U.S. NUCLEAR REGULATORY COMMISSION

AUDIT REPORT FOR THE REGULATORY AUDIT OF

NUSCALE POWER, LLC

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

I. BACKGROUND

NuScale Power, LLC (NuScale) submitted by letter dated February 15, 2017, to the U.S. Nuclear Regulatory Commission (NRC), a topical report (TR) titled, "Subchannel Analysis Methodology" (Reference 1). The NRC staff started its detailed technical review of the NuScale TR on January 4, 2017.

The purpose of the NRC's regulatory audit of NuScale's TR was to: (1) gain a better understanding of the basis underlying the NuScale Subchannel Analysis Methodology and confirm the NRC staff's understanding; (2) better understand if the methodology meets NRC regulations and conforms to any applicable regulatory guidance; and; (3) develop requests for additional information. Additional background is available in the audit plan associated with this audit summary (Reference 2).

II. REGULATORY AUDIT BASES

Title 10 of the *Code of Federal Regulations*, Part 50, Appendix A, General Design Criterion 10, "Reactor Design," states that:

> *The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.*

An audit is needed to confirm the basis for the safety conclusions made in the applicant's TR.

III. AUDIT LOCATION AND DATES

The audit was conducted from the NRC headquarters via NuScale's electronic reading room.

Location: NRC Headquarters
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

Dates: Stage 1: July 12, 2017 through August 14, 2017
Stage 2: November 3, 2017 through March 15, 2018

Enclosure 1

IV. AUDIT TEAM MEMBERS

Bruce Bavol (Office of New Reactors (NRO), Project Manager)
Syed Haider (NRO, Technical Reviewer)
Joe Kelly (Office of Nuclear Regulatory Research, Technical Reviewer)
Matt Thomas (NRO, Technical Reviewer)

V. APPLICANT AND INDUSTRY STAFF PARTICIPANTS

Jennie Wike (NuScale)
Darrell Gardner (NuScale)
Steven Mirsky (NuScale)

VI. DOCUMENTS AUDITED

1. ER-0000-2337, Rev. 4, "Subchannel Analysis Methodology"
2. EC-A025-3562, Rev. 1, "NuScale Hot Channel Factors Evaluation"
3. EC-0000-3369, Rev. 0, "Subchannel Application Parametric Sensitivity Analysis"
4. EC-A010-3204, Rev. 1, "RCS Loop CFD"
5. EC-0000-2347, Rev. 2, "Steady State Subchannel Analysis"
6. EC-0000-4696, Rev. 0, "Subchannel Input Sensitivity Analysis"
7. EC-A021-5178, Rev. 0, "The Enthalpy Rise Engineering Uncertainty Factor"
8. EC-0000-3157, Rev. 0, "Subchannel Code-to-Code Benchmarking of Reactor Core"
9. EC-0000-3610, Rev. 0, "Evaluation of the Void Drift Model in NVIPRE"
10. EC-0000-3632, Rev. 0, "Interbundle Diversion Crossflow (IBDCF) VIPRE-01 Benchmark"
11. EC-0000-2265, Rev. 2, "Subchannel Analysis Basemodel"
12. EC-0000-3633, Rev. 0, "VIPRE-01 Two-Phase Flow Models and Correlations"
13. EC-0000-4296, Rev. 0, "VIPRE-01 Heat Transfer Correlations"
14. EC-0000-2897, Rev. 0, "Subchannel Analysis of Control Rod Misoperation"
15. EC-0000-2898, Rev. 0, "Subchannel Analysis of Decrease in Feedwater Temperature"
16. EC-0000-3073, Rev. 0, "Subchannel Analysis of Loss of External Load, Turbine Trip, Condenser Vacuum"

17. ECN-0000-5122, Rev. 0, “Pressure Drop Plots for EC-0000-2139, Rev. 0”

VII. DESCRIPTION OF AUDIT ACTIVITIES AND SUMMARY OF OBSERVATIONS

Post-Critical Heat Flux Application Limitation

The applicant did not seek approval for the use of VIPRE-01, MOD-02, in the NuScale Subchannel Analysis Methodology (NSAM) for post- critical heat flux (CHF) calculations. All VIPRE-01 calculation results presented in various audit documents were generated using the NRC approved NSP2 CHF correlation within its prescribed limits. Therefore, post-CHF application limitation is satisfied.

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 information provided by the applicant’s calculation in EC-0000-4296, Rev. 0, with regard to the VIPRE-01 heat transfer correlations and confirmed that the NSAM VIPRE-01 application used the Thom+EPRI (THSP) correlation (subcooled/saturated nucleate boiling + single phase). However, the THSP correlation was used in the NSAM VIPRE-01 application outside of its stated range of applicability. The NRC staff conducted a sensitivity study to assess the choice of THSP for the control rod misoperation case as, being the limiting example design basis event (DBE), it has the lowest minimum critical heat flux ratio (MCHFR). The NRC staff audit EC-0000-4296, Rev. 0, and performed a sensitivity study of five nucleate boiling correlations at representatively high pressure, which confirmed that the THSP heat transfer correlations used in the methodology are appropriate. 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. The NRC staff concludes that the THSP correlation option used in the NSAM TR provides a conservative result.

Input, Model and Analysis Assumptions, and Boundary Conditions Review

The NRC staff audited the document EC-0000-3369, Rev. 0, which presents the results of a sensitivity study comparing the results of the “uniformly applied” bundle average grid spacer loss coefficients to a case where all 1388 subchannels were modeled with different grid loss coefficients applied for each subchannel type (normal or “unit,” wall, corner, and guide-tube or instrument-tube). The subchannel specific values were derived by Areva for their high thermal performance (HTP) grid spacer design using data for typical pressurized water reactor (PWR) normal and off-normal operating ranges. NuScale derived their bundle average grid spacer formula for HTP grids using data typical of the normal and off-normal operating ranges for the NPM. The NRC staff audit calculation files and confirmed that the grid spacer loss coefficient was calculated conservatively.

The NRC staff audited calculation files and confirmed that for the rod bow penalty, the applicant followed the Areva based fuel rod bowing methodology. The NRC staff notes that there is [ ]. The NRC staff further notes that that the limiting CHF and LHGR [ ].

The NRC staff audited calculation files and confirmed that for the assembly bow penalty, the approved Areva methodology indicates that CHF penalties are only applied for rod bow and not assembly bowing.

The NRC staff audited calculation files and confirmed that for the rod power part of the hot channel factor, $F^{E}_{\Delta H1}$, the applicant followed the Areva fuel methodology (TR-0116-20825 Rev 0,) (Reference 3).

The NRC staff noted that for the limiting assembly peak-to-average ratio, [ ], the applicant was tracking movement of the [ ] assumption from a non-quality assurance (QA) document into a QA document via ODI-15-0318. This open design item (ODI) has since been closed.

The NRC staff noted that ODI-15-0319 was implemented to track the assumption for the value of the enthalpy rise engineering uncertainty (flow area reduction factor). The value was originally based on a non-final fuel design and CHF correlation. This ODI has since been closed.

The NRC staff noted that ODI-15-0320 was implemented to track the uncertainty value for $F_{\Delta H}$ measurement uncertainty, since the measurement uncertainty value of instrumentation is unknown until procurement. The NRC staff notes that the applicant implements a representative value based on traditional plant engineering.

The NRC staff noted that for the flow leakage between the heavy reflector and core barrel, the applicant assumed this value to be zero because of the relative size of this region and because conservatisms exist in other bypass fractions.

The NRC staff noted that for the max guide tube bypass flow fraction, ODI-16-1004 was implemented to verify the assumptions used in generating this number. The applicant obtains this penalty factor from a bounding neutronics analysis.

The NRC staff noted that cold geometry conditions are used by the applicant because the change in flow area, and wetted and heated perimeters for 'hot' conditions is negligible since thermal expansion of M5 and zircaloy-4 materials are very similar. The NRC staff noted that a rod can undergo thermal expansion but will also go through creepdown with exposure. The NRC staff noted that axial changes can also occur; however, the applicant states that hot channel factors account for pellet densification among other variations.

The NRC staff reviewed the development of the applicant's basemodel and noted that the subchannel analyses utilize eighth-core nodalization and do not represent any cycle specific core. The NRC staff noted that the basemodel is constructed in a way to preserve the limiting core conditions along with the operational envelope specified in technical specifications or core operating limits report (COLR). The NRC staff also noted that it is established based on the design peaking factors in combination with the limiting reactor coolant system global parameters. The NRC staff reviewed several radial nodalizations, one being used for most of the DCD calculations, i.e. the 24-channel model, the 51-channel model, as well as a fully detailed model, i.e. 1388-channel model, used for select reactivity transients. The applicant conducted sensitivity runs on the three models above to determine adequacy of basemodel for NuScale core. The applicant confirmed in EC-0000-3369, Rev 0, that radial geometry nodalization accurately maintain the hot channel flow field (void fraction, mass flux, and crossflow profiles vs. elevation) and resulted in conservative MCHFR. The applicant conducted benchmarking for high pressure, low pressure, low power and flow, high power and flow, and high radial peaking augmentation. The applicant found that lumping channels more than a few rod rows from hot channel has a negligible impact on flow field. For the axial nodalization the applicant stated that MCHFR is typically observed to be calculated between fourth and fifth spacer grid; therefore, it uses a finer resolution in this "CHF region." The applicant conducted a sensitivity study in EC-0000-3369, Rev. 0, which used different uniform node heights within the CHF region (space between fourth spacer grid and the end of active fuel length) to determine that [ ] nodes are sufficient to resolve the flow field. The other nodes within the active fuel region are split into equal increments between the edges of the defined spacer grids which result in just below [ ] nodes. The applicant uses smaller nodes at the inlet and exit to properly resolve flow perturbations. The applicant uses an aspect ratio of less than 3 to ensure axial flow remains dominant in subchannel analysis.

The NRC staff noted that for the applicant's subchannel analysis methodology, the boundary conditions required for VIPRE-01 are inlet flow rate, inlet enthalpy or temperature, system pressure, bypass flow, power, exit pressure, and inlet and exit cross flows. The boundary conditions come from a system's code (e.g. RELAP5) and is done based on a once-through linear process (i.e. no iteration or code coupling). The applicant stated that the NuScale loop is single phase natural circulation. At powers greater than 15 percent rated thermal power (RTP), the Re Number in the core is well within the forced turbulent convection flow range, making correlations like Dittus-Boelter appropriate for approximating the surface heat transfer coefficient (EC-A030-2713, Rev. 1, "Primary and Secondary Steady State Parameters"). The applicant uses NRELAP5 to calculate natural circulation flow as input into subchannel analysis. The applicant uses VIPRE-01 with the RECIRC option. The NRC staff noted that the applicant could input core pressure drop and exit enthalpy into VIPRE-01 in place of inlet mass flow rate; however, they input inlet mass flow rate in order to more directly provide a way of accounting for bypass flow and treatment of flow imbalanced related uncertainties.

The applicant conducted a reactor coolant system (RCS) Loop computational fluid dynamics (CFD) analysis to determine the best-estimate reflector cooling bypass fraction based on core power levels and axial power distribution. Furthermore, the applicant said that it determined a bounding thermal design flow rate by applying large biases to the pressure drops within the RCS loop. The NRC staff did not review the CFD methodology to determine if this method of determining reflector cooling channel bypass is adequate for subchannel analysis methodology.

The NuScale power plant relies upon natural circulation to convect coolant through the core. To determine how this coolant is distributed amongst the core inlet, and thus to the assemblies which are modeled in the subchannel analysis, the applicant conducted a detailed CFD analysis. In the CFD analysis, the applicant ran many cases varying the power level, power distribution (axial and radial), and core average temperature to determine how the inlet flow distribution responds. [

] Finally, the applicant applied a penalty on the limiting assembly inlet flow fraction to account for uncertainty in the distribution of the core inlet flow. The applicant showed that the amount of flow maldistribution (i.e. the amount of penalty applied to the hot assembly) had an insignificant effect on MCHFR. The NRC staff noted that the reason for this insignificant effect is due to the assembly cross flow causing the axial flow to converge to the same mass flux at the same axial location regardless of the amount of flow maldistribution. Lastly, the applicant stated that the inlet flow distributions used for licensing calculations will be confirmed by geometry-specific testing. Due to the conservative approach taken by the applicant to develop the bounding inlet flow distribution, due to the insignificant effect the flow maldistribution penalty has on the MCHFR, and due to the applicant's planned geometry specific testing to confirm the bounding inlet flow distributions, the NRC staff finds the applicant's approach for developing the inlet flow distribution to be used in the subchannel analysis appropriate.

The applicant's RCS loop CFD calculation was also conducted to verify the core inlet temperature distribution for several power levels and power distributions. The applicant showed that, due to the unique design of the integral steam generators, the largest deviation in core inlet temperature for any case simulated was less than 1 degree Fahrenheit. Therefore, the applicant uses a uniform core inlet temperature distribution.

The applicant stated that the turbulent mixing model within VIPRE-01 accounts for the exchange of enthalpy and momentum between adjacent subchannels due to turbulent flow. The coefficient for turbulent mixing and the turbulent momentum factor are the two inputs needed for this model. This mixing model is incorporated into the energy and momentum equation, which is dependent on the amount of turbulent crossflow per unit length. Turbulent crossflow, w', is calculated as: $w' = ABETA * S * \bar{G}$, ABETA is the turbulent mixing coefficient, S is the flow gap width, and Gbar is the average axial mass flux velocities among the adjacent subchannels. ABETA is determined from thermal mixing tests and is [ ] as determined in EC-0000-2594, Rev. 0, "Assessment of Stern Thermal Mixing Test Data." ABETA is fuel design specific; however, the applicant stated that the value they use is conservatively low and representative of non-mixing vane grids. The applicant stated that the turbulent momentum parameters are not measured and are justified from sensitivity studies. The sensitivity study results demonstrate that NuScale basemodel is not sensitive to this value and [ ] is suitable (EC-0000-3369, Rev. 0). The applicant further stated that as the turbulent mixing parameter is derived for subchannel phenomenon, lumped channels, as used in the VIPRE-01 basemodel, require a reduction in the mixing coefficient ABETA. The mixing coefficient ABETA is reduced by dividing the standard subchannel value, [ ], by the ratio of the centroid distance to the fuel rod pitch. This equates to non-lumped subchannel gaps having the [ ] mixing coefficient, while the lumped channels are reduced accordingly to the centroid distance differences for each gap. The NRC staff noted that the effect from reducing the turbulent mixing coefficient has negligible differences in the lumped basemodel since the value of ABETA is already small.

The NRC staff noted from the audit that the radial power distribution is developed conservatively and provides a bounding distribution. The radial power distribution is characterized by FdH, which is the enthalpy rise factor describing the integrated rod power for a particular rod. The FdH value changes with exposure, fuel composition, burnable poison loading, operational history, and thermal hydraulic conditions. The applicant's radial power distributions assumed in the subchannel analyses are artificial and remain constant throughout the transient. For static 15.4 analyses, the NRC staff noted that the FdH power distribution will remain the same as that derived for the basemodel with one slight modification: an FdH augmentation peaking factor for asymmetric radial peaking is applied to all the rods within the central limiting assembly, which preserves the peak-to-average value. The assembly furthest from the central limiting distribution is reduced by a factor to maintain normalization of power. The applicant determines an augmentation factor in the nuclear analysis calculation as the ratio of change in FdH from post-event to the initial condition. The relative increase in FdH captures the peaking increase from control rod motion and is applied to the basemodel limiting central assembly as a multiplicative factor. For time dependent 15.4 analyses, the radial power distribution is severely skewed from the bounding radial power distribution used for the static analyses; therefore, the fully-detailed (1388 subchannels) basemodel is used. The applicant uses a pin-by-pin FdH power distribution in the subchannel analysis that is determined from an event specific nuclear analysis to be at the time of peak core power. The staff notes that using the power distribution at the time of peak power is conservative as it is used throughout the entire transient with the transient fluid boundary conditions from NRELAP5.

The applicant stated that the core design has imposed a limitation on the peak value of FdH; therefore, the highest value for any fuel rod at hot full power has been limited to 1.5; which is the FdH peaking design limit, also known as the technical specification (TS) peaking factor (LTSP). This limit is relaxed for lower power levels, allowing a linear increase to a hot zero power value of 1.65. The applicant stated that the subchannel methodology bounds any radial power distribution that occurs in the core prior to any anticipated operational occurrence (AOO), infrequent events (IE), or accident. The staff noted that by using a 95/95 acceptance criteria on the hot rod, the hot rod for the radial power distribution is set to the tech spec peaking factor (i.e. design limit), dependent upon the initial condition. For normal operation at HFP this means 1.5. Setting the tech spec FdH design limit at the core design limit implies uncertainties associated with FdH are not included. The applicant stated that uncertainties are accounted for in the subchannel analysis methodology as an increase on the LTSP value. The uncertainties account for measurement uncertainty related to instrumentation used for monitoring, and engineering hot channel uncertainty. The NRC staff noted that increases in FdH peaking for rodded configuration are also included in order to create the analytical limit. The applicant stated that for cases evaluated at partial power levels, the FdH distribution in the central assembly is increased by a factor that is the ratio of the partial power LTSP value divided by the HFP value (1.5). The central assembly is peaked by multiplying this factor to the FdH value and an outer assembly is reduced to maintain normalization of power. The applicant imposed an additional requirement on the FdH design limit; that is, the peak FdH rod for any assembly must not occur on the peripheral row. This allows the hot channel to not occur on the outer row, as the outer row would be influenced by direct crossflow from the annulus channel between assemblies. The applicant stated that this channel is not truly simulated in a CHF test and not a valid channel for CHF to occur anyways.

The applicant stated that radial tilt can have an impact on thermal margin because if the power tilts, then an assembly can have a higher FdH, which must be accounted for in the safety analysis. The applicant stated that the design enthalpy rise peaking factor safety limit will inherently account for radial tilt. The applicant further stated that the nuclear analysis group will verify the tech spec FdH design limit is met accounting for radial tilt in core design calculations caused by any xenon transients that disturb symmetric power peaking; therefore, the subchannel analysis requires no additional methodology to account for radial tilt.

The applicant stated that for the safety analysis, the core can be at 100 percent RTP with control rods inserted to the power dependent insertion limits (PDIL). With the enthalpy rise hot channel factor defined as an all rods out (ARO) maximum value, an additional peaking factor therefore must be accounted for on the hot rod for PDIL-to-ARO augmentation peaking differences. The applicant stated that a nuclear analysis determines the value for PDIL-ARO augmentation for all power levels. The NRC staff noted that the subchannel analysis methodology uses a bounding single value to be applied for use at every power level, a simplification such that only one value is applied as the factor for all transients, including those that initiate at partial powers. The applicant notes that this value is cycle-specific based on loading patterns and control rod worth; therefore, a bounding value is suggested in order to be cycle-independent, which is confirmed by a cycle-specific nuclear analysis calculation. EC-A021-1627, Rev. 1, “Core Design Parameters Analysis,” provides the PDIL-ARO augmentation factor for the DCD core design, which ranges from [ ] at HFP [ ] at HZP. The powers of [ ]. The applicant noted that for events that initiate from a partial power condition, an FdH augmentation factor is used that inherently accounts for this rodded peaking factor. Therefore, the applicant stated that the NuScale subchannel analysis will use a value of [ ] as the PDIL-ARO FdH augmentation factor for rodded operating conditions.

As mentioned earlier, the applicant stated that the radial power distribution is not intended to model any specific core design, but must be bounding in order to be used for future core designs that maintain a similar shuffle/loading pattern. The applicant stated that a sensitivity confirms that radial power distribution far removed from the hot subchannel has a negligible impact on the MCHFR results; therefore, the use of a radial power distribution with the hot rod at the design peaking limit is sufficient for any distribution in a cycle-specific core. The applicant stated that since the artificial radial power distribution used in the basemodel and most transients is not realistic or representative of the actual core conditions, the determination of MCHFR for meeting the acceptance criterion is only acceptable for the hot rod and subchannel. The purpose for the artificial radial power distribution is to capture the hot subchannel flow conditions, which is dependent upon the surrounding crossflow neighbor channels. The NRC staff noted that a flat power distribution is one in which nearly all rods provide similar power and therefore flow conditions; this power distribution limits the amount of turbulent mixing and diversion crossflow in the hot subchannel, and therefore, is conservative for thermal margin calculations. The applicant stated that the approach for the subchannel methodology is to set the hot rod at the tech spec value at HFP as discussed above, as well as accounting for the radial tilt and PDIL-ARO factor. The power distribution for an assembly may be characterized by its peak-to-average ratio, which is the maximum FdH rod in an assembly divided by the average FdH. A value closer to unity denotes a flat power distribution. Since a cycle-specific core design does not specifically have a distribution that has FdH at the tech spec value, the applicant utilizes a spectrum of peak-to-average FdH values for each assembly throughout the cycle burnup in the equilibrium cycle design to inform a bounding distribution. The applicant states that this method of peak-to-average FdH values generates several thousand values. The applicant filtered out assembly average relative power fraction values that were less than [ ] as the hot rod is too low to be considered limiting for MCHFR. The loading pattern for the NuScale DCD core [ ]. The applicant states that for the highest powered assemblies, the ratio of the peak-to-average is flatter, as all the rod FdH values are not far from the mean. [

] The applicant stated that this value is then used to set a “slope” for the artificial power distribution of the core. The hot rod is set to [ ] and the remaining rods within the 1/8th central assembly are reduced via the slope and distance the rod is from the hot rod. The NRC staff noted that this approach keeps hotter rods around the hot rod, which reduces the amount of potential turbulent mixing and diversion crossflow from the hot subchannel. The applicant stated that the other assemblies gradually reduced in power preserve the power generated, which means the lumped rods that are used to represent partial or full assemblies must take into account the total number of rods when used in preserving the total power. The NRC staff reviewed the radial power distribution for the subchannel basemodel with hot rod set to [ ].

The applicant stated that the radial power distribution, as described above, is penalized for FdH measurement uncertainty and FdH engineering uncertainty. The applicant stated that the radial power distribution prior to applying uncertainty penalties retains conservative peak-to-average for the hot assembly, while the rod where MCHFR is determined accounts for the applicable uncertainties. In order to accommodate the slight increase on the hot rod, the applicant applies a small reduction in the peripheral assembly to maintain normalization. The applicant stated that FdH measurement and FdH engineering uncertainty are applied to the hot rod via room-sum-square (RSS) methods, because the uncertainties are composed of independent parameters. The applicant further stated that the hot rod is increased by the total uncertainty and the furthest rod is slightly reduced to maintain power normalization.

The applicant stated that the axial power distribution is the distribution of power along the height of the active fuel length. This power peaking can be expressed as a core-average, rod-specific, or assembly specific shape. Thus, the axial power distribution is widely variable, dependent upon core design, exposure, and power history. The staff notes that the axial power distribution is not a strictly imposed limitation on the core design, but it is inherently limited by the total peaking factor FQ core design limit. The applicant stated that the subchannel methodology incorporates a CHF-limiting power shape for use in all steady-state analyses as well as most transients. Not included are Standard Review Plan (SRP) NUREG-0800, “Standard Review Plan,” Chapter 15, “Transient and Accident Analysis” events in which control rods are withdrawn during the transient. The axial power shape of interest is provided from a nuclear analysis calculation. The applicant derived a sufficient set of bounding and representative power shapes for various configurations of core cycle exposure, control rod configuration (ARO, PDIL, alignment uncertainty), xenon distribution (axial offsets), 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 window. The neutronics analysis ensures axial power shapes developed for each power level will cover the widths of the axial offset (AO) window. The applicant stated that from the many potential permutations of the core-average axial power shape from the nuclear analysis, the CHF limiting shape is obtained. It is known that the MCHFR for the subchannel basemodel is sensitive to the axial power shape, with a conclusion that a top-peaked axial power shape for the same magnitude of axial peaking is more limiting with respect to CHF. This means for each power level (in reasonable increments), power shapes will be provided with an emphasis on ensuring the top-peaked shapes reach the widths of the AO window. The staff noted that axial offset alone is not an indicator for the axial power shape, as the magnitude of Fz and the x/L (axial elevation) contribute to the MCHFR; therefore, the applicant selects the axial power shape as the one that results in the lowest MCHFR using the basemodel. The applicant conducted a subchannel analysis calculation to determine the limiting axial power distribution for use in steady-state and most transient analyses, from which a bounding axial power shape for each provided power level could be ascertained. The applicant stated that the CHF limiting axial power shape (based on core-average axial power) is sufficient to be used for most transient analyses. For transients that are not related to control rod motion (non-SRP 15.4 events), the core-average axial power shape does not deviate significantly from the spectrum already considered within the power shapes analysis, and therefore, the subchannel limiting axial power shape is appropriate. The applicant stated that the combination of the core-average axial power shape of initiating power level with the conservative radial power distribution and core hydraulic boundary conditions from NRELAP5 provides a conservative MCHFR calculation. For SRP 15.4 Analyses, the applicant stated that an event-specific nuclear analysis is performed to determine control rod worth, radial power distribution (FdH augmentation factor), and axial power distribution. The applicant stated that the initial conditions for control rod motion transients will start from the edges of the AO window, as the edge of the window is a permissible condition prior to an event actuation. If a permissible normal operation power swing results in the core-average axial power shape at the edges of the window, then once 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.

The NRC staff noted that the applicant uses a deterministic methodology for uncertainties, i.e., the uncertainty associated with a parameter is applied in the conservative direction, without regard of the combination nature of uncertainties. The NRC staff also noted that because axial and radial discretizations used for the subchannel analysis were used in the benchmarking, no additional penalty is needed for discretization. The NRC staff further noted that because the applicant developed the CHF correlation with the subchannel VIPRE-01 code, the CHF correlation inherently includes uncertainty from the computer code. Thus no additional penalties are necessary for this.

The NRC staff noted that the applicant applies a penalty directly to the 95/95 CHF limit due to heat flux engineering uncertainty. Heat flux engineering uncertainty accounts for fuel enrichment, pellet density, pellet diameter, and fuel rod surface area, which are small manufacturing uncertainties that affect the local heat flux. The applicant states that the value for this penalty is calculated in accordance with the Areva fuel methodology (TR-0116-20825). Similarly, the applicant applies another penalty for the LHGR engineering uncertainty.

The staff noted that no additional penalty is applied for the radial power distribution uncertainty because the hot rod is assumed to be at the design limit FdH. Furthermore, the staff noted that the applicant does account for any core exit pressure uncertainty because the open upper plenum design allows for immediate pressure equilibrium to occur.

Courant Number Criterion

The NRC staff audited calculation files and confirmed that the topical report's Courant number methodology is accurate and reasonable. The NRC staff audited the relevant subchannel analysis calculations for three design-basis example transients: (1) control rod misoperation (EC-0000-2897, Rev. 0); (2) decrease in feedwater temperature (EC-0000-2898, Rev. 0); and (3) loss of external load, turbine trip, and condenser vacuum (EC-0000-3073, Rev. 0). The NRC staff audited the results for both the MCHFR and the minimum value of the Courant number for each of the three AOOs, and NuScale's confirmation that the Courant number limitation had been met. Further, the staff independently checked the NuScale results to confirm their validity. In the applicant's calculation of Courant number, transient time step is selected on a case-by-case basis for the axial nodalization and coolant velocity, to ensure Courant number to be greater than one ($N_C > 1$). Audited calculations assume 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. The staff also performed independent Courant number confirmatory calculations for the three design-basis transients. The following Table compares the staff Courant number results with the NuScale values the staff viewed via audit:

Case	**NuScale Result**	**Staff Result**
#1: Loss of external load	[ ]	[ ]
#4: Control rod misoperation	[ ]	[ ]
#5: Decrease in feedwater temp.	[ ]	[ ]

The approximate Courant number values calculated by the staff were within 3.8 percent of the NuScale values, which is not significant. The NRC staff's calculated values were typically lower than the NuScale values. The staff noted that the NuScale value were 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 NRC staff audit showed that the Courant number criterion was met throughout all three boiling transients, and no numerical stability issues resulted from the transient time step selection.

<u>Qualification of VIPRE-01 MOD-02 Models</u>

To understand the justification and conservatism of the applicant's two-phase flow modeling assumptions, the NRC staff audited the applicant's documentation, EC-0000-3610, Rev. 0, of the drift flux model sensitivity cases applicable to NuScale conditions. The NRC staff audit confirmed that the use of the drift flux model had an insignificant impact on void fractions and CHFRs at any point along the hot channel, as stated in the TR. The void fraction results vary little across the three base models for the hot channel (i.e., a less than 2-percent change at any point along the model elevation). The NRC staff also confirmed that the modified code, NVIPRE-01 as described in EC-0000-3610, Rev. 0, without the void drift option predicted void fraction and MCHFR values and trends identical to that of VIPRE-01. The NRC 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.

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 audited EC-0000-3157, Rev. 0, and EC-0000-3633, Rev. 0, and confirmed that the CHFRs predicted by two different subchannel codes and different sets of two-phase flow closure models were in reasonable agreement. The audit helped the NRC staff conclude 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.

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. The staff conducted an audit and performed confirmatory calculations to ensure the consistency of the inlet flow boundary condition between VIPRE-01 and NRELAP5. 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. 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. Less of the core was in a two-phase condition for the other cases. The NRC staff observed a small but constant 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. 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. The NRC staff audited EC-0000-2139, Rev. 0 to clarify the locations of the "virtual" pressure taps that were used to provide the NRELAP5 values of core pressure drop. The NRC staff also audited ECN-0000-5122, Rev. 0, that showed that control variables 701 and 702 were added to the NRELAP5 input model to provide values for the core pressure drop. The audited material showed that NuScale 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 calculations 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 NRC staff calculated a gravity head adjustment of 0.126 psia. This value almost exactly matches the observed offset between the NRELAP5 and VIPRE-01 calculated pressure drop values. 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. Therefore, the NRC staff audit confirmed that the methodology's selection of inlet flow boundary condition is appropriately justified.

VIII. EXIT BRIEFING

The NRC staff conducted an audit closeout meeting on March 19, 2018. At the exit briefing the NRC staff reiterated the purpose of the audit and discussed their activities. The NRC staff stated that they had identified areas where additional information is being requested to support the review, and briefly discussed the scope of these information requests. References to the detailed questions are provided in Section IX of this audit summary.

IX. REQUESTS FOR ADDITIONAL INFORMATION RESULTING FROM AUDIT

The NRC staff issued 4 requests request for additional information (RAI) based on information observations made during the audit. These RAIs are available in Agencywide Documents Access and Management System (ADAMS). ADAMS Accession Nos. are provided in Table 1.

Table 1: RAIs Resulting from Audit

RAI Number	Question Reference	Response (Supplemental) Reference
9080	ML17254A439	ML17313B205 (ML18061A108)
9086	ML17252A688	ML17299A973
9099	ML17245A000	ML17251A368
9129	ML17321A597	ML18015A012

X. OPEN ITEMS AND PROPOSED CLOSURE PATHS

There are no open items associated with this audit.

XI. DEVIATIONS FROM THE AUDIT PLAN

There were no deviations from the audit plan.

XII. REFERENCES

1. 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. Audit Plan for the Regulatory Audit of NuScale Power, LLC Topical Report-0915-17564, Revision 1, “Subchannel Analysis Methodology,” July 12, 2017 (ADAMS Accession No. ML17193A276).

3. “NuScale Power, LLC Submittal of Topical Report TR-0116-20825, “Applicability of AREVA Fuel Methodology for the NuScale Design,” Revision 0 (NRC Project No. 0769)”, dated March 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16095A244).