligible (i.e., the contribution to the flux observed by the ex-core detectors is several orders of magnitude less than the uncertainty associated with a startup neutron count rate) (Reference 13). In addition, the applicant's response included a markup to the TR to clarify that the methodology is applicable to a single NuScale RXC, independent of the presence of additional NPMs. Based on the applicant's analysis showing

that multimodule effects are negligible, the NRC staff finds the response and associated TR markups acceptable.

During a regulatory audit, NRC staff asked the applicant to clarify the approach used to model the axial zoned CRAs (Reference 13). The applicant explained that the CRAs for the NuScale design have multiple axial segments with different compositions and geometry. From bottom to top, the absorbing region of the CRA contains four segments:

- the annular lower silver-indium-cadmium (AIC) slug with a central pin for stack support;
- the stack support head;
- the solid upper AIC slug; and
- the boron carbide pellet stack.

The applicant first modeled each segment with explicit material composition and radial geometry in CASMO5 in order to generate cross sections and nuclear data specific to each individual segment. It then modeled each segment in SIMULATE5 with the appropriate height and position. The normal axial nodalization of the SIMULATE5 core model is overlaid with subnodes for each CRA segment, as well as other neutronically distinct regions. The subnodes are broken down sufficiently to achieve axial homogeneity (i.e., no axial variation within a subnode), allowing for the use of explicit cross section data to more accurately represent axial discontinuities that may be present for reasons such as CRA positioning and axial composition. The NRC staff finds the modeling approach described by the applicant acceptable because it explicitly captures the axial heterogeneity in the CRAs.

During a regulatory audit, the NRC staff asked the applicant to clarify whether the analysis considered the impact of in-core instrumentation on reactivity and power distributions (Reference 13). The applicant explained that in-core instruments can be modeled explicitly in CASMO5. However, the detailed design of in-core instrumentation for the NuScale reactor is not yet finalized, so the CASMO5 and SIMULATE5 NuScale RXC models used for benchmarks in the TR do not contain in-core instrumentation information, and all instrumentation tubes are therefore modeled as water-filled. The applicant further explained that:

1. the impact to reactivity and power distributions is not considered significant because the in-core instruments reside in only 12 of the 37 assemblies in the reactor core, and the instrument tube is only 1 of 289 pin cells in an assembly; and

2. performing benchmarks of the NuScale RXC with in-core instrumentation modeled has negligible impact on the ability of the codes to accurately predict reactivity and power distributions.

The NRC staff further investigated the precedent and impact of accounting for in-core instrumentation in the lattice physics calculations. The NRC staff reviewed previously approved nuclear design methodologies and determined that explicit modeling of the in-core instrumentation in the lattice physics calculations is not performed for existing design methodologies (References 14 and 15), but that the uncertainties associated with using fixed in-core instrumentation are addressed when applying NRFs (Reference 5). Additionally, detailed

reactor physics calculations in the literature show that the impact of instrumented thimble tubes is small, with a negligible contribution to reactivity and a localized reduction of pin powers of 3 to 5 percent for fuel pins adjacent to the thimble tube (References 16 and 17). Based on the existing precedent, the small impact of the in-core instrumentation, and capturing the effect of in-core instrumentation in the NRFs, the NRC staff finds that the applicant is not required to explicitly model in-core instrumentation in the lattice physics calculations (i.e., CASMO5).

During a regulatory audit, the NRC staff asked the applicant to clarify the radial reflector modeling approach as presented in Figure 3-5 of the TR (Reference 13). The applicant explained that the modeling of the radial reflector in the NuScale RXC is split into multiple segments matching the square array and pitch of the fuel assemblies. The applicant identified radial reflector segments according to their position around the core and the relative material composition of the segment, as shown in Figure 3-5 of the TR. Each unique radial reflector segment has different relative proportions and locations of stainless steel and water and thus has different reflection capability. In Figure 3-5 of the TR, only reflector segments 1, 2, and 5 are adjacent to a fuel assembly. However, the method of modeling each reflector segment in CASMO5 requires an adjacent fuel assembly to provide a source of neutrons. The applicant modeled the radial reflector with multiple concentric reflector segments that are modeled in CASMO5 using the same technique. The applicant explained that the radial reflector modeling is performed in this manner in order to maintain fidelity of the reflector regions that are closest to the core while aligning the fuel submesh, radial reflector submesh, and neutronically distinct layers of the radial reflector. The NRC staff finds the applicant’s approach to modeling the radial reflector acceptable because it explicitly captures the neutronically distinct regions of the radial reflector.

3.2.2 Material Modeling

Section 2.0 of the TR describes the reactor design and discusses the materials used in the fuel, CRAs, and heavy reflector. Section 2.2 of the TR describes the fuel rods as consisting of ceramic pellets of up to 4.95-percent enriched UO_2 encapsulated in M5®, a zirconium-based alloy. Section 2.3 of the TR describes the CRAs as composed of 24 rods fastened to a hub, with each rod containing two absorber materials—AIC at the bottom of the rod and boron carbide. Section 2.4 of the TR describes the use of gadolinia (Gd_2O_3) ([ ]) as a burnable poison, which is used to compensate for the excess reactivity needed for full-power operation over the entire cycle. Section 2.6 of the TR describes the stainless steel heavy reflector used to reduce leakage from the NuScale RXC. The NRC staff compared the materials used in the NuScale RXC with the reactors and experiments used to validate the nuclear design methodology, described in Section 3.4 of this SE, and finds that the applicant applied the CMS5 code suite to materials within the validation basis and at the appropriate neutron energy spectrum.

The NRC staff further investigated the material modeling capabilities of the thermal-physical parameters. In the generic CMS5 methodology, CASMO5 performs the thermal expansion calculations such that hot cross sections and pin power data are calculated based on conditions representative of core operation (Reference 18), and the thermal-physical modeling capabilities in SIMULATE5 account for the impact of burnup and geometry changes resulting from swelling, densification, and thermal expansion (Reference 8). The thermal-physical models for the fuel, however, are provided for specific materials. The NRC staff considered these limitations as part the generic review for CMS5, and Studsvik further clarified the range of applicability for the

CMS5 code suite in response to an RAI (Reference 19, Question 1). As a result, the NRC staff imposed limitations on the application of CMS5 to specific materials, enrichments, and burnable poison loading (Reference 20). The NRC staff compared the limitations imposed on the generic CMS5 methodology against the NuScale RXC design described in Section 2.0 of the TR and finds that the applicant's application of CMS5 complies with the limitations identified by the NRC staff in the generic review of CMS5 (Reference 20).

The description of the NuScale RXC in Section 2.0 of the TR includes fuel cladding that is specific to a particular fuel vendor. The NRC staff recognizes that future application of this TR may involve different materials. Accordingly, the NRC staff imposed Limitation 1 (see Section 4.0 of this SE) to ensure that future use of this TR is limited to those materials identified by the NRC staff in its review of the generic CMS5 TR.

3.2.3 Physics Modeling

As described in Section 3.2.1 of this SE, the overall process incorporated into the CMS5 code suite is acceptable. The process begins with a simplified geometry and detailed energy group structure and then progresses to a detailed geometry with a simplified group structure. The CASMO5 methodology manual (Reference 6) describes the process of cross section condensation. The NRC staff reviewed this description and finds that it is consistent with the well-established methodology of preserving reaction rates as described in standard texts on reactor physics (References 10 and 11). The flux discontinuity factors, which are used in the SIMULATE5 nodal diffusion calculation to preserve the net currents on an interface calculated by CASMO5, are calculated for single assembly calculations and baffle/reflector calculations as described by Studsvik (Reference 18) and summarized as follows. The single assembly calculations, which are performed for individual fuel assemblies, use the lattice physics code CASMO5 with reflective boundary conditions (i.e., a net current of zero), and flux discontinuity factors are determined by the ratio between the surface and average fluxes. The baffle/reflector calculations are performed by imposing a black boundary condition (i.e., zero incoming current) on the reflector edge facing away from the fuel assembly, and discontinuity factors are calculated using the homogeneous flux distribution in the reflector, applying the same basis function as used in SIMULATE5. The NRC staff finds that the process for estimating discontinuity factors, as described in the CASMO5 methods and validation report (Reference 18), acceptable because it has been demonstrated to provide accurate approximations to equivalence parameters (Reference 21).

Additional physics modeling further investigated by the NRC staff includes the enhanced scattering kernel described in the TR, the applicability of the thermal-hydraulic modeling capabilities in SIMULATE5 to the NuScale design, isotope selection for microdepletion, and branch structure/histories used by the applicant to model the NuScale RXC.

Section 3.1 of the TR describes an enhanced scattering kernel as an improvement of CASMO5 over CASMO-4. The NRC staff further investigated the basis and justification for the updated scattering kernel. As part of the generic CMS5 TR review, Studsvik further described the enhanced scattering kernel in discussing the resonance upscatter model in response to an RAI, (Reference 19, Question 4d), as a set of correction factors that are applied to resonance absorption integrals. A Monte Carlo Slowing Down code calculates these intervals, and Studsvik verified CASMO5 model implementation by comparing it against a UO_2 Doppler benchmark (Reference 22). Based on the description provided by Studsvik and the CASMO5

benchmarking, the NRC staff found this model to be acceptable as part of the generic CMS5 TR review (Reference 20).

The PWR thermal-hydraulic modeling capabilities and closure correlations in SIMULATE5 are described in Chapters 9–11 of the SIMULATE5 methodology manual (Reference 8). The PWR thermal-hydraulic model in SIMULATE5 extends from the lower tie plate to the upper tie plate. The NRC staff reviewed this description and determined that the simplified steady-state model provided in SIMULATE5 conserves mass, energy, and momentum and is acceptable for use in steady-state reactor physics calculations. Additionally, the NRC staff reviewed the closure correlations and determined that they cover the operating parameters of an NPM. In regards to the prediction of void fraction, the NRC staff performed a confirmatory calculation to compare the correlation implemented into SIMULATE5 with the Chexall-Lellouche drift-flux correlation (Reference 23). This comparison showed that the void fraction correlation implemented into SIMULATE5 predicts void fractions consistent with the Chexall-Lellouche drift-flux correlation, with the SIMULATE5 model predicting higher void fractions of up to 5 percent at lower qualities (2–5 percent). For qualities above 40 percent, the two correlations predict void fractions within 1 percent of each other. Based on the closure correlations covering the range of operation of the NPM, the NRC staff finds the correlations acceptable for use in the analysis of a NuScale RXC.

Section 3.3 of the TR describes the depletion capabilities in SIMULATE5 as a hybrid microscopic-macroscopic model. Section 5.1 of the SIMULATE5 methodology manual (Reference 8) further clarifies that [

]

Section 3.4.3 of the TR briefly describes the branch structure and histories used in the CASMO5 case matrix. The CASMO5 case matrix covers the range of off-nominal conditions and is used to develop the neutronic data library for SIMULATE5. The NRC staff examined the supplemental information provided by the applicant (Reference 3) and determined that the applicant uses the default S5C option for generating the CASMO5 case matrix. Additionally, the NRC staff evaluated the PWR segment S5C case matrix and determined that the description of the histories, provided by the applicant in Section 3.4.3 of the TR, is consistent with the default histories recommended for use by the code vendor, Studsvik (Reference 7). The NRC staff compared the PWR segment S5C case matrix with the range of conditions required for the evaluation of the NuScale RXC and finds that the PWR segment S5C case matrix covers the range of moderator temperatures, moderator void fractions, fuel temperatures, boron concentrations, and rodded conditions necessary to evaluate the NuScale RXC during normal and off-nominal conditions. Because the applicant is using the default PWR segment S5C case matrix for the analyses supporting this TR and in order to ensure an adequate branch structure to cover the analysis of the NuScale RXC during normal and accident conditions, NRC staff established Condition 1 (see Section 4.0 of the SE), requiring the applicant to use PWR segment S5C case matrix.

3.3 Code-to-Code Benchmarking

In Section 5.1 of the TR, the applicant described benchmarking of the CMS5 code suite with MCNP6. As described in Section 2.3.2 of the CMS5 code vendor's generic TR (Reference 5),

higher order code comparisons, in which MCNP6 is considered a higher order code, are used to identify code-to-code trends and biases that may signal deficiencies in the code system, and to extend the range of benchmarking to materials and configurations that are not available in the experimental data. Because no experimental or operating data are presently available that are fully representative of the NuScale RXC configuration, the NRC staff agrees that code-to-code benchmarking is an appropriate method to identify potential deficiencies in the CMS5 code suite for modeling the NuScale RXC.

The applicant's code-to-code benchmarking calculations model the NuScale RXC at hot zero power (HZP) (cycle 1 and zero burnup) and at hot full power over cycles 1 through 4, with cycle average burnups of 0 and 5 gigawatt days per metric ton of uranium. The applicant simplified the code-to-code benchmarking calculations by:

1. smearing isotopic data in MCNP6 calculations for a given node across all the fuel rods in an assembly, with the exception of fuel rods that contain burnable poison;

2. relying on depletion calculations from CMS5 to obtain isotopic data for MCNP6 calculations;

3. turning off the thermal-hydraulic models in CMS5 for hot full-power calculations and explicitly specifying the fuel temperature, moderator temperature, and moderator density in CMS5 to be consistent with MCNP6;

4. imposing a fixed fission product concentration (either zero or steady-state concentration); and

5. for power distribution calculations, depositing all fission power in the fuel.

The NRC staff recognizes that simplifications may be necessary when performing code-to-code comparisons in order to accommodate differences in modeling capabilities between MCNP6 and the CMS5 code suite. The NRC staff finds the applicant's simplifications acceptable because they limit the potential for distortion attributed to modeling limitations that may be present in MCNP6.

In Section 5.1.4 of the TR, the applicant presented its results of the code-to-code comparison between MCNP6 and CMS5. Based on these code-to-code comparisons, the applicant developed tolerance limits for differential boron worth (DBW), isothermal temperature coefficient (ITC), moderator temperature coefficient (MTC), CRA worth, relative assembly power (RAP), pin power peaking factors, and axial offset (AO). However, in Section 7.0 of the TR, the applicant showed that other data bound all of the code-to-code-based tolerance limits (as evaluated further in Section 3.5 of this SE). Accordingly, the NRC staff finds that the code-to-code benchmarking shows that the CMS5 modeling fidelity, as compared to a higher order code, is nonlimiting in terms of NRF development. Additionally, based on the close agreement between the CMS5 and MCNP6 results, the NRC staff finds that the code-to-code benchmarking did not identify any deficiencies in the CMS5 code suite for modeling the NuScale RXC.

3.4 Code Validation

In this section of the SE, the NRC staff evaluates the benchmarking of the CMS5 code suite against data obtained through physical measurements (critical experiments, experimental reactors, and power reactor operation). In Section 5.2 of the TR, the applicant presented benchmarking calculations for the IPEN/MB-1, KRITZ-2, and DIMPLE experimental reactors. Additionally, Section 6.1 of the TR presents benchmarking calculations for the operating PWR TMI-1. The NRC staff also considered additional benchmarking performed by Studsvik, the CMS5 code vendor, as discussed below.

The applicant's benchmarking of the IPEN/MB-1 experimental reactor included experiments that use a heavy (Type 304 stainless steel) reflector and experiments that use UO_2-Gd_2O_3 fuel rods. The applicant performed benchmarking against the KRITZ-2 experimental data over a range of operating conditions. The applicant's benchmarking against the DIMPLE experimental data included experiments performed over a range of fuel configurations and included experiments with and without a stainless steel baffle surrounding the fuel assemblies. In Table 5-13 of the TR, the applicant presented calculated-to-measured eigenvalues for the IPEN/MB-1, KRITZ-2, and DIMPLE experiments. The applicant's results show that the CMS5 tends to predict [ ], with a mean difference of [ ] and a standard deviation of [ ]. In Table 5-18 of the TR, the applicant summarized the results of its calculated-to-measured values for individual pin fission rates. The applicant's results show that the largest deviation between the predicted-to-measured values is [ ] percent. Upon closer examination of the data, the NRC staff identified that, of the approximately [ ] included in the benchmark, only [ ] data points showed measured fission rates that were more than [ ] percent (the base value proposed by NuScale for radial peaking NRF) greater than the value predicted by CMS5. Furthermore, the NRC staff identified that all of these data points occurred in relatively low-power rods. Section 3.5 of this SE gives the NRC staff's evaluation of the impact of these benchmarking calculations on the establishment of the critical boron concentration (CBC) and radial peaking NRFs.

The applicant's benchmarking against the TMI-1 operating reactor data included calculated-to-measured values for CBC, DBW, ITC, power coefficient, CRA worth, and RAP. The applicant compared different cycles, burnups, and boron concentrations. Upon closer examination of the data, the NRC staff did not identify any adverse trend in the data (e.g., differences in predicted-to-measured values does not depend on burnup, cycle number, or boron concentration). The only potential outlier in the data identified by the NRC staff is the [ ]. Section 3.5 of this SE provides the NRC staff's evaluation of the impact of these benchmarking calculations on the development of the NRFs. Based on the applicant's benchmarking analyses for IPEN/MB-1, KRITZ-2, DIMPLE, and TMI-1 demonstrating results consistent with physical data, the NRC staff finds that the applicant has demonstrated technical competence with respect to execution of the CMS5 code suite, which is consistent with the expectations for licensees in accordance with Generic Letter 83-11, Supplement 1, "Licensee Qualification for Performing Safety Analyses," dated June 24, 1999 (Reference 24).

In addition to the analyses performed by the applicant, the NRC staff considered the validation documented by the code vendor, Studsvik. In particular, the NRC staff further investigated:

1. additional validation analysis of critical experiments performed by Studsvik in support of the CMS5 code suite that the NRC staff identified as applicable to the NuScale RXC;

2. validation analyses of depletion benchmarks performed by Studsvik in support of the CMS5 code suite; and

3. the basis for the generic NRFs developed by Studsvik.

In Section 2.3.1.1 of the CMS5 generic TR, Studsvik stated that CASMO5 was compared against 160 critical configurations in support of the generic CMS5 TR (Reference 5). The NRC staff identified the critical experiment validation analyses, performed by Studsvik, that address high leakage cores and heavy reflectors as being of particular interest for their applicability to the analysis of the NuScale RXC. Studsvik performed validation analysis using the Babcock & Wilcox (B&W) series 1484 critical experiments, described in more detail in Section 3.2.1.2 of the CASMO5 methods and validation report (Reference 18), to demonstrate that CMS5 is capable of accurately modeling core configurations that range from low to high leakage. Core I from the B&W 1484 test is a very high leakage core, with radial leakage representing nearly 35 percent of the total core reactivity. Core II from the B&W 1484 test is a relatively low leakage core, with the radial leakage representing roughly 15 percent of the total reactivity. For comparison to the NuScale RXC, the NRC staff performed confirmatory analyses of the NuScale RXC and determined that radial leakage represents [ ]. The results of the Studsvik analyses show that CASMO5 accurately predicted the eigenvalues for the Core I and Core II configurations.

Studsvik performed validation analyses using the tank-type critical assembly experiments, described in more detail in Section 3.2.1.5 of the CASMO5 methods and validation report (Reference 18) and by Tahara, Sekimoto, and Miyoshi (Reference 25), to demonstrate that CASMO5 can appropriately model reflectors of varying thicknesses and predict kinetics data. The results of the Studsvik analyses show that CASMO5 accurately captured the reactivity effect of the different reflector configurations. In addition, the Studsvik analyses show that the predicted kinetic values are consistent with the physical measurement within the experimental uncertainty.

Studsvik performed validation analyses using the Japan Atomic Energy Research Institute PWR isotopic benchmarks, described in more detail in Section 3.2.2.2 of the CASMO5 methods and validation report (Reference 18), to demonstrate that CASMO5 can accurately predict the isotopic changes of the fuel during depletion. The results of the Studsvik analyses show that CASMO5 can reasonably predict the isotopic burnup of PWR fuel, including fuel containing gadolinium. In addition, Studsvik has published information that demonstrates that the quadratic depletion model, used in CASMO5 for the depletion of gadolinium isotopes, can match the accuracy of the model used in CASMO-4 with larger time-steps (Reference 26). The gadolinium depletion capability of CASMO-4, which Studsvik used to benchmark the performance of the updated quadratic depletion model in CASMO5, has been demonstrated to accurately capture the depletion of gadolinium isotopes by performing validation studies for fuel assemblies with an initial Gd2O3 loading of 9 percent (References 27 and 28). The NRC staff notes that the Gd2O3 loading used in these validation studies [ ].

Section 4.0 of the generic CMS5 TR discusses the development of the generic NRFs (Reference 5). Studsvik used 63 cycles of measured plant data, obtained from seven PWR units located at four sites, to develop nuclear uncertainty factors. Studsvik used these uncertainty factors to select bounding NRFs. Previously, the NRC staff found these generic NRFs to be conservative in Section 3.9 of the NRC staff's SE for the generic CMS5 TR (Reference 20). For the purposes of this SE, the NRC staff considers the benchmarking against the 63 cycles of plant data to be further evidence of the CMS5 capability to model PWRs with similar characteristics to the NuScale RXC (e.g., fuel, control rod, and coolant geometry and material composition). In addition, the NRC staff considers the Studsvik discussion on generic NRF development in its evaluation of the applicable NRFs proposed for the NuScale RXC in Section 3.5 of this SE.

As described above, the applicant and code vendor have performed validation analyses that demonstrate that the CMS5 code suite is capable of modeling experiments and reactors with features representative of the NuScale RXC. Based on the results of these validation analyses, the NRC staff finds that the CMS5 code suite has been appropriately validated for use in performing analyses of the NuScale RXC.

3.5 Application Uncertainty

The NRC staff's evaluation of the application uncertainty includes:

1. an evaluation of the statistical methodology used to develop and update the base NRFs;
2. an evaluation of the treatment of reactor physics parameter uncertainty during operation; and
3. the detailed base NRF development and update methods for each specific NRF.

The NRC staff evaluated the statistical methodology used to develop and update NRFs to provide reasonable assurance that the reactor physics parameters used in the safety analyses are suitably conservative. The NRC staff evaluated the treatment of reactor physics parameters during operation to ensure that, should an unforeseen error in the prediction of a reactor physics parameter occur, the plant will be operated within the bounds of the safety analysis.

3.5.1 Statistical Methodology

Section 4.0 of the TR discusses the statistical methodology used in the development of the tolerance limits. This methodology begins with normality testing, described in Section 4.2 of the TR, which includes developing a histogram of the data, developing a normality plot, and conducting goodness-of-fit tests (Shapiro-Wilk normality test, D'Agonstino normality test, and Anderson-Darling normality test). If these tests determined that the data are normally distributed, the applicant determined the tolerance limit by adding the standard deviation times and appropriate tolerance factor to the distribution average. If these tests determined that the data cannot be treated as normally distributed, the applicant determined the tolerance limit using nonparametric statistics. The NRC staff compared the applicant's description of the methods to the existing literature and determined that the normality testing and nonparametric statistics are consistent with the existing literature (References 29 and 30), and the one-sided tolerance limit for a normal distribution is calculated consistent with the well-known and

previously accepted method described in Owen (Reference 31). Accordingly, the NRC staff finds the applicant's statistical methodology acceptable because the proposed statistical methods are consistent with established practice.

The applicant also discussed the NRF update methodology for the reactor physics parameters in Section 7.0 of the TR. The TR states that the NRFs are updated when a sufficient minimum number of measurements for acceptable statistics (a minimum of 10) are collected. The TR did not state the criteria for acceptable statistics or for establishing a minimum of 10 measurements. Accordingly, on May 8, 2017, the NRC staff issued RAI 8807, Question 29749, asking NuScale to provide the criteria (e.g., tolerance limit, confidence level) that are used to determine the NRF (Reference 12). In a letter dated July 6, 2017 (Reference 2), the applicant stated that NRFs will be updated to a 95/95 confidence/probability interval. The NRC staff finds the 95/95 confidence/probability interval acceptable because it is consistent with the standard review plan acceptance criteria (Reference 32).

3.5.2 Uncertainty Treatment during Plant Operation

Section 8.0 of the TR discusses the application of the nuclear analysis methodology. The TR did not describe how the uncertainties accounted for in the NRFs are translated into the appropriate Generic Technical Specifications (GTS) or other operational requirements. The NRC staff sought to ensure that unexpected uncertainties associated with reactor physics parameters would not result in operation of the NPM outside the bounds of the safety analysis. Accordingly, the NRC staff issued RAI 8807, Question 29752, asking NuScale to describe how each of the NRFs is translated into GTS or otherwise commit to preventing operation of the NPM outside the tolerance bounds of the NRFs (Reference 12).

In a letter dated July 6, 2017 (Reference 2), the applicant stated that the NRFs have been incorporated into the safety analysis and will be accounted for in determining appropriate operating limits. The applicant's response also included a table that described how each parameter is addressed and the action taken if a measurement exceeds the bounds of the NRF. This response clarified that, with a few exceptions (i.e., power coefficient, fuel temperature coefficient (FTC), and kinetics parameters), the NRFs associated with reactor physics parameters are addressed during startup testing and core follow procedures, and that the bounds for each NRF are captured either directly or indirectly through GTS. The applicant used analytically determined conservative values for the power coefficient, FTC, and kinetics parameters. The NRC staff recognizes that measuring the ITC includes both the MTC and the FTC (i.e., Doppler coefficient) (Reference 33). Additionally, the NRC staff recognizes that the kinetics parameters are checked during startup testing, which includes measurement of the reactor period. Therefore, the NRC staff finds that the power coefficient, FTC, and kinetics parameters are adequately verified during startup testing and GTS surveillance. Based on the verification of the NRFs through GTS surveillance and startup testing and the applicant's RAI response explaining that the safety analysis accounts for NRFs, the NRC staff finds the applicant's treatment of uncertainty in reactor physics parameters during operation acceptable.

3.5.3 Reactor Physics Parameter Nuclear Reliability Factors

3.5.3.1 Critical Boron Concentration

Section 7.1 of the TR discusses the development and update of the base NRFs for CBC. The applicant's base CBC NRFs are based on the operating benchmark for TMI-1, cycles 1 and 2. In addition, the applicant presented the industry standard (i.e., generic) CBC NRFs, which were developed by Studsvik and are based on 63 cycles of measured plant data, obtained from seven PWR units located at four sites (Reference 5). The NRC staff identified that the industry standard lower CBC NRF lies outside the bounds of the value proposed by NuScale. Accordingly, the NRC staff issued RAI 8807, Question 29754, asking NuScale to describe how the base CBC NRFs provide conservative bounds (Reference 12).

In letter dated July 6, 2017 (Reference 2), the applicant clarified that (1) the NuScale NRFs [ ] that NuScale observed in its benchmark of TMI-1, cycles 1 and 2, and (2) the NuScale NRFs actually account for a larger uncertainty than the industry standard NRFs. The NRC staff finds the applicant's response acceptable because it describes the reason the base NuScale NRFs appear nonlimiting. Additionally, the NRC staff compared proposed CBC NRFs with NuScale GTS (Reference 34) and determined that (1) the NuScale GTS require a reactivity balance that corresponds to uncertainty in the CBC of approximately [ ] parts per million and (2) this criteria for a reactivity balance is consistent with established practice (Reference 35). Based on the applicant's response to RAI 8807, Question 29754, and the fact that the base CBC NRFs provide a tighter tolerance than the NuScale GTS, the NRC staff finds the applicant's base CBC NRFs acceptable.

In addition to the base NRFs, the applicant described the CBC NRF update methodology, which includes taking measurements during startup and operation of the power plant. The applicant's description of the statistical methodology is consistent with the methodology evaluated in Section 3.5.1 of this SE. Based on the direct measurement of CBC, use of an acceptable statistical method (see Section 3.5.1 of this SE), and an acceptable treatment of uncertainty during plant operation (see Section 3.5.2 of this SE), the NRC staff finds the CBC NRF update methodology acceptable.

3.5.3.2 Differential Boron Worth

Section 7.2 of the TR discusses the development and update of the base NRFs for DBW. The applicant's base DBW NRFs are based on code-to-code benchmarking and the operating benchmark for TMI-1, cycles 1 and 2. The applicant's proposed NRFs account for more uncertainty than that observed in the code-to-code benchmarking and operating benchmark. Additionally, the base DBW NRFs provide for additional margin over the industry standard NRFs. Because the DBW NRFs bound the tolerance limits established by the benchmarking calculations and industry standard values, the NRC staff finds the base DBW NRFs acceptable.

In addition to the base NRFs, the applicant described the DBW NRF update methodology, which includes taking measurements at HZP during startup. The NRC staff recognizes that measuring the CBC at power includes the effect of DBW. The applicant's description of the statistical methodology is consistent with the methodology evaluated in Section 3.5.1 of this SE. Based on the direct measurement of the DBW at HZP, indirect measurement of the DBW at power, use of an acceptable statistical method (see Section 3.5.1 of this SE), and an acceptable treatment of uncertainty during plant operation (see Section 3.5.2 of this SE), the NRC staff finds the DBW NRF update methodology acceptable.

3.5.3.3 Isothermal Temperature Coefficient and Moderator Temperature Coefficient

Section 7.3 of the TR discusses the development and update of the base NRFs for ITC and MTC. The applicant's base ITC and MTC NRFs are based on code-to-code benchmarking and the operating benchmark for TMI-1, cycles 1 and 2. In addition, the applicant presented the industry standard NRFs, which were developed by Studsvik and are based on 63 cycles of measured plant data, obtained from seven PWR units located at four sites (Reference 5). The applicant's proposed NRFs account for more uncertainty than that observed in the code-to-code benchmarking and operating benchmark. Additionally, the base ITC and MTC NRFs provide for additional margin over the industry standard NRFs. Based on the ITC and MTC NRFs bounding the tolerance limits established by the benchmarking calculations and industry standard values, the NRC staff finds the base ITC and MTC NRFs acceptable.

In addition to the base NRFs, the applicant described the ITC and MTC NRF update methodology, which includes taking ITC measurements during HZP during startup, and that the updated ITC NRFs will be applied to the MTC NRFs. The NRC staff recognizes that a future combined license holder would perform additional MTC surveillances during power operation in accordance with the NuScale GTS (Reference 34). The applicant's description of the statistical methodology is consistent with the methodology evaluated in Section 3.5.1 of this SE. Based on the direct measurement of ITC and MTC, use of an acceptable statistical method (see Section 3.5.1 of this SE), and an acceptable treatment of uncertainty during plant operation (see Section 3.5.2 of this SE), the NRC staff finds the ITC and MTC NRF update methodology acceptable.

3.5.3.4 Power Coefficient and Fuel Temperature Coefficient

Section 7.4 of the TR discusses the development of the base NRFs for the power coefficient and FTC. The applicant's base power coefficient and FTC NRFs are informed by the operating benchmark for TMI-1, cycles 1 and 2, which included only two points. The applicant stated that the power coefficient is rarely measured and the operational data are very limited. Similarly, in Section 3.6.2 of the generic CMS5 TR (Reference 5), Studsvik stated that direct determination of an NRF for Doppler feedback remains very difficult. However, Studsvik's benchmarking of CASMO5 Doppler temperature defects to MCNP6 demonstrated that the CASMO5 methodology contributed little bias. Additionally, CMS5 benchmarking against operating plant transients has demonstrated the ability of CMS5 to accurately capture xenon transients (Reference 5), which demonstrates that xenon-Doppler balance is well predicted. The industry standard NRFs for the power coefficient and FTC are based on historical precedent and have been accepted previously by the NRC staff. In Section 7.4 of the TR, the applicant proposed power coefficient and FTC NRFs that increase the tolerance band by [ ] percent over the industry standard values. Based on the demonstrated ability to accurately predict xenon transients and proposed power coefficient and FTC NRFs that bound the currently accepted industry standard values and operating benchmark calculation, the NRC staff finds the base power coefficient and FTC NRFs acceptable.

3.5.3.5 Control Rod Assembly Bank Worth

Section 7.5 of the TR discusses the development and update of the base NRFs for CRA bank worth. The applicant stated that the base CRA bank worth NRFs are based on code-to-code benchmarking and the operating benchmark for TMI-1, cycles 1 and 2. In addition, the applicant

presented the industry standard CRA bank worth NRFs, which were developed by Studsvik and are based on 63 cycles of measured plant data, obtained from seven PWR units located at four sites (Reference 5). The applicant's proposed NRFs account for more uncertainty than that observed in the code-to-code benchmarking, but [ ]. However, the base CRA bank worth NRFs provide for additional margin over the industry standard NRFs, which are based on 340 observations from operating plants (Reference 5). Because the CRA worth NRFs bound the industry standard values, the NRC staff finds the base CRA worth NRFs acceptable.

In addition to the base NRFs, the applicant described the CRA bank worth NRF update methodology, which includes taking measurements during low-power physics testing. The applicant's description of the statistical methodology is consistent with the methodology evaluated in Section 3.5.1 of this SE. Based on the direct measurement of CRA bank worth, use of an acceptable statistical method (see Section 3.5.1 of this SE), and an acceptable treatment of uncertainty during plant operation (see Section 3.5.2 of this SE), the NRC staff finds the CRA bank worth NRF update methodology acceptable.

3.5.3.6 Assembly Radial Peaking

Section 7.6 of the TR discusses the development and update of the base NRF for RAP. The applicant proposed only an upper bound RAP NRF because the concern about assembly peaking uncertainty is the underprediction of assembly power. The applicant's proposed base RAP NRF is based on code-to-code benchmarking and the operating benchmark for TMI-1, cycles 1 and 2. In addition, the NRC staff compared the proposed base RAP NRF to the 2D integral reaction rate uncertainty factors developed by Studsvik in the CMS5 TR, which are based on 63 cycles of flux map data, obtained from seven PWR units located at four sites (Reference 5). The applicant's proposed NRFs account for more uncertainty than that observed in the code-to-code benchmarking and operating benchmark. Additionally, the proposed base RAP NRF provides additional margin over the 2D integral reaction rate uncertainty factor developed by Studsvik (Reference 5). Because the RAP NRF bounds the tolerance limit established by the benchmarking calculations and the value developed by Studsvik, the NRC staff finds the proposed base RAP NRF acceptable.

In addition to the base NRF, Section 7.6 of the TR describes the RAP NRF update methodology, which includes taking measurements at intermediate and full power. The applicant's description of the statistical methodology is consistent with the methodology evaluated in Section 3.5.1 of this SE. Based on the direct measurement of RAP, use of an acceptable statistical method (see Section 3.5.1 of this SE), and an acceptable treatment of uncertainty during plant operation (see Section 3.5.2 of this SE), the NRC staff finds the RAP NRF update methodology acceptable.

3.5.3.7 Pin Peaking

Section 7.7 of the TR discusses the development and update of the base NRFs for the pin peaking factors ($F_{\Delta H}$, and F_Q). The applicant proposed only an upper bound NRF because the concern about pin power peaking uncertainty is the underprediction of power. The applicant stated that the code-to-code benchmarks and empirical data are used collectively to develop the NRFs, but the empirical data are only presented as a demonstration of the fidelity of CMS5 in

predicting values related to pin power. The CMS5 TR contains more information on the development of pin peaking NRFs (Reference 5). The CMS5 TR identifies the following considerations for the development of the pin peaking NRFs:

- The $F_{\Delta H}$ NRF is a combination of the 2D integral reaction rate uncertainty and the pin-to-box bias.

- The F_Q NRF is a combination of a 3D reaction rate uncertainty, the pin-to-box bias, and (for fixed in-core detectors) an additional bias term.

- Studsvik calculated the pin-to-box bias using a combination of CASMO5 to experiment benchmarking calculations and CASMO5 to SIMULATE5 benchmarking calculations (to capture the uncertainty associated with pin power reconstruction). The CMS5 TR further clarifies that no mechanism is presented to allow users of the CMS5 code to change the pin-to-box bias in the CMS5 TR.

- Studsvik calculated the bias associated with fixed in-core detector systems by comparing movable in-core detector data to effectively collapsed nodal data. The CMS5 TR further clarifies that no mechanism is presented to allow users of the CMS5 code to change the fixed in-core detector bias in the CMS5 TR.

The NRC staff evaluated the applicability of the pin-to-box bias, developed by Studsvik, to the NuScale RXC. The NRC staff identified that the experimental basis used to develop the pin-to-box bias included test reactors (e.g., B&W 1810, DIMPLE, KRITZ3) with a range of core configurations that are similar to the NuScale RXC. Based on the range of core configurations included in the development of the pin-to-box bias and the similarity of these configurations to the NuScale RXC, the NRC staff finds that the pin-to-box bias developed by Studsvik in the CMS5 TR is applicable to the analysis of the NuScale RXC.

The NRC staff evaluated the applicability of the fixed in-core detector bias, developed by Studsvik, to the NuScale RXC. To calculate the bias, Studsvik assumed that in-core instrumentation is located at four axial positions, which is consistent with the NuScale in-core instrumentation system. The NRC staff identifies that the NuScale RXC is approximately half the height of the reactor cores for which the bias was developed; however, the normalized axial flux shape is consistent with those reactor cores. Based on the similarity in the relative axial power shapes and in-core detector locations, the NRC staff finds that the in-core detector bias developed by Studsvik in the CMS5 TR is applicable to the analysis of the NuScale RXC.

The applicant’s proposed pin peaking NRFs bound the code-to-code benchmarking calculations and industry standard values. The NRC staff also verified that the applicant’s proposed $F_{\Delta H}$ NRF bounds all but one of the fission rate benchmarks performed in Section 5.2.3.2 of the TR for relative pin powers above 50 percent. In addition, the NRC staff performed a confirmatory analysis, to include the pin-to-box bias and fixed in-core detector bias, that showed that the applicant’s proposed pin peaking NRFs are conservative. Based on the applicant’s proposal of conservative values, the NRC staff finds the base pin peaking NRFs acceptable.

In addition to the base NRFs, Section 7.7 of the TR describes the pin peaking NRF update methodology, which includes taking measurements at intermediate and full power. The applicant proposed the use of a pin-to-box bias, which it developed by performing code-to-code

(CMS5 to MCNP6) benchmarking. However, the applicant's proposed bias is less restrictive than the value developed by Studsvik. Additionally, the applicant did not address the bias associated with fixed in-core instrumentation. As described in the paragraphs above, the pin-to-box bias and in-core instrumentation bias developed by Studsvik are based on physical data, and the NRC staff found them to be applicable to the analysis of the NuScale RXC. Accordingly, the NRC staff established Condition 2 (see Section 4.0 of this SE), requiring any subsequent updates to the pin peaking factors to account for the pin-to-box bias and in-core instrumentation bias in accordance with the CMS5 TR. The NRC staff has previously approved the update of the pin peaking factors in accordance with the CMS5 TR (Reference 20).

3.5.3.8 Axial Offset

Section 7.8 of the TR discusses the development and update of the base NRFs for AO. The applicant's base AO NRFs are informed by code-to-code benchmarking but are based on engineering judgment. The NRC staff recognizes that AO is treated conservatively in the plant's safety analyses and is surveilled during operation in accordance with the NuScale GTS (Reference 34). Based on the conservative treatment of the AO in the safety analysis and verification of the AO in accordance with the NuScale GTS, the NRC staff finds the applicant's base AO NRFs acceptable.

In addition to the base NRFs, the applicant described the AO NRF update methodology, which includes taking measurements at intermediate and full power. The applicant's description of the statistical methodology is consistent with the methodology evaluated in Section 3.5.1 of this SE. Based on the direct measurement of AO, use of an acceptable statistical method (see Section 3.5.1 of this SE), and an acceptable treatment of uncertainty during plant operation (see Section 3.5.2 of this SE), the NRC staff finds the AO NRF update methodology acceptable.

3.5.3.9 Kinetics

Section 7.9 of the TR discusses the development and update of the base NRFs for the kinetics parameters (neutron lifetime, delayed neutron fractions, and delayed neutron precursor decay constants). The applicant selected the kinetics NRFs to be the same as the DBW NRFs and to be updated accordingly based on the premise that neutron lifetime is proportional to the soluble boron worth. The NRC staff questioned the underlying basis for stating that neutron lifetime is proportional to soluble boron worth. Accordingly, on May 8, 2017, the NRC staff issued RAI 8807, Question 29750, asking NuScale to provide evidence to show that the DBW NRF is applicable to the kinetics parameters (Reference 12).

In a letter dated July 6, 2017 (Reference 2), the applicant referenced an analysis showing that the neutron lifetime can be determined by perturbing the reactor with a $1/v$ absorber, and that the neutron lifetime is proportional to the associated reactivity perturbation. The NRC staff was able to reproduce these results by performing a confirmatory reactivity perturbation analysis. Based on the applicant's RAI response, the NRC staff finds the use of DBW NRFs acceptable for application to the neutron lifetime. Additionally, the applicant's RAI response referenced literature indicating that uncertainty in effective delayed neutron fraction is about 5 percent, and the delayed neutron yield is the primary contributor to the uncertainty (Reference 36). Based on the limitations of the physical data for delayed neutron yield, the NRC staff imposed Limitation 2 (see Section 4.0 of this report) to place a lower bound (of 5 percent) on the magnitude of any

updated delayed neutron parameter NRF. Because the base kinetics NRFs bound both the NRC staff's Limitation 2 and the previously accepted industry standard values, the NRC staff finds the base kinetics NRFs acceptable. Additionally, based on the discussion above and pursuant to Limitation 2, the NRC staff finds the kinetics NRF update methodology acceptable.

4.0 LIMITATIONS AND CONDITIONS

The NRC staff's conclusions about TR-0616-48793 are subject to the following limitations and conditions:

Limitation 1	Application of this TR is limited to the materials identified in the SE for the generic CMS5 methodology (Section 4.0 of Reference 20). Section 3.2.2 of this SE describes the basis for this limitation.
Limitation 2	Updates to any delayed neutron parameter NRF cannot reduce the magnitude of the NRF below 5 percent. Section 3.5.3.9 of this SE describes the basis for this limitation.
Condition 1	Generation of the case matrix for use in SIMULATE5 must be performed using the S5C option in CASMO5 (i.e., analyses are to be performed using the PWR segment S5C case matrix). Section 3.2.3 of this SE describes the basis for this condition.
Condition 2	Updates to the pin peaking NRFs ($F_{\Delta H}$, and F_Q) must include the pin-to-box bias and fixed in-core detector bias in accordance with the generic CMS5 TR (Reference 5). Section 3.5.3.7 of this SE describes the basis for this condition.

5.0 CONCLUSIONS

The NRC staff approves the use of NuScale TR-0616-48793, "Nuclear Analysis Codes and Methods Qualification," subject to the condition and limitations identified in Section 4.0 of this SE. In particular, the NRC staff finds that (1) the CMS5 code suite is applicable to the NuScale RXC design, subject to Limitation 1 and Condition 1, (2) the base NRFs proposed in TR-0616-48793 are acceptable, and (3) the NRF update methodology proposed by the applicant is acceptable, subject to Limitation 2 and Condition 2. These findings are based on the following six points:

1. The methodology presented in TR-0616-48793 is capable of modeling the geometry of the NuScale RXC (see Section 3.2.1 of this SE).

2. The methodology presented in TR-0616-48793 is capable of modeling the materials of the NuScale RXC (see Section 3.2.2 of this SE).

3. The methodology presented in TR-0616-48793 is capable of modeling the necessary physics of the NuScale RXC (see Section 3.2.3 of this SE).

4. The methodology presented in TR-0616-48793 has been adequately assessed against appropriate benchmarks (see Sections 3.3 and 3.4 of this SE).

5. The NRFs are developed and updated using appropriate data and statistical methods (see Sections 3.5.1 and 3.5.3 of this SE).

6. Uncertainty in reactor physics parameters during operation is managed appropriately (see Section 3.5.2 of this SE).

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