UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

June 15, 2018

Mr. Victor M. McCree
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: SAFETY EVALUATION OF THE NUSCALE POWER, LLC TOPICAL REPORT TR-0616-48793, REVISION 0, "NUCLEAR ANALYSIS CODES AND METHODS QUALIFICATION" AND SAFETY EVALUATION OF THE NUSCALE POWER, LLC TOPICAL REPORT TR-0116-21012, REVISION 1, "NUSCALE POWER CRITICAL HEAT FLUX CORRELATIONS"

Dear Mr. McCree:

During the 654th meeting of the Advisory Committee on Reactor Safeguards, June 6-7, 2018, we reviewed the NRC staff's safety evaluations for the NuScale Power, LLC (NuScale) topical reports TR-0616-48793, Revision 0, "Nuclear Analysis Codes and Methods Qualification," and TR-0116-21012, Revision 1, "NuScale Power Critical Heat Flux Correlations." Our NuScale Subcommittee also reviewed this matter during a meeting on May 15, 2018. During these meetings, we had the benefit of discussions with the staff and representatives of NuScale. We also had the benefit of the referenced documents.

CONCLUSIONS AND RECOMMENDATIONS

1. Topical report 0616-48793, Revision 0, provides an acceptable methodology for the design and steady-state analysis of the NuScale reactor core, subject to the staff's limitations and conditions. The staff safety evaluation should be issued.

2. Topical report 0116-21012, Revision 1, provides an acceptable basis for use of the NuScale critical heat flux correlations, NSP2 and NSP4 and associated critical heat flux ratio limits, within their range of applicability, subject to the staff's limitations. The staff safety evaluation should be issued.

BACKGROUND

Applicants use nuclear analysis and thermal-hydraulic methodologies to design and determine reactor core performance under steady-state and normal operations, as well as anticipated operational occurrences and accident conditions. Approved methods and correlations are used to generate reactor physics and thermal-hydraulic parameters for use in safety evaluations and to demonstrate compliance with applicable general design criteria in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

DISCUSSION

Nuclear Analysis Methods

NuScale submitted in August 2016, TR-0616-48793, "Nuclear Analysis Codes and Methods Qualification," as supplemented by further information in March and July 2017. This topical report describes the neutronic methods for design and steady-state analysis of the NuScale reactor core. The methodology includes the use of the Studsvik Scandpower, Inc. (Studsvik), Core Management Software, Version 5 (CMS5) suite of codes: CASMO5, CMSLINK5, and SIMULATE5. Studsvik previously submitted in 2015, a topical report on the generic application of CMS5 to pressurized water reactors (PWRs) that has been approved.

CASMO5, a two-dimensional transport theory code based on the method of characteristics, generates pin cell and assembly lattice physics parameters using cross section libraries based on the Evaluated Nuclear Data File (ENDF)/B-VII.0 or newer. The topical report describes, in general terms, the process followed to generate cross sections using CASMO5; however, all NuScale validations were performed using the Studsvik PWR case matrix, S5C, as recommended by the CASMO5 vendor. The staff will impose a condition requiring NuScale to use the S5C case matrix for all future applications of these methods. This safety evaluation condition is warranted.

CMSLINK5 processes the CASMO5 reactor physics results and creates a neutronic data library that is then passed to SIMULATE5. In SIMULATE5, a three-dimensional, steady-state, multi-group, nodal diffusion method is utilized to model the three-dimensional arrangement of the core. The code also accounts for burnup and depletion histories by using a hybrid microscopic-macroscopic cross section model. Geometric and material heterogeneities in the radial and axial directions are then treated explicitly in code noding selections. SIMULATE5 also includes a coupled, four-equation, thermal-hydraulic model for calculating power, coolant density, and fuel temperature distributions, and then iterates these solutions with the reactor physics model to determine reactor physics parameters. These include critical boron concentration, differential boron worth, isothermal temperature coefficient and moderator temperature coefficient, moderator density coefficient, power coefficient and fuel temperature coefficient, control rod assembly bank worth, relative assembly power, pin power peaking factors, axial offset, and kinetics parameters (neutron lifetime, delayed neutron fractions, and delayed neutron precursor decay constants).

The NuScale reactor core design is generally derivative of current PWR designs, albeit reduced in diameter (number of fuel assemblies) and reduced in height, hence power. The actual fuel assembly proposed for use in the NuScale reactor core, NuFuel-HTP2™, is consistent with existing deployed PWR fuel assembly designs. The staff examined application of these methods to the NuScale core design in treatment of axial zoning in control rod assemblies, effects of fixed in-core instrumentation, and modeling of the radial reflector unique to this design. The staff also questioned and examined multiple-module effects on neutron flux at the boundary of an individual module and found these negligible. In applying CMS5 to modeling of the reactor core, the staff found: 1) the axial nodalization applied by the applicant sufficient to deal with axial discontinuities such as control rod assembly positioning and axial material composition, 2) the impact of in-core instrumentation to be small and a negligible contribution to reactivity, and 3) the radial reflector modeling to be acceptable in capturing the neutronically distinct regions of the stainless steel-water reflector design.

Because no experimental or operating data are presently available that are fully representative of the NuScale reactor core configuration, code-to-code benchmarking is used to identify potential deficiencies in the CMS5 code suite, and to extend the range of benchmarking to materials and configurations that are not available in the experimental data. MCNP6, a high-fidelity Monte Carlo neutron-particle transport code, was used for reference comparisons. The simplified benchmarking comparisons were made at hot zero power and at hot full power over cycles 1 through 4, with cycle average burnups of 0 to 5,000 MWD/MTU. Based on the close agreement between CMS5 and MCNP results, the staff found that the code-to-code benchmarking did not identify any deficiencies for modeling the NuScale reactor core.

Code validation was performed against data obtained from critical experiments, experimental reactors, and an operating PWR (Three Mile Island Unit 1 (TMI-1)), as well as prior benchmarking by Studsvik. Empirical benchmarking results for the IPEN/MB-1 research reactor, and the KRITZ and DIMPLE experimental reactors, with similar geometrical and material features to the NuScale reactor core, showed good agreement in predicted effective neutron multiplication factor, k_{eff}. Benchmarking against TMI-1 data included calculated-to-measured values for key reactor physics parameters at different cycles, burnups, and boron concentrations. The staff did not identify any adverse trends in the predicted-to-measured values and determined that the applicant demonstrated technical competence with respect to execution of the CMS5 code suite.

Of note from prior CMS5 benchmarking for the generic topical report was the comparison of CASMO5 predicted results to data from Babcock and Wilcox series 1484 critical experiments, with high and low radial leakage cores, bounding that expected for the NuScale reactor core. The results showed prediction of eigenvalues for each of the two core configurations to be within the acceptable range. The generic CMS5 topical report also documents validation analyses to demonstrate appropriate modeling of varying reflector thicknesses and prediction of isotopic changes of fuel during depletion, and includes the development of nuclear uncertainty factors and generic nuclear reliability factors (NRFs) based on 63 cycles of measured plant data at seven PWRs.

The staff evaluated the application uncertainty, including the statistical methodology to develop and update the NuScale base NRFs, uncertainty treatment during plant operation, and the bases for updating base NRFs during startup testing and initial plant operations. The staff determined that the methodology to update each of the key NRFs is acceptable. We agree with the staff conclusions regarding acceptability of the methods and their application to the NuScale design as supported by the code benchmarking approach, the experimental data benchmarks, and the NRFs development. The uncertainty bounds for these methods have been determined for the NuScale application to be conservative in comparison with those developed with the generic PWR datasets. This is appropriate given the absence of specific operational experience with the NuScale design. It is expected that the ranges in NRF values will be amended when startup and operational experience is obtained.

The staff approved the use of TR-0616-48793, subject to the following limitations and condition:

> Limitation 1: Application of this topical report is limited to the materials identified in the safety evaluation for the generic CMS5 methodology.
>
> Limitation 2: Updates to any delayed neutron parameter NRF cannot reduce the magnitude of the NRF below five percent.

Condition 1: Updates to pin peaking NRFs must include pin-to-box bias and fixed in-core detector bias in accordance with the generic CMS5 topical report.

The staff review of the subject topical report was thorough and complete. The staff safety evaluation report should be issued.

We further note that the NuScale transient methodology is based on a point kinetics approximation. SIMULATE5 calculates effective coefficients for the transient calculation, such as differential boron worth, moderator density coefficient, and moderator and fuel temperature coefficients, by performing small linear perturbations around the steady state solution. However, for transients that significantly depart from this initial condition, that may occur at conditions of low pressure and higher void fraction, these coefficients may vary significantly, and a table of coefficient values as a function of transient conditions may be needed. NuScale's transient analysis methodology will be reviewed by the staff. Code validation in regimes more associated with boiling water reactor-like operating conditions may be appropriate to verify conservatism of key NRFs.

Critical Heat Flux Correlations

NuScale submitted in November 2017, TR-0116-21012, Revision 1, "Critical Heat Flux Correlations." This revision replaced an earlier submittal, "NuScale Power Critical Heat Flux Correlation NSP2," to implement an additional critical heat flux (CHF) correlation NSP4. The purpose of this topical report is to provide the bases for use of the NSP2 and NSP4 correlations in a subchannel analysis code VIPRE-01, within their range of applicability, along with associated critical heat flux ratio (CHFR) limits for the NuScale design certification application and safety analyses using NuFuel-HTP2™. An approved CHF correlation is used to demonstrate that specified acceptable fuel design limits are not exceeded during normal operations, including the effects of anticipated operational occurences, and to determine the number of fuel failures associated with CHF that need to be included in estimating radiological consequences for postulated accidents.

The staff conducted its technical evaluation of this topical report against the "Critical Boiling Transition Model Assessment Framework," which was developed by the staff and first documented in an appendix to the safety evaluation. Critical boiling transition here is defined by the staff as a transition from a flow regime of high heat transfer rate to one of significantly lower heat transfer rate, and the terminology is meant to describe phenomena including CHF, departure from nucleate boiling, and dryout. The framework, informed by engineering judgment and significant past experience in evaluating such correlations, is based on applying a top-down set of goals to demonstrate that the critical boiling transition model can be trusted in reactor safety analyses, primarily: 1) the experimental data support the model, 2) the model was generated in a logical fashion, and 3) sufficient validation has been demonstrated with appropriate uncertainty quantification.

CHF Testing

The CHF testing for development of the correlations was conducted at Stern Laboratories in Hamilton, Ontario, Canada and at the KArlstein Thermal HYdraulic test loop (KATHY) in Karlstein, Germany. The staff conducted an inspection of the Stern facility and an audit of the KATHY facility to address quality assurance matters related to data generation and collection. Conditions prototypic of those expected in the NuScale reactor core, including axial power profiles, were tested. The applicant used a non-uniform flux factor to adjust CHF predictions for the effect of variations of axial power profiles and demonstrated that this factor adequately adjusts data obtained for a uniform power profile to non-uniform shapes. Legacy data from Stern test sections using preliminary fuel designs were used to develop the NSP1 CHF correlation. Legacy data from KATHY using a test section with HMP™ grids were used to develop the NSPX factor as applied to the NSP1 correlation in developing the NSP2 CHF correlation. This dataset was generated in test sections not fully representative of the prototypical NuScale design. For the prototypical testing at KATHY, Series K-9000-K9300, a 5x5 bundle, with prototypical dimensions and prototypical spacer grids and axial positions, was used to simulate NuFuel-HTP2™. Pin power peaking was used to produce peak heat flux values on interior rods to negate wall effects and produce an equivalent hot or limiting subchannel at the interior of the bundle. The dataset from this test series was subsequently used to validate the NSP2 and NSP4 CHF correlations.

CHF Model Development and Validation

The NSP2 and NSP4 correlations include pressure, mass flux, equilibrium quality, boiling length, and hydraulic diameter in its basic form. This approach is consistent with previously approved CHF correlations. The formulations selected, based on global sensitivity analyses, were demonstrated to produce consistency with known physical trends.

The final NSP2 CHF correlation introduces an NSPX factor to adjust the base correlation for analyzing the thermal performance of NuFuel-HTP2™ fuel. As mentioned previously, the base correlation was developed with data from a non-prototypical geometry (i.e., spacer grid); the factor introduces a mean predicted-to-measured ratio of less than one when applying the NSP2 CHF correlation to K8500 data. Based on the conservative development of this factor and its use in the validation process over the correlation's range of applicability, the staff found the NSPX factor acceptable. The non-uniform flux factor, based on the original Tong factor, is applied to adjust the NSP2 CHF predictions for the effect of variations of axial power profiles. The applicant placed an additional multiplier on the original Tong factor based on Stern test data. The effect is to reduce the predicted non-uniform CHF. The staff put a limitation on application of this non-uniform flux factor used in the NSP2 and NSP4 CHF correlation that it always be greater than or equal to one.

The final NSP4 CHF correlation was developed using data from experiments that are prototypical of the NuFuel-HTP2™. For the non-uniform flux factor, the applicant used data obtained from these experiments (i.e., KATHY K9000-K9300). The correlation validation process verified adequate performance over its intended range of applicability.

To develop coefficients for the above models the applicant used the statistical technique of randomly apportioning data into *k* subsets, then held one subset for validation testing and used the remaining data to "train" the correlation (i.e., obtain the correlation coefficients). The process is then repeated k times such that all the data have been used in validation. This process is appropriate because it allows the correlation coefficients to be obtained from a broader dataset than used in the final validation.

The final model validation data were obtained from the tests using the test bundle prototypic of NuFuel-HTP2™ (i.e., KATHY K9000-K9300). This required application of the VIPRE-01 subchannel analysis code to determine local conditions at CHF. Model predictions of expected mass flux and pressure regions are well covered for both correlations. Model predictions for low mass flux and high quality regions show predicted-to-measured data points exceeding CHFR limits, hence the range of applicability for each correlation is limited accordingly. The CHFR 95/95 confidence limits were set conservatively by NuScale at 1.17 for the NSP2 CHF correlation, and 1.21 for the NSP4. The staff found these acceptable.

NSP2 and NSP4 CHF Correlation Use

The VIPRE-01 subchannel analysis code was used to perform the data reduction calculations in developing the NSP2 and NSP4 correlations. The staff has imposed a second limitation that analyses using these correlations must be performed in accordance with TR-0915-17564, Revision 1, e.g., the models are expected to be implemented in the VIPRE-01 code.

The staff safety evaluation should be issued.

SUMMARY

Topical report 0616-48793, Revision 0, provides an acceptable methodology for the design and steady-state analysis of the NuScale reactor core, subject to the staff's limitations and conditions. The staff safety evaluation should be issued.

Topical report 0116-21012, Revision 1, provides an acceptable basis for use of the NuScale CHF correlations, NSP2 and NSP4 and associated CHFR limits, within their range of applicability, subject to the staff's limitations. The staff safety evaluation should be issued.

The staff review of the subject topical report against its critical boiling transition framework was comprehensive and complete. This critical boiling transition framework should be published as a public document, as part of the NRC's Knowledge Management Program.

Members Rempe and Riccardella did not participate.

Sincerely,

/RA/

Michael L. Corradini
Chairman

REFERENCES

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2. NuScale Power, "NuScale Power, LLC Submittal of Topical Report TR-0616-48793, 'Nuclear Analysis Codes and Methods Qualification,' Revision 0 (NRC Project No. 0769)," August 30, 2016 (ML16243A517).

3. U.S. Nuclear Regulatory Commission, "U.S. Nuclear Regulatory Commission Safety Evaluation by the Office of New Reactors Topical Report - 0116-21012, Revision 1 'NuScale Power Critical Heat Flux Correlations' NuScale Power, LLC (Project No. PROJ0769)," Draft, April 2018 (ML18100A873).

4. NuScale Power, "NuScale Power, LLC, Submittal of Topical Report 'Critical Heat Flux Correlations,' TR-0116-21012, Revision 1," November 30, 2017 (ML17335A089) .

5. NuScale Power, "NuScale Power, LLC Supplemental Information in Support of NuScale Topical Reports (NRC Project No. 0769)," March 7, 2017 (ML17066A463).

6. Studsvik, SSP-14/P01-028-TR-NP, "Generic Application of the Studsvik Scandpower Core Management System to Pressurized Water Reactors," December 2015 (ML15355A285).

7. NuScale Power, "NuScale Power, LLC Proprietary Marking Changes to 'Subchannel Analysis Methodology' topical report, Revision 1 (NRC Project No. 0769)," February 15, 2017 (ML17046A333).

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1. U.S. Nuclear Regulatory Commission, "U.S. Nuclear Regulatory Commission Safety Evaluation by the Office of New Reactors Topical Report TR-0616-48793, Revis 0, 'Nuclear Analysis Codes and Methods Qualification,' NuScale Power, LLC, Project No. 0769," Draft, February 2018 (ML18033A779).

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