U.S. NUCLEAR REGULATORY COMMISSION

SAFETY EVALUATION REPORT BY THE OFFICE OF NEW REACTORS

TOPICAL REPORT TR-0915-17564, "SUBCHANNEL ANALAYSIS METHODOLOGY,"

REVISION 2, NUSCALE POWER, LLC

DOCKET NO. 52-048

1.0 INTRODUCTION

By letter dated February 15, 2017, NuScale Power, LLC (NuScale or the applicant), submitted Topical Report (TR)-0915-17564-P, "Subchannel Analysis Methodology," Revision 1, to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval in support of the application for the design certification of NuScale's small modular reactor. The NRC issued an acceptance letter on March 20, 2017.

The NuScale subchannel analysis methodology (NSAM) uses an NRC-approved thermal-hydraulic computer code (VIPRE-01) for reactor cores. For pressurized-water reactors (PWRs), VIPRE-01 is typically used for the departure from nucleate boiling ratio (DNBR), fuel and cladding temperatures, and reactor coolant flow and temperature calculations during both normal and off-normal conditions. Two VIPRE-01 code versions, MOD-01 and MOD-02, are available and were approved by the NRC in safety evaluation reports (SERs). NuScale plans to use the MOD-02 version of VIPRE-01 for steady-state and transient subchannel calculations for the design certification application (DCA) and for future design analyses such as those in Chapters 4 and 15 of the NuScale final safety analysis report (FSAR). In TR-0915-17564-P, NuScale uses the NSP2 critical heat flux (CHF) correlation as an example correlation for the VIPRE-01 safety analysis of the NuScale power module (NPM) with NuFuel-HTP2™ fuel, and has submitted a separate TR on the NSP2 CHF correlation (Ref. 6) for NRC review and approval. A separately issued SER will evaluate the applicability of the NSP2 CHF correlation to the NPM with NuFuel-HTP2™ fuel. However, the evaluation of this TR is independent of the use of the NSP2 CHF correlation in TR-0915-17564-P.

By letter dated October 31, 2018, NuScale submitted Revision 2, TR-0915-17564, "Subchannel Analysis Methodology," which incorporate changes from request for additional information (RAI) responses.

This SER is divided into eight sections. Section 1 is the introduction; Section 2 summarizes the information presented in the TR; Section 3 summarizes the applicable regulatory criteria; Section 4 contains the technical evaluation of TR-0915-17564-P, Revision 1; Section 5 presents the limitations and conditions; Section 6 presents the conclusions of this review; Section 7 lists the references; and Section 8 defines the acronyms used in the document.

2.0 SUMMARY OF THE TOPICAL REPORT

The applicant asked the NRC to review and approve the subchannel analysis methodology described in TR-0915-17564-P for the NuScale thermal-hydraulic design and reactor safety analyses. The objective of TR-0915-17564-P is to describe the NuScale subchannel analysis

assumptions, codes, and methodologies and to justify them as technically acceptable. TR-0915-17564-P discusses how NuScale meets the NRC's requirements for use of the VIPRE-01 code based on the modeling methodology for performing steady-state and transient subchannel analyses and the qualification of the code for application to the NuScale design. Specifically, NuScale asked for NRC approval of TR-0915-17564-P for the following four elements, as stated in the TR:

1. VIPRE-01 applies to the NuScale steady-state and transient subchannel analysis using the methodology presented.

2. The NuScale methodology fulfills the NRC's requirements in the SERs for VIPRE-01, MOD-01 and MOD-02.

3. The methodology presented is independent of any specific CHF correlation and is used for NuScale applications if the methodology requirements are satisfied and if the NRC approves the CHF correlation.

4. The methodology for treatment of uncertainties in the NuScale subchannel methodology is appropriate.

For the given geometry of the NPM reactor core and coolant channels and the boundary conditions, VIPRE-01 calculates core flow distributions, coolant conditions, fuel rod temperature, and the minimum critical heat flux ratio (MCHFR) for steady-state and certain operational transients and abnormal events. The calculations of the reactor system code, NRELAP5, provide the specified boundary conditions. This SER documents the NRC staff's review of the NSAM, which uses VIPRE-01. This subchannel analysis methodology is used to calculate margin to fuel thermal limits, such as critical heat flux ratio (CHFR) and fuel centerline temperature. The NSAM TR provides the methodology for performing a subchannel analysis, and NuScale stated that the TR-0915-17564 does not provide final detailed reactor core design or final values of any other associated accident evaluations. Further, NuScale stated that the results of the subchannel analysis presented in the TR-0915-17564 demonstrate the analytical methodology to provide an understanding of the context of the application; however, NuScale is not seeking NRC approval of these results as part of the present SER.

3.0 REGULATORY BASIS

Under Title 10 of the *Code of Federal Regulations* (CFR), Section 52.47, "Contents of Applications; Technical Information," and 10 CFR 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report," the NRC requires an FSAR to analyze the design and performance of the structures, systems, and components (SSCs). The NRC staff evaluates the FSAR to determine whether it meets the NRC regulatory requirements for thermal margin under normal operations, including anticipated operational occurrences (AOOs), and accident conditions. These safety evaluations use an approved subchannel analysis methodology to establish a partial basis for demonstrating compliance with the applicable General Design Criteria (GDC) in 10 CFR Part 50, "Domestic Licensing of Production and Utilization of Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants."

"Reactor Design," GDC 10, requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to provide assurance that specified acceptable fuel design limits are not exceeded during any condition of normal

operation, including the effects of AOOs. GDC 10 is relevant to the DNBR because it is used to establish safety-related margins for the fuel and cladding integrity. To ensure compliance with GDC 10 per NuScale DSRS Section 4.4, the NRC staff needs to confirm that the thermal-hydraulic design of the core and the reactor coolant system (1) is accomplished using acceptable analytical methods, (2) is equivalent to or is a justified extrapolation from proven designs, (3) provides adequate margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability.

The regulations in 10 CFR 50.34, "Contents of Applications; Technical Information," apply to transient and accident analysis methods and require each plant to include the analysis of transients and accidents in its FSAR. The regulations in 10 CFR 50.34 require licensees to submit SARs that analyze the design and performance of SSCs that are important for the prevention of accidents and the mitigation of consequences of accidents. As part of the core reload design process, licensees are responsible for reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle. To confirm that the analyses are bounding, licensees confirm that those key inputs to the safety analyses (e.g., the MCHFR) are conservative with respect to the design cycle. Additionally, because the results of the transient and accident analysis methods are important to the safety of nuclear power plants, these methods must be maintained under a quality assurance program that meets the criteria in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

NuScale DSRS Section 4.4 stipulates the NRC staff's review process for thermal and hydraulic design applications. One acceptance criterion specified in DSRS Section 4.4 for the evaluation of fuel design limits is to ensure that the hot fuel rod in the core does not experience departure from nucleate boiling during normal operation or AOOs. To accomplish this, the licensee must address the uncertainties in the values of process parameters, core design parameters, calculation methods, and instrumentation in the assessment of thermal margin. According to Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," to SRP Section 4.2, "Fuel System Design," fuel cladding failure is presumed if local heat flux exceeds the thermal design limits.

The scope of the NRC staff's review addresses the applicability of the NSAM methodology and VIPRE-01, MOD-02, to the NuScale NPM design. SRP Section 15.0.2, "Review of Transient and Accident Analysis Method," describes six areas that the NRC staff reviews for transient and accident analysis methods: documentation, the evaluation model, the accident scenario identification process, code assessment, the uncertainty analysis, and the quality assurance program. The NRC staff based its review of VIPRE-01 on the SRP guidance for these six areas, an evaluation of the technical merit of the submittal, and the applicable regulations.

4.0 <u>TECHNICAL EVALUATION</u>

The purpose of the applicant's TR on subchannel analysis methodology is to conform to the regulatory requirements identified in the SERs for VIPRE-01, MOD-01 and MOD-02 (Refs. 4 and 5). The TR provides the supporting documentation and justification for its specific modeling assumptions, choice of models/correlations, and input values of plant-specific data used with VIPRE-01 for the NuScale design licensing calculations. The staff reviewed the applicant's TR and the supporting documentation through an audit (Ref. 7) to review the adequacy of the information for making safety findings. The technical evaluation below documents the staff's

basis for making the safety findings.

In May 1986, the NRC generically approved VIPRE-01, MOD-01, for referencing in license applications to the extent specified and under the limitations delineated in the licensing TR and the associated NRC SER for VIPRE-01, MOD-01 (Ref. 4). Section 4.1 of this SER describes the conditions for use delineated in the NRC's SER for VIPRE-01, MOD-01. In October 1993, the NRC approved the use of VIPRE-01, MOD-02, for referencing in license applications to the extent specified and under the limitations delineated in the licensing TR and the associated NRC SER for VIPRE-01, MOD-02 (Ref. 5). Section 4.2 of this SER describes the conditions for use delineated in the NRC SER for VIPRE-01, MOD-02. In accordance with the subsequent NRC generic SER for VIPRE-01, MOD-02, the limitations on use of VIPRE-01, MOD-01, remain applicable to the PWR applications of VIPRE-01, MOD-02 (Ref. 5). Sections 4.1 and 4.2 of this SER describe the staff's evaluation of the fulfillment of each of the conditions for use of VIPRE-01 by the NuScale methodology.

4.1 VIPRE-01, MOD-01, Safety Evaluation Report Conditions

The sections below list the conditions for use delineated in the NRC SER for VIPRE-01 and the staff's evaluation relative to each condition.

1. **Post-CHF Application Limitation** The application of VIPRE-01 is limited to PWR licensing calculations with heat transfer regime up to CHF. Any use of VIPRE-01 in boiling-water reactor (BWR) calculations or post-CHF calculations will require prior NRC review and approval.

Even though this post-CHF condition of limiting the VIPRE-01 code to PWR licensing applications for heat transfer regimes up to the point of CHF was imposed on MOD-01 of VIPRE-01, in accordance with the subsequent NRC generic SER for VIPRE-01, MOD-02, it also applies to the PWR applications of VIPRE-01, MOD-02. In this regard, TR-0915-17564-P states that NuScale does not seek approval for the use of VIPRE-01, MOD-02, in the NSAM for post-CHF calculations. Therefore, the NRC staff concludes that TR-0915-17564-P has provided the necessary information and that the NSAM meets the post-CHF application limitation on the use of the VIPRE-01 computer code for the NuScale design.

2. **CHF Correlation Requirement** Use of a steady state CHF correlation with VIPRE-01 is acceptable for reactor transient analysis provided that the CHF correlation and its DNBR limit have been reviewed and approved by NRC and that the application is within the range of applicability of the correlation including fuel assembly geometry, spacer grid design, pressure, coolant mass velocity, quality, etc. Use of any CHF correlation that has not been approved will require the submittal of a separate topical report for staff review and approval. The use of a CHF correlation that has been previously approved for application in connection with another thermal hydraulic code other than VIPRE-01 will require an analysis showing that, given the correlation data base, VIPRE-01 gives the same or a conservative safety limit, or a new higher DNBR limit must be used, based on the analysis results.

A key objective of the NSAM is to use the VIPRE-01 code to determine the MCHFR for the NuScale design. For this reason, NuScale used VIPRE-01 to reduce the experimental CHF data to calculate the limiting local thermal-hydraulic parameters that were used to develop the

NSP2 CHF correlation. The NSP2 CHF correlation was developed by fitting CHF values against local fluid parameters obtained using VIPRE-01 analyses. The resulting NSP2 CHF correlation is used with VIPRE-01 to calculate event-specific CHFR values and limits. In alignment with the CHF correlation requirement cited above, NuScale submitted a separate TR for NRC review and approval of the use of the NSP2 CHF correlation in VIPRE-01 for the NuScale DCA and safety analysis of normal operation and AOOs in the NPM with NuFuel-HTP2™ fuel. The applicant submitted a separate TR to address the use of design-specific CHF correlations. However, application of this TR to the safety analysis of the NPM must be performed using an approved CHF correlation. Accordingly, NRC staff created Condition 1 requiring the use of an approved CHF correlation when this TR is referenced in the safety analysis. Based on the submittal of a separate TR and pursuant to Condition 1 in Section 5.0 of this SER, NRC staff finds that the applicant satisfies Condition 2 above.

Section 2.2 of Volume 5 of the VIPRE-01 manual identifies a spectrum of VIPRE code limitations. Condition 3 in the SER for VIPRE-01, MOD-02, stipulates that each user should ensure that the code is not being used in violation of these limitations. Section 2.2 of the VIPRE-01 manual states that the VIPRE code should not be applied to situations that entail conditions such as low-flow boiloff, annular flow, phase separation that involves a sharp liquid/vapor interface, or countercurrent flow. Furthermore, Section 2.2 of Volume 5 of the VIPRE-01 manual identifies another VIPRE-01 limitation that arises from the omission of several cross-coupling terms from the lateral momentum equation that leads the code to accurately predict the flow field only when wall friction is significant and lateral flow resistance is fairly large compared to axial flow resistance.

Table 3-1 of the TR shows the parameter ranges used to demonstrate the applicability of the NSAM with the example NSP2 CHF correlation. The example NuScale normal/off-normal parameter ranges are 11.7 – 15.2 MPa (1,700–2,200 pounds per square inch absolute (psia)) for pressure, 136 – 678 kg/s-m^2 (0.1–0.5 million pounds mass per hour per square foot (Mlbm/hr-ft^2)) for local coolant mass flux, and less than 20 percent for the local equilibrium quality. The NRC staff was concerned that all example ranges chosen in the NSAM TR to demonstrate VIPRE application are narrower than the corresponding NSP2 CHF correlation applicability ranges of 2.0 – 15.9 MPa (300–2,300 psia) for pressure, 150 – 950 kg/s-m^2 (0.11 - 0.70 Mlbm/hr-ft^2) for local coolant mass flux, and less than 95 percent for the local equilibrium quality, as listed in Table 7-2 of the NSP2 CHF correlation TR (Ref. 6). For instance, the example local equilibrium quality used in the NSAM TR is up to only 20 percent, whereas the NSP2 CHF correlation limit for local equilibrium quality is up to 95 percent. Likewise, the example lower limit of pressure used in the NSAM TR is 1,700 psia, which is significantly higher than the 300-psia lower limit of the NSP2 CHF correlation. The NRC staff needed to evaluate the applicability of NSAM for the safety of the NuScale design over the full range of the NSP2 CHF correlation application, as VIPRE-01 was used for the development of the NSP2 CHF correlation and the VIPRE-01 based NSAM is expected to be approved for the entire NSP2 CHF correlation range. Therefore, the NRC staff issued Request for Additional Information (RAI) No. 9080, Question 04.04-12, dated September 11, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17254A439), to ask the applicant to justify the applicability of VIPRE-01 over the spectrum of transients and two-phase phenomena involving the complete NSP2 CHF correlation range. The applicant needed to demonstrate that the NuScale application of VIPRE would not violate the code's limitations as documented in Section 2.2 of Volume 5 of the VIPRE-01 manual throughout the

NSP2 CHF correlation's full range. The NRC staff required the additional information to make a safety finding on the overall applicability of the NSAM to the NuScale core safety design.

In its response to RAI No. 9080, Question 04.04-12, dated November 9, 2017 (ADAMS Accession No. ML17313B205), the applicant explained that Table 3-1 of the TR presents the expected parameter ranges for application of the VIPRE-01 code as part of the NSAM. The table was intended to show that the applicability range of the NSP2 CHF correlation encompasses the range of off-normal core conditions of the reactor over which the NSAM is expected to apply. For example, although the CHF correlation remains valid up to 95-percent local quality, the NuScale core is not expected to operate at more than 20-percent quality. However, in a teleconference on February 7, 2018, NuScale stated that the ranges in the last column of Table 3-1 bound the actual safety analyses presented in the DCA. The NRC staff asked the applicant to clarify in its response whether the values in the last column of Table 3-1 of the TR are the actual normal/off-normal ranges of the applicability of NSAM to the NuScale design and do not bound the applicability of NSAM to the entire CHF correlation range. In its supplemental response to RAI No. 9080, Question 04.04-12, dated March 2, 2018 (ADAMS Accession No. ML18061A109), the applicant stated that Table 3-1 of the NSAM TR presents the pressure, local mass flux, and local equilibrium quality parameter ranges in the last column to demonstrate that the conditions expected to be analyzed fall well within the applicability range of the NSP2 CHF correlation. The values presented in Table 3-1 are representative of the actual safety analyses presented in the DCA and do not imply or implement any application limitations for the NSAM or the VIPRE-01 code.

In its November 9, 2017, response to RAI No. 9080, the applicant stated that "in addition to the Subchannel Analysis Methodology, NuScale uses the VIPRE-01 code for evaluation of local conditions in support of the development of the CHF correlations" (Ref. 6). This statement implies that NuScale distinguished the use of the VIPRE-01 code in the NSAM from its use for the development of the CHF correlation. However, in the February 7, 2018, conference call, the applicant clarified that it does not differentiate between the VIPRE-01 code applications for NuScale safety analyses and the CHF correlation development; instead, it considers use of the VIPRE-01 code with the NSAM to be applicable for the entire range of the CHF correlation. In its March 2, 2018, supplemental response to RAI No. 9080, the applicant stated that it did not intend to differentiate separate applications of VIPRE-01 in its quoted statement. Section 3.3 of the NSAM TR lists the NSAM applicability criteria, one of which is to use the CHF correlation within its applicable parameter ranges.

In its response to RAI No. 9080, Question 04.04-12, the applicant also explained the NSAM use of the CHF correlation as an additional two-phase flow closure model that ties the VIPRE-01 CHF prediction to the measured CHF test data at the 95/95 level (i.e., 95-percent probability at the 95-percent confidence level). A separate NRC staff SER (Ref. 8) found the CHF correlation acceptable because its predictions are reasonable across a broad range of CHF parameters (e.g., mass flux, pressure, and quality). The applicant extended the fluid conditions during CHF testing to conditions (e.g., high equilibrium quality) that are not expected in any NuScale operation or off-normal events but that are needed to establish the CHF trends and limits for NuScale. In its RAI No. 9080 response, the applicant stated that the use of the VIPRE-01 code options and closure relationships in the NSAM are consistent with those used to evaluate the CHF test local conditions. In its response, the applicant also stated that Section 5.4 of the NSAM TR evaluates several combinations of closure models to assess the impact of using different models on local fluid conditions and the MCHFR. The NRC staff's review concluded that Table 5-3 of the TR demonstrates limited sensitivity of the closure model combinations on

the MCHFR evaluation using the NSAM. Moreover, Section 5.8.2 of the NSAM TR describes benchmarking of VIPRE-01 with AREVA's COBRA-FLX subchannel code for NuScale reactor transients. The comparisons were based on different sets of two-phase flow closure models that affected the predicted local conditions and showed that, when using the same CHF correlation, the CHFRs predicted by two different subchannel codes demonstrate reasonable agreement. The NRC staff concludes that the code-to-code benchmarking provided an additional technical basis to holistically demonstrate the applicability of VIPRE-01 to the safety analyses of the NuScale design.

The RAI No. 9080 response also stated that, in the revaluation of the same CHF test database with conservative homogeneous equilibrium models with no subcooled boiling or two-phase friction effects, the different resulting local conditions predicted by VIPRE-01 led to an inconsequential change in the CHF correlation 95/95 design limit. During the February 7, 2018, teleconference, the NRC staff asked the applicant to elaborate on why it considered the difference in the CHFR limit to be inconsequential. This way the applicant could address the staff's concern in the original RAI about using VIPRE-01 at a very high quality. In its supplemental response to RAI No. 9080, Question 04.04-12, dated March 2, 2018 (ADAMS Accession No. ML18061A109), the applicant also explained that, as documented in the CHF TR, VIPRE-01 local conditions are generated with the Electric Power Research Institute (EPRI) closure model correlations for the subcooled boiling, bulk void, and two-phase friction multiplier. In its November 9, 2017, RAI response, NuScale had quantified the effect of these correlations on the 95/95 design limit. NuScale regenerated the VIPRE-01 CHF testing under local conditions, assuming no subcooled boiling, homogeneous bulk voiding, and homogeneous two-phase friction multiplier correlations. This assumption, which essentially treats the fluid in a subchannel as perfectly mixed and in equilibrium, is conservative because subcooled boiling or two-phase friction provides no benefit. The 95/95 CHFR limit that resulted from this assumption, which accounts for statistical uncertainties, including subregions with higher sensitivities, was similar to the value obtained using the EPRI closure models. In design applications for steady-state and transient analyses, which are characterized by qualities that are much lower than those in the CHF testing database, this effect is even less than the calculated difference in CHFR, as documented in Section 5.4 of the NSAM TR. Therefore, the NRC staff agrees that the CHFR difference observed in the sensitivity study is a conservative upper limit of this effect and demonstrates that the VIPRE-01 two-phase correlations used for the NuScale applications, as presented in the NSAM TR, are appropriate.

RAI No. 9080 communicated the NRC staff's concern that VIPRE-01's NuScale applications, as documented in the subchannel methodology TR, are for much narrower ranges than the full range of the NuScale NSP2 CHF correlation. The NRC staff sought justification for using VIPRE-01 in the NSAM for the entire CHF correlation range. The applicant's supplemental response to RAI No. 9080, Question 04.04-12, dated March 2, 2018 (ADAMS Accession No. ML18061A109), explained that, as the CHF correlation acts as a closure relation between the VIPRE-01 predicted local conditions and test data, its 95/95 limit captures all uncertainties in predicting the actual CHF value. Further, the closure models used in the NSAM for the determination of CHF local conditions and for safety analysis calculations are identical. Using the closure models in this manner removes any fundamental effects that may arise from predicting local thermal-hydraulic conditions differently. In addition, the 95/95 limit captures any potential inaccuracies in VIPRE-01 related to high local equilibrium quality. The NRC staff accepts that the information in the original and supplemental responses to RAI No. 9080 demonstrates that VIPRE-01 and the selected two-phase closure models, when they are used with the NSAM and evaluation model, are appropriate for the full range of the NuScale CHF

correlation. The NRC staff agrees that the use of identical closure models in the NSAM and the entire range of CHF correlation development along with the 95/95 limit fully addresses all modeling uncertainties and is therefore appropriate. Therefore, the NRC staff concludes that NSAM applies to the entire range of the applicable approved CHF correlation; therefore, RAI No. 9080, Question 04.04-12, is resolved and closed.

The NSAM TR states that the methodology presented is independent of any specific CHF correlation, which is also one of the four specific elements of approval that NuScale has sought for the application of the NSAM using VIPRE-01. The TR further stresses that additional CHF correlations may be used with the code in the future. Each correlation must be approved for use by its own SER and submitted to the NRC for review and approval before licensing use. Because the NRC found the NSP2 CHF correlation used for the NuScale design acceptable for a range that encompasses the expected range for the NSAM application, the staff infers that the approved CHF correlation and its 95/95 MCHFR safety limit capture the phenomenological uncertainty in VIPRE-01 through validation with the test data. Therefore, the staff accepts that the NSAM is independent of any specific NRC-approved CHF correlation that is developed by applying VIPRE-01 with the identical closure models used in the subchannel analysis methodology (e.g., NSP2 CHF correlation).

Implementation of the NSP2 Critical Heat Flux Correlation in the NuScale VIPRE-01 Model

The NSP2 CHF correlation is implemented into VIPRE-01 by a dynamic linked library (DLL) file. This method allows VIPRE-01 to be used with user-programmed CHF correlations without any modifications to the VIPRE-01 source code. The staff needed to confirm that the NSP2 CHF correlation was correctly implemented in the NuScale DLL that is used with VIPRE-01. For this purpose, the staff coded the NSP2 correlation into a spreadsheet and compared the results to those calculated by VIPRE-01 for one steady-state case and three transient cases. Evaluating the MCHFR using the NSP2 CHF correlation by the spreadsheet required input of the following values from the VIPRE-01 output file at the time and axial location at the point of the MCHFR: pressure, mass flux, equilibrium quality, rod axial heat flux profile, elevation of the MCHFR point, and boiling length. A comparison of the MCHFR values calculated by the NRC staff's spreadsheet with the NuScale VIPRE-01 DLL output showed a discrepancy of less than 1 percent for all four cases. Exact matches would not be expected because of the imprecision associated with the printed output values that were used as input to the spreadsheet. The NRC staff concludes that the implementation of the NSP2 correlation into the NuScale DLL file is correct.

3. **VIPRE-1 Modeling Assumptions & Correlations:** Each organization using VIPRE-01 for licensing calculations should submit separate documentation describing how they intend to use VIPRE-01 and providing justifications for their specific modeling assumptions, choice of particular two-phase flow models and correlations, heat transfer correlations, CHF correlation and DNBR limit, input values of plant specific data such as turbulent mixing coefficient, slip ratio, and grid loss coefficient, etc., including defaults.

Turbulent Mixing Coefficient Sensitivity

In the NSAM TR, the applicant presented a sensitivity study that showed a negligible impact of the turbulent mixing parameter upon the MCHFR. To corroborate this finding for a different set of flow conditions, the staff conducted a sensitivity study of the effect of the turbulent mixing

coefficient on the MCHFR for the steady-state part of the control rod misoperation case. The nominal value of the mixing coefficient used by NuScale is {{ }}[1]. In this study, as in the NuScale study, it was varied from [ ]. As stated by NuScale in the TR, the staff also found a negligible impact of the mixing parameter on the MCHFR because the maximum change in [ ], which is the percentage variation from the nominal MCHFR value. In addition, the staff-calculated impact value compares favorably to that of the [ ] given in Table 6-9 of the TR for the high-power case. Therefore, the NRC staff confirmed that the MCHFR for the NuScale design is insensitive to the value of the turbulent mixing coefficient used in VIPRE-01. The NRC staff does not see a need to confirm this conclusion for transient conditions because the AOO transients analyzed by VIPRE-01 are relatively slow, almost like a steady-state succession. The control rod misoperation case starts from a VIPRE-01 steady state that is partially through the transient, just before the MCHFR point used for the confirmatory sensitivity study. In addition, NuScale biased the radial power profile in a conservative manner; therefore, the importance of the mixing coefficient is greatly diminished.

Grid Spacer Loss Coefficient

The subchannel analysis methodology requires fuel design-specific information as input into the subchannel base model. The applicant stated that the grid spacer loss coefficients and friction factor are derived from pressure drop tests. However, the staff notes that the grid spacer loss coefficients and friction factors derived are indicative of assemblywide losses, not subchannel-specific losses, because the pressure drop tests are instrumented in order to obtain the pressure drop across the entire grid. Therefore, the staff notes that the applicant used a "smearing approach" for grid spacer loss coefficient application in the subchannel analysis methodology. In short, the applicant took the assemblywide-derived loss coefficient and applied it uniformly to each subchannel being simulated by VIPRE-01.

In TR-0915-17564-P, Section 3.6, the applicant stated that a parametric sensitivity analysis in Section 6.3 that demonstrates that the impact of the grid loss coefficients applied globally is negligible justifies the use of the grid loss coefficient smearing approach. Because the staff could not locate this justification for the modeling method for the smearing loss coefficient to each subchannel in Section 6.3 of the TR, the staff issued RAI No. 9086, Question 04.04-11, dated September 9, 2017 (ADAMS Accession No. ML17252A688), to ask the applicant to provide additional information in the TR justifying the approach used for the subchannel grid loss coefficient, as described in Section 3.6 of the TR. In its response dated October 30, 2017, to RAI 9086, Question 04.04-11 (ADAMS Accession No. ML17299A973), the applicant provided a TR markup showing that the missing information will be added to Section 6.4.7. Revision 2 of TR-0915-17564, was verified to include this information. The staff found this response acceptable because the applicant provided a markup showing that the smearing approach justification would be added to the TR. The NRC staff reviewed the justification for using smeared loss coefficients and found that the use of smeared loss coefficients for the NuScale subchannel methodology is acceptable because it results in conservative calculations of the MCHFR.

[1] {{ }} - Export Controlled Information

Thom Nucleate Boiling Correlation

The wall heat transfer coefficient is relevant for transients such as control rod misoperation, whereby one of the figures of merit is the fuel melt temperature. Therefore, the NRC staff audited the applicant's calculation with regard to the VIPRE-01 heat transfer correlations and confirmed that the NSAM VIPRE-01 application used the Thom+EPRI (THSP) correlation for subcooled/saturated nucleate boiling plus single phase outside of its stated range of applicability. The NRC staff conducted a sensitivity study to assess the choice of the THSP correlation for the control rod misoperation case as it has the lowest MCHFR. Of the five nucleate boiling correlations compared at 2,000 psia, only the Chen correlation yielded lower values for the wall heat flux than did the THSP correlation. However, the MCHFR was unaffected by the selection of the boiling heat transfer model and was calculated as [ ] by both the THSP and Chen correlations. The peak fuel centerline temperature using the Chen correlation was about [ ]. Because a large margin exits for fuel melt, this small temperature difference is considered negligible. The NRC staff concludes that the value of the boiling heat transfer coefficient is so large that its uncertainty only results in a few degrees of difference in the fuel centerline temperature.

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Figure 1. Effect of the boiling heat transfer correlation on the fuel centerline temperature for the control rod misoperation case

In modern boiling heat transfer models, the forced convection and nucleate boiling components are added in a similar fashion to that used in the THSP correlation option. The NRC staff concludes that the THSP correlation option provides a conservative result because of three factors:

1. The forced convection component is calculated using the Dittus-Boelter heat transfer correlation. The Dittus-Boelter correlation was developed for heat transfer in tubes and has a consistent bias of a 20- to 30-percent underprediction compared to rod bundle data.

2. The forced convection component of the THSP correlation is calculated for single-phase convection only. Because there is no two-phase multiplier, the forced convection

component would be underpredicted by about a factor of about 2 for a void fraction of 50 percent.

3. When compared to modern nucleate boiling correlations for high-pressure conditions, the THSP correlation is quite conservative, as shown in Figure 2. For example, at 13.79 MPa (2,000 psia) and a wall superheat of 10 degrees C (18 °F), the THSP correlation gives a wall heat flux about one-quarter that of the Gorenflo model that is used in the TRACE code.

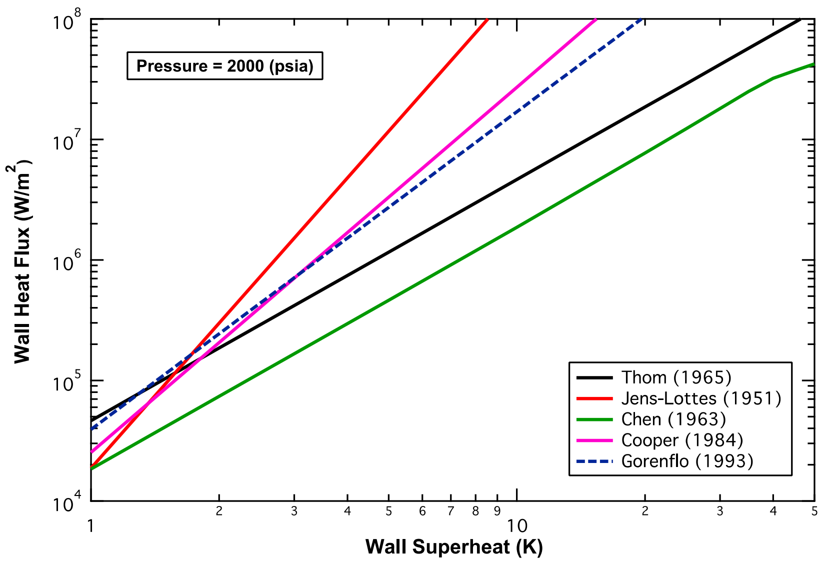

Figure 2. Comparison of nucleate boiling heat transfer correlations

Model Geometry

The NRC staff reviewed the applicant's subchannel methodology to determine the adequacy of the applicant's model discretization. The NuScale core contains 37 assemblies, and the applicant's subchannel analysis uses an eighth-core nodalization that does not represent any cycle-specific core. The applicant constructed a base model to use for licensing calculations that had a sufficient amount of radial spatial resolution while maintaining an economical computer simulation time. The applicant's base model is a 24-channel model that lumps subchannels in increasing size the further away they are from the hot subchannel. Similarly, the fuel rods are modeled as single fuel rods in the hot assembly and as lumped fuel rods in the rest of the assemblies of the eighth-core base model. The applicant also constructed two other finer resolution models (a 51-channel intermediate model and the 1,388-channel fully detailed model) to conduct sensitivity runs on the base model for ensuring adequate base-model resolution. The NRC staff reviewed the sensitivity analyses presented in the TR and confirmed that the 24-channel base model's radial nodalization accurately maintained the hot channel flow field and resulted in conservative MCHFRs compared to that of the fully detailed model. The applicant also showed that lumping channels further away from the hot channel had an insignificant effect on the flow field. For transients that result in control rod motion and thus a significant change in power distribution, the applicant uses the fully detailed 1,388-channel model. For the reasons listed above, the NRC staff finds the applicant's radial nodalization of the subchannel analysis base model and fully detailed model adequate for predicting thermal-hydraulic behavior.

The NRC staff also reviewed the applicant's axial nodalization to determine adequacy. With regard to the axial nodalization, CHF occurs in a region that is defined as the space between the fourth spacer grid and the top of active fuel; therefore, the applicant refined this part of the axial discretization more and conducted a sensitivity on the different axial resolutions. The NRC staff reviewed the sensitivity analyses presented in the TR and confirmed that the applicant's axial discretization is adequate for capturing the flow field throughout the height of the assembly. For the reasons listed above, the NRC staff finds the applicant's axial nodalization of the subchannel analysis base model and fully detailed model adequate for predicting thermal-hydraulic behavior.

The NRC staff noted that the applicant assumed "cold" conditions for geometrical input because the expansion of fuel and spacer grid materials is very similar; therefore, the change in flow area and wetted and heated perimeters for "hot" conditions is negligible. Furthermore, other phenomena such as rod thermal expansion and creepdown can cause slight changes in geometry; however, hot channel penalty factors and uncertainties account for this. For the reasons listed above, the NRC staff finds the use of "cold" geometry conditions acceptable.

Boundary Conditions

The NRC staff notes that VIPRE-01 requires certain boundary conditions, including inlet flow rate, inlet enthalpy or temperature, system pressure, bypass flow, power, exit pressure, and inlet and exit cross flows. The applicant inputs the boundary conditions into VIPRE-01 using a once-through linear process (i.e., no coupling of codes is required). The applicant employed NRELAP5 to calculate the system-level thermal-hydraulic parameters that VIPRE-01 uses as input. More specifically, NRELAP5 computes the core inlet mass flow rate that the applicant inputs into the VIPRE-01 code as a boundary condition. The applicant used the inlet mass flow rate as a boundary condition as opposed to using the core pressure drop in order to directly account for bypass flow. As documented below in the section titled, "Inlet Boundary Condition Consistency," the NRC staff conducted a confirmatory analysis to provide assurance that the core pressure drop computed by both NRELAP5 and VIPRE-01 is consistent. Because the system-level code consistently calculates thermal-hydraulic behavior, the NRC staff finds the applicant's boundary condition implementation adequate for the subchannel analysis methodology.

Bypass Flow

The applicant accounted for bypass flow in the subchannel analysis methodology by (1) using the NRELAP5 inlet mass flow rate as a boundary condition and (2) accounting for bypass flowpaths in the reflector cooling channel and in the guide tube/instrument tube channels.

The applicant subtracted a penalty from the inlet mass flow rate to account for the reflector cooling channel bypass flow. The NRC staff finds that applying a penalty to the inlet mass flow rate because of the reflector cooling channel bypass flow is acceptable for the NSAM; however, because the applicant did not provide a final reflector cooling channel bypass flow value for use in the subchannel methodology or a justification for developing this value, the staff does not approve the value to be used for this penalty nor the methodology for developing this penalty as part of this TR safety evaluation.

The NRC staff reviewed the applicant's justification for not subtracting a bypass flow penalty

from the inlet mass flow rate because of flow leakage between the heavy reflector and core barrel. The applicant stated that this flow area is negligible because of its size, which becomes smaller when thermal expansion causes the reflector cooling block to expand. Because the area between the heavy reflector and core barrel is initially small and because this area becomes smaller with thermal expansion, the NRC staff finds the applicant's reasoning for not subtracting a penalty from the inlet mass flow rate because of this flow leakage acceptable.

The applicant subtracted another penalty from the inlet mass flow rate to account for the guide tube and instrument tube bypass flow. The NRC staff finds that applying a penalty to the inlet mass flow rate because of the guide tube and instrument tube bypass flow is acceptable for the NSAM; however, because the applicant did not provide a final guide tube and instrument tube bypass flow value for use in the subchannel methodology or a justification for developing this value, the staff does not approve the value to be used for this penalty nor the methodology for developing this penalty as part of this TR safety evaluation.

Overall, the total bypass flow fraction that is subtracted from the inlet mass flowrate is calculated to be a sum of the reflector cooling channel bypass flow, the flow leakage between the heavy reflector and core barrel, and the guide tube/instrument tube bypass flow. As noted above, the NRC staff finds that applying a penalty to the inlet mass flowrate to account for bypass flow mechanisms is acceptable for the NSAM. However, because the TR does not provide the final values of bypass flow penalties or the methodologies for developing these final values, the NRC staff does not approve the value to be used for the total bypass flow fraction penalty nor the methodology for developing the total bypass flow fraction penalty as part of this TR safety evaluation.

Inlet Flow Distribution

The NRC staff reviewed the applicant's assumptions about the inlet flow distribution for the subchannel analysis methodology. The NRC staff notes that the applicant's subchannel analysis methodology requires the use of a maldistribution penalty (i.e., a reduction in flow on the hot assembly to provide assurance of conservative MCHFR results), which is consistent throughout industry applications. The results of a sensitivity study presented in the TR showed that large maldistribution penalties applied to the hot channel have insignificant effects on the resulting MCHFR because flow in the core redistributes early (i.e., in the lower portions of the fuel assemblies). The NRC staff finds that the applicant's application of a maldistribution penalty on the hot channel is conservative for the NSAM and therefore acceptable. However, because the TR does not provide a final value for inlet flow maldistribution penalty or the methodology for developing the final maldistribution penalty, the NRC staff does not approve the value to be used for the inlet flow maldistribution penalty nor the methodology for developing this penalty as part of this TR safety evaluation. The NRC staff notes that approval of a specific value for the maldistribution penalty not required as part of this TR safety evaluation because the maldistribution penalty has such an insignificant effect on the MCHFR, as shown by the applicant's sensitivity studies.

Inlet Temperature Distribution

The NRC staff reviewed the applicant's assumptions about the inlet temperature distribution for the subchannel analysis methodology. The NRC staff noted that the unique design of the NuScale integral helical coil steam generator precludes any possibility for asymmetric steam generator effects on the temperature of the coolant. Based on the NRC staff's review and

because of the unique design of the integral helical coil steam generator, the staff finds that a uniform core inlet temperature distribution is adequate for the subchannel analysis methodology.

Radial Power Distribution

The NRC staff reviewed the applicant's radial power distribution assumptions to assure adequacy for use in the subchannel analysis. The staff notes that the enthalpy rise factor, $F_{\Delta}H$, which describes a fuel rod's integrated power, changes as a result of exposure, fuel composition, burnable poison loading, operational history, and thermal-hydraulic conditions. The rod with the maximum $F_{\Delta}H$ or the peak $F_{\Delta}H$ is referred to as the hot rod, and its surrounding subchannel is referred to as the hot subchannel. The location of the hot subchannel can change throughout the cycle for the reasons above. The NRC staff notes that the applicant's subchannel methodology selects a radial power distribution that is conservative and bounds the worst distribution throughout the cycle. The NRC staff also notes that in Section 3.10 of the TR, the applicant stated that, for each cycle, an analysis will confirm that the cycle-specific radial power distribution is bounded by the conservative radial power distribution developed as part of the subchannel analysis methodology. Because the applicant confirmed that the cycle-specific radial power distribution is bounded, the NRC staff finds that the applicant's approach for using a bounding radial power distribution is acceptable for this TR methodology.

The applicant stated that the radial power distribution will remain constant throughout the transient for subchannel analysis. The NRC staff finds this acceptable because the constant, artificial radial power distribution used by the applicant bounds the dynamic power distribution that results from an actual event. For analyses associated with reactivity and power distribution anomalies, the radial power distribution used is still constant throughout the transient; however, the applicant applied an $F_{\Delta}H$ augmentation peaking factor to the limiting assembly to account for the asymmetric radial peaking that results from reactivity changes in the core. A nuclear analysis determines this $F_{\Delta}H$ augmentation peaking factor as the ratio of the change in the $F_{\Delta}H$ from the postevent to the initial condition. The NRC staff notes that the relative increase in the $F_{\Delta}H$ captures the peaking increase from control rod motion during an actual transient. For transients that result in significant reactivity changes and severely skewed power distribution changes, the applicant stated that the radial power distribution used in the subchannel analysis is at the time of peak core power, as determined from an event-specific nuclear analysis. The NRC staff notes that using the power distribution at the time of peak power is conservative because it is used throughout the entire transient. Furthermore, these transients, which result in severely skewed power distributions, are analyzed with the fully detailed 1,388-subchannel model. For the reasons listed above, the NRC staff finds the applicant's approach for implementing the radial power distribution appropriate.

To determine the actual values used in the radial power distribution, the applicant set the hot rod to be the technical specification limit on $F_{\Delta}H$ (i.e., the applicant ensured that the highest $F_{\Delta}H$ value for any fuel rod is limited to the technical specification limit at whatever power level the transient starts). However, setting the hot rod at the technical specification limit implies that uncertainties are not included. Therefore, the applicant accounted for measurement, engineering, and rodded configuration uncertainties by increasing the technical specification value that is applied to the hot rod. Furthermore, for cases evaluated at partial power levels, the applicant ensured that the power distribution in the central assembly is increased by a factor corresponding to the ratio of the technical specification limit at that power level divided by the technical specification limit at hot full power. The NRC staff notes that this procedure ensures

that appropriate power peaking of the hot assembly is captured for partial power level initial conditions. Furthermore, the applicant stated that an additional requirement on the power distribution is that the peak $F_{\Delta}H$ rod for any assembly cannot occur on the peripheral row. This is a result of the way the CHF correlation was developed (i.e., the peripheral row is not truly simulated in CHF tests used to develop the correlation). The NRC staff notes that this is conservative because a rod on the outer row is influenced by direct crossflow (i.e., the requirement that the hot rod must occur in the center of an assembly is typically conservative).

The NRC staff notes that radial tilt is a valid phenomenon that can occur in the NuScale reactor core. Radial tilt is a condition in which power is not symmetric between azimuthally symmetric fuel assemblies, as noted by the applicant. Radial tilt is also associated with a technical specification. The NRC staff notes that radial tilt can affect thermal margin because, if the power distribution tilts, one assembly may obtain a higher $F_{\Delta}H$ value. The applicant considered this phenomena in its subchannel analysis methodology. The applicant stated that the design enthalpy rise peaking factor safety limit (i.e., the technical specification limit) inherently accounts for radial tilt. The applicant stated that the nuclear analysis group completes core design calculations that account for radial tilt caused by any xenon transients. The nuclear analysis group then verifies that those core design calculations produce $F_{\Delta}H$ values below the technical specification limit. Because the core design calculations account for radial tilt and because the results of these calculations do not exceed the technical specification limit on the $F_{\Delta}H$, the NRC staff notes that the subchannel analysis methodology does not then require any additional accounting of radial tilt because the subchannel analysis methodology uses the technical specification limit on $F_{\Delta}H$ as its starting point for developing the conservative hot channel peaking factor. Because the core operating limit $F_{\Delta}H$ is met while nuclear analyses account for radial tilt, the NRC staff finds that the subchannel analysis methodology, which has no explicit radial tilt uncertainty, is acceptable.

The NRC staff notes that, for the NuScale reactor design, the core can be at 100-percent rated thermal power with control rods inserted to the power-dependent insertion limits (PDILs). Because the enthalpy rise hot channel factor is defined as an all rods out (ARO) maximum value, the applicant accounted for an additional peaking factor on the hot rod for PDIL-to-ARO peaking differences. The applicant stated that a nuclear analysis determines the PDIL-to-ARO augmentation factor for all power levels, and that the subchannel analysis uses a single bounding value to be applied at every power level. The applicant further stated that this value is cycle specific in nature (e.g., based on loading patterns, control rod worth); therefore, the subchannel analysis methodology uses a cycle-independent bounding value that is ultimately confirmed by a cycle-specific nuclear analysis calculation. The NRC staff finds the applicant's approach for selecting a bounding PDIL-to-ARO augmentation factor acceptable because a cycle-specific nuclear analysis will confirm the bounding value as part of the subchannel analysis methodology. However, because the applicant did not provide a final value to be used for the PDIL-to-ARO augmentation factor, the staff does not approve the PDIL-to-ARO augmentation factor value as part of this TR safety evaluation.

The applicant stated that the purpose of the artificial bounding radial power distribution is to capture the hot subchannel flow conditions, which is dependent on the surrounding crossflow neighbor channels. The applicant used a relatively flat power distribution within the limiting hot assembly to ensure that each channel around the hot channel has similar flow conditions and that turbulent mixing and diversion crossflow in the hot subchannel are minimal. The NRC staff finds this approach acceptable because minimizing turbulent mixing and diversion crossflow will minimize the MCHFR, which is conservative. By characterizing the power distribution in the hot

assembly using a peak-to-average ratio of $F_{\Delta}H$, the applicant can ensure that the power distribution in the hot assembly is characterized by a peak-to-average ratio close to 1 (i.e., flat). The applicant used a sensitivity analysis on a large spectrum of peak-to-average $F_{\Delta}H$ values for each assembly throughout the cycle burnup in the equilibrium cycle design. This sensitivity informs the applicant's selection of the bounding, relatively flat, power distribution. The NRC staff audited the applicant's sensitivity analysis and confirmed that the method used to determine the flat peak-to-average ratio for the limiting assembly's power distribution is conservative. The NRC staff further confirmed that the $F_{\Delta}H$ of the hot rod is set at the peak value (i.e., technical specification limit multiplied by the PDIL-to-ARO augmentation factor), and the $F_{\Delta}H$ of the remaining rods that are in the hot assembly is gradually sloped using the conservative peak-to-average ratio and the distance the rod is from the hot rod to ensure that hotter rods remain around the hot rod to ensure that turbulent mixing is minimal. The remaining rods in the hot assembly that are gradually reduced in power by the conservative peak-to-average ratio and by the rod's distance from the hot rod are such that the total power generated is preserved. For the reasons listed above, the NRC staff finds the applicant's radial power distribution development conservative and thus acceptable. However, because the applicant does not provide final values to be used for the bounding radial power distribution in the subchannel analysis methodology, the NRC staff does not approve the bounding radial power distribution values as part of this TR safety evaluation.

The last piece to the applicant's radial power distribution is to appropriately apply any uncertainties associated with $F_{\Delta}H$. These uncertainties include measurement uncertainty and engineering uncertainty. The uncertainties on $F_{\Delta}H$ are applied only to the hot rod, whereas the rods in the peripheral assembly are slightly reduced to maintain normalization of power. The staff confirmed through an audit of the applicant's sensitivity analysis that the radial power distribution far removed from the hot subchannel has a negligible effect on the MCHFR results. The applicant used the root-sum-square method to apply the $F_{\Delta}H$ measurement uncertainty and $F_{\Delta}H$ engineering uncertainty to the hot rod. The NRC staff finds this acceptable because the uncertainties are composed of independent parameters.

Section 3.12.4 of the TR describes that the enthalpy rise engineering uncertainty is composed of two components. The first component is the rod power component, which the applicant uses to account for uncertainties in the fuel stack length and uranium loading, and is calculated using the methodology described in TR-0116-20825-A, "Applicability of AREVA Fuel Methodology for the NuScale Design," dated February 9, 2013 (Ref. 13).

The second component of the enthalpy rise engineering uncertainty is the flow area reduction factor, $F^{E}_{\Delta H2}$, which accounts for uncertainties in the fuel rod diameter and fuel rod pitch. The NRC staff noted that the applicant's methodology for computing the flow area reduction factor is based on the fuel geometry (e.g., rod diameter, pitch, and associated tolerances) and the CHF correlation used to determine the factor. If any of these parameters change (e.g., fuel geometry design values or CHF correlation), the staff notes that the value calculated for use in subchannel analyses will also need to be recomputed. Despite the dependence of the flow area reduction factor on fuel geometry and CHF correlation, the staff reviewed the applicant's methodology for calculating this value and found it acceptable because the factor itself is simply an increase in the hot rod $F_{\Delta}H$ that would result in the same MCHFR if the rod pitch was reduced by its manufacturing tolerance and the fuel rod and guide tube outer diameters were increased by their tolerances. However, Section 3.12.4, "Enthalpy Rise Engineering Uncertainty," of the TR provides a final value for this factor. The NRC staff notes that, because the value is dependent on the fuel geometry and the CHF correlation, the value presented in the

TR should only be reported as an example. To eliminate confusion and to clarify that the NRC staff is approving the method for calculating this factor, not the value itself, the staff issued RAI No. 9099, Question 04.04-10, dated October 2, 2017 (ADAMS Accession No. ML17251A368) asking the applicant to clearly identify that the value presented for the flow area reduction factor is simply an example value. In its response to RAI No. 9099, Question 04.04-10, dated September 13, 2017 (ADAMS Accession No. ML17251A368), the applicant clarified that the flow area reduction factor depends on the fuel design and CHF correlation and that the value presented is only an example. Revision 2 of TR-0915-17564, was verified to include this information. Because the applicant clarified that the flow area reduction factor depends on the fuel design and CHF correlation and that the value presented in the TR is only an example value and because the applicant provided TR markups to support this response, the staff finds the response acceptable.

The NRC staff notes that the applicant also employed an $F_{\Delta}H$ measurement uncertainty as part of the subchannel analysis methodology. The NRC staff finds this acceptable because measurement of system parameters will always have some associated uncertainty with it; furthermore, these measurement uncertainties are plant specific. However, because the applicant did not provide a final value to be used for the measurement uncertainty in the subchannel methodology, the NRC staff does not approve the measurement uncertainty value as part of this TR safety evaluation.

Axial Power Distribution

Because the axial power distribution is widely variable (i.e., it is dependent on core design, exposure, and power history), the applicant used a CHF-limiting axial power shape in the subchannel analysis methodology. To develop this bounding axial power profile, the applicant considered various configurations of core cycle exposure, control rod configuration, xenon distribution, and core thermal-hydraulic conditions. The applicant stated that these power shapes consider any possible scenario that can occur for normal and anticipated operation within or on the axial offset (AO) window. The applicant stated that the CHF-limiting shape is obtained from the many potential permutations of the core-average axial power shape from the nuclear analysis. The staff notes that the MCHFR for the subchannel base model is sensitive to axial power shape.

The applicant stated that a top-peaked axial power shape for the same magnitude of axial peaking is more limiting with respect to CHF. The applicant's methodology ensures that the top-peaked power shapes reach the widths of the AO window. The applicant used the subchannel base model to select the axial power shape that results in the lowest MCHFR. For events involving control rod motion, the applicant stated that it performs an event-specific nuclear analysis to determine control rod worth, radial power distribution (the $F_{\Delta}H$ augmentation factor), and axial power distribution. The applicant stated that the initial conditions for control rod motion transients start from the edges of the AO window. If rods leave the core for an uncontrolled bank withdrawal or single rod withdrawal, the core-average axial power shape has the potential to go beyond the AO limits for a brief amount of time.

Because the AO limit could be exceeded and because no axial power shapes were analyzed (to determine the bounding axial shape) outside of the AO limit, the applicant found that event-specific nuclear analyses are necessary to provide axial power shapes in addition to radial peaking augmentation and control rod worth information for specific transients. However, based on further review, the NRC staff was unable to clearly understand which axial power

methodology was used for each of the SRP Section 15.4 events involving control rod motion (i.e., an axial power shape from an event-specific nuclear analysis or an axial power shape used in non-SRP Section 15.4 events, as described above). Therefore, the staff issued RAI 9129, Question 04.04-13, dated December 7/2017 (ADAMS Accession No. ML17321A597), to rectify this issue.

In its response dated January 15, 2018, to RAI No. 9129, Question 04.04-13 (ADAMS Accession No. ML18015A012), the applicant clarified that it performs an event-specific nuclear analysis for each of the SRP Section 15.4 events that involve control rod motion to determine the postevent axial power shape. However, in some cases that have the available margin, the axial power shape from the generic axial power shapes analysis detailed above for non-SRP Section 15.4 events (i.e., the event-initiating power level axial power shape) may be even more conservative for use in the subchannel analysis than use of the postevent axial power shape because of the larger than allowed magnitudes of axial peaking caused by a wider AO at lower power levels. In any case, the NRC staff notes that the axial power shape used in the analysis is conservative for the event and is held constant throughout the transient.

Because the applicant used a bounding axial power distribution that results in the most conservative prediction of the MCHFR for non-SRP Section 15.4 events and that informs the axial power shape selection for SRP Section 15.4 events that involve control rod motion using an event-specific nuclear analysis, the NRC staff finds the applicant’s axial power distribution approach adequate for the subchannel analysis methodology. As part of its response to RAI No. 9129, Question 04.04-13, the applicant provided markups to the TR showing the changes that will be made in the next revision to clarify the applicant’s axial power shape approach for the subchannel analysis methodology. Revision 2 of TR-0915-17564, was verified to include this information. Lastly, because the applicant does not provide final values to be used for the axial power distribution in the subchannel analysis methodology, the NRC staff does not approve the axial power distribution values as part of this TR safety evaluation.

Rod and Assembly Bow Penalty

As a result of exposure, a fuel rod can bow and reduce the channel flow area, which negatively affects the CHF and linear heat generation rate (LHGR). The NRC staff reviewed the applicant’s subchannel methodology to ensure that it appropriately accounts for the effects of rod bow. The NRC staff notes that NuScale applied a penalty externally to VIPRE-01 calculations by increasing the CHF analysis limit that is used for margin comparison. The NRC staff further notes that the applicant calculated the penalty in accordance with NRC-approved TR-0116-20825-A. The NRC staff finds that the application of a CHF penalty external to the VIPRE-01 calculations that was calculated in accordance with an NRC-approved methodology is conservative and acceptable.

Similarly, as a result of exposure, an entire fuel assembly can bow and negatively affect the CHF and LHGR. The NRC staff reviewed the applicant’s subchannel methodology to ensure that it has appropriately accounted for the effects of assembly bow. The NRC staff notes that large flux gradients typically increase the potential for an entire assembly bow; however, large flux gradients are encountered toward the edge of the core where fresh fuel is loaded and where CHF is not a concern. The NRC staff notes that the approved AREVA fuel methodology considers the effects of fuel assembly bow; however, it deems the effects insignificant to plant operating thermal margins mostly because of where assembly bow could occur. The NRC staff further notes that a shorter fuel assembly, like the NuScale fuel assembly, would be less

susceptible to bow. The NRC staff notes that the applicant did not apply an assembly bow penalty in the subchannel analysis methodology in accordance with approved TR-0116-20825-A; for the reasons listed above, the staff finds this acceptable.

Heat Flux Engineering Uncertainty

The NRC staff notes that NuScale applied an engineering uncertainty on heat flux, which accounts for fuel enrichment, pellet density, pellet diameter, and fuel rod surface area. These uncertainties manifest from manufacturing processes and affect the local heat flux. The NRC staff notes that the applicant calculated this uncertainty in accordance with NRC-approved TR-0116-20825-A. The NRC staff finds that the application of an engineering uncertainty on heat flux that was calculated in accordance with an NRC-approved methodology is conservative and acceptable.

Linear Heat Generation Rate Engineering Uncertainty

The NRC staff notes that NuScale also applied an engineering uncertainty on the peak LHGR for fuel centerline melt calculations, which is similar to the heat flux engineering uncertainty. The NRC staff notes that the applicant calculated this uncertainty in accordance with NRC-approved TR-0116-20825-A. The NRC staff finds that the application of an engineering uncertainty on LHGR that was calculated in accordance with an NRC-approved methodology is conservative and acceptable.

Application of Uncertainty

The NRC staff notes that the applicant used a deterministic methodology for uncertainties (i.e., the uncertainty associated with a parameter is applied in the conservative direction without regard for the combination nature of uncertainties). The NRC staff finds this approach acceptable because it is a conservative way to bias parameters.

CHF correlation and DNBR limit

The details of the NSP2 CHF correlation and DNBR limit have been provided by the applicant, and have been discussed under Condition 2.

4. **Courant Number Criterion** If a profile fit subcooled boiling model (such as Levy and EPRI models), which was developed based on steady state data is used in boiling transients, care should be taken in the time step size used for transient analysis to avoid the Courant number less than 1.

The SER for VIPRE-01, MOD-01, identified the dimensionless Courant number as a necessary condition for the convergence of the numerical solution of the mass, momentum, and energy conservation equations. The sensitivity studies described in the SER for VIPRE-01, MOD-01, show that a subcooled void correlation based on steady-state data is not suitable in boiling transients in which the Courant number is less than 1. Because the NSAM uses the existing VIPRE-01 steady-state subcooled boiling correlations for modeling transients, the user needs to ensure that the Courant number is greater than 1 to satisfy the corresponding SER condition ($N_C > 1$). Because VIPRE-01 uses an implicit solution technique, the staff does not anticipate any numerical stability issues resulting from the transient time step selection as long as the Courant number criterion is met.

Section 4.3 of the NSAM TR describes the methodology for addressing the Courant number criterion and states that “Selection of the transient time step is achieved on a case-by-case basis based on the axial nodalization and the coolant velocity to ensure that the SER condition is satisfied.” Section 6.3 of the NSAM TR presents results for various system parameters for three design-basis example transients for the NuScale design and states that “Additionally, all convergence and methodology criteria, including the Courant limit criterion, are satisfied.” However, the TR does not include any quantitative information to show that the Courant number criterion was satisfied. The NRC staff needed to establish that the applicant’s selected time step and axial nodalization ensured that the Courant number condition was met throughout the transient. Therefore, the NRC staff reviewed the relevant subchannel analysis calculations for control rod misoperation; decrease in feedwater temperature; and loss of external load, turbine trip, and condenser vacuum. The staff audited the results for both the MCHFR and the minimum value of the Courant number, and NuScale’s confirmation that the Courant number limitation had been met. Further, the staff independently checked the NuScale results to confirm their validity. The Courant number is simply given by the following equation:

$$N_C = V_z \frac{\Delta t}{\Delta z} \tag{1}$$

where V_z is the axial velocity, Δt is the time step, and Δz is the node length. This calculation assumes that the lateral velocity is negligible relative to the axial velocity, which is irrelevant with respect to finding a minimum Courant number because the vector addition of any lateral velocity component would only serve to augment the Courant number.

Because VIPRE-01 does not output the Courant number, the staff used an approximate approach to verify the NuScale results. To minimize the value given by Equation 1, the staff used the minimum velocity and the maximum axial node size, thereby minimizing the numerator while maximizing the denominator. For the transients considered here, the minimum velocity always occurs at the inlet of the hot channel not only because it is biased low, but also because the axial velocity increases as the mixture density decreases as a result of heat input. Consequently, the staff determined the minimum hot channel inlet velocity from the output files for each transient and evaluated Equation 1 using the maximum node size of [ ]. This relatively simple approximate method would give a lower bounding value of the Courant number because some flow acceleration would be expected to occur before the flow reached the level of the first [ ] node.

The approximate Courant number values calculated by the staff are within 3.8 percent of the NuScale values, which is not significant. The staff’s calculated values are lower than the NuScale values with the one exception—the value for Condition 4. Although the cause of this small discrepancy has not been determined, the staff notes that the NuScale value would be conservative. The audited calculations also included the hot channel Courant number plots for the respective example transient that verified the Courant numbers for the respective MCHFR. The staff finds that the Courant number methodology used by NuScale in the NSAM TR to ensure the convergence of a numerical solution appears to be accurate and reasonable. Therefore, the staff concludes that the Courant number criterion was satisfied and that there are no concerns with the overall stability of the NuScale numerical model implemented in the example VIPRE-01 transients presented in the TR.

5. <u>**Quality Assurance**</u> The VIPRE-01 user should abide by the quality assurance procedures described in Section 2.6 of the VIPRE-01 MOD-01 SER.

Section 2.6 of the SER for VIPRE-1, MOD-01, summarizes the quality control requirements to ensure consistency in the development and application of the VIPRE-01 code for safety analysis. It identified the quality assurance needs to ensure that only the approved version of VIPRE-01 is used and any modification to VIPRE-01 is implemented under quality control. The SER for VIPRE-01, MOD-02, that was approved in 1993 did not report any quality assurance issues or identify additional quality control requirements since the approval of VIPRE-01, MOD-01, in 1986. As described in the Section 2.2 of the TR, no modifications were made to the constitutive models and algorithms in VIPRE-01, MOD-02, and the computational philosophy remains unchanged. The TR also stressed that "The only enhancement that was made to VIPRE-01 for NuScale use is the addition of NuScale specific CHF correlations to the existing suite of VIPRE-01 CHF correlations." The NRC staff concluded that implementation of the NSP2 CHF correlation into VIPRE-01 by a DLL file would allow the VIPRE-01 code to be used with user-programmed CHF correlations without any modifications to the VIPRE-01 source code. Therefore, the NRC staff has no concerns about the quality assurance for the version of VIPRE-01, MOD-02, used with the NSAM.

4.2 VIPRE-01, MOD-02, Safety Evaluation Report Conditions

The TR provided information on the fulfillment of the four additional conditions for the use of VIPRE-01, MOD-02. These conditions are listed below, followed by the NRC staff's evaluation relevant to each condition.

6. **<u>Qualification of VIPRE-01 MOD-02 Models</u>** The use of this code [VIPRE-01 MOD-02] is contingent upon full qualification of the models described in Section 3.2.2 of VIPRE-01 MOD-02 technical evaluation report (TER) before implementation into a licensing model. For example, since no model verification or qualification was provided with the submittal, each user of the drift flux model should be required to justify the use of the model and the selected parameters related thereto on a transient-by-transient basis over the range of two-phase flow conditions expected to be encountered.

While describing VIPRE-01, page 10 of the TR states that "Although the formulation is based on the fluid being homogeneous, non-mechanistic empirical models are included for subcooled boiling non-equilibrium and vapor/liquid phase slip in two-phase flow." Page 63 of the TR also states that "The flow field is assumed to be incompressible and homogeneous, although models are added to reflect subcooled boiling and concurrent liquid/vapor slip." In accordance with Table 7-1 of the NSAM TR, a slip ratio is not used as a NuScale modeling assumption, which, to the staff, meant that NuScale did not use the VIPRE user option to specify a slip ratio as an input to the model. As stipulated by Condition 3 of the SER for VIPRE-01, MOD-01, and Conditions 6 and 8 of the SER for VIPRE-01, MOD-02, to support qualification of the modeling assumptions, the applicant needed to justify its two-phase modeling for using VIPRE-01 for the NuScale design application.

Section 3.2.2 of the TER for VIPRE-01, MOD-02, as cited in Condition 6 above, discusses the models added to the code specifically for use in BWR applications that include (1) water tube channel modeling, (2) leakage flowpath connection, and (3) a drift flux model. The "water tube channel" model simulates the behavior of the hollow "water" rods that replace fuel rods in some BWR designs. VIPRE-01 can model lateral leakage paths that exist in the BWR lower regions (below the active core) as gap connections that exist only over a few axial nodes instead of the

full length of the channel. Therefore, the NRC staff concludes that the water tube channel modeling and leakage flowpath connections are specific to BWR applications and are not a concern for the NuScale design. However, the NRC staff could not conclude the same for the drift flux model.

In accordance with Section 2.2 of Volume 5 of the VIPRE-01 manual (Ref. 11) and Section 3.2.2 of the TER for VIPRE-01, MOD-02 (Ref. 10), the homogeneous equilibrium formulation of VIPRE-01 is not sufficient to apply to cases with large relative phase velocities. Because of the homogeneous flow assumption in VIPRE-01, convergence problems may arise in transient boiling calculations when using the subcooled boiling models and bulk void correlations to account for phase slip. The NRC staff agrees that the drift flux model in VIPRE-01 calculates the void fraction by using the physically more realistic conservation of vapor mass generated by wall superheat instead of defining the void as a function of flow and quality with an empirical slip correlation. The NRC staff's close scrutiny of Section 3.2.2.3 of the TER for VIPRE-01, MOD-02, revealed that there is nothing specific in the drift flux model that would limit it to BWR applications. Even the original text of the condition, as concluded in Section 4.0 of the TER for VIPRE-01, MOD-02, does not include the specific language "use of this code for BWR licensing applications." In addition, Section 4.0 of the TER for VIPRE-01, MOD-02, states that the validation of VIPRE-01 against the submitted void fraction test data was inconclusive. In this regard, as described in Section 5.10 of the NSAM TR, NuScale took the approach of demonstrating the global results from the VIPRE-01 computations to be appropriate while using the drift flux model for NuScale applications.

To understand the justification and conservatism of the applicant's two-phase flow modeling assumptions, the staff audited the applicant's documentation of the drift flux model sensitivity cases applicable to NuScale conditions and confirmed that use of the drift flux model had an insignificant impact on void fractions and CHFR, as stated in the TR. The void fraction results vary little across the three base models for the hot channel (i.e., less than a 2-percent change at any point along the model elevation). The staff also confirmed that the modified code, NVIPRE-01, without the void drift option predicted void fraction and MCHFR values and trends identical to that of VIPRE-01. Considering the scarcity of the transient void fraction data, the staff found the applicant's use of the drift flux model to justify its VIPRE-01 two-phase modeling assumptions appropriate and consistent with the conclusions of the TER for VIPRE-01, MOD-02.

7. **GEXL Correlation** The GEXL correlation is the only correlation currently having NRC approval for use in critical power ratio (CPR) calculations of a core containing GE fuels. However, use of the GEXL correlation for other vendors' fuels or use of any other correlation requires a separate submittal for NRC review and approval.

This condition from the SER for VIPRE-01, MOD-02, requires the applicant to declare any possible use of the GEXL correlation with the proposed subchannel analysis methodology. With respect to this condition, the NSAM TR states that "NuScale does not perform CPR calculations for BWR fuel with VIPRE-01." Although the TR statement does not categorically confirm that NuScale will not use the GEXL correlation with the NSAM with the VIPRE-01 code, the NRC staff concludes that the above cited statement means that NuScale will not use the GEXL correlation for the NSAM VIPRE-01 applications. Therefore, no further clarification is needed in this regard.

8. **Code Limitations** Section 2.2 of Volume 5 of the submittal [VIPRE-01 Manual] (Ref. 11) identifies a spectrum of limitations of the code. Each user, in its documentation for NRC approval, should certify that the code is not being used in violation of these limitations.

Section 2.2 of Volume 5 of the VIPRE-01 manual on guidelines identifies a spectrum of limitations of the code. Each user needs to ensure that VIPRE-01 is not being used outside of these limitations.

As described in Section 2.2, the conservation equations in VIPRE are based on the assumptions of homogeneous equilibrium incompressible flow. The homogeneous equilibrium formulation of VIPRE-01 is not sufficient to apply to cases with large relative phase velocities, countercurrent flow, or conditions under which the flow regime changes radically. VIPRE should not be applied to situations that entail conditions such as low-flow boiloff, annular flow, phase separation involving a sharp liquid/vapor interface, or countercurrent flow. Another limitation arises from the omission of several cross-coupling terms from the lateral momentum equation. The VIPRE-01 equations are still valid wherever the lateral flow resistance is large compared to axial flow resistance. Therefore, VIPRE predicts the flow field accurately only in cases where wall friction is significant and lateral flow resistance is fairly large. As described under Condition 2, the applicant has appropriately addressed the staff's concerns in RAI No. 9080, Question 30583, about VIPRE-01 code limitations for NuScale application.

Operating Conditions: Boiling-Water Reactors or Pressurized-Water Reactors

In its initial review of the NSAM TR with regard to the input selection, uncertainties, and sensitivity analyses, the NRC staff conducted a study to determine whether VIPRE-01 was used for NuScale normal and off-normal conditions that are more typical of a BWR than those of a PWR. The relevant concern was whether or not the two-phase flow regimes encountered in the NuScale normal and off-normal operating conditions involve annular flow with large relative phase velocities. The investigation was performed to determine whether any known code limitations were violated in the NuScale application model of VIPRE-01.

The NRC staff used VIPRE-01, MOD-2.5, to analyze the steady state and three transients to examine the resulting two-phase conditions in detail and compared them to the NuScale results. The four analyses are as follows:

1. Base Model: steady state
2. Case 1: loss of external load transient (Ref. 9)
3. Case 4: control rod misoperation transient (Ref. 9)
4. Case 5: decrease in feedwater temperature transient (Ref. 9)

Figure 3 shows the resulting transient void fraction results calculated by the NRC staff.

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Figure 3. Maximum void fraction versus time for NuScale transients

Table 1 summarizes the staff's confirmatory VIPRE-01 calculation results for the one steady-state and three transient input models that NuScale made available for the NRC staff's audit. The maximum void fraction and MCHFR values calculated by the NRC staff matched exactly the values for all four cases. The table shows that the most limiting VIPRE-01 transient analyzed, both from the standpoint of a "lowest MCHFR" and highest "maximum void fraction," was Case 4 for control rod misoperation (single rod withdrawal), which corroborates the applicant's identification of the limiting transient and evaluation of the MCHFR. Consequently, the NRC staff used Case 4 to investigate the limiting two-phase flow conditions and for the other sensitivity calculations. For this case at the time of the MCHFR, the hot channel exit void fraction is [ ]. However, about two-thirds of the core exit plane is subcooled, and the bulk core exit is also subcooled. Consequently, the riser would also be single-phase flow.

Table 1. [

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Based on the above discussion, the staff concludes that the VIPRE-01 code is not being used in NSAM in violation of the limitations identified in Section 2.2 of Volume 5 of the submittal [VIPRE-01 Manual].

9. **<u>Input Selection, Uncertainties and Sensitivity Analyses</u>** By acceptance of this code version [VIPRE-01 MOD-02], the NRC does not endorse procedures and uses of this code described in [VIPRE-01 Code Manual] Volume 5 (Ref. 11) as appropriate for licensing applications. As the code developer stated (Ref. 12), the materials were provided by the code developers as their non-binding advice on efficient use of the code. Each user is advised to note that values of input recommended by the code

developers are for best-estimate use only and do not necessarily incorporate the conservatism appropriate for licensing type analysis. Therefore, the user is expected to justify or qualify its use [i.e., input selections] for licensing applications.

Inlet Flow Boundary Condition Consistency

The NPM operates in a natural circulation mode instead of using forced circulation as in a conventional PWR. Standard practice in subchannel analysis is to impose a transient core inlet flow rate calculated by a system's analysis code as the inlet boundary condition. When the core inlet flow rate results from a change in pump speed or pump coastdown, this approach is clearly reasonable. However, because the NPM is a natural circulation system, the consistency of this boundary condition would come into question if the core pressure drop calculated by VIPRE-01 is substantially different from that calculated by NRELAP5, which provided the imposed inlet flow boundary condition. The NRC staff needed to confirm the consistency because the NRELAP5 core model with one lumped channel predicted a subcooled core exit, whereas in the VIPRE-01 simulation for the control rod misoperation case, one-third of the core exit was in a two-phase condition and two-thirds of the core exit was subcooled. Less of the core was in a two-phase condition for the other cases. Therefore, the NRC staff used VIPRE-01 to simulate the limiting transient (Case 4, control rod misoperation) and compared the resulting core pressure drop with the NuScale NRELAP5 calculations (Ref. 9) reviewed by the staff. Figure 4 shows the results. The NRC staff observed a small difference (approximately 0.127 psia) between the VIPRE-01 and NRELAP5 calculated values of core pressure drop; the NRELAP5 values were higher. This offset did not appear to be a function of either the core flow rate or the extent of the two-phase region in the VIPRE-01 calculation and was nearly constant over the duration of the transient. From this, the NRC staff inferred that the most likely reason for the offset would be an inconsistency between the axial length and, hence, the gravity head component used in the NRELAP5 core pressure drop calculation compared with that of the VIPRE-01 core model.

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Figure 3. Core pressure drop comparison for Case 4

In this regard, the NRC staff reviewed NuScale's documentation for the locations of the "virtual" pressure taps that were used to provide the NREALP5 values of core pressure drop. In the reviewed material, NuScale stated that it calculated the NRELAP5 core pressure drop for the active core region plus about 9 inches of unheated region above the core and about 2.5 inches of unheated region below the core. Compared to the VIPRE-01 model, the core in the

NRELAP5 calculation is about 5.25 inches longer with an extra 3.32 inches above the active core and an extra 1.93 inches below it. Using these values combined with the core inlet and outlet mixture densities from Case 4 (at time zero), the staff calculated a gravity head adjustment of 0.126 psia. This value almost exactly matches the observed offset. Therefore, the NRC staff has no concern about inconsistency between the imposed flow boundary condition calculated by the NRELAP5 system code and the core pressure drop calculated by VIPRE-01 in the subchannel analysis.

Flow Perturbation Sensitivity Study

Because nearly one-third of the NuScale core exit plane is in a two-phase condition at the time of the MCHFR for Case 4 and because the flow is driven by a relatively low-gravity head, the staff performed a sensitivity study to investigate the sensitivity of core pressure drop to a perturbation in core flow rate. The NRC staff selected Case 4 (control rod misoperation) for this sensitivity study because it is the transient that results in the largest core exit quality. NuScale simulated this transient with VIPRE-01 by first running a steady-state calculation at a time of about 7 seconds before the point of the MCHFR. At this time, the core two-phase flow conditions were almost the same as those at the time of the MCHFR; therefore, the NRC staff used this steady-state calculation for the sensitivity study. The NRC staff performed the sensitivity study using this steady-state operating point and perturbing the core inlet flow rate by ±10 percent at a constant heat flux condition. Figure 5 and Figure 6 show the results of the NRC staff's analysis for the core pressure drop and MCHFR.

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Figure 4. Sensitivity results for core pressure drop as a function of flow rate at constant power

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Figure 5. Sensitivity results for the MCHFR as a result of core flow rate perturbation

These simulations showed that the core pressure drop varied with the inlet flow rate almost linearly with a positive slope. For a 10-percent decrease in the inlet flow rate, the MCHFR decreased by [ ], whereas for a 10-percent increase in the inlet flow, the MCHFR increased by [ ]. In addition, the behavior of the NSP2-calculated MCHFR is almost linear with flow except for the point corresponding to a 10-percent increase in core flow. Furthermore, the NSP2-calculated values of the MCHFR are always conservative with respect to those of the default Electric Power Research Institute (EPRI)-1 correlation. Finally, even for a 10-percent reduction in the core flow rate, the NSP2 value of the MCHFR is above the NuScale limiting value of 1.262.

5.0 <u>LIMTATIONS AND CONDITIONS</u>

Condition 1 An applicant referencing this TR in the safety analysis must also reference an approved CHF correlation which has been demonstrated to be applicable for use with TR-0915-17564. The basis for this Condition is provided in Section 4.1 of this SER.

6.0 <u>CONCLUSIONS</u>

The NRC staff reviewed TR-0915-17564-P and the applicant's responses to the staff's RAIs and audited supporting documentation. As a result of the staff's review of the TR in accordance with the applicable NRC regulations documented in Section 3.0 of this SER, the NRC staff has reasonable assurance that the use of the VIPRE-01 code with the NSAM described in the TR is appropriate for the thermal-hydraulic performance and plant safety analyses of the NuScale fuel design. Specifically, NRC staff finds that (1) VIPRE-01 applies to the NuScale steady-state and transient subchannel analysis using the methodology presented., (2) NSAM fulfills the NRC's requirements in the SERs for VIPRE-01, MOD-01 and MOD-02, (3) NSAM is independent of any specific CHF correlation and is used for NuScale applications if the methodology requirements are satisfied and if the NRC approves the CHF correlation as confirmed by Condition 1 of this SER, and (4) NSAM describes a methodology for treatment of uncertainties in the NuScale subchannel analysis that is appropriate. As a result of its review, the NRC staff considers all RAI questions closed and resolved.

The NSAM TR uses many example values of input parameters just to demonstrate the application of NSAM to perform subchannel analyses, whereas the TR only includes the subchannel analysis results to enhance the understanding of the analytical methods. Therefore, this SER does not approve the use of any specific example value input or result presented in the TR. In various subsections of this SER, the NRC staff has documented the review of various input parameters and the determination of whether or not their aspects are approved. The NRC staff will approve specific input values and ensuing results for the reactor core design for the subsequent licensing submittals (e.g., DCAs) referencing the NSAM TR. The NRC staff would review these final design- and plant-specific input values used in the subchannel analyses as a part of these subsequent licensing submittals.

The NRC staff based its review on an evaluation of the technical merit of the submittal and its compliance with the applicable regulations. If the NRC's regulations or acceptance criteria change such that the conclusions about the acceptability of the thermal-hydraulic methods presented in this TR are invalidated, the NRC will expect the licensee or applicant referencing the TR to revise and resubmit its documentation or submit justification for the continued effective applicability of these methodologies without revising the respective documentation.

7.0 REFERENCES

1. Letter from T. Bergman (NuScale Power, LLC) to the NRC, "NuScale Power, LLC Proprietary Marking Changes to 'Subchannel Analysis Methodology' Topical Report, Revision 1 (NRC Project No. 0769)," February 15, 2017 (ADAMS Accession No. ML17046A333).

2. TR-0915-17564-P (proprietary version), "Subchannel Analysis Methodology," Revision 1, NuScale Power, LLC, February 2017(ADAMS Accession No. ML17046A334).

3. Letter from S. Lee (NRC) to T. Bergman (NuScale Power, LLC), "Acceptance Letter for the Review of Topical Report TR-0915-17564-P, 'Subchannel Analysis Methodology,' Revision 0 (PROJ0769)," March 20, 2017 (ADAMS Accession No. ML17072A389).

4. Letter from C.F. Rossi (NRC) to J.A. Blaisdell (Northeast Utilities Service Company), "Acceptance for Referencing of Licensing Topical Report, EPRI NP-2511-CCM, 'VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores', Volumes 1, 2, 3, and 4," May 1986 (Only Proprietary Version Available - ADAMS Accession No. ML102100004).

5. Letter from A.C. Thadani (NRC) to Y.Y. Yung (VIPRE-01 Maintenance Group (VMP)), "Acceptance for Referencing of the Modified Licensing Topical Report, EPRI NP-2511-CCM, Revision 3, 'VIPRE-01: A Thermal Hydraulic Analysis Code for Reactor Cores' (TAC NO. M79498)," October 30, 1993 (Only Proprietary Version Available - ADAMS Accession No. ML102100004).

6. LO-1018-62243, "NuScale Power, LLC Submittal of TR-0915-17564, "Subchannel Analysis Methodology," Revision 2 (NRC Project No. 0769)," NuScale Power, LLC, October 31, 2018 (ADAMS Accession No. ML18305B216).

7. NuScale Power, LLC, Topical Report TR-0116-21012, Rev. 0, "NuScale Power Critical Heat Flux Correlation," October 5, 2016 (ADAMS Accession No. ML16279A363).

8. “Subchannel Analysis Methodology,” Audit Summary, November 14, 2018, (ADAMS Accession No. ML18317A274).

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10. Letter from T. Bergman (NuScale Power, LLC) to the NRC, “NuScale Power Supplemental Information in Support of NuScale Topical Reports (NRC Project No. 0769),” March 7, 2017 (ADAMS Accession No. ML17066A463).

11. EPRI-NP-2511-CCM-A, “VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores,” Revision 3, August 1989, prepared for the NRC by International Technical Services, Inc., New York, NY (Only Proprietary Version Available - ADAMS Accession No. ML102100004).

12. Stewart, C.W., and J.M. Cuta, “VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores,” Computer Code Manual, Volume 5 (Revision 4), “Guidelines,” NP-2511-CCM-A, Electric Power Research Institute, March 1988 (Only Proprietary Version Available - ADAMS Accession No. ML102040785).

13. Letter to R.C. Jones (NRC) from Y.Y. Yung (VIPRE-01 Maintenance Group), “Response to Request for Additional Information on VIPRE-01 MOD-02 Documentation EPRI NP-2511-CCM, VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores,” March 16, 1992.

14. Letter from T. Bergman (NuScale Power, LLC) to the NRC, Submittal of the Approved Version of NuScale Topical Report TR-0116-20825-A, “Applicability of AREVA Fuel Methodology for the NuScale Design,” Revision 1, February 9, 2018 (ADAMS Accession No. ML18040B306).

8.0 LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
AO	axial offset
AOO	anticipated operational occurrence
ARO	all rods out
BWR	boiling-water reactor
C	Celsius
CFR	*Code of Federal Regulations*
CHF	critical heat flux
CHFR	critical heat flux ratio
CPR	critical power ratio
DCA	design safety application
DLL	dynamic linked library
DNBR	departure from nucleate boiling ratio
DSRS	design-specific review standard
EPRI	Electric Power Research Institute
F	Fahrenheit
FSAR	final safety analysis report
GDC	general design criterion (criteria)
LHGR	linear heat generation rate

MCHFR	minimum critical heat flux ratio
NPM	NuScale power module
NRC	U.S. Nuclear Regulatory Commission
NSAM	NuScale subchannel analysis methodology
NuScale	NuScale Power, LLC
PDIL	power-dependent insertion limits
psia	pounds per square inch absolute
PWR	pressurized-water reactor
RAI	request for additional information
RCS	reactor coolant system
SER	safety evaluation report
SSC	structure, system, and component
SRP	Standard Review Plan
TER	technical evaluation report
THSP	Thom+EPRI (correlation)
TR	topical report